Computation Of Neutron Fluence Information Exchange

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Background

- Chapter 15, “Accident Analysis”, of NUREG-0800 (“Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants”) assumes the reactor pressure vessel (RPV) does not fail.
- Specific fracture toughness requirements for normal operation and for anticipated operational occurrences for power reactors are set forth in Appendix G, “Fracture Toughness Requirements”, of 10 CFR Part 50 (“Domestic Licensing of Production and Utilization Facilities”).
  In addition, the NRC issued 10 CFR 50.61, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.”
- The fracture toughness of pressure vessel materials is related to a parameter called the material's "reference temperature for nil-ductility transition," or simply reference temperature, RT_{NDT}.

Background

- The RT_{NDT} is defined in Revision 2 of Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," by a correlation of the fluence (E > 1 MeV), material chemistry (concentrations of Cu and Ni), initial reference temperature, and margin to account for uncertainties in the correlation and input values.
- In 10 CFR 50.61, evaluation of the reference temperature based on the best estimate of the fast neutron fluence at the end of the license period is required. The corresponding reference temperature is termed RT_{PTS}.
- Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” was issued in March 2001 to provide start-of-the-art calculations and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence.

Background

- Historically, the main region of concern for RPV fluence has been the beltline region.
- With the continuing trend of plant life extension and power uprates at nuclear power plants in the U.S., there is growing concern about fluence levels in regions outside the beltline and in reactor vessel internals.
- Calculation of fluence levels outside the beltline region may be subject to phenomena that are not important for beltline region fluence calculations. Hence, the guidance in RG 1.190 may not be adequate for fluence calculations outside the beltline region.

What is the “Beltline Region”?

- Section II.F of 10 CFR Part 50, Appendix G, defines the beltline region as

  “The region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient radiation damage to be considered in the selection of the most limiting material with regard to radiation damage.”

What is the “Beltline Region”?

- Section III of Appendix H (“Reactor Vessel Material Surveillance Program Requirements”) of 10 CFR Part 50 requires that reactor vessels for which the peak neutron fluence at end of the design life of the vessel exceeds 10^{17} n/cm^2 (E > 1 MeV) must have their beltline materials monitored by a surveillance program complying with ASTM E185, as modified by Appendix H.
- The NRC has used a fluence value of 10^{17} n/cm^2 (E > 1 MeV) to define the extent of the RPV beltline (NUREG-1511, “Reactor Pressure Vessel Status Report,” Section 2.3).

What is the "Beltline Region"?

- *Integrity of reactor pressure vessels in nuclear power plants: Assessment of irradiation embrittlement effects in reactor pressure vessel steels* (IAEA Nuclear Energy Series, 2009) refers to the beltline as

  "the region of shell material directly surrounding the effective height of the fuel element assemblies, plus an additional volume of shell material both below and above the active core, with an EOL fluence of more than 10^{21} m^{-2} (E > 1 MeV)." (10^{17} n/cm^2)

- Note that this is consistent with NUREG-1511.

Typical EOL Neutron Fluences

- Chapter 12 of *Nuclear Power - Control, Reliability, and Human Factors* (Pavel V. Tsvetkov, Editor) states that typical end-of-life (EOL) design neutron fluences are

	on the order of	**10^{18} n/cm^2**	for BWRs
and	on the order of	**10^{19} n/cm^2**	for PWRs.

The IAEA Nuclear Energy Series reference lists values of

4×10^{18} n/cm^2	for BWRs,
4×10^{19} n/cm^2	for Westinghouse PWRs, and
1.2×10^{19} n/cm^2	for B&W PWRs.

The PWR fluence values are noted as corresponding to a lifetime of 32 EFPY (1.01×10^9 s). No lifetime is noted for BWRs.

Today's Presentation

- We noted on Slide 4 that the calculation of fluence levels outside the beltline region may be subject to phenomena that are not important for beltline region fluence calculations. In today's presentation we will briefly discuss our analysis methodology for fluence calculations and present the results of some parametric studies that illustrate effects of source, geometry, and materials that affect 'beyond the beltline' fluence calculations in ways that they do not affect beltline region calculations.

Calculational Methodology

- RG 1.190 discusses two fundamental methods of neutron flux calculations: deterministic (discrete ordinates) and stochastic (Monte Carlo).
- Deterministic calculations have been used for decades to obtain ‘global’ neutron flux solutions based on a transport model that is discretized in space, energy, and angle. Extensive studies have been performed to assess the sensitivity of RPV neutron flux calculations to each of these discretized variables. These studies have led to recommendations on multigroup cross-section libraries, angular quadrature schemes, and spatial discretization methods.

Calculational Methodology

- Monte Carlo calculations have the advantage of allowing a more accurate representation of the problem geometry and continuous energy cross sections. Thus, they are not subject to the same parameter sensitivity effects that occur in deterministic calculations.
- The primary shortcoming of Monte Carlo methods for many decades has been the excessive computational time required to obtain well-converged solutions. In addition, Monte Carlo calculations have typically been used to calculate fluxes (or responses based on the flux) at only a few selected locations, rather than a global solution.

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Calculational Methodology

- For our analyses we have applied the hybrid FW-CADIS methodology, which uses ‘moderate fidelity’ determinstic calculations to generate variance reduction parameters (weight windows and source biasing) that significantly improve the convergence rate of Monte Carlo calculations.
- The FW-CADIS method has been validated with existing vessel fluence benchmarks (PCA, VENUS) and using the PWR and BWR vessel fluence calculational benchmarks in NUREG/CR-6115 (“PWR and BWR Pressure Vessel Fluence Calculation Benchmark Problems and Solutions”).
- Use of the hybrid method allows us to obtain ‘global’ solutions with well-converged mesh tallies in an acceptable amount of computational time.

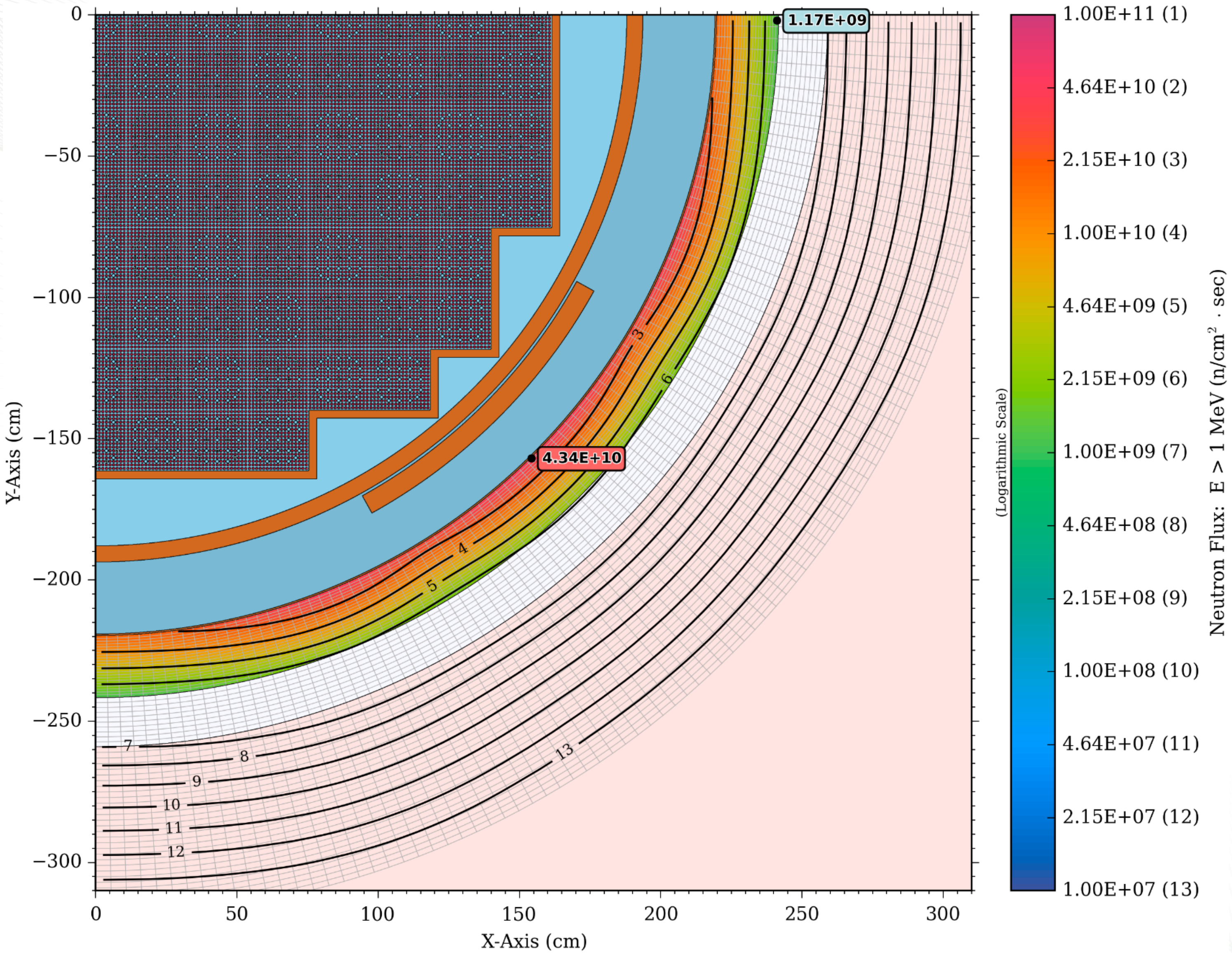
Parameter Study Results for a Representative PWR model

- To assess the sensitivity of fluence calculations outside the beltline region to a number of modeling parameters, we applied a model of the Watts Bar Unit 2 reactor. We used both an explicit core model (pin-by-pin modeling) and a homogenized core representation.

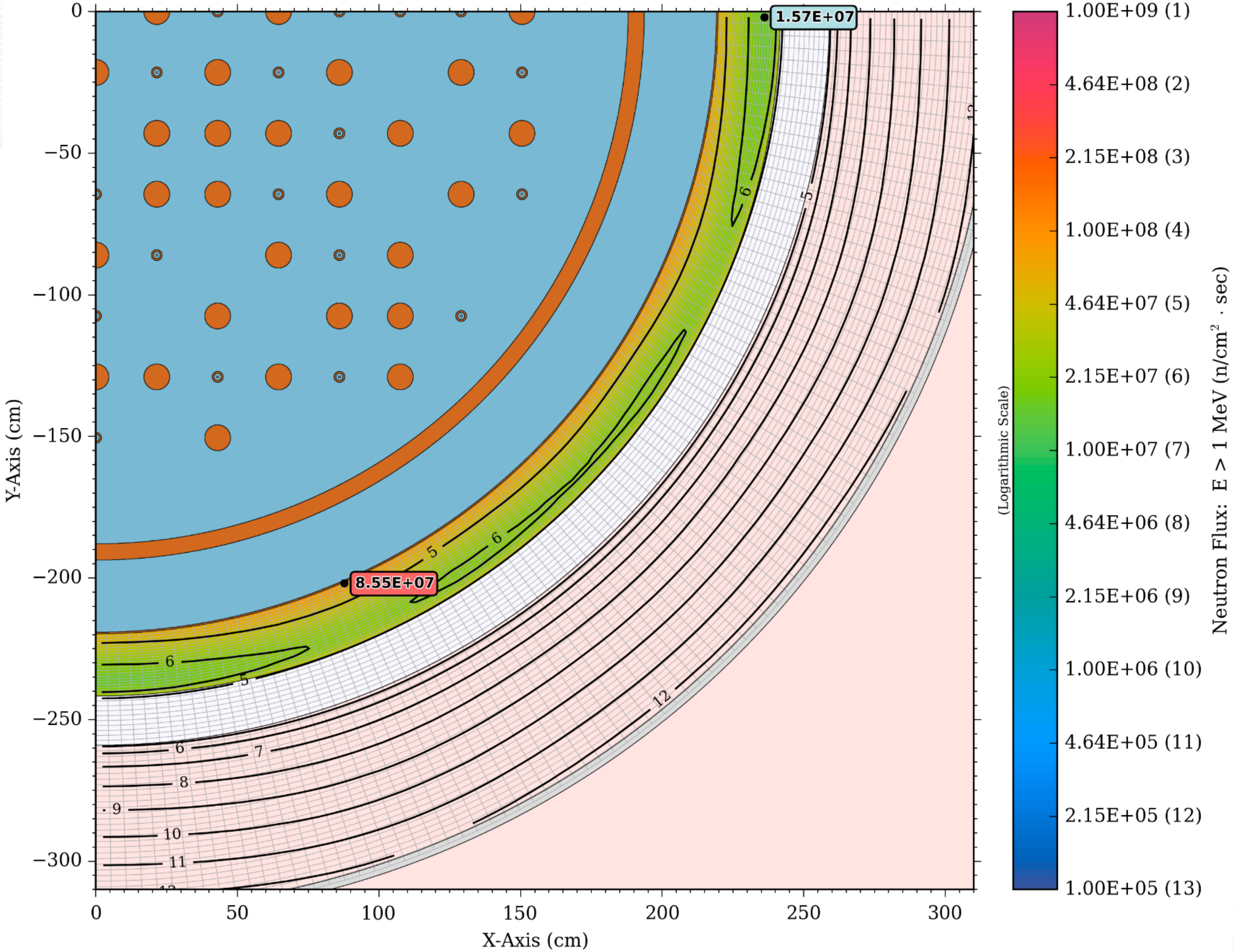
- Our hybrid calculations were performed using MCNP5 and ADVANTG. MCNP is a widely-used Monte Carlo code developed at LANL. ADVANTG is an automated variance reduction generator code developed at ORNL. Both codes are distributed by the Radiation Safety Information Computational Center (RSICC).

