NRC Rules and Regulatory Guides Concerning Reactor Pressure Vessel Integrity

Mark Kirk
Senior Materials Engineer
United States Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Division of Engineering, Component Integrity Branch
mark.kirk@nrc.gov

KINS, Daejeon, Republic of Korea, 24th to 25th September 2012
Seoul National University, Seoul, Republic of Korea, 26th September 2012

Objectives of Visit

- Present and discuss information on the
  - Background of,
  - Technical bases for, and
  - Experience with

  NRC regulations and regulatory guides concerning RPV integrity
- Specific focus on
  - History & background: 10 CFR 50 App H & G, PTS Rules
  - Master Curve
  - Regulatory experience
    - Low upper shelf energy
    - High RT_{PTS}

Outline

- **Conceptual framework for RPV Integrity**
- **Organizational**
  - Relationships and roles: NRC / ASME / ASTM
  - RPV integrity documents: inter-relationships between NRC, ASME, and ASTM
- **Technical**
  - Evolution of requirements & technical basis for RPV integrity regulations and reg. guides
  - Current status
    - Technical basis
    - Description of efforts to update, if any
  - Experience in use

Outline

- **Conceptual framework for RPV Integrity**
- **Organizational**
  - Relationships and roles: NRC / ASME / ASTM
  - RPV integrity documents: inter-relationships between NRC, ASME, and ASTM
- **Technical**
  - Evolution of requirements & technical basis for RPV integrity regulations and reg. guides
  - Current status
    - Technical basis
    - Description of efforts to update, if any
  - Experience in use

RPV Integrity
Conceptual Framework

1. RPV Integrity, a balance of applied driving force and fracture resistance.
2. Schematic K_{Ic} *vs*. $K_{APPLIED}$ comparisons.
3. Deterministic *vs*. probabilistic assessments; more in common than you may think.
4. Evolution of RPV integrity standards over time.

RPV Integrity

<u>Words</u>:

The structural integrity of nuclear reactor pressure vessels (RPVs) is ensured by requiring that the resistance to fracture (toughness) always exceeds the driving force for fracture that is produced by loading

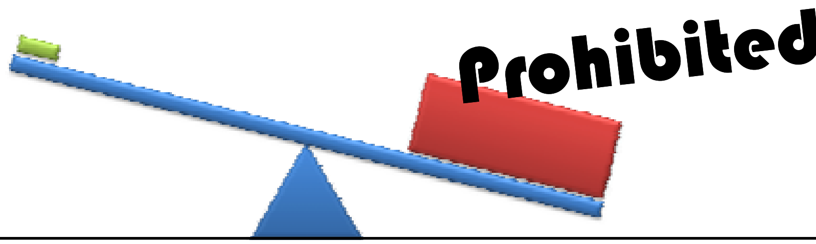
RPV Integrity

Words:

The structural integrity of nuclear reactor pressure vessels (RPVs) is ensured by requiring that the resistance to fracture (toughness) always exceeds the driving force for fracture that is produced by loading

Equation:

Fracture Toughness $<$ Fracture Driving Force



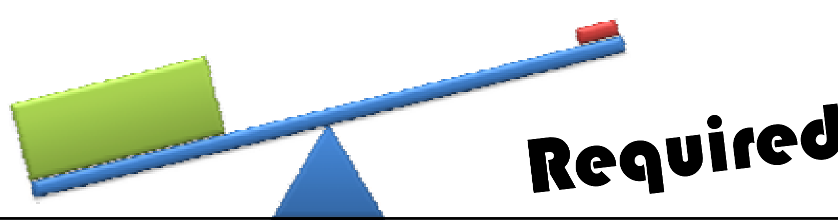
RPV Integrity

Words:

The structural integrity of nuclear reactor pressure vessels (RPVs) is ensured by requiring that the resistance to fracture (toughness) always exceeds the driving force for fracture that is produced by loading

Equation:

Fracture Toughness $>$ Fracture Driving Force



RPV Integrity

Words:

The structural integrity of nuclear reactor pressure vessels (RPVs) is ensured by requiring that the resistance to fracture (toughness) always exceeds the driving force for fracture that is produced by loading

Equation:

Fracture Toughness > Fracture Driving Force

$$K_{Ic}(T, RT_{NDT}) > \sigma \times \sqrt{\pi a} \times F$$

Toughness, Embrittlement, Stress, Crack Size, Geometry

RPV Integrity

All RPV integrity guidelines address and/or control one or more of these five factors

Fracture Toughness > Fracture Driving Force

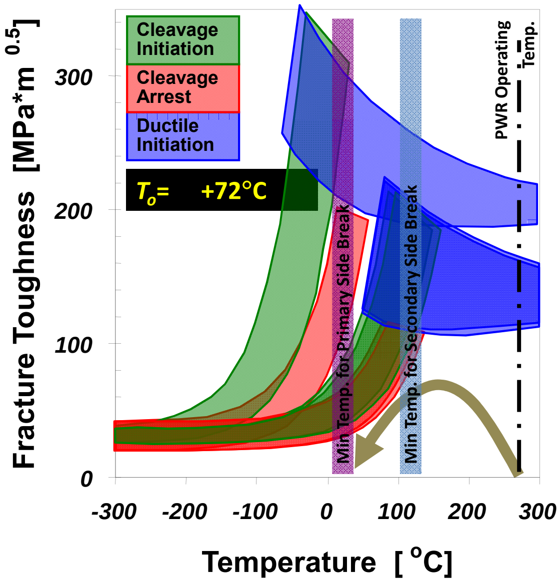
$$K_{Ic}(T, RT_{NDT}) > \sigma \times \sqrt{\pi a} \times F$$

Toughness, Embrittlement, Stress, Crack Size, Geometry

K_{Ic} *vs.* $K_{APPLIED}$

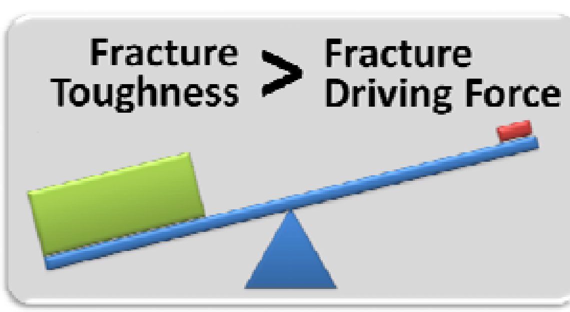
- Toughness curves for the most embrittled axial weld in a highly embrittlement PWR
  - → At beginning of life
  - → At 40 years
  - → At 60 years

  Toughness evolves over plant live
- Driving force remains fixed over plant life
- K_{Ic} & $K_{APPLIED}$
  - Are both estimates, can have varying levels of conservatism
  - Both quantities are distributed
  - Distributions can be accounted for implicitly of explicitly

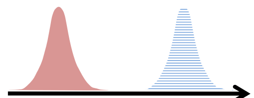
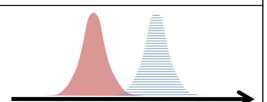
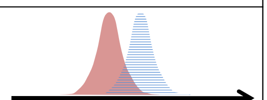


Deterministic *vs.* Probabilistic
Overview

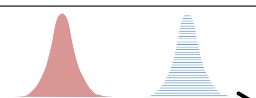
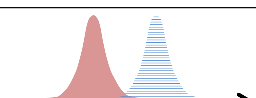
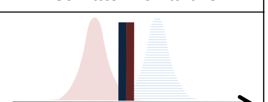
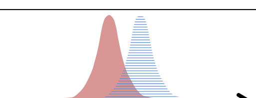
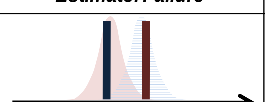
- To perform an RPV assessment, you compare
  - Structural driving force (*DF*) to fracture
  - Materials' resistance (*R*) to fracture (toughness)
- *DF* and *R* are inherently distributed quantities
- Deterministic & probabilistic analyses both provide a mathematical representation of these distributed quantities



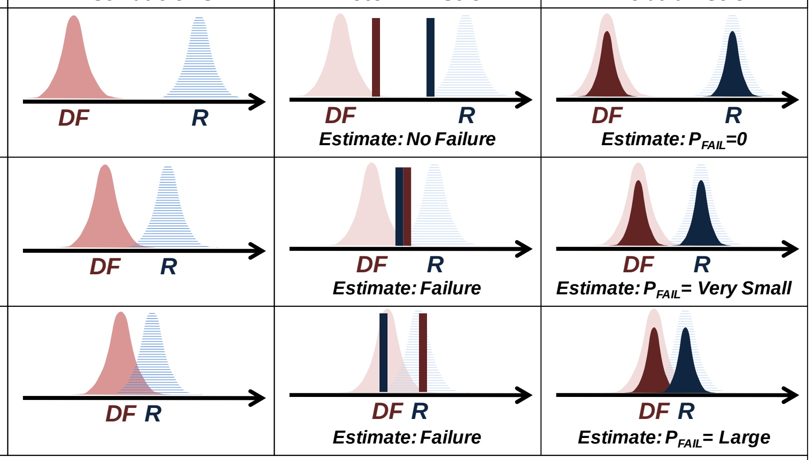
Deterministic *vs*. Probabilistic
The Reality we Seek to Represent

Actual Situation	
$P_{FAILURE}$	Distributions
Zero	 *DF* *R*
Small	 *DF* *R*
Large	 *DF* *R*

Deterministic *vs*. Probabilistic
Mathematical Models of Reality (1 of 2)

Actual Situation		Representation
$P_{FAILURE}$	Distributions	Deterministic
Zero	 *DF* *R*	 *DF* *R* *Estimate: No Failure*
Small	 *DF* *R*	 *DF* *R* *Estimate: Failure*
Large	 *DF* *R*	 *DF* *R* *Estimate: Failure*

Deterministic *vs.* Probabilistic
Mathematical Models of Reality (2 of 2)



Actual Situation		Representation of Distributions	
$P_{FAILURE}$	Distributions	Deterministic	Probabilistic
Zero	*DF* *R*	*DF* *R* *Estimate: No Failure*	*DF* *R* *Estimate:* P_{FAIL}*=0*
Small	*DF* *R*	*DF* *R* *Estimate: Failure*	*DF* *R* *Estimate:* P_{FAIL}*= Very Small*
Large	*DF* *R*	*DF* *R* *Estimate: Failure*	*DF* *R* *Estimate:* P_{FAIL}*= Large*

<u>Note</u>: The actual *DF* and *R* distributions are also shown, lightly, in the deterministic and probabilistic columns.

Deterministic *vs.* Probabilistic
Similarities and Differences

<u>Similarities</u>

- Both treat uncertainty
  - Deterministic models ***<u>bound</u>*** uncertainty
  - Probabilistic models ***<u>quantify</u>*** uncertainty
- Probabilistic models may contain deterministic aspects where full information is lacking, e.g.:
  - Conservative models
  - Bounding inputs
  - And so on …

<u>Differences</u>

- How result is expressed
  - **<u>Deterministic</u>**: "Failed" or "Not Failed"
  - **<u>Probabilistic</u>**: A failure probability
- Who the decisionmaker is
  - **<u>Deterministic</u>**: Only the engineering analyst (because "failure" is unacceptable)
  - **<u>Probabilistic</u>**: Many people (because some failure probability can be accepted)

Evolution of RPV Integrity Standards

- **RPV integrity assessment in 1973**
  - The 1st significant documents governing RPV integrity (10 CFR 50 Appendix G & ASME Section XI Appendix G) were adopted
    - Conservatisms were embedded in the assessment (e.g., ¼-T flaw, factor of 2 on pressure, bounding K_{Ic} curve, …) in the hope that unknown unknowns would be thus covered.
  - These documents have changed only in detail since that time
- **Technical state of knowledge in 1973**
  - Fracture mechanics
    - LEFM had been around since 1952, but was only used in the assessment of aircraft and other "brittle" metals. There was significant debate regarding if LEFM could be applied to pressure vessels at all.
    - EPFM was new [Rice: 1968]. Use of ductile materials still guided mostly on experience and Charpy tests. No good understanding of the relationship between Charpy and upper shelf toughness.
  - Embrittlement
    - Beginning to understand the deleterious effects of copper
    - Beginning to understand that embrittlement reduces toughness
    - Three years from the 1st predictive equations for embrittlement
- **USA operating experience in 1973**
  - Commercial reactors operating since 1957 (Shippingport, Pennsylvania)
  - Nearly 30 commercial nuclear reactors were operating in the USA in 1973

In 1973 …

- Knowledge of RPV integrity, fracture, and embrittlement was incomplete
  - Guidelines to ensure operating safety were developed based on available information
- Those guidelines were based on many judgments, mostly conservative
  - Conservative guidelines were not restrictive at the time because most RPVs were then new and, therefore, not embrittled very much
  - Those guidelines mostly survive today
- If RPV integrity cannot be demonstrated relative to these guidelines it does not necessarily imply that public safety is threatened
  - The evolution of RPV standards represents technological improvement, and thereby enhanced safety

Shippingport Reactor

Outline

- **Conceptual framework for RPV integrity**
- **Organizational**
  - Relationships and roles: NRC / ASME / ASTM
  - RPV integrity documents: inter-relationships between NRC, ASME, and ASTM
- **Technical**
  - Evolution of requirements & technical basis for RPV integrity regulations and reg. guides
  - Current status
    - Technical basis
    - Description of efforts to update, if any
  - Experience in use
- **Summary of current activities**
  - NRC research
  - NRC regulations and regulatory guides
  - ASME SC-XI
  - ASTM E10.02

Relationships and Roles

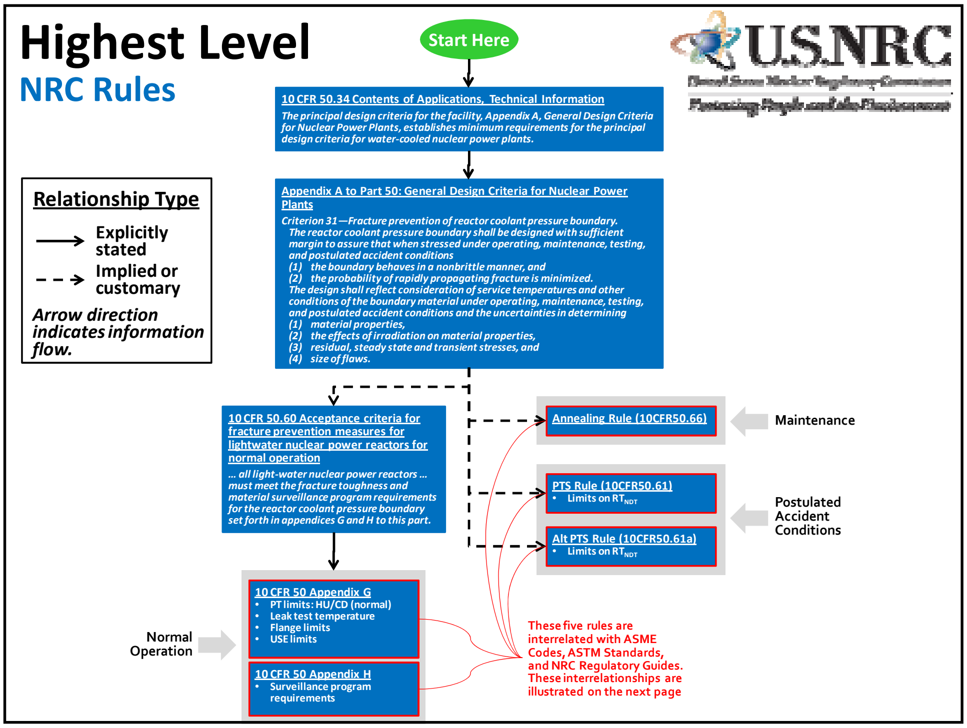
- **NRC**
  - Office of Nuclear Reactor Regulation (NRR)
    - Issues and administers licenses to operate commercial power reactors under 10 CFR Part 50
    - Cognizance over Part 50 rules
  - Office of New Reactors (NRO)
    - Issues and administers licenses to operate commercial power reactors under 10 CFR Part 52
    - Cognizance over Part 52 rules
  - Office of Nuclear Regulatory Research (RES)
    - Supports NRR and NRO needs through short and long term research, and through technical support
    - Cognizance over Regulatory Guides

