Correlation of Pressurized Water Reactor Vessel Material Properties Variation with Neutron Fluence by Surveillance Program

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Outline

- Introduction
- Neutron Fluence Calculation
- Charpy V-notch Impact Test
  - Tensile Test
- Summary
Introduction

- **Important factors in ensuring safety in the nuclear industry**
  - Pressure vessel containing the reactor core
  - The primary coolant to resist fracture

- **The beltline region of the reactor pressure vessel**
  - Significant fast neutron bombardment.

- **Low alloy ferritic materials under certain conditions of irradiation**
  - Increase in hardness and tensile properties
  - Decrease in ductility and toughness.
Appendix G to Sec. III of the ASME B & PV Code

• **Reference nil-ductility transition temperature (RT\text{NDT})**
  – The drop weight nil-ductility transition temperature (T\text{NDT} per ASTM E-208)
  – The temperature 60°F less than the 50 ft-1b (or 35-mil lateral expansion) temperature

• **Reference stress intensity factor curve (K\text{IC} curve)**
  – Lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel.
  – Allowable stress intensity factors can be obtained for this material as a function of temperature, and hence allowable operating limits.
Radiation Embrittlement

• The radiation embrittlement changes in mechanical properties of a given reactor pressure vessel steel can be monitored by a reactor surveillance program.
  
  – Adjusted $RT_{NDT}$ (initial $RT_{NDT} + \Delta RT_{NDT}$) is used to index the material to the $K_{IC}$ curve and, in turn, to set operating limits for the nuclear power plant
  
  – $\Delta RT_{NDT}$: the average Charpy V-notch 30 ft-1b temperature due to irradiation
RPV Surveillance Program

- Six surveillance capsules were inserted in the reactor vessel prior to initial plant start-up.
  - Charpy V-notch, tensile, and 1/2T compact tension (1/2T-CT) specimens.
  - Dosimeter wires of pure copper, iron, nickel, and aluminum-0.15 weight percent cobalt wire.
  - Cadmium shielded dosimeters of neptunium ($\text{Np}^{237}$) and uranium ($\text{U}^{238}$).
RADIATION ANALYSIS AND NEUTRON DOSIMETRY

• Neutron environment, such as energy spectrum, and fluence
  – Interpreting the neutron-induced material property changes
  – Combination of rigorous analytical techniques and measurements obtained with passive radiometric monitors

• Relationship between the neutron environment at various positions and that experienced by the test specimens
  – Relating the changes observed in the test specimens to the present and future conditions of the reactor vessel
  – Solely from analysis
Discrete Ordinates Analysis

• The 2-D calculating model (R-Θ and R-Z geometry) with DORT code
  – Neutron fluence rate distribution for different azimuthal angle and height respectively.

• The 1-D calculations were by ANISN code
  – Neutron fluence rate in a cylindrical geometry.
# Radiometric Monitors

<table>
<thead>
<tr>
<th>Monitor Material</th>
<th>Reaction of Interest</th>
<th>Target Weight Fraction</th>
<th>Product Half Life</th>
<th>Fission Yield (%)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Iron wire</td>
<td>$^{54}\text{Fe} \ (n, p) \ ^{54}\text{Mn}$</td>
<td>0.058</td>
<td>312.1 d</td>
<td></td>
</tr>
<tr>
<td>Nickel wire</td>
<td>$^{58}\text{Ni} \ (n, p) \ ^{58}\text{Co}$</td>
<td>0.6829</td>
<td>70.83 d</td>
<td></td>
</tr>
<tr>
<td>Copper wire</td>
<td>$^{63}\text{Cu} \ (n,\alpha) \ ^{60}\text{Co}$</td>
<td>0.6917</td>
<td>5.271 y</td>
<td></td>
</tr>
<tr>
<td>Uranium-238$^{(a)}$ in $\text{U}_3\text{O}_8$</td>
<td>$^{238}\text{U}(n, f) \ ^{137}\text{Cs}$</td>
<td>1.0</td>
<td>30.05 y</td>
<td>6.05</td>
</tr>
<tr>
<td>Neptunium-237$^{(a)}$ in $\text{NpO}_2$</td>
<td>$^{237}\text{Np}(n, f) \ ^{137}\text{Cs}$</td>
<td>1.0</td>
<td>30.05 y</td>
<td>6.25</td>
</tr>
</tbody>
</table>

(a) Denotes that the monitor is cadmium-shielded
HPGe Gamma Detection System

Gamma Spectrometer
HPGe Detector
Neutron Spectrum Analysis Results

- **Neutron Transport Analysis Results**
  - The lead factor deduced from the ratio of neutron flux at the center of the surveillance capsule to that at the peak location on the reactor pressure vessel surface is 2.48.

- **Dosimetry Results**
  - Measurement of the integrated-effect that the time and energy dependent neutron flux has on the target material over the course of the irradiation period.

