Examination of ENDF/B-VIII.1-Based Nuclear Data for LWR Pressure Vessel Fluence and Dosimetry Applications

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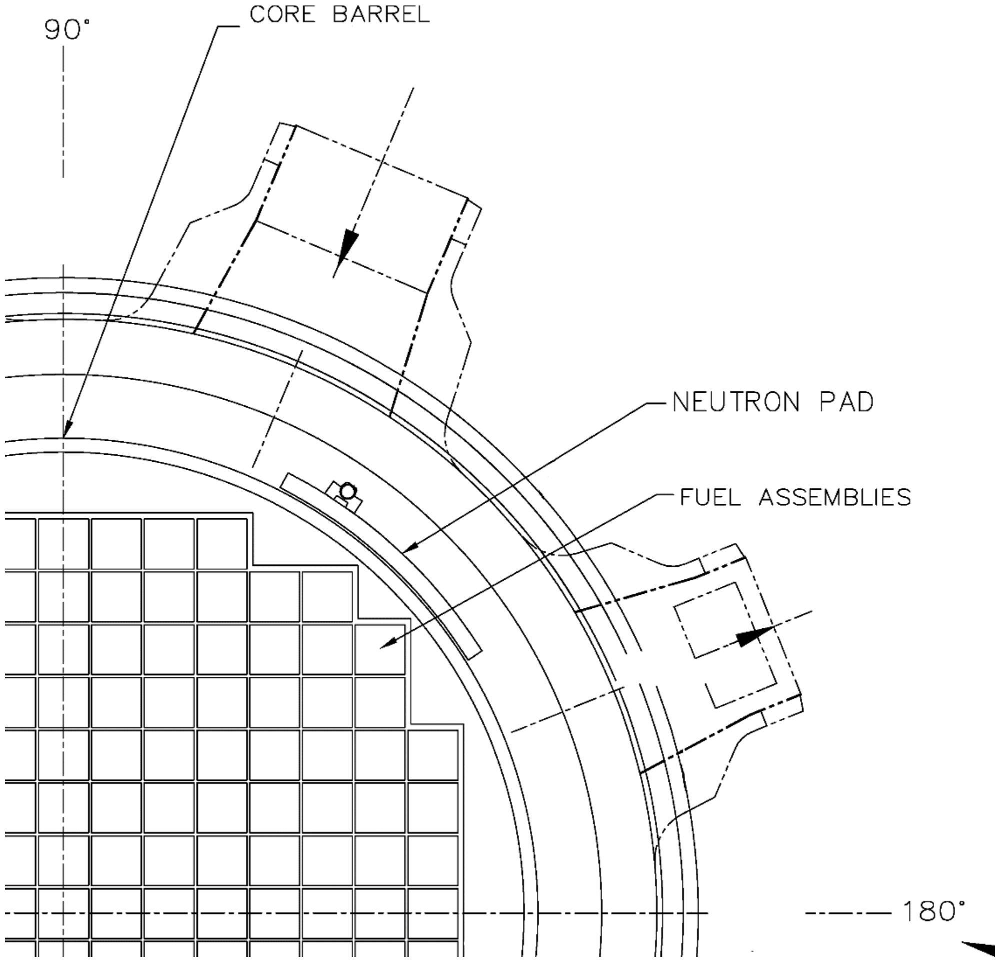
Overview

- Background on Fluence Calculations
- Nuclear Data Processing
- Data-Level Comparisons
- Dosimetry Comparisons
- Conclusions

Background on Fluence Calculations

- Most fission neutrons remain inside the core, supporting the self-sustaining chain reaction
- We're interested in the relatively few neutrons that leak outside of the core and impinge upon the reactor vessel



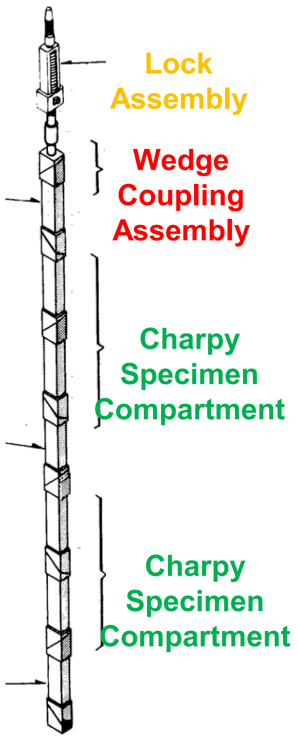
Background on Fluence Calculations

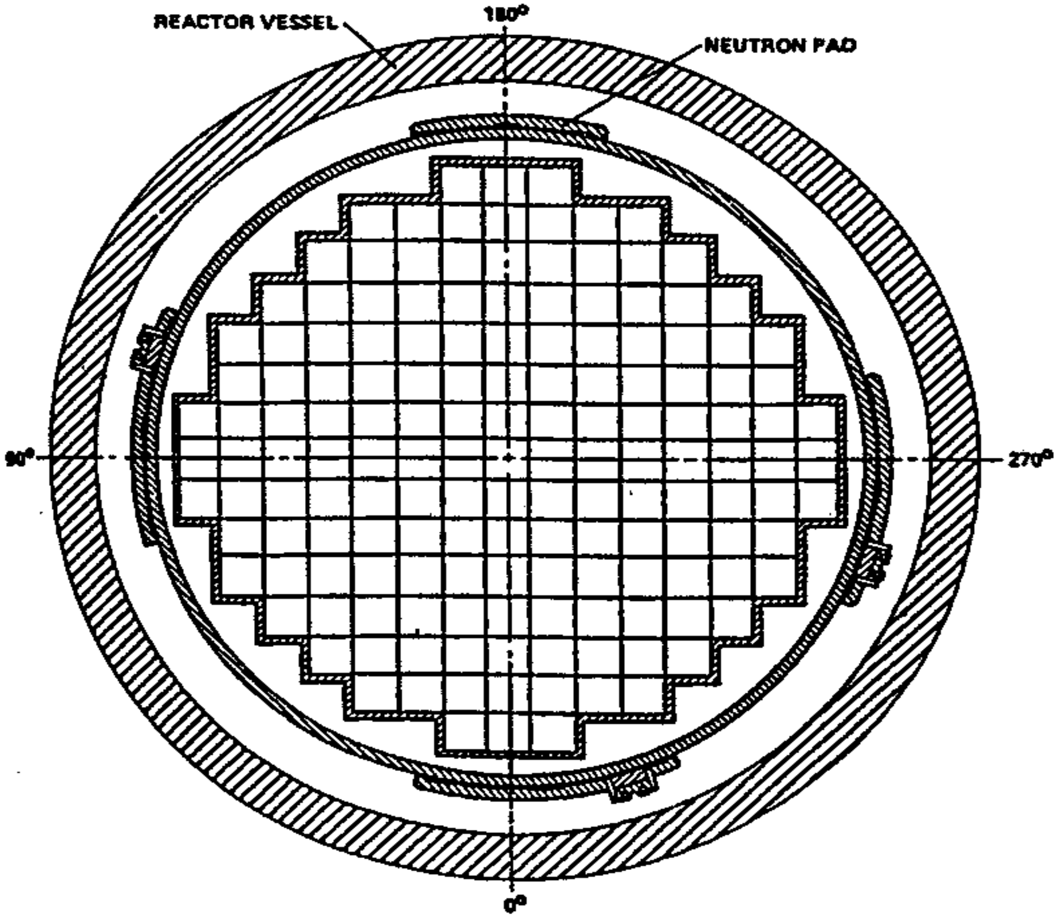
- The reactor pressure vessel and internals components are susceptible to irradiation-induced degradation of material properties