“Baseline” Model Results

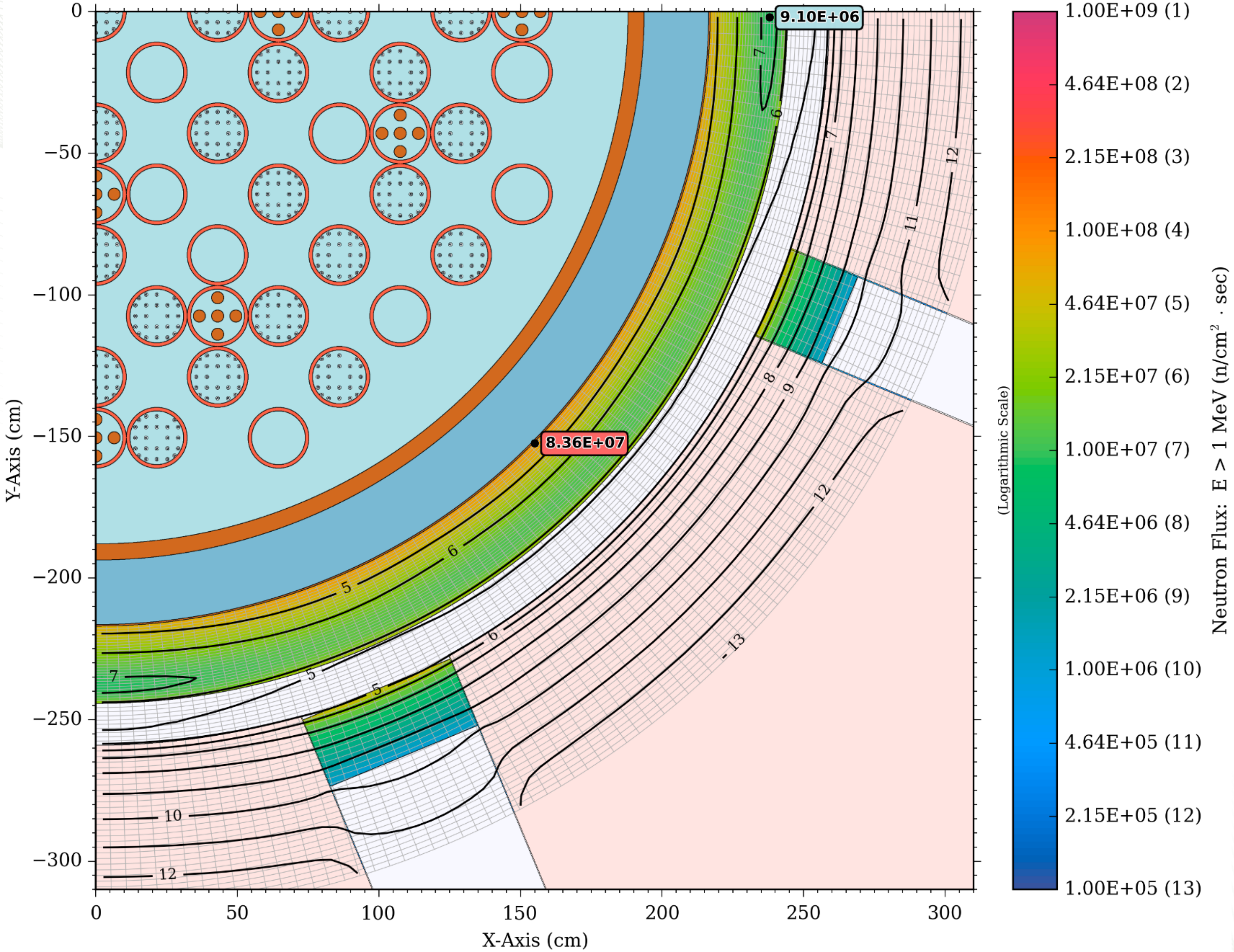
- We begin with a ‘baseline’ model and present representative fast flux levels for a spatially uniform U-235 fission source. While a spatially uniform source is certainly not intended to be a representative source at any time in life, it provides a reasonable base case for parameter sensitivity studies.



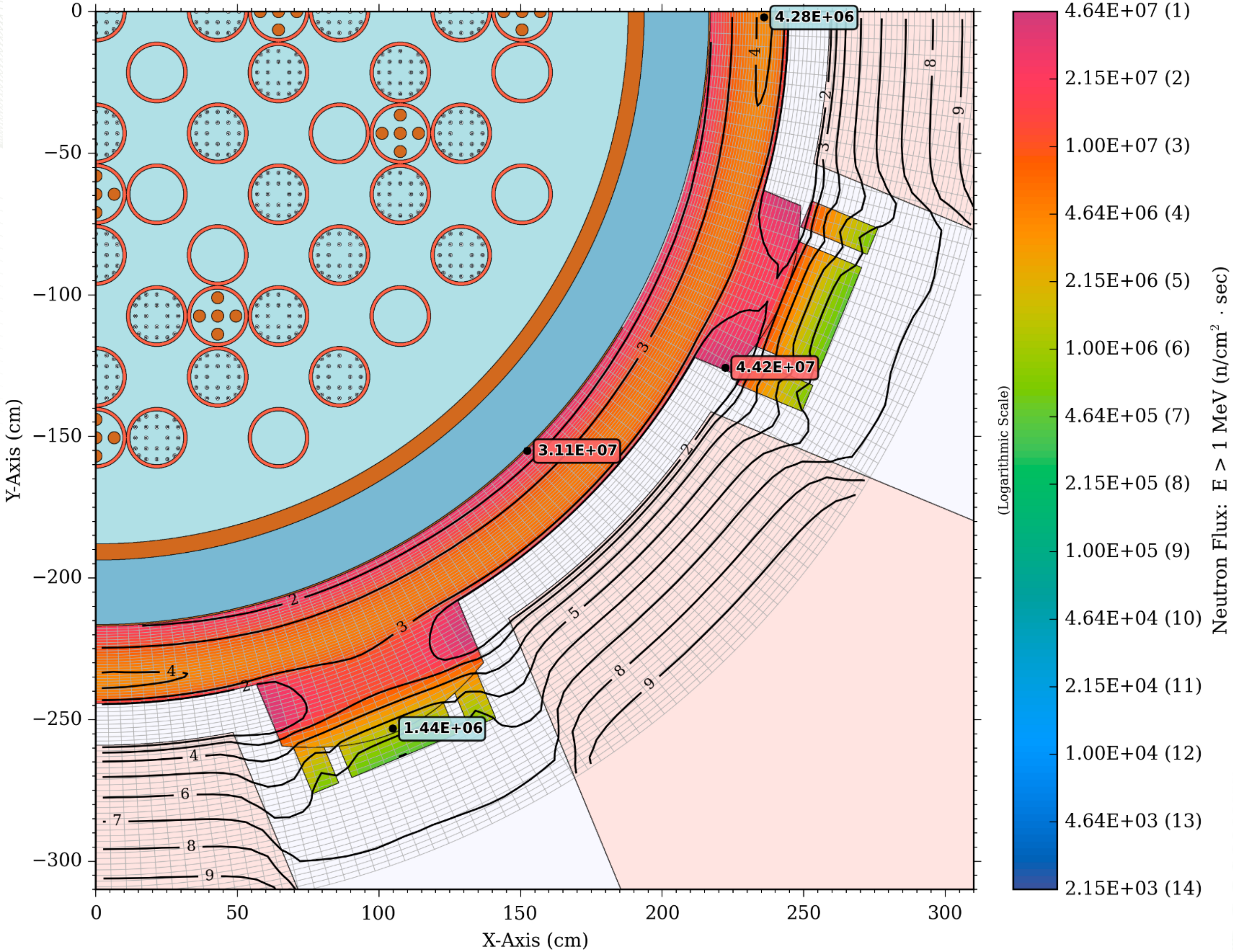
Fast neutron flux (E > 1 MeV) at Z = 200 cm (near the core midplane elevation) for the PWR reference model. Note that the peak flux value of 4.34E+10 n/cm^2-s is consistent with the 32-year EOL fluence value for a Westinghouse PWR from the IAEA report.



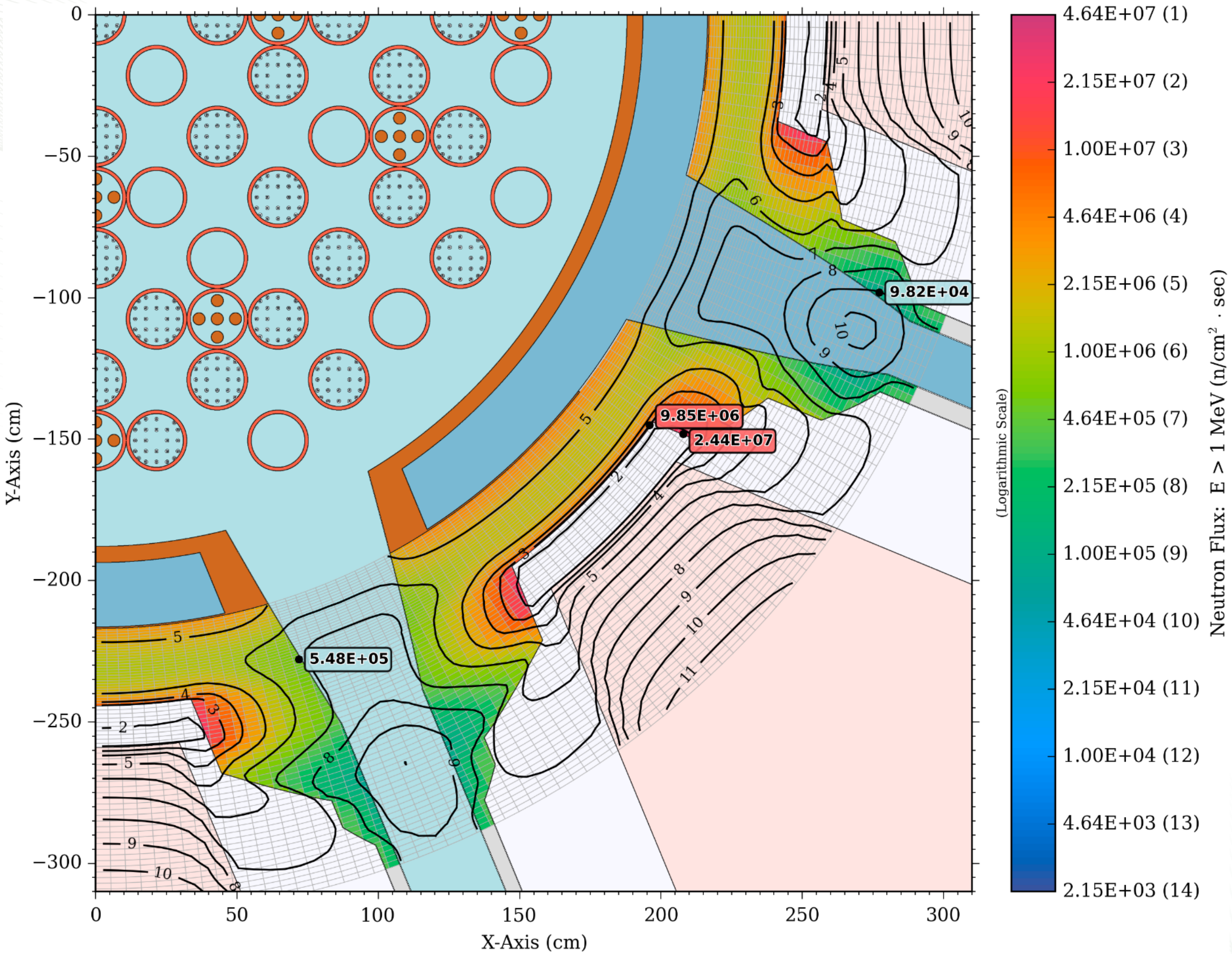
Fast neutron flux (E > 1 MeV) at Z = -60 cm (approximately 70 cm below the bottom of the active fuel).