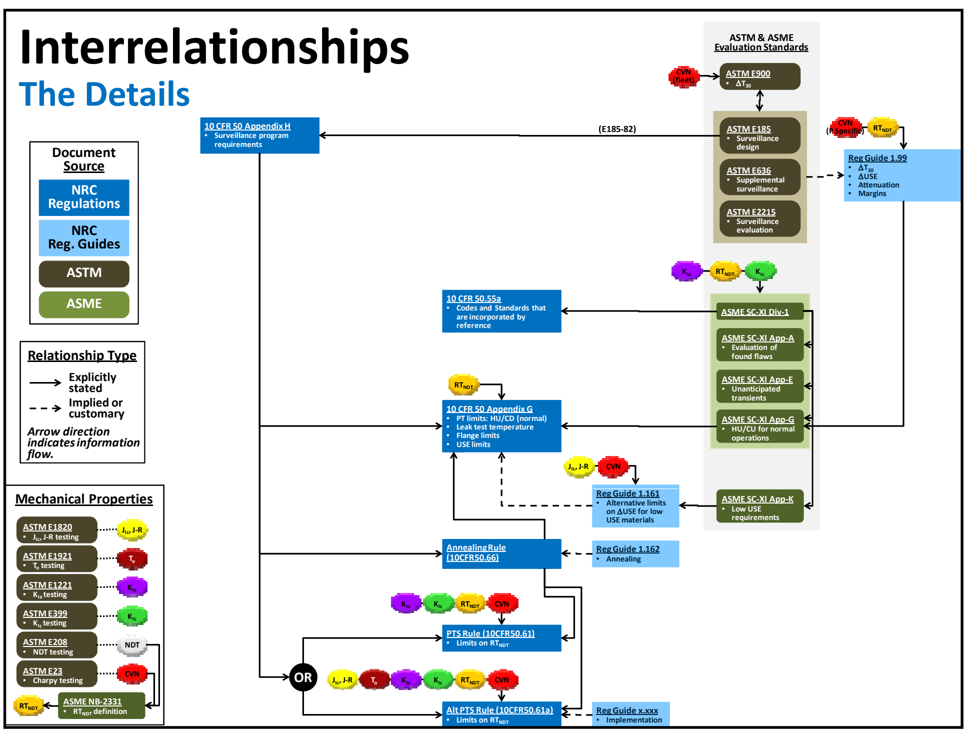
Relationships and Roles
Continued

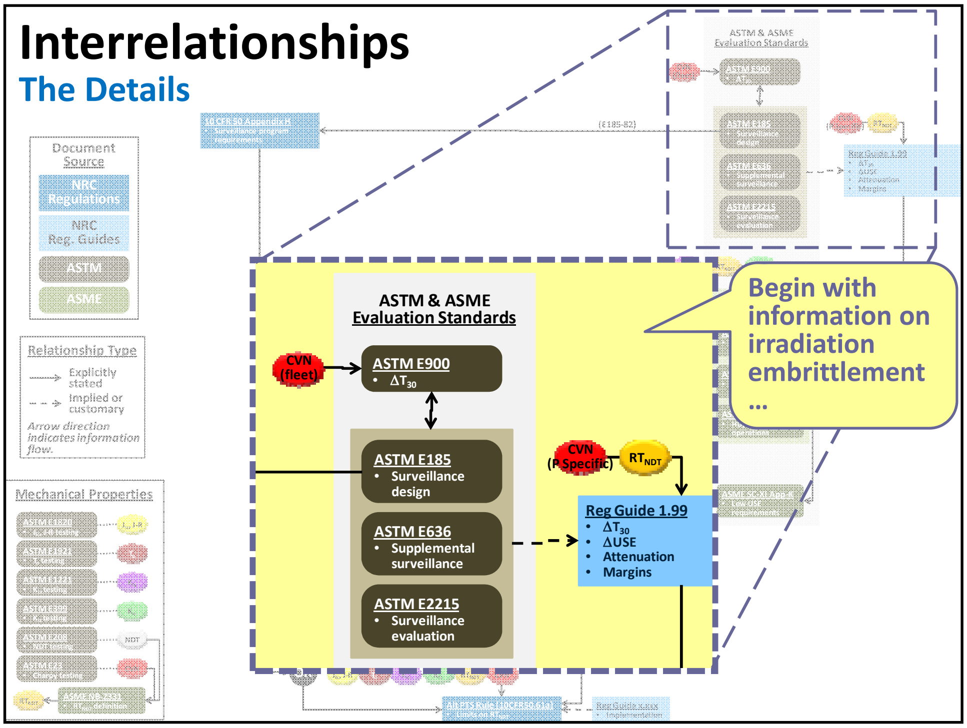
- **ASME**
  - Independent consensus technical body
  - Includes representatives from NRC, industry, academia, and the public
  - Develops / maintains guidelines for the design, inspection, and assessment of nuclear RPVs
    - Section III, Division 1 – *Rules for Construction of Nuclear Facility Components*
    - Section XI – *Inservice Inspection of Nuclear Power Plant Components*
- **ASTM**
  - Independent consensus technical body
  - Includes representatives from NRC, industry, academia, and the public
  - Develops / maintains guidelines for the design, inspection, and assessment of nuclear RPVs
    - Subcommittee E10.02 – *Behavior and Use of Nuclear Structural Materials*

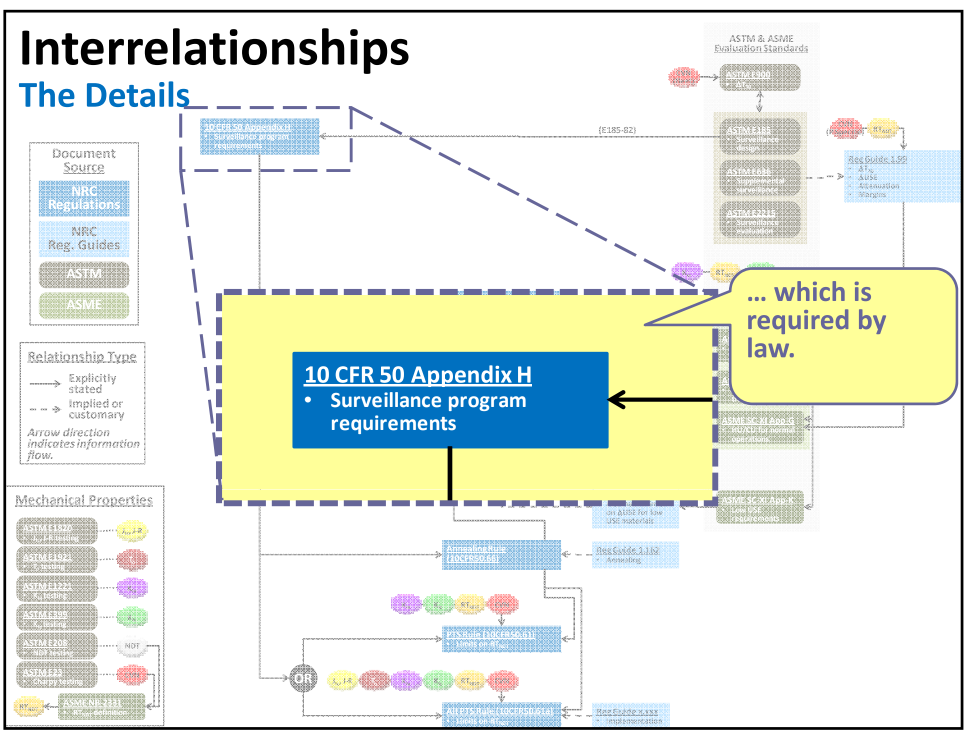
NRC, ASME, ASTM Interrelationships

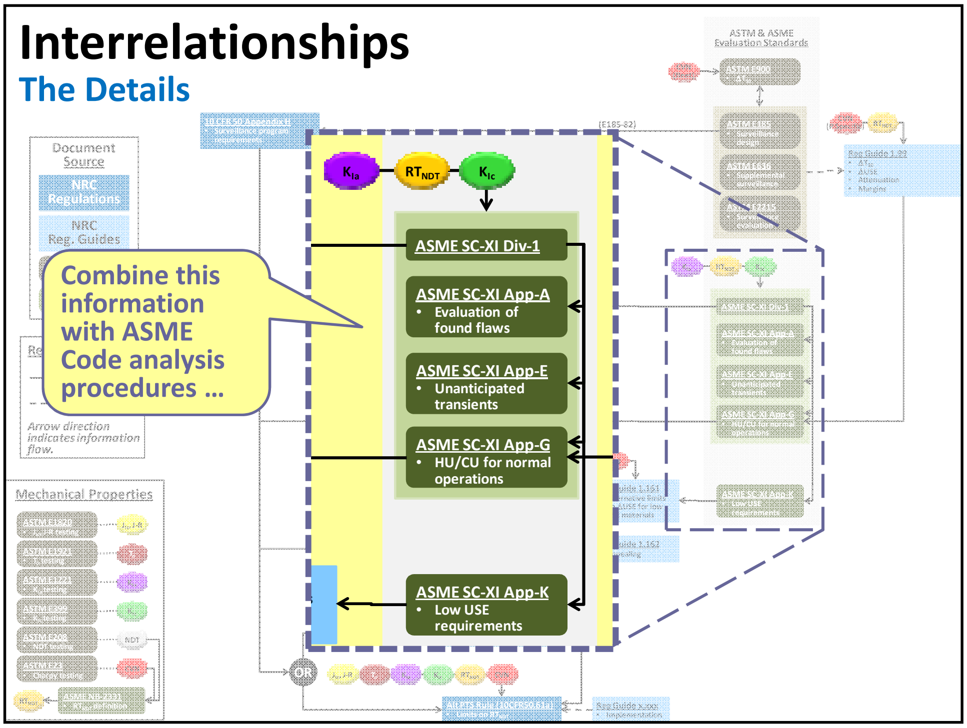
- All of the following are evolutionary and living documents that have developed since the early 1970s
  - **NRC documents**
    - Rules (10 CFR Parts 50 & 52)
    - Regulatory Guides
  - **ASME Code**
  - **ASTM Standards**
- Because they are evolutionary, there is no single source that captures all provisions of the RPV integrity framework
- The following pages illustrate the inter-relationships between these various documents

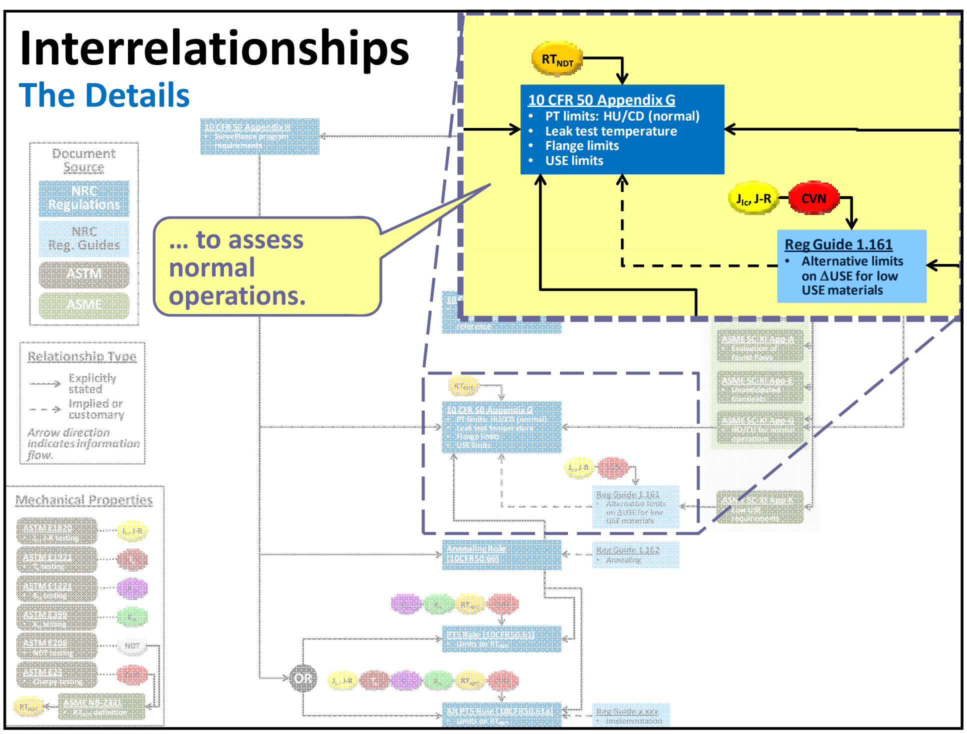

Highest Level
NRC Rules
Start Here
U.S.NRC
10 CFR 50.34 Contents of Applications, Technical Information
The principal design criteria for the facility, Appendix A, General Design Criteria for Nuclear Power Plants, establishes minimum requirements for the principal design criteria for water-cooled nuclear power plants.
Appendix A to Part 50: General Design Criteria for Nuclear Power Plants
Criterion 31—Fracture prevention of reactor coolant pressure boundary.
The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions
(1) the boundary behaves in a nonbrittle manner, and
(2) the probability of rapidly propagating fracture is minimized.
The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining
(1) material properties,
(2) the effects of irradiation on material properties,
(3) residual, steady state and transient stresses, and
(4) size of flaws.
Relationship Type
Explicitly stated
Implied or customary
Arrow direction indicates information flow.
10 CFR 50.60 Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation
… all light-water nuclear power reactors … must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part.
Annealing Rule (10CFR50.66)
Maintenance
PTS Rule (10CFR50.61)
• Limits on RT_{NDT}
Alt PTS Rule (10CFR50.61a)
• Limits on RT_{NDT}
Postulated Accident Conditions
10 CFR 50 Appendix G
• PT limits: HU/CD (normal)
• Leak test temperature
• Flange limits
• USE limits
Normal Operation
10 CFR 50 Appendix H
• Surveillance program requirements
These five rules are interrelated with ASME Codes, ASTM Standards, and NRC Regulatory Guides. These interrelationships are illustrated on the next page

