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Basics of Operating Criteria and Material Property Requirements for RPVs in the US

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Presentation Outline

- Relevant Codes/Standards/Regulations
- Other Aspects of RPV Surveillance

Relevant Codes/Standards/Regulations

Relevant Codes/Standards/Regulations

- Overview
- RT_{To}
- Pressurized thermal shock
- Low upper shelf energy (USE) materials
- Normal heatup and cooldown (P/T limits)
- Low temperature overpressurization (LTOP) in PWRs
- BWR cold hydro and leak test temperatures

ASME Code Rules for Prevention of Brittle Fracture in Vessels

- ASME Section III, Appendix G fracture mechanics method established margins for safe operation (1972)
- ASME Section III method became mandatory for all plants through 10CFR50, Appendix G (1972)
- ASME Code rules for in-service inspection established in new Section XI (1970)
- Reference nil-ductility transition temperature (RT_{NDT}) index defined in ASME Section III, Subsection NB-2331 for establishing reference toughness (1972)
- ASME Section XI, Appendix E established Code methodology to evaluate overpressurization and overcooling events and the effect on reactor vessel integrity (1990)
- ASME Section XI adopted Appendix G method to determine allowable pressure-temperature limits (1992)
- Code Case N-512 approved by ASME Code for low upper shelf energy materials evaluations (1993)
- Code Case N-514 issued and Appendix G procedure established for determining LTOP setpoints (1993)

NRC Regulations on Reactor Vessel Embrittlement and Integrity

- Code of Federal Regulations adopted 10CFR50, Appendices G and H requirements for fracture toughness and materials surveillance (1972)
- Requirements to maintain 50 ft-lbs upper shelf energy first defined in 10CFR50, Appendix G (1972)
- Regulatory Guide (RG) 1.99, Rev. 1 established embrittlement trend curve prediction method (1977)
- PTS Rule published in 10CFR50.61 established PTS screening criteria limits (1984)
- RG 1.154 issued for plant-specific analyses of plants exceeding PTS screening criteria (1987)
- RG 1.99, Rev. 2 updated trend curves to include effects of Cu and Ni for predicting embrittlement in vessel materials (1988)
- 10CFR50.61 (PTS Rule) updated to include RG 1.99, Rev. 2 trend curve method (1990)
- RG 1.161 issued for evaluating RPV materials with USE < 50 ft-lbs (1995)
- Thermal Annealing Rule (10CFR50.66) and Regulatory Guide 1.162 issued (1995)
- Code Cases N-629 and N-631 approved (1999)

Alternative Reference Fracture Toughness Temperature RT_{To}

- Basis
- Application to irradiated materials

RT_{To} Alternative Index Parameter to RT_{NDT}

- ASME Code Case N-631 (unirradiated data for Section III) and N-629 (unirradiated and irradiated data for Section XI) define an alternative index (RT_{To}) for the ASME reference toughness curves
- RT_{To} is derived from the Master Curve T_o value of ASTM E 1921 as:

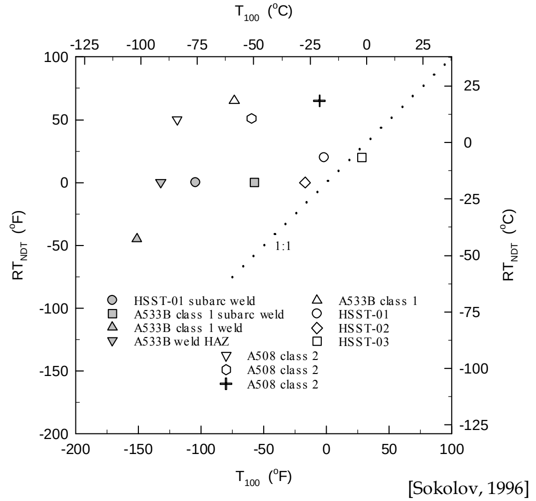
$$T_o + 35^oF = T_o + 19.4^oC$$

- RT_{To} is a direct measure of reference temperature for irradiated materials, but application may require some normalization to project to slightly different fluences

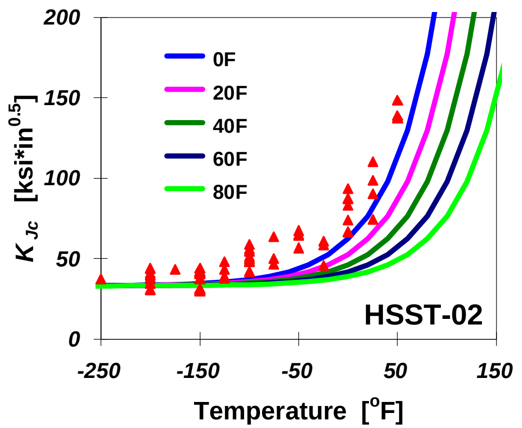
Non-Relationship Between RT_{To} and RT_{NDT}

What RT_{To} is *not*, what RT_{To} is, and why we selected 35°F?

RT_{To} is *not* correlated with RT_{NDT}



RT_{To} appropriately bounds available data (that's why 35°F)

