Determination of Inventories and Power Distributions for the NBSR

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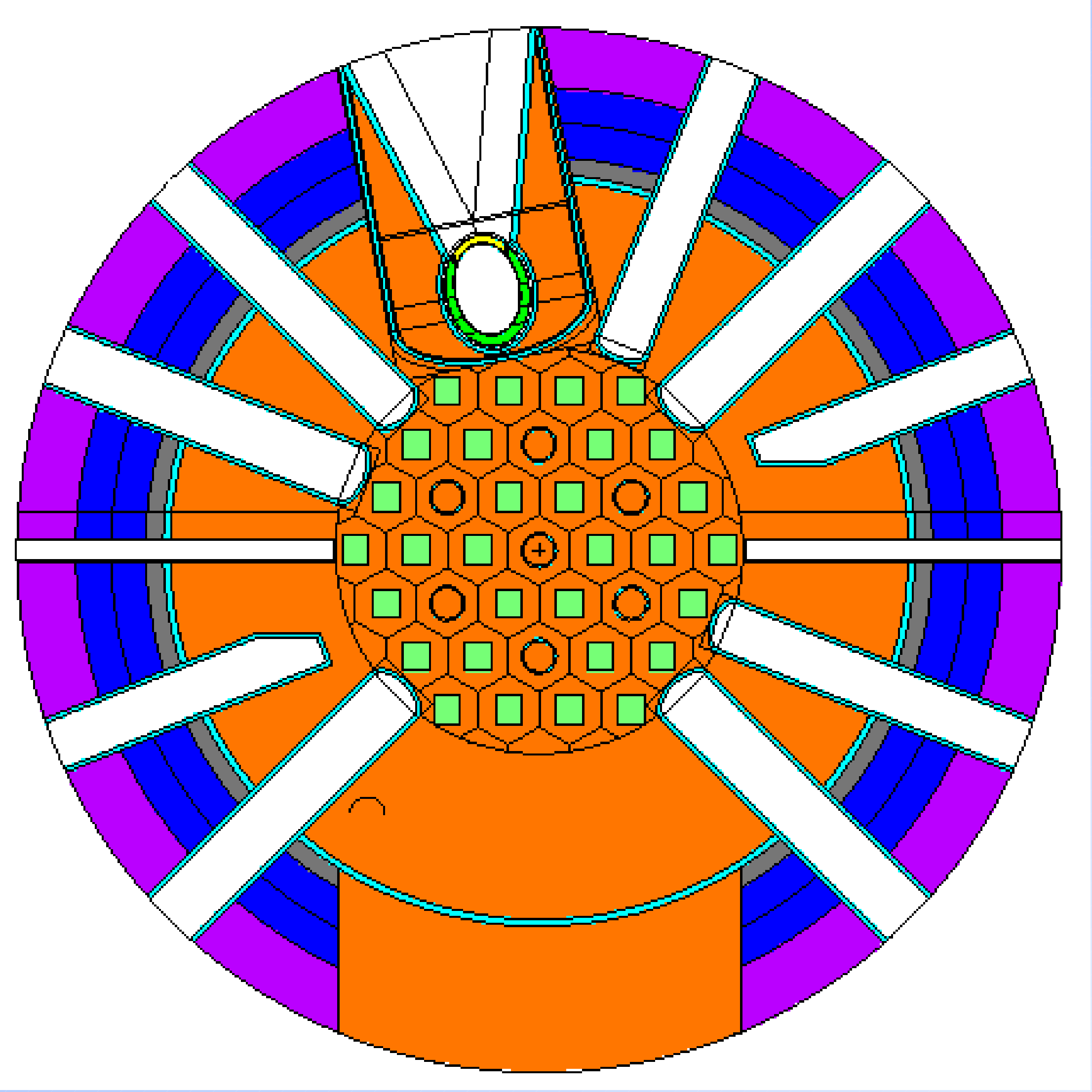
NBSR Characteristics