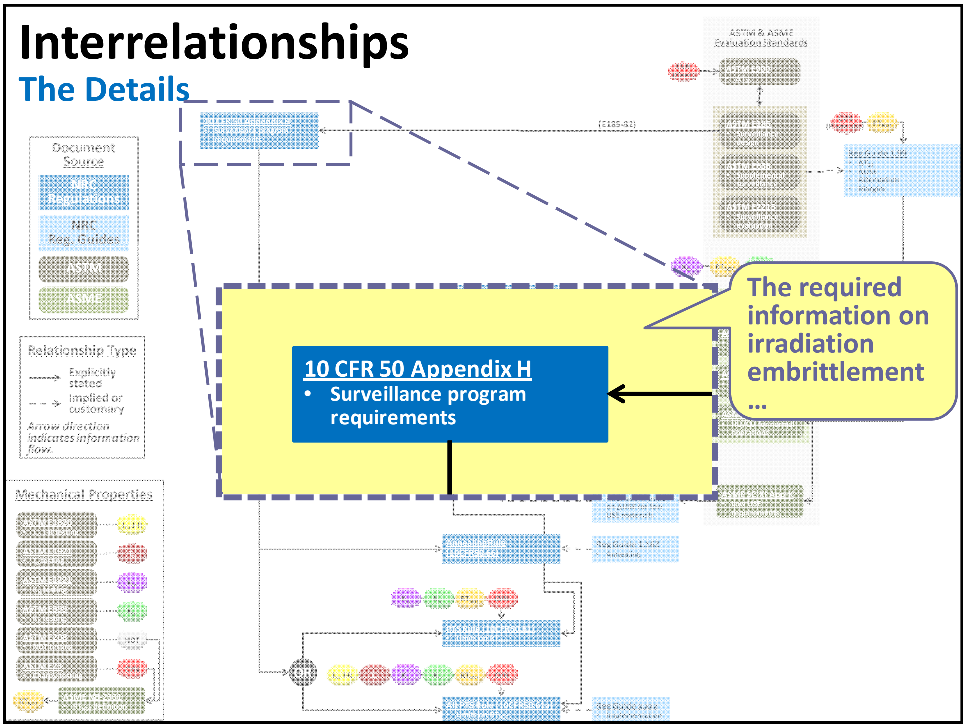

Interrelationships
The Details
Document Source
NRC Regulations
NRC Reg. Guides
ASTM
ASME
Relationship Type
Explicitly stated
Implied or customary
Arrow direction indicates information flow.
Mechanical Properties
ASTM E1820
• J_{Ic}, J-R testing
ASTM E1921
• T_o testing
ASTM E1221
• K_{Ia} testing
ASTM E399
• K_{Ic} testing
ASTM E208
• NDT testing
ASTM E23
• Charpy testing
ASME NB-2331
• RT_{NDT} definition
J_{Ic}, J-R
T_o
K_{Ia}
K_{Ic}
NDT
CVN
RT_{NDT}
10 CFR 50 Appendix H
• Surveillance program requirements
(E185-82)
ASTM & ASME Evaluation Standards
CVN (fleet)
ASTM E900
• ΔT_{30}
ASTM E185
• Surveillance design
ASTM E636
• Supplemental surveillance
ASTM E2215
• Surveillance evaluation
CVN (P Specific)
Reg Guide 1.99
• ΔT_{30}
• ΔUSE
• Attenuation
• Margins
10 CFR 50.55a
• Codes and Standards that are incorporated by reference
ASME SC-XI Div-1
ASME SC-XI App-A
• Evaluation of found flaws
ASME SC-XI App-E
• Unanticipated transients
ASME SC-XI App-G
• HU/CU for normal operations
ASME SC-XI App-K
• Low USE requirements
10 CFR 50 Appendix G
• PT limits: HU/CD (normal)
• Leak test temperature
• Flange limits
• USE limits
Reg Guide 1.161
• Alternative limits on ΔUSE for low USE materials
Annealing Rule (10CFR50.66)
Reg Guide 1.162
• Annealing
PTS Rule (10CFR50.61)
• Limits on RT_{NDT}
OR
Alt PTS Rule (10CFR50.61a)
• Limits on RT_{NDT}
Reg Guide x.xxx
• Implementation

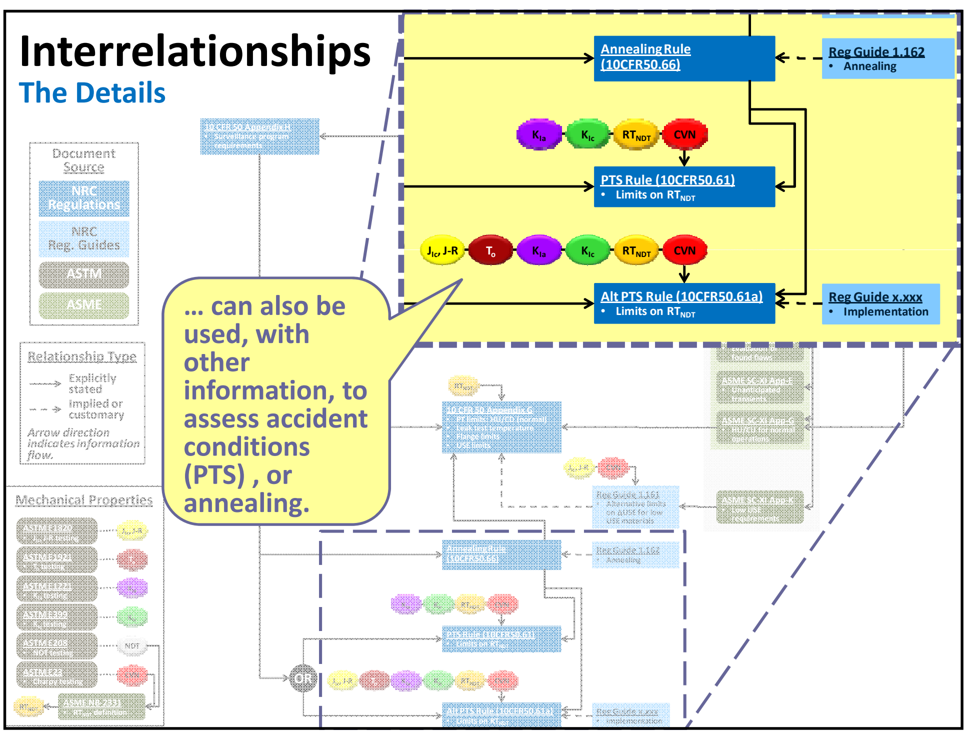

Interrelationships
The Details
Document Source
NRC Regulations
NRC Reg. Guides
ASTM
ASME
Relationship Type
Explicitly stated
Implied or customary
Arrow direction indicates information flow.
Mechanical Properties
ASTM & ASME Evaluation Standards
CVN (fleet)
ASTM E900
• ΔT30
ASTM E185
• Surveillance design
ASTM E636
• Supplemental surveillance
ASTM E2215
• Surveillance evaluation
CVN (P-Specific)
RTNDT
Reg Guide 1.99
• ΔT30
• ΔUSE
• Attenuation
• Margins
Begin with information on irradiation embrittlement …


Interrelationships
The Details
Document Source
NRC Regulations
NRC Reg. Guides
ASTM
ASME
Relationship Type
Explicitly stated
Implied or customary
Arrow direction indicates information flow.
Mechanical Properties
ASTM & ASME Evaluation Standards
10 CFR 50 Appendix H
• Surveillance program requirements
… which is required by law.


Interrelationships
The Details
ASTM & ASME
Evaluation Standards
Document
Source
NRC
Regulations
NRC
Reg. Guides
Combine this information with ASME Code analysis procedures …
Arrow direction indicates information flow.
Mechanical Properties
K_{Ia}
RT_{NDT}
K_{Ic}
ASME SC-XI Div-1
ASME SC-XI App-A
• Evaluation of found flaws
ASME SC-XI App-E
• Unanticipated transients
ASME SC-XI App-G
• HU/CU for normal operations
ASME SC-XI App-K
• Low USE requirements
OR


Interrelationships
The Details
Document
Source
NRC
Regulations
NRC
Reg. Guides
ASTM
ASME
Relationship Type
Explicitly stated
Implied or customary
Arrow direction indicates information flow.
Mechanical Properties
RT_{NDT}
10 CFR 50 Appendix G
• PT limits: HU/CD (normal)
• Leak test temperature
• Flange limits
• USE limits
J_{Ic}, J-R
CVN
Reg Guide 1.161
• Alternative limits on ΔUSE for low USE materials
… to assess normal operations.
OR


Interrelationships
The Details
ASTM & ASME Evaluation Standards
Document Source
NRC Regulations
NRC Reg. Guides
ASTM
ASME
Relationship Type
Explicitly stated
Implied or customary
Arrow direction indicates information flow.
Mechanical Properties
10 CFR 50 Appendix H
• Surveillance program requirements
The required information on irradiation embrittlement …
OR


Interrelationships
The Details
Document Source
NRC Regulations
NRC Reg. Guides
ASTM
ASME
Relationship Type
Explicitly stated
Implied or customary
Arrow direction indicates information flow.
Mechanical Properties
Annealing Rule (10CFR50.66)
Reg Guide 1.162
• Annealing
K$_{Ia}$
K$_{Ic}$
RT$_{NDT}$
CVN
PTS Rule (10CFR50.61)
• Limits on RT$_{NDT}$
J$_{Ic}$, J-R
T$_{0}$
Alt PTS Rule (10CFR50.61a)
• Limits on RT$_{NDT}$
Reg Guide x.xxx
• Implementation
… can also be used, with other information, to assess accident conditions (PTS) , or annealing.
OR

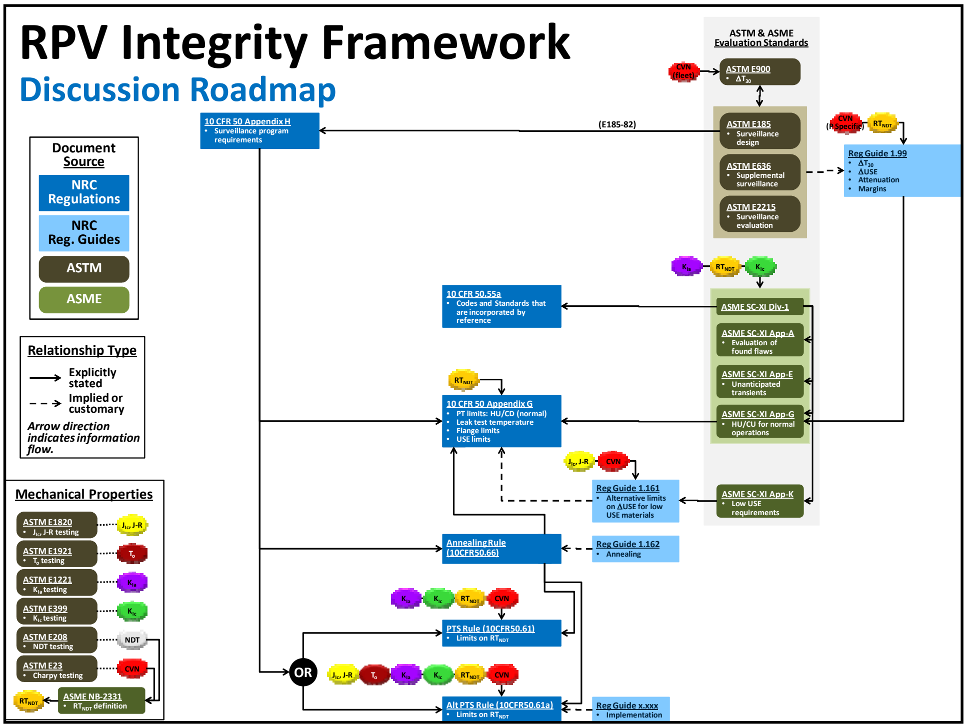
Outline

- **Conceptual framework for RPV integrity**
- **Organizational**
  - Relationships and roles: NRC / ASME / ASTM
  - RPV integrity documents: inter-relationships between NRC, ASME, and ASTM
- **Technical**
  - Evolution of requirements & technical basis for RPV integrity regulations and reg. guides
  - Current status
    - Technical basis
    - Description of efforts to update, if any
  - Experience in use

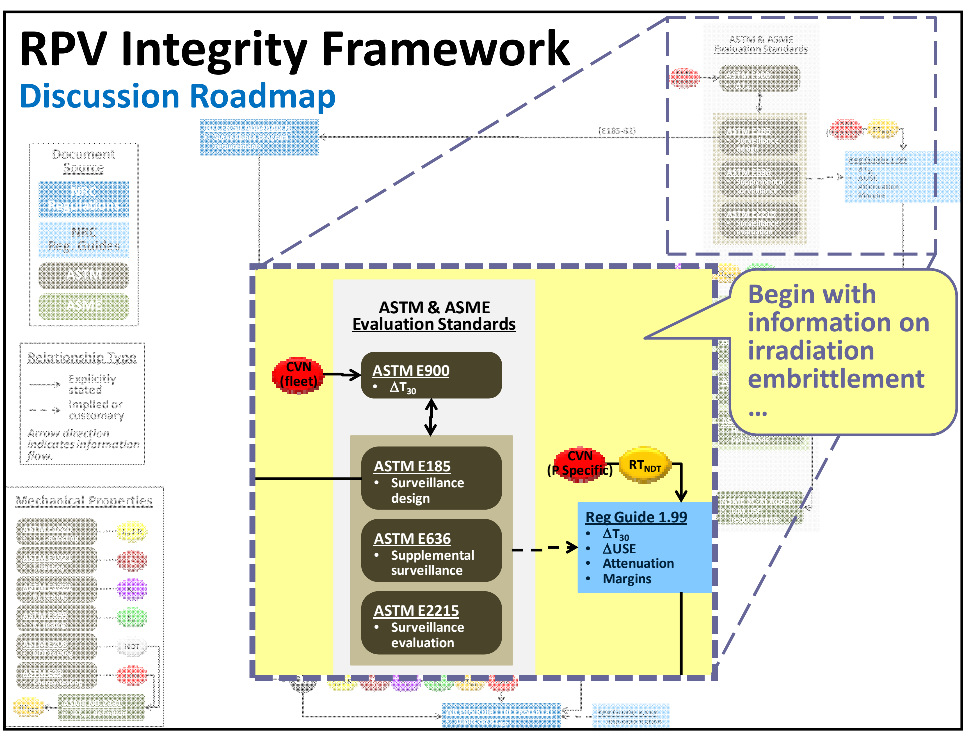
RPV Integrity Framework

List View – Focus on Highlighted Items

NRC Document	Topics	ASTM	ASME
Reg. Guide 1.99 *(Embrittlement Prediction)*	• ΔT_{41J} trend curve • ΔUSE trend curve • Use of surveillance data • Attenuation	• E900	----
10 CFR 50 App. H ***(Surveillance)***	• Materials to include • Tests to perform • Pull schedule	• E185 • E636 • E2215	----
10 CFR 50 App. G ***(Normal Operations)***	• **50 ft-lb limit** • P/T limits • Leak test limits • Flange limits • Beltline definition	----	Section XI Appendix G
Reg. Guide 1.161 ***(Low Upper Shelf)***	• Assessment procedure • J-R curve estimation		Section XI Appendix K
10 CFR 50.61 ***(PTS)***	RT_{PTS} limit	----	----
10 CFR 50.61a ***(PTS)***	• RT_{MAX-XX} limits • Inspection requirements	----	----
---	**Master Curve**	E1921	Code Cases N629 & N631