- **SAND-II Code**
  - Combining the results of neutron transport calculations with neutron dosimetry measurements to obtain an optimal neutron spectrum with assigned deviation.
## Comparison of Measurement & Calculation

<table>
<thead>
<tr>
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<th>Measured value</th>
<th>Calculated value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fast fluence rate (E &gt; 1 MeV)</td>
<td>$1.185\times10^{11}$&lt;sup&gt;(a)&lt;/sup&gt;</td>
<td>-</td>
</tr>
<tr>
<td>Fast fluence rate (E &gt; 1 MeV)</td>
<td>$1.220\times10^{11}$&lt;sup&gt;(b)&lt;/sup&gt;</td>
<td>-</td>
</tr>
<tr>
<td>Fast fluence rate (E &gt; 1 MeV)</td>
<td>$1.203\times10^{11}$&lt;sup&gt;(c)&lt;/sup&gt;</td>
<td>$1.125\times10^{11}$&lt;sup&gt;(e)&lt;/sup&gt;</td>
</tr>
<tr>
<td>Fast fluence (E &gt;1 MeV)</td>
<td>$8.407\times10^{19}$&lt;sup&gt;(d)&lt;/sup&gt;</td>
<td>-</td>
</tr>
</tbody>
</table>

- **a)** Based on SAILOR dosimetry file.
- **b)** Based on SAND-II calculation.
- **c)** An average of values derived from (a) and (b).
- **d)** The product of fast fluence rate ($1.203\times10^{11}$) and effective full power years (22.17).
- **e)** Obtained from neutron transport analysis result.
Tensile Test

• **ASTM Standard E8-09 and E21-09, and INER 70-NF-SOP-009**
  - Constant crosshead speed of 0.01 centimeters per minute
  - Specimen elongation: linear variable displacement transducer (LVDT) extensometer over a gage length of 1.00 inch.

• **Load-extension curve**
  - The yield load, ultimate load, fracture load, total elongation, and uniform elongation
  - The yield strength, ultimate strength, and fracture strength were calculated using the original cross-sectional area.
Intron 4505

- All pull rods, grips, and pins were made of Inconel 718.
- Elevated test temperatures were obtained with a three-zone electric resistance split-tube furnace with a 5-inch hot zone.
The strength of the specimens increases with the irradiation dose, evidencing that the specimens were hardened after being irradiated, whereas the relationship between ductility and irradiation dose is not clear.

The yield strength falls between 345~621MPa.
Charpy V-notch Impact Test

- **ASTM Standard E23-07a and INER 70-NF-SOP-010**
  - Percent shear: post-fracture photographs using the ratio-of-areas methods.
  - Lateral expansion: dial gage rig.

- **Yield stress** ($\sigma_Y$) **was calculated from the three-point bend formula.**
  - Flow stress: the average of the yield and maximum loads
Tinius-Olsen 74

- The maximum impact capacity is 359 Joules.
- Instrumented with the data acquisition system
Test Results  Charpy V-notch impact test

- Base metal - longitudinal
  - NL
  - USE = 138 J (54°C)

- Weld
  - NW
  - USE = 110 J (49°C)

- Base metal - transverse
  - NT
  - USE = 75 J (59°C)

- Heat-affected Zone
  - NH
  - USE = 133 J (68°C)

USE values are all > 68 J
ΔRT\text{NDT} Evaluation Models

- **NRC R.G. 1.99 rev.1 Model**
  \[ \Delta RT_{NDT}(^\circ F) = [40 + 1000(\%Cu - 0.08) + 5000(\%P - 0.008)] \left( \frac{f}{10^{19}} \right)^{0.5} \]

- **NRC R.G. 1.99 rev.2 Model**
  \[ \Delta RT_{NDT}(^\circ F) = CF \times (f/10^{19})^{[0.28-0.10 \log(f/10^{19})]} \]

- **ASTM E900-02 Model**
  \[ \Delta RT_{NDT}(^\circ F) = SMD + CRP \]
  - \( SMD = 6.70 \times 10^{-18} e^{\left( \frac{20730}{T_{c}+460} \right)} f^{0.5076} \)
  - \( CRP = B[1 + 2.106Ni^{1.173}]F(Cu)G(f) \)
  - \[ F(Cu) = \begin{cases} \frac{0}{(Cu-0.072)^{0.577}} & , Cu > 0.072\text{wt}\% \\ 0 & , Cu \leq 0.072\text{wt}\% \end{cases} \]
  \( = \Delta RT_{NDT} \)
Comparison of $\Delta RT_{NDT}$ Evaluation Models

Base Metal
M=9°C

Weld
M=16°C
Summary

• **Tensile test**
  – The yield strength of the surveillance specimens meets the requirement of 10CFR App. G (345~621MPa).

• **Charpy V-notch impact test**
  – The upper shelf energy of the surveillance specimens meets the requirement of 10CFR App. G for plant operation after 55EFPY (22.17EFPY*2.48).

• **ΔRT\textsubscript{NDT} Evaluation**
  – From the results of surveillance program over the years, ΔRT\textsubscript{NDT} meets the evaluation of NRC R.G. 1.99 rev.2 (1988, 2014 reviewed).
Thanks for your attention.