**REACTOR PRESSURE VESSEL (RPV) INTEGRITY WORKSHOP**

International Light Water Reactor Materials Reliability Conference and Exhibition 2016
August 1, 2016 • Hyatt Regency McCormick Place, Chicago, Illinois USA

*Room S101B*

<table>
<thead>
<tr>
<th>TIME</th>
<th>TOPIC</th>
<th>PRESENTER</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>8:00 AM</strong></td>
<td>1.0 Introduction</td>
<td><em>Tim Hardin (EPRI)</em></td>
</tr>
<tr>
<td></td>
<td>• Purpose of workshop</td>
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<tr>
<td><strong>8:15 – 8:45 AM</strong></td>
<td>2.0 Reactor Vessel Function and Degradation Mechanisms</td>
<td><em>Tim Hardin (EPRI)</em></td>
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<tr>
<td></td>
<td>• RPV Embrittlement</td>
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<td>• Other potential degradation mechanisms (e.g., thermal aging)</td>
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<tr>
<td><strong>8:45 – 9:15 AM</strong></td>
<td>3.0 RPV Fabrication</td>
<td><em>Tim Hardin (EPRI)</em></td>
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<tr>
<td><strong>9:15 – 9:45 AM</strong></td>
<td>4.0 Elements of Fracture Mechanics</td>
<td><em>Nathan Palm (EPRI)</em></td>
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<td>• Linear Elastic Fracture Mechanics</td>
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<td>• Elastic-Plastic Fracture Mechanics</td>
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<td><strong>9:45 AM</strong></td>
<td>Break</td>
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<tr>
<td><strong>10:00 - 10:45 AM</strong></td>
<td>5.0 Basic Concepts of Embrittlement Monitoring and Management</td>
<td><em>William Server (ATI Consulting)</em></td>
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<tr>
<td></td>
<td>• Fracture toughness and radiation effects, Reference Temperature,</td>
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<tr>
<td></td>
<td>Reference Toughness Curves, Embrittlement trend correlations,</td>
<td></td>
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<td>Adjusted Reference Temperature</td>
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</tr>
<tr>
<td><strong>10:45 – 11:30 AM</strong></td>
<td>6.0 Criteria and Requirements for Vessel Fracture Toughness and Fracture Prevention</td>
<td><em>Tim Hardin (EPRI)</em></td>
</tr>
<tr>
<td></td>
<td>• Current regulatory requirements</td>
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<tr>
<td><strong>11:30 – 12:00 PM</strong></td>
<td>7.0 Vessel Monitoring through In-service Inspection</td>
<td><em>Jack Spanner (EPRI)</em></td>
</tr>
<tr>
<td></td>
<td>• Requirements, methods, frequency</td>
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<tr>
<td><strong>12:00 PM</strong></td>
<td>Lunch</td>
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</tr>
<tr>
<td><strong>1:00 – 1:45 PM</strong></td>
<td>8.0 Vessel Monitoring through Surveillance Programs</td>
<td><em>William Server (ATI Consulting)</em></td>
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<tr>
<td></td>
<td>• 10 CFR 50 Appendix H</td>
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<td>• Integrated surveillance programs</td>
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<td>• Supplemental surveillance programs</td>
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<tr>
<td><strong>1:45 – 2:15 PM</strong></td>
<td>9.0 Upper Shelf Energy</td>
<td><em>Tim Hardin (EPRI)</em></td>
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Together we’re Shaping the Future of Electricity
## AUGUST 1, 2016

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<td>Tim Hardin (EPRI)</td>
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<tr>
<td>3:15 – 4:00 PM</td>
<td>11.1 Pressurized Thermal Shock: PTS Rule</td>
<td>Nathan Palm (EPRI)</td>
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<td></td>
<td>11.2 Alternative PTS Rule</td>
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<tr>
<td>4:00 – 4:45 PM</td>
<td>12.0 Current Challenges and Future Directions</td>
<td>Tim Hardin (EPRI)</td>
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<tr>
<td></td>
<td>• Vessel fabrication issues (hydrogen flaking and high carbon); BTP 5-3 Issues; Appendix G Small Flaw Issues; Direct Use of Fracture Toughness and Small Specimen Testing (mini-CT); Vessel Annealing</td>
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</tr>
<tr>
<td>4:45 PM</td>
<td>Open Discussion/ Q&amp;A</td>
<td>All</td>
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<tr>
<td>5:00 PM</td>
<td>Adjourn</td>
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</table>
1 - Introduction and Agenda

Tim Hardin
Technical Executive

RPV Integrity Workshop,
2016 International LWR Materials Reliability Conference, Chicago, USA
August 1, 2016
Content

- Purpose / Objective
- Ground Rules
- Agenda
Purpose / Objective

- Present broad review of
  - Fundamental concepts related to RPV Integrity
  - Current regulations and the rationale behind them
  - Existing and developing technologies / techniques for evaluating and assuring RPV integrity

- Focus is on the traditional vessel and the degradation mechanisms that impact operating limits (e.g., embrittlement)

  - Revision 1 is being prepared for publication by end of 2016
Ground Rules

- Informal presentation format
  - Interactive participation, questions strongly encouraged

- Workshop is designed for attendees with all levels of experience in reactor vessel embrittlement and integrity issues

- Detailed presentation slides are designed to be more stand alone than typical seminar slides
  - Slides are designed to be a useful reference after attendance at this workshop is forgotten
  - Slides will allow attendees to focus on content, not note-taking
Ground Rules

- Presenters will clearly identify in the presentations:
  - Mandatory requirements in the regulations – as “required” or as directing actions that “must” be taken
  - Technical information - and suggestions regarding good practice that “should” be taken - which may facilitate compliance with regulations
  - Opinions will be presented based on experience of the presenters

- These presentations and the Primer (MRP-278) are technical resources only and not part of a Guideline mandated by NEI 03-08, “Guideline for the Management of Materials Issues.” They do not contain any mandatory or required elements that need to be addressed by utilities to comply with NEI 03-08
Training Development Credit

- Curricula and material were developed consistent with ACAD02-004 "Guidelines for the Conduct of Training and Qualification Activities" and ACAD 85-006 "Principles of Training Systems Development"
  - To encourage utility training programs to award formal credit for attending this training
  - Certificates will be emailed
- Attendees’ training programs decide whether or not credit for attending will be given
<table>
<thead>
<tr>
<th>Time</th>
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</thead>
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<tr>
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## Agenda – Afternoon Session

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<tr>
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<th>Speaker</th>
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<td>Nathan Palm (EPRI)</td>
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<td>12.0 Future Directions and Challenges</td>
<td>Tim Hardin (EPRI)</td>
</tr>
<tr>
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<td>All</td>
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Together...Shaping the Future of Electricity
2 - Reactor Vessel Function & Degradation Mechanisms

Tim Hardin
Technical Executive

RPV Integrity Workshop,
2016 International LWR Materials Reliability Conference,
Chicago, USA
August 1, 2016
Learning Objectives

- Importance of RPV integrity
- General configuration and functions of PWR and BWR vessels, including significant differences
- Materials Degradation Matrix (MDM)
- Potential degradation mechanisms for RPV Low Alloy Steel
  - Thermal embrittlement
  - Environmental (hydrogen effects)
  - BWR Stress Corrosion Cracking (SCC)
  - Neutron Embrittlement
Importance of RPV Integrity

- The reactor vessel is one of the four barriers of protection from irradiation release to the general public
  - Ceramic fuel pellets, fuel rods, RPV, containment
  - Protecting the integrity of the RPV is of the utmost importance to safety

- Failure of the vessel is beyond the design basis
  - Containment was not designed for RPV failure; therefore, early focus of ASME Section XI (1973) was the RPV

- Neutrons from the reactor core irradiate the vessel during operation and cause localized embrittlement of the vessel steel

- Reactor vessel embrittlement is the dominant aging concern that may limit the useful life of a nuclear plant
  - Note that PWRs are more susceptible than BWRs because PWR vessels receive significantly more neutron irradiation (~10X)
Configuration and Function of PWR RPV

- **Function:** Support and contain the core; direct coolant flow through fuel bundles
- **Essential characteristics:**
  - Low alloy steel (LAS) vessel with stainless steel clad
  - No penetrations in beltline area
  - Removable top head, nonremovable bottom head
  - Control rod penetrations in top head
Configuration of BWR RPV

- Function: Support and contain the core; direct coolant flow through fuel bundles
  - Also, serve as a steam generator; house steam dryer and steam separator assembly

- Essential characteristics:
  - LAS vessel, clad with SS (except removable top head)
  - Removable top head, nonremovable bottom head
  - Differences from PWR:
    - Many nozzles penetrations near or in beltline
    - Control rod penetrations in bottom head
RPV Materials and Degradation Mechanisms


- Provides a comprehensive review of degradation mechanisms applicable to light water reactor (LWR) plant nuclear steam supply system (NSSS) components, and assesses the extent to which these degradation mechanisms are understood.

- Summarizes the state of industry knowledge regarding degradation mechanisms and related research and development activities.
  - MDM results are used as direct inputs into the BWR and PWR Issue Management Tables (IMTs) which are used to guide EPRI research.

- Available to public (no fee)
### MDM Summary for PWR RPV Materials

#### Table 3-1: PWR Primary Pressure Boundary (1)

<table>
<thead>
<tr>
<th>Material</th>
<th>WtGt</th>
<th>Pitting</th>
<th>FAC</th>
<th>Foul</th>
<th>Wear</th>
<th>IG/TG</th>
<th>IA</th>
<th>HC</th>
<th>EAF</th>
<th>Th</th>
<th>Env</th>
<th>Emb</th>
<th>VS</th>
<th>IC / SR</th>
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</thead>
<tbody>
<tr>
<td>C&amp;LAS: Base Metal &amp; HAZ</td>
<td>Y</td>
<td>N</td>
<td>N</td>
<td>N</td>
<td>N</td>
<td>N</td>
<td>?</td>
<td>N</td>
<td>?</td>
<td>N</td>
<td>?</td>
<td>Y</td>
<td>Y</td>
<td>N</td>
</tr>
<tr>
<td>C&amp;LAS: Welds</td>
<td>Y</td>
<td>N</td>
<td>N</td>
<td>N</td>
<td>N</td>
<td>N</td>
<td>?</td>
<td>N</td>
<td>?</td>
<td>N</td>
<td>?</td>
<td>Y</td>
<td>Y</td>
<td>N</td>
</tr>
<tr>
<td>SS: 300 Series SS Base Metal &amp; HAZ</td>
<td>N</td>
<td>Y</td>
<td>p1-2c</td>
<td>N</td>
<td>N</td>
<td>N</td>
<td>Y</td>
<td>p1-6c</td>
<td>N</td>
<td>Y</td>
<td>p1-10c</td>
<td>Y</td>
<td>N</td>
<td>N</td>
</tr>
<tr>
<td>SS: 300 Series SS Welds &amp; Clad</td>
<td>N</td>
<td>Y</td>
<td>p1-2d</td>
<td>N</td>
<td>N</td>
<td>N</td>
<td>Y</td>
<td>p1-6d</td>
<td>N</td>
<td>Y</td>
<td>p1-10d</td>
<td>Y</td>
<td>N</td>
<td>N</td>
</tr>
</tbody>
</table>

**Colors and Descriptions**
- **Blue**: Lack of data to establish degradation applicability
- **Green**: Well characterized, little or no additional research is needed
- **Yellow**: Ongoing R&D efforts to resolve uncertainties in near-term time frame
- **Orange**: Insufficient R&D to resolve uncertainties in a near-term time frame
# MDM Summary for BWR RPV Materials

## Table 4-1: BWR Primary Pressure Boundary

<table>
<thead>
<tr>
<th>MATERIAL</th>
<th>Corrosion</th>
<th>Wear</th>
<th>SCC</th>
<th>Fatigue</th>
<th>Reduction in Fract Properties</th>
<th>Irradiation Effects</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Wstg</td>
<td>Pitting</td>
<td>FAC</td>
<td>Foul</td>
<td>Wear</td>
<td>IG/TG</td>
</tr>
<tr>
<td>C&amp;LAS: Base Metal &amp; HAZ</td>
<td>N</td>
<td>N</td>
<td>Y</td>
<td>b1-3a</td>
<td>N</td>
<td></td>
</tr>
<tr>
<td>C&amp;LAS: Welds</td>
<td>N</td>
<td>N</td>
<td>Y</td>
<td>b1-3b</td>
<td>N</td>
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</tr>
</tbody>
</table>

**Legend**

- **Blue**: lack of data to establish degradation applicability
- **Green**: well characterized, little or no additional research is needed
- **Yellow**: ongoing R&D efforts to resolve uncertainties in near-term time frame
- **Orange**: insufficient R&D to resolve uncertainties in a near-term time frame
U.S. NRC Expanded MDA

  - Environmental Effects on Fracture Resistance
  - Thermal Embrittlement of RPV Steels
  - Long-term Integrity of Dissimilar Welds
  - Environmental Assisted Fatigue
  - Neutron Embrittlement
Thermal Embrittlement
Background

- LAS (plate, forging, welds, and HAZ) is susceptible to thermal aging
- Thermal aging is caused by slow diffusion and segregation of impurity elements (e.g., primarily phosphorous) into grain boundaries at operating temperature
  - Results in increase in DBTT but not yield strength/hardness; also known as a “non-hardening embrittlement”
  - Degree of embrittlement is largely a function of P and temperature

Overview & summary of the literature through ~2003:
LAS Components Susceptible to Thermal Ageing

- Ferritic primary pressure boundary LAS operating at higher temperatures is susceptible to thermal aging embrittlement
  - RPV LAS components, e.g., RPV flange, nozzle shell ring, and outlet nozzles, experience temperatures of up to 315°C (~600°F)
  - Vessel heads of some reactors may be near the hot leg temperature of about 315°C (~600°F)
  - Pressurizers experience highest operating temperature at around 343 °C (650°F)
- HAZ of pressurizer welds viewed as most susceptible
Regulatory Approaches

- Nuclear safety codes in France (e.g., French Association for Nuclear Codes and Standards [AFCEN]) and Russia require consideration of thermal embrittlement effects when ductile-to-brittle transition temperature (DBTT) shift is calculated.

- U.S. regulations do not require consideration of thermal aging.
  - However, per 10CFR50 Appendix H, surveillance capsule specimens reflect not only effects of neutron environment but also the thermal environment to which RPV walls are exposed.
  
  - To the extent that surveillance test results are used for integrity assessment, effects of thermal aging on RPV beltline shell (exposed to $T_{COLD}$) are considered.
Recent Research / References

  - Accelerated aging at 400°C up to 30,000 hours and 450°C up to 20,000 hours
  - Base metal, weld & HAZ exhibited significant embrittlement

  - Base metal weld & HAZ aged in collars attached to PWR cold leg & aged @ 290°C for 30 years
  - No aging effects were found; thermal embrittlement is negligible at cold leg temperatures
Recent Studies on Effect of Thermal Aging on RPV LAS

- P. Todeschini et.al. [EDF], “Effects of thermal ageing on toughness properties of vessel steel,” Fontevraud 8, France, September 2014.
  - Aged representative RPV materials at 300°C and 350°C for up to 80,000 hours (equivalent to 43 years at 325°C)
  - Observed embrittlement was lower than that predicted by French code
  - A report from the Materials Aging Institute (MAI, EDF Les Renardières, France) is expected in late 2016
Environmental Effects on Fracture Resistance (Hydrogen)
Background: Hydrogen Effects in Low Alloy Steel

- Hydrogen in ferritic steels can cause loss of ductility and time-dependent failure at stresses considerably less than yield strength (e.g., delayed failure)

- Effects of hydrogen vary as a function of hydrogen concentration and strength level: high-strength steels are significantly more susceptible than low-strength steels
  - Catastrophic failure can occur in high-strength steels at 1 ppm H or less, whereas such failure may require > 4 ppm H in lower strength steels

- In early 1960s, Atomic Energy Commission (AEC, predecessor to U.S. NRC) was aware of the potential effects of hydrogen and commissioned several studies
Early Studies: Harries & Broomfield 1963

  - The only significant source of hydrogen is that due to the corrosion reaction at the steel-water interface
    - Negligible contribution from hydrogen in water or from radiolysis
  - Assuming a ‘pessimistic” corrosion rate (75 mg/dm² x month) on an unclad vessel and that all the corrosion induced hydrogen enters the steel, Harries and Broomfield concluded the information on effects of hydrogen in low alloy PWR RPV steels available at that time suggest immunity to hydrogen embrittlement
    - However, carbon steel vessels could experience hydrogen embrittlement
  - Assessment did not consider effects of high stress areas or effects of irradiation
Early Studies: ANL-7266, 1967

  - Studied hydrogen embrittlement in irradiated steel: Type 4340, A212-B
  - “Catastrophic embrittlement due to hydrogen (like delayed failures at 25 to 50% of the notch-tensile strength in Type 4340 steel) was not observed in 212-B, even for irradiated material that had been charged to produce high hydrogen contents. Therefore, catastrophic hydrogen embrittlement of a well-designed nuclear-reactor pressure vessel is not believed credible.”
  - “As long as [hydrogen] concentrations are in the 1 to 2 ppm range, the necessary conditions for catastrophic delayed failure do not exist in 212-B reactor pressure vessels.”
Early Studies: Brinkman and Beeston, 1970


- Influence of hydrogen on ductility, fracture strength, and tendency towards delayed failure was investigated for several irradiated pressure vessel steels: ASTM A302B, A542, and HY-80 steel irradiated at fluences from $8 \times 10^{18}$ to $4 \times 10^{20}$ n/cm$^2$, $E > 1$ MeV.

- “Reductions in ductility and true fracture strength occurred with increasing hydrogen content but were not extensive at strength levels less than 180 ksi in specimens containing 1 to 2 ppm hydrogen.”

- “The data reported herein therefore support the conclusion that medium strength pressure vessel steels in the normal quenched and tempered condition used for pressurized water and boiling water reactor plants operating under normal conditions will not be embrittled by the presence of 1 to 2 ppm hydrogen to the point where catastrophic failure will occur.”
Susceptibility of LAS to Stress Corrosion Cracking
BWR SCC in RPV Low Alloy Steels

- Extremely difficult to initiate or maintain SCC crack growth in LAS under steady state loading in normal BWR environment
  - In the absence of high loads (K in excess of 50-60 MPa√m), abnormal water chemistry including transients or load cycling, it is difficult to produce and maintain SCC extension

- Field experience supports the inherent SCC resistance of LAS in the BWR environment
  - Tsuruga shroud support weld cracking, but no SCC growth in LAS
  - Hamaoka CRD stub tube to vessel weld cracking, but no SCC crack extension into low alloy steel bottom head
  - Cracking had occurred in BWR feedwater nozzles (due to thermal fatigue) but there was no apparent SCC crack extension

- Potential SCC might result from cracking in Alloy 182 attachments to RPV and/or from water chemistry transients, particularly associated with chloride concentrations above 3 ppb
BWR SCC in RPV Low Alloy Steels

- To address SCC concerns the BWRVIP has developed crack growth disposition curves
- Additional work ongoing to address chloride transients and memory effect (the time it takes to restore the crack tip to bulk water chemistry conditions)
Neutron Irradiation Embrittlement
Behavior of Carbon Steels

- RPV is fabricated of ferritic low alloy steel, which exhibits good combination of strength and ductility.

- Carbon and low alloy steels behave in ductile manner at high temperatures but brittle at low temperatures – exhibit a ductile-to-brittle transition over temperature.

- Ductile: Crack propagation consumes large amounts of energy and involves plastic deformation.

- Brittle: Crack propagation is rapid with low energy absorption and little plastic deformation; fracture is sudden and catastrophic.

- Both failure modes are to be avoided, but brittle fracture warrants special attention.
The Safety Concern of Embrittlement

- Neutron irradiation from the core causes damage to the vessel steel by reducing material toughness
  - Raises ductile-brittle transition temperature
  - Reduces upper shelf energy (USE)

- A small initial flaw may exist somewhere in the vessel welds or heat-affected-zone just at a critical location

- High stresses from pressure and thermal loads could cause crack initiation of the small pre-existing flaw

- A rapidly growing crack in an embrittled vessel could cause a sudden and catastrophic rupture, either by brittle fracture (if in transition region) or ductile tear (if on upper shelf)

- Safety margins must be assured to prevent fracture of the vessel as it gradually embrittles over time with increasing neutron exposure
Embrittlement Concerns Can Lead to Plant Shutdown

- In 1988, Yankee Rowe was selected as the industry lead pressurized water reactor (PWR) plant in an EPRI-Department of Energy program to help define license renewal needs.

- Yankee Rowe RPV embrittlement became a seemingly insurmountable concern with the NRC.

- In 1992, Yankee Rowe was shutdown due to unfavorable economics posed in part by RPV embrittlement issue.
  - Showed industry that it is necessary to manage embrittlement to prevent plant shutdown.

- Fortunately, the circumstances at Yankee Rowe were relatively unique; vessel embrittlement issues since then have proven manageable.
Mechanistic Understanding of Embrittlement

- Neutron irradiation hardens ferritic steel causing an increase in strength and a corresponding decrease in fracture toughness, as a result of several factors
  - Microstructure of bcc (ferritic) RPV steels experiences complex changes when exposed to neutron irradiation, which in turn leads to degradation in mechanical properties as the cumulative exposure increases
  - High energy neutrons interact with atomic nuclei in the steel’s crystal lattice structure and create high-energy primary recoil atoms (PRAs); various features are induced by this process (e.g., solute clusters / copper-enriched clusters)
  - These features act as obstacles to movement of dislocations in the crystal matrix (movement of dislocations is necessary for ductile behavior)
  - Irradiation conditions (flux, fluence, energy spectrum, and irradiation temperature) strongly influence the process
- Complex!
Ongoing Research

- Numerous organizations worldwide are engaged in research to understand the various mechanisms of embrittlement
  - International Group on Radiation Damage Mechanisms (IGRDM) is a forum where world-wide discussion and interaction occurs
    - CRIEPI, UCSB, ORNL, NNL, University of Manchester, SCK-CEN, Rolls-Royce, University of Tokyo, U. of Rouen, EDF, VTT, Kiev Institute, EPRI & many others!

- Overview of current state of knowledge (including the many unknowns and gaps in current understanding) was provided by CRIEPI in a presentation available at
Around 1972, it was conclusively found that Cu and P increase irradiation damage; Ni also reduces toughness, especially for steels with moderate to high copper contents.

Higher impurity content makes weld metal more susceptible to the effects of irradiation than plates and forgings.

These findings led to tighter controls on Cu (< 0.10 wt%) and P (<0.012 wt%) in the plates and welds used to fabricate RPVs and to development of improved prediction methods for radiation damage as a function of Cu and Ni and neutron fluence; vessels ordered and fabricated before the early 1970s may exhibit greater sensitivity to neutron embrittlement.

Knowledge gained starting in the late 1990s has identified the importance of other alloying elements (Mn and Si).

Currently, uncertainty exists regarding high fluence effects – e.g., so-called “late blooming phase” of NiMnSi clusters that may introduce markedly increased embrittlement at high fluence >~7&E+19 n/cm².
Summary

- RPV steels are susceptible to several potential degradation mechanisms
  - Neutron embrittlement, thermal aging, environmental effects (hydrogen)

- Neutron embrittlement has the most immediate and significant impact that requires operational limits to assure RPV integrity
  - Risk of brittle fracture at temperatures closer to operating temperatures due to shift in DBTT to higher temperatures
  - Risk of low energy ductile tear due to reduction in USE

- The operational limits used to effectively mitigate these risks are discussed in the following modules of this workshop
Together…Shaping the Future of Electricity
3 - RPV Fabrication

Tim Hardin
Technical Executive

RPV Integrity Workshop,
2016 International LWR Materials
Reliability Conference
Chicago, USA
August 1, 2016
Reactor Pressure Vessel Fabrication

Learning Objectives:
- Two Basic Methods of RPV Fabrication
- Materials Selection and Evolution
- Hydrogen Flaking Issues
- Macrostructural Segregation in Steel Ingots
- Fabrication of Plates, Forged Rings, & Forged Nozzles
- Similarities and Differences of PWR and BWR vessels
- Basic RPV Fabrication Process
- Cladding Techniques and Practices
- Vessel Welding Practices
- RPV Fabrication Records
Methods of Vessel Fabrication

- Two basic methods of vessel fabrication have been used in the construction of RPVs
  - Rolled and welded plates to form separate shell courses
  - Large ring forgings
    - Improved component reliability and reduced inspection costs because of the absence of longitudinal beltline welds
    - However, much more expensive to fabricate
Materials

- Materials selection plays key role in fabrication process
- Factors influencing materials selection:
  - Mechanical and physical properties, fabricability, weldability, cost
  - Production capacity for product form (e.g., size limits)
- Essentially, steel is iron and carbon
  - Alloying elements are added to improve properties: manganese, nickel, chromium, molybdenum, silicon, aluminum
  - Always present but unwanted: hydrogen, nitrogen, sulfur, phosphorus
  - Carbon is the element that has greatest influence on mechanical properties
    - Carbon increases strength but reduces notch toughness
      - Notch toughness: the ability to resist brittle crack initiation & propagation
      - Notch toughness is measured by energy required to break a Charpy V-notch specimen
Chemical Composition from Alloying

- **Alloying composition** is a key issue for radiation damage
  - **C**: Cementite improves hardness and strength of steel, but decreases toughness properties. Should maintain to the minimum level needed to achieve the desired strength.
  - **Si**: Silicon is used as deoxidizers. Improves the transition temperature.
  - **Mn**: Manganese is also an deoxidizer. Improves the strength and hardenability of low carbon steel.
  - **Ni**: Is beneficial to steel with less than about 0.40% C. Improves low temperature toughness and weldability.
  - **Cr**: Increases resistance to oxidation. Increases the response to heat treatment, thus improving hardenability and strength.
  - **Mo**: Carbide former creating complex carbides \((\text{FeMo})_6\text{C}, \text{Fe}_{21}\text{Mo}_2\text{C}_6\). Improves yield strength, tensile strength and lowers DBTT of the steels.
  - **V**: Vanadium carbides make the material relatively resistant to thermal ageing. Creates fine grained tempered bainite and strength improves, but weldability is not good (pre-heat treatment).
RPV Steel Impurities

- **Impurities** from steel making create greater radiation damage
  - **Cu:**
    - Base metal in older plants came from the automotive scrap (wiring) used in the production of ingots
    - Welding used to use copper-coated wire electrodes to minimize the oxidation during storage and improves electrical conductivity
  - **P, S:** Impurity from the scrap in both base metal and weld wire
  - **Al, Sn, N, As, Co** also can be present in small amounts
Vessel Materials Evolution

- Over time, steelmaking was refined to improve strength and toughness and reduce content of elements which produce undesirable effects
  - Ni increased toughness but was found to increase neutron embrittlement
- During 30+ years that U.S. RPVs were fabricated major material changes included:
  - Plates SA-302-B → SA-302-B Mod → SA-533 Grade B Class 1
  - Forgings SA-336-F1 → SA-508 Class 1/2/3
  - Weld wire heats: a fairly abrupt end to the practice of coating weld wire with Cu once the role of Cu in embrittlement was discovered
- Reasons:
  - Improve fabrication
  - Reduce welding defects
  - Reduce embrittlement
Basics of Fabrication

- Melting practices
  - Open hearth and electric furnace
  - Use of electric furnace process is preferred for nuclear materials
    - Electric furnace provides more control over the process
    - Charge materials are lower in phosphorus content
      - Phosphorus is another element that reduces notch toughness (e.g., raises transition temperature)
    - Electric furnaces can be tilted, so slag volumes can be reduced
      - Steel is cleaner (fewer sulfide and oxide inclusions)
- Fabrication of plates and forgings begins with pouring of a steel ingot
Hydrogen Flaking Issues

- Hydrogen has highly detrimental effect in steelmaking
- Water vapor is present during steelmaking and hydrogen is formed when it reacts with liquid metal
- Hydrogen is very soluble in liquid metal but much less soluble in solid steel; when steel cools H₂ precipitates at inclusions, microvoids
  - Pressure builds to fracture strength and causes localized cracking ("flakes," "shatter cracks")
- These flaws have caused catastrophic failures for some types of components, such as turbine rotors
  - There has been recent experience regarding hydrogen flaking in RPV forged rings (e.g., Doel 3 / Tihange 2) but the flakes tend to be laminar / quasi-laminar and so have less structural significance
- Hydrogen also has detrimental effect on weldability – can cause cold cracking in heat-affected zones
- Hydrogen degassing was introduced in the 1950s but process is not perfect
Illustration of Classic Segregation Macrostructures in Static Cast Hot Topped Steel Ingot

- A- and V-segregates form during solidification
- During blooming, the top and bottom of the ingot are cropped to remove top positive and bottom negative segregation
- If forging a ring, the center core is usually trepanned, which removes V-segregates, but A-segregates remain
- Ingot is not trepanned if a plate is to be rolled

- A-segregates are significantly enriched with Mn, Ni, Mo and exhibit much greater hardenability and propensity to precipitate carbides.