- These changes are manifested as:
  - Reduction in the toughness during ductile fracture
  - Tendency for brittle fracture to onset at increasing temperatures

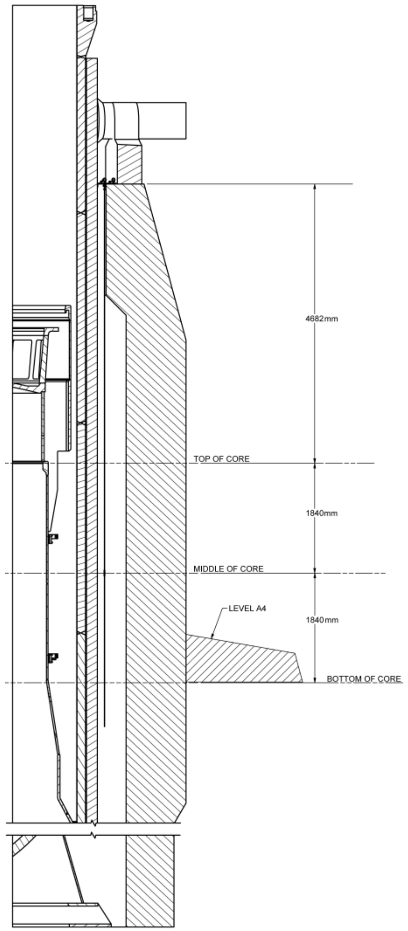
- All reactors are required to have a materials surveillance program to monitor changes in reactor vessel materials properties

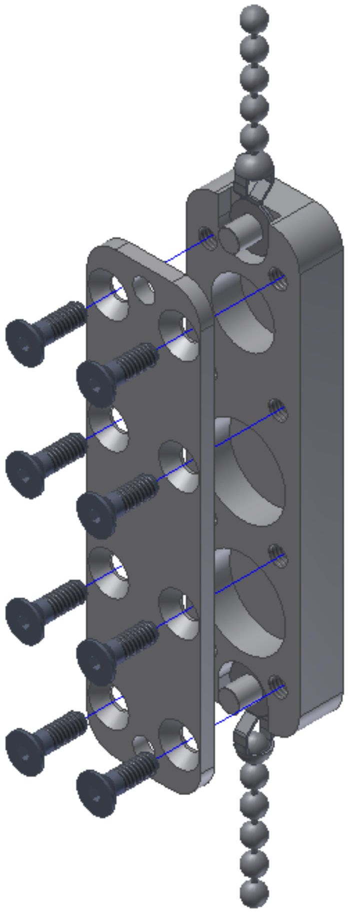
Background on Fluence Calculations





Background on Fluence Calculations





Background on Fluence Calculations

- Westinghouse fluence determination methods are based on BUGLE-96 transport cross section data and the SNLRML dosimetry cross section library.

  With each major release of ENDF, the updates have been examined.

  - ENDF/B-VIII.0 (2018) exhibited significant problems with the iron evaluations that yielded poor agreement with reactor dosimetry measurements and rendered the library unsuitable for RPV fluence applications.

Background on Fluence Calculations

- ENDF/B-VIII.1 was released in August 2024.
  - This release addressed the known problems with the iron evaluations.
- The IRDFF-II library was released in January 2020.
  - International consensus dosimetry reaction cross sections
- This work compares dosimetry analysis results for a 4-Loop PWR obtained with BUGLE-96 and SNLRML to results from an ENDF-B/VIII.1-based multigroup library and IRDFF-II.

Nuclear Data Processing

- Multi-group transport cross sections were processed with NJOY and TRANSX into 200n47g structure using ENDF/B-VIII.1 evaluations.

- Calculated spectra from a 200n47g solution of a representative 2D problem were used to collapse a 45n20g “broad” library.