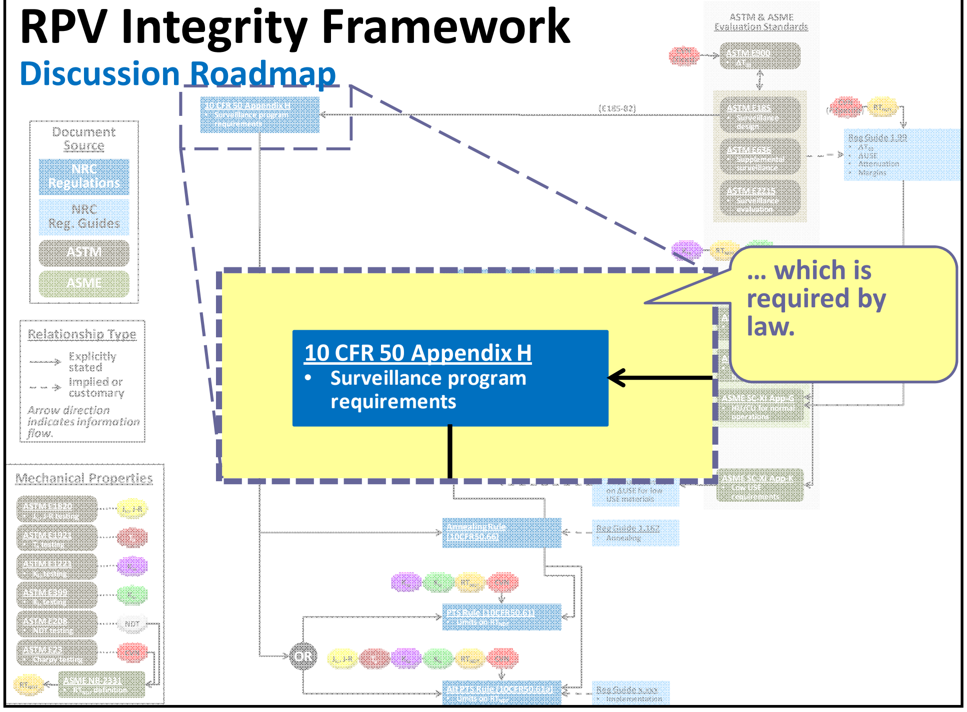
RPV Integrity Framework
Discussion Roadmap
Document Source
NRC Regulations
NRC Reg. Guides
ASTM
ASME
Relationship Type
Explicitly stated
Implied or customary
Arrow direction indicates information flow.
Mechanical Properties
ASTM E1820
ASTM E1921
ASTM E1221
ASTM E399
ASTM E208
ASTM E23
Charpy testing
NDT testing
CVN
NDT
ASME NB-2331
ASTM & ASME Evaluation Standards
ASTM E900
ASTM E185
Surveillance design
ASTM E636
Supplemental surveillance
ASTM E2215
Surveillance evaluation
CVN (fleet)
(E185-82)
10 CFR 50 Appendix H
Surveillance program requirements
Reg Guide 1.99
Attenuation
Margins
10 CFR 50.55a
Codes and Standards that are incorporated by reference
ASME SC-XI Div-1
ASME SC-XI App-A
Evaluation of found flaws
ASME SC-XI App-E
Unanticipated transients
ASME SC-XI App-G
HU/CU for normal operations
10 CFR 50 Appendix G
PT limits: HU/CD (normal)
Leak test temperature
Flange limits
USE limits
Reg Guide 1.161
Alternative limits on ΔUSE for low USE materials
ASME SC-XI App-K
Low USE requirements
Annealing Rule (10CFR50.66)
Reg Guide 1.162
Annealing
PTS Rule (10CFR50.61)
OR
Alt PTS Rule (10CFR50.61a)
Reg Guide x.xxx
Implementation



RPV Integrity Framework
Discussion Roadmap
ASTM & ASME Evaluation Standards
CVN (fleet)
ASTM E900
• ΔT30
ASTM E185
• Surveillance design
ASTM E636
• Supplemental surveillance
ASTM E2215
• Surveillance evaluation
CVN (P Specific)
RTNDT
Reg Guide 1.99
• ΔT30
• ΔUSE
• Attenuation
• Margins
Begin with information on irradiation embrittlement …

RPV Integrity Framework

Discussion Roadmap



10 CFR 50 Appendix H

- This rules concerns surveillance monitoring of RPV embrittlement
  - Incorporates ASTM standards on surveillance by reference

Directly, but the 1982 version

Designation: E185 – 10

Standard Practice for
Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels[1]

Not directly (this was in E185 in 1982)

Designation: E2215 – 10

Standard Practice for
Evaluation of Surveillance Capsules from Light-Water Moderated Nuclear Power Reactor Vessels[1]

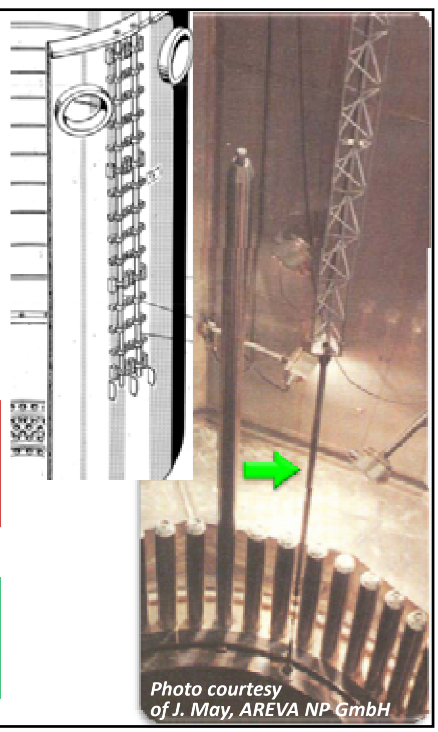


Photo courtesy of J. May, AREVA NP GmbH

ASTM E185 Revision History

TABLE X1.1 Significant Difference Between ASTM E185 Revisions

ASTM E185 Revision	Materials Monitored	No. of Capsules	No. of Unirradiated Specimens	No. of Irradiated Specimens per Exposure Set	Charpy Specimen Orientation	Withdrawal Schedule
1961	Materials used in fabrication	Not specified	Not specified	Charpy – minimum of 8 Tension – not specified	Not specified	Not specified
1966	Base metal with highest init. trans. temp. weld and HAZ	3 or more	Charpy – at least 15 Tension – at least 3	Charpy – 8 or more Tension – 2 or more	Parallel to working direction Notch ⊥ to surface	3 or more different times
1970	Base metal with highest init. trans. temp and flux, weld and HAZ with highest flux	Same as above	Same as above	Charpy – ≥ 8 Tension – ≥ 2 base and weld	Parallel to working direction Notch ⊥ to surface Illustrations provided	1 at 30 % of design life, 1 at EOL, 1 standby
1973	Beltline base, weld, and HAZ metal with highest init. trans. temp, largest shift in trans. temp or decrease in upper shelf energy (USE)	3 or 5 based on predicted temp shift and EOL fluence value	Charpy – at least 15 Tension – at least 3 (not required if temp shift ≥ 100°F or EOL fluence ≥ 5E18 n/cm²)	Charpy – 12 Tension – 2 base and weld only if temp shift ≥ 100°F or EOL fluence ≥ 5E18 n/cm²	Normal to working direction Notch ⊥ to surface Illustrations provided	Specified % of life or specified trans. temp. shift with last capsule as a standby
1979	Beltline base, weld, and HAZ metal with highest init. trans. temp, largest shift in trans. temp, decrease in USE, or most limiting for setting temp/press limits	3, 4, or 5 based on predicted temp shift	Charpy – at least 18 Tension – at least 3 for base and weld only	Charpy – 12 Tension – 3 base and weld only	Same as above	At specified effective full-power years based on shifts in trans. temp with last capsule as a standby
1982	Same as above	Same as above	Same as above	Same as above	Same as above	Same as above
1993	Beltline base, and weld, metal only with highest init. trans. temp, largest shift in trans. temp, decrease in USE, or most limiting for setting temp/press limits	Same as above	Same as above	Same as above	Same as above	Same as above
1998	Same as above	Same as above	Same as above	Same as above	Same as above	Same as above
2002	Same as above	2, 3, or 4 based on predicted temp shift	Charpy – at least 15 Tension – at least 6 for base and weld only Fracture toughness – at least 8	Charpy – at least 15 Tension – 3 base and weld only Fracture toughness – 8 for limiting weld and base metal	Same as above	At specified fluences
2010	Limiting beltline base, weld and other; highest init + shift	3, or 4 based on predicted temp shift	Charpy – at least 15 Tension – at least 6 for base and weld only Fracture toughness – at least 8	Charpy – 15 Tension – 3 base and weld only Fracture toughness – 8 for limiting weld and base metal	Same as above	At fractions of RPV EOL fluence

10 CFR 50 Appendix H

NRC Efforts to Update

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

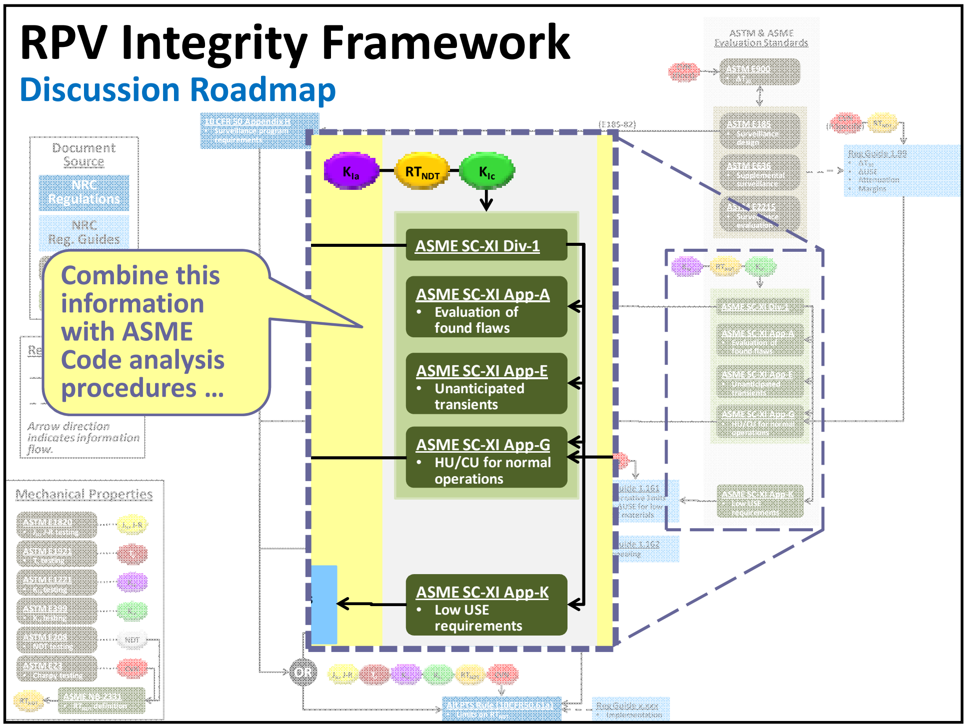
Priority for rulemaking to support update is under discussion

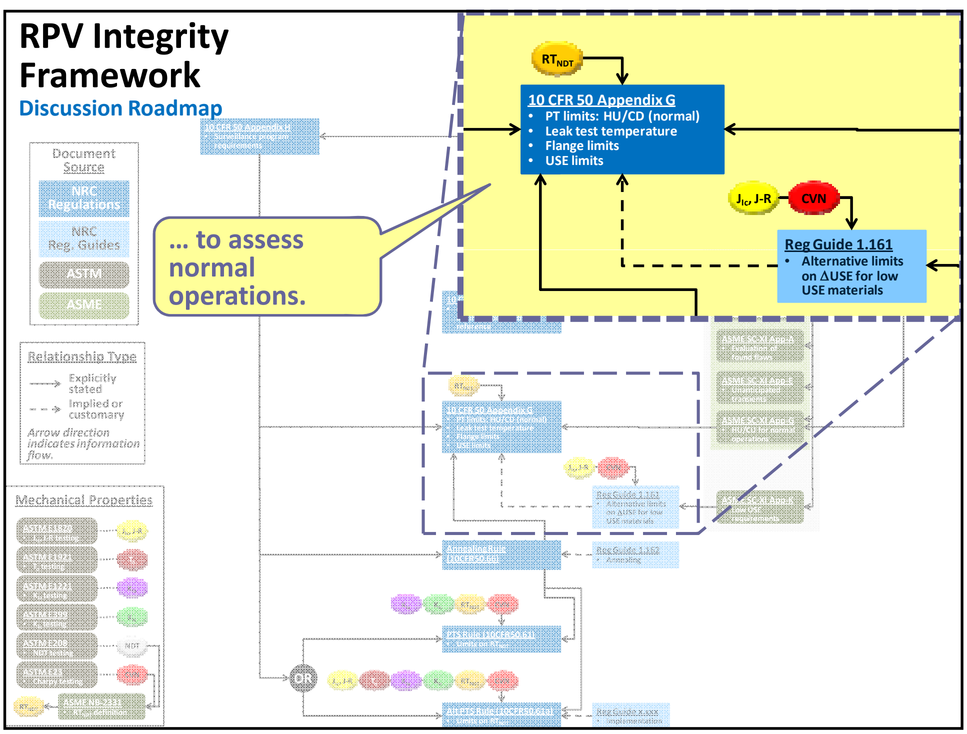
Factors Supported

- Reduction in # of capsules for low shift materials
- Change to fluence-based (rather than EFPY-based) withdraw schedule
- Elimination of HAZ specimens

Factors Not Supported

- E900 trend curve – the trend curve used should be one supported by the NRC
- Requirement that fracture toughness specimens be inserted
  - Option to include toughness specimens is agreed to, and encouraged


RPV Integrity Framework
Discussion Roadmap
Document Source
NRC Regulations
NRC Reg. Guides
Combine this information with ASME Code analysis procedures …
Arrow direction indicates information flow.
Mechanical Properties
K_Ia
RT_NDT
K_Ic
ASME SC-XI Div-1
ASME SC-XI App-A
Evaluation of found flaws
ASME SC-XI App-E
Unanticipated transients
ASME SC-XI App-G
HU/CU for normal operations
ASME SC-XI App-K
Low USE requirements


RPV Integrity Framework
Discussion Roadmap
Document Source
NRC Regulations
NRC Reg. Guides
ASTM
ASME
Relationship Type
Explicitly stated
Implied or customary
Arrow direction indicates information flow.
Mechanical Properties
… to assess normal operations.
RT_NDT
10 CFR 50 Appendix G
PT limits: HU/CD (normal)
Leak test temperature
Flange limits
USE limits
J_Ic, J-R
CVN
Reg Guide 1.161
Alternative limits on ΔUSE for low USE materials