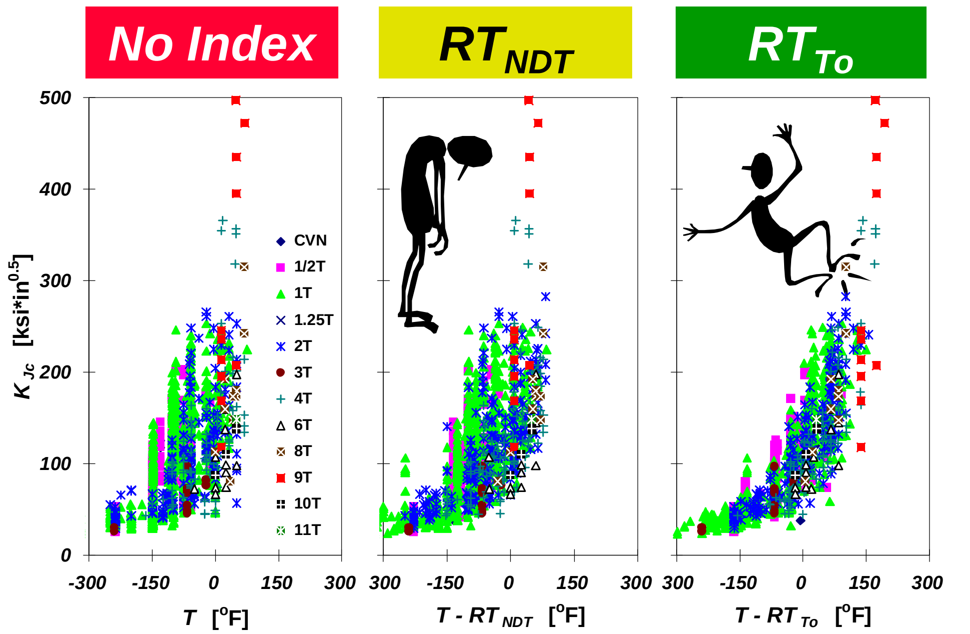


$$RT_{To} \equiv T_o + 35^\circ \mathrm{F}$$

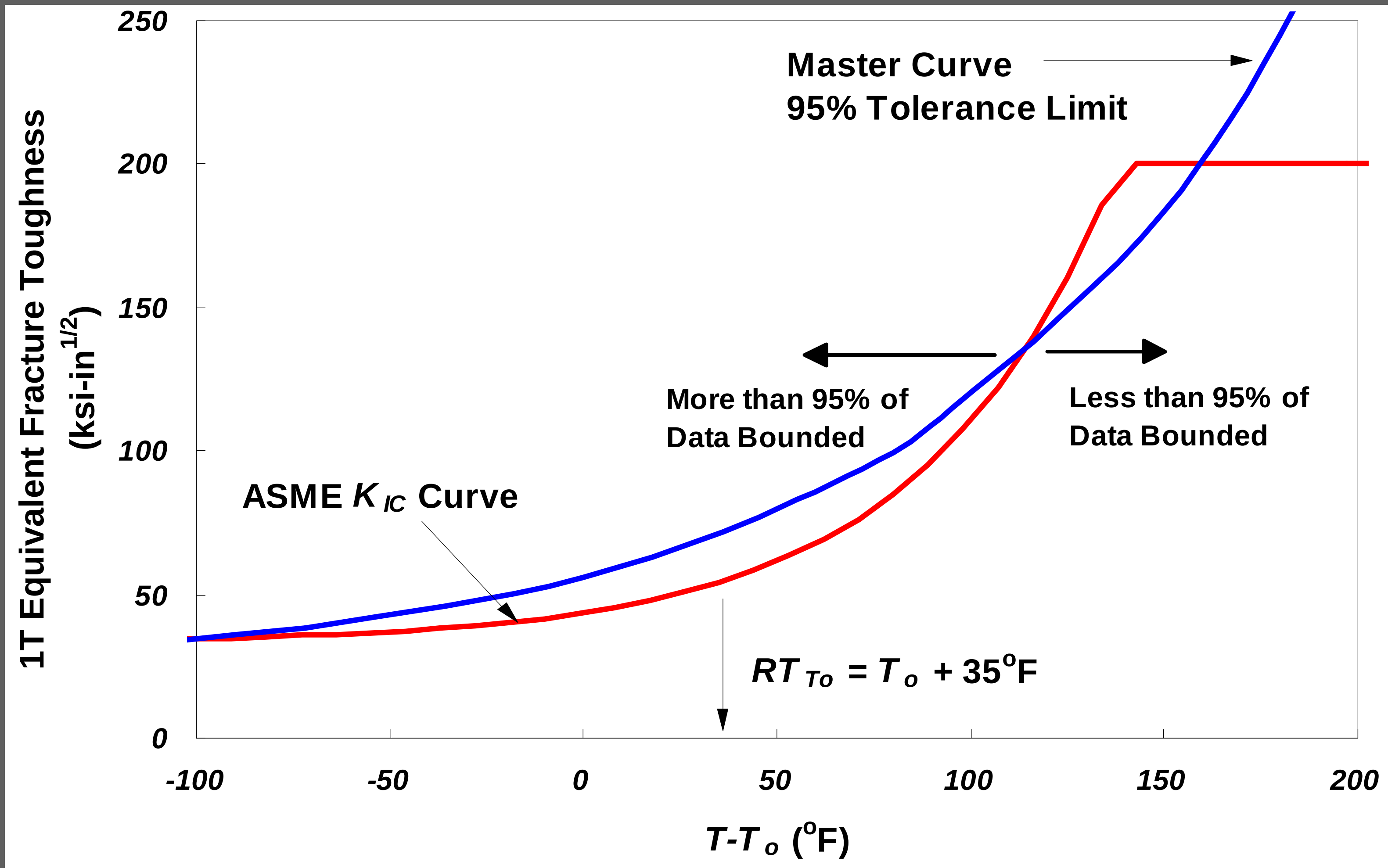
VG 7

Master Curve and RT_{To} Works

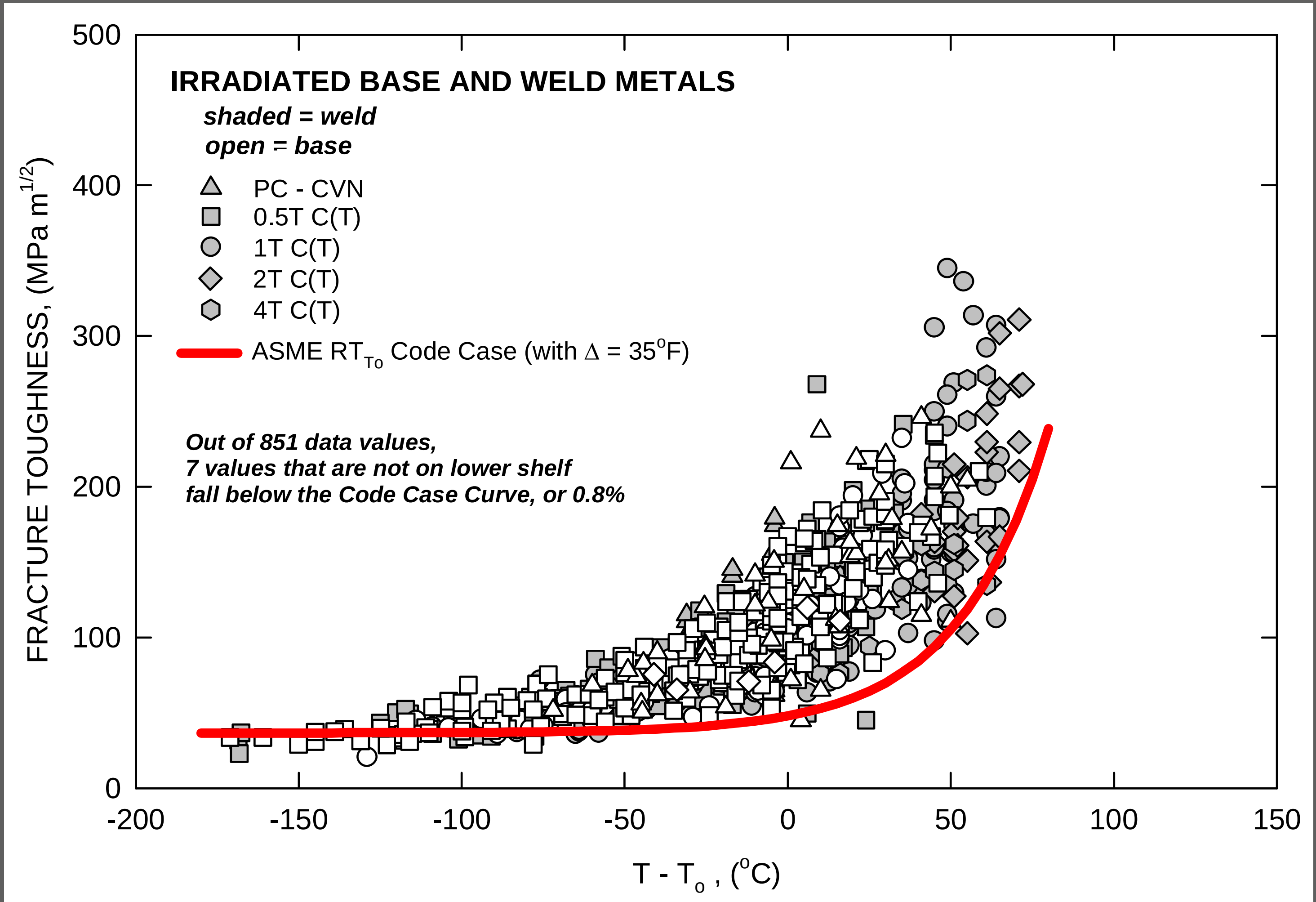
RT_{To} Reduces Scatter Relative to RT_{NDT}