- MnS inclusions in A-segregates act as hydrogen traps and thus preferential location for flaking, if it occurs.
Vessel Fabrication – Plates

- Ingots are broken down into a slab and soaked at high temperature (>2150°F, 1177°C) which helps to homogenize the microstructure.
- A rolling operation is conducted when steel is in austenitic temperature range.
- In general the minimum reduction in thickness is 3 to 1, with 4 to 1 more typical.
- An attempt is made to cross-roll plates which makes properties more uniform.
  - Because process is imperfect, plates exhibit different strength properties in the different directions.
- After rolling, plate can be subjected to a forming process to attain the required shape for use in a shell course, or the torus section of a top or bottom head.
Vessel Fabrication – Forged Rings

- For shell ring forgings and flange ring forgings, there are 2 processes have been typically used
  - Mandrel forging – the forged component is elongated or enlarged in a series of discreet, incremental steps
  - Ring rolling – the forged component is worked in a continuous manner
- Advantage of forged rings is the absence of axial welds in the vessel; only circumferential welds are necessary to join the various rings
- Ring forgings for RPV shells require very large ingots which tend to exhibit greater nonhomogeneity than smaller ingots
  - The potential effects of nonhomogeneity on vessel integrity are receiving increased attention
Vessel Fabrication – Nozzles

- Nozzles typically forged by one of the following processes, depending on size:
  - Tight mandrel process (for larger nozzles)
    - In tight mandrel process, a cored billet is lengthened but not enlarged
    - Several nozzles can be produced from a single tube; individual nozzles are cut and contour machined
  - Machining of forged bar stock (for small nozzles)

- Because many nozzles can be fabricated from a single ingot (single heat of material), then different plants can (and sometimes do) have nozzles from the same heat of material
  - Most common for BWRs, because BWR vessels contain many nozzles, most of which are much smaller than PWR nozzles
  - Material properties can differ; although all nozzles from a billet must be austenitized at the same time, they may be individually quenched
U.S. Vessel Fabrication Background

- Fabrication of an RPV required many years to complete
- Similar fabrication techniques were used to construct both BWR and PWR vessels
- Most vessels in U.S. were fabricated either by Combustion Engineering (Chattanooga) or Babcock & Wilcox (Barberton and Mount Vernon)
- Slight majority of BWR vessels were fabricated by Chicago Bridge & Iron (CB&I); 1 by Hitachi
- 8 PWR vessels were fabricated in Europe by Rotterdam Shipyard and 2 by Creusot-Loire
- In some cases, vessels were constructed by more than one fabricator due to scheduling problems in the shops
- Because of difficulties in transporting the large BWR vessels overland in one piece, field fabrication was used in the construction and completion of some vessels
Similarities and Differences Between PWRs and BWRs

- **Similarities:**
  - Welded on hemispherical bottom head and removable hemispherical top head

- **Main Differences:**
  - Only penetrations in cylindrical portion of PWR are inlet/outlet nozzles, well above top of core
  - BWR has many nozzles in the cylindrical shells both above and below core level
    - Why? It serves as both a reactor vessel and a steam generator
  - PWR CRDM penetrations are in top head; BWR in bottom head
PWR and BWR Vessel Differences

- Main Differences:
  - Size
    - BWR diameter ~ 5.4 – 6.4 m; overall height ~22 m
    - PWR diameter ~3.3 – 4.9 m; overall height ~14 m
    - BWR much taller due to water-steam separator
  - However, *weights* of the vessels are surprisingly similar, 400-500 tons
    - Larger BWR vessel offset by thinner wall thickness
  - Operating conditions
    - PWR: 15.2 MPa @ 270-315°C
    - BWR: 6.8 MPa @ saturation temperature ~290°C
    - BWR has large water annulus which shields vessel wall; RPV fluence (~1.1 – 7.6 E+18 n/cm² @ 60 years) significantly lower than PWR

- All U.S. BWRs are fabricated from formed/welded plate – no forged ring vessels
Basic Vessel Fabrication Process (1/4)

1. Hot form the plates to the proper cylindrical radius for the cylindrical shell; heat plate to 1600°F (871°C) and bend in a hydraulic plate bending press
   - The bending process was a line bend using three point dies; considerable skill required to obtain the proper uniform radius

2. After hot forming, plate was reheated to 1600°F (871°C) (above the transition temperature) and quenched and tempered in a flowing cold water quench tank

3. Test specimens were cut from plate prolongation (tensile, Charpy V-notch, and drop weight specimens where required); final properties and NDT or RT\textsubscript{NDT} of that particular plate were determined
Basic Vessel Fabrication Process (2/4)

4. Plates were cut to length and longitudinal weld preps were machined with weld root approximately one-third of the plate thickness from the ID

5. 3 curved plate segments were fit-up to form a cylinder (i.e. one shell course); held together with U-shaped tie straps (or backing bars) spanning the weld seams on the ID

6. Welding began on the OD using the automated submerged arc (ASA) process with Code required preheat supplied by external and internal gas torches and temperature carefully monitored

7. Upon completion of the external welding, the internal tie straps were removed; weld root ground to remove any slag inclusions; and weld completed from the ID
Basic Vessel Fabrication Process (3/4)

8. Shell courses were then given a post-weld heat treatment (PWHT)
   - ASME III requires a post weld heat treatment at a temperature 1100 – 1250°F (593 - 677°C) for various hold times
     ▪ Temperature is below material transition temp so that the temper enhanced properties are minimally affected
     ▪ But temperature is sufficiently high to rapidly reduce weld residual stresses and to metallurgically improve the weld
   - Hold times established to ensure thermal uniformity through-thickness and enough time to attain desired metallurgical changes

9. Welds were examined by radiographic examination (RT) and magnetic particle examination (MT)
Basic Vessel Fabrication Process (4/4)

10. Weld repairs made by manual welding (MMA)
11. Shell courses were then clad (discussed later)
12. Other vessel subassemblies were made in a similar fashion to the cylindrical shell courses
   - Bottom hemispherical head constructed of formed plate, typically with central dome section and a segmented toroidal section formed from “orange peel” segments
13. Final assembly required locating the position of the axial welds and nozzles which were indexed azimuthally in each shell course
14. Final circumferential welds were made with ASA
Circumferential Weld by ASA
Cladding

- Interior surfaces of the low alloy steel portions of the PWR reactor vessel, closure head, and flange area of vessels were typically clad with corrosion resistant material (Type 309/308) to prevent general corrosion of the materials by borated reactor coolant
- BWR vessels were typically clad below the level of the steam-water interface
- Cladding may increase the tendency for a small, near-surface flaw to propagate into the reactor vessel wall
- Vessels have “single layer” or “multi-layer” cladding
  - Early vessels were multi-layer; later vessels usually have single layer
- Cladding information is not readily available and relatively difficult to retrieve
Welding During Vessel Fabrication

- Many beads of weld material were required for vessel weld seams; used a large volume of weld wire
  - Important when determining the properties of each individual weld in the vessel beltline for sensitivity to neutron irradiation
  - Weld chemistry (Cu, Ni) may vary through thickness and around circumference because of variations in the weld wire used in fabrication
  - Each weld in the vessel can be traced by the unique weld wire/flux lot combination used

- Closing girth seam weld was made much later than longitudinal welds and is usually different heat weld wire/flux
CE Welding Practices

- CE used different welding procedures at different periods of time for RPV fabrication, changing weld wire specifications and weld wire suppliers.

- The earliest commercial RPVs were produced with SA302 Grade B plate and high Mn-Mo weld wire.

- ~1965, it was decided to add nickel to the plate material (SA302 Grade B Modified) and nickel to the weldments.
  - The SA302 Grade B Ni Modified plate specification eventually became the SA533 Grade B, Class 1 plate specification.

- Beginning late 1966, CE ordered submerged arc wire to a Mil B-4 Modified (Mn-Mo-Ni) specification that included Ni content of approximately 0.9 to 1.1 wt%.

- Later, Ni content was dropped to ~ 0.6 to 0.8 wt% when the synergistic adverse embrittlement effects of copper and nickel were identified.

- Copper was used as a coating for weld wire to reduce corrosion during storage and to increase conductivity during the welding process.

- When copper was discovered to be detrimental to embrittlement of the welds, its use as a coating was discontinued after 1972 for welds in the RPV beltline.

- However, many of the RPVs in the U.S. had already been fabricated prior to that date.

- Determining the amount of copper in a particular weld is critical for evaluating the amount of radiation embrittlement in the RPV.
B&W Welding Practices

- Babcock & Wilcox (B&W) used similar ASA welding techniques to those of CE, and few changes were made in the welding procedures.
- B&W used a relatively few number of different weld wire/flux lot combinations.
- All B&W RPV welds were made with Linde 80 flux type.
- Many B&W vessels were made with same specific weld number because B&W used fewer number of different weld wire/flux lots.
- One unfortunate commonality is low Charpy upper shelf energy (USE) which is due to Linde 80 flux.
  - The Linde 80 flux was chosen because it resulted in very small and finely dispersed non-metallic inclusions, producing good radiographs and requiring fewer weld repairs.
  - Unfortunately, the small inclusion size requires less local strain to debond an inclusion, so Linde 80 welds have relatively low initial USE.
RPV Fabrication Records

- ASME Code requires retentions of Lifetime Quality Records for each RPV
  - Design Specification
  - Overpressure Protection Report
  - Certified Material Test reports (CMTRs)
  - Heat treatment records
  - Final hydrostatic test results
  - Final non-destructive examination (NDE) results

- Form N-1 Certificate Holder’s Data Report for Nuclear Vessels
Together…Shaping the Future of Electricity
4 – Elements of Fracture Mechanics

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Sr. Technical Leader

RPV Integrity Workshop,
2016 International LWR Materials Reliability Conference, Chicago, USA
August 1, 2016
Learning Objectives

- Obtain a basic understanding of the use of fracture mechanics as it is applied to reactor pressure vessel integrity
- Recognize the role of fracture mechanics in the industry standards and regulations regarding vessel integrity and embrittlement
- Be able to identify the three categories of fracture mechanics and associated test methods
- Be aware of differences between probabilistic vs. deterministic fracture mechanics methods
Introduction to Fracture Mechanics

- Fracture is a deformation process whereby regions of a material body separate and load-carrying capacity decreases significantly, approaching zero

- Fracture is defined when the applied loading of a cracked body (crack driving force) exceeds the material’s resistance to failure (fracture toughness)

- Fracture toughness is a material property for a given material condition

- Fracture mechanics is an engineering discipline which concerns the behavior of crack-like defects in structures or components and their effect on integrity
Introduction to Fracture Mechanics

- Fracture mechanics was conceived by Griffith during World War I
  - Early applications were limited to fracture in brittle, cracked bodies
  - Interest intensified when approximately 25% of the all-welded U.S. Liberty ships in WW II experienced brittle fracture, exposing the urgent need to understand failure in ferritic structural steels and weldments
Introduction to Fracture Mechanics

- First use of fracture mechanics in mid-1950’s for missile and rocket motor cases
  - Brittle fracture in the high strength materials
  - U.S. Secretary of Defense requested assistance from American Society for Testing and Materials (ASTM)

- ASTM formed special committee in 1959 to develop technology and test methods for brittle fracture in that class of materials

- Linear Elastic Fracture Mechanics (LEFM) technology developed rapidly

- In 1965 special committee changed to standing Committee E24 on Fracture Testing of Materials (now part of E08 committee on Fatigue and Fracture)
Introduction to Fracture Mechanics

- FM technology was successfully extended into the elastic-plastic behavior regime
  - Elastic Plastic Fracture Mechanics (EPFM) has general engineering usefulness with a state of the art equivalent to LEFM

- Work on extending EPFM to high temperature time-dependent fracture mechanics (HTTDFM) started in 1976
  - HTTDFM is currently in an advanced state of research and development
  - Crack growth and fracture under creep and creep-fatigue loading conditions
  - Potential applications: high temperature turbines, coal conversion systems, advanced reactors
Introduction to Fracture Mechanics

- Since ~1970 fracture mechanics has been used as basis for ASME Boiler & Pressure Vessel Code Section III fracture prevention criteria
  - Provides the technical basis for development of P-T operating curves, ASME Section XI flaw evaluation, etc.

- Crack driving force is a function of the applied stresses, the size of the crack in the subject body, and body geometry factors

- The basic concept: the relationship between the applied cracked body (structure) loading conditions and the material resistance to crack growth and fracture
Fracture Mechanics – Basic Concept

- If a material's resistance to failure in the presence of a sharp crack is less than the crack-tip stress-strain conditions imposed by the loading and geometry conditions, structural failure will occur.

- To avoid failure:

  **Material Resistance (MR) > Crack Driving Force (CDF)**

- In all three generalized categories of fracture mechanics (LEFM, EPFM, and HTTDFM) the basic idea is to describe both sides of this equation in common terms representing both the crack-driving conditions and the material behavior (response) to these conditions.

  - Different crack-tip characterizing parameters have been developed for each of the general categories.
Elements of Fracture Mechanics
Variables Affecting Material Fracture Toughness of Ferritic Steels

- External and mechanical variables
  - Temperature
  - Loading rate
  - Environment (neutron irradiation, corrosive, etc.)

- Material variables
  - Chemical composition/impurities
  - Heat treatment
  - Microstructure
  - Strength level
  - Fabrication (welding method, rolling practice, etc.)
  - Time-temperature metallurgical changes (temper embrittlement)
Generalized Categories Fracture Mechanics

LEFM
Linear-elastic
Localized yielding at crack tip

EPFM
Elastic-plastic
Net section yielding

HTTDFM
High temperature-time dependent
Entire section yielding
Linear Elastic Fracture Mechanics

LEFM is based on elastic stress analysis of relatively brittle materials containing infinitely sharp cracks.

The intensity of the localized, elastic, stress-strain field in the vicinity near the crack-tip is described in terms of a singular term called the stress intensity factor, K.

K parameter usually includes a subscript (I, II, or III) which refers to the 3 different modes of loading a cracked body:

- Mode I: the opening mode where the cracked body is loaded by normal stresses (technically the most important loading mode)
- Mode II: sliding or in-plane shearing mode
- Mode III: tearing mode caused by out-of-plane shear
Modes of Cracking

- **Mode I** – Opening Mode
- **Mode II** – Sliding Mode
- **Mode III** – Tearing Mode
Stress Intensity Factor, $K$

- $K_I$ is the LEFM parameter of concern for RPV integrity

The magnitude of the intensification of the elastic stresses in the region of the crack tip can be described by a unique singular term, $K$.

\[ K = \text{Stress intensity factor} \]

(units, ksi$\sqrt{\text{in.}}$ or MPa$\sqrt{\text{m}}$)
Stress Intensity Factor, $K$

- $K$ is dependent on:
  - Externally applied load, $\sigma$
  - Length of crack, $a$
  - Geometry of cracked body and method of load application, $G$

- $K = f\sqrt{a} \ G$

- Crack initiation occurs if:
  $$K > K_{lc}$$

  \[
  \begin{align*}
  \text{Applied Stress} & \quad \text{Material}
  \\
  \text{Intensity} & \quad \text{Toughness}
  \end{align*}
  \]
Stress Intensity Factor, $K$

- Continuum mechanics solutions for prescribed applied loadings and geometries permit characterization of stress (and deformation) fields near a crack tip
  - Functional form of the local asymptotic field includes a scalar amplitude value of $K$ that can be expressed in Mode I loading for crack opening in the $yy$-direction as

$$\sigma_{yy} = \frac{K_I}{(2\pi r)^{1/2}}$$

as $r \to 0$
Plane-Strain Fracture Toughness, $K_{lc}$

- $K_{lc}$ is a measure of the plane-strain, brittle fracture resistance of a material and is commonly referred to as plane strain fracture toughness

- Unstable, rapid crack extension is predicted to occur when $K$ reaches $K_{lc}$

- $K_{lc}$ is a unique material property for a given material condition, temperature, and loading rate

- $K_{lc}$ can be measured in the laboratory using the specimen and test procedures described in ASTM Standard Test Method E 399 for “Plane-Strain Fracture Toughness of Metallic Materials”
Plane Strain vs. Plane Stress

- Transverse contractions in z-direction are opposed by unyielding faces of fatigue crack area resulting in transverse stresses $\sigma_{xx}$ and $\sigma_{zz}$ ahead of the crack.

- Plane strain occurs when $\varepsilon_{zz} = 0$

- Plane stress occurs when $\sigma_{zz} = 0$
Plane-Strain Fracture Toughness, $K_{lc}$

- $K_{lc}$ can be used to evaluate the brittle fracture conditions in any other linear-elastic loaded cracked body of practical interest so long as it contains the same material, is loaded at the same rate, and is at the same temperature as the laboratory test.

- The unified fracture toughness standard ASTM Test Method E 1820 for “Measurement of Fracture Toughness” covers both potential LEFM and EPFM fracture toughness measurement; although ASTM E 399 is still used for LEFM.
Plane-Strain Crack Arrest Fracture Toughness, $K_{la}$

- $K_{la}$ is a measure of the plane-strain crack-arrest toughness of ferritic steels
  - represents the level of $K$ at which a rapidly running crack can be arrested. It is a material property that has a unique value for a given material at a given temperature
    - ASTM E 1221

- A crack can initiate in a local region of a structure due to a combination of factors (e.g., low temp., high stresses, embrittled material, and high loading rates).

- The fast running crack may eventually run into a region of higher temperatures, lower stresses, or greater toughness.

- The crack will arrest only when the applied $K$ (crack driving force) is less than $K_{la}$
Comparison of $K_{la}$ with $K_{lc}$ for a Ferritic Pressure Vessel Steel

Data for a typical thick plate of SA533B-1 pressure vessel steel

Fracture Toughness, ksi $\sqrt{\text{in}}$

$K_{lc}$ fracture toughness

$K_{la}$ Crack-arrest fracture toughness

Temperature, °F
Material Resistance to Crack Growth

- Termination of the life of a component may be based on the applied stress intensity factor reaching a critical value ($K_{IC}$), representing the material's brittle fracture resistance.

- Useful life of a component depends on the rate of growth of flaws (cracks) from some subcritical size to the critical size when $K$ reaches $K_{IC}$.

- $K$ is a function of applied stress ($\sigma$), the cracked body geometry, and the crack size ($a$).
  - In order to utilize LEFM concepts it is necessary to characterize the material resistance to crack growth in terms of $K$ under cyclic loading (e.g., fatigue) and/or static loading (e.g., stress corrosion) conditions.
Fatigue Crack Growth

- The rate of crack growth expressed as $da/dN$ (change in crack size, $a$, with respect to elapsed cycles of loading, $N$) depends primarily on the cyclic range of the applied stress intensity factor, $\Delta K$, above a threshold $\Delta K_{th}$
  - $\Delta K$ is analogous to $\Delta \sigma$ as commonly used in conventional fatigue analyses

- Select a $K$ expression applicable to the geometry and loading method of practical interest and substitute $\Delta \sigma$ for $\sigma$ in order to compute $\Delta K$ rather than $K$
  - For Mode I loading (simple tensile opening of a crack): $K_I = \sigma(\pi a)^{\frac{1}{2}}$
  - For fatigue analysis: $\Delta K_I = \Delta \sigma(\pi a)^{\frac{1}{2}}$
Schematic Representation of Fatigue Crack Growth Behavior in a Non-hostile Environment
Stress Corrosion Crack Growth

- For some combinations of materials and environments, it is possible for cracks to grow under sustained (constant) loading conditions.

- Two characterization parameters:
  - Material behavior under constant loading conditions in an aggressive environment can be characterized in terms of $K_{\text{Iscc}}$, a threshold level of K below which a crack will not grow for a given material and environment.
  - If the applied K level is greater than the $K_{\text{Iscc}}$ threshold, the material crack growth rate behavior can be characterized in terms of $da/dt$ versus K (crack growth rate per unit time as a function of the applied K).
Schematic Representation of Stress Corrosion Crack Growth Behavior in an Aggressive Environment

- **Region I**: K dependent region
- **Region II**: Constant growth rate
- **Region III**: Approaching final failure
- **Specimen under a static load in an aggressive environment**

Log Crack Growth Rate, \(da/dt\)

Stress intensity factor, \(K\)
## LEFM Material Parameters and Test Methods

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Characterizes</th>
<th>Comments</th>
<th>ASTM Test Method</th>
</tr>
</thead>
<tbody>
<tr>
<td>$K_{ic}$</td>
<td>Plane strain, brittle fracture toughness</td>
<td>Material property, static &amp; dynamic</td>
<td>E399-12e3, E1820-15a</td>
</tr>
<tr>
<td>$K_{ia}$</td>
<td>Plane strain, crack arrest toughness</td>
<td>$K_i$ when running crack is arrested</td>
<td>E1221-12a</td>
</tr>
<tr>
<td>$K_{sc}$</td>
<td>Threshold for SCC propagation</td>
<td>Sustained loading and environment</td>
<td>E1681-03 (2013)</td>
</tr>
<tr>
<td>$\frac{da}{dt}$ vs. $K$</td>
<td>Growth rate for SCC</td>
<td>Sustained loading and environment</td>
<td>Under development</td>
</tr>
<tr>
<td>$\Delta K_{th}$</td>
<td>Fatigue crack growth threshold</td>
<td>Region I crack growth</td>
<td>Under development</td>
</tr>
<tr>
<td>$\frac{da}{dn}$ vs. $\Delta K$</td>
<td>Fatigue crack growth rates</td>
<td>Region II crack growth</td>
<td>E647-15</td>
</tr>
</tbody>
</table>
Limitations of LEFM Concepts

- The entire concept of LEFM is based on elastic stress analysis of cracked bodies
  - Limited to small-scale yielding

- The most significant limitation of LEFM is that the amount of local crack-tip plasticity prior to and/or during the crack growth or fracture processes must be small by comparison to the K zone in which the elastic-stress field equations apply
Elastic Plastic Fracture Mechanics

- J-integral approach to EPFM is most popular
- J is a field parameter that defines the plastic stress and strain intensity in the region around a crack tip
  - A function of stress, strain, crack size, and geometry of the crack and body
  - J is directly analogous to K used in LEFM
Elastic Plastic Fracture Mechanics

- $J_{lc}$ defines the level of applied $J$ at the onset of ductile, stable crack extension during monotonic loading of a precracked specimen at a temperature within the ductile behavior regime (i.e., the ductile upper shelf for ferritic materials).

- $J_{lc}$ is a basic material property representing a lower-bound measure of ductile fracture toughness in the presence of an initial sharp crack (fatigue precrack).