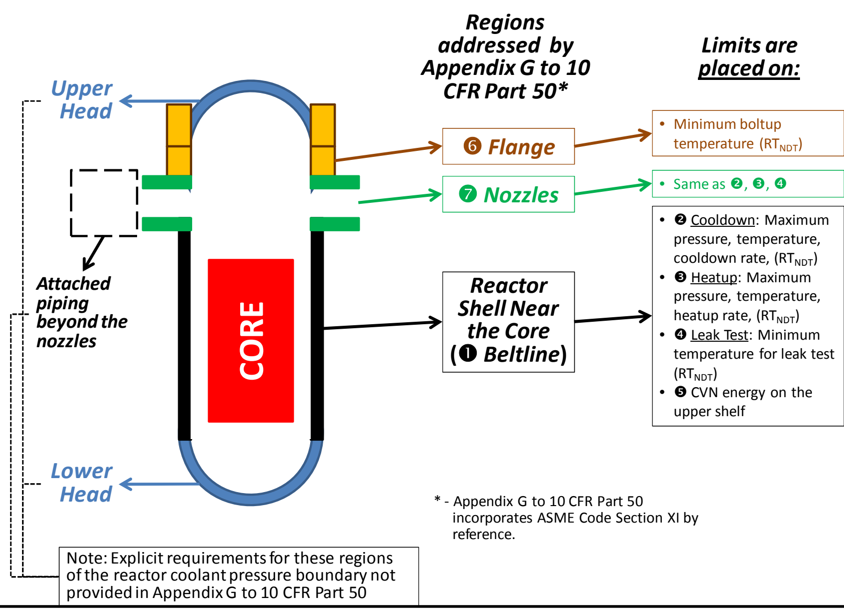
Topics Covered

10 CFR 50 App. G ∪ ASME SC-XI App. G

- 10 CFR 50 Appendix G incorporates ASME Section XI Appendix G, so both are discussed here
- Topics covered
  - Beltline
    - Normal operating P-T limits
      - Cooldown
      - Heatup
      - Leak Test
    - Upper shelf energy (USE) [& RG1.161]
  - Flange
  - Nozzles

Topics Covered

10 CFR 50 App. G ∪ ASME SC-XI App. G



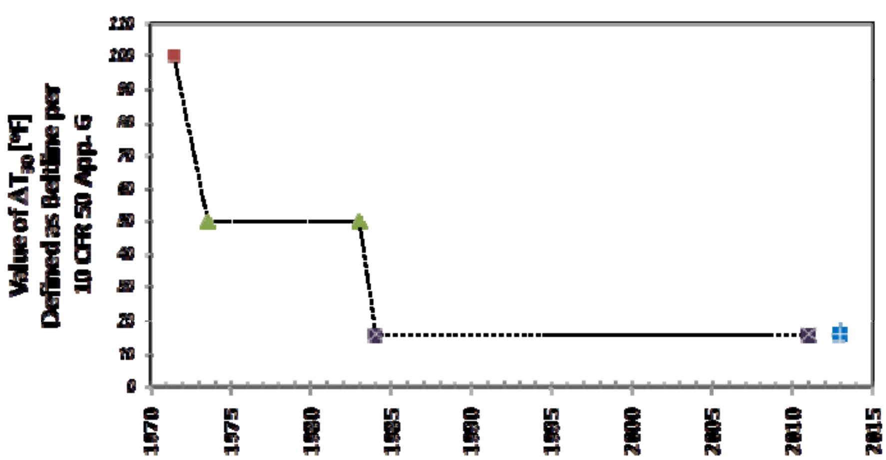
Topics Covered

10 CFR 50 App. G ∪ ASME SC-XI App. G

- 10 CFR 50 Appendix G incorporates ASME Section XI Appendix G, so both are discussed here
- Topics covered
  - Beltline
    - Normal operating P-T limits
      - Cooldown
      - Heatup
      - Leak Test
    - Upper shelf energy (USE) [& RG1.161]
  - Flange
  - Nozzles

Beltline

Summary: Definition of Beltline vs. Time



Beltline

Evolution of Definition – 1980 to 1983

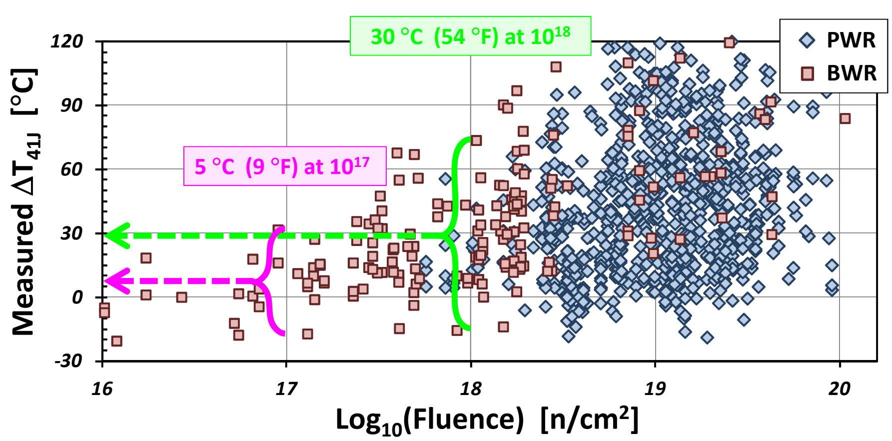
U.S.NRC

- Citations
  - 14 November 1980, Federal Register, Vol. 45, No. 122, pp. 75536-75539.
  - 27 May 1983, Federal Register, Vol 48, No. 104, pp. 24008-24011.
- In 1980-1983 revisions were proposed and made to both Appendix G and Appendix H
  - No changes to Appendix H were made with regards to beltline
  - Appendix G was changed to read: *"Beltline" or "Beltline region of reactor vessel" <u>means</u> the region of the reactor vessel (shell material including welds, heat-affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*
- This change was effective on July 26, 1983 and has remained unchanged since.
- With this change the definition of beltline in Appendix G became implied. Common practice since has been to defined beltline based on the requirements of Appendix H, which requires surveillance for all materials experiencing fluence above 10^{17}.

Beltline

Measured USA Surveillance Data *vs.* Fluence

U.S.NRC



These data were not available in 1973

Topics Covered

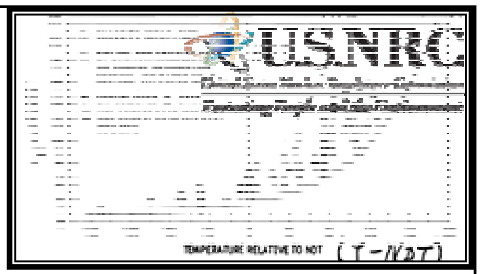
10 CFR 50 App. G ∪ ASME SC-XI App. G

- 10 CFR 50 Appendix G incorporates ASME Section XI Appendix G, so both are discussed here
- Topics covered
  - Beltline
    - Normal operating P-T limits
      - Cooldown
      - Heatup
      - Leak Test
    - Upper shelf energy (USE) [& RG1.161]
  - Flange
  - Nozzles

Normal Operation Limits

WRC-175 (1973)

- Conservatisms in the assessment method
  1. Bounding K_{IR} curve
  2. ¼ T flaw
  3. Safety factors



$$K_{I\,pressure} + K_{I\,therm.} \leq K_{IR} \text{ from Fig. 4-1} \quad (4\text{-}4)$$

Normal Operation Limits

Current & R-I Approach

Requirement of ASME Code for P-T limits & Leak Tests

$$K_{It} + \alpha K_{Im} < K_{Ic}$$

Cladding neglected

For ¼T Flaw

Thermal Loading

$$K_{It} = F_2 \times Rate \times t^{2.5}$$

Pressure Loading

$$K_{Im} = F_1 \sqrt{t} \times (pR_i/t)$$

Fracture Toughness

$$K_{Ic} = 33.2 + 20.734\exp[0.02(T - \{RT_{NDT} + \beta\})]$$

Normal Operation Limits

Current & R-I Approach

Requirement of ASME Code for P-T limits & Leak Tests

$$K_{It} + \alpha K_{Im} < K_{Ic}$$

Cladding neglected

For ¼T Flaw

Thermal Loading

$$K_{It} = F_2 \times Rate \times t^{2.5}$$

Pressure Loading

$$K_{Im} = F_1 \sqrt{t} \times (pR_i/t)$$

Fracture Toughness

$$K_{Ic} = 33.2 + 20.734\exp[0.02(T - \{RT_{NDT} + \beta\})]$$

Leak Test Temperature

$$T_{LEAK} - RT_{NDT(I)} = Margin + \beta + \frac{1}{0.02} \times \ln\left\{\frac{F_2 \times Rate \times t^{2.5} + \frac{\alpha \times F_1 \times P_{OPER} \times R_i}{\sqrt{t}} - 33.2}{20.734}\right\}$$

Max Pressure for PT Limits

$$P_{MAX} = [33.2 + 20.734\exp[0.02(T - \{RT_{NDT} + \beta\})] - F_2 \times Rate \times t^{2.5}] \times \left[\frac{t}{R_i}\right] \times \left[\frac{1}{\alpha F_1 \sqrt{t}}\right]$$

Normal Operation Limits

Current & R-I Approach

		Current	R-I
PL Limit	α	2	1
	β	0	110
Leak	α	1.5	1
	β	0	60
ETC		RG1.99	50.61a
Margin		RG1.99	zero

$$K_{It} + \alpha K_{Im} < K_{Ic}$$

Requirement of ASME Code for P-T limits & Leak Tests

Thermal Loading: $K_{It} = F_2 \times Rate \times t^{2.5}$

Pressure Loading: $K_{Im} = F_1 \sqrt{t} \times (pR_i/t)$

Fracture Toughness: $K_{Ic} = 33.2 + 20.734\exp[0.02(T - \{RT_{NDT} + \beta\})]$

Leak Test Temperature

$$T_{LEAK} - RT_{NDT(I)} = Margin + \beta + \frac{1}{0.02} \times \ln\left\{\frac{F_2 \times Rate \times t^{2.5} + \frac{\alpha \times F_1 \times P_{OPER} \times R_i}{\sqrt{t}} - 33.2}{20.734}\right\}$$

Max Pressure for PT Limits

$$P_{MAX} = [33.2 + 20.734\exp[0.02(T - \{RT_{NDT} + \beta\})] - F_2 \times Rate \times t^{2.5}] \times \left[\frac{t}{R_i}\right] \times \left[\frac{1}{\alpha F_1 \sqrt{t}}\right]$$

Topics Covered

10 CFR 50 App. G ∪ ASME SC-XI App. G

- 10 CFR 50 Appendix G incorporates ASME Section XI Appendix G, so both are discussed here
- Topics covered
  - Beltline
    - Normal operating P-T limits
      - Cooldown
      - Heatup
      - Leak Test
    - Upper shelf energy (USE) [& RG1.161]
  - Flange
  - Nozzles

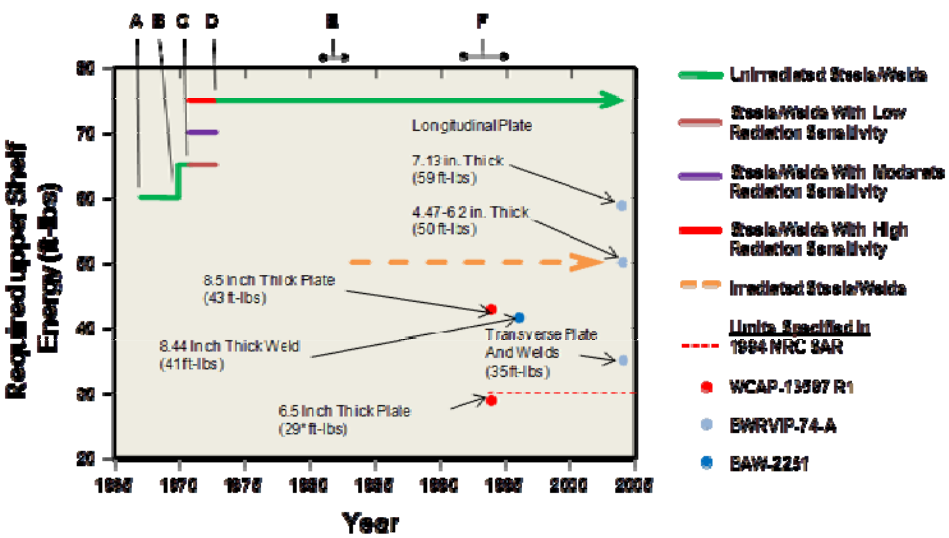
USE Limits

Evolution of USE Limits – Before 1973

- While Appendix G first was being developed ('67-'73)
  - The application of fracture mechanics was in its infancy for thick-walled, relatively tough steels
  - Early USE limits were therefore based on
    - Engineering judgment
    - Perceptions of what toughness (Charpy energy) levels were achievable.
- 10 CFR 50 App. G 1st issued on 17th July 1973

USE Limits

Evolution of USE Limits – Timeline



Equivalent Margins Analysis

- Generic Safety Issue A-11
  - 10 CFR 50 Appendix G requires and analysis whenever USE < 50 ft-lb
  - Task A-11 developed the necessary elastic-plastic analysis methodologies to perform these analyses
  - NUREG-0744, published in 1982, resolved GSI A-11
- NRC and ASME guidance on these elastic-plastic procedures were published in the mid 1990s
  - NRC: Regulatory Guide 1.161
  - ASME: Appendix K to Section XI

Elements of a RG 1.161 Analysis (EMA)

- Requirements
  - No crack initiation (based on $J_{0.1}$)
  - Ductile tearing stability at 0.1-inches crack growth
- J-R curve estimates
  - CVN → J-R correlations
  - Direct measurements

  These are included in RG1.161, ASME Appendix K only says that you need them, it does not tell you how to get them.
- Significant conservatisms exist in many parts of the analysis

$$J_{applied} < J_{material}$$

$$\frac{dJ_{applied}}{da} < \frac{dJ_{material}}{da}$$

<u>Note</u>: A successful EMA satisfies these mathematical requirements. These requirements can be re-arranged (algebra) to infer an acceptable USE that is based on fracture toughness (J-R) data.