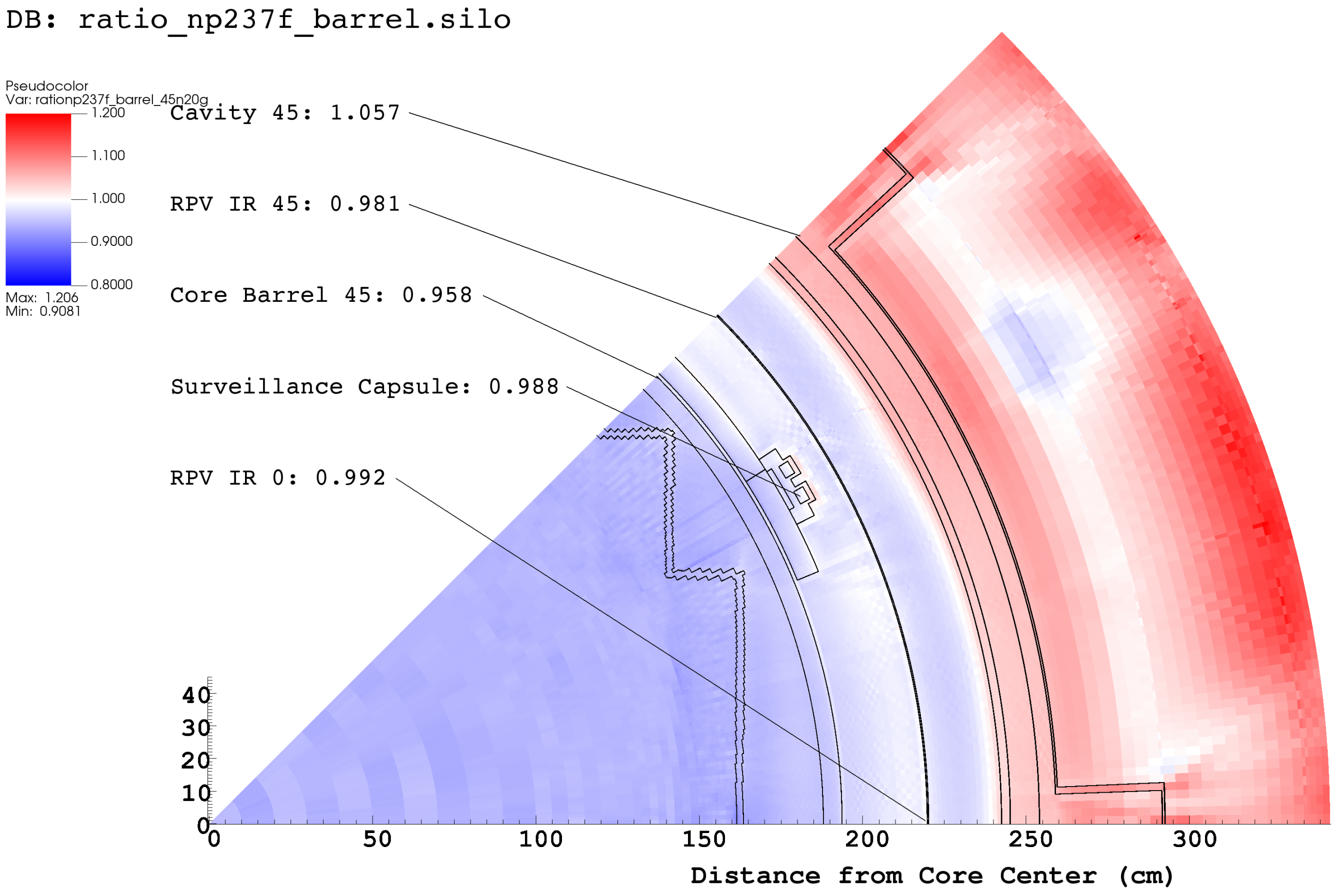
- IRDFF-II dosimetry library for dosimetry comparisons was similarly processed using NJOY: “fine” 200n xsec and “broad” 114n covariances

Data-Level Comparisons

- Changes in the transport cross sections were examined by calculating and comparing reaction rates for common dosimetry sensors
  - All comparisons were performed with SNLRML dosimetry cross sections
  - ("latest" / "current")

Reaction of Interest	Neutron Energy Response(a)
$^{63}Cu(n,\alpha)^{60}Co$	4.53-11.0 MeV
$^{46}Ti(n,p)^{46}Sc$	3.70-9.43 MeV
$^{54}Fe(n,p)^{54}Mn$	2.27-7.54 MeV
$^{58}Ni(n,p)^{58}Co$	1.98-7.51 MeV
$^{238}U(n,f)^{137}Cs$	1.44-6.69 MeV
$^{93}Nb(n,n')^{93m}Nb$	0.95-5.79 MeV
$^{237}Np(n,f)^{137}Cs$	0.68-5.61 MeV