Master Curve 5% Tolerance Limit & Code K_{IC} Curve (Using RT_{To})



Measured Irradiated Toughness Data Also Are Bounded



Application to Several USA Reactor Pressure Vessels

- B&W Owners Group indirect justification for a lower initial RT_{NDT} for Zion-1/2 Linde 80 weld metals – approved
- Kewaunee application to Linde 1092 circumferential weld metal – approved
- Beaver Valley-1 application to limiting SA533B-1 plate – submitted to NRC, but later withdrawn
- B&W Owners Group (B&WOG) generic application for Linde 80 weld metal heats using initial unirradiated RT_{To} plus Charpy shift – now approved by NRC
- Point Beach-2 application to Linde 1092 axial welds –eventually applied B&WOG approach

Pressurized Thermal Shock (PTS)

- Background
- PTS Rule and Screening Criteria Limits
- Additional PTS Items
- Past changes to PTS Rule
- Ongoing PTS Activities

Background

- Prior to 1980 there were at least eight actual overcooling transients that were identified as PTS events in PWRs
- In 1981 NRC identified PTS to be an unresolved generic safety issue for PWR vessels
  - Concerned with challenge to integrity of an embrittled vessel from a severe overcooling event in combination with repressurization
- Two events were severe pressure and thermal transients:
  - An event at Rancho Seco was attributed to a relatively minor human error, vessel integrity was questioned if event had occurred in an embrittled vessel
  - Three Mile Island-1 event focused operator trade-offs between keeping the core cooled and protecting vessel integrity
- Significant NRC and industry research contributed to issuance of the final PTS Rule (10CFR 50.61) in 1985
- Plant-specific PTS analyses ultimately showed no near-term safety concerns for PWR vessels

PTS Screening Criteria

- NRC monitors RT_{PTS} and compares end-of-life value to screening criteria limits defined in PTS Rule (10CFR50.61)
- RT_{PTS} = 270°F (132°C) for plates, forgings and axial welds
- RT_{PTS} = 300°F (149°) for circumferential welds
- RT_{PTS} determined at inside surface and includes initial RT_{NDT} plus shift in RT_{NDT} due to irradiation plus a margin term for uncertainties in initial and irradiated values
- Vessels projected to exceed PTS screening criteria must perform additional analyses to demonstrate acceptable level of risk

Additional PTS Items

- Regulatory Guide 1.154 issued to provide guidance for probabilistic fracture mechanics analyses (1987)
- Generic Letter 92-01 and Generic Letter 92-01, Supplement 1 Requests for utility information
  - NRC questioned accuracy and completeness of vessel data for calculating RT_{PTS} values
  - Include consideration of “sister” plant data
  - Use all data for determining Cu and Ni content
  - Need for ratioing of surveillance data?
- NRC published NUREG-1511 containing plant-specific status of reactor pressure vessel issues and released RVID licensing database (1994)
- NRC issued revised PTS Rule (10CFR50.61) with clarifications for calculating RT_{PTS} values (1995)

Past Changes to PTS Rule

- PTS Rule (10CFR50.61) amended in 1995. The following changes were made:
  - Directly incorporated RG 1.99, Revision 2 method for determining RT_{NDT} including initial RT_{NDT} value, the margin term, and the explicit definition of "credible" surveillance data
  - Clarified the determination of RT_{PTS} and adds adjustment in CF for difference between surveillance material and vessel material
  - Requires that other plant surveillance data and test reactor data be used to verify that RT_{NDT} values calculated for each beltline material are bounding values for the specific reactor vessel
- These words reflected a changing regulatory philosophy toward a bounding approach for regulating PTS

Latest PTS Activities

- Extensive program underway to perform complete re-evaluation of PTS Screening Criteria
  - Joint activity between NRC and U.S. industry (through EPRI Materials Reliability Program)
  - Review all technical disciplines and improve analysis based on enhanced knowledge and operating experience
    - Probabilistic risk assessment (PRA)
    - Thermal hydraulics (TH)
    - Probabilistic Fracture Mechanics (PFM)
    - Risk acceptance criteria
- Regulatory revision to 10CFR50.61a (PTS Rule) as an alternative approach has been approved and should be published this year

Upper Shelf Toughness

- Background
- Evaluation Criteria

Upper Shelf Toughness Background

- Prior to 1973
  - 30 ft-lb (41J) energy required at 60°F (33°C) below the lowest service temperature (strong direction, unirradiated material)
- 1973 to May 1983
  - 75 ft-lb minimum (weak direction, unirradiated material)
  - CVN shift measured at 50 ft-lb (68J) energy level
- After May 1983
  - 75 ft-lb (102J) minimum (weak direction, unirradiated material)
  - 50 ft-lb (68J) minimum during service (weak direction, irradiated condition)
  - Charpy shift measured at 30 ft-lb (41J) energy level
- Current
  - Methodology to predict upper shelf toughness being reviewed as part of ongoing NRC effort to revise RG 1.99

Upper Shelf Toughness Evaluation Criteria

- For materials falling below 50 ft-lb, perform an equivalent margin fracture mechanics analysis:
  - ASME Code recommended Service Level A and B criteria in 1987
  - ASME Code recommended Service Level C and D criteria in 1989
  - ASME Code Case N-512 approved by Section XI in August 1992 for evaluation of low USE materials
- NRC RG 1.161 analysis methodology based on approach developed in ASME Section XI Appendix K
  - RG provides additional guidance on transient selection and material properties
- Not an issue in U.S. for 40-year operating term
  - Extended operation – should not be a problem, but must be checked

Normal Heatup and Cooldown Plant Operating Criteria

- ASME Section XI, Appendix G
- 10CFR50, Appendix G

Normal Heatup and Cooldown
ASME Section XI, Appendix G

- Prior to 1973
  - Assure structure was not structurally loaded unless metal temperature was (a specified amount) above the ductile/brittle transition temperature
  - Charpy energy > 30 ft-lb (41J) at 60°F (33°C) below the lowest service temperature
- 1973 – early 1990s
  - Quantitative fracture mechanics procedure used to assure defined margin against failure as a vessel is structurally loaded
  - Safety margin defined as

$$2K_{IP} + K_{IT} < K_{Ic}$$

K_{IP} = membrane stress intensity
K_{IT} = thermal stress intensity
K_{Ic} = reference toughness

