

# Resolution Summary of Acceptance Review Results (RV Integrity at Weld Location)

- Background
- Resolution Summary
- Future Plan

# Background

- **16<sup>th</sup> PARM (June 18 ~ 19, 2014)**
  - DCD Section 5.3
    - Section 5.3.2 P-T Limits, PTS, and Charpy Upper Shelf Energy Data and Analyses
    - Omission of evaluation at the reactor vessel welds

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# Resolution Summary (1/11)

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# Resolution Summary (2/11)

- RV Beltline Welds

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## Resolution Summary (3/11)

- **RV Neutron Fluence Calculation**
  - Neutron Flux Calculation Methodology
    - RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence”, 2001
  - Evaluation of Neutron Flux Distributions
    - CCC-650/DOORS3.2, “One-, Two- and Three-Dimensional Discrete Neutron./Photon Transport Code System”, ORNL, 1998
  - Peak Fast Neutron Fluence
    - $9.5 \times 10^{19}$  n/cm<sup>2</sup> based on the 93% capacity factor for 60 years

# Resolution Summary (4/11)

- Input Data for Evaluation at Weld Location

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## Resolution Summary (5/11)

- DCD Section 5.3.2.1.1 Material Properties

Before	After	Remarks
Based on considerations of existing material property test data, an initial $RT_{NDT}$ for the reactor vessel material is $-23.3\text{ }^{\circ}\text{C}$ ( $-10\text{ }^{\circ}\text{F}$ ), and an initial $RT_{NDT}$ for the remaining material of the reactor coolant system is $-12.2\text{ }^{\circ}\text{C}$ ( $10\text{ }^{\circ}\text{F}$ ).	Based on considerations of existing material property test data, an initial $RT_{NDT}$ for the reactor vessel <b>forging</b> is $-23.3\text{ }^{\circ}\text{C}$ ( $-10\text{ }^{\circ}\text{F}$ ), and an initial $RT_{NDT}$ for the remaining material <b>including weld materials</b> of the reactor coolant system is $-12.2\text{ }^{\circ}\text{C}$ ( $10\text{ }^{\circ}\text{F}$ ).	Pages 5.3-17~18
Because there are no longitudinal seam welds and circumferential seam welds are designed to be away the center of the core belt, the forging material at the beltline is critical and is the only material that is evaluated.	<b>There</b> are no longitudinal seam welds and <b>two (2)</b> of circumferential seam welds are <b>located near the fringes of core beltline region</b> . It is <b>conservatively assumed that weld materials are subjected to the maximum neutron fluence of <math>9.5 \times 10^{19}\text{ n/cm}^2</math> for evaluating <math>RT_{NDT}</math></b> .	Page 5.3-18

## Resolution Summary (6/11)

- DCD Section 5.3.2.1.1 Material Properties (cont'd)

Before	After	Remarks
The RT <sub>NDT</sub> values at the end-of-life are expected to be 21.1 °C (70 °F) per NRC RG 1.99 based on the weight percent of residual elements in Subsection 5.2.3.1.	The RT <sub>NDT</sub> values (1/4T location) at the end-of-life are expected to be 20.2 °C (68.4 °F) for beltline forging and 47.2 °C (117 °F) for weld material per NRC RG 1.99 based on the weight percent of residual elements in Subsection 5.2.3.1.	Page 5.3-18



# Resolution Summary (7/11)

- **DCD Section 5.3.2.1.2 Determine of P-T Limitation Curves**

Before	After	Remarks
<p>The margin required to be added for uncertainties by NRC RG 1.99 is 16.8 °C (30.2 °F) for the vessel beltline base materials.</p>	<p><b>For the beltline forging, the margin required to be added for uncertainties by NRC RG 1.99 is 15.8 °C (28.4 °F).</b></p>	<p>Page 5.3-23</p>
<p>However, to cover unanticipated long-term aging phenomena of 27.8 °C (50 °F) for the vessel beltline base materials is used for additional conservatism. Once confirmatory data are available from the reactor vessel surveillance program, the margin can be reduced in subsequent shift calculations consistent with the NRC RG 1.99.</p>	<p>However, to cover unanticipated long-term aging phenomena of 27.8 °C (50 °F) for the vessel beltline base materials is used for additional conservatism. <b>In case of welding materials, no additional conservatism is applied in evaluating the margin as maximum neutron fluence is applied to the weld location.</b> Once confirmatory data are available from the reactor vessel surveillance program, the margin can be reduced in subsequent shift calculations consistent with the NRC RG 1.99.</p>	<p>Pages 5.3-23~24</p>