Data-Level Comparisons



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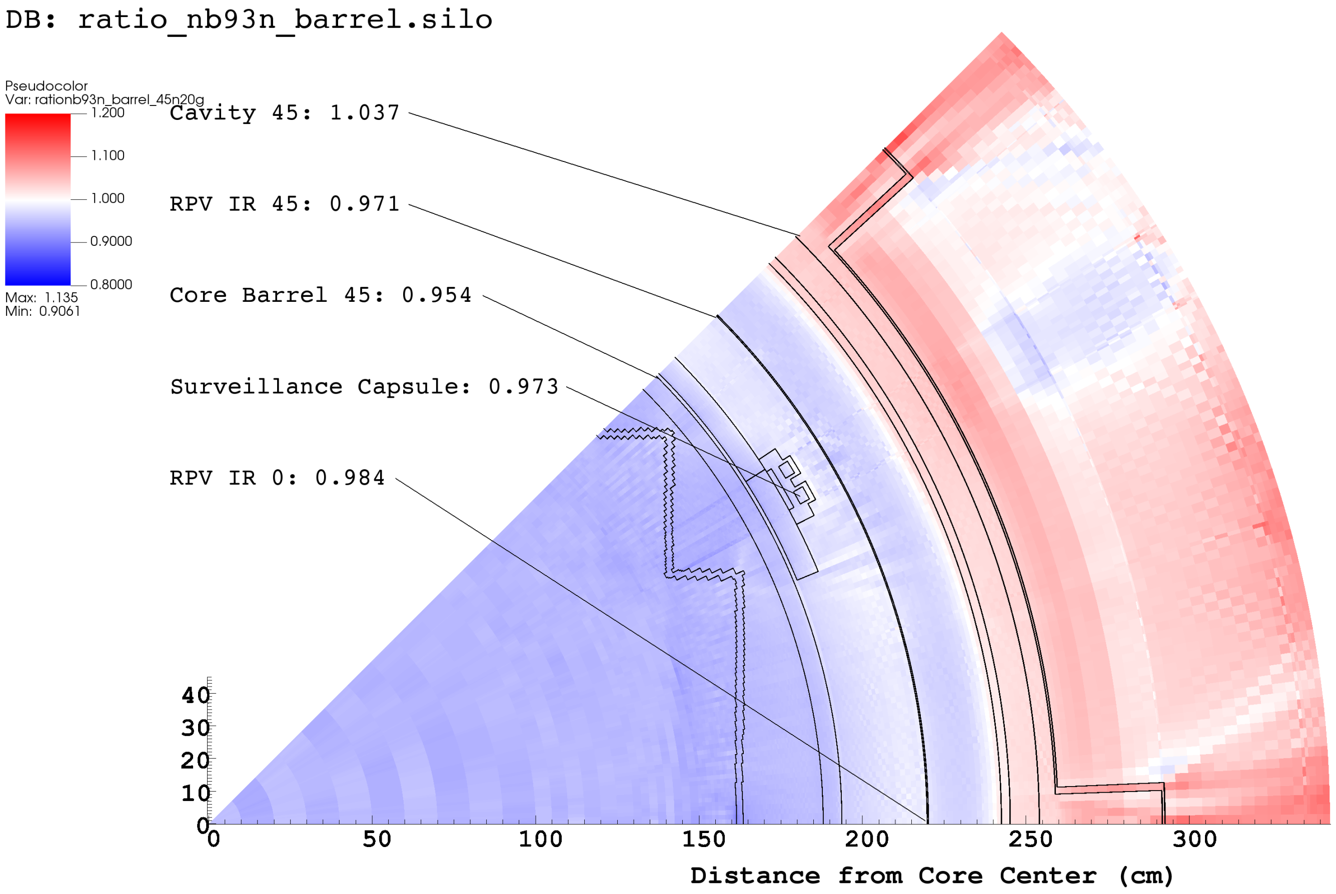
Data-Level Comparisons



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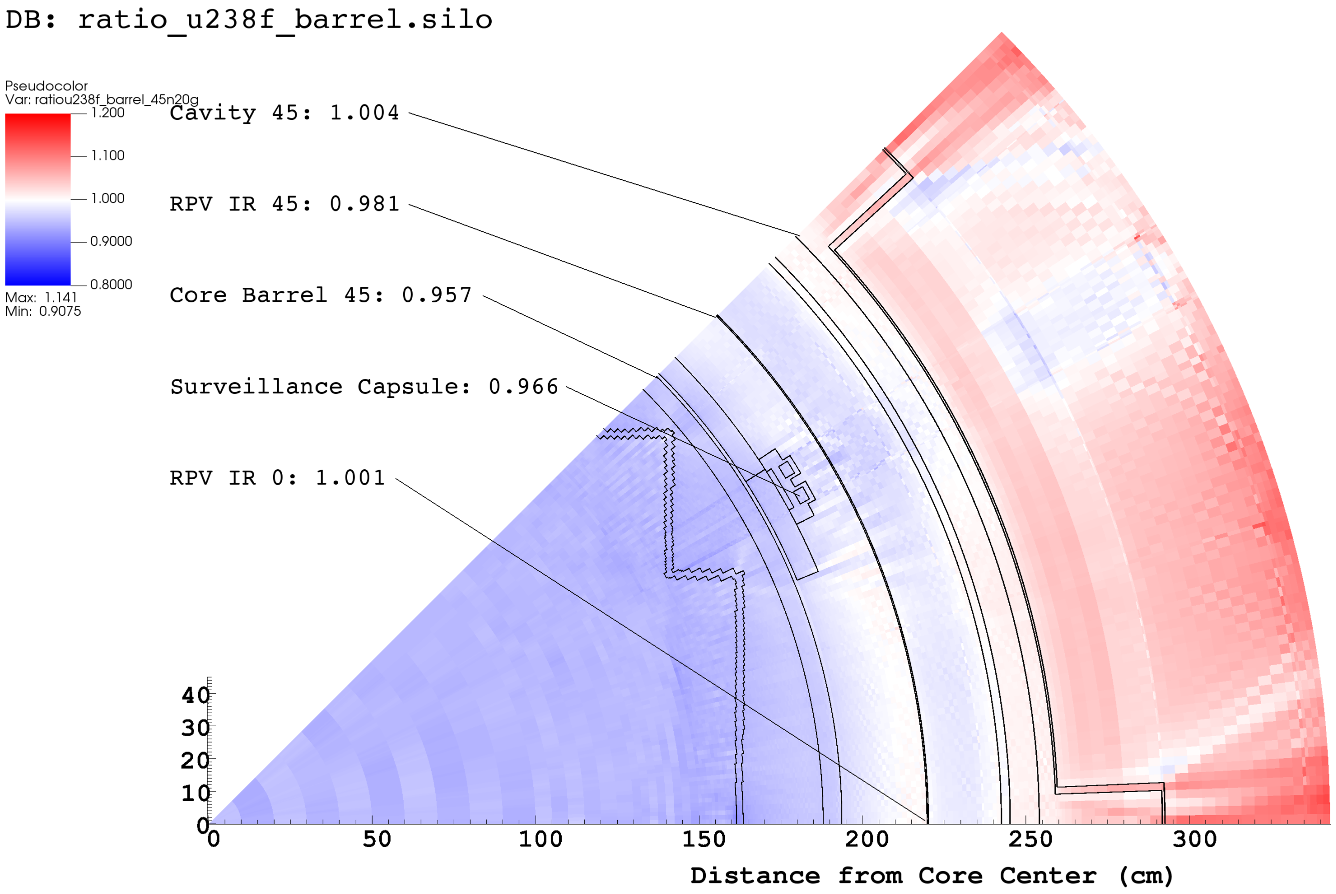