Normal Heatup and Cooldown
ASME Section XI, Appendix G

- 1992
  - Code Case N-514 and Code Change approved permitting LTOP setpoints to limit the maximum pressure in the vessel to 110% of the pressure determined at the low temperature end of the P/T limit curve.
- 1996
  - Code Case N-588 and Code Change approved allowing use of a circumferential reference flaw in circumferentially-oriented vessel welds
  - Revised equations for K_{IP} and K_{IT} to be consistent with modern finite element solutions when calculating P-T limit curves. This change also involved adding the stress intensity factor solutions to Appendix G for circumferentially-oriented reference flaws.
- 1998
  - Code Case N-640 and Code Change approved allowing the use of the K_{IC} (static) reference fracture toughness in lieu of the K_{IR} (dynamic) reference fracture toughness for P/T limits and BWR hydrostatic test temperatures.
    - LTOP setpoint *reduced* to 100% of Appendix G pressure

Normal Heatup and Cooldown ASME Section XI, Appendix G

- 2000
  - Code Case N-641 approved allowing LTOP system temperature and pressure setpoints to be determined using vessel specific geometries and material properties, including aspects of circumferential reference flaws in vessels with circumferential welds.
- Present
  - Efforts underway to revisit Code Case N-640 requirement to limit LTOP setpoint to 100% of Appendix G pressure
    - "Compromise" in order to gain NRC acceptance
    - Technical basis prepared to demonstrate adequate safety margin against K_{Ic} reference toughness curve while maintaining LTOP setpoint of 110% of Appendix G pressure

Normal Heatup and Cooldown
ASME Section XI, Appendix G

- Present (continued)
  - Efforts underway to eliminate flange requirement in Appendix G
    - 10CFR50 Appendix G requires that the RPV can not be pressurized beyond 20% of the preservice hydrotest pressure until the temperature is at least RT_{NDT} + 120°F (66.7°C)
      - Analysis based on use of K_{Ia} toughness curve
      - Results in “notch” in P-T limit curve – can become significant
    - Code Case N-640 is now in the Section XI Code and allows use of K_{Ic} curve; flange requirement should be based on analysis using K_{Ic} curve
      - Results in flange requirement of RT_{NDT} + 45°F (25°C) and a minimum pressure of 33% of the hydrotest pressure

Normal Heatup and Cooldown 10CFR50, Appendix G

- Fracture toughness requirements to prevent brittle fracture in vessels specify:
  - Limits on maximum pressure as defined in ASME Section XI, Appendix G
    - Assume 1/4-thickness reference flaw with semi-elliptical (6:1) shape
    - Use lower bound (K_{Ic}) toughness
    - Use safety factor of 2 on pressure
    - Axial flaw orientation in axial welds and plates
    - Circumferential flaw orientation in circumferential welds
  - Limits on minimum temperatures as defined in 10CFR50, Appendix G
    - Flange RT_{NDT} + 120°F, normal operation
    - Flange RT_{NDT} + 90°F, hydro and leak tests
- These requirements are used to define the full range of plant operating (P-T) limits

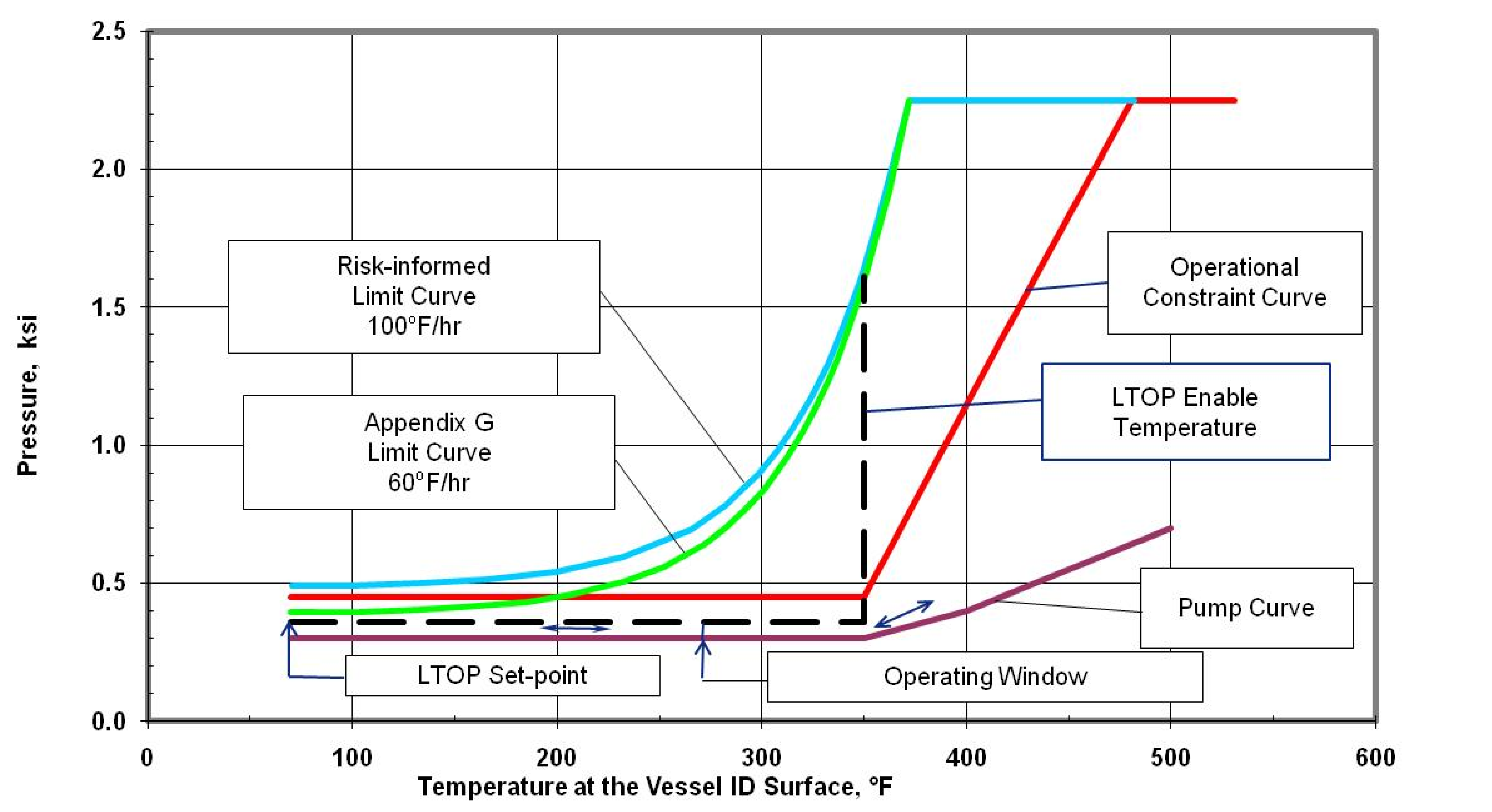
Low Temperature Overpressurization (LTOP)