Fast neutron flux (E > 1 MeV) at Z = 460 cm (approximately 80 cm above the top of the active fuel).



Fast neutron flux (E > 1 MeV) at Z = 475 cm (nozzles and nozzle supports shown).



Fast neutron flux (E > 1 MeV) at Z = 500 cm (through the lower portion of the inlet and outlet nozzles).

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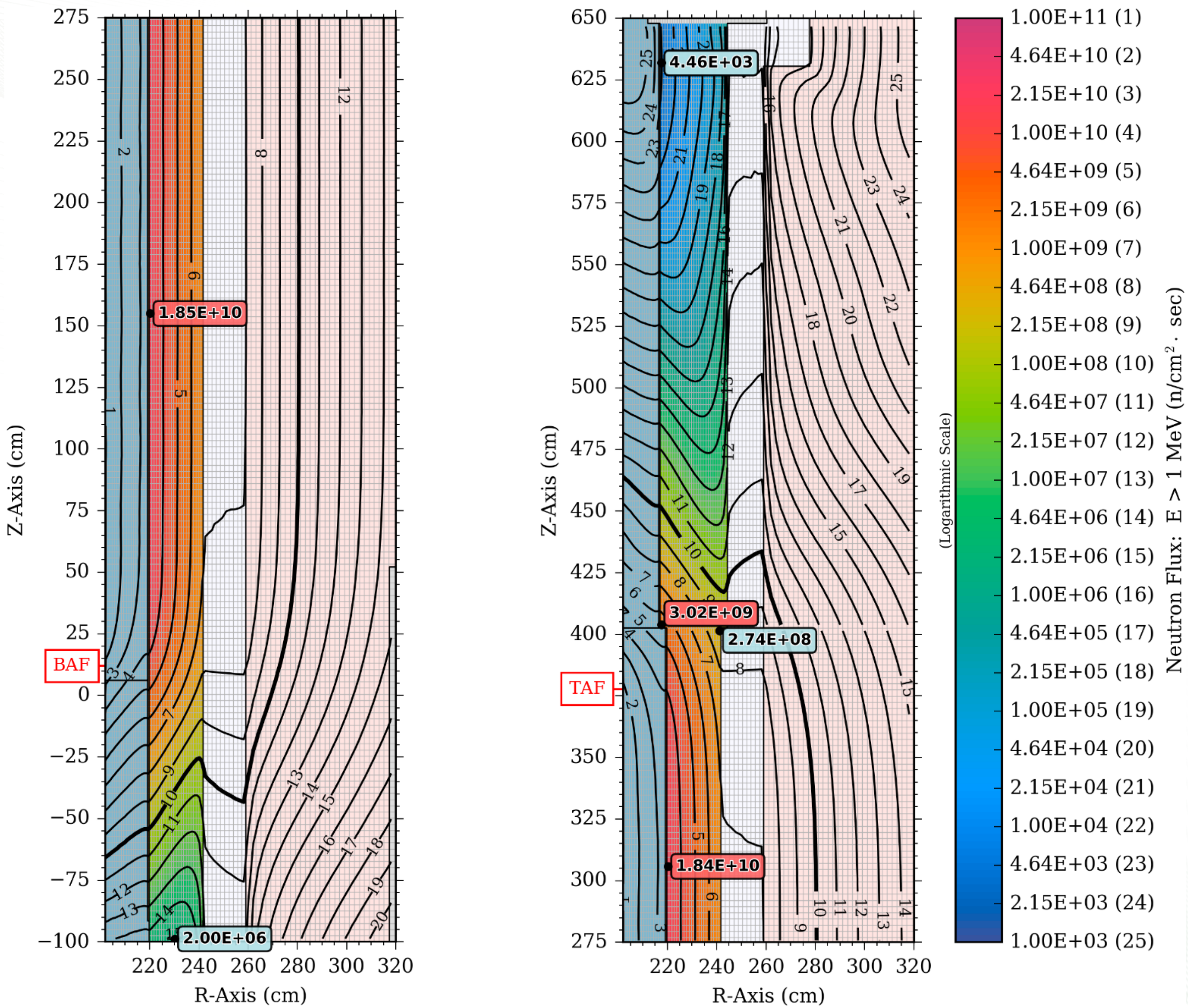
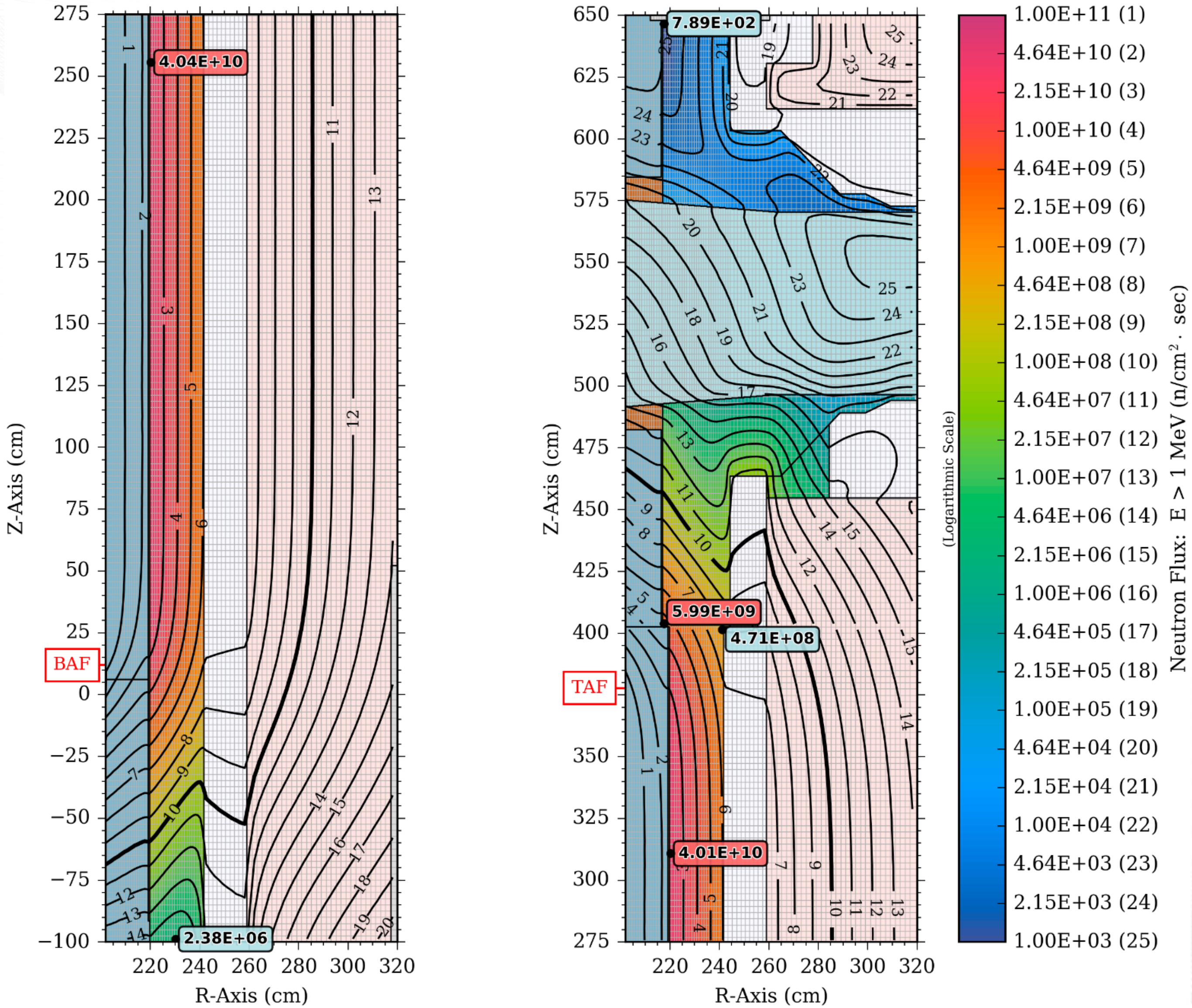
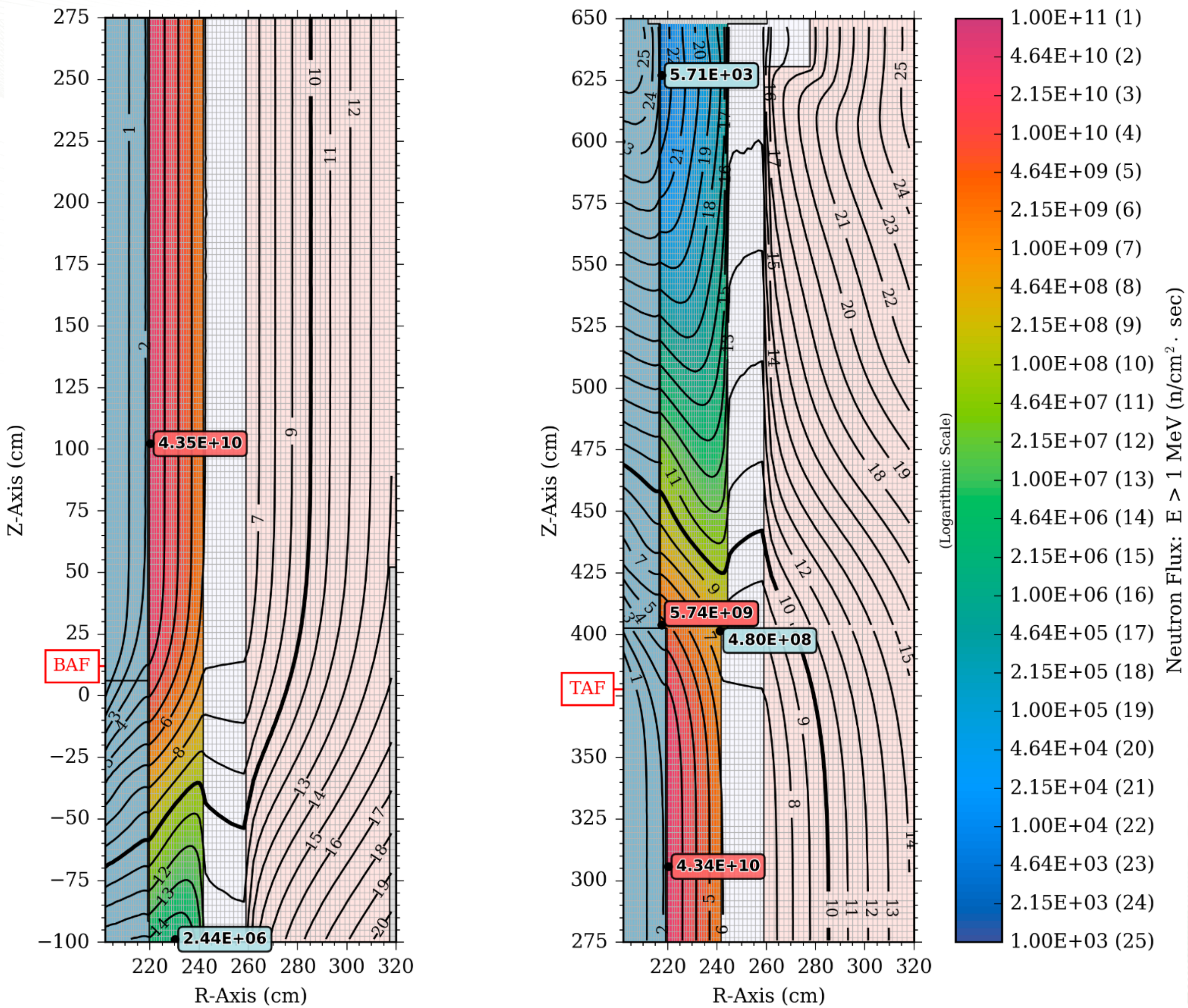
Axial Flux Profile at Theta = 270.5° for the PWR Reference Model With a Spatially Uniform U-235 fission Source