- Tearing modulus, $T$, accounts for sustained stable crack growth and is treated as a material parameter:

$$T = \left[ \frac{E}{\sigma_0^2} \right] (dJ/da)$$
Stages of Ductile Fracture Process (J-resistance or J-R Curve)
<table>
<thead>
<tr>
<th>Parameter</th>
<th>Characterizes</th>
<th>Comments</th>
<th>ASTM Test Method</th>
</tr>
</thead>
<tbody>
<tr>
<td>$J_{lc}$ or $J_c$</td>
<td>Initiation J for ductile crack extension</td>
<td>Material property, static &amp; dynamic</td>
<td>Old E 813, now E1820-15a; also E1921-15ae1 for ferritic steels</td>
</tr>
<tr>
<td>J-R Curve</td>
<td>Resistance to stable, ductile crack growth</td>
<td>J-$\Delta$a under monotonic loading</td>
<td>Old E1152, now E1820-15a (unified)</td>
</tr>
<tr>
<td>$T_0$</td>
<td>Ductile-cleavage transition temperature</td>
<td>Master Curve application</td>
<td>E1921-15ae1</td>
</tr>
<tr>
<td>da/dn vs. $\Delta J$</td>
<td>Fatigue crack growth rates</td>
<td>Crack extension per cycle of $\Delta J$</td>
<td>Under consideration</td>
</tr>
<tr>
<td>da/dt vs. C*</td>
<td>Creep crack growth rate</td>
<td>High temperature, time-dependent</td>
<td>E1457-15</td>
</tr>
</tbody>
</table>
Two Most Common Specimens Used for $J_{lc}$ and J-R Curve Fracture Toughness Tests

a.) Compact specimen for J testing

b.) Three-point bend specimen for J testing

Thickness "B" for both specimen types is nominally 0.5W
LEFM Requires Large Specimens for Fracture Toughness Characterization of RPV Steels – Evolution of EPFM Allows for Use of Small Specimens
Selecting Appropriate Fracture Toughness Criterion for Materials Exhibiting Ductile-to-Brittle Transition Behavior

Region I
LEFM

Brittle Cleavage Fracture Mode
Use $K_{lc}$

Region II
Transition Range
LEFM or EPFM

Mixed fracture mode
Use $K_{lc}$, $J_C$ or $J_{IC}$

Region III
Ductile Upper Shelf
EPFM

Ductile Fracture Mode
Use $J_{lc}$ and J-R curves

Fracture Toughness

Temperature
Probabilistic/Deterministic Fracture Mechanics

- Fracture mechanics analyses can be performed either deterministically or probabilistically
  - Deterministic methodology is used when all the input information to the analysis is considered to be known with certainty or when conservative estimates provide acceptable results
  - Probabilistic fracture mechanics has evolved from the need to provide results more representative of actual situations than conservative lower bound analyses could offer
    - Used for reliability analyses of components such as reactor pressure vessels, piping, steam turbine generator rotors and gas turbine disks and blades
Probabilistic Fracture Mechanics

- Purpose of Probabilistic Fracture Mechanics (PFM) is:
  - To estimate or bound the reliability \([1 - (\text{probability of failure})]\) of a component subject to cracking, and
  - To quantify the influence of engineering and management decisions on component reliability

- Deterministic methods assume inputs are known or use conservative or “worst-case,” estimates
  - Multiple, compounded conservatisms can lead to overly pessimistic results
  - PFM combines conventional fracture mechanics calculations with appropriate statistical methods to minimize stacking of conservatisms
FAVOR

- FAVOR code (Fracture Analysis of Vessels – Oak Ridge)
  - Developed at Oak Ridge National Laboratory (ORNL) under NRC Research funding
  - Used for the reassessment of pressurized thermal shock (PTS) rule and for risk-informed approach for RPV operating curves

- FAVOR PFM model uses Monte Carlo techniques
  - Deterministic fracture analyses are performed on a large number of stochastically generated RPV trials or realizations
  - Each trial considers the uncertainties of the vessel’s properties and the postulated flaw population, which are described by statistical distributions
  - Trials propagate the input uncertainties (with associated interactions) through the model and determine probability of crack initiation and through-wall cracking
Review of Learning Objectives

- Obtain a basic understanding of the use of fracture mechanics as it is applied to reactor pressure vessel integrity
- Recognize the role of fracture mechanics in the industry standards and regulations regarding vessel integrity and embrittlement
- Be able to identify the three categories of fracture mechanics and associated test methods
- Be aware of differences between probabilistic vs. deterministic fracture mechanics methods

Additional resource:
Together...Shaping the Future of Electricity
5 - Basic Concepts of Embrittlement Monitoring and Management

William Server
ATI Consulting

RPV Integrity Workshop,
2016 International LWR Materials Reliability Conference,
Chicago, USA
August 1, 2016
Basic Concepts

- Learning Objectives:
  - What is Reference Temperature?
  - What are Reference Toughness Curves?
  - What is Adjusted Reference Temperature?
  - How is RPV fluence determined?
  - How does fluence attenuate through an RPV wall?
Reference Temperature
Reference Temperature

- Direct fracture toughness testing of RPV steels is not required by ASME.
- The ductile-to-brittle behavior of ferritic RPV steels is used to define a reference temperature which can be indexed to a reference toughness curve to determine lower-bound fracture toughness for the material at any temperature.
- Reference temperature can also be adjusted for neutron embrittlement effects.
- ASME Code Section III, NB-2300 contains a definition for the initial (unirradiated) reference temperature, $RT_{NDT}$.
- Requires 2 tests: $T_{NDT}$ and Charpy.
- WRC-175 task group thought that use of both NDT and CVN tests for determining reference temperature gives protection against the possibility of errors in conducting the tests or the reporting of test results.
Temperature for Nil Ductility Transition, \( T_{NDT} \)
Nil Ductility Transition Temperature ($T_{NDT}$)

- The upper-most temperature for the lower shelf can be correlated with the nil-ductility transition (NDT)
- NDT marks the beginning of the transition from brittle to ductile
  - As temperature increases above NDT, a material exhibits increased toughness and the potential for brittle fracture diminishes
- NDT is determined using ASTM Standard Test Method E 208
  - A drop-weight test
  - Test specimen has notched brittle weld bead
Nil Ductility Transition Temperature ($T_{\text{NDT}}$)

- NDT is determined using ASTM Standard Test Method E 208
  - A drop-weight test
  - Test specimen has notched brittle weld bead
- Several specimens are tested over a range of temperatures to determine the highest temperature at which the specimen will break when a weight is dropped on it
- NDT is the highest temperature where a specimen breaks if two tests at $5^\circ\text{C}$ ($10^\circ\text{F}$) higher do not result in a break
Charpy V-Notch Test
Charpy V-notch Testing

- CVN testing was one of the earliest procedures to be employed for determining a brittle-to-ductile transition.
- A notched bar specimen is supported at both ends and struck behind the notch by a striker mounted at the end of a swinging pendulum.
- CVN energy (the energy absorbed by the specimen) is calculated from the height to which the pendulum rises after the strike (compared to the height it would rise had there been no specimen).
- Test standard: ASTM E 23.
- Test results: CVN energy, lateral expansion, and fracture appearance (percent shear).
CVN Test Shows Ductile-to-Brittle Transition

From NUREG-1807
Charpy V-notch Testing: Measured parameters

- **Impact energy (CVE)** - the energy needed to fracture the test specimen
  - determined directly from the impact test machine scale (corrected for windage and friction losses)

- **Lateral expansion (LE)** - the amount of deformation caused by the pendulum striking the specimen resulting in the expansion of the specimen thickness

- **Shear fracture (% Shear)** - the percent area on the face of the fractured specimen that is attributed to brittle failure
  - Not commonly used as a specification
  - Plays important role in determination of Upper Shelf Energy (USE)
CVN Index Temperatures

- CVN data can be plotted using tanh curve fit
- From plot, index temperatures are defined
  - From plot of CVN Absorbed energy vs. Temperature:
    - $T_{30}$ [30 ft-lb (41J) Transition Temperature]
    - $T_{50}$ [50 ft lb (68 J) Transition Temperature]
  - From plot of Lateral Expansion (mils) vs. Temp:
    - $T_{35\text{mil}}$ [35 mil (0.89 mm) Lateral Expansion Temperature]
- Comparing the unirradiated material CVN index temperatures to the irradiated plot index temperatures characterizes embrittlement in terms of $\Delta T_{30}$, $\Delta T_{50}$, $\Delta T_{35\text{mils}}$
Index Temperature Shift Due to Neutron Embrittlement

\[ \Delta T_{30} = \Delta R_{NDT} \]

Temperature (°F)

CVN Energy (ft-lb)

30 ft-lbs

Unirradiated

Irradiated
Using Charpy V-Notch Test and $T_{NDT}$ to Determine Initial (Unirradiated) Reference Temperature
Determination of Initial RT$_{NDT}$ (ASME III, NB-2300)

1. Determine a temperature $T_{NDT}$ that is at or above the nil-ductility transition temperature obtained by drop weight testing.

2. At a temperature not greater than $T_{NDT} + 60^\circ$F (33°C), each of 3 transverse-oriented CVN test specimens shall exhibit at least 35 mils lateral expansion and not less than 50 ft-lb (68 J) absorbed energy. When these requirements are met, then $T_{NDT}$ is the reference temperature RT$_{NDT}$. [Note: the lateral expansion requirement almost never enters into the setting of RT$_{NDT}$ for pressure vessel steels.]

3. In the event the requirements of (2) are not met, conduct additional CVN tests in groups of 3 specimens to determine the temperature $T_{cv}$ at which the requirements of (2) are met. In this case, the reference temperature RT$_{NDT} = T_{cv} - 60^\circ$F (33°C). Thus RT$_{NDT}$ is the higher of $T_{NDT}$ or ($T_{cv} - 60^\circ$F [33°C]). [Note: Interpolation of the minimum data points is permitted for (2) or (3).]

**Note**: Full Charpy curves are not required for determination of RT$_{NDT}$. 
Summary - Reference Temperature, $RT_{NDT}$

- $RT_{NDT}$ is both a heat-to-heat material normalizing parameter and an indexing parameter to determine the fracture toughness properties used in the fracture mechanics-based analyses.
- $RT_{NDT}$ permits a single set of RPV toughness rules to be established for vessels fabricated of many different heats of LAS, all with different fracture toughness.
- $RT_{NDT}$ is not a direct fracture toughness index temperature because it is based on non-fracture toughness tests (NDT and Charpy); however, it can be correlated with fracture toughness and is used as an indexing parameter.
- $RT_{NDT}$ is a tool that allows application of a single reference toughness curve to all RPV steels.
Considerations Regarding Determination of $RT_{NDT}$ for Older Plants

- Fracture toughness testing performed for older plants on vessel material did not usually include all tests necessary to determine $RT_{NDT}$ per the NB-2331 requirements
  - Acceptable estimation methods for the most common cases are provided in Branch Technical Position 5-3 for determining $RT_{NDT}$ when measured values are not available
  - There are current issues with BTP 5-3 that will be discussed later
Fracture Toughness Reference Curves
Development of Reference Fracture Toughness Curves

- All available fracture toughness test data available at the time (e.g., early 1970’s) were gathered
  - Crack arrest data ($K_{Ia}$) separate from static initiation data ($K_{Ic}$)
- Data: a fracture toughness / test temperature pair
- Each data point was normalized and plotted by subtracting the $RT_{NDT}$ of the tested material from the test temperature at which that test was conducted
- A bounding curve was drawn

Note: $K_{IR} = K_{IA} = K_{ID}$
Normalized Fracture Toughness Data (Static) Used for $K_{IC}$ Curve Derivation

From NUREG-1807
Comparison of $K_{IC}$ and $K_{IA}$ (K$_{IR}$) Curves

Fracture toughness for crack initiation under slow loading rates ($K_{IC}$) is significantly higher at any given temperature compared to fracture toughness under dynamic loading conditions ($K_{ID}/K_{IA}$).
How Toughness Curves are Adjusted for Irradiation

- Irradiated toughness curve shape is sufficiently similar to unirradiated curve that the shapes are assumed to be the same
- Therefore, the toughness after irradiation can be determined simply by adding a shift term to $RT_{NDT}$
- Determining irradiation shift by irradiating actual fracture toughness specimens (that meet size requirements) in a typical RPV was impracticable. How else could the shift term be determined?
- Charpy impact specimens were small enough for irradiation in surveillance capsules and the Charpy shift was used to measure shift in fracture toughness curves
$\Delta R_{N_D T}$ Assumed Equal to Charpy $\Delta T_{30}$
Relationship between $\Delta T_o$ and $\Delta T_{41J}$ is not always 1:1

→Testing by Nanstad and Sokolov has shown that $\Delta R_{T_{NDT}}$ is not exactly equal to Charpy $\Delta T_{30}$, however; but in practice it is still assumed to be equal (except in FAVOR)

Figure from ORNL/TM-2007/030
“Fracture Analysis of Vessels – Oak Ridge, FAVOR, v06.1, Computer Code: Theory and Implementation of Algorithms, Methods, and Correlations” (ML12193A386)
Calculation of Adjusted Reference Temperature
Adjusted Reference Temperature (ART)

- ART is a parameter developed to incorporate the irradiation effects encountered in various pressure vessel steels.
- ART is defined by Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials”.
- ART = Initial $RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$
- ART is a key concept for the regulations governing RPV embrittlement and integrity analyses. ART varies through wall.
  - For P-T limit curves, ART is calculated at the ¼ and ¾ vessel wall thickness locations (tip of reference flaws).
  - For Pressurized Thermal Shock screening analysis, ART is calculated at 0T (the inner wall) and is called $RT_{PTS}$.
Adjusted Reference Temperature

- \( \text{ART} = \text{Initial RT}_{\text{NDT}} + \Delta \text{RT}_{\text{NDT}} + \text{Margin} \)

- Initial \( \text{RT}_{\text{NDT}} \) is the reference temperature for the unirradiated material and is a function of material chemistry, heat treatment, and fabrication process.
  - Determined per ASME Code Section III, paragraph NB-2331 as described earlier
Adjusted Reference Temperature

- ART = Initial RT<sub>NDT</sub> + ΔRT<sub>NDT</sub> + Margin

- ΔRT<sub>NDT</sub> is the mean value of the adjustment in RT<sub>NDT</sub> caused by irradiation.
  - a function of Cu and Ni content, irradiation temperature and fluence (usually the projected fluence at some future milestone, such as End of Life or a specific future EFPY).
  - Expressed as the product of a Chemistry Factor (CF) and a Fluence Factor (FF).
    \[
    ΔRT_{NDT} = (CF) f^{(0.28 - 0.10 \log f)}
    \]
Adjusted Reference Temperature

- ART = Initial RT<sub>NDT</sub> + ΔRT<sub>NDT</sub> + Margin

\[
ΔRT_{NDT} = (CF) f^{(0.28 - 0.10 \log f)}
\]

- CF (°F) is the chemistry factor, given in tables in Reg. Guide 1.99 Rev. 2, as a function of material chemistry and product form – or, alternatively, CF may be derived from plant surveillance data if certain criteria are met.

- \( f^{(0.28 - 0.10 \log f)} \) is the fluence factor (FF).

- \( f \) which is the neutron fluence at any depth in the vessel wall (in units of \( 10^{19} \text{ n/cm}^2, \ E > 1 \text{ MeV} \)), determined as
  - \( f = f_{\text{surf}} \left( e^{-0.24x} \right) \)
Attenuation of $f$

$f = f_{\text{surf}} \left( e^{-0.24x} \right)$

- $f_{\text{surf}} (10^{19} \text{ n/cm}^2, E > 1 \text{ MeV})$ is calculated at the inner wetted surface of the vessel at the location of the postulated defect, and $x$ (in inches) is the depth into the vessel wall measured from the inner surface.
- $e^{-0.24x}$ converts fluence to a “dpa equivalent” based on calculations published in 1982 which showed attenuation of dpa is less than fluence.
- If dpa is available, the ratio of dpa at the depth in question to dpa at the surface may be used as a substitute for the exponential attenuation factor.
Adjusted Reference Temperature

\[
\text{ART} = \text{Initial RT}_{NDT} + \Delta \text{RT}_{NDT} + \text{Margin}
\]

\[
\text{Margin} = 2 \sqrt{\sigma_{I}^2 + \sigma_{\Delta}^2}
\]

- \(\sigma_I\) is the standard deviation for the initial \(\text{RT}_{NDT}\). If a measured value of initial \(\text{RT}_{NDT}\) for the material in question is available, \(\sigma_I\) is to be estimated from the precision of the test method (and it is normally taken to be 0°F [0°C]). If not, and if generic mean values for the class of material are used, \(\sigma_I\) is the standard deviation obtained from the set of data used to establish the mean. If the generic mean Initial \(\text{RT}_{NDT}\) value of a Linde 80, 0091, 1092 and 124 or ARCOS B-5 weld is used, then \(\sigma_I\) is 17°F (9°C) [weld \(\sigma_I\) is from 10CFR50.61]

- The standard deviation for \(\Delta \text{RT}_{NDT}\), \(\sigma_\Delta\), is 28°F (16°C) for welds and 17°F (9°C) for base metal, except that \(\sigma_\Delta\) need not exceed 0.50 times the mean value of \(\Delta \text{RT}_{NDT}\).
Determination of Fluence
Determination of Fluence

- Accurate calculation of fluence values in the RPV beltline and surveillance capsules is an essential element of determining the effect of irradiation embrittlement on fracture toughness of the vessel materials.

- Each reactor is required to have an adequate dosimetry program (per 10 CFR 50 Appendix H).
Flux and Fluence

- Neutrons generated by fission “leak” out of the core and reach the vessel wall
- Neutrons generated by fission have different energies
  - Fast, intermediate, and thermal
- Two terms, flux and fluence, are used to quantify
  - Flux: product of the number of neutrons per unit volume and the neutron speed
    - Typical values in LWRs range from $10^{10}$ to $10^{14}$ n/cm$^2$-s
  - Fluence: the time integration of flux; the number of neutrons received per unit surface area of the irradiated material (n/cm$^2$)
    - Fluence is cumulative
Dosimetry Fundamentals

- Flux and fluence at any particular location are highly dependent on location relative to the core
  - Axial and azimuthal variation

Rollout diagram of beltline materials and representative fluence maps, from NUREG-1806
Displacements per Atom (dpa)

- dpa is another, more precise way to characterize embrittlement damage
- Definition: the number of times, on the average, an atom has been displaced from its lattice site during an irradiation.
  - Another definition says dpa is the total number of Frenkel pairs (one vacancy + one self-interstitial atom) created in a given volume by the neutrons, divided by the number of atoms in the volume.

- Whereas fluence is simply a measure of the neutrons to which a material is exposed, dpa measures the response of a material to those neutrons.
  - dpa increases with the amount of energy deposited by the neutrons and so is a function of the neutron spectrum.
  - dpa is also sometimes called dose; when expressed by time unit (dpa/s), is called dose rate.
Regulatory Guide 1.190

- Regulatory Guide 1.190 provides guidance for calculation and measurement of LWR vessel fluence
- Yields a best estimate fluence, rather than a bounding or conservative fluence
  - Uncertainty regarding fluence calculation is accounted for in the margin term of ART
- While the E > 1 MeV fluence is used as the exposure parameter for RT_{NDT} and RT_{PTS} correlations, RG 1.190 methodology determines the damage fluence spectrum (from 0.1 to 15 MeV) and is applicable to other exposure units, such as iron dpa
Determination of RPV fluence is based both on calculations and on dosimetry measurements:
- Calculations are performed to predict a fluence.
- Dosimetry measurements are used to qualify the calculational methodology by comparing the predictions to actual measurements.
- Comparison identifies biases (i.e., systematic errors) in the calculations; permits reliable estimates of fluence uncertainties and determination of effects of computer modeling approximations.
- If appropriate, the fluence calculations are modified by applying a correction to account for bias and/or by adjusting the computer model.
Regulatory Guide 1.190

- Calculation of fluence consists of the following steps:
  1. determination of geometrical and material input data
  2. determination of the core neutron source
  3. propagation of the neutron fluence from the core to the vessel and into the cavity, and
  4. qualification of the calculational procedure

- BWR Vessel and Internals Program (BWRVIP) developed RAMA (Radiation Analysis Modeling Application) Fluence Methodology software package
  - Approved for fluence evaluations in LWR components in compliance with Reg. Guide 1.190
  - Available from EPRI (BWRVIP-126, Revision 2)
Through-wall Attenuation Issues

- ART values are calculated at the ¼-thickness and ¾-thickness locations in the RPV wall (e.g., at postulated references flaw crack tips)
- $\Delta RT_{NDT}$ varies as a function of the radial depth into the wall because the neutron energy spectrum and fluence (of which the shift is a function) change significantly through the vessel wall
- As previously mentioned, Reg. Guide 1.99, Revision 2 uses an exponential decay in fluence (based on the decay in dpa) as the predictive model

$$f_x = f_{surf} \left( e^{-0.24x} \right)$$

where $f_{surf} \left( 10^{19} \text{ n/cm}^2, E > 1 \text{ MeV} \right)$ is calculated at the inner wetted surface of the vessel at the location of the postulated defect, and $x$ (in inches) is the depth into the vessel wall measured from the inner surface
Through-wall Attenuation Issues

- Fluence attenuation through-wall is more severe than actual damage attenuation; damage more closely follows dpa attenuation, but dpa attenuates less than fluence
  - Using fluence attenuation model would be non-conservative (it would underestimate damage)
  - However, when Reg. Guide 1.99 Rev. 2 was written, it was the common practice to use fluence
  - Therefore, the formula \( e^{-0.24x} \) approximates attenuation of damage while still using fluence terms
  - Formula was based on some dpa calculations performed in early 1980’s

Decay in Effective Fluence Through the Thickness of the Simulated RPV Wall (from MRP-243)
Attenuation based on dpa

- Reg. Guide 1.99 Rev. 2 allows option that, if damage in dpa has been established as a function of depth by approved neutron transport calculations and supporting dosimetry, the calculated dpa can be used directly to attenuate fluence by:

\[ f_x = f_{\text{surf}} \frac{D_x}{D_{\text{surf}}} \]

where \( D_x \) is the damage in dpa at a depth, \( x \), and \( D_{\text{surf}} \) is the damage at the inner surface of the vessel.

- In either case \([f_{\text{surf}} \left( e^{-0.24x} \right) \) or \( f_{\text{surf}} \frac{D_x}{D_{\text{surf}}} \)], fluence at any depth is based indirectly or directly on the attenuation of dpa through the vessel wall.
Attenuation in Nozzles

- $f_x = f_{\text{surf}} (e^{-0.24x})$ is valid only through the thickness of the vessel beltline shell.

- Recent studies (MRP-345, *Fluence Attenuation Profile in the PWR Nozzle Shell Course* (Inlet/Outlet Nozzles)) suggest the relationship does not correctly characterize attenuation in nozzles due to possible neutron streaming from the cavity.

- It is recommended that fluence at depth $x$ in a PWR nozzle be based on analysis which explicitly models the nozzles and considers streaming effect.
Together…Shaping the Future of Electricity
6 – Criteria and Requirements for Vessel Fracture Toughness and Fracture Prevention

Tim Hardin
Technical Executive

RPV Integrity Workshop,
2016 International LWR Materials Reliability Conference, Chicago, USA
August 1, 2016
Learning Objectives

- Design criteria relative to vessel integrity and fracture toughness
- Role and relationship of NRC regulations and the ASME Boiler & Pressure Vessel Code
- Overview of the primary governing regulations for vessel fracture toughness and fracture prevention
- Future directions
Relationship of 10 CFR 50 and ASME Code (1/3)

- Rules for design & operation of reactor pressure vessels are provided in the Code of Federal Regulations (10 CFR 50, Domestic Licensing of Production and Utilization Facilities) and the ASME Boiler & Pressure Vessel Code Sections III and XI
- 10 CFR 50 and the ASME Code have a synergistic relationship
- 10 CFR 50 is ultimately the governing document by law but it invokes by reference selected sections of the ASME Code
Relationship of 10 CFR 50 and ASME Code (2/3)

- 10 CFR has used provisions of the ASME Code since 1971 as one part of the 10 CFR framework to establish the necessary design, fabrication, construction, testing, and performance requirements for components important to safety.

- 10 CFR 50.55a(g), “Inservice Inspection Requirements,” requires, in part, that certain component (including RPVs) meet the requirements of ASME Code Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” or equivalent quality standards.
  - The latest editions and addenda of Section XI that the NRC has approved for use are specified in 10 CFR 50.55a(b).
Relationship of 10 CFR 50 and ASME Code (3/3)

- ASME publishes Code Cases quarterly
  - Code Cases provide alternatives to existing Code requirements that the ASME developed and approved

- Regulatory Guide 1.147 (updated periodically) identifies the Section XI Code Cases acceptable to NRC for use without prior NRC authorization
  - RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III” for ASME III CC
  - RG 1.193, “ASME Code Cases Not Approved For Use” identifies the Code Cases not approved by NRC

- Section XI Code Cases that the NRC has not yet endorsed may be implemented through 10 CFR 50.55a(a)(3)
  - 10 CFR 50.55a(a)(3) permits use of alternatives if they result in an acceptable level of quality and safety and if the NRC’s Office of Nuclear Reactor Regulation (NRR) specifically authorizes the alternative
General Design Criteria, 10 CFR 50 Appendix A (1/3)

- Reactor vessel integrity begins with design, fabrication, and quality assurance practices
- Appendix A to 10 CFR 50 provides the General Design Criteria for Nuclear Power Plants
  - Govern applications for construction permits for nuclear power plants
  - Not directly applicable to the licensing basis for many plants - especially older plants - and do not provide explicit fracture toughness criteria for operating plants
  - Provide insight into design philosophy
General Design Criteria, 10 CFR 50 Appendix A (2/3)

- GDC 14: reactor coolant pressure boundary (RCPB) must have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture

- GDC 31: RCPB shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions
  - Pressure boundary behaves in a nonbrittle manner
  - Probability of rapidly propagating fracture is minimized
  - Design shall reflect consideration of service temperatures…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
GDC 32: components which are part of the RCPB shall be designed to permit

1. periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity, and
2. an appropriate material surveillance program for the reactor pressure vessel
Applicable ASME Design Codes

- Many older vessels were designed and built to different design codes
- The code of construction was usually the latest version of the ASME Code in place at the time the vessel was ordered
  - ASME Section I and VIII (pre-1960)
  - ASME Section III (after 1963; also, a major revision of Section III was adopted in 1972)
ASME Code Rules for Prevention of Brittle Fracture in Vessels

- ASME Section III, Appendix G was adopted in 1972, providing procedures for establishing allowable loading in order to prevent nonductile failure
  - Based on recommendations of PVRC Welding Research Council Bulletin No. 175, based on fracture mechanics methods
    - \( RT_{NDT} \) reference temperature defined in Subsection NB-2331
- ASME Section XI, Appendix G adopts methodology for P-T limits and establishes procedure for LTOP setpoints (1993) replacing ASME Section III, App. G
- ASME Section XI, Appendix E provides methodology to evaluate overpressurization events on reactor vessel integrity
- ASME Section XI, Appendix K provides evaluation methods for low upper shelf energy (USE) materials
NRC Regulations on Embrittlement and Integrity of Reactor Pressure Vessels

- 10CFR50 Appendix G establishes requirements for vessel fracture toughness
- 10CFR50 Appendix H establishes requirements for vessel material surveillance programs
- PTS Rule (10CFR50.61) establishes screening criteria limits for pressurized thermal shock (PTS) in PWR vessels
  - Alternative PTS Rule (10CFR50.61a) establishes alternative PTS screening limits for PWR vessels
  - PTS rules not applicable to BWRs
Regulations/Guidance for Predicting Effects of Neutron Irradiation

- To demonstrate compliance with regulations throughout the service life of an RPV, a means of predicting future toughness is required.
- In the PTS Rules (both 10CFR50.61 and 61a), trend curves for making embrittlement ($\Delta T_{30}$) predictions are specified within the Rule.
- 10CFR50 Appendix G does not provide a trend curve but states:
  - “For the reactor vessel beltline materials, including welds, plates and forgings, the values of $R T_{NDT}$ and Charpy upper shelf energy must account for the effects of neutron radiation, including the results of the surveillance program of Appendix H of this part.”
- Regulatory Guide 1.99, Rev. 2 provides guidance for calculating embrittlement effects on $R T_{NDT}$ shift for P-T curves, and USE drop.
  - This trend curve for $\Delta T_{30}$ is identical to the $\Delta T_{30}$ trend curve in 10CFR50.61.
  - Although Regulatory Guides simply provide one method acceptable to the NRC and are not regulations, no other trend curve is used for P-T and USE.
# Primary Reactor Vessel Integrity Regulations

<table>
<thead>
<tr>
<th>Low Upper Shelf Toughness</th>
<th>Pressure-Temperature Limits</th>
<th>Reactor Vessel Material Surveillance</th>
<th>Pressurized Thermal Shock</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>10CFR50 Appendix G</strong></td>
<td><strong>10CFR50 Appendix G</strong></td>
<td><strong>10CFR50 Appendix H</strong></td>
<td><strong>10CFR50.61</strong></td>
</tr>
<tr>
<td><strong>ASME Section XI Appendix K (for Equivalent Margins Analysis [EMA])</strong></td>
<td><strong>ASME Section XI Appendices G &amp; E</strong></td>
<td><strong>ASTM E185 (-73, -79, -82) (Testing &amp; reporting procedures should meet ASTM E185-82)</strong></td>
<td></td>
</tr>
</tbody>
</table>
Overview of RPV Integrity Evaluation Process for Compliance with Fracture Toughness Regulations
Descriptions of RPV Integrity Regulations and Special Requirements
# Summary of PWR Embrittlement Requirements

<table>
<thead>
<tr>
<th>LWR Type</th>
<th>Regulatory Requirement / Limit for Embrittlement</th>
<th>Compensatory Actions</th>
</tr>
</thead>
</table>
| PWR      | 10 CFR 50 Appendix G Toughness (heatup and cooldown curves) | • Flux reduction  
          |                                                                 | • Anneal the vessel |
|          | 10 CFR 50 Appendix G Upper Shelf Energy            | • EMA  
          |                                                                 | • Flux reduction  
          |                                                                 | • Anneal the vessel |
|          | 10 CFR 50.61 PTS                                   | • 10 CFR 50.61a  
          |                                                                 | • Flux reduction  
          |                                                                 | • Anneal the vessel  
          |                                                                 | • Request exemption from 10 CFR 50.61 based on plant-specific fracture toughness measurements etc. |
|          | Partial relief from vessel weld inspection requirements of 10 CFR 50.55a by extension of ISI interval from 10-yr. to 20-yr. | • Demonstrate vessel embrittlement is bounded by analyses of WCAP-16168-NP-A Rev. 3. Otherwise, 10-yr. interval cannot be extended. |

License Renewal (if applicable)

- TLAA’s related to RPV embrittlement:
  - upper-shelf energy,  
  - PTS,  
  - heat-up and cool-down (pressure-temperature limits) curves,  
  - other plant-specific TLAA’s on reactor vessel neutron embrittlement.