Conservatisms in RG 1.161

U.S.NRC

Aspect of EMA		Service Level: A – Normal	Service Level: B – Upset	Service Level: C – Emergency	Service Level: D – Faulted
Flaw Depth		0.25t + 0.1-inch		0.1t + 0.1-inch, not to exceed 1-in. total	
Margin Factor (MF) on material toughness (J_R) - Eq. (3-3)		2σ margins as achieved via MF values from Table 3-1			MF=1 (no margin)
Pressure used to estimate $J_{applied}$		Maximum accumulator pressure (for PWRs, for BWRs)		Maximum operating pressure (≈2,250 For PWRs, ≈1,050 For BWRs)	
Safety Factor (SF) on the $J_{applied}$ value caused by pressure use in the tearing initiation (i.e., tearing > 0.1-in.) calculation (Eq. (3-1))		1.15		1 (no safety factor)	
Safety Factor (SF) on the $J_{applied}$ value caused by pressure use in the tearing stability calculation (Eq. (3-2))		1.25		1 (no safety factor)	
Acceptance Criteria Applied	No initiation of tearing beyond 0.1-in. (Eq. (3-1))	Required			Not required
	Tearing stability (Eq. (3-2))	Required			Required before Δa > 0.75t
	Tensile ligament stability (Eq. (3-15))	Not Required			Eq. (3-15) must be satisfied
Loading Severity		Least	Intermediate (lower)	Intermediate (higher)	Greatest

A/B normally control due to conservatisms

deep

Low estimate

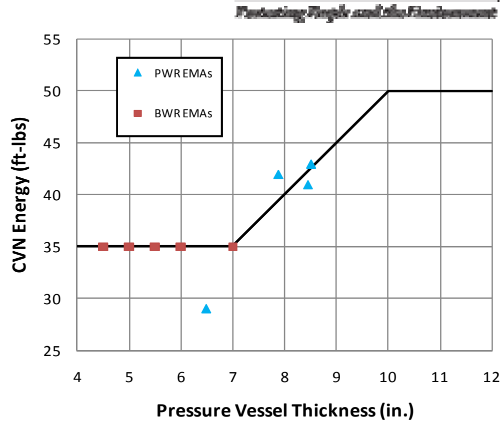
High estimate

Some Experience with EMAs

U.S.NRC

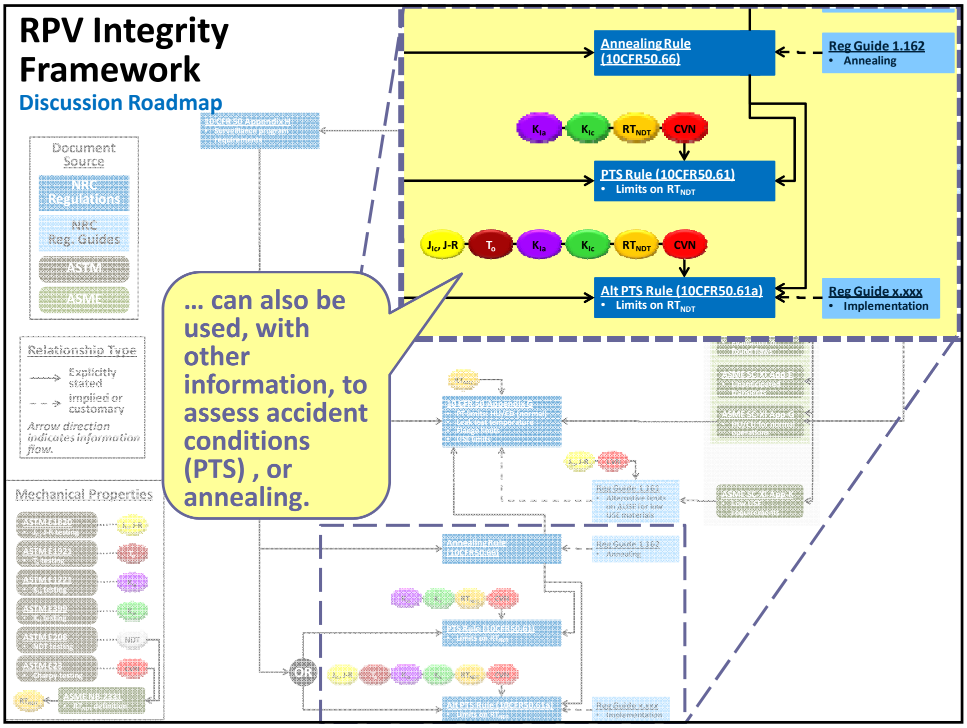
Sample of some EMAs

- For RPVs with $t_{wall} < 7$-in.
  - BWRs and two-loop PWRs
  - Citations
    - WCAP-13587-1
    - BWRVIP-74-A
  - **Accepted USE: 35 & 29 ft-lbs**
  - Caveat: fracture mechanics equations used may not be valid below 30 ft-lbs.
- For RPVs with $t_{wall} \approx 8½$-in.
  - Three and four loop PWRs
  - Citations
    - WCAP 13587-1
    - BAW-2251-A
  - **Accepted USE: 42-43 ft-lbs**



RPV Integrity Framework

Discussion Roadmap



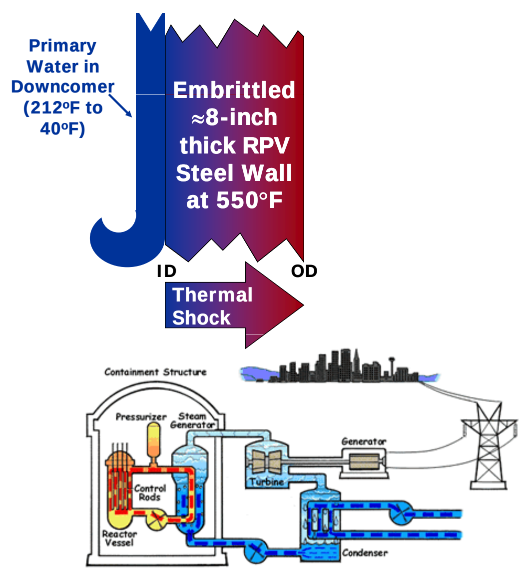
What is PTS?

Primary Side Break

- Inventory (water & steam) lost through the break is replaced by colder (40-70°F) water held in external tanks

Secondary Side Break

- Loss of pressurization in the secondary leaves water boiling (212°F) at atmospheric pressure
- Primary side inventory just across the heat exchanger also approaches 212°F
- Natural circulation in primary draws colder water into downcomer



10 CFR 50.61

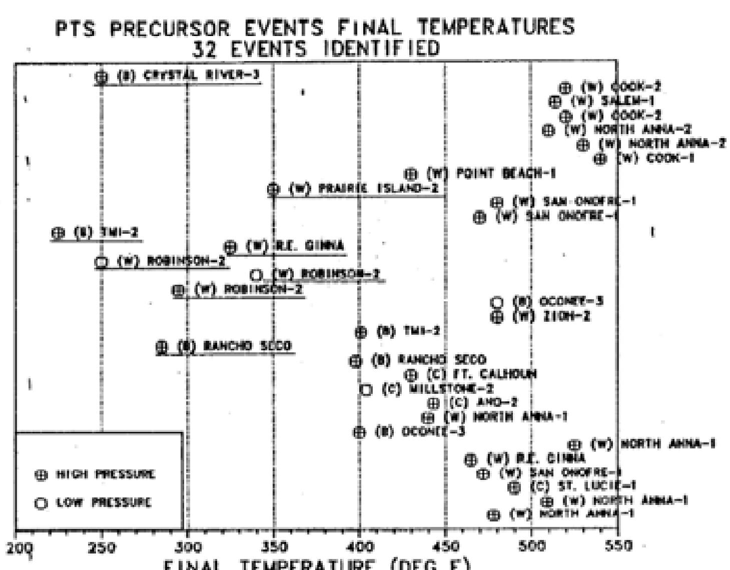
Promulgated 1984

- Motivation (from technical basis: SECY-82-465)
  - *"As a result of operating experience, it is now recognized that transients can occur in PWRs characterized by severe overcooling causing thermal shock to the vessel, concurrent with or followed by repressurization."*
  - *"*[These] *unanticipated loadings … could contribute significantly to the failure probability of the RPV."*
  - *"In addition, operating experience and research programs over the past few years have provided additional information that more clearly defines both material property variations in RPVs and the effect of neutron irradiation on the material's resistance to fracture."*

- Specific motivators (incidents)
  - TMI Action Plan
    - NUREG-0737, Item II.K.2.13 "Thermal Mechanical Report Effect of High-Pressure Injection on Vessel Integrity for Small-Break Loss of Coolant Accident with No Auxiliary Feedwater"
  - Rancho Seco – 20th March 1978 – Excessive feedwater transient
    - *"the RCS was cooled from 582° F to about 285° F in slightly more than one hour (approximately 300°F/hr), while RCS pressure was about 2000 psig."*

10 CFR 50.61

Cooldowns Prior to 1982



[SECY-82-465]

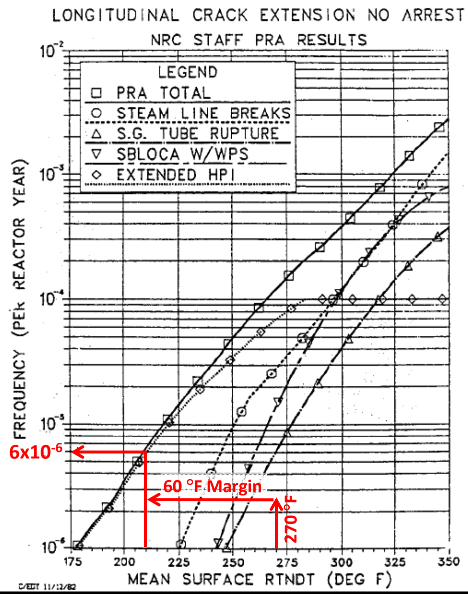
10 CFR 50.61

SECY-82-465 – RT_{PTS} Limits

Interpretation of 270 °F RT_{PTS} Limit

- *"The steepness of the curves … shows a high sensitivity of RPV failure probability to the value of RT_{NDT}. A change in RT_{NDT} as small as 20-30°F changes the calculated probability by a factor of 10 …yet we know neither the actual value of RT_{NDT} for a given RPV, nor the severity of a given transient, to within this order of accuracy. … For this reason, the NRC staff recommends that the PTS criteria, screening or otherwise, should not be determined by where these curves cross some acceptable value of risk. Rather, the probabilistic curves are used to estimate the margin of safety for vessel approaching the screening criterion."*

- *"A plant evaluated to be at the 270 °F screening criterion is likely to have a true RT_{NDT} of 150-270 °F (two sigma is ±60 °F). For the mean of 210 °F,* [this gives] *$F = 6x10^{-6}$ per reactor-year for the NRC curve."*



10 CFR 50.61

SECY-82-465 Outcome

Recommendation

- ***"The risk from PTS events for reactor vessels with RT values less than the proposed screening criterion (270 °F for axial welds, and 300°F for circumferential welds) is acceptable.***

Remediation

- ***"Most plants can avoid reaching the screening criterion … by timely implementation of flux reduction programs."***
- ***"Any plant for which the value of RT is projected to reach the screening criterion before the end of service life … should submit plant-specific evaluations … to determine what, if any, modifications to equipment, systems and procedures should be required …"***

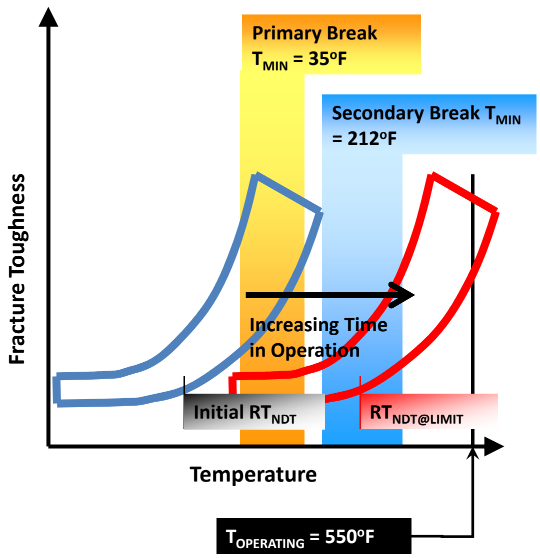
Approach

- ***"It will be evident from this report that the staff is not proposing to resolve PTS issues by requiring a "design-basis pressurized thermal shock event" to be analyzed by a prescribed conservative evaluation model with the results to be compared to specified acceptance criteria. Rather than this traditional approach, the staff has used analyses of overcooling event sequences actually experienced, plus a wide spectrum of possible sequences that have not occurred, together with explicit consideration of the frequencies or probabilities of occurrence of the various events. Moreover, the staff has used analysis models as realistic as the state of the art permits, with a few explicit conservatisms to provide the needed margin of safety. The overall level of safety thus provided has been estimated (very approximately) using probabilistic analysis."***

10 CFR 50.61

RT_{PTS} Limits Max. Embrittlement

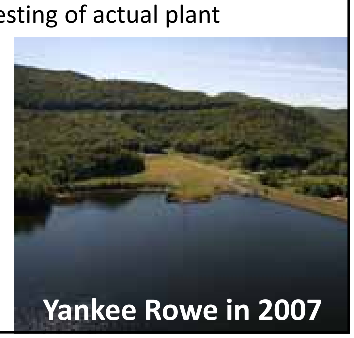
- Embrittlement monitored using surveillance to estimate RT_{NDT}
- If RT_{NDT} exceeds 300 °F (for circ. welds) or 270 °F (for all other materials) before end of license (EOL), the licensee must
  - Do something to keep RT_{NDT} below 300 °F or 270 °F
    - Reduce Flux: Reduce embrittlement rate
    - Anneal: De-embrittle the material (see RG 1.162)
  - Show that RT_{NDT} above 300 °F or 270 °F is safe
    - Analyze: Plant specific analysis per RG 1.154



10 CFR 50.61

Practical Experience – Yankee Rowe (1992)

- To address exceeding the PTS screening limits, YAEC attempted a plant-specific PTS analysis per RG 1.154
  - Important information was not available, including, but not limited to, copper content of the welds, surveillance data, etc.
  - Operating temperature was 500 °F, leading to concerns about additional embrittlement
  - Significant disagreements between the licensee and the NRC regarding the appropriate assumptions and details of the analysis. The NRC review determined that because of the uncertainties the risk may have been greater than previously estimated.
- The NRC recommended shutting the plant until testing of actual plant conditions could be performed to address uncertainties
  - Trepaning to obtain samples of the weld materials
  - Use of UT to measure flaw sizes
- YEAC
  - Concluded the tests costs too much ($23 million) to justify continued operation of a small (185 MW) plant
  - Voluntarily removed the plant from service


Yankee Rowe in 2007

10 CFR 50.61

Practical Experience Since Yankee Rowe

- None
- Several licensees have gotten close to the RT_{PTS} limits, but all have found a means to stay below it, including:
  - Physical means
    - Flux reduction
  - Analytical means
    - Fluence re-calc
    - Master curve
    - Discovering the RPV is actually larger in diameter than was previously thought
    - Using 10 CFR 50.61a

PTS Rule Revision

The Path to 10 CFR 50.61a

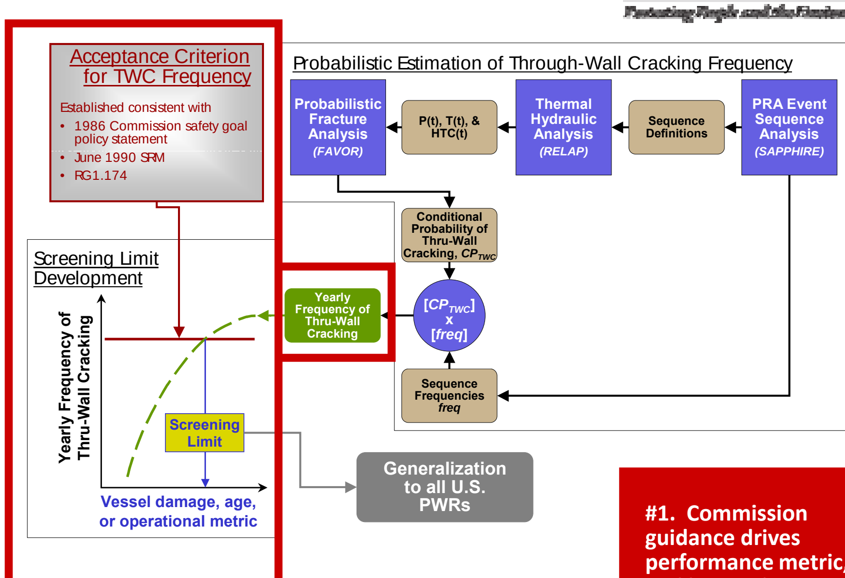
- Why revise the PTS rule?
- Process supporting revision of the PTS technical basis
- Details of the analysis
- Key results
- The alternative PTS rule (10 CFR 50.61a)