Axial Flux Profile at Theta = 292.5° for the PWR Reference Model With a Spatially Uniform U-235 fission Source



Axial Flux Profile at Theta = 315.5° for the PWR Reference Model With a Spatially Uniform U-235 fission Source



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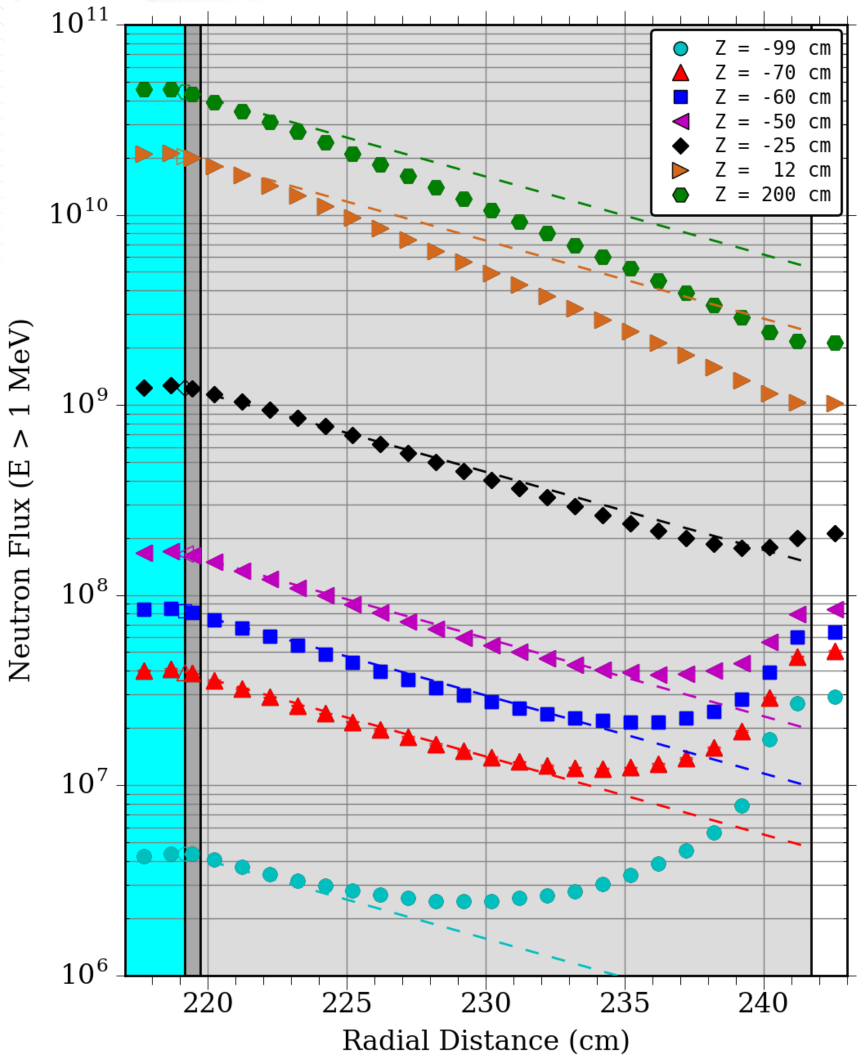
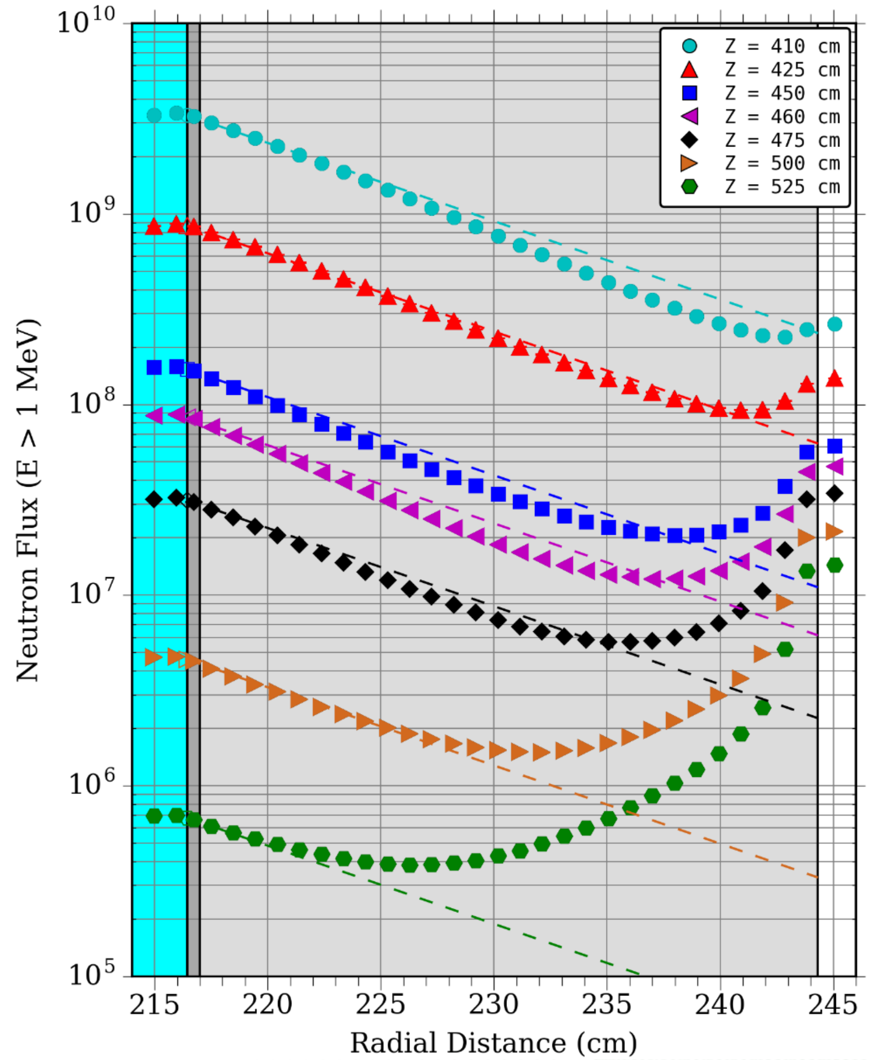
Radial Fast Flux Profiles at θ = 315°

Fitted curves are of the form $\phi(r) = \phi_s e^{-0.24x}$, where ϕ_s is the flux at the wetted surface of the vessel and x is the depth into the vessel wall in inches





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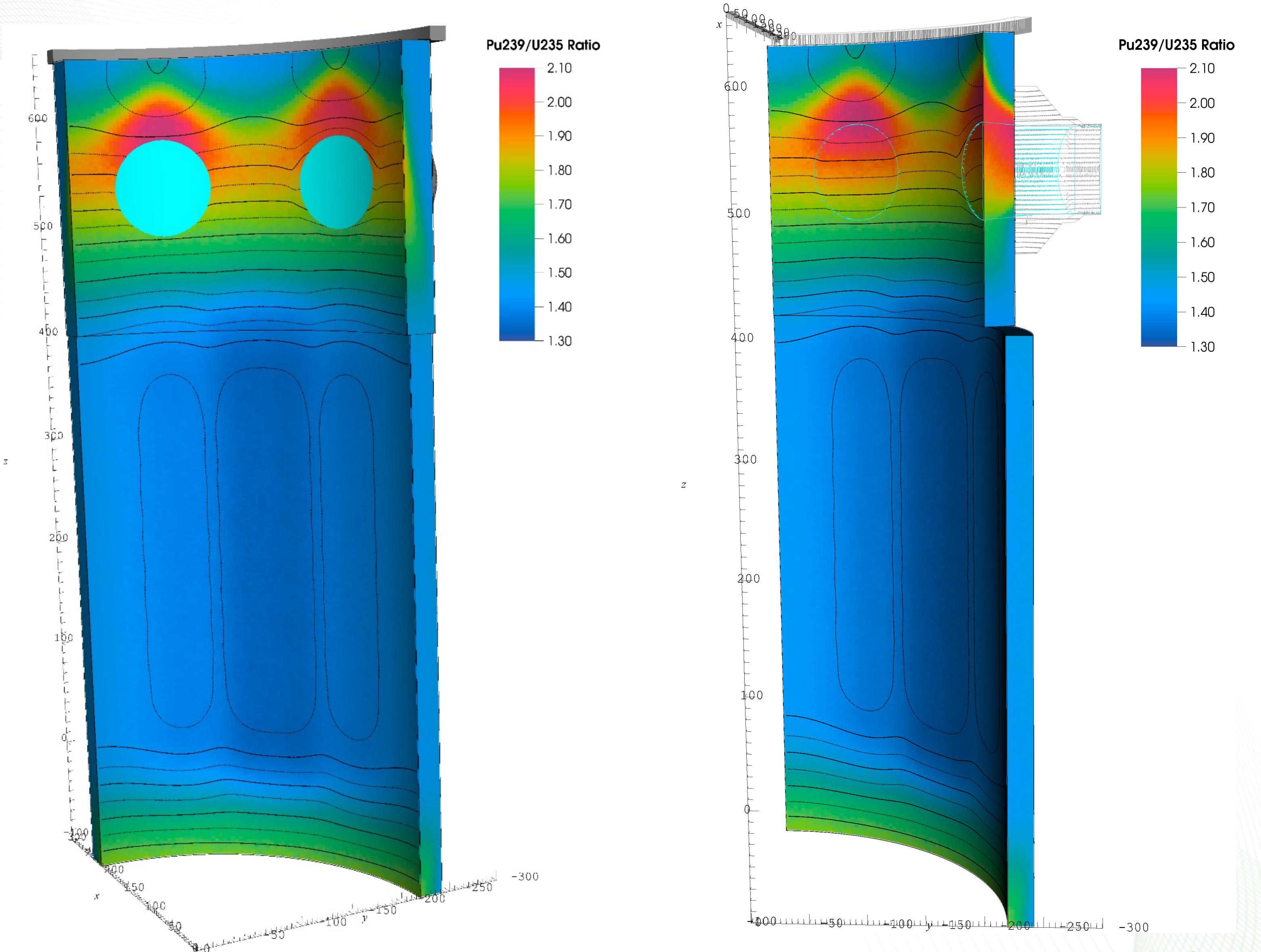
General Note on Plotting of Parameter Study Results

- On Slides 14 - 21, the flooded contours and contour lines both correspond to the fast flux values.
- In the following parameter study slides, the flooded contours are used to represent the ratio between two solutions (typically a 'perturbed' calculation relative to the baseline model), while the contour lines are used to represent the fast flux values for the baseline solution.
- The minimum and maximum ratio values are shown within the RPV and, for plots that include any portion of the nozzles, within the combination of the RPV and the nozzles.
- In some cases, a separate set of minimum and maximum ratio values is shown within the first 2 cm of the RPV.
- For the parameter studies, a model with no RPV supports was used. For the cavity gap study the RPV supports would have to be modeled differently for each perturbed case. Using a model with no supports makes it possible to have more 'direct' comparisons.

Effect of Fission in Pu-239 vs. U-235

- As LWR fuel burns up, an increasing amount of fission occurs in Pu-239. Fission in Pu-239 has a ‘harder’ neutron spectrum (shifted toward higher energies) than U-235, and has a higher average number of neutrons produced per fission. Both of these effects tend to result in higher fast flux levels in the RPV. The following calculations demonstrate that the impact of Pu-239 fission is more significant at vessel locations outside the beltline region.