- Appendix H RPV Surveillance Program

Pursuant to 10 CFR 54.21(c)(1)(i) - (iii), an applicant must demonstrate one of the following:

(i) The analyses remain valid for the period of extended operation;  
(ii) The analyses have been projected to the end of the extended period of operation; or  
(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Plant adopts an Aging Management Plan (AMP) for its App H RPV surveillance program consistent with the requirements of NUREG-1801, Rev. 2 (GALL report) Chapter XI.M31
# Summary of BWR Embrittlement Requirements

<table>
<thead>
<tr>
<th>LWR Type</th>
<th>Regulatory Requirement / Limit for Embrittlement</th>
<th>Compensatory Actions</th>
</tr>
</thead>
</table>
| BWR      | 10 CFR 50 Appendix G Toughness (heatup and cooldown curves) | • Flux reduction  
            • Anneal the vessel | |
|          | 10 CFR 50 Appendix G Upper Shelf Energy         | • EMA  
            • Flux reduction  
            • Anneal the vessel |
|          | Circumferential weld examination relief (if applicable) | • Flux reduction |
|          | Axial weld failure probability / embrittlement monitoring | • Flux reduction to reduce mean RT_{NDT} of the limiting axial weld  
            • A probabilistic analysis to show the NDT failure frequency for axial welds is < 5 x 10^-8 per reactor year |
| License Renewal (if applicable) | • TLAA's related to RPV embrittlement:  
                                          o upper-shelf energy,  
                                          o heat-up and cool-down (pressure-temperature limits) curves,  
                                          o BWR Vessel and Internals Project BWRVIP-05 analysis for elimination of circumferential weld inspection and analysis of the axial welds, and  
                                          o other plant-specific TLAA's on reactor vessel neutron embrittlement (e.g., reflood thermal shock analysis).  
                                          • Appendix H RPV Surveillance Program | Pursuant to 10 CFR 54.21(c)(1)(i) – (iii), an applicant must demonstrate one of the following:  
                                          (i) The analyses remain valid for the period of extended operation;  
                                          (ii) The analyses have been projected to the end of the extended period of operation; or  
                                          (iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.  
                                          Plant adopts an Aging Management Plan (AMP) for its App H RPV surveillance program consistent with the requirements of NUREG-1801, Rev. 2 (GALL report) Chapter XI M31 |

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Appendix G to 10 CFR 50, “Fracture Toughness Requirements”

- RPVs can continue to operate only for the service period where the specified fracture toughness criteria are satisfied
  - Upper Shelf Energy (USE)
    - Requirement applies to “beltline materials” only
  - Pressure-Temperature Limits and Minimum Temperature Requirements
    - These requirements apply to all ferritic materials in the vessel
    - $RT_{NDT}$ must be determined for each heat of vessel material
What is the definition of “beltline materials”?

- Definition has evolved over time (see below)
- In October 2014, NRC issued Regulatory Issue Summary 2014-11, “Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components” (ML14149A165) to clarify requirements

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>“…the shell material (including welds and weld heat-affected zones) that directly surrounds the effective height of the fuel element assemblies and any additional height of shell material for which the predicted adjustment of reference temperature at the end of service life of the reactor vessel exceeds 50°F.”</td>
<td>The region of the reactor vessel 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.”</td>
<td>“…the beltline definition in 10 CFR Part 50, Appendix G is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than $1 \times 10^{17}$ n/cm$^2$ (E &gt; 1 MeV)”</td>
</tr>
</tbody>
</table>
$10^{17}$ n/cm² is the fluence at which a statistically significant shift in $T_{41J}$ is observed.
Appendix G to 10 CFR 50: Upper Shelf Energy

- RPV beltline materials must have Charpy upper-shelf energy of no less than 75 ft-lb (102J) and must maintain upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb (68J), unless equivalent margins of safety can be shown.

- Orientation of the CVN specimens should be T-L.

- Although not explicitly stated, this is the requirement for USE at the 1/4T position in the vessel.
  - Appendix G references the ASME Code for requirements for the test specimens, which in turn requires the specimens to be taken at the 1/4T.

- If the minimum USE requirement at end of life is not met, an Equivalent Margins Analysis is required.
  - EMA is conducted at 2 locations (depending on service levels): 1/4T(Service Levels A and B) and 1/10T (Service Levels C and D).
Appendix G to 10 CFR 50: P-T Limits

- Pressure-Temperature Limits and Minimum Temperature Requirements
  - Invokes ASME XI Appendix G + additional requirements
  - P-T limits
    - Minimum temperatures for performing any hydrostatic test involving pressurization of the reactor vessel after installation in the system
    - Minimum temperatures for all leak and hydrostatic tests performed after the plant is in service
    - Maximum pressure-minimum temperature curves for operation, including startup, upset, and cooldown conditions
    - Maximum pressure-minimum temperature curves for core operation

- P-T Limits / Min. Temp. requirements will be covered in detail in a later presentation
Appendix G to 10 CFR 50

- If requirements cannot be met, “the reactor vessel beltline may be given a thermal annealing treatment to recover the fracture toughness of the material, subject to the requirements of § 50.66 [10 CFR 50.66, “Requirements for thermal annealing of the reactor pressure vessel”].

- “The reactor vessel may continue to be operated only for that service period within which the predicted fracture toughness of the beltline region materials satisfies the requirements of Section IV.A. of this appendix using the values of RT_{NDT} and Charpy upper-shelf energy that include the effects of annealing and subsequent irradiation.”
Special Note on Appendix G Compliance for Older Plants (1/2)

- Until 1973, ASME Section III required that materials used in the pressure-retaining components of RPVs be qualified by CVN impact tests on specimens oriented in the strong direction (Longitudinal, L-T).
- In Summer 1972 Addenda, the Code adopted a new approach to RPV integrity, based on LEFM with toughness characterized by $R_{NDT}$:
  - Pellici $T_{NDT}$ and $T_{cv}$ (lower-bound Charpy $T_{50 \text{ ft-lb}}$ / $T_{35 \text{ mils lat. exp.}}$) for specimens oriented in weak direction.
  - $RT_{NDT} = \text{MAX} [T_{NDT}, T_{cv} - 60^\circ F]$.
- 10 CFR 50, Appendix G, introduced in August 1973, required all operating power reactors to assess vessel integrity based on weak direction properties.
- Plants fabricated to version of Code earlier than ASME III Summer 1972 Addenda may not have the weak-direction test data to fully comply with present requirements in all respects.
Special Note on Appendix G Compliance for Older Plants (2/2)

- For those plants, alternative methods must be used to estimate $R_{NDT}$ and USE based on weak direction properties
  - A small number of plants use generic data
  - Many older PWRs use NUREG-0800 Branch Technical Position (BTP) 5-3
    - 9+ PWRs also use a GE Method for estimating $R_{NDT}$ of nozzle forgings
  - Nearly all older BWRs use the GE Method for estimating Initial $R_{NDT}$ and use the BTP 5-3 B1.2 for estimating USE

- In 2014 a vendor identified that some methods in BTP 5-3 are potentially non-conservative (PVP2014-28897)

- Since then, NRC has determined that both BTP and GE methods for $R_{NDT}$ are potentially non-conservative but there is no immediate safety issue
  - NRC found BTP B1.2 method for estimating weak-direction USE to be adequate
  - It is possible NRC will revise BTP 5-3

- Both Industry and NRC are presently evaluating and attempting to resolve this issue, which may necessitate some future action by plants
Appendix H to 10 CFR 50

- Establishes requirement for comprehensive surveillance programs
  - Prior to the introduction of Appendix H in 1973, plants had installed irradiation test samples using the guidance of the 1961 (tentative), 1962, 1966, 1970 or the then-emerging 1973 version of ASTM E-185

- RPVs that exceed $10^{17}$ n/cm$^2$ (E > 1 MeV) at the end-of-license are required to have an RPV material surveillance program

- The intent of surveillance programs is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline that result from exposure to neutron irradiation and the thermal environment

- Appendix H surveillance requirements will be covered in greater detail later
Fracture toughness requirements for protection against pressurized thermal shock events

\( RT_{PTS} \) is determined for all reactor vessel beltline plates and/or forgings, and axial and circumferential welds
- \( RT_{PTS} \) is ART at the vessel ID surface (not the 1/4T as for P-T curves)
  - PWR \( RT_{PTS} \) values will also be impacted by the BTP 5-3 issue discussed earlier
- Based on end of license (EOL) fluence
- Screening limits:
  - 270°F (132°C) for plates, forgings, and axial welds
  - 300°F (149°C) for circumferential welds
If RT\textsubscript{PTS} exceeds these screening limits, actions required:
- Implement flux reduction programs reasonably practicable
- Supplemental plant-specific analyses
- Thermal annealing
- Or, apply for licensing under Alternative PTS Rule, 10 CFR 50.61a

PTS will be covered in greater detail in later presentation
Future Regulatory Directions

- In recent years, NRC intended to pursue rulemaking to update 10CFR50 Appendix G and Reg. Guide 1.99 Rev. 2, and 10CFR50 Appendix H
- In 2016, NRC abandoned the Appendix G rulemaking effort for budgetary reasons
  - It is also unlikely that Reg. Guide 1.99 Rev 2 will be revised (insufficient safety benefit to justify a revision)
- However, NRC intends to proceed with rulemaking to update 10CFR50 Appendix H (surveillance program requirements) starting in 2017
  - Appendix H currently cites E185-82; intent is to cite a more recent version of E185 and also incorporate by reference E2215
    - E185 provides guidance for design of new surveillance programs
    - E2215 provides guidance for testing and evaluation of capsules
  - Also other changes; see ML16021A005 (NRC presentation 1/19/2016)
Together…Shaping the Future of Electricity
7 – Vessel Monitoring Through In-service Inspection

Jack Spanner
Technical Leader, Principal

RPV Integrity Workshop,
2016 International LWR Materials Reliability Conference,
Chicago, USA
August 1, 2016
Topics

- Ultrasonic fundamentals
  - Extent and frequency of RPV weld examinations
    - Examination volumes
- Volumetric exam of welds
  - UT requirements
  - UT qualifications
  - RPV UT scanners
- UT for Pressurized Thermal Shock
- Non-Visual examinations of Bottom Mounted Nozzles
UT - Rectified A-scan
Dolphins & Bats Use Ultrasonic Longitudinal Waves (Straight Beam)

Reflected sound energy is displayed versus time, and inspector can visualize a cross section of the specimen showing the depth of features that reflect sound. The distance ($D$) is readily calculated by $D = \text{Time} \times \text{Sound Velocity}$.
Ultrasonic Fundamentals
Example of Conventional Ultrasonic Testing
Section XI RPV Examinations

Meridional Weld not shown
# Frequency of RPV Weld Examinations - Every 10 Years

## Table IWB-2500-1 (B-A)

**Examination Category B-A, Pressure Retaining Welds in Reactor Vessel**

<table>
<thead>
<tr>
<th>Item No.</th>
<th>Parts Examined</th>
<th>Examination Requirements/Figure No.</th>
<th>Examination Method</th>
<th>Acceptance Standard</th>
<th>Extent and Frequency of Examination</th>
<th>Deferral of Examination to End of Interval</th>
</tr>
</thead>
<tbody>
<tr>
<td>BL10</td>
<td>Shell welds</td>
<td>IWB-2500-1</td>
<td>Volumetric</td>
<td>IWB-3510</td>
<td>All welds [Note (2)]</td>
<td>Permissible</td>
</tr>
<tr>
<td>BL11</td>
<td>Circumferential</td>
<td>IWB-2500-2</td>
<td>Volumetric</td>
<td>IWB-3510</td>
<td>Accessible length of all welds [Note (2)]</td>
<td>Permissible</td>
</tr>
<tr>
<td>BL12</td>
<td>Longitudinal</td>
<td>IWB-2500-3</td>
<td>Volumetric</td>
<td>IWB-3510</td>
<td>Weld [Note (2)]</td>
<td>Permissible [Note (3), [Note (4)]</td>
</tr>
<tr>
<td>BL20</td>
<td>Head welds</td>
<td>IWB-2500-4</td>
<td>Volumetric</td>
<td>IWB-3510</td>
<td>Weld [Note (2)]</td>
<td>Permissible [Note (4), [Note (5)]</td>
</tr>
<tr>
<td>BL21</td>
<td>Circumferential</td>
<td>IWB-2500-5</td>
<td>Volumetric and surface</td>
<td>IWB-3510</td>
<td>Weld [Note (2)]</td>
<td>Permissible</td>
</tr>
<tr>
<td>BL22</td>
<td>Meridional</td>
<td>IWB-2500-6</td>
<td>Volumetric</td>
<td>IWB-3510</td>
<td>All weld repair areas</td>
<td>Permissible</td>
</tr>
</tbody>
</table>

### NOTES:

1. Material (base metal) weld repairs where repair depth exceeds 10% nominal of the vessel wall. If the location of the repair is not positively and accurately known, then the individual shell plate, forging, or shell course containing the repair shall be included.
2. Includes essentially 100% of the weld length.
3. The shell-to-flange weld examination may be performed during the first and third periods in which case 50% of the weld-to-flange weld shall be examined by the end of the first period, and the remainder by the end of the third period. During the first period, the examination need only be performed from the flange face, provided this same portion is examined from the shell during the third period.
4. During the first and second periods, the examination may be performed from the flange face, provided these same portions are examined from the head during the third period.
5. Deferral in the first inspection interval is not permitted. Deferral in successive inspection intervals is permitted provided that the shell-to-flange weld area where repair activities have been performed either on the shell-to-flange weld or head-to-flange weld, and neither the shell-to-flange weld nor the head-to-flange weld contains identified flaws or relevant conditions that require corrective inspections in accordance with IWB-2420(B).
### Frequency of RPV Nozzle Examinations- Every 10 Year Interval

<table>
<thead>
<tr>
<th>Item No.</th>
<th>Parts Examined</th>
<th>Examination Requirements/ Figure No.</th>
<th>Examination Method</th>
<th>Acceptance Standard</th>
<th>Extent and Frequency of Examination</th>
<th>Deferral of Examination to End of Interval</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Reactor Vessel</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>B3.90</td>
<td>Nozzle-to-Vessel Welds</td>
<td>IWB-2500-7</td>
<td>Volumetric</td>
<td>IWB-3512</td>
<td>All nozzles [Note (1)] Same as for 1st Interval</td>
<td>See [Note (2)], [Note (3)], [Note (5)]</td>
</tr>
<tr>
<td>B31.00</td>
<td>Nozzle Inside Radius Section</td>
<td>IWB-2500-7</td>
<td>Volumetric</td>
<td>IWB-3512</td>
<td>All nozzles [Note (1)] Same as for 1st Interval</td>
<td>See [Note (2)], [Note (5)]</td>
</tr>
</tbody>
</table>
1. Includes all full penetration nozzle welds
2. At least 25% and no more than 50% of nozzles need to be examined within 3 1/3 years (Period)
3. If UT with straight beam from nozzle bore can delay UT from shell side to end of interval
4. Examination volumes are in IWB-2500-7
5. For PWRs after first interval nozzle UTs can be performed at end of interval if there were no repair replacements or flaws/relevant conditions
Examination Volumes

- UT entire thickness
- UT ½ thickness from each toe of the weld
Other Examination Volumes

UT entire thickness
UT ½ thickness from each toe of the weld

Longitudinal (Axial) Welds

Spherical Head Welds
Flange Welds

Head to Flange Weld

Shell to Flange Weld
Nozzle Weld Volumes
Welds and Nozzle Inside Corner Region

Figure NB-2500-7(a)
Nozzle in Shell or Head
(Examination Zones in Barrel-Type Nozzles Joined by Full-Penetration Corner Welds)

Figure NB-2500-7(b)
Nozzle in Shell or Head
(Examination Zones in Flange Type Nozzles Joined by Full-Penetration Butt Welds)
Nozzle Inside Corner Region UT
ASME Section XI Volumetric Examinations - IWA-2230

- IWA 2232: Ultrasonic Testing (UT)
  - Section XI, Appendix I, III, VII and VIII
    - Appendix I directs users to UT requirements based on component
    - Appendix III – piping and thin vessels
    - Appendix VII – UT personnel qualifications
    - Appendix VIII – UT system qualifications
  - ASME Section V, Article 4 (welds) and 5 (Components –ie Bolts) for components not covered by App I or App VIII
  - Non-Mandatory Appendix M – Modeling – Used for nozzle examinations
### Appendix VIII Component Applicability

#### Table VIII-3110-1

<table>
<thead>
<tr>
<th>Component Type</th>
<th>Component Qualification Supplements</th>
<th>Applicable Supplement</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Piping Welds</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Wrought austenitic</td>
<td></td>
<td>2</td>
</tr>
<tr>
<td>Ferritic</td>
<td></td>
<td>3</td>
</tr>
<tr>
<td>Cast austenitic [Note (1)]</td>
<td>In the course of preparation</td>
<td></td>
</tr>
<tr>
<td>Structural weld inlay (corrosion- resistant clad) austenitic[Note (1)]</td>
<td>In the course of preparation</td>
<td></td>
</tr>
<tr>
<td>Dissimilar metal</td>
<td></td>
<td>10</td>
</tr>
<tr>
<td>Overlay</td>
<td></td>
<td>11</td>
</tr>
<tr>
<td>Coordinated implementation</td>
<td></td>
<td>12</td>
</tr>
<tr>
<td><strong>Vessels</strong></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Clad/base metal interface region</td>
<td></td>
<td>4</td>
</tr>
<tr>
<td>Nozzle examinations from the outside surface</td>
<td></td>
<td>5</td>
</tr>
<tr>
<td>Reactor vessel welds other than clad/base metal interface</td>
<td></td>
<td>6</td>
</tr>
<tr>
<td>Nozzle examinations from the inside surface</td>
<td></td>
<td>7</td>
</tr>
<tr>
<td><strong>Bolts and Studs</strong></td>
<td></td>
<td>8</td>
</tr>
</tbody>
</table>
### Inservice Examination Acceptance Standards

<table>
<thead>
<tr>
<th>Examination Category</th>
<th>Component and Part Examined</th>
<th>Acceptance Standard</th>
</tr>
</thead>
<tbody>
<tr>
<td>B-A, B-B</td>
<td>Vessel welds</td>
<td>IWB-3510</td>
</tr>
<tr>
<td>B-D</td>
<td>Full penetration welded nozzles in vessels</td>
<td>IWB-3512</td>
</tr>
<tr>
<td>B-F, B-J</td>
<td>Dissimilar and similar metal welds in piping and vessel nozzles</td>
<td>IWB-3514</td>
</tr>
<tr>
<td>B-G-1</td>
<td>Bolting greater than 2 in. (50 mm) in diameter</td>
<td>IWB-3515</td>
</tr>
<tr>
<td>B-G-2</td>
<td>Bolting 2 in. (50 mm) in diameter and less</td>
<td>IWB-3517</td>
</tr>
<tr>
<td>B-K</td>
<td>Welded attachments for vessels, piping, pumps, and valves</td>
<td>IWB-3516</td>
</tr>
<tr>
<td>B-L-1, B-M-1</td>
<td>Welds in pumps and valves</td>
<td>IWB-3518</td>
</tr>
<tr>
<td>B-L-2, B-M-2</td>
<td>Pump casings and valve bodies</td>
<td>IWB-3519</td>
</tr>
<tr>
<td>B-N-1, B-N-2, B-N-3</td>
<td>Interior surfaces and internal components of reactor vessels</td>
<td>IWB-3520</td>
</tr>
<tr>
<td>B-O</td>
<td>Control rod drive and instrument nozzle housing welds</td>
<td>IWB-3523</td>
</tr>
<tr>
<td>B-P</td>
<td>Pressure retaining boundary</td>
<td>IWB-3522</td>
</tr>
<tr>
<td>B-Q</td>
<td>Steam generator tubing</td>
<td>IWB-3521</td>
</tr>
</tbody>
</table>
Examining a PWR Vessel with UT

- Scanners designed to examine shell welds during 10 Yr ISI
- Scanners designed to UT the Hot Leg Nozzles

- TWS in OL3.wmv
Sizing of Laminar Indications in Reactor Pressure Vessels – Modeling & Simulation

- CEA’s CIVA software was utilized to simulate the ultrasonic responses from laminar indications to predict sizing results.
Sizing of Laminar Indications in Reactor Pressure Vessels – Mockups (Cluster #3) – 50 mm Focal Distance Top, Side, & End Views – Preliminary Data with 2.5 MHz Array

Can resolve all six 3 mm diameter holes in cluster at the same depth.
UT for Alternative PTS Rule 10CFR50.61a

- Need to determine flaw density at inner surface of PWRs
- Can use procedures qualified to App VIII Supp 4 & 6 to determine flaw density
- If flaw density is exceeded then owner can use other UT techniques to reduce NDE sizing uncertainties.

### Table 3—Allowable Number of Flaws in Plates and Forgings

<table>
<thead>
<tr>
<th>Through-wall extent, TWE [in.]</th>
<th>TWE(_{\text{MIN}})</th>
<th>TWE(_{\text{MAX}})</th>
<th>Maximum number of flaws per 1000 square-inches of inside surface area in the inspection volume that are greater than or equal to TWE(<em>{\text{MIN}}) and less than TWE(</em>{\text{MAX}}). This flaw density does not include underclad cracks in forgings.</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>0.075</td>
<td>0.375</td>
<td>No Limit</td>
</tr>
<tr>
<td>0.075</td>
<td>0.075</td>
<td>0.375</td>
<td>0.05</td>
</tr>
<tr>
<td>0.125</td>
<td>0.075</td>
<td>0.375</td>
<td>0.05</td>
</tr>
<tr>
<td>0.175</td>
<td>0.075</td>
<td>0.375</td>
<td>0.08</td>
</tr>
<tr>
<td>0.225</td>
<td>0.075</td>
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<td>0.275</td>
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<td>0.325</td>
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<tr>
<td>0.375</td>
<td>Infinite</td>
<td>0.375</td>
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</tr>
</tbody>
</table>
UT for PTS - Potential Issues for UT

- Smallest flaws to go into density ‘bins’ are smaller than the flaws in the Appendix VIII test sets
  - Smallest flaw in test set is .075 in.
- Small flaws tend to be oversized so small flaw may be measured so they go into a bin that allows many fewer flaws
  - For plates, a .125 in. flaw (3.15/1000 square ft) could be oversized to > .175 ( >85/1000 square ft)
- Ultrasonic physics limits the depth sizing accuracy to ~1-2 wavelengths which is ~ .064 to .128 in. for a 2MHz transducer
- Beaver Valley flaw densities were determined and they met the tables.
Non-Visual Examination of BMNs

- BMNs are visually examined (VT) in accordance with Code Case N-722
- Almost ½ of fleet with BMNs have been UT’d and Eddy Current Tested
  - Voluntary exam at MRP request
  - Alternate to bare metal visual exams to avoid dose
  - Tube is examined, not weld
- Considered cold location and weld will prevent ejection
- Examination issues
  - B&W units have an un-inspectable region
  - 2-loop BMNs are thick and difficult to UT
  - Welds have numerous fabrication flaws and cannot discriminate between them and PWSCC
  - Weld surface typically not smooth so very difficult to examine with eddy current
Together…Shaping the Future of Electricity
8 - Vessel Monitoring through Surveillance Programs

William Server
ATI Consulting

RPV Integrity Workshop,
2016 International LWR Materials Reliability Conference,
Chicago, USA
August 1, 2016
Vessel Monitoring through Surveillance Programs

Learning Objectives:
- Purpose of surveillance programs
- Requirements for surveillance programs
- Contents of surveillance capsules
- Mechanical testing of specimens
- Analysis of Charpy data
- Use of surveillance data
- Integrated surveillance programs
- Supplemental programs to generate high-fluence data
Surveillance Program Requirements
Roadmap: RPV Integrity Rules, Guidance & Standards

Low Upper Shelf Toughness
- 10CFR50 Appendix G
- Reg. Guide 1.99 Rev. 2 (for calculating embrittlement effects)
- Reg. Guide 1.161 (for Equivalent Margins Analysis (EMA))
- ASME Section XI Appendix K (for Equivalent Margins Analysis (EMA))

Pressure-Temperature Limits
- 10CFR50 Appendix G
- Reg. Guide 1.99 Rev. 2 (for calculating embrittlement effects)
- ASME Section XI Appendices G & E

Reactor Vessel Material Surveillance
- 10CFR50 Appendix H
- NUREG-1801 Rev. 2 XI M31 (re: Surveillance Requirements in License Renewal)

Pressurized Thermal Shock
- 10CFR50.61
- 10CFR50.61a
- Reg. Guide under development for implementation of 10CFR50.61a

ASTM E185 (-73, -79, -82) (Testing & reporting procedures should meet ASTM E185-82)
Monitoring of Vessel Irradiation Effects

- Variables that can influence the rate and degree of vessel embrittlement:
  - Number and energy level of neutrons impacting the vessel wall
  - Temperature of the vessel wall during irradiation (essentially, coolant inlet temperature during power operations)
  - Correlation of neutron impact rate to material damage

- Surveillance programs obtain data on these variables by placing capsules near the pressure vessel wall
Typical Placement of Capsules in PWRs (left) and BWRs (right)

Notes: This drawing is not to scale.
F = Fuel bundle locations.
(Locations shown only for the northeast quadrant.)
* = Control rod locations.