Motivations

Developments since the 1980s suggested the overall conservatism of the rule

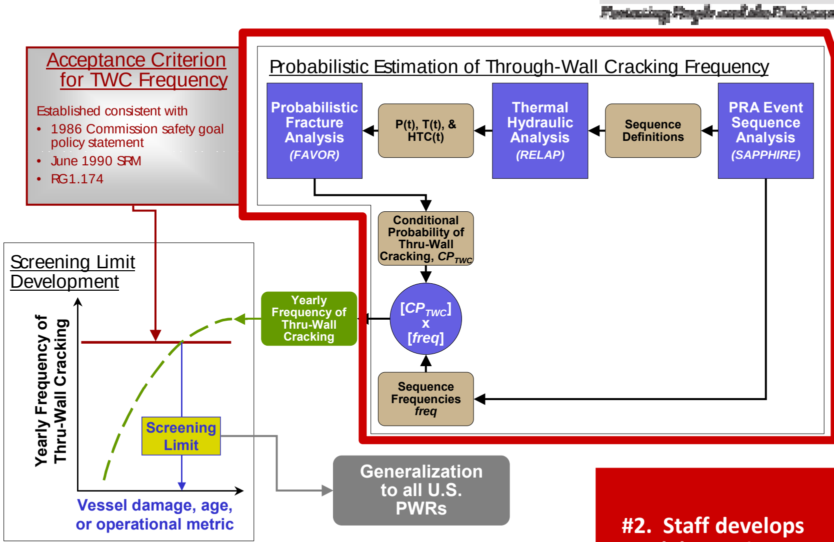
- **PRA**
  - Use of latest PRA/HRA data
  - More refined binning
  - Operator action credited
  - Acts of commission considered
  - External events considered
  - Medium and large-break LOCAs considered
- **TH**
  - Many more TH sequences modeled
  - TH code improved
- **PFM**
  - Significant conservative bias in toughness model removed
  - Spatial variation in fluence recognized
  - Most flaws now embedded rather than on the surface, also smaller
  - Material region dependent embrittlement props.
  - Non-conservatisms in arrest and embrittlement models removed

PTS Project – Overall Approach



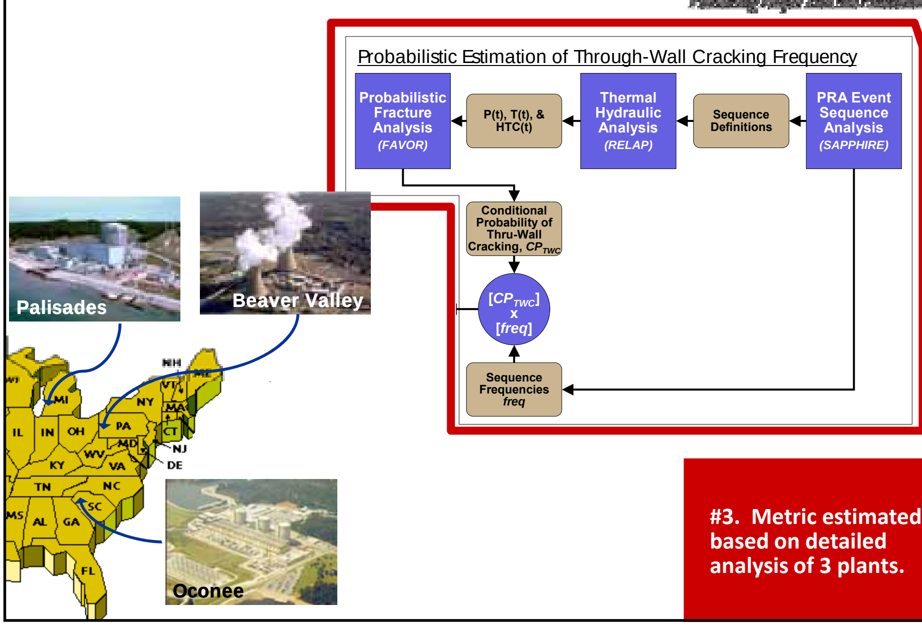
#1. Commission guidance drives performance metric, and limit value.

PTS Project – Overall Approach



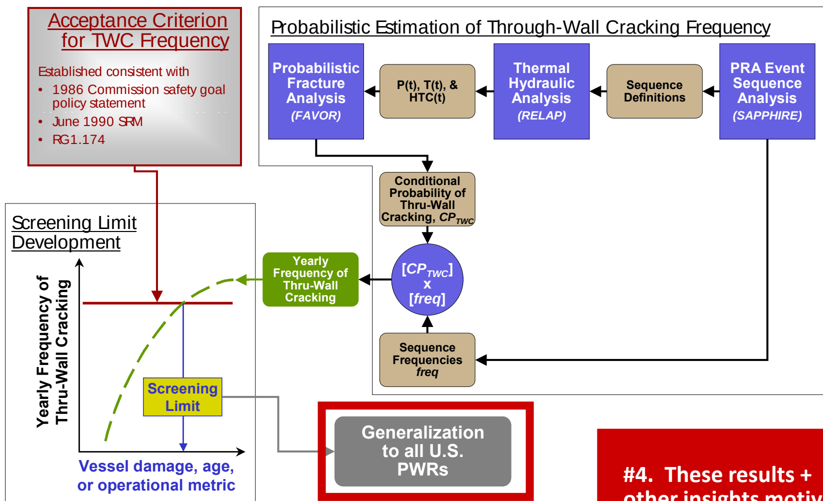
#2. Staff develops model to estimate performance metric.

PTS Project – Overall Approach



#3. Metric estimated based on detailed analysis of 3 plants.

PTS Project – Overall Approach



#4. These results + other insights motivate generalization to all plants.

Linkage of PTS Limits on TWCF to CDF & LERF Policy Decisions

51 FR 28044, *Safety Goal Policy Statement (1986)*

SECY-00-0077, *Modifications to Safety Goal Policy Statement*

Regulatory Guide 1.174

10 CFR 50.61a
Voluntary Alternative Pressurized Thermal Shock Rule

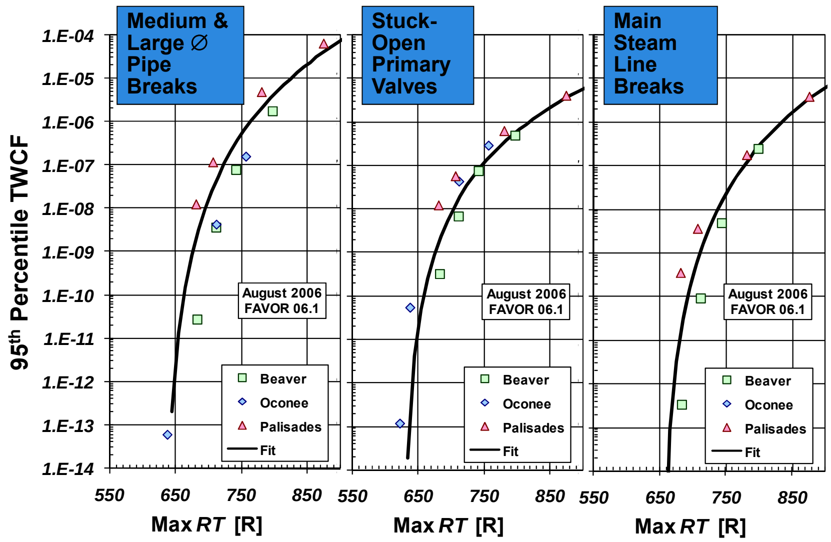
QHOs < 0.1% of the total public risk (prompt & latent)

CDF < 1×10^{-4}/ry
CDF & QHO limits for generic decisions

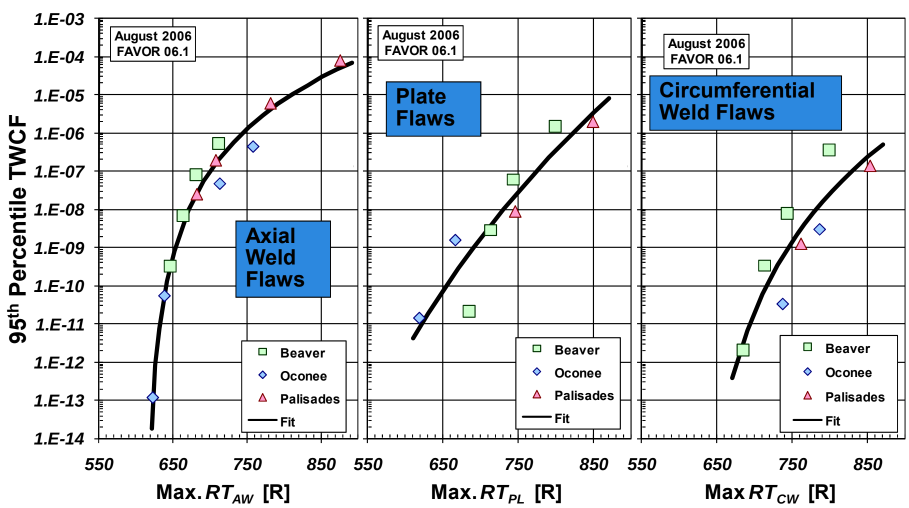
	Mean	Δ-Mean
CDF	10^{-4}/ry	10^{-5}/ry
LERF	10^{-5}/ry	10^{-6}/ry

- Accident sequence progression study shows that through-wall cracking rarely leads to LERF
- Conservatively assumes equivalence of LERF and the yearly through-wall cracking frequency (TWCF) of the reactor pressure vessel
- **Tolerable limit on TWCF established as 10^{-6}/ry**

Important Transient Classes

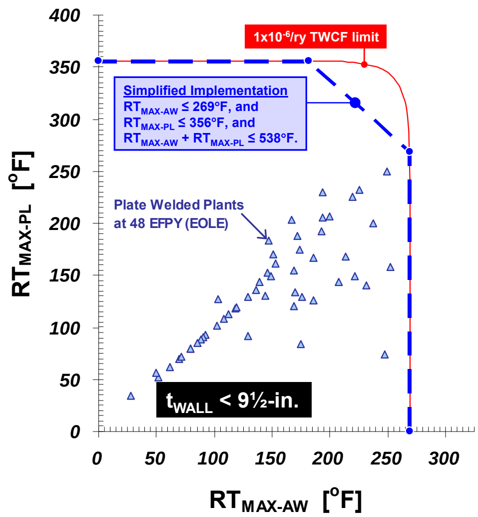


Important Material Features



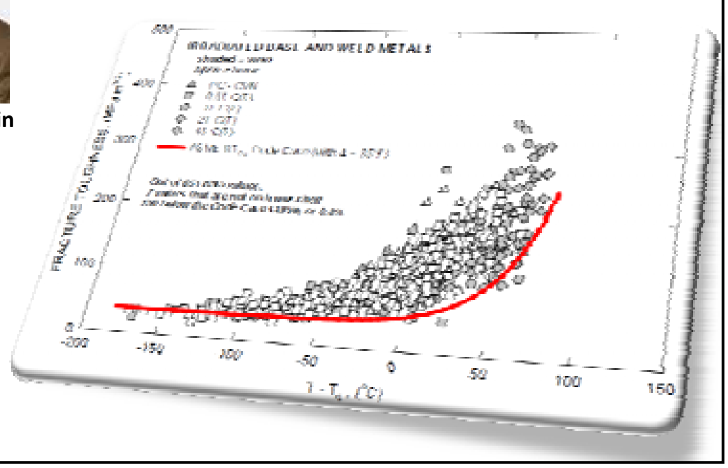
10 CFR 50.61a Summary

- To use less restrictive limits of 10CFR50.61a, plant specific aspects must be checked (defense in depth)
  - Flaw distribution
  - Embrittlement
- Limits apply to all currently operating U.S. PWRs
- All plants assessable based only on available materials and fluence information
- All currently operating PWRs conform to limits, even through 60 years of operation



THE MASTER CURVE

Prof. Kim Wallin
VTT, Finland



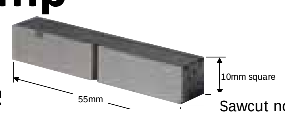
The Master Curve

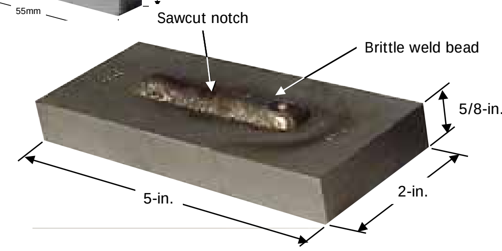
Topics Discussed

- Currently used index temperatures (RT_{NDT})
- Master curve index temperatures (T_o)
- Why is the Master Curve of interest in RPV integrity assessment?
- Examples of how Master Curve has been used in USA RPV regulation
  - Zion … leads to generic estimate of unirradiated RT_{NDT} for Linde 80 welds (BAW-2308)
  - Kewaunee lead plant application
  - Probabilistic fracture model used in developing the alternative PTS limits in 10 CFR 50.61a

Current Index Temp

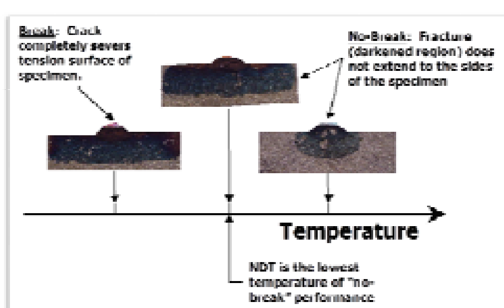
RT_{NDT}





- The transition fracture toughness curve (K_{Ic}) is indexed (positioned) using a parameter that accounts for embrittlement
- The RT_{NDT} index temperature is based on
  - NDT data
  - CVN data

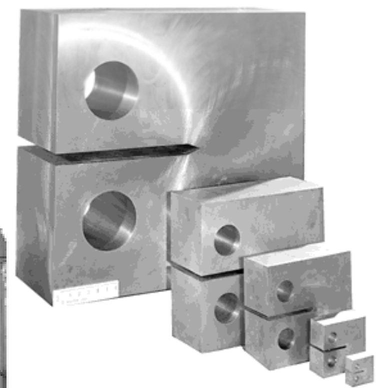
  both of which are ***correlated to*** fracture toughness, but neither of which ***measures*** fracture toughness



Current Index Temperature

RT_{NDT}

- Correlation was used to estimate the RT_{NDT} index temperature because LEFM-valid specimens were too big for practical use

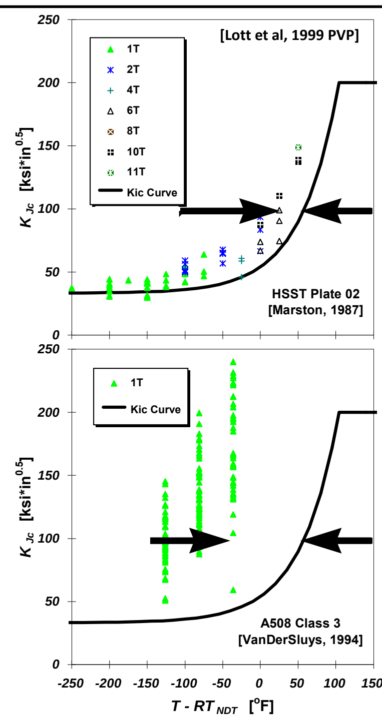


Current Index Temp.