Pu-239/U-235 Fast Flux Ratio on the RV Inner Surface



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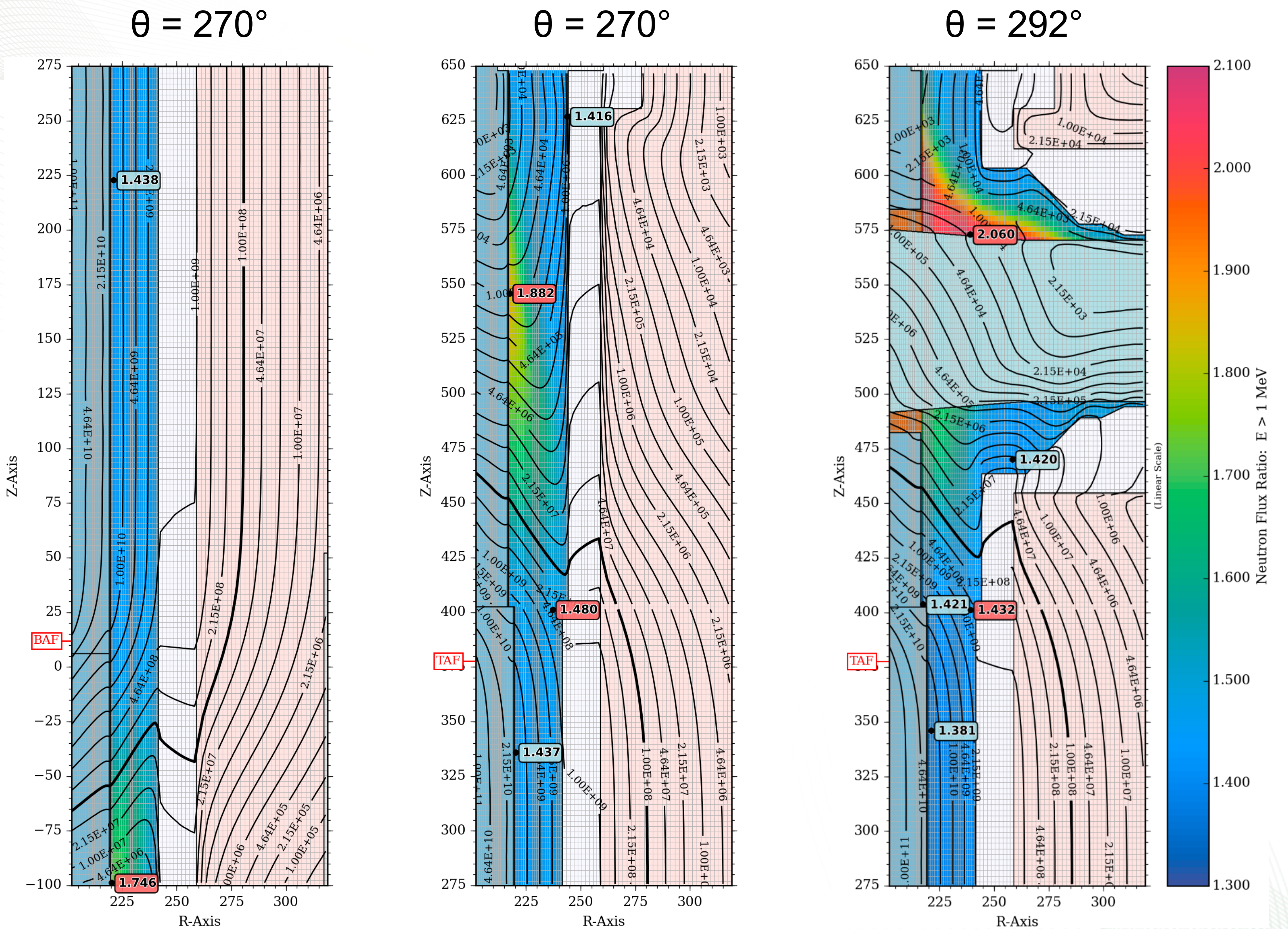
Pu-239/U-235 Fast Flux Ratio at θ = 270° and 292°



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Effect of Concrete Composition

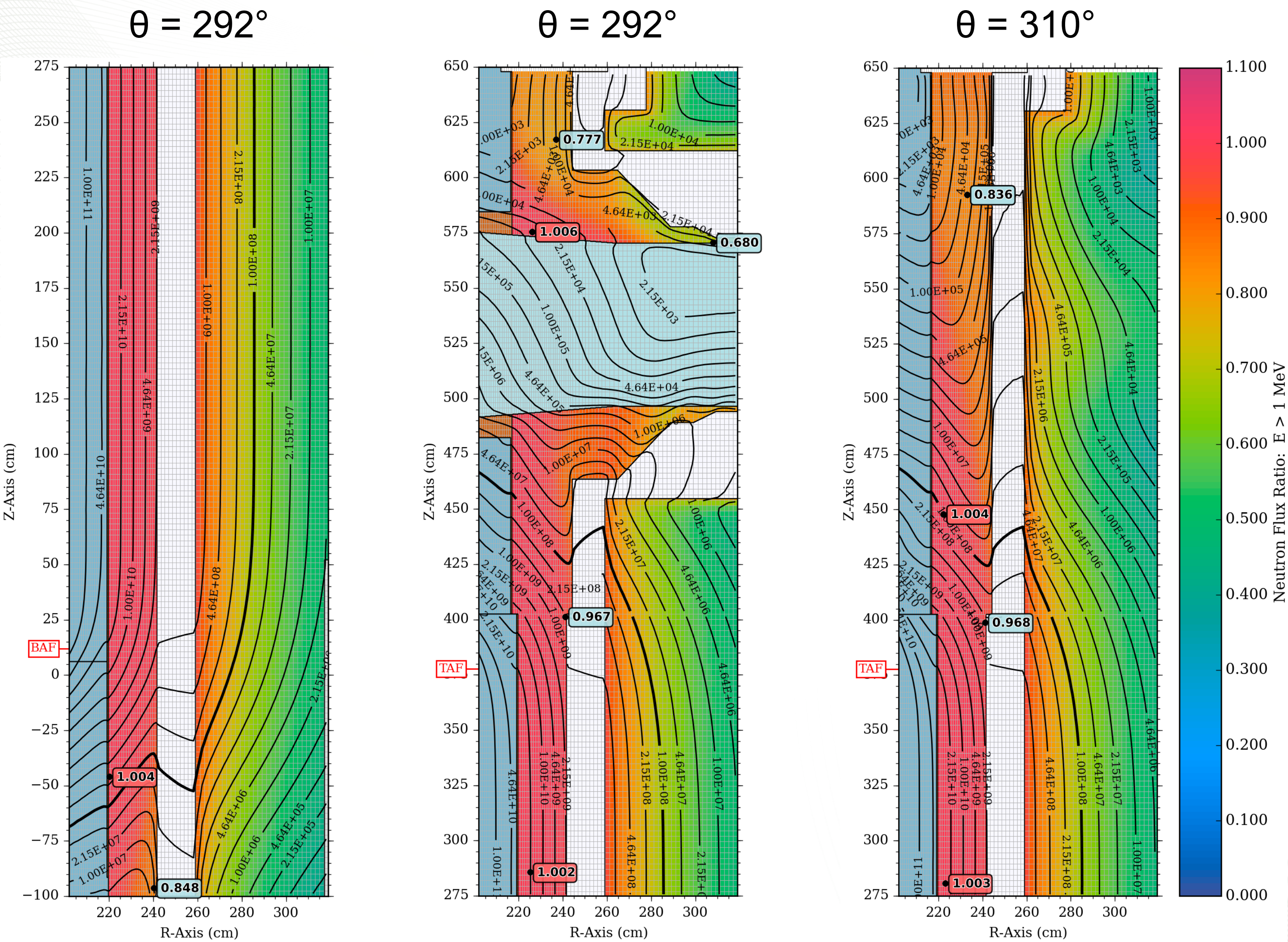
- The radial flux profiles shown on Slide 22 indicate the increasing importance of cavity streaming as the axial distance from the active fuel region increases. Thus, we would expect that changes to the cavity geometry and/or materials might have a noticeable impact on fast flux levels outside the beltline region. Here we look at the effect of changing the concrete composition.
- The baseline model uses NBS-04 concrete (ANSI/ANS-6.4-2006). We also consider NBS-03 concrete, 'regulatory' concrete (from the SCALE manual), and PNNL 'ordinary' concrete.

Effect of Concrete Composition

Concrete Elemental Compositions by Weight Fraction

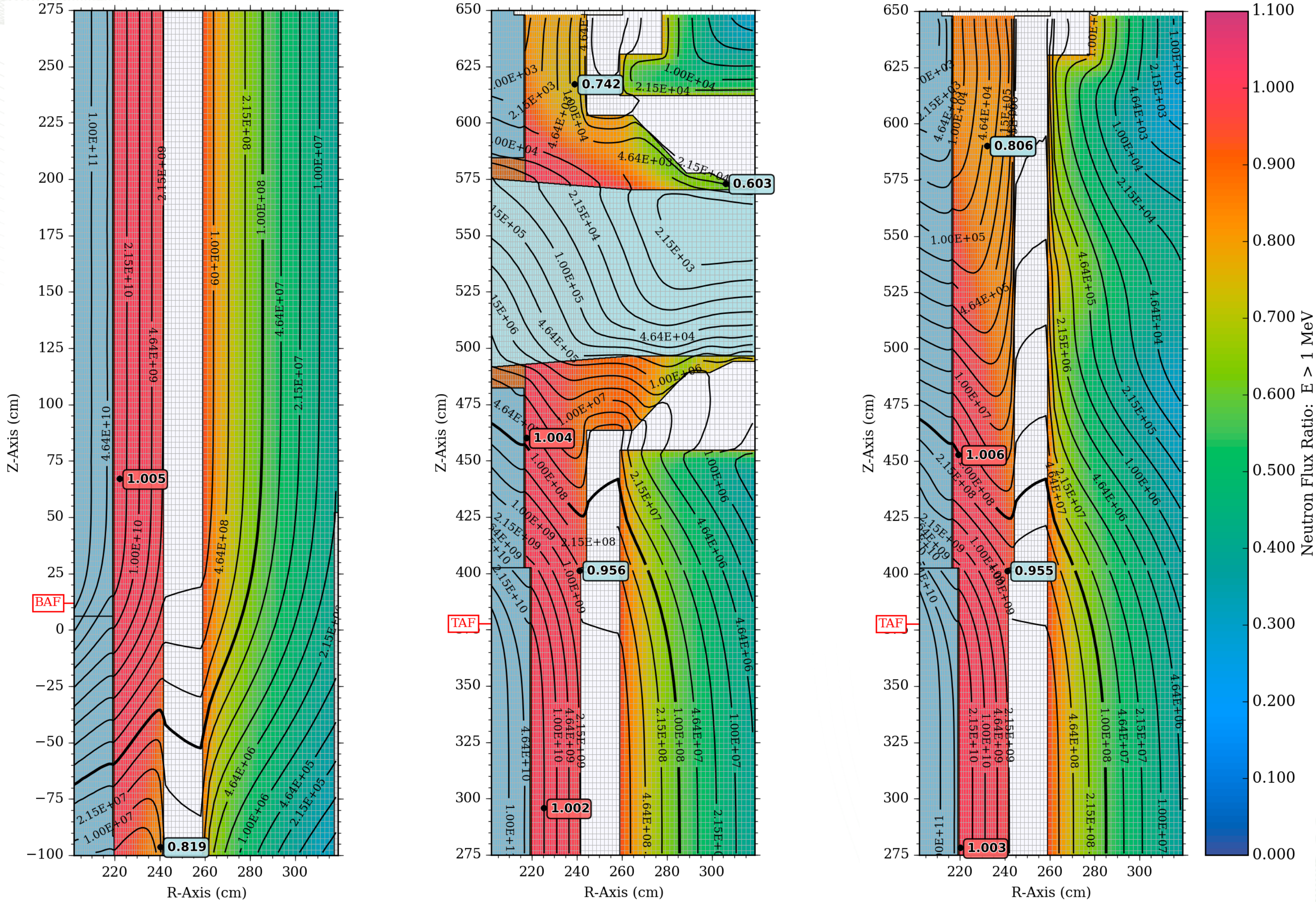
Element	NBS-04 ρ = 2.35 g/cc	NBS-03 ρ = 2.39 g/cc	Regulatory ρ = 2.30 g/cc	PNNL Ordinary ρ = 2.30 g/cc
H	5.5319E-03	8.3682E-03	1.00E-02	2.21E-02
O	4.9830E-01	4.7657E-01	5.32E-01	5.7493E-01
Si	3.1574E-01	1.4310E-01	3.37E-01	3.04627E-01
Ca	8.2553E-02	2.4351E-01	4.40E-02	4.2951E-02
C	—	4.9372E-02	—	2.484E-03
Na	1.7021E-02	—	2.90E-02	1.5208E-02
Mg	2.5532E-03	2.3849E-02	—	1.266E-03
Al	4.5532E-02	3.5565E-02	3.40E-02	1.9953E-02
S	1.2766E-03	2.9289E-03	—	—
K	1.9149E-02	1.6736E-03	—	1.0045E-02
Fe	1.2340E-02	1.2552E-03	1.40E-02	6.435E-03
Ni	—	1.0879E-02	—	—
P	—	2.9289E-03	—	—