From EPRI TR-113891
Monitoring of Vessel Irradiation Effects

- RPV surveillance programs (1) assess actual material embrittlement levels and (2) provide data upon which a predictive trend correlation can be based
- Capsules contain material specimens, temperature monitors and dosimetry to measure the level of neutron bombardment
- Periodic withdrawal and testing of capsules allows for verification of analytical predictions of vessel embrittlement
- In some cases - depending on the quality of the data – surveillance data may be used to change the estimate/prediction of vessel embrittlement
Monitoring of Vessel Irradiation Effects

- In July 1973, 10 CFR Part 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” established the first legal requirements for a comprehensive surveillance program
  - Plants already licensed had generally installed irradiation test samples (usually Charpy V-notch specimens) of RPV material per ASTM E 185, “Surveillance Tests on Structural Materials in Nuclear Reactors”
Appendix H to 10 CFR 50

  - ASTM E 185-73, -79, or -82, depending on which was in effect on date the RPV was purchased

- Proposed capsule withdrawal schedule must be approved by NRC
  - NRC Administrative Letter AL 97-004

- Capsule test report must be submitted to NRC within one year of capsule withdrawal

- Appendix H also provides the requirements for integrated surveillance programs (e.g., B&W, BWRVIP)
NRC Administrative Letter AL 97-004

- Formerly available in ADAMS under ML031210296 but is no longer available?
- AL 97-004 was issued to inform all plants of the NRC decision promulgated in CLI-96-13 (specific to Perry plant)
  - Changes to reactor vessel surveillance specimen capsule withdrawal schedules that do not conform to the required ASTM standard referenced in Appendix H will be treated as license amendments requiring public notice and opportunity for a hearing
  - Changes to RVSP capsule withdrawal schedules that do conform to ASTM E 185 only need prior NRC approval to verify conformance with the ASTM (as required by Appendix H)
  - This is accomplished via letter
Monitoring of Vessel Irradiation Effects

- Specific requirements for surveillance program design have evolved over time and are very detailed
  - each vessel program is designed to “...the edition of ASTM E 185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased.”
  - Later editions may be used, but only those editions through 1982
- RPVs which have peak neutron fluence greater than $10^{17}$ n/cm$^2$ (E $>$ 1 MeV) at end of design life must have the beltline materials (base metal and weld metal) monitored
### Sampling Requirements

- 2 capsules – Archive stock for 2 capsules
- 6 capsules uncut – Material held in reserve
- EODL $\Phi \times 2$ – Maximum allowed neutron exposure
- EODL $\Phi \times 3.25$ – Maximum required neutron exposure
- 0 or 2 – # of standby capsules
- 5 minus # required
- $\Phi$ only – Basis of pull schedule
- Same as matl. accept tests – Location for specimen removal
- $\frac{1}{4}$ t for base, away from root for weld

### Irradiation Condition Requirements

- Eutectic alloys (melt wires) required
- Monitor coolant temperature, may use eutectic alloys
- E482
- E482 & E844
- E844 – Dosimetry

- Thermal monitors
- Thermal capsules (optional)
- Accelerated Capsules (optional)

- Lead factor for wall (required) capsules
- Monitor materials with $\Phi > 10^{17}$

### Material Requirements

- Include correlation monitors?
- Materials to monitor
  - Chemistry
  - Processing

- Fully representative of RPV thermal & stress relief history
- Limited base & weld, + HAZ
- Limited base & weld – x Regulatory limits
- For representational via E184
- Chemical analysis required

### Specimen Requirements

- for low USE, no # specified
- 8 for $T_s$, limiting matl. only
  - Toughness
  - $4(u, 2l)$
  - $4(u, 3l)$
  - $6(u, 3l)$
  - $8(l) \rightarrow 15(u, 8l) \rightarrow 15(u, 12l)$
  - Tensile
  - Charpy

### ASTM E185 Revision Dates

- 1961
- 1966
- 1970
- 1973
- 1979
- 1981
- 1994
- 1998
- 2002
- 2010
- 2015

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**ASTM E 185 Evolution**

Slide by Dr. Mark Kirk (NRC-RES), from presentation “The Bases for LWR Vessel Surveillance Programs, General Requirements, Part 2: ASTM E185 & E2215 History,” ASTM E10.02 Workshop on Nuclear Reactor Pressure Vessel Surveillance Programs, Chicago, June 29, 2016
Requirements for Materials in RVSP

- Actual material used in the construction of the beltline
- Include at least one heat of base metal, weld metal, and heat-affected-zone (HAZ) material
- Selection of materials based on materials predicted to be most limiting, with regard to setting P-T limits at end-of-license (e.g., those with highest projected ART); any material projected to be < 50 ft-lb (68 J) at 1/4T must also be included
- Fabrication history of surveillance specimens is to be fully representative of fabrication history of the vessel materials (e.g., thermal annealing, austenizing treatment, quenching and tempering, and PWHT)
Minimum Recommended Number of Surveillance Capsules and Their Withdrawal Schedule

<table>
<thead>
<tr>
<th>Predicted Transition Temperature Shift at Vessel Inside Surface</th>
<th>&gt;56°C</th>
<th>≤ 56°C</th>
<th>&gt;111°C</th>
<th>≤111°C</th>
</tr>
</thead>
<tbody>
<tr>
<td>(&gt;100°F)</td>
<td></td>
<td>(&gt;200°F)</td>
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<tr>
<td>(≤ 100°F)</td>
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</table>

<table>
<thead>
<tr>
<th>Minimum Number of Capsules</th>
<th>3</th>
<th>4</th>
<th>5</th>
</tr>
</thead>
<tbody>
<tr>
<td>Withdrawal Sequence:</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>First</td>
<td>6⁴</td>
<td>3⁴</td>
<td>1.5⁴</td>
</tr>
<tr>
<td>Second</td>
<td>15⁹</td>
<td>6⁶</td>
<td>3⁹</td>
</tr>
<tr>
<td>Third</td>
<td>EOL⁶</td>
<td>15⁹</td>
<td>6⁶</td>
</tr>
<tr>
<td>Fourth</td>
<td>EOL⁶</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fifth</td>
<td>EOL⁶</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

¹ Or at the time when the accumulated neutron fluence of the capsule exceeds $5 \times 10^{22}$ n/m² ($5 \times 10^{18}$ n/cm²), or at the time when the highest predicted $\Delta RT_{\text{MAX}}$ of all encapsulated materials is approximately 28°C (50°F), whichever comes first.

⁴ Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel inner wall location, whichever comes first.

⁹ Or at the time when the accumulated neutron fluence of the capsule corresponds to the approximate EOL fluence at the reactor vessel 1/4 T location, whichever comes first.

⁶ Or at the time when the accumulated neutron fluence of the capsule corresponds to a value midway between that of the first and third capsules.

EOL⁶ Not less than once or greater than twice the peak EOL vessel fluence. This may be modified on the basis of previous tests. This capsule may be held without testing following withdrawal.

From TR-101975
Test Specimen Requirements

- CVN and tension specimens
- Fracture toughness specimens included if surveillance materials are predicted to exhibit marginal properties
- Tension and CVN specimens for base metal & HAZ shall be from 1/4T locations; weld specimens may be from any thickness except near root or surface of weld
- Base metal tension & CVN specimens: major axis normal to principal rolling direction of the plate, normal to the major working direction for forgings
  - Charpy specimens should have the transverse (T-L) orientation (weak direction)
Specimen Position and Orientation Requirements
Specimen Orientation Issues

- Prior to 1973, the specification required Charpy specimens to be longitudinal (L-T), so older plants may not have T-L specimens
  - Because the change from L-T to T-L was anticipated, plants built within a few years of the transition often have both L-T and T-L
- NUREG-0800 Branch Technical Position 5-3 provides guidance for estimating T-L values from L-T data
  - For some product forms, the BTP 5-3 methods of estimating T-L $T_{50}$ (50 ft-lb [68 J] Charpy index temperature) from L-T data may be potentially nonconservative
  - This issue is discussed separately in this workshop
## Minimum Number of Test Specimens Required by ASTM E185-82

<table>
<thead>
<tr>
<th>Material</th>
<th>Quantities of Specimens</th>
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<tbody>
<tr>
<td></td>
<td>Unirradiated</td>
<td></td>
<td>Irradiated¹</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Charpy²</td>
<td>Tension³</td>
<td>Charpy</td>
<td>Tension</td>
</tr>
<tr>
<td>Base Metal</td>
<td>18</td>
<td>3</td>
<td>12</td>
<td>3</td>
</tr>
<tr>
<td>Weld Metal</td>
<td>18</td>
<td>3</td>
<td>12</td>
<td>3</td>
</tr>
<tr>
<td>Correlation Monitor</td>
<td>18</td>
<td>0</td>
<td>12</td>
<td>0</td>
</tr>
</tbody>
</table>

¹Number of test specimens per exposure set (capsule)
²A minimum of 15 shall be tested
³Later versions of ASTM E185 (e.g., ASTM E185-92) require 6, not 3, unirradiated tension specimens
Capsule Dosimetry

- Dosimeters are placed in the surveillance capsules when the capsules are fabricated during reactor construction.
  - Because each dosimeter reacts to neutrons of a particular energy in the spectrum, a set of dosimeters are used in each capsule to provide adequate spectrum coverage.
  - Most are thin circular activation foils, although other shapes are available.
  - In addition to fast-neutron threshold monitors, a thermal monitor such as $^{59}\text{Co}$ is typically included to determine thermal neutron fluence.
Temperature Monitors

- Mechanical and impact properties of irradiated specimens depend on the temperature at which the material is irradiated.
- ASTM E 185 requires temperature monitors in capsules.
- Low-melting alloys or pure metals are inserted inside of the capsules to monitor peak temperature.
  - Including different detectors of varying composition of the alloys allows for a range of temperatures to be monitored.
  - Provide indication of the maximum temperature experienced by the capsule specimens (and RPV wall).
- Recommended to have time-averaged $T_{\text{cold}}$ over the time of irradiation (via analysis).
ASTM E 185-82 Reporting Requirements

- Administrative: Both conventional and SI units reported
- Surveillance Program Description
- Surveillance Material Selection & Characterization
- Test results
  - Tension Tests
  - Charpy Tests
  - Hardness tests (optional)
  - Other Fracture Toughness Tests (if performed)
  - Temperature & neutron radiation environment
- Application of Test Results (compare with predicted)
- Deviations
Charpy Test Results Reporting Requirements

- Description of testing machine
- Test data from each specimen
  - Test temp., specimen ID, CVE, fracture appearance, lateral expansion (LE)
  - Test data for each specimen
    - Index temperatures for Charpy 41J (30 ft-lb), 68J (50 ft-lb), 0.89mm (35-mil) LE for irradiated & unirradiated specimens
      - $T_{30}$, $T_{50}$, $T_{35}$mil
    - Test data for each material
      - Upper shelf energy (USE) absorbed before and after irradiation
      - Initial & adjusted reference temperatures
Surveillance Program Requirements for License Renewal Period (60 years)

- NUREG-1801, Rev 2 stipulates requirements for an acceptable reactor vessel surveillance program during the license renewal period.

  "The plant-specific or integrated surveillance program shall have at least one capsule with a projected neutron fluence equal to or exceeding the 60-year peak reactor vessel wall neutron fluence prior to the end of the period of extended operation.

- The program withdraws one capsule at an outage in which the capsule receives a neutron fluence of between one and two times the peak reactor vessel wall neutron fluence at the end of the period of extended operation [emphasis added] and tests the capsule in accordance with the requirements of ASTM E 185-82.”
Highlighted GALL Rev 2 RVSP Requirements

- “Any changes to the capsule withdrawal schedule, including spare capsules [emphasis added], must be approved by the Nuclear Regulatory Commission (NRC) prior to implementation.

- Untested capsules placed in storage must be maintained for possible future insertion.

- Pulled and tested samples…. are placed in storage (these specimens and capsules are saved for future reconstitution and reinsertion use) unless the applicant has gained NRC approval to discard the pulled and tested samples or capsules.”
Analysis of Charpy Test Data
Charpy Data Curve Fitting Technical Description

- ASTM E 185-82 requires that irradiation effects be determined from Charpy data by measuring the differences in the 30 ft-lb (41 J) energy, 50 ft-lb (68 J) energy, and 35 mil (0.89 mm) lateral expansion index temperatures before and after irradiation.

- “The index temperatures shall be obtained from the average curves.”
  - The ASTM guide does not specify how the average curves are to be determined.
Charpy Data Curve Fitting Technical Description

- The general shape of Charpy test data (energy versus temperature, or lateral expansion versus temperature) is that of an "S"
- The hyperbolic tangent (tanh) function is used as a simple statistical curve-fit tool to describe the "S"-shaped response
  - Tanh curve fitting of Charpy V-notch data is a standard practice within industry
Use of Surveillance Data
Use of Surveillance Data

- Specific guidance for use of surveillance data depends on the application

<table>
<thead>
<tr>
<th>Application/Use</th>
<th>Guidance</th>
</tr>
</thead>
<tbody>
<tr>
<td>PTS</td>
<td>10CFR50.61</td>
</tr>
<tr>
<td>Alternate PTS</td>
<td>10CFR50.61a</td>
</tr>
</tbody>
</table>

- Detailed discussion of Reg. Guide 1.99 R2 guidance – see backup slides
- Note:
  - Reg Guide 1.99 Rev 2 and 10CFR50.61 guidance for treatment of surveillance data are nearly identical and provide for direct use of surveillance data
  - 10CFR50.61a says to use the ETC in preference to surveillance data, except when the data significantly and nonconservatively deviate from the ETC.
Integrated Surveillance Programs
Integrated Surveillance Programs

- In an ISP, the representative materials are irradiated in one or more reactors that have similar design and operating features.

- Requirements for ISPs are given in 10CFR50 Appendix H:
  - similarity of irradiation conditions; adequate dosimetry programs in each reactor; adequate data sharing between plants; contingency plan for outages in host reactors

- Examples of integrated surveillance programs:
  - B&W Master Integrated Reactor Vessel Surveillance Program (MIRVP) (BAW-1543);
  - BWRVIP Integrated Surveillance Program (BWRVIP ISP) (BWRVIP-86-A)
    - All U.S. BWRs participate in BWRVIP ISP
Coordinated and Supplemental Surveillance Programs
Background: RPV Surveillance/ETC Issues

- Irradiated surveillance data are used in development of embrittlement trend curves (ETCs) to predict RPV irradiation embrittlement (transition temperature shifts)

- Current ETCs are based on surveillance database data predominately < ~3x10^{19} n/cm^2
  - Regulatory Guide 1.99, Rev.2 (177 data points, 1988)
    - Used for Appendix G P-T curves; same ETC for PTS Rule 10CFR50.61
  - “EONY” ETC (775 data points, 2006; last data ~2003)
    - Used for 10 CFR 50.61a, Alternate PTS Rule only
MRP Programs to Address Paucity of High-Fluence PWR Surveillance Data

- High-fluence surveillance data is needed to inform development of an embrittlement trend correlation applicable for RPV operation to high fluences

- MRP has developed 2 programs to meet this need:
  1. Coordinated Reactor Vessel Surveillance Program, CRVSP: Withdraw/Test current licensing basis capsules at a higher fluence (see MRP-326)
     - ~13 plants were selected to defer withdrawal of next scheduled capsule so that higher fluence would be attained
     - CRVSP was fully implemented in 2012
  2. PWR Supplemental Surveillance Program, PSSP: Design, fabricate, irradiate and test 2 supplemental capsules (MRP-412 – publication pending)
PWR Supplemental Surveillance Program (PSSP)

- Design/Fabricate/Irradiate 2 supplemental surveillance capsules containing previously-irradiated PWR materials
  - Selected weld and base metal surveillance materials to fill data gaps for development of future embrittlement trend curves
  - **Reconstitute** previously-irradiated specimens (per ASTM E1253) (see figure)
  - Obtain ~24 new high-fluence transition temperature shift (TTS) data points
    - Each capsule will contain ~12 materials
- Insert capsule(s) in 2016 and 2018
- Irradiate ~10 years
  - Withdraw capsules & obtain data by ~2026/2028
Recent Surveillance Capsule Operating Experience

- In 2013, a Westinghouse-design PWR relocated 2 surveillance capsules from existing locations to available (empty) capsule holder positions with higher lead factors in order to increase fluence for license renewal.

- During next outage, loose parts were found on fuel assemblies as they were off-loaded, and on bottom of core plate.

- Cause: Both of the re-located capsules had failed and spread debris, including capsule contents.

- Root cause analysis: Installation errors and installation procedural inadequacies (ML15215A656)

- NRC INFORMATION NOTICE 2016-02: Improper Seating of Reactor Vessel Surveillance Capsules (January 15, 2016) (ML15278A472)
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Backup Slides –
Use of Surveillance Data per Reg. Guide 1.99 Rev. 2
Regulatory Guide 1.99, Rev. 2 Guidance for Assessing Embrittlement of RPV Materials

- Provides guidance for
  - Prediction of Adjusted reference temperature (Adjusted $RT_{NDT}$)
  - Prediction of percent decrease in USE

- Details two methods for determination of assessing Adjusted $RT_{NDT}$ and $\Delta USE$
  - Surveillance Data Not Available (No data or only 1 irradiated data point)
    - Regulatory Position 1.1 (for ART)
    - Regulatory Position 1.2 (for USE decrease)
    - Limitations given by Regulatory Position 1.3
  - Surveillance Data Available (2 or more data points)
    - Regulatory Position 2.1 (for ART)
    - Regulatory Position 2.2 (for USE decrease)
Adjusted Reference Temperature (P-T Curves)

- ART = Initial RT\textsubscript{NDT} + ΔRT\textsubscript{NDT} + Margin

ΔRT\textsubscript{NDT} = (CF) f^{(0.28 – 0.10 \log f)}

- CF (°F) is the chemistry factor, given in tables in Reg. Guide 1.99 Rev. 2, as a function of material chemistry, or CF may be derived from plant surveillance data if certain criteria are met
- f^{(0.28 – 0.10 \log f)} is the fluence factor (FF)
  - f which is the neutron fluence at any depth in the vessel wall (in units of 10^{19} n/cm^2, E > 1 MeV), determined as
    - f = f\textsubscript{surf} (e^{-0.24x})
Adjusted Reference Temperature

- ART = Initial RT<sub>NDT</sub> + ΔRT<sub>NDT</sub> + Margin

\[
\text{Margin} = 2 \sqrt{\sigma_I^2 + \sigma_\Delta^2}
\]

- \(\sigma_I\) is the standard deviation for the initial RT<sub>NDT</sub>. If a measured value of initial RT<sub>NDT</sub> for the material in question is available, \(\sigma_I\) is to be estimated from the precision of the test method (and it is normally taken to be 0°F [0°C]). If not, and if generic mean values for the class of material are used, \(\sigma_I\) is the standard deviation obtained from the set of data used to establish the mean.

- The standard deviation for ΔRT<sub>NDT</sub>, \(\sigma_\Delta\), is 28°F (16°C) for welds and 17°F (9°C) for base metal, except that \(\sigma_\Delta\) need not exceed 0.50 times the mean value of ΔRT<sub>NDT</sub>.

- If credible surveillance data are available, \(\sigma_\Delta\) may be cut in half.
Surveillance Data Not Available (No data or only 1 data point)

- **Regulatory Position 1.1 (for ART)**
  - Chemistry Factor (CF) determined from Table 1 (weld) or Table 2 (plate)

- **Regulatory Position 1.2 (for USE)**
  - USE is assumed to decrease as a function of fluence and Cu content as depicted in Figure 2

- **Limitations given by Regulatory Position 1.3**
  - Common RPV materials w/ min yield strength 50ksi
  - Irradiation temperature lower than 525°F (274°C) or higher than 590°F (310°C) should be corrected
  - Justify any Cu/Ni or fluence beyond those covered by the figures & tables
Reg Guide 1.99, Rev. 2: Chemistry Factor Tables (Table 1 – Welds, Table 2- Base Metal)

**TABLE 1**

CHEMISTRY FACTOR FOR WELDS, °F

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<thead>
<tr>
<th>Copper, Wt-%</th>
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</table>
Reg. Guide 1.99, Rev. 2 Figure 2 – Predicted USE Decrease (as function of Cu & fluence)
Detailed Discussion of Regulatory Position 2.1 (Use of Surveillance Data for Calculation of Vessel ART)
Surveillance Data Available (2 or more data points)

- Regulatory Position 2.1 (for ART)
  - Calculate the fluence function (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 the new CF which reflects a minimization of the sum of the squares of the errors

\[
CF = \sum \frac{\Delta T_{30} \times FF}{FF^2}
\]
Surveillance Data Available (2 or more data points)

- Regulatory Position 2.2 (for USE)
  - Decrease in USE obtained by plotting the reduced plant data with a line drawn parallel to existing lines as the upper bound of all data; use the line in preference to the existing graph
  - To be discussed in detail in USE section
Reg Guide 1.99, Rev. 2: Surveillance Data Available (2 or more data points)

- ART is calculated for a vessel material per Position 2.1 (using surveillance data) only for the vessel heat that matches the surveillance material heat.
- For all other vessel heats for which no matching-heat surveillance data are available, ART is calculated per Position 1.1.
- Before surveillance data results are extrapolated to predict embrittlement behavior of the vessel, it must be adjusted for any differences that would make the vessel material behavior different from the surveillance material behavior:
  - Chemistry (weld only)
  - Irradiation temperature (weld and plate)
Adjust Surveillance Data for Chemistry (Weld only)

- If there is clear evidence that the copper or nickel content of the surveillance weld differs from that of (best estimate of) the vessel weld, the measured values of $\Delta R_{\text{NDT}}$ should be adjusted by multiplying the ratio of the chemistry factor for the vessel weld to that for the surveillance weld:

$$\text{RatioAdjusted}_{\Delta R_{\text{NDT}}} = \left( \frac{\text{TableCF}_{\text{Vessel Chem.}}}{\text{TableCF}_{\text{Surv Chem.}}} \right) \times \text{Measured}_{\Delta R_{\text{NDT}}}$$
Adjust Surveillance Data for Irradiation Temperature (Weld and Plate)

- If the irradiation temperature of the surveillance material was different from the irradiation temperature of the vessel material to which the data is to be applied, adjustment is made to the surveillance data:
  - NRC: “Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in \( \Delta R_{NT} \)”
Calculate the Chemistry Factor (CF) for the Best Fit of the Data

- After any required adjustments (chemistry, irradiation temperature) are made to the shift data, a fitted CF for the vessel material can be calculated.

- According to methods of Position 2.1 of RG 1.99, Revision 2, a CF based on the surveillance data is determined as follows:

\[
CF = \frac{\sum (FF \times \Delta RT_{NDT-Adj})}{\sum (FF^2)}
\]
Surveillance Data Credibility Requirements
(Regulatory Guide 1.99, Rev. 2)

- Use of surveillance data in lieu of RG1.99R2 requires considerable engineering judgment to evaluate the credibility and assign suitable margins.
- When surveillance data become available (e.g., 2 or more data points), the weight given them depends on the credibility of the data.
- If data credible, the values for $\sigma_{\Delta}$ (standard deviation for $\Delta RT_{NDT}$) may be cut in half (in effect, reduce the margin term by half if $\sigma_i$ is 0).
Surveillance Data Credibility Requirements
(Regulatory Guide 1.99, Rev. 2)

1) Materials in the capsules should be judged to be most limiting for the vessel with regard to radiation embrittlement

2) Scatter in the Charpy energy data should be small enough to determine $T_{30}$ and USE unambiguously

3) Scatter of the measured shift vs. fluence about a best fit line should be within 1-sigma

4) The irradiation temperature for the specimens should match the vessel wall temperature within $\pm 25^\circ$F

5) The surveillance results for the correlation monitor material should fall within the scatter band for that specific material
Surveillance Data Credibility Guidance

- Guidance regarding credibility further addressed in 1990’s
Surveillance Data Credibility Guidance

- NRC staff provided a set of examples of surveillance data application examples/situations and methods acceptable to the NRC staff
  - If credible and fitted CF is lower than table CF, may use fitted CF for ART calculation, and margin may be reduced
  - If credible and fitted CF is higher than table CF, must use fitted CF, and margin may be reduced
  - If not credible, use the higher of fitted CF or table CF, and must use full margin
9 – Upper Shelf Energy

Tim Hardin
Technical Executive

RPV Integrity Workshop,
2016 International LWR Materials Reliability
Conference, Chicago, USA
August 1, 2016
Learning Objectives

- Understand definition of Upper Shelf Toughness / Energy (USE)
- Be able to identify and understand the application of Regulations, Regulatory Guide(s), and Code(s) pertaining to USE
- Be able to state the 10CFR50, Appendix G criteria limits for USE in both the un-irradiated and irradiated conditions
- Understand Charpy specimen orientation requirements for determining USE
- Be capable of identifying the type of analyses required when USE fails to meet the regulation criteria
Definition of Upper Shelf Energy (USE)

- RPV steel Charpy V-notch energy increases with increasing temperature until a plateau is reached where it remains relatively high and constant with further increases in temperature – a level known as Upper Shelf Energy.

- Neutron irradiation of beltline materials causes USE to drop. Decrease in USE with irradiation is a function of product form, chemistry content, and fluence.
Definition of Upper Shelf Energy (continued)

- Definition of USE from ASTM E185-82:
  
  "upper shelf energy level - the average energy value for all Charpy specimens (normally three) whose test temperature is above the upper end of the transition region. For specimens tested in sets of three at each test temperature, the set having the highest average may be regarded as defining the upper shelf energy."