- MTR type plate fuel
- HEU
- U_3O_8 sintered with aluminum and clad in aluminum
- 30 fuel elements
  - 16 irradiated for 8 cycles (38days/cycle)
  - 14 irradiated for 7 cycles
- Split core
  - Each fuel element has 28 inches of fuel
  - There is a 7 inch gap between the upper and lower portions of the fuel
  - Beam tubes face the gap in the fuel

NBSR Radial Geometry at Core Midplane – MCNP Model



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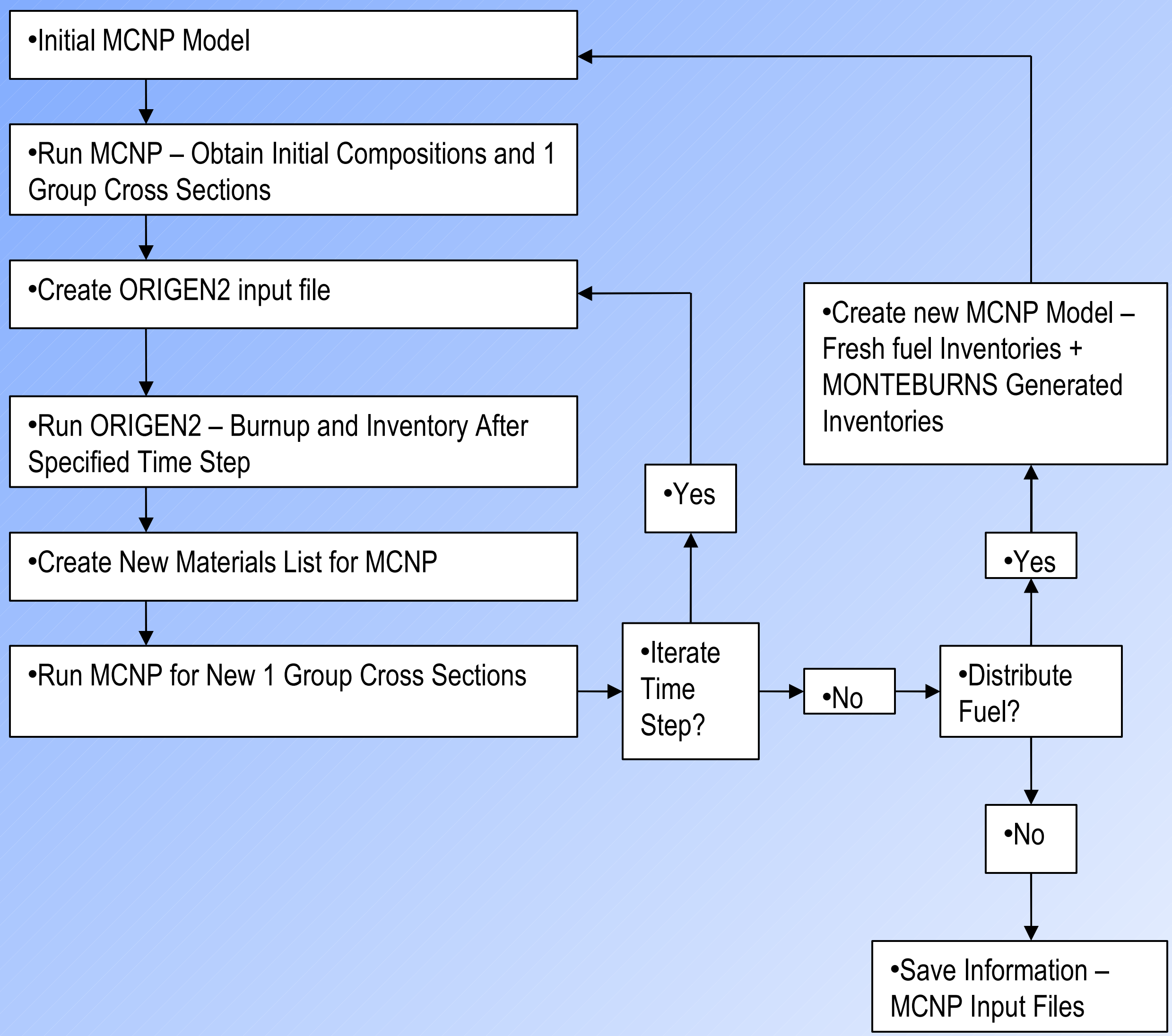
MCNP Model