NBS-03 / NBS-04 at θ = 292° and 310°



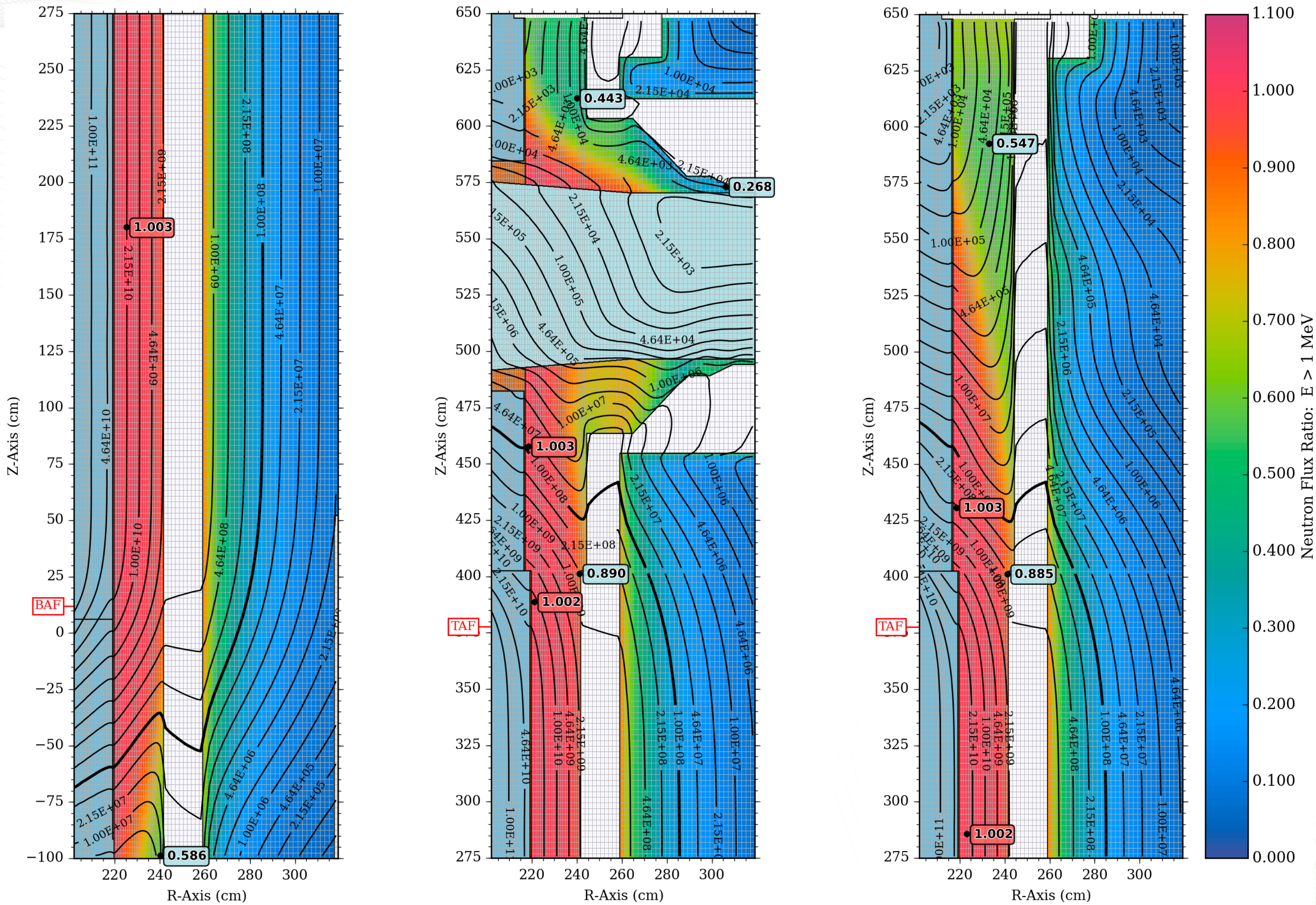
“Regulatory” / NBS-04 at θ = 292° and 310°

θ = 292° θ = 292° θ = 310°



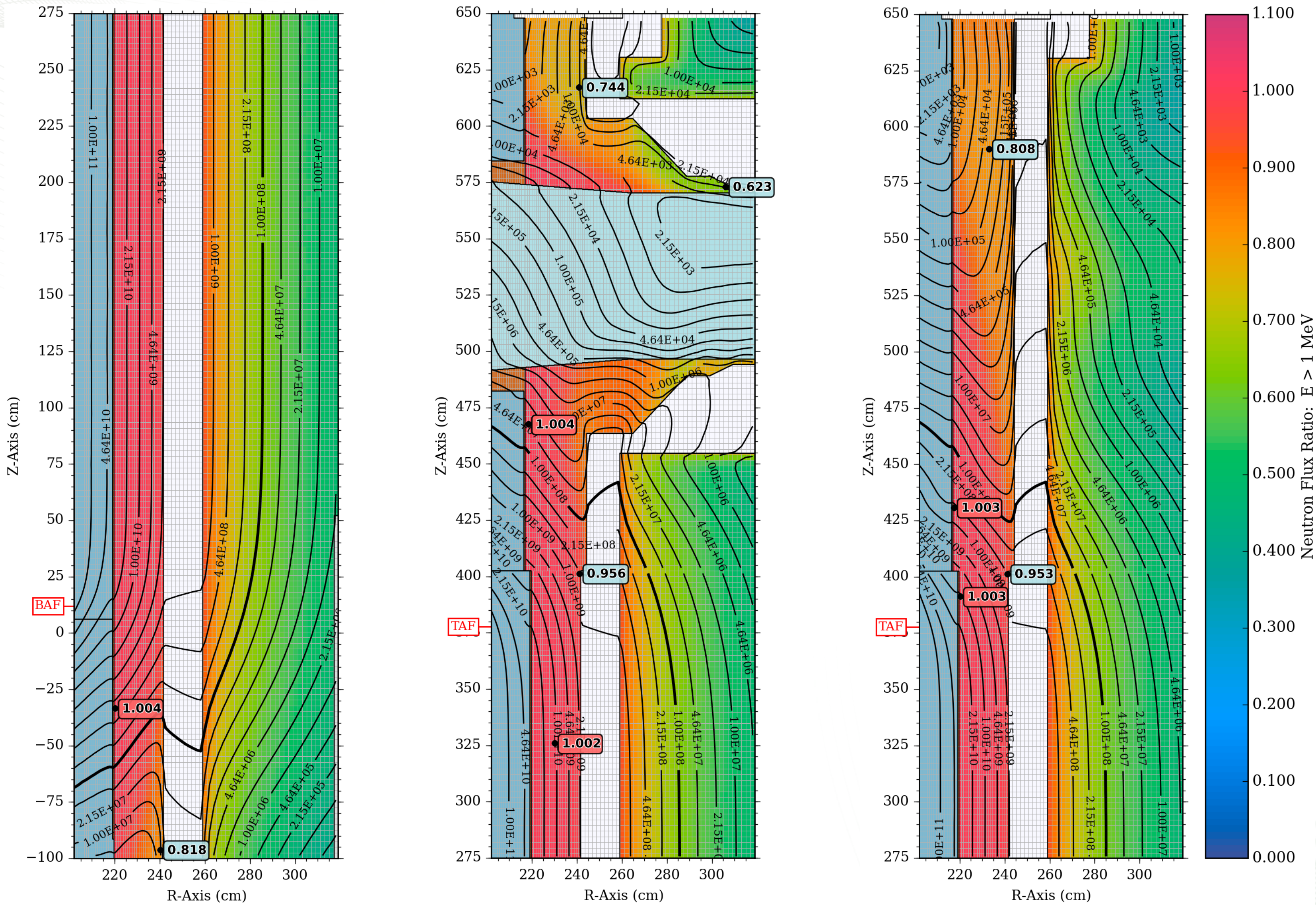
PNNL "Ordinary" / NBS-04 at θ = 292° and 310°

θ = 292° θ = 292° θ = 310°



PNNL “Ordinary” with Reduced H / NBS-04 at θ = 292° and 310°

θ = 292° | θ = 292° | θ = 310°



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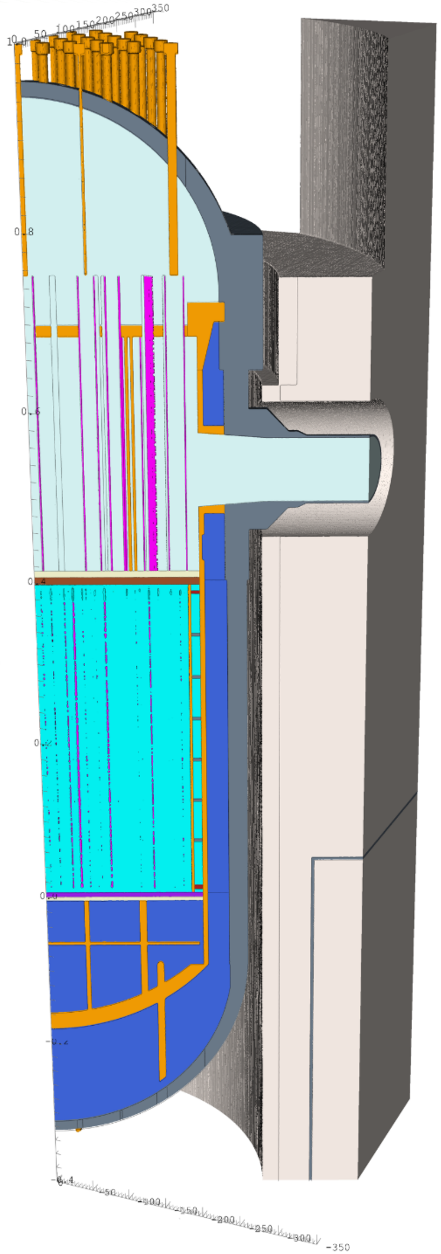
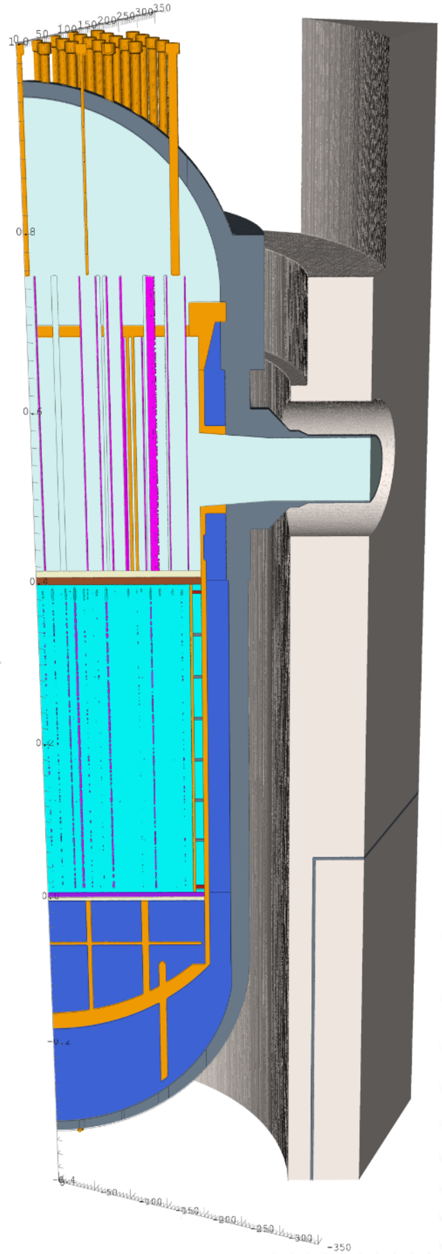
Effect of RV/Bioshield Cavity Gap Width