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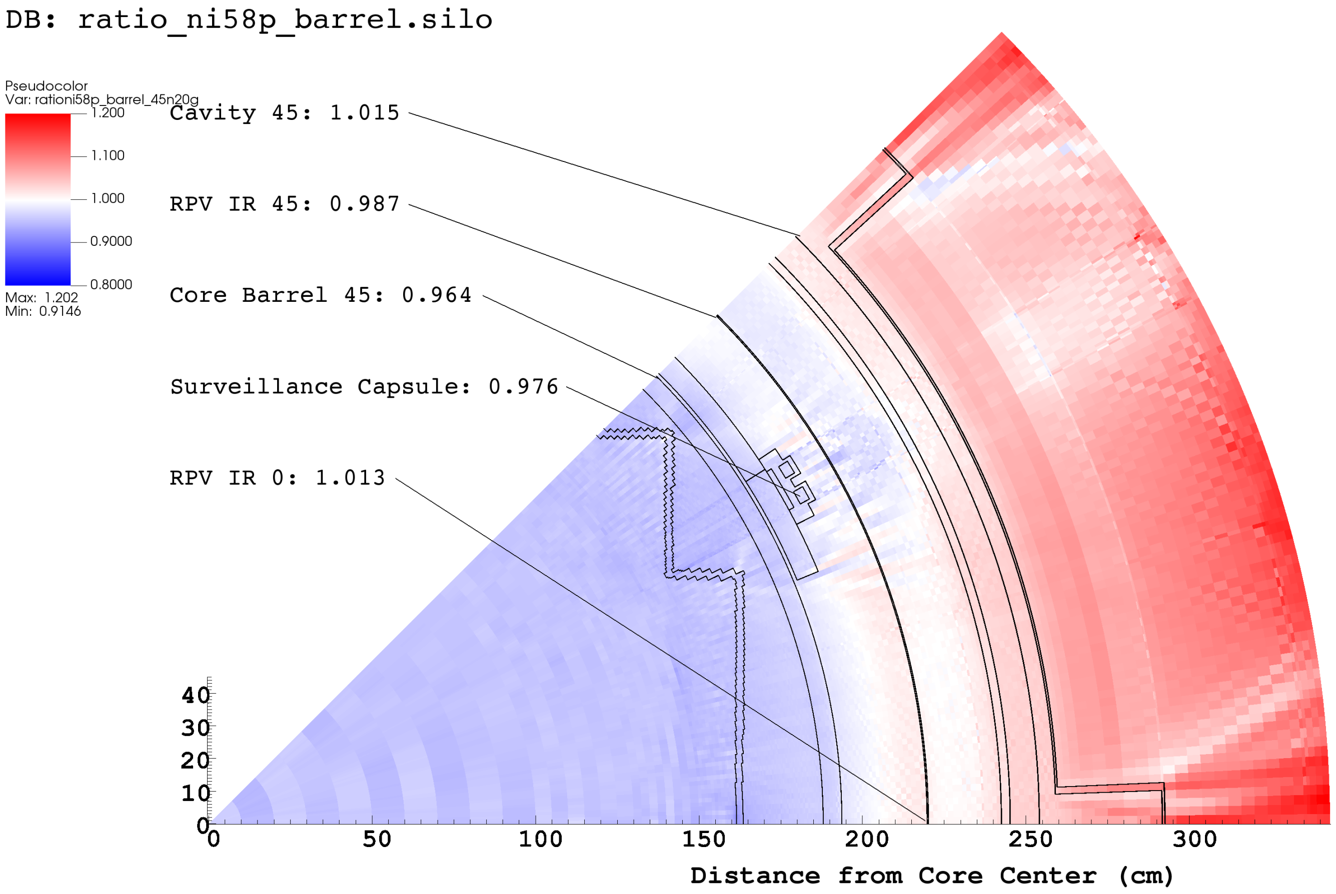

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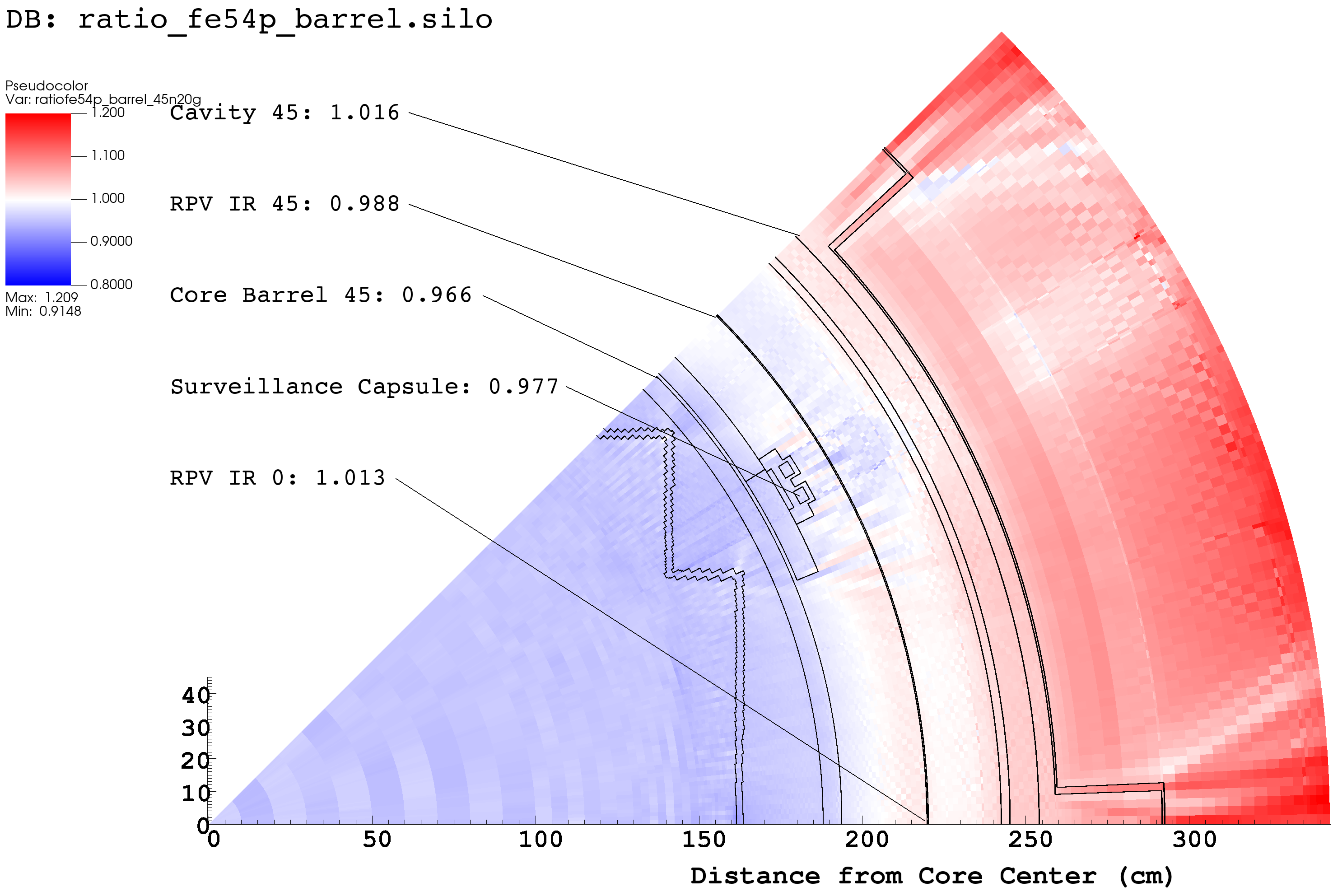