- Initial inventories was a “best guess“ based on burnup
- Some fission products lumped with aluminum
- 30 different fuel materials were used
  - Different materials for upper and lower halves of each fuel element
  - Assumed East-West symmetry
  - MONTEBURNS has a limit of 49 materials

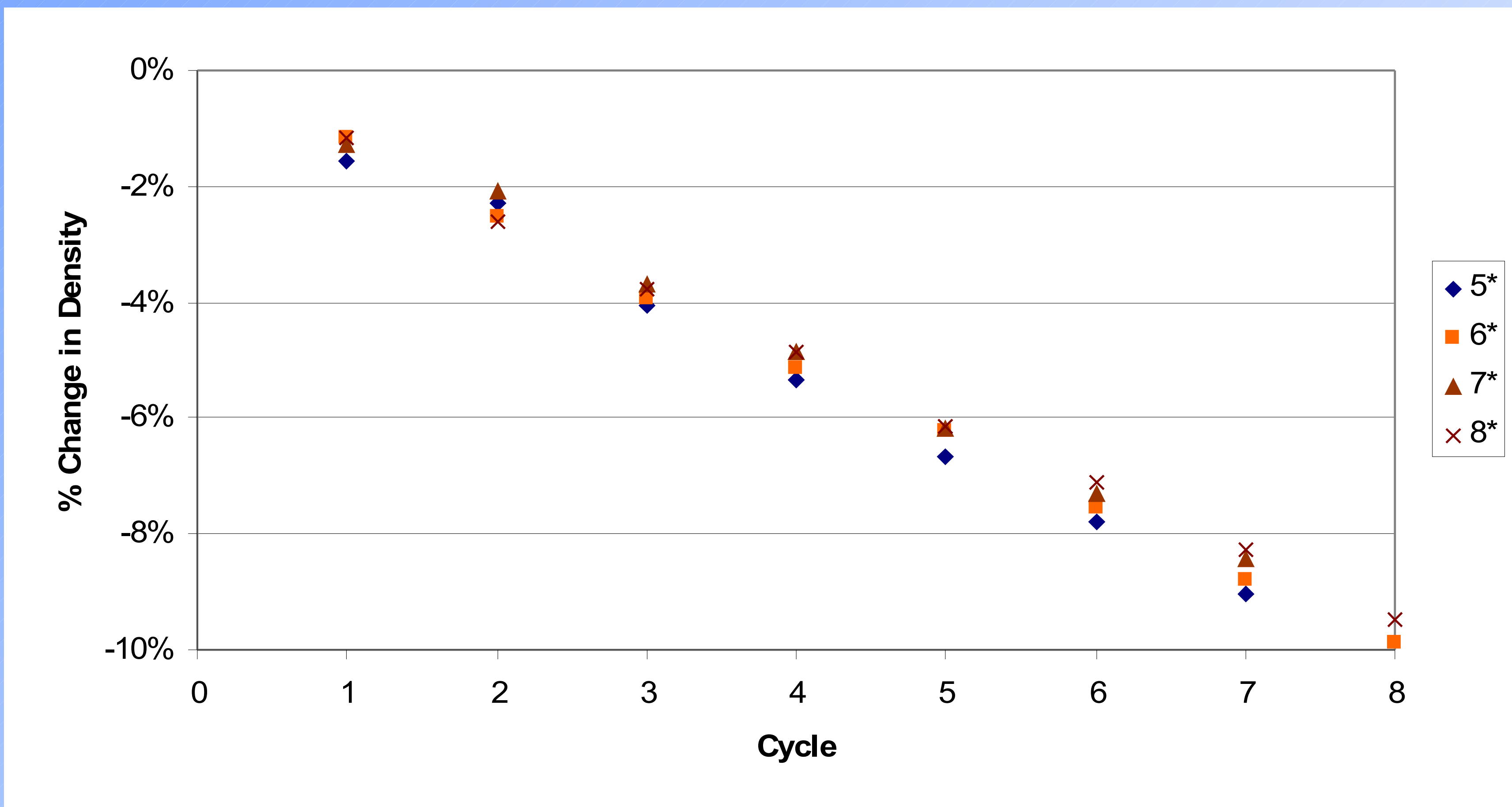
MONTEBURNS Flow Chart



Problem

- The neutron cross section files distributed with MCNP do not support most radioactive fission products
  - Most models lump the non-supported isotopes into representative fission products
- MONTEBURNS approach:
  - Determine the mass of non-supported fission products
  - Discard the non-supported fission products
  - Renormalize the mass fractions to sum to unity
  - Adjust the densities of the materials to maintain the mass of the actinides
  - Result: the end-of-cycle mass is less than the start-of-cycle mass
- Burnup capability is being implemented in MCNPX (presently in alpha testing) – The approach is the same

Density Change in NBSR MONTEBURNS Analysis



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Dealing With the Issue

- In our model, the total number of isotopes a material up to 60
- One can download cross section files for many of the major radioisotopes
  - This solution cannot account for 100% of the mass
  - Computation time increases substantially
- Desire to use real fuel densities
  - Important for power distributions

Our Solution

- Extract density and mass fractions for each material
- Multiply mass fractions by the ratio ρ_{adj}/ρ_{actual}
- Return the aluminum and oxygen mass fractions to original values
- Sum all mass fractions, Σ
- The balance $(1-\Sigma)$ is distributed equally between Sn, ^{138}Ba, and ^{133}Cs as representative isotopes
- This becomes the EOC inventory

Isotopic Adjustments

- The choice of representative isotopes was
  - To include some cross section for fission products