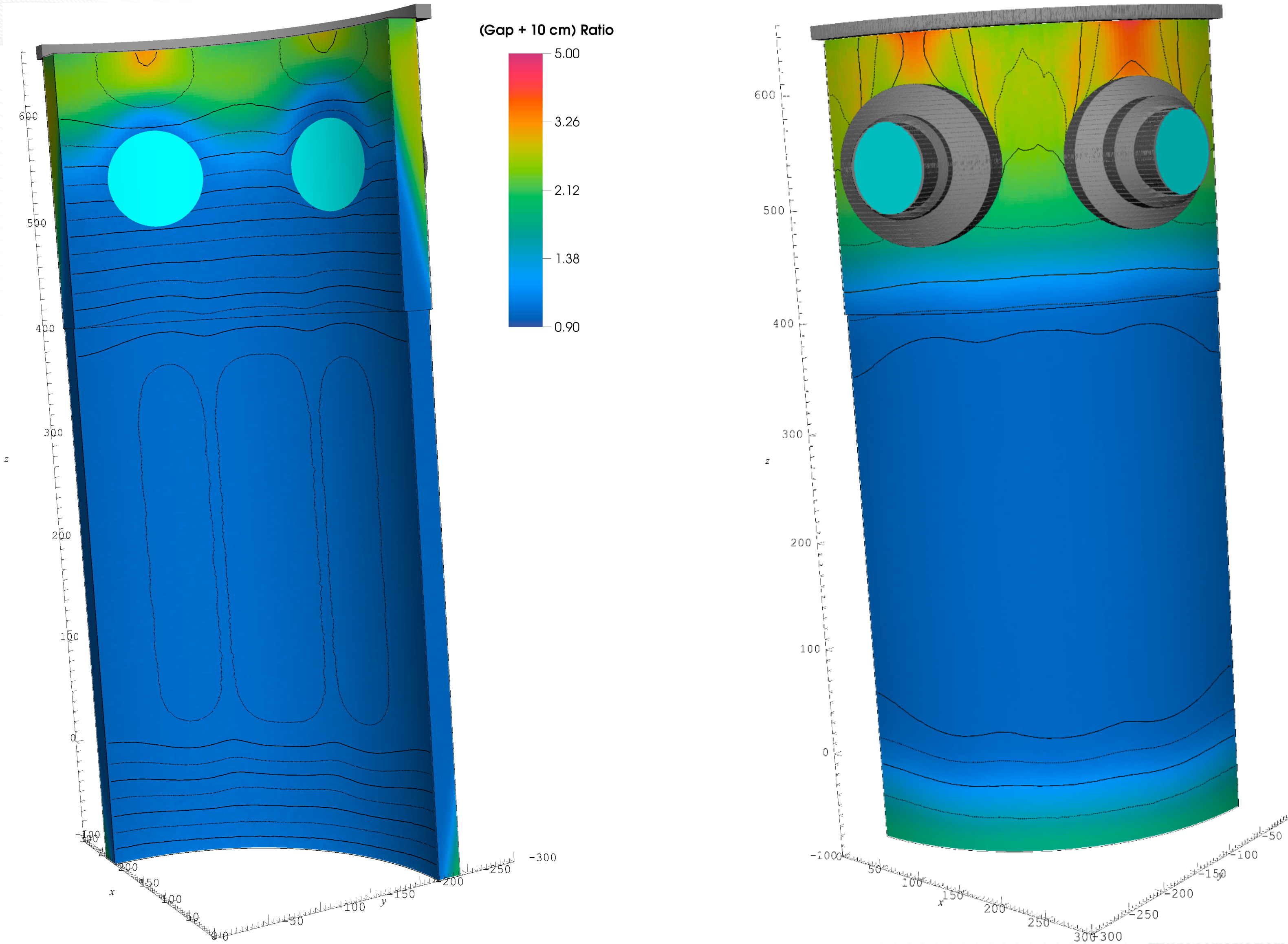
- For the next analysis we consider increasing the width of the reactor cavity gap, which will change the scattering/streaming geometry. The baseline model has cavity gap widths of 17.38 cm below 402.59 cm (the elevation where the RPV thickness increases slightly), and 14.76 cm from 402.59 cm to 647.86 cm (where the vessel flange begins.)
- We evaluated gap width increases of 10, 20, and 30 cm. Results for the 10-cm and 30-cm increases are shown.

Effect of RV/Bioshield Cavity Gap Width

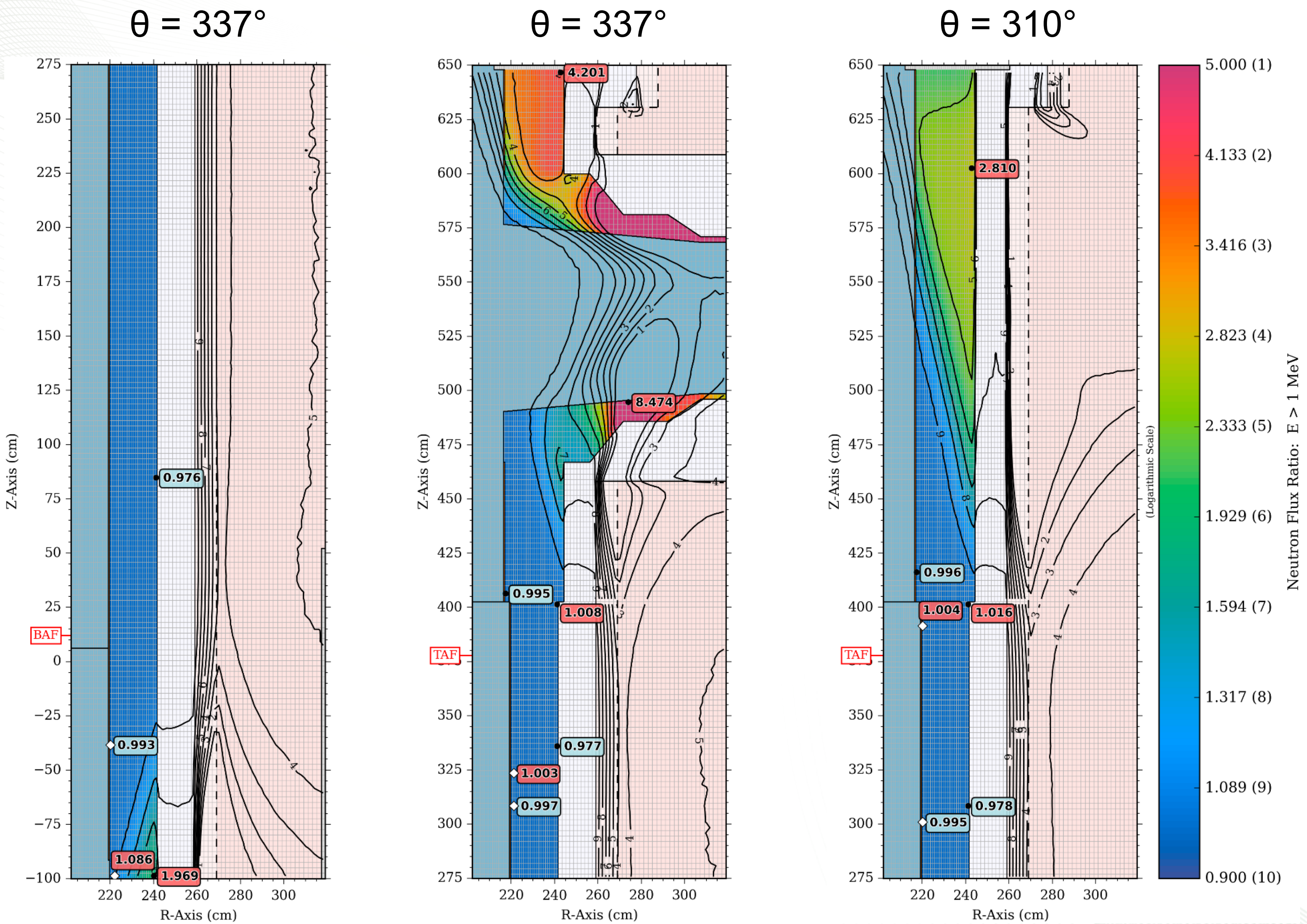




Increasing the Cavity Gap by 10 cm: Fast Flux Ratio



Increasing the Cavity Gap by 10 cm: Fast Flux Ratio



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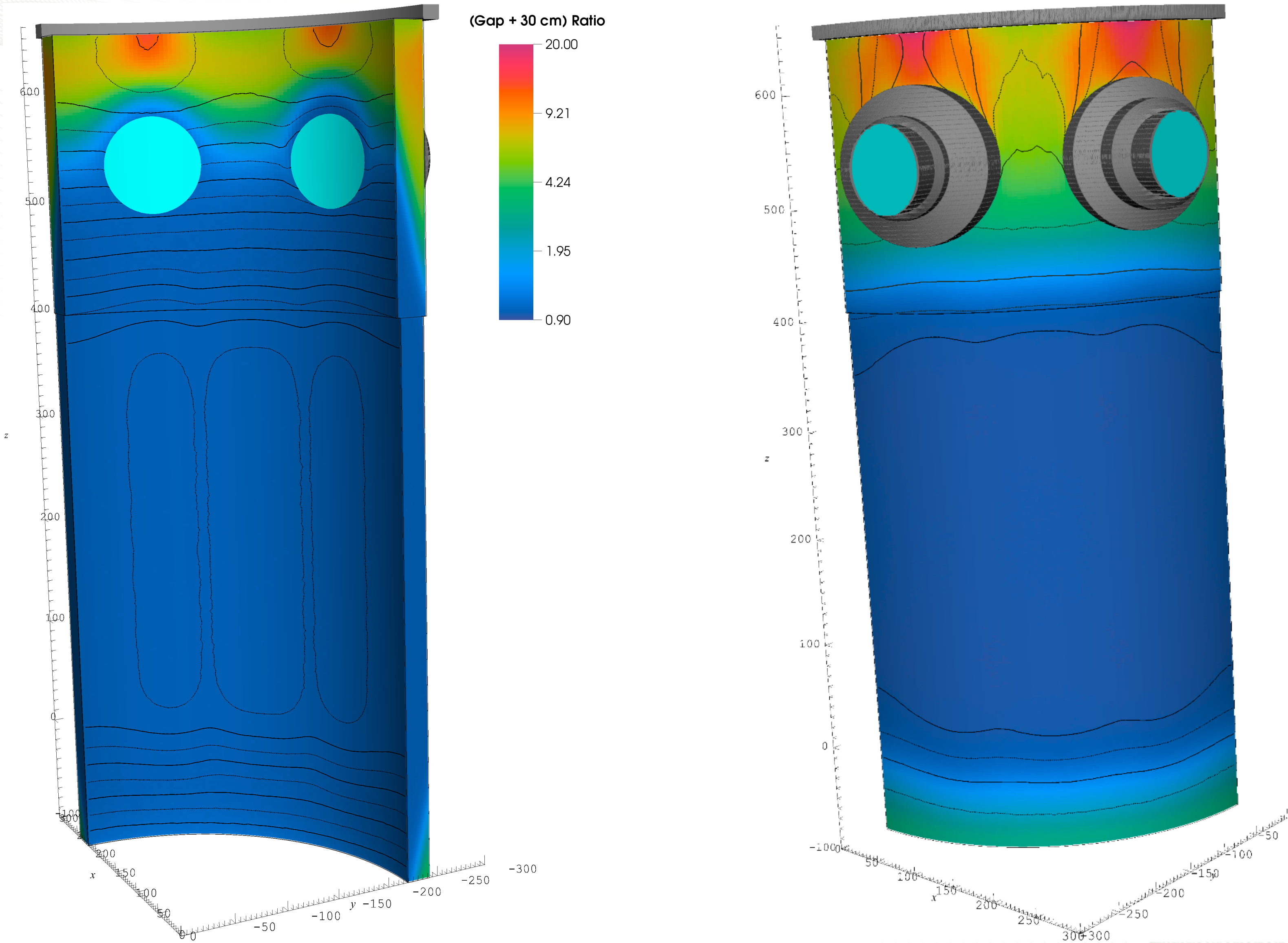
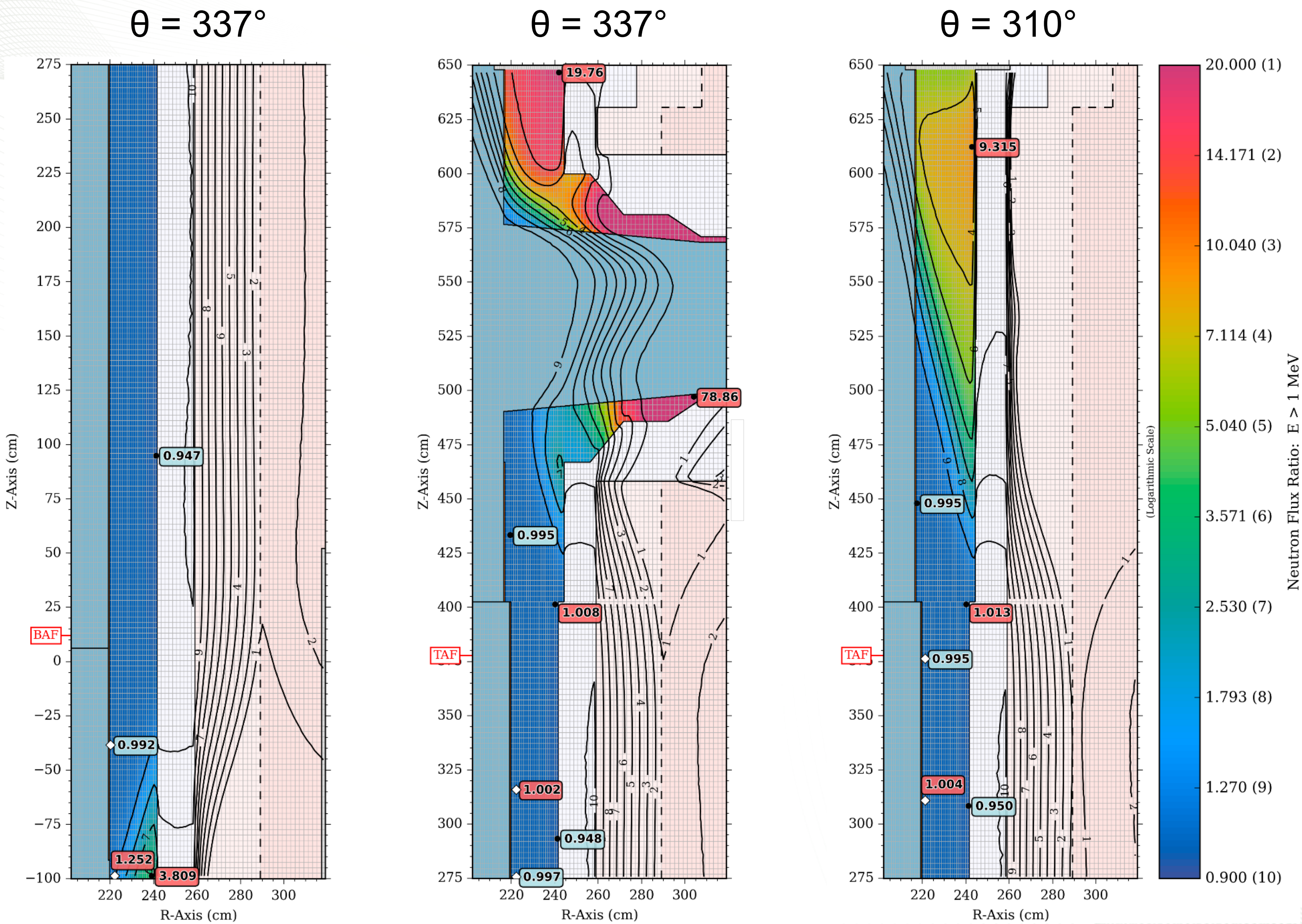
Increasing the Cavity Gap by 30 cm: Fast Flux Ratio