- Conventional practice: USE is average of all Charpy data exhibiting ≥ 95% shear

- Charpy USE definition clarified in ASTM E185 (2010 version):
  
  - The average energy value for all Charpy specimen tests (preferably three or more) whose test temperature is at or above the Charpy upper-shelf onset;
    - Specimens tested at >150°F (83°C) above the Charpy upper-shelf onset shall not be included, unless no data are available between the onset temperature and onset plus 150°F (83°C)
    - Charpy upper-shelf onset is the test temperature above which the fracture appearance of all Charpy specimens tested is ≥ 95% shear
Measuring and Reporting Upper Shelf Energy

- Requirements for minimum Charpy USE pertain to test results of CVN specimens oriented in the weak (transverse, T-L) direction
  - Longitudinal axis of the CVN specimen is transverse (perpendicular) to the materials’ primary rolling/working direction

- If only longitudinal (L-T) specimen test data are available, as the case for some older plants, Branch Technical Position 5-3 provides that transverse (T-L) properties can be estimated as 65% of the longitudinal (L-T) values:

  \[
  \text{Charpy USE}_{(T-L)} = 0.65 \times \text{Charpy USE}_{(L-T)}
  \]

* (Note: This relationship may not be used in reverse to estimate strong direction (L-T) properties from weak (T-L) direction data.)
# Reactor Vessel Integrity Rules

## Low Upper Shelf Toughness
- **10CFR50 Appendix G**
  - Reg. Guide 1.99 Rev. 2 (for calculating embrittlement effects)
  - Reg. Guide 1.161 (for Equivalent Margins Analysis [EMA])
- **ASME Section XI Appendix K (for Equivalent Margins Analysis [EMA])**

## Pressure-Temperature Limits
- **10CFR50 Appendix G**
  - Reg. Guide 1.99 Rev. 2 (for calculating embrittlement effects)

## Reactor Vessel Material Surveillance
- **10CFR50 Appendix H**
  - NUREG-1801 Rev. 2 XI M31 (re: Surveillance Requirements in License Renewal)
- **ASTM E185 (-73, -79, -82)** (Testing & reporting procedures should meet ASTM E185-82)

## Pressurized Thermal Shock
- **10CFR50.61**
- **10CFR50.61a**
  - Reg. Guide under development for implementation of 10CFR50.61a
10CFR50 Appendix G, “Fracture Toughness Requirements”

- Minimum USE requirements were established to address concerns for susceptibilities of quenched & tempered steels to ductile tear fractures of low energy absorption (“low energy shear fracture”)

- Pellini (developer of Pellini Drop Weight Test) was a major proponent of concern for failure on upper shelf
  - Pellini studied low-energy shear fracture in high strength quench & tempered steels
  - There had been a service failure of an experimental high-strength steel pressure vessel via ductile fracture that focused attention on the phenomenon
  - After irradiation, low alloy RPV steels are not unlike the higher-strength steels

- As early as 1967, the Atomic Energy Commission (AEC) began considering safety requirements for the upper shelf temperature region (existing requirements focused on brittle failure on the lower shelf/transition region)
First formal USE requirement was issued by AEC in 1971: minimum initial (unirradiated) USE of 68 J (50 ft-lb) for vessels with walls > 127 mm (5 in.) thick.

The basis for 50 ft-lb (68 J) is not well documented, but some background information is provided in NUREG/CR-5552 (ORNL/TM-11314), “An Overview of the Low-Upper-Shelf Toughness Safety Margin Issue” (John Merkle)

- “...a draft AEC technical document [gave] the bases for the 68 J (50 ft-lb) criterion as the NRL ratio analysis diagram [see WRC Bulletin 130], a Charpy USE vs. toughness correlation developed by Rolfe and Novak, and a fracture mechanics leak-before-break calculation for a 254 mm-thick (10 in.) reactor vessel wall.”

It is also noted that, until ~1980, shift in RT\textsubscript{NDT} was measured at the 68 J (50 ft-lb) energy level; if USE were less than 68 J, shift due to neutron irradiation could not be measured, so this may have also been a practical consideration for requiring at least 68 J.
Current Appendix G Requirements

- 10 CFR 50, Appendix G, Para. IV.A.1, currently specifies that vessel materials:
  - Must have Charpy USE, measured in the weak (transverse) direction, for base material and along the weld of **no less than 75 ft-lb (102 J) initially** (i.e., in the un-irradiated condition)
  - Must maintain Charpy USE of **no less than 50 ft-lb (68 J) throughout vessel life**

- Surveillance data show that neutron irradiation can produce USE < 50 ft-lb for radiation sensitive materials (e.g., A302B plate materials and Linde 80 welds, etc.)

- For materials falling below the 10CFR50 Appendix G limits, an Equivalent Margin Analysis (EMA) must be performed demonstrating margins of safety against fracture equivalent to those required by **Appendix G of ASME Code Section XI**

- EMA must be submitted to NRC, for review and approval, no less than three years before the material is predicted to fall below 50 ft-lb
How to Predict End-of-Life USE?


- Figure 2 curves define the predicted % USE decrease vs. fluence, as a function of Cu content, material (base metal or weld), and 1/4T fluence.

- Allows for application of credible surveillance test data to establish additional “material-specific” curves on Figure 2 for estimating % USE decrease.
Use of Reg. Guide 1.99 Rev. 2 for Predicting USE Decrease (1/2)

- Projecting USE Drop for Vessel Materials not within the Surveillance Program (Regulatory Position 1.2)
  
  - The predicted drop in USE is taken directly from Figure 2 of Reg. Guide 1.99 Rev. 2
  
  - Enter the appropriate curve (base metal or weld) with the Cu content of the material and the projected EOL 1/4T fluence (from x-axis)
    
    - Interpolation is acceptable
  
  - Read the predicted USE drop obtained from the y-axis and apply it to the material’s unirradiated USE to predict the EOL USE
Use of Reg. Guide 1.99 Rev. 2 for Predicting USE Decrease (2/2)

- For materials with credible surveillance data, use Regulatory Position 2.2:
  - The measured % USE drops are plotted (see example), and a new curve is drawn parallel to existing curves, from which % USE drop at end of life (EOL) fluence can be predicted.
- For materials with non-credible surveillance data, Regulatory Position 1.2 should be used.
Equation Form of Reg. Guide 1.99 R2 Figure 2 Curves

As alternative to use of the figure, equations have been developed for Reg. Guide 1.99 Rev 2, Figure 2; available in Reg. Guide 1.162 (Thermal Annealing) and NUREG/CR-5799 (Review of Yankee Rowe Reactor Vessel Evaluation)

For base metal, predicted drop in USE is
$$\Delta CVN(\%) = (100 \text{ Cu} + 9) f^{0.2368}$$
where $f$ is fluence (in terms of $10^{19}$ n/cm$^2$)

For weld metal, predicted drop in USE is
$$\Delta CVN(\%) = (100 \text{ Cu} + 14) f^{0.2368}$$
ASME Code Section XI, Appendix K

- 10CFR50 Appendix G requires an Equivalent Margin Analysis (EMA) for materials with USE falling below 50 ft-lb (68 J) in order to demonstrate fracture toughness margins are essentially equivalent to those when the USE ≥ 50 ft-lb
- In 1982, NRC requested ASME Section XI to develop criteria for these analyses
- ~1993, Appendix K, “Assessment of Reactor Vessels with Low Upper Shelf Charpy Impact Energy Levels” was issued
  - Provides procedures for developing EMA using actual geometry
  - Technical bases for the analysis methods are in WRC Bulletin 413
- EMA approaches use Elastic Plastic Fracture Mechanics (EPFM) for evaluating resistance to ductile fracture with postulated flaws of various sizes in a limiting material, considering flaw orientation
  - Evaluates margins on fracture initiation and ductile crack extension
Regulatory Guide 1.161

- NRC had deemed ASME XI Appendix K EMA criteria to be generally acceptable, but noted that Appendix K is missing guidance for:
  - Determining event sequences and transients to consider
  - Appropriate material properties to be assumed for the EMA

  - Provides specific guidance regarding the selection of material properties and transients to be considered within the EMA
Industry Status on Low USE and EMAs

- NSSS vendors (GE, W, CE, B&W) submitted generic bounding USE EMAs in response to NRC Generic Letter 92-01

- USE values well below 50 ft-lb have been justified by EMA for all currently operating RPVs
  - In a Safety Evaluation approving as low as 29 ft-lb in one application, NRC gave notice that no future applications below 30 ft-lb would be approved

- Based on vendor EMAs and its own generic study, NRC found that all RPVs have adequate upper shelf toughness throughout the initial license period

- EMAs have also been updated to address the license renewal period
  - Industry is beginning work to address needs for Second License Renewal (SLR)

- As plants test capsules and additional irradiated USE data become available, plants must demonstrate that materials continue to be bounded by the generic analyses; otherwise, a plant-specific analysis is required
Review Learning Objectives

• Understand definition of Upper Shelf Toughness / Energy (USE)

• Be able to identify and understand the application of Regulations, Regulatory Guide(s), and Code(s) pertaining to USE

• Be able to state the 10CFR50, Appendix G criteria limits for USE in both the un-irradiated and irradiated conditions

• Understand Charpy specimen orientation requirements for determining USE

• Be capable of identifying the type of analyses required when USE fails to meet the regulation criteria
Together…Shaping the Future of Electricity
10 – Pressure-Temperature Limit Curves

Tim Hardin
Technical Executive
RPV Integrity Workshop,
2016 International LWR Materials Reliability Conference, Chicago, USA
August 1, 2016
Pressure-Temperature (P-T) Limit Curves

Learning Objectives
- The purpose of P-T limits
- The regulations pertaining to P-T limits
- The analytical approach
- The composite nature of P-T limit curves and the basis for each component
- The purpose and regulations for LTOP
- Guidance for evaluating violations of P-T limits
Regulations for P-T Limits

- The purpose of P-T limits is to protect against nonductile failure of the RPV during normal reactor heatup and cooldown, and leak tests / hydro tests
- 10 CFR 50 Appendix G governs and establishes 2 sets of requirements:
  - Requirements for pressure-temperature limits for heatup, cooldown, and pressure test conditions
  - Minimum temperature requirements for specific vessel components during various operating conditions
Roadmap: RPV Integrity Rules, Guidance & Standards

Low Upper Shelf Toughness
- 10CFR50 Appendix G
- Reg. Guide 1.99 Rev. 2 (for calculating embrittlement effects)
- Reg. Guide 1.161 (for Equivalent Margins Analysis [EMA])
- ASME Section XI Appendix K (for Equivalent Margins Analysis [EMA])

Pressure-Temperature Limits
- 10CFR50 Appendix G
- Reg. Guide 1.99 Rev. 2 (for calculating embrittlement effects)
- ASME Section XI Appendices G & E

Reactor Vessel Material Surveillance
- 10CFR50 Appendix H
- NUREG-1801 Rev. 2 XI M3.11 (re: Surveillance Requirements in License Renewal)
- ASTM E185 (-73, -79, -82) (Testing & reporting procedures should meet ASTM E185-82)

Pressurized Thermal Shock
- 10CFR50.61
- 10CFR50.61a
- Reg. Guide under development for implementation of 10CFR50.61a
Plant Operating Window

- 10CFR50 Appendix G P-T limits are just one of many limits that constrain RCS pressure and temperature during plant operations
  - Reactor coolant pump net positive suction head (NPSH) – establishes a minimum pressure for operation of reactor coolant pumps
  - Minimum margin to subcooling
  - Low Temperature Overpressure (LTOP) protection system (PWRs only) (discussed later)
  - Steam generator D/P limit (some plants)
  - Pressurizer spray ΔT limit
  - Flange minimum temp. limit (more later)
  - RHR system relief setpoint
- Taken together, these limits define an “operating window” for the RCS
### 10 CFR 50 Appendix G P-T and Minimum Temperature Requirements

<table>
<thead>
<tr>
<th>Operating condition</th>
<th>Vessel pressure(^1)</th>
<th>Requirements for pressure-temperature limits</th>
<th>Minimum temperature requirements</th>
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<tbody>
<tr>
<td>1. Hydrostatic pressure and leak tests (core is not critical):</td>
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</tr>
<tr>
<td>1.a Fuel in the vessel</td>
<td>&lt;20%</td>
<td>ASME Appendix G Limits</td>
<td>((^2))</td>
</tr>
<tr>
<td>1.b Fuel in the vessel</td>
<td>&gt;20%</td>
<td>ASME Appendix G Limits</td>
<td>((^2)) +90 ° F((^6))</td>
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<tr>
<td>1.c No fuel in the vessel (Preservice Hydrotest Only)</td>
<td>ALL</td>
<td>(Not Applicable)</td>
<td>((^3)) +60 ° F</td>
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<tr>
<td>2. Normal operation (incl. heat-up and cool-down), including anticipated operational occurrences:</td>
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<td></td>
<td></td>
</tr>
<tr>
<td>2.a Core not critical</td>
<td>&lt;20%</td>
<td>ASME Appendix G Limits</td>
<td>((^2))</td>
</tr>
<tr>
<td>2.b Core not critical</td>
<td>&gt;20%</td>
<td>ASME Appendix G Limits</td>
<td>((^2)) + 120 ° F.</td>
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<tr>
<td>2.c Core critical</td>
<td>&lt;20%</td>
<td>ASME Appendix G Limits + 40 ° F.</td>
<td>Larger of ([(^4)]) or ([(^2)] + 40° F.]</td>
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<tr>
<td>2.d Core critical</td>
<td>&gt;20%</td>
<td>ASME Appendix G Limits + 40 ° F.</td>
<td>Larger of ([(^4)]) or ([(^2)]+160°F]</td>
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<tr>
<td>2.e Core critical for BWR ((^5))</td>
<td>&lt;20%</td>
<td>ASME Appendix G Limits + 40 ° F.</td>
<td>((^2))+60°F</td>
</tr>
</tbody>
</table>

1 Percent of the preservice system hydrostatic test pressure.
2 The highest reference temperature of the material in the closure flange region that is highly stressed by the bolt preload.
3 The highest reference temperature of the vessel.
4 The minimum permissible temperature for the inservice system hydrostatic pressure test.
5 For boiling water reactors (BWR) with water level within the normal range for power operation.
6 Lower temperatures are permissible if they can be justified by showing that the margins of safety of the controlling region are equivalent to those required for the beltline when it is controlling.
Pressure-Temperature Limit Curves
10 CFR 50 Appendix G Requirements for P-T Limits

- “ASME Appendix G Limits” are minimum requirements
  - Plants may determine P-T limits using any method, but the limits must be at least as conservative as those that would be generated by using the methods of ASME XI Appendix G
  - 10CFR50 Appendix G makes nonmandatory ASME XI Appendix G mandatory

- P-T limits must account for the effects of neutron irradiation on RPV materials which receive fluence $\geq 1 \times 10^{17}$ n/cm$^2$ ($E > 1$ MeV)
  - P-T limits are based, in part on the reference temperature, RT$_{NDT}$, of the vessel limiting material
  - NRC requires that the Adjusted Reference Temperature (ART) as defined in Reg. Guide 1.99, Rev. 2, be used as RT$_{NDT}$
    - ART accounts for embrittlement by shifting reference temperature higher
What is Scope of P-T Limits Curves? (1/2)

- 10CFR50 Appendix G requires consideration of all ferritic RPV components
  - Vessel shell and nozzles

- Until recently, PWR nozzles were not always being explicitly considered in P-T curves

- A 1975 Westinghouse analysis (WCAP-7924-A) demonstrated beltline is limiting
  - That analysis assumed a $\frac{1}{10} T$ flaw and used WRC Bulletin 175 methodology

  - B.2.2.2, "Calculations need only be performed for the beltline region, if the $RT_{NDT}$ of the beltline is demonstrated to be adequately higher than the $RT_{NDT}$ for all higher stressed regions."
What is Scope of P-T Limits Curves? (2/2)

- In a 2012 public meeting with NRC, PWROG reported results of a study that found, in some cases, vessel nozzles can be more limiting for P-T curves than the beltline area, even when beltline is irradiated to 60 year fluence (ML12116A087)
  - Nozzle stress concentration due to structural discontinuity
- NRC became concerned regarding the adequacy of existing P-T curves
- NRC issued Regulatory Issue Summary 2014-11, “Information on Licensing Applications For Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components” (ML14149A165)
  - “…all ferritic components within the entire reactor vessel must be considered in the development of P-T limits, and the effects of neutron radiation must be considered for any locations that are predicted to experience a neutron fluence exposure greater than 1x10¹⁷ n/cm² (E > 1 MeV) at the end of the licensed operating period.”
Effective Period of P-T Limit Curves

- Normally, P-T limit operating curves are calculated for a particular fluence period (e.g., through a selected value of EFPY)
  - This corresponds to a maximum fluence and limiting ART value for the specified EFPY

- If new surveillance data, fluence evaluations, or plant capacity factor cause the ART of the vessel limiting material to be greater than that assumed for the current P-T curves, the curves may not be bounding for the EFPY
  - The period of validity of the curves may be shortened
  - It has sometimes happened that new data show the curves are no longer valid at all

- Highly embrittled vessels may face restrictive operating limits that make it necessary for heatup and cooldown transients to be conducted more slowly, increasing costs
RPV P-T Curves are Composite Curves

- RPV P-T curves are composite curves constructed from the most limiting pressure at a given temperature for various analytical conditions.

- Vessel shell (beltline) analysis
  - Consider both I.D. and O.D. axial and circumferential 1/4T elliptical surface flaws
    - For cooldown, O.D. surface flaw does not need to be considered because compressive thermal stresses counter tensile pressure stresses
    - Code Case N-588 (1997) permits circumferentially oriented 1/4T reference flaw to be assumed for vessel girth welds
  - Consider various heatup and cooldown rates, and steady state conditions

- Nozzle analysis
  - 1/4T circular corner flaw (I.D.)
    - Postulation of O.D. (3/4T) flaw not required for nozzles because pressure stress is significantly lower than at nozzle inside surface (see ML13249A386, pg. 34 of 40)
  - ORNL/TM-2010/246, Rev. 1 (ML12181A162) provides simplified closed-form LEFM solutions for nozzle corner cracks
Deterministic Procedures for P-T Curves

- ASME Section XI, Appendix G allowable limits are based on
  - large (¼-thickness) assumed reference flaw with 6:1 aspect ratio
  - $K_{IC}$ (lower bound crack initiation) reference toughness
    - Some plants still use $K_{IR}$ (lower bound crack arrest curve)
  - Adjusted $RT_{NDT}$ (i.e., ART) to account for vessel embrittlement
  - Safety factor on membrane pressure stress (SF = 1.5 for hydrostatic/leak test, SF = 2 for heatup/cooldown)
  - Thermal stresses for maximum heatup/cooldown rates (safety factor of 1 on thermal stress intensity)

- Note: Appendix G also provides a risk-informed P-T limit methodology that is not approved by NRC; not addressed here
Basic equation for determining P-T operating limit curves:

\[(SF) \, K_{im} + K_{lt} < K_{IC}\]

where:

- \(K_{ic}\) = lower bound, plane strain, crack initiation fracture toughness for the material as represented in Figure 4-3 (depends on metal temperature and material RT\(_{NDT}\))
- \(K_{im}\) = stress intensity factor caused by membrane (pressure) stress
- \(K_{lt}\) = stress intensity factor caused by the thermal gradients through the vessel wall (depends on heatup/cooldown rate and vessel wall thickness)

Requires flaw size assumption (e.g., 1/4T)
Deterministic Procedures for P-T Curves

To actually calculate a P-T limit curve, equation is algebraically rearranged to solve for allowable pressure at each specified temperature:

\[
\text{allowable pressure} = \frac{t \times (K_{lc} - K_{lt})}{(SF \times M_m \times R_i)},
\]

where

- \( t \) = vessel wall thickness, in.
- \( M_m \) = An influence coefficient to convert applied stress to crack tip stress intensity; depends on flaw orientation (axial flaws for plates, forgings, and axial welds; circumferential flaws for circumferential welds) and material thickness. ASME XI Appendix G specifies values for \( M_m \)
- \( R_i \) = vessel inside radius, in.
- \( SF \) = safety factor of 2 for level A and level B service limits (for heatup and cooldown); 1.5 for hydrostatic and leak test conditions when the reactor core is not critical
Key Elements of P-T Limits

Curves are calculated for several conditions:

- Normal operation (heatup and cooldown)
  - core critical operation
  - core not critical operation
  - includes boltup temperature limits

- Hydrostatic pressure and leak tests
  - includes boltup temperature limits
Steps in Calculating Appendix G Curves

1. Perform Thermal Analysis of the Vessel
   – calculate through-wall temperatures

2. Perform Stress Analysis of the Vessel
   – calculate thermal and pressure stresses

3. Fracture Mechanics Analysis of the Vessel
   – calculate crack tip SIF and toughness

4. Determine Allowable Pressure vs. Temperature
   
   \[
   \text{allowable pressure} = t \times (K_{lc} - K_{lt}) / (SF \times M_m \times R_i)
   \]
Thermal Analysis of the Vessel

- ASME XI Appendix G provide conservative optional procedures for estimating stress intensity due to effect thermal gradients

- Branch Technical Position 5-3 also permits use of ORNL/NRC/LTR-03/03, “Tabulation of Thermally-Induced Stress Intensity Factors (K_{IT}) and Crack Tip Temperatures for Generating P-T Curves per ASME Section XI-Appendix G” (ADAMS ML100840745)
  - 120 tables of computer-generated crack-tip temperatures and K_{IT} corresponding to the ¼-t and ¾-t crack tip locations for BWR and PWR vessel geometries for varying CDR and HUR
  - Used by NRC to check P-T submissions
  - These methods are applicable only for shell regions, not discontinuities
Solving for $K_{IC}$

- Critical stress intensity factor ($K_{IC}$) is determined using the curve shown in ASME XI Appendix G Figure G 2210-1
  - Shows the relationship between $K_{IC}$ and $(T - RT_{NDT})$; see equation below
- Adjusted $RT_{NDT}$ is calculated at deepest point of postulated defect, °F (per Regulatory Guide 1.99, Revision 2)

$$K_{IC} = 33.2 + 20.734 \exp[0.02(T - RT_{NDT})] \text{ksi}\sqrt{\text{in}}$$

- For $RT_{NDT}$, use the ART of the vessel limiting material (1/4T for I.D. flaw, 3/4T for O.D. flaw) for the shell region, or if evaluating a non-shell region, the specific ART of that region:

$$K_{IC} = 33.2 + 20.734 \exp[0.02(T - ART)] \text{ksi}\sqrt{\text{in}}$$
Evolution of Requirements in ASME XI Appendix G

- Code Case N-588 (1997)
  - Permitted circumferentially oriented 1/4T reference flaw to be assumed for vessel girth welds
- Code Case N-640 (1999)
  - Permitted use of $K_{IC}$ reference fracture toughness (instead of $K_{IR}$) for P-T limit curves
- Code Case N-641 (2000)
  - Combined Code Cases N-588 and N-640 and provides a method for plant-specific LTOP $T_{enable}$ temperature
- Now incorporated into ASME XI Appendix G
Closure Flange Minimum Temperature Limits
Closure Flange Minimum Temperature Limit

- 10 CFR 50 Appendix G, Table 1, establishes minimum temperature requirements for the vessel that are based on the highest reference temperature, $RT_{NDT}$, of the material in the closure flange region that is highly stressed by the bolt preload (tensioning of the RPV closure studs).

- For normal operation / core not critical, vessel pressure may not exceed 20% of the pre-service hydrostatic test pressure (~ 621 psig for a PWR and ~ 300 psig for a BWR) until the metal temperature of the closure flange region is greater than Initial $RT_{NDT} + 120^\circ F$.
Exemption from Flange Limits

- During heatup the closure flange may become limiting for P-T limits
- Some plants have eliminated this flange notch limit via exemption request
- WCAP-15315, Rev. 1, “Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants” (2002) petitioned for removal of requirements
  - NRC did not generically approve, but several (~8) plants have been approved based on plant-specific licensing submittals
  - NRC intended to address petition during the Appendix G rulemaking which has now been abandoned
What is the Basis for Flange Minimum Temperature Requirements?

- NRC staff evaluated Appendix G minimum temperature requirements in PVP2011-57208
- Basis of minimum temperature requirements was detailed in SECY-83-80; requirements are based on $K_{IA}$ fracture toughness curve and use of a 0.1t flaw
  - Limits have not been updated to use $K_{IC}$ curve (which is now used for P-T curves)
  - If $K_{IC}$ fracture toughness were used, minimum temp. requirements would be significantly lower

Minimum (T-RT<sub>NDT</sub>) based on $2K_i$ for a 0.1t deep flaw ($L/a=6$).

Figure from E. Focht, “Evaluation Of The Minimum Temperature Requirements For The RPV Closure Head Flange Region In 10 CFR 50 Appendix G,” PVP2011-57208
## Conservatism of the “Initial RT_{NDT} + 120°F” Requirement

<table>
<thead>
<tr>
<th>Plant</th>
<th>Column A T-RT_{NDT} (°F) using K_{la} (a/t = 0.1, SF=2)</th>
<th>Column B T-RT_{NDT} (°F) using K_{IC} (a/t = 0.1, SF=2)</th>
<th>Column C Amount of Conservatism (°F)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>PWR</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>CE</td>
<td>68</td>
<td>13</td>
<td>120 - 13 = 107</td>
</tr>
<tr>
<td>B&amp;W</td>
<td>100</td>
<td>41</td>
<td>120 – 41 = 79</td>
</tr>
<tr>
<td>Westinghouse 4 Loop</td>
<td>1</td>
<td>0</td>
<td>120 – 0 = 120</td>
</tr>
<tr>
<td>Westinghouse 3 Loop</td>
<td>0</td>
<td>0</td>
<td>120 – 0 = 120</td>
</tr>
<tr>
<td><strong>BWR</strong></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>GE (CB&amp;I 251-in.)</td>
<td>97</td>
<td>38</td>
<td>120 – 38 = 82</td>
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<tr>
<td>GE (B&amp;W 251-in.)</td>
<td>118</td>
<td>56</td>
<td>120 – 56 = 64</td>
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<tr>
<td>GE (CE 218-in.)</td>
<td>43</td>
<td>0</td>
<td>120 – 0 = 120</td>
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</tbody>
</table>
Documenting Inputs for P-T Curve Submittals
Documenting Inputs for P-T Limit Submittals (1/2)

- If P-T limits are not generated per an approved Pressure-Temperature Limit Report (PTLR), then changes must be reviewed and approved by NRC per 10CFR50.90.

- The NRC staff provided this template for “documenting inputs for licensee submittals that would facilitate the staff’s review, reduce the need for requests for additional information (RAIs), and make the submittal review process more efficient” (ML15155B464)

- NRC requests P-T curves be submitted 18 months in advance.