## Resolution Summary (8/11)

### ● DCD Section 5.3.2.3 Pressurized Thermal Shock

Before	After	Remarks
<p>The calculated <math>RT_{PTS}</math> is 21.2 °C (70.2 °F) for the vessel beltline base materials, which satisfies the screening criteria in 10 CFR 50.61(b)(2).</p> <p>This number has been calculated with the following assumptions:</p> <p>a. The maximum initial <math>RT_{NDT}</math> for the vessel beltline materials is -23.3 °C (-10 °F).</p> <p>b. The maximum integrated fast neutron flux exposure of the reactor vessel wall opposite the midplane of the core is less than <math>9.5 \times 10^{19}</math> n/cm<sup>2</sup>.</p>	<p>The PTS screening criterion is</p> <ul style="list-style-type: none"> <li>- 132.2 °C (270 °F) for plates, forgings, and axial weld materials</li> <li>- 148 °C (300 °F) for circumferential weld materials</li> </ul> <p>In calculating <math>RT_{PTS}</math> for the limiting beltline material, the following assumptions are applied:</p> <p>a. The limiting case is the weld material subjected to the maximum integrated fast neutron fluence of <math>9.5 \times 10^{19}</math> n/cm<sup>2</sup>.</p>	<p>Page 5.3-27</p>

## Resolution Summary (9/11)

### ● DCD Section 5.3.2.3 Pressurized Thermal Shock (cont'd)

Before	After	Remarks
<p>b. (continued) The adjustment in the reference temperature caused by irradiation (<math>\Delta RT_{PTS}</math>) is 16.8 °C (30.2 °F) for the vessel beltline base materials. This calculated value assumes a maximum copper content of 0.03 percent by weight, and a maximum nickel content of 1.00 percent by weight in the forgings.</p> <p>c. The margin required to be added for uncertainties per 10 CFR 50.61 is 16.8 °C (30.2 °F) for the vessel beltline base materials. However, a margin of 27.8 °C (50 °F) for the vessel beltline base materials is used for additional conservatism.</p>	<p>b. For the weld material, maximum copper content is 0.05 wt% and maximum nickel content is 0.10 wt%, and maximum initial <math>RT_{NDT}</math> is -12.2°C (10 °F).</p> <p>c. The adjustment in the reference temperature caused by irradiation (<math>\Delta RT_{PTS}</math>) is calculated to be 31.4 °C (56.6 °F). The margin required by 10CFR50.61 is 31.1 °C (56 °F) for the weld materials.</p> <p>The calculated <math>RT_{PTS}</math> is 50.6 °C(123 °F) which satisfies PTS screening criterion above.</p>	<p>Page 5.3-27</p>

# Resolution Summary (10/11)

- DCD Section 5.3.2.4 Upper-Shelf Energy

Before	After	Remarks
<p>Even with the limiting value of 102 Joules (75 ft-lbs) assumed for the initial USE, the end-of life (EOL) USE is estimated to be 69.4 Joules (51 ft-lbs), which is higher than the limiting EOL value of 68 Joules (50 ft-lbs) based on the maximum neutron fluence of <math>9.5 \times 10^{19}</math> n/cm<sup>2</sup>, and maximum copper content of 0.03 percent by weight for forging materials, as assumed in subsection 5.3.2.3. Previous data on the OPR1000 reactor vessel materials show that initial USE values are well above 200 joules (147 ft-lbs), and actual EOL values of USE are therefore expected to be much higher than 68 Joules (50 ft-lbs).</p>	<p>Even with the limiting value of 102 Joules (75 ft-lbs) assumed for the initial USE, the end-of life (EOL) USE is estimated to be 69.4 Joules (51 ft-lbs), which is higher than the limiting EOL value of 68 Joules (50 ft-lbs) based on the maximum neutron fluence of <math>9.5 \times 10^{19}</math> n/cm<sup>2</sup>, and maximum copper content of 0.03 percent by weight for forging materials <b>and 0.05 percent by weight for weld materials</b>, as assumed in subsection 5.3.2.3. Previous data <b>of the OPR1000 and Shin-Kori 3&amp;4 reactor vessel materials including weld metal</b> show that initial USE values are <b>mostly higher than</b> 200 joules (147 ft-lbs), and actual EOL values of USE are <b>therefore</b> expected to be much higher than 68 Joules (50 ft-lbs).</p>	<p>Page 5.3-28</p>

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# Resolution Summary (11/11)

- DCD Figure 5.3.7 Pressure Temperature Limit Curve (60 years)

Before	After	Remarks
	<p>Figure 5.3-7 Pressure Temperature Limit Curve (60 years)</p>	<p>Page 5.3-56</p>

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