- Background
- Integrity Requirements

Low Temperature Overpressurization Background

- By the late 1970s, 29 events occurred that produced pressure excursions above the P-T limits during PWR operation at low temperature
- Service experience indicates most events are isothermal and occur between 100 – 200°F (38 – 93°C)
- LTOP events occur from several different initiating sources
  - Safety injection related events
  - Charging and letdown events
  - Residual heat removal isolation
  - Reactor coolant pump start events
- Based on the event frequency NRC classified LTOP as anticipated operational occurrences and required protection systems during these events

Illustration of Narrowing Operating Window for PWR Normal Operation



Risk Informed Appendix G

- An alternative method for determining Appendix G heat-up and cool-down curves is being considered by the ASME Code Section XI
- Similar in use as current deterministic-based approach, but is based upon extensive probabilistic analyses
- $p = \{36.5 + 22.783 \text{ x } \exp[0.036(T - RT_{NDT} - 61)] - K_{It}\} \text{ x } \{t/R_i\} \text{ x } \{1/M_m\}$ – SI units
- $p = \{33.2 + 20.734 \text{ x } \exp[0.02(T - RT_{NDT} - 110)] - K_{It}\} \text{ x } \{t/R_i\} \text{ x } \{1/M_m\}$ – US customary units

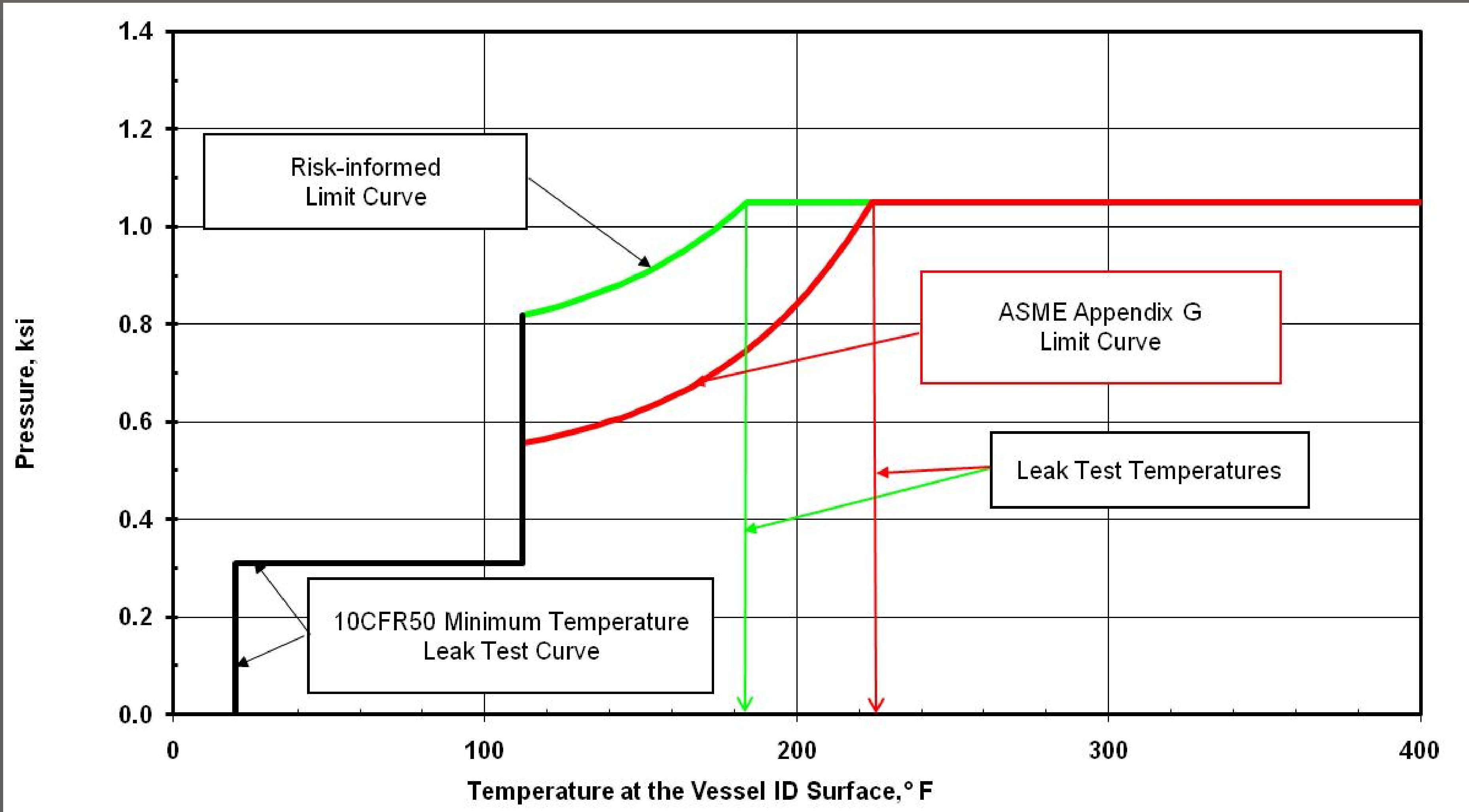
Low Temperature Overpressurization Integrity Requirements

- LTOP protection system requirements in NRC Standard Review Plan 5.2.2
  - Maximum pressure: App. G curve allowable pressure limit
  - Enable temperature: RT_{NDT} + 90 °F
- LTOP protection system requirements in ASME Code Case N-514
  - Maximum pressure: 110% of Appendix G curve allowable pressure limit
  - Enable temperature: RT_{NDT} + 50 °F
- Code Case recently approved allowing LTOP system temperature and pressure setpoints to be determined using vessel specific geometries and material properties, including aspects of circumferential reference flaws in vessels with circumferential welds

BWR Cold Hydro Test and Leak Test Temperatures

- Embrittlement in BWR vessels raises the temperature required to perform cold hydro and leak tests
- Test temperatures determined by margins defined in the ASME Code for prevention of brittle fracture
- Hydro tests performed above 200°F (93°C) pose severe operational problems and personnel safety risks
  - ECCS must be operational
  - Primary isolation required
  - Additional pump heating or auxiliary heating needed to achieve test temperature
  - Leak detection by conventional methods becomes more difficult and possibly dangerous
- In addition, hydro tests above 200°F (93°C) would become critical path leading to delays in plant outages