<table>
<thead>
<tr>
<th>GEOMETRY DATA</th>
<th>Description</th>
<th>Value</th>
<th>Units</th>
<th>Source*</th>
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<tr>
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<td>Value</td>
<td>Units</td>
<td>Source*</td>
<td></td>
</tr>
<tr>
<td>(to inside surface of clad)</td>
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<td></td>
<td></td>
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<tr>
<td>Cladding thickness</td>
<td>Value</td>
<td>Units</td>
<td>Source*</td>
<td></td>
</tr>
<tr>
<td>RPV base metal wall thickness</td>
<td>Value</td>
<td>Units</td>
<td>Source*</td>
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</table>

<table>
<thead>
<tr>
<th>REFERENCE TEMPERATURE DATA (for all materials with EOL fluence &gt; 1x10^{17} n/cm^2)</th>
<th>Description</th>
<th>Value</th>
<th>Units</th>
<th>Source*</th>
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<tbody>
<tr>
<td>Material description</td>
<td>Value</td>
<td>Units</td>
<td>Source*</td>
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<td>Material type (i.e., SA-533 Gr. B Cl. 1)</td>
<td>Value</td>
<td>Units</td>
<td>Source*</td>
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<td>Product form (plate, weld, forging)</td>
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<td>Units</td>
<td>Source*</td>
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<tr>
<td>Thickness (if different than above)</td>
<td>Value</td>
<td>Units</td>
<td>Source*</td>
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<tr>
<td>Heat No.</td>
<td>Value</td>
<td>Units</td>
<td>Source*</td>
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<tr>
<td>Initial reference temperature, IRT_{NDT}</td>
<td>Value</td>
<td>Units</td>
<td>Source*</td>
<td></td>
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<tr>
<td>(including description of method used to determine IRT_{NDT})</td>
<td>Value</td>
<td>Units</td>
<td>Source*</td>
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<tr>
<td>Copper content, Cu</td>
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<td>Nickel content, Ni</td>
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<td>(including description of method used to determine CF)</td>
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<td>Calculated irradiation shift, ΔRT_{NDT}</td>
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<td>Source*</td>
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<td>Units</td>
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<tr>
<td>Calculated ART</td>
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<td>Units</td>
<td>Source*</td>
<td></td>
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* Note: Source should refer to a docketed submittal or be a reference included with the submittal.
## TRANSIENT DEFINITIONS (specify for each all transients evaluated)

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<tr>
<th>Time</th>
<th>Temperature (units)</th>
<th>Inside Surface Heat Transfer Coefficient (units)</th>
<th>Outside Surface Heat Transfer Coefficient (units)</th>
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Specify all time-history points here

### OTHER DATA

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<td>Preservice Hydrostatic Test Pressure</td>
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### MATERIAL PROPERTY DATA (for both base metal and clad)

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<td>Mean coefficient of thermal expansion</td>
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<td>Stress-free temperature</td>
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### LIMITING NOZZLE DATA

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<tr>
<td>Description of how the stress analysis was performed for the nozzle</td>
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<td>Pressure stress coefficients used to calculate stress intensity factor</td>
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<tr>
<td>Thermal stress coefficients used to calculate stress intensity factor</td>
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<tr>
<td>Initial reference temperature, IRT&lt;sub&gt;NDT&lt;/sub&gt;</td>
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<td>(including description of method used to determine IRT&lt;sub&gt;NDT&lt;/sub&gt;)</td>
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<td></td>
<td></td>
</tr>
</tbody>
</table>

*Note: Source should refer to a docketed submittal or be a reference included with the submittal.
Low Temperature Overpressure (LTOP) Systems
Overpressure Events

- A pressure transient is classified as an overpressure event if the pressure exceeds the ASME Section XI Appendix G allowable pressure.
- There were 29 overpressure events in PWRs through 1976 (28 were PWRs).
- The overpressure events:
  - primarily occurred during startup when RCS was water solid.
  - generally were attributed to operator error.
  - were isothermal and occurred at temperatures ranging from 80°F to 300°F.
Low Temperature Overpressure Protection (LTOP)

- The NRC Standard Review Plan (NUREG-0800) Branch Technical Position 5-2 defines the requirements for low temperature overpressurization protection for PWRs
- LTOP systems must be in place to protect the vessel against brittle fracture due to high pressure (i.e., water solid) operation while at low temperature
  - Protects the low end of the Appendix G curve
- ASME Section XI, Appendix G provides criteria for temperature and pressure setpoints of LTOP systems
Pressure Transients Feb 1972 – Oct 1987

29 Events
Feb 1972- Sept 1976

61 Events
April 1979 - Oct 1987
LTOP Setpoints (1/2)

- Branch Technical Position 5-2 states:

1. A system should be designed and installed that will prevent exceeding the applicable technical specifications and Appendix G limits for the RCS while operating at low temperatures.
   - capable of relieving pressure during all anticipated overpressurization events particularly while the RCS is in a water-solid condition

2. The low-temperature overpressure protection system should be operable during startup and shutdown conditions below the enable temperature, defined as the water temperature corresponding to a metal temperature of at least $RT_{NDT} + 90^\circ F$ (50°C) at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations
   (additional requirements are also given)
LTOP Setpoints (2/2)

- ASME Section XI, Appendix G states that:
  - LTOP systems shall be effective at coolant temperatures less than 200°F (95°C) or at coolant temperatures corresponding to a reactor vessel metal temperature less than $RT_{NDT} + 50°F$ (28°C), whichever is greater
- ASME enable temp is less conservative than BTP 5-2 but NRC approves setpoints that use ASME criteria
- Plants implement LTOP with either single-setpoint systems or variable setpoint systems
  - Variable setpoint systems allow operation closer to the P-T limit curve and do not constrict the operating envelop as severely
- LTOP limits should be set based on the steady state P-T curve
  - Why: LTOP events are most likely to occur during isothermal conditions
ASME XI NONMANDATORY APPENDIX E, EVALUATION OF UNANTICIPATED OPERATING EVENTS
What Happens If P-T Limits Are Exceeded?

- ASME XI IWB-3720 requires that “When an operating event causes an excursion outside the normal operating pressure and temperature limits defined in the plant Technical Specifications, an engineering evaluation shall be performed to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System”

- Developed in 1985, ASME Section XI Nonmandatory Appendix E provides procedures to evaluate RPV integrity following a unanticipated pressure excursion which exceeds Appendix G limits
Nonmandatory Appendix E

- Paragraph E-1200 provides conservative screening criteria for quickly assessing vessel integrity after an unanticipated event (e.g., isothermal pressure transient or pressurized thermal transient) without significant analyses.

- In the event the screening criteria of E-1200 cannot be met, Paragraph E-1300 provides a procedure for performing a more detailed transient-specific fracture mechanics analysis.

- PVP2011-57201 documented an analysis performed to demonstrate Appendix E is consistent with NRC risk-informed criteria.

- Appendix E is not applicable to BWRs.
  - BWRs typically address these events by calculating a P-T curve for the event, or a bounding event, then comparing the actual transient to the P-T curve to demonstrate whether adequate margins to nonductile failure were maintained.
Administrative Margins
Administrative Margins

- Tech Spec or plant operating limit curves must include some additional administrative margin to accommodate instrument uncertainty and to assure that the “real” P-T curves are not violated, which is a reportable event.
- Administrative margins are related to the accuracy and uncertainty of the instruments used (i.e., narrow range vs. wide range instruments).
- Typical administrative margins are 60 psi and 10°F
  - Not all plants adopt instrument uncertainty.
Administrative Control of P-T Limits
Administrative Control of P-T Limits

- In the past, P-T limits and Low Temperature Over Pressure (LTOP) limits were part of a plant’s Technical Specifications, and updating them required submission of plant license amendments, which require NRC review and approval (a burden on plant and NRC resources)

- Generic Letter 96-03 permitted plants to define P-T limit/LTOP methodology in a Pressure and Temperature Limits Report (PTLR), which is pre-approved by the NRC
  - A unit-specific document that provide the P-T limits, heatup / cooldown rates, and LTOP setpoints
  - P-T updates do not require a change in plant tech specs if generated in accordance with the NRC-approved methodology in the PTLR
    - After it is first approved, future PTLR revisions do not require NRC approval unless they invoke a change in methodology
Together…Shaping the Future of Electricity
11.1 – Pressurized Thermal Shock – PTS Rule

Nathan Palm
Sr. Technical Leader

RPV Integrity Workshop,
2016 International LWR Materials Reliability Conference, Chicago, USA
August 1, 2016
Learning Objectives

- Be able to define a PTS event
- Be able to identify the PTS regulations within the CFR
- Be able to identify the PTS Screening Limits
- Be able to identify the available remedial actions to prevent exceeding PTS screening limits
- Have a basic understanding of how $R_{PTS}$ is determined
## Roadmap for RPV Integrity Rules, Guidance and Standards

<table>
<thead>
<tr>
<th>Category</th>
<th>Relevant Documents/Standards</th>
</tr>
</thead>
</table>
Description of PTS Event

- PTS is a rapid cooldown of the inside wall of the vessel, accompanied by either sustained high reactor coolant system pressure or a subsequent re-pressurization of the system
  - Produces high transient thermal stresses, tensile on the inner wall and compressive at the outer wall; in combination with pressure stresses (which are always tensile and greatest at the inner wall)
  - Rapid cooldown → reduced temperature → reduced toughness
  - Neutron embrittlement → even lower toughness
  - High inner wall tensile stresses coupled with low toughness create significant potential for flaw extension
Potential Causes of PTS Events

- PTS events can result from
  - System transients initiated by instrumentation and control system malfunctions (e.g., stuck open valves in either the primary or secondary system)
  - Small break loss-of-coolant accidents (SBLOCA), main steam line breaks, and feedwater pipe breaks
  - Several incidents in the late 1970s and early 1980s drew attention to the impact of operator action on PTS events

- PTS not a significant concern for BWRs
  - BWRs have a larger volume of water in the vessel at saturated conditions than PWRs; a sudden drop in temperature will condense steam, decrease pressure
10 CFR 50.61, “PTS Rule”

- 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events” (PTS Rule) - adopted in 1983
  - Establishes screening criteria below which the potential for a reactor vessel to fail due to a PTS event is deemed to be acceptably low
  - The screening criteria effectively define a limiting level of embrittlement beyond which a PWR may not operate without further plant-specific evaluation
    - $R_{PTS}$ (a calculated ART at the vessel inner surface) must remain below 270°F for plates, forgings, and axial welds, or 300°F for circumferential welds for the entire operating life of the vessel
Basis of PTS Screening Criteria

- PTS screening criteria were based on goal of limiting the through-wall cracking frequency (TWCF) to a value < $5 \times 10^{-6}$ per reactor-year (ry)
- TWCF was assumed to be equivalent to reactor vessel failure and core damage
- SECY-82-465 (Proposed PTS rule, 1982) found that acceptable TWCF was in the range of $6 \times 10^{-6}$ to $1 \times 10^{-5}$/ry, a value deemed consistent with the intent of 10 CFR 50 Appendix A, General Design Criteria (GDC) 14, “Reactor Coolant Pressure Boundary,” and GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary”
- RPV failure due to a PTS event is deemed to be acceptably low if $R_{PTS}$ is below the screening limit
Corrective Actions

- If a vessel is projected to exceed the screening limits by end of life (EOL), actions must be taken to keep the vessel below the screening limit; details of the approach selected must to be submitted for NRC approval at least 3 years before the reactor vessel is projected to exceed the PTS screening criteria
  - (1) Apply for licensing under Alternative PTS Rule 10CFR50.61a (optional), or
  - (2) Implement “reasonably practicable” flux reductions to reduce the embrittlement rate
  - (3) If practicable flux reduction cannot prevent $RT_{\text{PTS}}$ from exceeding 270°F/300°F, perform a plant-specific analysis to show that $RT_{\text{PTS}}$ above 270°F/300°F is safe
    - Modifications to plant equipment, systems, and operation may be necessary
    - Analysis must be submitted to NRC for review and approval
    - Use of Reg. Guide 1.154 was previously suggested but has been removed
  - (4) If (2) and (3) are unsuccessful, de-embrittle the vessel by thermal annealing in accordance with 10 CFR 50.61(b)(7), 10 CFR 50.66, and guidance in Reg. Guide 1.162
Determination of $\text{RT}_{\text{PTS}}$

- The procedure for calculating $\text{RT}_{\text{PTS}}$ closely mirrors the procedure for calculating Adjusted Reference Temperature (ART) detailed in Reg. Guide 1.99, Rev. 2
  - The essential difference is that while ART is typically calculated for the $\frac{1}{4}$-T and $\frac{3}{4}$-T vessel wall depths at any fluence (e.g., EFPY) of interest, $\text{RT}_{\text{PTS}}$ is calculated at the inside surface (specifically, the clad-base-metal interface) for the end-of-license (EOL) fluence only
  - There is also some difference in terminology
  - 10 CFR 50.61 itself details the procedure for determination of $\text{RT}_{\text{PTS}}$; it does not refer to RG1.99R2
Determination of $RT_{PTS}$

$$RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$$

- $RT_{NDT(U)}$ is the reference temperature for unirradiated material as defined in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code (also termed Initial $RT_{NDT}$)
  - Some plants have measured values of Initial $RT_{NDT}$; other plants use generic values
  - For generic values of weld metal, the following generic mean values must be used unless justification for different values is provided:
    - 0°F (-18°C) for welds made with Linde 80 flux
    - -56°F (-49°C) for welds made with Linde 0091, 1092, and 124 and ARCOS B-5 weld fluxes.
    - For Linde 80 welds, the approach defined in BAW-2308, Rev. 2 can be used to define lower values of $RT_{NDT(U)}$
Determination of $RT_{PTS}$

$$RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$$

- $\Delta RT_{PTS}$ is $\Delta RT_{NDT}$ evaluated at the EOL fluence. $\Delta RT_{NDT}$ is the mean value of the adjustment in reference temperature caused by irradiation, as calculated by:

$$\Delta RT_{NDT} = (CF) f^{(0.28 - 0.1 \log f)}$$

where,

CF ($^\circ$F) is the chemistry factor (CF can either be from 10 CFR 50.61 Table 1 (welds) or 2 (base metals), or a factor based on the “best fit” of two or more surveillance test data);

and $f$, is the best estimate fluence, in units of $10^{19}$ n/cm$^2$ (E > 1 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence for the period of service in question (end of license)

- Same as RG1.99R2: $\Delta RT_{NDT} = CF \times FF$; only difference is location (vessel ID, not $1/4T$)
**Determination of RT<sub>PTS</sub>**

\[
RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}
\]

- “Margin” is added to account for uncertainties in the values of \(RT_{NDT(U)}\), copper and nickel contents, fluence, and calculation procedures:

where,

\[
\text{Margin} = 2\sqrt{\sigma_U^2 + \sigma_\Delta^2}
\]

\(\sigma_U\) is the standard deviation for \(RT_{NDT(U)}\). If a measured value for the material is available, \(\sigma_U\) is estimated from the precision of the test method (and is normally taken to be 0°). If not, and generic mean values for the class of material are used, \(\sigma_U\) is the std. deviation (SD) obtained from the set of data used to establish the mean. If the generic mean Initial \(RT_{NDT}\) value of a Linde 80, 0091, 1092 and 124 or ARCOS B-5 weld is used, then \(\sigma_U\) is 17°F

\(\sigma_\Delta\) is SD for \(\Delta RT_{NDT}\). \(\sigma_\Delta\) is 28°F for welds and 17°F for base metal, except that \(\sigma_\Delta\) need not exceed 0.50 times the mean value of \(\Delta RT_{NDT}\).
Current Status of Compliance with PTS Rule

- Although PTS is a low-probability event, the PTS screening criteria are so conservative that a few PWRs are close to the 10 CFR 50.61 screening limits
  - These plants are likely to exceed PTS screening limit during the 20 year extended license period
  - A few additional plants will likely exceed the PTS screening limits prior to 80 years of operation

- Some plants maintain low $RT_{PTS}$ by implementing flux reduction programs which penalize the economic operation of the reactor

- Many B&W fabricated plants had $RT_{PTS}$ values approaching the PTS screening criteria but were able to reduce their values through implementation of BAW-2308 (use Master Curve for Initial $RT_{NDT}$)
Alternative PTS Rule

- NRC in 2010 added an alternative PTS regulation, 10 CFR 50.61a, for those plants which have difficulty meeting the 10 CFR 50.61 PTS criteria
  - Only available for use by plants whose construction permit was issued before the effective date of the 10 CFR 50.61a rule and only if the vessel was designed/constructed to the 1998 (or earlier) edition of the ASME Code
  - A PWR licensee may either continue to comply with the “old PTS rule” 10 CFR 50.61 or may apply for a license amendment to implement the new requirements of 10 CFR 50.61a
  - Will be discussed more later
Review of Learning Objectives

- Be able to define a PTS event
- Be able to identify the PTS regulations within the CFR
- Be able to identify the PTS Screening Limits
- Be able to identify the available remedial actions to prevent exceeding PTS screening limits
- Have a basic understanding of how $RT_{PTS}$ is determined
Together…Shaping the Future of Electricity
11.2 – Alternate PTS Rule, 10 CFR 50.61a

Nathan Palm
Sr. Technical Leader

RPV Integrity Workshop,
2016 International LWR Materials Reliability Conference, Chicago, USA
August 1, 2016
Learning Objectives

- Reasons why Alternate PTS Rule was issued
- Major differences between Original and Alternate PTS Rules
- Key requirements of the Alternate PTS Rule
Background

- It had been more than 30 years since 10 CFR 50.61 PTS Rule was issued.
- 10 CFR 50.61 was based on PFM technology available in the 1980s but the state of knowledge (PFM, transient analysis, etc.) greatly improved.
- NRC decided to update the PTS Rule because:
  - PTS screening criteria were creating an artificial impediment to license renewal for some plants.
  - Original 10 CFR 50.61 screening limits were known to be overly conservative because of the 1980’s PFM technology.
Conservatisms Identified in 1980s PTS Analysis

- Treatment of plant transients was highly simplified
- Little credit for operator action
- Fracture toughness was characterized by RT_{NDT}, a parameter which was intentionally conservative
- Assumed flaw distribution placed all flaws on the interior surface of the RPV and sized them much larger than those found during inservice inspections
- RPV model assumed the RPV was constructed entirely of its most embrittled constituents
- Fluence model for assessing embrittlement assumed that all interior surfaces of the RPV experience peak fluence
Alternate PTS Rule

- In 2010, NRC added an Alternate PTS regulation, 10 CFR 50.61a, for those plants which have difficulty meeting the 10 CFR 50.61 PTS criteria
  - Available for use only by plants whose construction permit was issued before the effective date of the 10 CFR 50.61a rule and only if the vessel was designed/constructed to the 1998 (or earlier) edition of the ASME Code
  - A PWR licensee may either continue to comply with the “old PTS rule” 10 CFR 50.61 or may apply for a license amendment to implement the new requirements of 10 CFR 50.61a
  - Still a relatively new regulation, industry experience is just starting
Alternate PTS Rule

- 10 CFR 50.61a, “Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events” (Alternate PTS Rule)
  - The technical bases were based on a risk-informed / probabilistic approach, more sophisticated than the original PTS Rule bases
  - NUREG-1874, NUREG-1806
  - Considerably more complex to implement but is likely to reduce or eliminate PTS concerns for those plants that have problem meeting the 10 CFR 50.61 PTS Rule screening criteria
Technical Bases of Alternate PTS Rule

- **Detailed PRA/PFM analyses**
  - 3 pilot plants evaluated (Beaver Valley 1, Oconee 1 and Palisades)
  - PFM performed using newly developed FAVOR code
  - More realistic flaw distributions from NUREG/CR-6817 used

- **Indirect use of Master Curve (MC) toughness instead of $R_{NT}$, which has an inherent conservative bias**
  - MC removes the epistemic uncertainty associated with $R_{NT}$, leaving only the aleatory uncertainty associated with $K_{IC}$, which results from material variability

- **Findings:**
  - Primary side faults (e.g., LOCA) dominate risk; secondary side faults (MSLB) make minor contribution
  - Axial cracks dominate the risk
    - Circumferential cracks arrest, while axial cracks less likely to arrest
Alternate PTS Technical Approach

Acceptance Criterion for TWC Frequency
Established consistent with
- 1986 Commission safety goal policy statement
- June 1990 SRM
- RG1.174

Probabilistic Estimation of Through-Wall Cracking Frequency

- Probabilistic Fracture Analysis (FAVOR)
- P(t), T(t), & HTQ(t)
- Thermal Hydraulic Analysis (RELAP)
- Sequence Definitions
- PRA Event Sequence Analysis (SAPHIRE)

Screening Limit Development

Yearly Frequency of Thru-Wall Cracking

Vessel damage, age, or operational metric

Generalization to all U.S. PWRs

Figure from NUREG-1874
Metric Used for Acceptance Criteria

- Establishes tolerable limit on Through-Wall Cracking Frequency (TWCF) as $10^{-6}$ events/reactor year
  - The legal Large Early Release Frequency (LERF) limit is $10^{-6}$
  - TWCF rarely leads to LERF
  - Equivalence between TWCF and LERF was assumed as the basis for developing the TWCF limit
    - Conservative
    - Eliminated need to model containment response under RPV failure
PTS Generalization Study

- Additional studies of 5 plants with relatively high embrittlement (Salem 1, TMI 1, Ft. Calhoun, Diablo Canyon 1, and Sequoyah 1)

- Verified plant features were sufficiently similar to support generalization of pilot plant results to remainder of the U.S. PWR fleet

- Identified parameters which required some customization of screening criteria to cover all cases:
  - Vessel thickness (thicker vessels are subject to higher thermal stresses and contain a larger volume of postulated flaws)
  - Presence of underclad cracks in forgings
Summary of Regulation 10 CFR 50.61a

- To use 10 CFR 50.61a, must submit application for license amendment to NRC at least 3 years before the limiting $RT_{PTS}$ value calculated under 10 CFR 50.61 is projected to exceed the 10 CFR 50.61 screening criteria.

- Rule consists of 3 main technical elements
  - $RT_{MAX}$ Based Screening Criteria
  - Surveillance Data Statistical Evaluation
  - Examination and Assessment of Flaws
Major Differences between 10 CFR 50.61 and 10 CFR 50.61a

- The ETC and method of calculation of $RT_{MAX-X}$ values in §50.61a differ significantly from the rules for calculation of $RT_{PTS}$ in §50.61
  - Less restrictive but more complex to calculate
- The “credibility” evaluation in §50.61 (also in RG 1.99R2) is replaced with statistical evaluations
  - Three evaluations in §50.61a as opposed to one in §50.61
  - Evaluations in §50.61a are more complex to perform
- Alternate PTS rule requires detailed analysis of ASME Section XI ISI volumetric exam results to verify that RPV flaw density and size distribution are bounded by the flaw distribution used in the §50.61a technical basis PFM analyses
Determination of Reference Temperatures Under 10 CFR 50.61a

- Reference temperatures (defined as $RT_{\text{MAX-X}}$) for beltline materials are calculated as a function of chemistry (Cu, Mn, P, and Ni wt.%), reactor cold leg temperature, and fast neutron flux and fluence values, and Initial $RT_{\text{NDT}}$
- $RT_{\text{MAX-X}}$ values are predicted using a generic embrittlement trend curve (the “EONY” ETC, used to predict $\Delta T_{30}$)
  - Basis for EONY ETC is given in ORNL/TM-2006/530
- $RT_{\text{MAX-X}}$ values are calculated for each product form in the beltline (plate, forging, axial weld, circum weld)
- For welds, the $RT_{\text{MAX-X}}$ value is computed using the weld fluence, but the more limiting of either the weld or adjacent plate/forging properties
  - $RT_{\text{MAX-X}}$ values for welds may be controlled by the adjacent plate or forging materials
### RT<sub>M</sub>AX Based Screening Criteria for 10 CFR 50.61a

<table>
<thead>
<tr>
<th>Product Form and RT&lt;sub&gt;M&lt;/sub&gt;AX-X Values</th>
<th>RT&lt;sub&gt;M&lt;/sub&gt;AX-X Limits [°F (°C)] for Different Vessel Wall Thicknesses&lt;sup&gt;1&lt;/sup&gt; (T&lt;sub&gt;WALL&lt;/sub&gt;)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>T&lt;sub&gt;WALL&lt;/sub&gt; ≤ 9.5 in (24.1 cm)</td>
</tr>
<tr>
<td>Axial Weld RT&lt;sub&gt;M&lt;/sub&gt;AX-AW</td>
<td>269 (132)</td>
</tr>
<tr>
<td>Plate RT&lt;sub&gt;M&lt;/sub&gt;AX-PL</td>
<td>356 (180)</td>
</tr>
<tr>
<td>Forging without underclad cracks RT&lt;sub&gt;M&lt;/sub&gt;AX-FO&lt;sup&gt;2&lt;/sup&gt;</td>
<td>356 (180)</td>
</tr>
<tr>
<td>Axial Weld and Plate RT&lt;sub&gt;M&lt;/sub&gt;AX-AW + RT&lt;sub&gt;M&lt;/sub&gt;AX-PL</td>
<td>538 (281)</td>
</tr>
<tr>
<td>Circumferential Weld RT&lt;sub&gt;M&lt;/sub&gt;AX-CW&lt;sup&gt;3&lt;/sup&gt;</td>
<td>312 (156)</td>
</tr>
<tr>
<td>Forging with underclad cracks RT&lt;sub&gt;M&lt;/sub&gt;AX-FO&lt;sup&gt;4&lt;/sup&gt;</td>
<td>246 (119)</td>
</tr>
</tbody>
</table>

<sup>1</sup> Wall thickness is the beltline wall thickness including the clad thickness.

<sup>2</sup> Forgings without underclad cracks apply to forgings for which no underclad cracks have been detected and that were fabricated in accordance with Regulatory Guide 1.43.

<sup>3</sup> RT<sub>P</sub>TS limits contribute 1x10<sup>-8</sup> per reactor year to the reactor vessel TWCF.