Increasing the Cavity Gap by 30 cm: Fast Flux Ratio

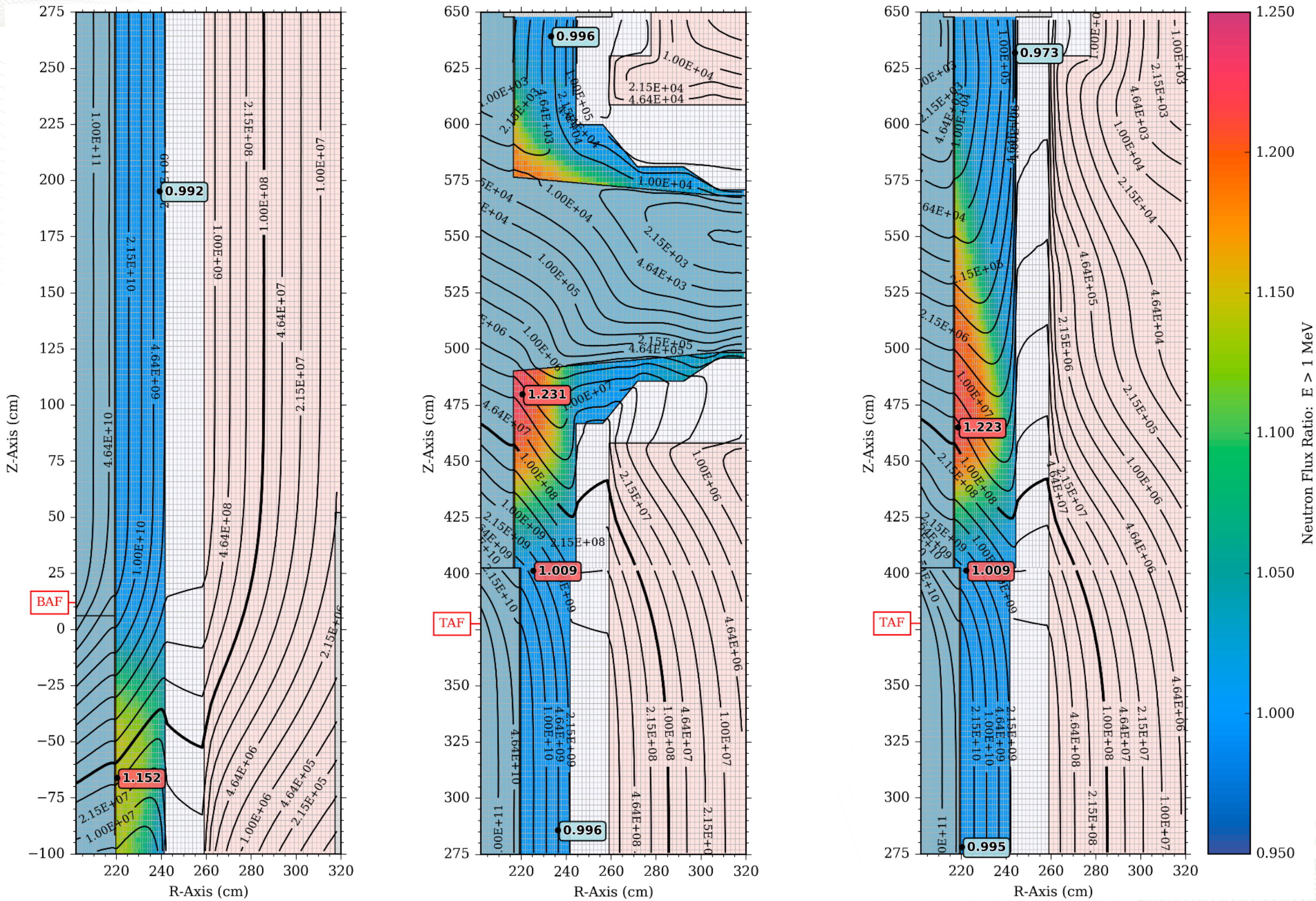


Effect of the M/W Density in the Top and Bottom Nozzles and the Upper and Lower Core Plates

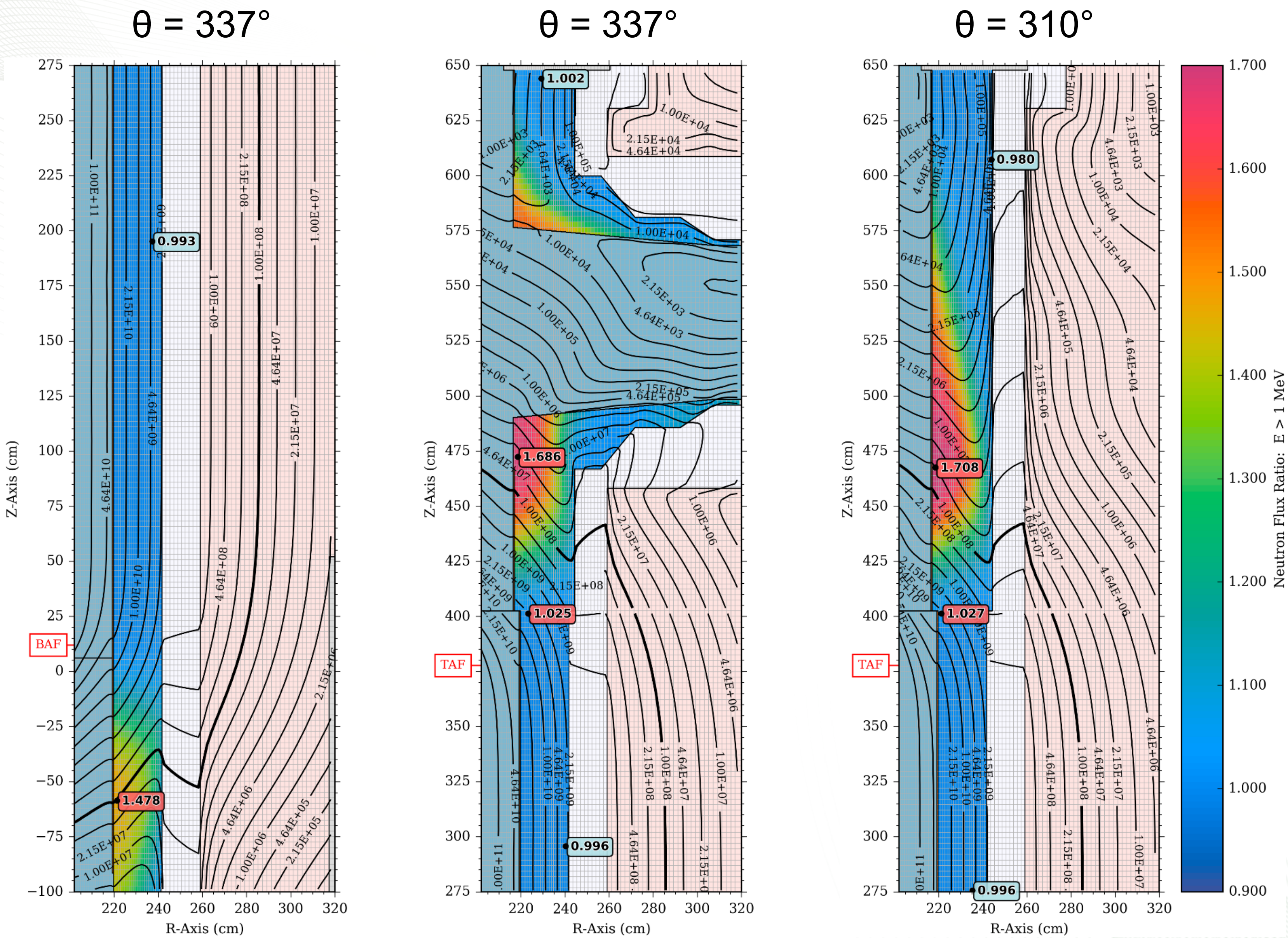
- The axial regions immediately above and below the active fuel are modeled as metal/water mixtures for the top and bottom nozzles and core plates. (This was done because of the available information on the Watts Bar core.) These regions have little to no effect on fast flux levels in the RPV within the active fuel height, but may be significant for locations outside the beltline region.
- For this study, we simply modify the density of these regions to obtain an indication of where this modeling is important for RPV flux calculations.
- It may be necessary to obtain more detailed information to model the top and bottom nozzles and core plates explicitly.

Decreasing the M/W Density by 10%: Fast Flux Ratio

θ = 337°   θ = 337°   θ = 310°



Decreasing the M/W Density by 25%: Fast Flux Ratio



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