Illustration of Conventional & Risk-informed P/T Limits for BWR Leak Test



Risk Informed Appendix G Leak Test Temperature Requirements

- *Maximum of outside surface (OS) flaw or inside surface (IS) flaw* for reactor heat-up and cool-down at rates not to exceed 100°F/hr (56°C/hr) *:*
  - $T = RT_{NDT} + 33 + \ln[(K_{Im} + K_{It} - 36.5)/22.783]/0.036$ – SI (OS)
  - $T = RT_{NDT} + 33 + \ln[(K_{im} - 36.5)/22.783]/0.036$ – SI units (IS)

  or

  - $T = RT_{NDT} + 60 + \ln[(K_{Im} + K_{It} - 33.2)/20.734]/0.02$ – US units (OS)
  - $T = RT_{NDT} + 60 + \ln[(K_{im} - 33.2)/20.734]/0.02$ – US units (IS)
- Heat-up and cool-down rates not to exceed 40°F/hr (22°C/hr)
  - $p = \{36.5 + 22.783 \times \exp[0.036(T - RT_{NDT} - 33)] - K_{It}\} \times \{t/R_i\} \times \{1/M_m\}$
  - $p = \{33.2 + 20.734 \times \exp[0.02(T - RT_{NDT} - 60)] - K_{It}\} \times \{t/R_i\} \times \{1/M_m\}$

Other Aspects of RPV Surveillance

Monitoring Vessel Embrittlement Surveillance Programs

- Neutrons from the core impact the vessel wall and cause changes in the microstructure and mechanical properties (toughness and strength)
- Property changes can be significant enough to reduce structural integrity to unacceptable levels
- Surveillance monitoring programs have been developed to physically monitor changes over time
- Results from these programs have shown that these programs must be more than simple design verification -- trend curves coupled with surveillance data are required to adequately assess RPV embrittlement

Monitoring Vessel Embrittlement Surveillance Requirements

- 10 CFR Part 50, Appendix H (effective late 1973)
- ASTM E185 (1961, 1966, 1970, 1973, 1979, 1982, 1994, 1998, 2002)
- Specifics for surveillance program design have evolved over time and are very detailed; each vessel program is designed to a version of ASTM E185 (e.g., ASTM E185-73) depending upon the date of the ASME Code of record for the vessel
- Specifics for testing and reporting have been updated and the most current version of E185 should be used
  - Efforts in progress within ASTM to remove testing requirements to a new standard
- Regulatory Guide 1.99, Revision 2 is used for predicting embrittlement (shift and upper shelf energy changes)

Types of Surveillance Programs

- Plant-Specific
  - Original
  - Supplemental or augmented
  - Ex-vessel dosimetry
- Integrated
  - B&WOG (lost original plant-specific capsule holders, now use host sites)
  - BWROG SSP (now an integrated supplemental program & host sites)
  - Original from another plant (generally supplemented)
  - Test reactor supplemental (e.g. Maine Yankee)

Other Key ASTM Standards for RPV Embrittlement

- Test reactor - E 184-79 (now not published)
- Surveillance program design and testing – older E 185
  - E-185 has been split into two standards
    - Designing a surveillance program (retains designation E 185-02)
    - Testing surveillance capsule materials (new E 2215-02)
- Annealing - E 509-03 (2008)
- Supplemental surveillance tests - E 636-09
- dpa - E 693-01 (2007)
- Surveillance dosimetry - E 853-01 (2008)
- Embrittlement shift - E 900-02 (2007)
- Reconstitution of Charpy specimens - E 1253-07
- Melt wire temperature monitors – E 1214-06

RG 1.190 for Calculational and Dosimetry Methods for Determining Pressure Vessel Fluence

- This regulatory guide provides a description of acceptable methodologies for:
  - Performing neutron transport calculations for the assessment of reactor vessel fluence (including the use of ENDF/B-VI or later cross-sections)
  - Specifying neutron dosimetry sensor sets and evaluation procedures for use in performing surveillance capsule and reactor cavity measurements
  - Validation and qualification of both analytical and measurement techniques
- The guide also specifies reporting requirements for:
  - Analytical results from neutron transport calculations
  - Measurement results and associated uncertainties
  - Derived pressure vessel fluence and associated uncertainties

Plant Specific Measurement Data

- Internal surveillance capsules
  - Provide 3-4 data points over the first 1/3 to 1/2 of plant life
  - Data are limited to the geometric location of the capsules
  - For long irradiations, sensors tend to saturate; this increases the uncertainty in the fluence evaluations
- Reactor cavity dosimetry
  - Provides complete azimuthal and axial mapping of exposure over the beltline region of the pressure vessel
  - Can provide measured data at the location of critical materials
  - Can provide a direct measure of the effectiveness of flux reduction initiatives
  - Measurement intervals can be chosen to optimize the use of the neutron sensors (i.e., avoid the saturation problem)
  - Provides dosimetry data beyond the data of the last scheduled surveillance capsule withdrawal

Reactor Cavity Dosimetry Program Objectives

- Provide a measurement data base sufficient to:
  - Remove biases that may be present in analytical predictions of neutron exposure
  - Support the methodology for projection of exposure gradients through the thickness of the pressure vessel wall
- Establish uncertainties in the best estimate fluence projections for the pressure vessel wall
- Provide a long term continuous monitoring capability for the beltline region of the pressure vessel

Vessel Surveillance
Testing Results and Application

- Until two sets of credible surveillance capsule results are available, the utility must use Regulatory Guide 1.99, Revision 2 (or acceptable alternative) to predict embrittlement
- Ideal surveillance programs contain specimens of the RPV limiting material(s) with regard to predicted embrittlement because the limiting material may change over time
- Plate or forging materials should be oriented in the weak direction, i.e., transverse (T-L or C-L) at the 1/4-thickness location
- Chemistry and baseline properties for materials should be documented (initial RT_{NDT}, upper shelf energy, 30 ft-lb (41 J) temperature [T_{30}], Cu content, and Ni content).

Vessel Surveillance
RG 1.99, Revision 2

- Predictions are based upon the Cu and Ni contents and the fluence following the product of a chemistry factor (CF) and a fluence function (FF):
  - Transition temperature shift = CF x FF
  - CF values are based upon the levels of Cu and Ni; 2 tables list the appropriate CF values depending upon the product form, base metal or weld metal
  - $FF = f^{(0.28 - 0.10 \log f)}$
  - Margin terms (approximate standard deviations) are provided for base metal (17°F) and weld metal (28°F)

Vessel Surveillance
RG 1.99, Revision 2

- Final adjusted RT_{NDT} (adjusted reference temperature [ART]) is used in P/T limits and assessing PTS concerns:
  - ART = Initial RT_{NDT} + Shift + Margin
  - Shift is equated to the shift in RT_{NDT} due to neutron exposure as measured at T_{30} from CVN energy curves
  - Margin is initially 2 standard deviations with some added effect if the initial RT_{NDT} is not measured directly

Vessel Surveillance
RG 1.99, Rev. 2 Credibility Requirements

- Most limiting material with regard to radiation embrittlement should be available in surveillance capsule
- Scatter in the CVN energy data should be small enough to permit an unambiguous determination of the 30 ft-lb (41 J) temperature (T_{30})
- Scatter of measured shift vs fluence data about a best fit line should be within one standard deviation for the appropriate material. If this criterion is failed, can still use data for upper shelf energy determinations
- Irradiation temperature for the specimens should match the RPV wall at cladding/base metal interface within ± 25°F (±14°C)
- Surveillance results for correlation monitor material should fall within scatter typical for that material

Vessel Surveillance
If Credibility Requirement Are Met

- Simplified least squares procedure is used to derive a modified chemistry factor
  - Calculate the FF for each data set
  - Multiply individual FF by the measured shift value and sum individual products
  - Square each value of FF and sum individual squares
  - Divide sum of (FF x shift) by sum of squares of FF; this quantity is new CF which reflects a minimization of the sum of the squares of the errors
- RG position 2.1 indicates need to make adjustment to measured surveillance data reflecting the chemistry differences (variability) between surveillance weld and RPV weld.