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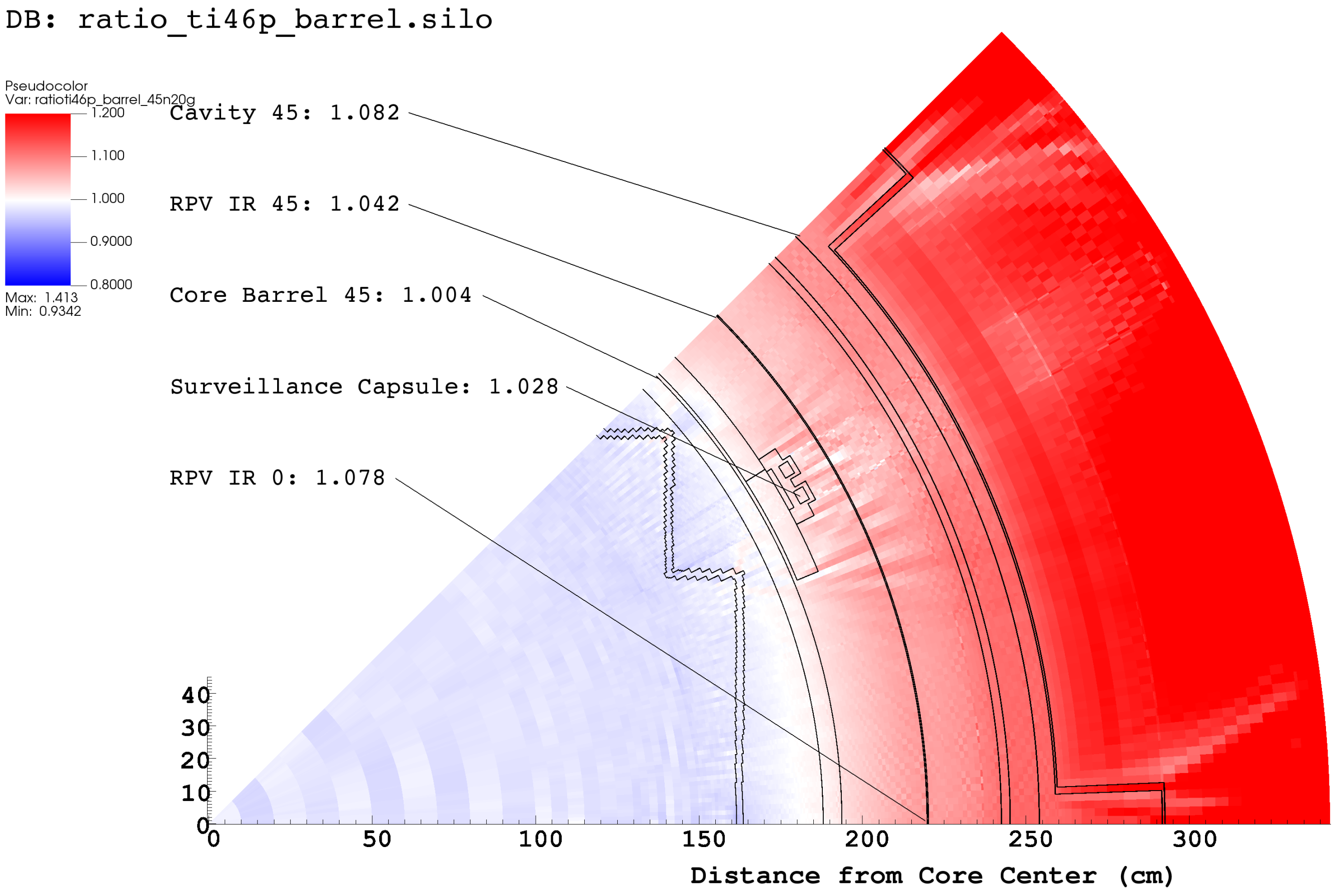