  - Average fission product cross section is ~25 b
  - High absorbing radioisotopes are included:
    - ^{105}Rh σ_a=33000 b
    - ^{135}Xe σ_a=2700000 b
    - ^{149}Pm σ^a=1400 b
    - ^{147}Nd σ_a=400 b
  - The average cross section for the three materials chosen ~10 b

Critical Angles and Predicted k_{eff}

Time step	Angle from Vertical (measured)	k_{eff} (predicted from model)
Startup Core	-19.3°	1.00101 ± 0.00029
BOC	-14.6°	1.00006 ± 0.00028
¼ cycle	-11.5°	1.00502 ± 0.00028
Mid cycle	-9.0°	1.00311 ± 0.00027
¾ cycle	-5.0°	1.00393 ± 0.00027
EOC	°0°	1.00125 ± 0.00027

Power Distributions in Upper and Lower Halves

UPPER

			0.92		1.02		1.06		0.95			
		0.90		0.97		<>		0.90		0.75		
	0.69		<>		0.86		0.86		<>		0.66	
0.60		0.68		0.79		<>		0.80		0.68		0.61
	0.64		<>		0.73		0.74		<>		0.68	
		0.72		0.82		<RR>		0.92		0.90		
			0.98		0.99		1.02		1.06			

LOWER

			1.05		1.14		1.21		1.13			
		1.21		1.23		<>		1.25		1.23		
	1.22		<>		1.22		1.22		<>		1.22	
1.26		1.16		1.19		<>		1.18		1.14		1.22
	1.20		<>		1.06		1.06		<>		1.17	
		1.15		1.15		<RR>		1.15		1.15		
			1.14		1.12		1.13		1.16			

Summary

- Inventories have been developed for the NBSR using MONTEBURNS
  - Total of 30 different fuel materials
  - Split core between upper and lower halves
  - Assumed East-West symmetry
- The MONTEBURNS methodology for calculating inventories invokes some assumptions
  - MONTEBURNS deals with the unsupported fission product problem by reducing material densities
- This requires some adjustments of the inventories before they are used

Problem

- ORIGEN2 calculates the existence of thousands of fission products
- MCNP ENDF/B files have cross sections for only a few radioactive fission products
- MONTEBURNS does not include those fission products when it rewrites the MCNP materials
- Those fission products are lost to the calculation
- Therefore there the end-of-cycle fuel element mass is less than the start-of-cycle mass