Typical Application
No Chemistry Adjustment

- Application of RG 1.99, Revision 2, Position 2.1 without chemistry adjustment

Capsule	f (10^{19})	FF	ΔRT_{NDT}	FF x ΔRT_{NDT}	FF^2
A	0.75	0.92	185.0	170.1	0.85
B	1.93	1.18	235.0	277.2	1.39
C	2.88	1.28	245.0	313.8	1.64
			Sum:	761.1	3.88

$$CF = \sum(FF \times \Delta RT_{NDT}) \div \sum(FF^2) = 761.1/3.88 = 196.3$$

- Revised CF can be used in correlation equation to predict future material behavior

Vessel Surveillance
Temperature Effects

- Lower temperature irradiations produce more embrittlement than higher temperature irradiations for the same neutron exposures
- Regulatory Guide 1.99, Rev. 2 recognizes this effect and puts limits on applicability to not less than 525°F (274°C); a correction factor for lower temperature data can be justified by reference to actual measured data
- But, very little data exist; the actual temperature effect is material dependent
- Yankee Rowe operated at 500 – 510°F (260 – 266°C) and was forced to use an additional degree of damage (1°F [0.6°C]) for each degree of irradiation below 550°F (which is the nominal temperature of the NRC database used for RG 1.99, Revision 2)

Vessel Surveillance
Neutron Energy Effects

- Generally, we treat the neutron energy spectrum as that above 1 MeV for light water reactor environments since most of the damaging neutrons are of similar magnitude and energy levels
- However, it is important to note that energies less than or equal to 1 MeV can also be very damaging due to their high numbers and their high collision cross-sections (i.e., probabilities for displacing iron atoms)
- There have been efforts in the international community to use displacements per atom (dpa) to measure and monitor radiation damage
- There is some degree of difficulty in retroactive analysis to backfit existing old data to the dpa methodology

Vessel Surveillance
Neutron Flux (Fluence Rate) Effects

- Do short term, high flux irradiations produce the same level of damage as lower flux, longer term irradiations?
  - It depends on the fluence rate, the fluence, and the materials
- This question can even be extended into surveillance lead factors greater than unity
- Recent workshop on dose rate effects on RPV materials
  - Sponsored by EPRI/CRIEPI – November 2001
  - Attempted to reach international consensus on dose rate effects
- Regulatory Guide 1.99, Rev. 2 reflects only power reactor surveillance data typical of longer term, low fluxes
- Embrittlement correlation in ASTM E900 does not include fluence rate effects, but latest (EONY) does

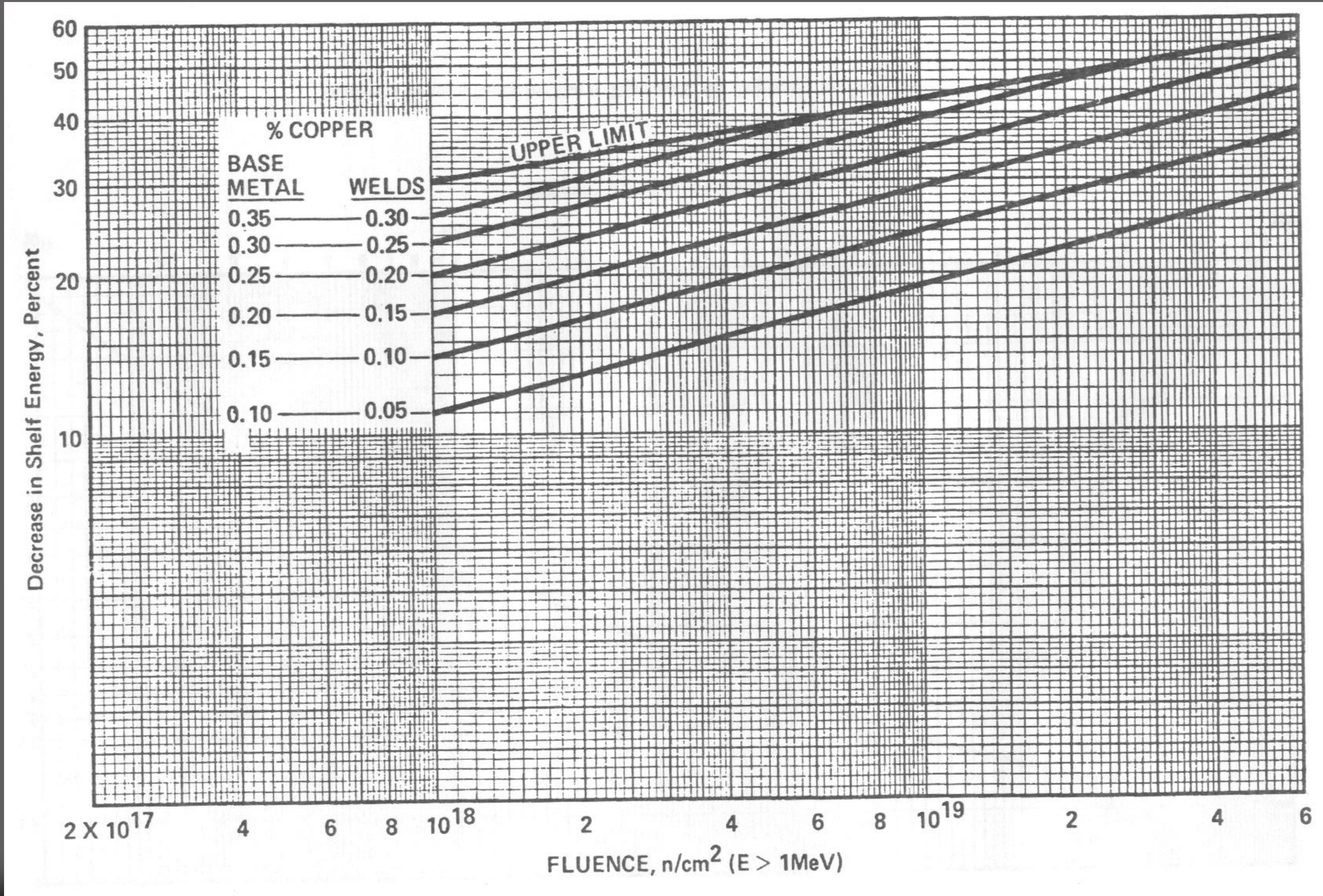
Vessel Surveillance
Through Wall Attenuation