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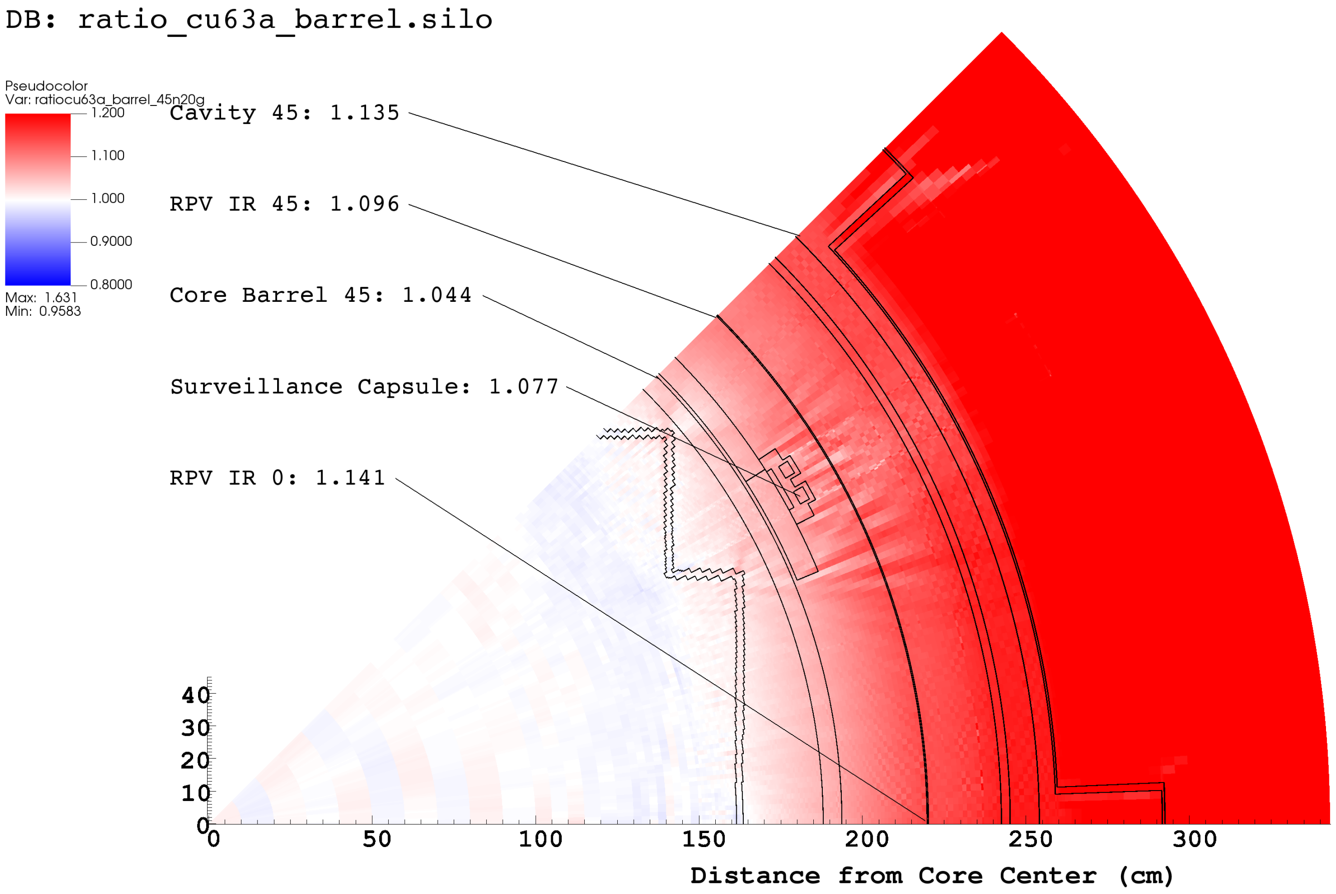
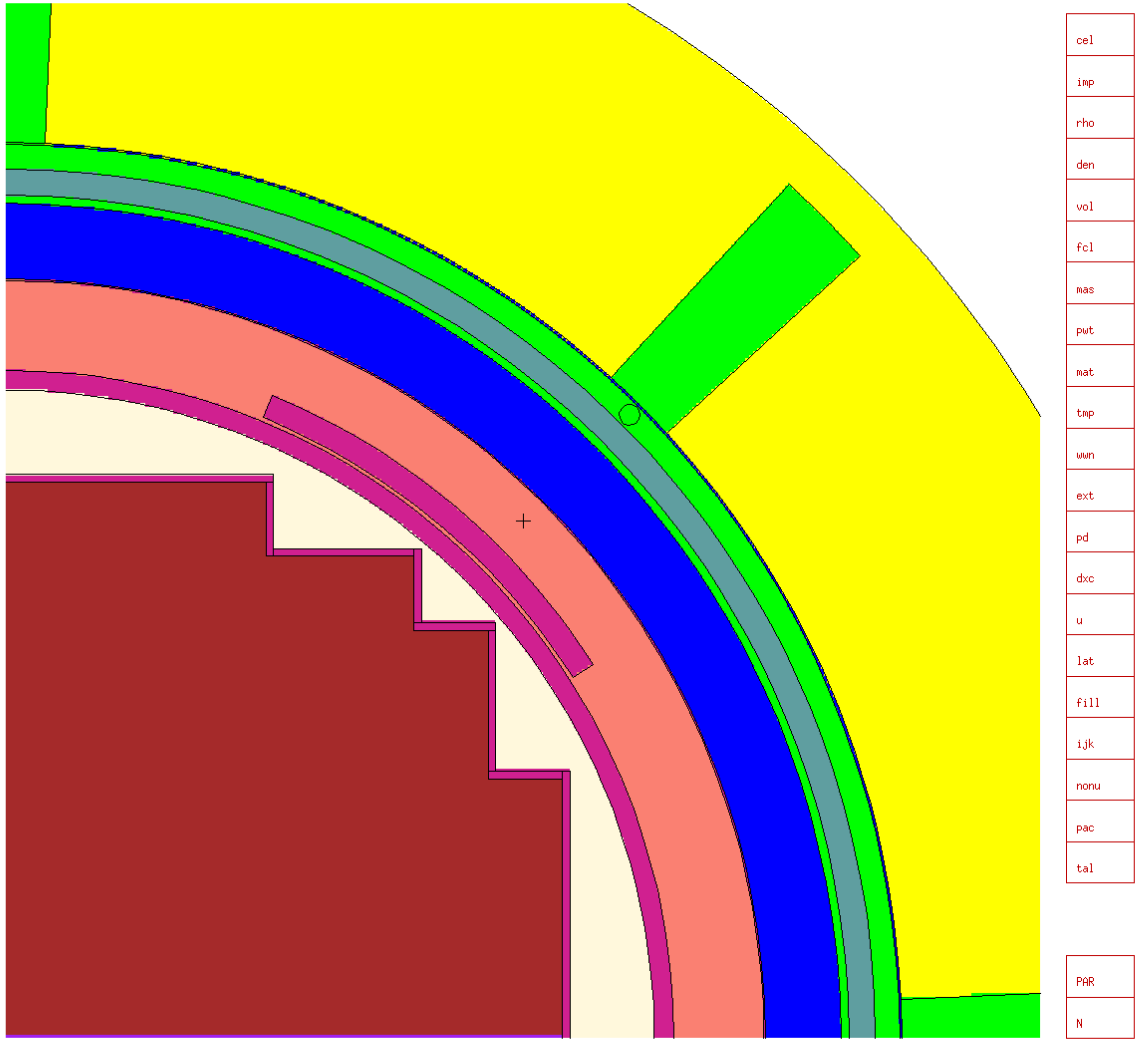