RT_{NDT}

- Since the RT_{NDT} index temperature only correlates to toughness it does not position the bounding K_{Ic} curve consistently relative to the data

Individual data sets

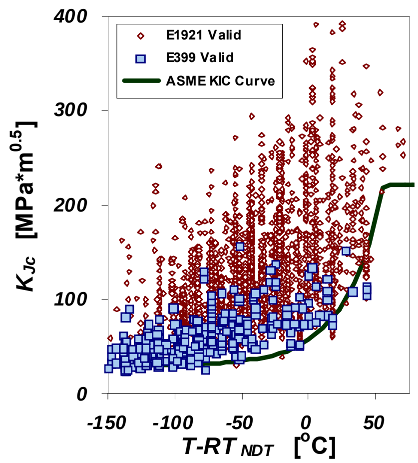


Current Index Temperature

RT_{NDT}

- Since the RT_{NDT} index temperature only correlates to toughness it does not position the bounding K_{Ic} curve consistently relative to the data

A large population of RPV toughness data

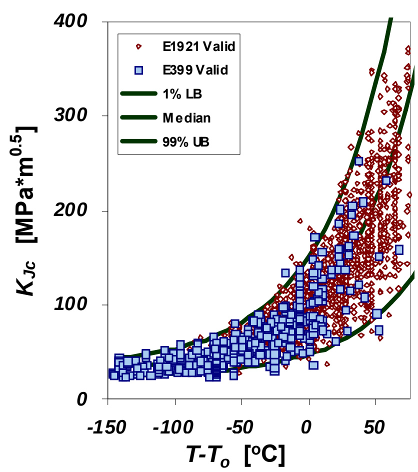


New Index Temperature

T_o

- $T_o \equiv$ temperate at which median K_{Jc} = 100 MPa√m
- Tied to the "Master Curve" as proposed by Wallin in 1984
  - Universal temperature dependence
  - Universal distribution of K_{Jc} data

  for all ferritic steels
- T_o can be measured using reasonably sized specimens (1T C(T), PC-CVN) with ASTM E1921



Comparing the Methods

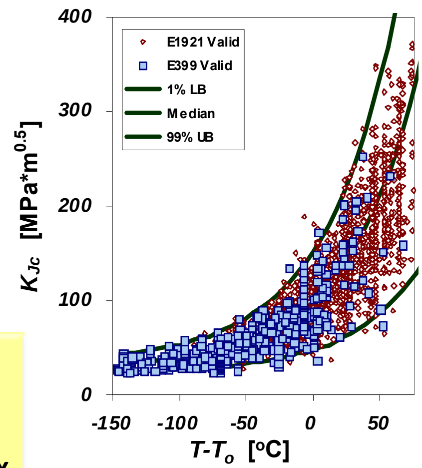
- Current Index Temperature

  RT_{NDT}

- New Index Temperature

  T_o

- T_o orders data consistently across broad populations
- T_o eliminates uncertainty associated with correlative index temperature



Master Curve Applications

… to RPV Integrity Assessment

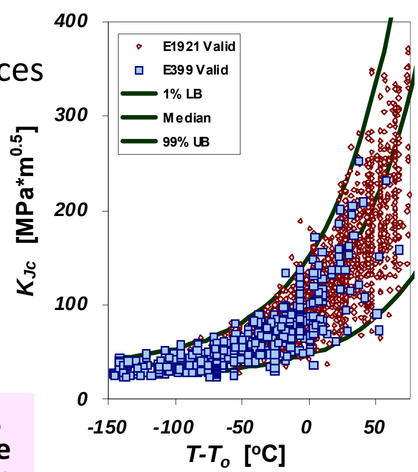
- Advantage: Consistent placement of assessment curve relative to data reduces uncertainty

- Index temperature estimation formula:

$$RT_{NDT(Irrad)} = RT_{NDT(Un\text{-}Irrad)} + \Delta RT_{NDT}$$

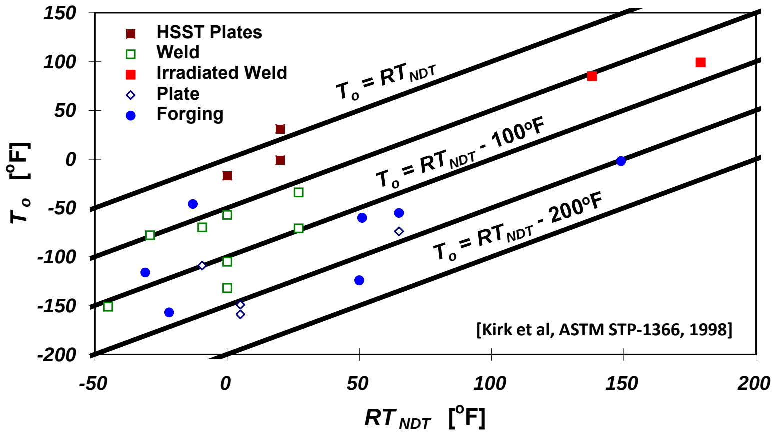
How does unirradiated T_o compare to unirradiated RT_{NDT}?

How does T_o shift compare to RT_{NDT} shift?



Unirradiated Index Temperature

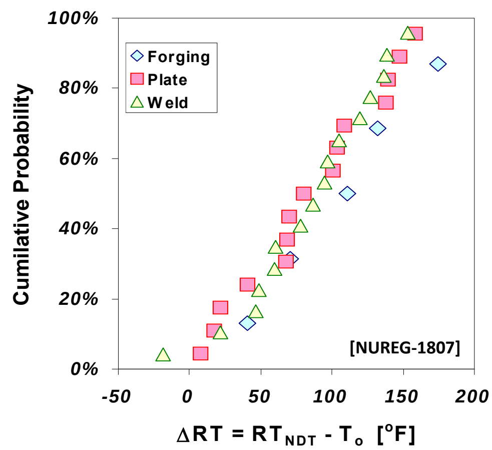
T_o or RT_{To} *vs.* RT_{NDT}



T_o or RT_{To} ranges from about the same as RT_{NDT} to considerably less than RT_{NDT}

Unirradiated Index Temperature

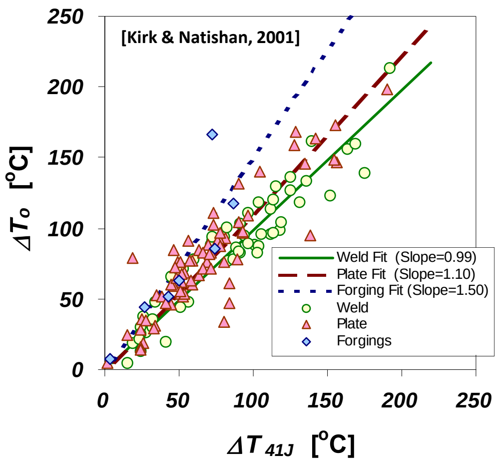
T_o or RT_{To} *vs.* RT_{NDT}



T_o or RT_{To} ranges from about the same as RT_{NDT} to considerably less than RT_{NDT}

Index Temperature Shift

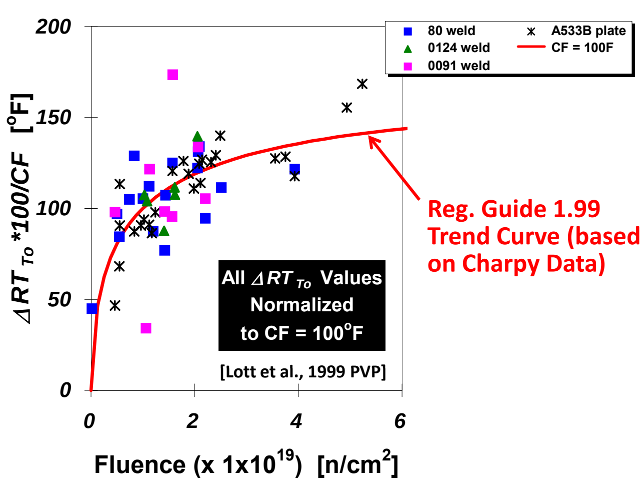
ΔT_o *vs.* ΔT_{41J}



ΔT_o is about the same as ΔT_{41J}

Index Temperature Shift

ΔT_o *vs.* ΔT_{41J}

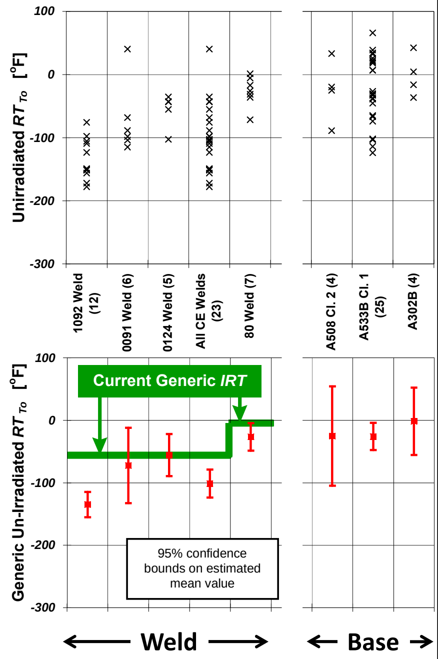


ΔT_o is about the same as ΔT_{41J}

Summary

Index Temperature Comparison

- Unirradiated
  - T_o lower to much lower than RT_{NDT}
- Irradiation shift
  - ΔT_o and ΔRT_{NDT} are about the same
- Suggests that amount of operating life that could be justified by use of Master Curve scales roughly with
  - T_o - RT_{NDT}
  - 1 year of operation is about equal to 1°C as plants approach 40 years of operation



[Kirk et al, 1999 PVP]

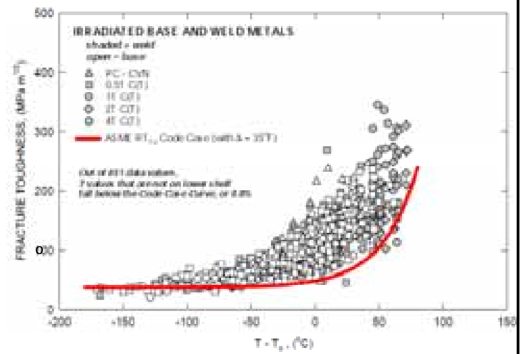
USA Regulatory Application of Master Curve

Year	Event	Key Citations	Notes
1993	Zion Application	• BAW-2202	First use of toughness data instead of Charpy and NDT to set an alternate unirradiated RT_{NDT} value, reduced from -5 to -26 °F
1997	ASME RT_{To} Code Case	• EPRI PWR MRP-1 (TR-108390-R1) • ASME CC-N629	Established technical basis in ASME Code to use T_o to estimate a reference temperature (RT_{To}) that can be used as an alternative to RT_{NDT}
1998-2001	Kewaunee Application	• WCAP-15075 • ML011210180	First application of N-629 code case along with unirradiated and irradiated T_o values to adjust RT_{PTS}
≈2000 - 2005	*Initial RT_{NDT} of Linde 80 Weld Materials*	• BAW-2308-R2 • ML052070408 • ML051180260 • ML081270388	Comprehensive use of toughness data instead of Charpy and NDT to set alternate unirradiated RT_{NDT} values for several plants
≈2000 - 2010	Alternative PTS Rule	• NUREG-1806 • NUREG-1874 • 10 CFR 50.61a	Quasi-master curve use (used toughness data but not full T_o framework) in a probabilistic framework

N629 Code Case

RT_{To} as an Alternative to RT_{NDT}

- $RT_{To} = T_o + 35$ °F provides equivalent bounding (approximately 95%) to that associated with the K_{Ic} curve indexed to RT_{NDT}
  - Since T_o is based on toughness data, the degree of bounding is consistent for all materials, which is not the case for RT_{NDT}
- Applies to irradiated as well as un-irradiated RPV steels



Kewaunee Lead Plant

1st Regulatory Use of N-629

Addressed Regulatory Concerns

- With Master Curve technology
  - Universal curve shape
  - Universal scatter characterization
  - Statistical size effect
- With Master Curve applications
  - Use of pre-cracked Charpy specimens
  - Use of Charpy embrittlement trend curve with ΔT_o data
- With use of weld data from sister plants
  - 1P3571 data obtained from both the Kewaunee and Maine Yankee reactors

-50 °F	$RT_{NDT(u)}$
- -109 °F	$RT_{To(u)}$
59 °F	Potential gain

297 °F	Old RT_{PTS} @ EOLE
-288 °F	New RT_{PTS}
9 °F	Actual gain

BAW-2308

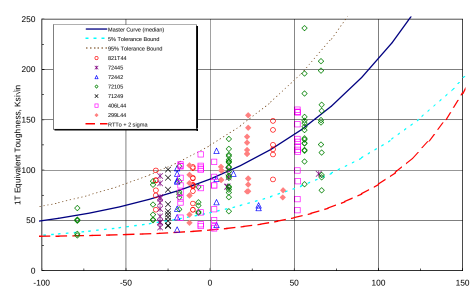
Objective

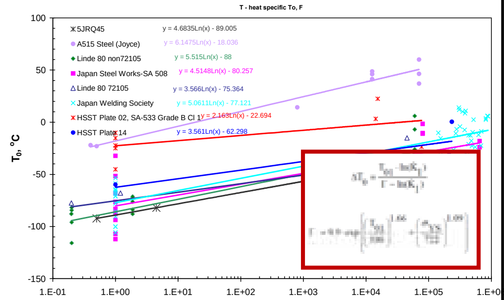
"Establish alternative IRT_{NDT} (RT_{To}) and associated uncertainty (σ_I) for the unirradiated Linde 80 welds for the B&W Owners Group member utilities in their licensing calculations. This alternative IRT_{NDT} is obtained by using the B&W Owners Group Master Curve reference temperature database and ASME Code Case N-629."

BAW-2308 IRT Computation

- Based on 314 fracture toughness tests
- Initial reference temperature begins with T_o and is adjusted to account for:

Factor		Amount
RT_{To} Code Case (N-629)		+35 °F
Pre-cracked Charpy specimen bias (constraint loss)		+18 °F
Loading rate adjustment		See figure
σ_I –SRSS Combo	test procedure, material variability	7 - 17 °F *(heat specific)*
	finite sample size	2 - 10 °F *(heat specific)*





BAW-2308 (Rev 2)
Outcome

Basis	Linde 80 Heat ID	IRT_{To} [°F]	Margin: σ_I [°F]
Master Curve Toughness Data	406L44	-98.0	11.6
	71249	-53.5	12.8
	72105	-31.1	13.7
	821T44	-84.2	9.6
	299L44	-74.3	12.8
	72442	-33.2	12.2
	72445	-72.5	12.0
	61782	-58.5	15.4
	Generic Value for Other Heats	-48.6**	18.0
RT_{NDT} Based Values	All heats	-7 to +10	17

** Includes 20 °F addition to address NRC's concerns concerning adequate bounding.

BAW-2308
Applications

Utility	Plants	NSSS Vendor	Date of BAW 2308 Usage
Entergy Operations, Inc.	**ANO 1**	**B&W**	
FirstEntergy Nuclear Operations, Inc.	**Davis Besse**	**B&W**	**2010**
Duke Energy Company	**Oconee 1,2,3**	**B&W**	**2011** ***requested***
Exelon Corporation	**TMI-1**	**B&W**	***request expected soon***
Florida Power Corporation	**Crystal River-3**	**B&W**	
Florida Power & Light Company	**Turkey Point 1, 2**	**W**	**2010**
Dominion	**Surry 1, 2**	**W**	**2007**
Nuclear Management Company	**Point Beach 1, 2**	**W**	**2003**

Master Curve
Summary

- Use of toughness data instead of NDT and Charpy data enables
  - reduction of transition temperature estimates, or
  - increase in screening limits

  justified based on
  - more accurate knowledge about actual toughness properties, and
  - reduction in implicit margins
- The magnitude of benefit justified by toughness data has increased in the last two decades because of increased familiarity and use in regulatory practice
- Benefit can be obtained within either "deterministic" or "probabilistic" assessments