- Important for calculation of damage at 1/4-thickness and 3/4-thickness (P/T curves, upper shelf evaluation, etc.)
- RG 1.99, Revision 2 uses a dpa-based attenuation formula which supposedly accounts for spectrum changes:
  - $f(x) = f_o \exp^{(-0.24\,x)}$
  - f_o is the inside wall fluence and x is the distance from the inside wall in inches
- Insufficient data available to support an alternate attenuation form
- U.S. utilities may have sufficient information to determine through-wall attenuation based on dpa calculations performed as part of surveillance capsule evaluation
- Comprehensive test reactor program with international consortium are evaluating through-wall attenuation (EPRI, CRIEPI, NRI, NRC)

Vessel Surveillance
Upper Shelf Energy

- 10 CFR Part 50, Appendix G sets the screening criteria for Charpy upper shelf energy:
  - 75 ft-lb (102J) initial shelf minimum
  - Never dropping below 50 ft-lb (68J) during service
- Decrease in upper shelf energy (USE) is conservatively assessed using Regulatory Guide 1.99, Revision 2
- If 50 ft-lb (68J) level is expected to be transgressed, it must be demonstrated that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section III of the ASME Code
- In some cases, 100% volumetric examination and/or supplemental fracture toughness testing may be required to assess equivalent margins
- Section XI of the ASME Code has developed acceptance criteria and analytical procedures for assessing adequate margins

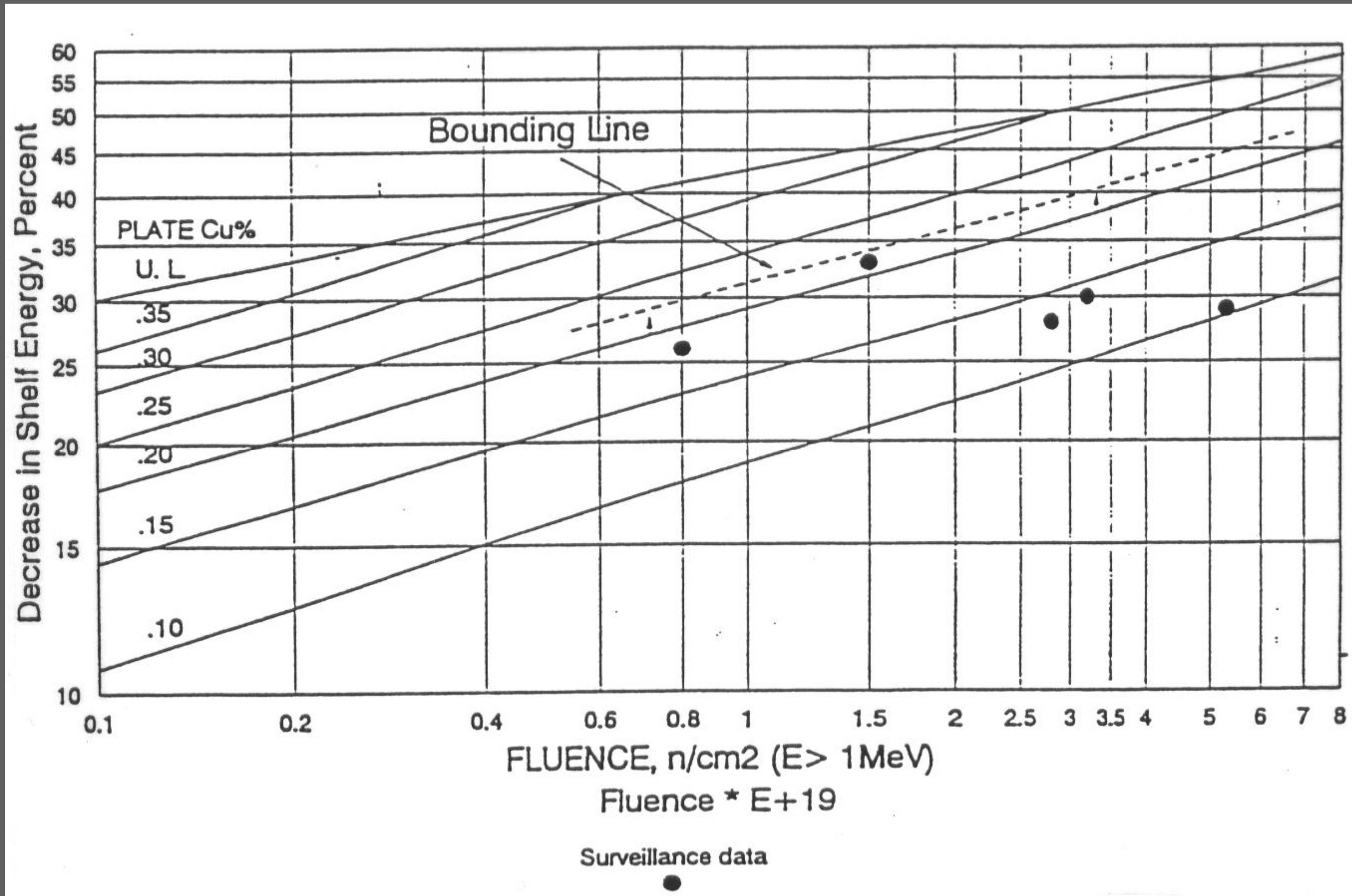
RG 1.99, Revision 2 Upper Shelf Energy Decrease Methodology



Vessel Surveillance
Upper Shelf Energy Drop

- Regulatory Guide 1.99, Revision 2 assumes that the USE decreases only as a function of fluence and copper content
- Actual surveillance data can be used to adjust the drop in USE:
  - Plot the plant-specific surveillance data on the Reg. Guide plot
  - Draw an upper bound to all of the data parallel to the existing lines
  - Use the upper bound to determine the decrease in USE

RG 1.99, Revision 2 Predicted Decrease in USE Plate Metal Showing Surveillance Data



Integrated and Supplemental Surveillance Programs

- Integrated programs involve irradiations at different facilities and tying the results to a specific reactor pressure vessel
- Supplemental programs involve additional capsules and testing outside of the original surveillance program for a specific vessel
- Generally an integrated program will require supplemental capsules and testing to validate the integrated program approach
- Several examples of integrated and supplemental surveillance programs: B&WOG, Calvert Cliffs-1/McGuire-1, McGuire-1/Diablo Canyon-2, BWROG SSP, Palisades, EPRI-CRIEPI

Obtaining Credit for Integrated/ Related Surveillance Results

- 10 CFR Part 50, Appendix H
  - Reactors with similar designs and operating features
  - Adequate dosimetry for each individual reactor
  - Adequate arrangement for data sharing between plants
  - Contingency plan to keep plants independent if operating issues change
  - Number of materials and specimen types/numbers must remain the same
  - Must be substantial advantages to be realized
- Requires approval on a case-by-case basis from NRC, Director of NRR

Supplemental Program Considerations

- Use of reconstituted irradiated specimens from previous capsules
- Use of surrogate or similar materials
- Application of ex-vessel dosimetry
- Aspects of ASTM E 185 that can be altered
- Fracture toughness testing of precracked Charpy specimens in addition to or in lieu of standard Charpy testing
- Annealing requires a supplemental surveillance program
  - Guidance provided in ASTM E 509