Data-Level Comparisons

- Monte Carlo calculations performed with ENDF/B-VI.8 and ENDF/B-VIII.1 transport cross sections and IRDFF-II dosimetry cross sections

- Reaction rates tallied at the midplane of a 4-Loop reactor model.



Data-Level Comparisons

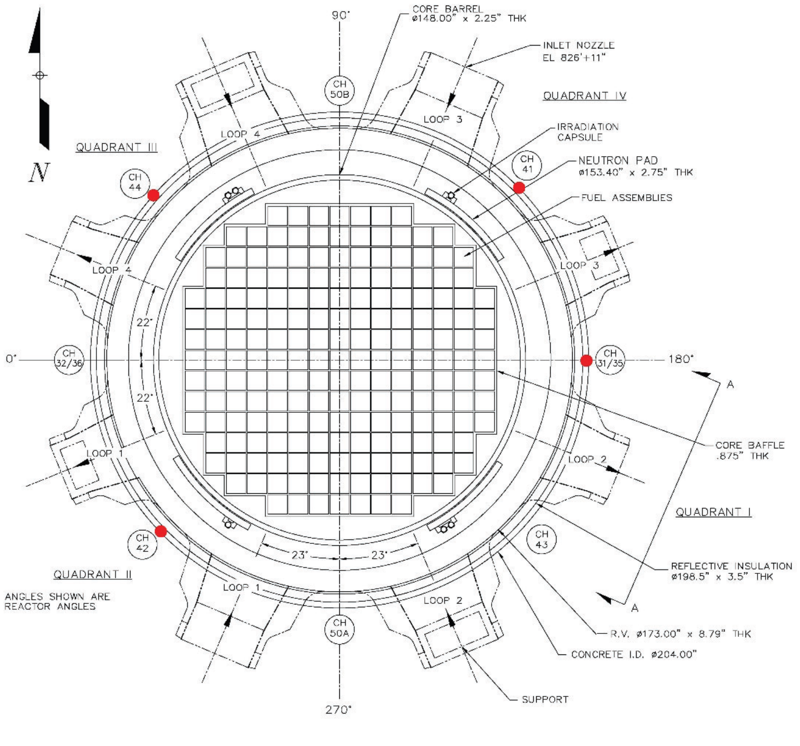
Reaction	ENDF/B-VI.8	ENDF/B-VIII.1
Cu-63 (n,α) Co-60	1.00	1.09
Ti-46 (n,p) Sc-46	1.00	1.04
Fe-54 (n,p) Mn-54	1.00	0.99
Ni-58 (n,p) Co-58	1.00	0.99
U-238 (n,f) Cs-137	1.00	0.99
Nb-93 (n,n) Nb-93m	1.00	0.96
Np-237 (n,f) Cs-137	1.00	0.97

Data-Level Comparisons

- Lower energy (< 4 MeV) neutrons exhibit performance that seems consistent with current transport data.
- High energy (> 4 MeV) neutron attenuation through water with ENDF/B-VIII.1 seems to result in higher reaction rates.
- This is contrary to what we see in our database of measurement data, where calculated reaction rates tend to (already) be higher than measurements.

Dosimetry Comparisons

- To assess the combined effects of new transport and dosimetry cross sections, EVND measurements from a 4-Loop PWR were re-evaluated.
  - Measurements were collected at core midplane, nozzle support, and RPV bottom head elevations.
  - Re-evaluation applies “broad” 45n20g transport cross sections based on ENDF/B-VIII.1 and IRDFF-II dosimetry cross sections.



PLAN VIEW AT CORE MIDPLANE (EL.815'+7.31")

Dosimetry Comparisons

	M/C Reaction Rate Ratios: E8.1, IRDFF-II (BUGLE-96, SNLRML)					
Reaction	**Core Mid. Θ = 0°**	**Core Mid. Θ = 45°**	**Nozzle Θ = 0°**	**Nozzle Θ = 45°**	**Bot. Head Θ = 0°**	**Bot. Head Θ = 45°**
Cu-63 (n,α) Co-60	0.73 (0.86)	0.82 (0.93)	0.51 (0.60)	0.47 (0.53)	1.05 (1.28)	0.87 (1.03)
Ti-46 (n,p) Sc-46	0.72 (0.89)	0.76 (0.91)	0.60 (0.73)	0.50 (0.59)	0.89 (1.11)	0.77 (0.94)
Fe-54 (n,p) Mn-54	0.89 (0.89)	0.90 (0.88)	0.65 (0.63)	0.52 (0.49)	1.01 (1.00)	0.89 (0.87)
Ni-58 (n,p) Co-58	0.87 (0.87)	0.92 (0.89)	0.77 (0.74)	0.58 (0.54)	1.05 (1.03)	0.93 (0.90)
Nb-93 (n,n) Nb-93m	0.93 (0.93)	1.03 (1.04)	1.00 (1.02)	0.59 (0.58)	1.15 (1.20)	0.84 (0.88)

Conclusions

- Known defects in the ENDF/B-VIII.0 iron evaluations have been corrected in ENDF/B-VIII.1.

- Transport cross section changes from ENDF/B-VIII.1 cause reaction rates for high-energy sensors to increase, which generally leads to worse agreement with measured data.

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