<sup>4</sup> Forgings with underclad cracks apply to forgings that have detected underclad cracking or were not fabricated in accordance with Regulatory Guide 1.43.
Surveillance Data Statistical Evaluation (1/2)

- Plant-specific and “sister” surveillance data are required to be checked against embrittlement trend in §50.61a
- Purpose is to assess how well the surveillance data are represented by the generic embrittlement trend curve
  - If surveillance data generally follow trend curve, then embrittlement trend curve is used
  - If it deviates significantly (non-conservatively) from trend curve, then plant must propose/justify alternative methods to predict the shift trend used in the calculation of $RT_{MAX-X}$
- 3 Statistical tests are required
  - Mean Test
  - Slope Test
  - Outlier Test
Surveillance Data Statistical Evaluation (2/2)
Examination and Assessment of Flaws

- Flaws located within the inner 1” or 1/10th of the RPV ID must meet limits in §50.61a
  - Different limits for plate and weld flaws
  - Number of allowable flaws is based on flaw size
- Flaws located between 1-inch and 3/8t from the RPV ID must meet the requirements of ASME Section XI, Table IWB-3510-1
- Axial flaws greater than 0.075” TWE at the clad/base metal interface must be verified to not open to the RPV inside surface
- Analysis must be performed using results of a ASME Section XI, Appendix VIII exam
Implementation of 10 CFR 50.61a

- NRC has developed a regulatory guide for implementation of the Alternate PTS Rule
  - DG-1299 / RG 1.230
  - Tech Basis Document is NUREG-2163
  - Expected to be published in Summer 2016
  - Previously, EPRI provided recommendations for issues/ambiguities in the Rule that should be addressed in the proposed Reg. Guide
    - MRP-334, product number 1024811

- One U.S. plant has been approved to implement 10 CFR 50.61a
  - No significant NRC requests for additional information
  - Plant did perform specialized examinations of flaws in the clad-to-base-metal interface (none were through cladding)
Review of Learning Objectives

- Reasons why Alternate PTS Rule was issued
- Major differences between Original and Alternate PTS Rules
- Key requirements of the Alternate PTS Rule
Together…Shaping the Future of Electricity
12 – Current Challenges and Future Directions for RPV Integrity

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Technical Executive

RPV Integrity Workshop,
2016 International LWR Materials Reliability Conference,
Chicago, USA
August 1, 2016
Topics

- Current RPV Issues and Challenges
  - Appendix G P-T curve small surface flaw issue
  - Issues related to vessel fabrication / hydrogen flaking
  - Issues related to the initial fracture toughness values for pre-1973 plants

- Future directions
  - Mitigation of vessel embrittlement / Vessel Annealing
  - Direct Use of Fracture Toughness for RPV Evaluation
Impact of Small Surface Flaws on P-T Limit Curves and Leak Test Temperatures
(Shallow, internal surface-breaking flaws may cause RPV CPF > 1E-6)
Concern: As Plants Operate Longer and $RT_{NDT}$ Increases, Operating Envelope Becomes More Constrained

Result: longer startup and shutdown times, higher leak test temperatures, higher costs, and more chance of violating limits
Risk-Informed P-T Curve Methodology was Developed to Relieve Constrained Operating Window

- In 2009, MRP and BWRVIP developed a risk-informed method for generating P-T curves as an alternative to the deterministic methods of ASME XI App G
  - Used FAVOR, Fracture Analysis of Vessels, Oak Ridge
  - Opens the envelope at low temps for PWRs
  - Reduces leak test temperatures for BWRs
  - Analyses postulated only buried flaws; did not consider surface flaws
- In 2011, was adopted in ASME XI Appendix G (paragraph G-2216)
- NRC & ORNL evaluated the risk-informed approach for acceptance
NRC Methodology

- NRC evaluated both risk-informed P-T methodology and the traditional (1/4T) Appendix G methodology
  - Assume RPV cooldown or heatup follows $P_{\text{MAX}}$ vs. time determined per ASME equations
  - Use FAVOR to calculate CPI & CPF estimates
    
    $\text{CPI} = \text{conditional probability of initiation}$
    $\text{CPF} = \text{conditional probability of failure}$

- Unlike MRP-250, considered surface-breaking flaws of various sizes

- Compared results vs. RG1.174-motivated limit values ($10^{-6}$ failures/reactor year)

Figure from ORNL/TM-2012/489
(2b) The ¼T flaw does not produce the highest risk for cooldown, shallow flaws do.

Reason the shallow flaw has high CPF: crack tip of shallow flaw is subjected to high stress due to difference in coefficient of thermal expansion between austenitic clad and ferritic base metal; that stress peaks at end of cooldown.

Implication: PT curves based on ¼T flaws are not conservative for all conditions.
Status of ASME Appendix G P-T Curve Issue

- 2012-2013 MRP performed independent FAVOR analyses (MRP-368)
- 2014: MRP work continued, seeking to define cooldown rate limits that ensure safety goals are met under all conditions
  - Goal was to develop a proposed change to ASME XI Appendix G (via Code Case) that limits cooldown rate if necessary to ensure CPF < 1E-6
- 2014: BWRVIP began investigation of impact of shallow surface flaws on BWR leak test conditions
- Before work could be completed, computational anomalies in the FAVOR software were identified (a “bug”)
- MRP informed NRC in August 2014
- NRC and ORNL are revising FAVOR to address the bug; new version expected soon
Stress Free Temperature Approach

- Much of the small surface flaw issue is attributable to FAVOR stress model at clad-base metal interface
- FAVOR uses a Stress-Free Temperature (SFT) approach for handling cladding residual stress
  - SFT approach assumes: At SFT, RPV shell is free from thermal strain; stress due to differences in Coefficient of Thermal Expansion (CTE) between clad and base metal, and the clad residual stresses, offset each other
- Derivation of the recommended Stress Free Temperature (SFT) used in FAVOR was originally detailed in NUREG/CP-0166 (ML042230476)
  - Measured strains were resolved into circumferential and longitudinal stress components. The circumferential stress component was larger and was used to derive SFT = 488°F

Figure from ORNL/TM-2015/59531/REV-01 (ML16043A170)
Plans for Future Appendix G Work

- To date both the NRC and industry analyses have used SFT = 488°F (based on the circumferential stress component, which acts on an axial flaw)
- However, the small surface flaw issue is a circumferential flaw issue
  - “All inner-surface flaws [in the 10CFR50.61a basis work] are oriented circumferentially because in the examinations of the PVRUF vessel all of the observed flaws in the cladding were circumferentially oriented. This observation was consistent with expectations, because weld-deposited cladding is applied to vessel inner surfaces as a series of circumferential weld passes. In the FAVOR model, all inner surface-breaking flaws are associated only with the vessel cladding process.” (from ORNL/TM-2015/59531/REV-01)
- In response to question posed by EPRI (Since we are analyzing a circ flaw, what’s the SFT for a circ flaw?), ORNL made a presentation at 01/2016 public meeting and showed that SFT for a circ flaw would be 364°F (ML16021A008)
- After the new version of FAVOR is released, MRP & BWRVIP will revisit the small flaw issues using a SFT appropriate for the circ flaw being analyzed
Discovery of Fabrication Flaws
(Quasi-Laminar Flaws / Hydrogen Flaking)
Doel 3 / Tihange 2 (D3T2) RPV Quasi-Laminar Flaws

- Westinghouse 3-loop design PWRs built in 1974 by Rotterdam Dockyards (RDM) to ASME III 1973 (1974 addenda), forged ring vessels; SA 508 Cl 3
- Summer 2012: inspection of base metal for possible underclad defects in the pressure vessel beltline revealed thousands of quasi-laminar flaws
  - ~8,000 in Doel 3; 2,000 in Tihange 2 (Fabrication UT did not record)
  - Attributed to hydrogen flaking
    - Large number of indications
    - Clear link between distribution profile in vessel wall and expected positive macro-segregation profile
Doel 3 / Tihange 2 RPV Quasi-Laminar Flaws

- Hydrogen flakes are short, discontinuous internal fissures caused by stresses produced by localized accumulation of gaseous hydrogen after a forging operation
  - Cause: decreased solubility of hydrogen as the metal cools and transforms from austenite to ferrite
Safety Cases Justified Restart in 2013

- Electrabel (licensee) developed safety cases to justify restart
  - Safety cases analyses were based on estimated toughness properties of irradiated flaked material because no irradiated data for flaked material was available

- FANC (Belgian regulator) authorized restart of both units in June 2013
  - However, FANC imposed several post-restart requirements, including need to obtain material property data for irradiated RPV steel containing flakes

- Flaked forging material VB395 was irradiated in BR2 test reactor (Belgium) Jan-Feb 2014
  - Test result: Twice the RT_{NDT} shift that would normally be predicted – safety cases potentially nonconservative
  - Plants shut down again in March 2014

- 3 additional test reactor irradiations of flaked material were conducted, including a new material, KS 02, and Doel 3 nozzle dropout material
  - Ultimately, VB395 deemed not representative
  - Presence of hydrogen flakes does not affect local material properties (PVP2016-63632)
Increase in Number and Size of Quasi-laminar Flaws at D3T2 Reported February 2015 vs. 2012

- Electrabel was also required to demonstrate that the UT method used in 2012 was able to detect all flakes
  - New UT inspections were conducted in 2014 using qualified method
- The 2014 inspections resulted in a significant increase in the reported number and size of flaws

<table>
<thead>
<tr>
<th>Vessel Ring Forging</th>
<th>Number of Detected Flaws</th>
<th>Maximum Size of Flaws (mm)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>2012</td>
<td>2015</td>
</tr>
<tr>
<td>Doel 3 Upper core shell</td>
<td>857</td>
<td>1440</td>
</tr>
<tr>
<td>Doel 3 Lower core shell</td>
<td>7205</td>
<td>11607</td>
</tr>
<tr>
<td>Total for Doel 3:</td>
<td>~8,000</td>
<td>~13,000</td>
</tr>
<tr>
<td>Tihange 2 Upper core shell</td>
<td>1931</td>
<td>3064</td>
</tr>
<tr>
<td>Tihange 2 Lower core shell</td>
<td>80</td>
<td>85</td>
</tr>
</tbody>
</table>
Hydrogen Hypothesis

- Belgian, German, and U.S corrosion experts (e.g., W. Bogaerts [University of Leuven] and Digby Macdonald [UC Berkeley]) theorized that corrosion is a factor that could be affecting D3T2 specifically, and more generally, all vessels worldwide.
  - Postulated that there is a continuing source of H from corrosion of clad.
    - H migrates into the low alloy steel, is trapped by the flakes, via a mechanism similar to the well-known “Hydrogen blistering” experienced in vessels of the petroleum or chemical industries.
    - Macdonald speculated that this process could account for the apparent increase in number and size of flaws and could affect all vessels.
      - Recommended that all vessels should be inspected.
**EPRI Solicited Expert Opinion**

- MRP requested GE Global Research (Dr. Peter Andresen) to evaluate Macdonald/Bogaerts theory
  - “The H permeation data overwhelmingly and conclusively show that H2 in gases, dissolved in water, or present in metals can readily dissociate and permeate through iron and nickel alloys.

- Since the outside surface of the RPV permits hydrogen to escape, and because the ~6 mm thick stainless steel cladding is a larger barrier to H permeation than 200 mm of pressure vessel steel, it is estimated that the maximum steady state pressure in any defect in the RPV is 0.07atm ~93% vacuum, far below even atmospheric pressure.”

- Under steady state conditions at operating temperatures, a buildup of hydrogen pressure with time is not possible based on the experimental evidence

- Conclusions were limited to steady state conditions; analyses by SCK-CEN have addressed concern of pressure buildup during rapid cooldown

**Approximate representation of the H fugacity in the metal vs. through-wall thickness based on the H diffusivity through 6 mm of stainless steel cladding and 200 mm of low alloy steel, and out into the surrounding atmosphere**

*Note: Fugacity relates to chemical activity.*
Current Status of D3T2

- FANC provided clarification regarding 2014 flaw population:
  - The increase in size and number was due to increased sensitivity of the UT method used in 2014, compared to the method used in 2012
  - The flaws detected in 2012 did not evolve between 2012 and 2014
  - Flaws are not actually larger; UT method used in 2014 caused multiple flaws to be detected as single, larger flaws

- On November 17, 2015, FANC authorized restart of Doel 3 & Tihange 2

  - “The FANC concludes that Electrabel has demonstrated that the hydrogen flakes in the walls of the reactor vessels do not pose an unacceptable safety risk for the reactors. Electrabel is therefore authorized to restart and operate the Doel 3 and Tihange 2 reactors until they are permanently decommissioned. By law this is planned on 1 October 2022 for Doel 3 and on 1 February 2023 for Tihange 2.”
2013 MRP and PWROG Response to Doel 3 OE

- Industry assessed the implications for forged-ring PWRs
  - PWROG evaluated pre-service UT exam methods and determined that UT conducted during vessel fabrication was capable of detecting a Q-L flaw condition like Doel (WCAP-17786-NP Rev 1)
  - All U.S. forged-ring vessels reviewed pre-service UT records; no flaking found
  - In 2013 MRP performed PFM analysis (MRP-367) and damage mechanics analysis (MRP-388) for the bounding U.S. forged-ring PWR vessel
    - Safety targets are met (risk < 10^{-6} /yr) even if a Doel-like condition exists
    - Flaw population and distribution for this analysis was conservative representation of 2012 Doel flaws; 2014 UT has not yet been performed
  - MRP-367 was provided to NRC for information in early 2014
NRC Response to Q-L Flaw RPV Issue

In November 2015 NRC released a LIC-504 evaluation, “Technical Assessment of Potential Quasi-laminar Indications in Reactor Pressure Vessel Forgings” dated September 8, 2015 (ML15282A218)

– “Even if quasi-laminar indications similar to those discovered at Doel 3 and Tihange 2 existed in U.S. plants, the indications are not expected to significantly affect reactor pressure vessel integrity under normal or accident conditions. The basis for this conclusion is the industry analysis described in [MRP-367]…”

– This conclusion was based on a high level review of MRP-367
Beznau 1 Reports Quasi-Laminar (Q-L) Indications

- In July 2015, flaw indications were reported at Beznau-1, “similar to D3T2”
  - 2-loop Westinghouse design forged-ring PWR (ASME III 1965) fabricated by SFAC (Société des Forges et Ateliers du Creusot), material similar to SA-508 Cl3
  - First forged-ring vessel not fabricated at RDM to report a Doel-like condition
- Significantly fewer and smaller flaws than D3T2
- In a May 3, 2016 media release, AXPO (Licensee) attributed indications to aluminum oxide inclusions, not flaking
- Swiss regulator ENSI requiring safety case before restart will be permitted
Issues Related to Macrosegregation
(Flamanville 3 EPR Head issues)
Flamanville EPR (FA3) Issues

- EDF is constructing a new EPR (European Pressurized Reactor) at Flamanville
- In 2015, an issue was identified regarding toughness of the FA3 EPR closure head and bottom head
- Detailed information: CODEP-DEP-2015-037971, “Analysis of the procedure proposed by AREVA to prove adequate toughness of the domes of the Flamanville 3 EPR reactor pressure vessel (RPV) lower head and closure head”
- The FA3 EPR closure and bottom heads were fabricated by Creusot Forge
- In order to demonstrate compliance with the French toughness regulation, 2 other heads fabricated by Creusot Forge using same method as FA3 heads were destructively tested
  - Impact tests averaged 52J, below the minimum 60 J
- Carbon measurements at surface revealed large area of positive macrosegregation (high carbon)
- AREVA is conducting further tests to demonstrate the acceptability of the FA3 head
Issues Related to the Initial Fracture Toughness of RPV Steels

(Branch Technical Position 5-3 Issue)
Background (1/2)

- 10CFR50 Appendix G requires that vessel integrity evaluations be based on $RT_{NDT}$
- $RT_{NDT}$ is defined in ASME Section III NB-2331 and is derived from results of two tests: Pellini dropweight tests and Charpy V-notch tests of specimens oriented in transverse direction (T-L)
- Until ASME III introduced the concept of $RT_{NDT}$ in 1972, the Code required testing of either dropweight specimens or Charpy V-notch specimens oriented in the longitudinal direction (L-T)
- Therefore, many vessels fabricated prior to the effective date of the Code change (January 1, 1973) may not have all the test data – and the correct orientation – to determine $RT_{NDT}$ per ASME III
  - Must use alternative method to determine $RT_{NDT}$, using available data
Background (2/2)

- NUREG 0800 Branch Technical Position 5-3 provides methods for older (pre-1973) plants to estimate RT\textsubscript{NDT} and Upper Shelf Energy (USE) in cases where the plant does not have drop weight (T\textsubscript{NDT}) and/or weak-direction Charpy V-notch (CVN) test data
  - NRC basis document for methods is unknown
- In 2014 a vendor informed NRC that some methods in BTP 5-3 may be potentially non-conservative (ML14038A265)
- MRP/BWRVIP and PWROG conducted projects to address potential non-conservatisms in BTP
  - MRP/BWRVIP project evaluated the conservatism of the BTP methods and assessed safety significance of nonconservatisms
  - PWROG’s “Material-Orientation Toughness Assessment” (MOTA) showed existing P-T curve methods for PWRs have available conservative margin to offset BTP in most cases


MRP/BWRVIP Evaluation of BTP 5-3

- An expanded database of RPV steel toughness data was assembled, against which the BTP methods were assessed
- Evaluation found some BTP methods are potentially non-conservative, depending on product form
- Performed probabilistic fracture mechanics (PFM) analyses to assess risk of PWRs continuing to use existing BTP through 60 years
  - Risk to vessel integrity is negligible; no safety benefit to be gained from revising vessel P-T limits to address BTP nonconservativeness
  - Because of the bug in FAVOR discovered in 2014, which affects analysis of BWR vessels, no PFM analyses were performed for BWRs
    - Few BWRs use BTP 5-3 for shell toughness; use GE procedure
MRP/BWRVIP Final Report

- MRP-401/BWRVIP-287 was published in September 2015
  - Copy was provided to NRC in September 2015, as requested
- PWROG “MOTA” report (PWROG-15003-NP, Rev. 0) was also provided to NRC
GE Procedure for Estimating Initial $RT_{NDT}$

- GE developed an alternate methodology for determination of Initial $RT_{NDT}$ for materials in BWR vessels, similar to BTP 5-3 Y1006A006, Rev. 1 “Methods for Establishing Initial Reference Temperature ($RT_{NDT}$) for Vessel Steels for Certain Plants”
- Basis documented in 1994 BWR Owner’s Group report (NEDC-32399P), which was reviewed & approved by NRC
- 9 PWRs use GE procedure to estimate Initial $RT_{NDT}$ of nozzles
- When BTP 5-3 issue was first raised, NRC said its review would also include the adequacy of the GE procedure
- In 01/2016 public meeting, NRC expressed its belief that this procedure is potentially nonconservative
Update on NRC Activities Related to RPV Integrity (3/3)
May 10, 2016

- BTP 5-3
  - NRC staff’s evaluation and the industry’s reports indicated that no PWRs need to update their P-T limits and PTS evaluations for 60 years of operations
  - NRC currently performing risk-informed PFM analyses to determine whether PWRs or BWRs need to update P-T limits for 80 years
  - BTP 5-3 will probably be revised.
    - Change driven by PTS.
    - Δ Risk approach not allowed for PTS, plants exceeding PTS screening criteria must use 50.61a.
    - Plants would determine new $RT_{PTS}$ and use 50.61a if necessary (or pursue other options such as adjusting operations, obtaining additional information, etc.)
Present Status of BTP 5-3 Issue

- This is an ongoing and developing issue on which NRC and industry continue to engage
- Unknown if BTP 5-3 will be revised, and if it is, what requirements would be promulgated for implementing revised guidance
  - Because Initial RT\textsubscript{NDT} is a fundamental input and starting point for all RPV integrity analyses, changing the way Initial RT\textsubscript{NDT} is calculated would potentially cause a significant burden on all plants that have used BTP 5-3 (a majority of the fleet)
Remedial Options for RPV Embrittlement
Remedial Options for RPV Embrittlement Management

- Regulations such as the PTS Rule (10CFR50.61) and 10CFR50 Appendix G essentially establish a limit on allowable RPV embrittlement.

- Acceptable compensatory actions for addressing RPV embrittlement specifically mentioned in regulations are:
  - Neutron flux reduction
    - Low leakage cores
    - Part length shield assemblies (PLSAs) made of stainless steel to selectively reduce flux at the vessel welds; hafnium rods
  - Plant-specific analyses, and
  - Annealing
Vessel Annealing

- Thermal annealing is the only option that can, to some degree, recover irradiated beltline region transition temperature shift and recover upper shelf energy properties lost during radiation exposure and extend RPV service life.

- Beltline region is heated to 650 to 900°F (343 to 482°C).

- Degree of mechanical property recovery is function of:
  - difference between the irradiation temperature and thermal anneal temperature
  - time of annealing
  - material chemistry
  - degree of pre-existing irradiation damage
Irradiation Damage / Annealing Recovery

- Selection of anneal temperature requires some trade-offs
  - Higher temperatures (and longer annealing times) produce greater recovery of fracture toughness properties
  - Higher temperatures also pose greater engineering challenges to assure that the annealing operation does not distort or damage the vessel, supports, primary coolant piping, pipe supports, adjacent concrete, insulation, etc.
Vessel Annealing – Wet Anneal

- Two basic types of anneals
  - Wet anneal
  - Dry anneal

- Wet anneal is performed at temp <650°F (<343°C); reactor coolant water is generally heated by the RCPs
  - Wet annealing is not as complicated from an engineering viewpoint because primary water temperature is controlled by pump heat up to the vessel design temperature of 650°F (343°C)
  - But because temperature differential between normal operating temp and 650°F annealing temp is not high enough to obtain substantial mechanical property recovery, wet anneal is only partially effective and not a solution for PWRs
Dry Annealing

- Dry anneals are performed at higher temperatures than wet anneals
  - Use air as the heating medium and radiant heat
  - 850°F (454°C) is regarded as an optimum “dry” annealing temperature

- Dry annealing requires removal of core internal structures and primary water so that a radiant heating source can be inserted near the vessel wall to locally heat the embrittled beltline region
  - Engineering difficulties of dry anneal process are quite complex and may need plant-specific evaluations to assure that other portions of the plant (concrete, for example) are not harmed by the high annealing temperatures

- ~15 dry thermal anneals of VVER-440 vessels have been successfully conducted in the former Soviet Union (but these are almost all unclad, which poses much less dose issue than clad vessels would have)
  - US utility and NRC staff witnessed some of these annealing events
Marble Hill Demonstration Project

In 1990’s a joint DOE/industry-sponsored Annealing Demonstration Project (ADP) was conducted at the Marble Hill facility (a partially completed W plant) to demonstrate feasibility

- Nozzle-supported four loop Westinghouse design vessel -- canceled plant (unirradiated vessel)
- Indirect gas-fired heating method
- Annealing temperature - 800 - 900°F (427 - 482°C); Annealing hold time - 168 hours
- EPRI released a Marble Hill report (EPRI TR-104934) and a final NRC report (NUREG/CR-6552) was also later published

Demonstration successful from all aspects
- Excellent control of heat exchanger during both heatup and cooldown
- No measurable vessel “distortion” after anneal
Midland Demonstration Project

- Skirt-supported Babcock & Wilcox-design vessel
- Electric resistance heating method (Russian technology and experience)
- Project approximately 50 percent complete when DOE funding eliminated
- Electric resistance heater fabricated and tested
- Demonstration never completed
NRC Regulations/Guidance

- Annealing Rule in 10 CFR Part 50.66
  - Permits thermal annealing of LWRs; Requires a plan for conducting the thermal annealing be submitted at least three years before fracture toughness criteria are exceeded

- Regulatory Guide 1.162 - Annealing requirements and reporting
  - Describes the format and content of an acceptable Thermal Annealing Report (TAR)

- Both reference NUREG/CR-6327
  - Predictive model for annealing recovery utilizing microhardness and CVN data to cover a broad range of annealing conditions
  - Model incorporates annealing time and temperature and neutron irradiation dose rate (flux)
Industry Standards

  - Provides guidance for supplemental surveillance program to monitor reirradiation embrittlement
  - Cautions regarding phosphorus segregation to grain boundaries in ferritic steels during thermal annealing sequences, potentially leading to non-hardening intergranular fracture

- ASME Code Case N-557, “In-Place Dry Annealing of a PWR Nuclear Reactor Vessel (Section XI, Division 1)”
  - Provides Code guidance for assuring design conformance after performing a thermal anneal heat treatment
    - Limits magnitude of thermally induced stresses in nozzle region
    - Effectively limits the maximum temperature of annealing to 505°C (940°F)
    - Passed in 1995 in anticipation of a Palisades NPP thermal anneal
  - Technical basis published by EPRI in TR-106967
Key Annealing Issues as Related to Long Term Operation

- Based on NRC Regulations and guidance – evidence of dose rate on annealing recovery at low annealing temperatures (less than 427°C)

- Based on guidance in ASTM E 509
  - Re-embrittlement rate and surveillance during extended life (including any effect of dose rate)
  - Potential enhancement of P segregation and intergranular fracture

- Based on ASME Code Case – must minimize thermally induced stresses in nozzle region, which effectively limits maximum temperature of annealing to 505°C (940°F)
Application of Master Curve Technology for RPV Integrity Evaluation
Master Curve (MC)

- Transitioning from the current RT\textsubscript{NDT}-based approach for RPV integrity evaluation to a Master Curve-based approach is expected to benefit RPV operation through extended life by reducing the margins typically applied to account for uncertainties
  - The RT\textsubscript{NDT}-based approach has redundant conservatisms because RT\textsubscript{NDT} is only a representation of fracture toughness, whereas the Master Curve approach uses fracture toughness directly
  - Master Curve was developed by Kim Wallin at VTT and describes the characteristic toughness-temperature curve of all ferritic steels
    - The only difference between ferritic steels is the absolute position of the Master Curve with respect to temperature

\[ K_{JC} = 30 + 70 \exp(0.019(T - T_0)) \]

\( T_0 \) is the temperature at which the median toughness of a steel is 100 MPa√m
Master Curve – ASME Code Action to Date

  - Permits use of fracture toughness test data to establish a fracture toughness-based reference temperature $RT_{T0}$, for pressure-retaining materials (other than bolting)
  - Defines a new reference temperature:
    - $RT_{T0} = T_0 + 35°F (19°C)$
    - MC may be used as an alternative indexing reference temperature to $RT_{NDT}$ for the $K_{lc}$ and $K_{la}$ toughness curves in Appendices A and G

- 2013 - provisions of Code Case N-629 incorporated in Section XI, Article G-2000

- Code Case N-830 (2014) allowed a material-specific MC fracture toughness curve (i.e., by E1921) as an alternative fracture toughness curve, $K_{lc}$, in Section XI, Appendices A and G (curve is 95% lower tolerance bound)
  - Applicable to base metal or weld materials in the irradiated or nonirradiated condition
Master Curve (MC) – Further ASME Code Efforts

- MRP (and others) are currently supporting development of Revision 1 to Code Case N-830, and its adoption in the ASME Code
  - Code Case N-830 Rev. 1 will adopt the complete Master Curve methodology with all of the associated toughness model links as an alternative to the current RT_{NDT} approach in the ASME Code, Section XI, initially in Appendices A and K, and later in Appendix G
- PWROG is considering a significant Master Curve testing program to support a transition to use of direct fracture toughness

- CC N-830 Rev 1 authors: Erickson, Kirk, Server, Stevens, Cipolla
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