Fast Reactor Physics - 1
Spectrum and Cross Sections

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NRC Fast Reactor Technology Training Curriculum
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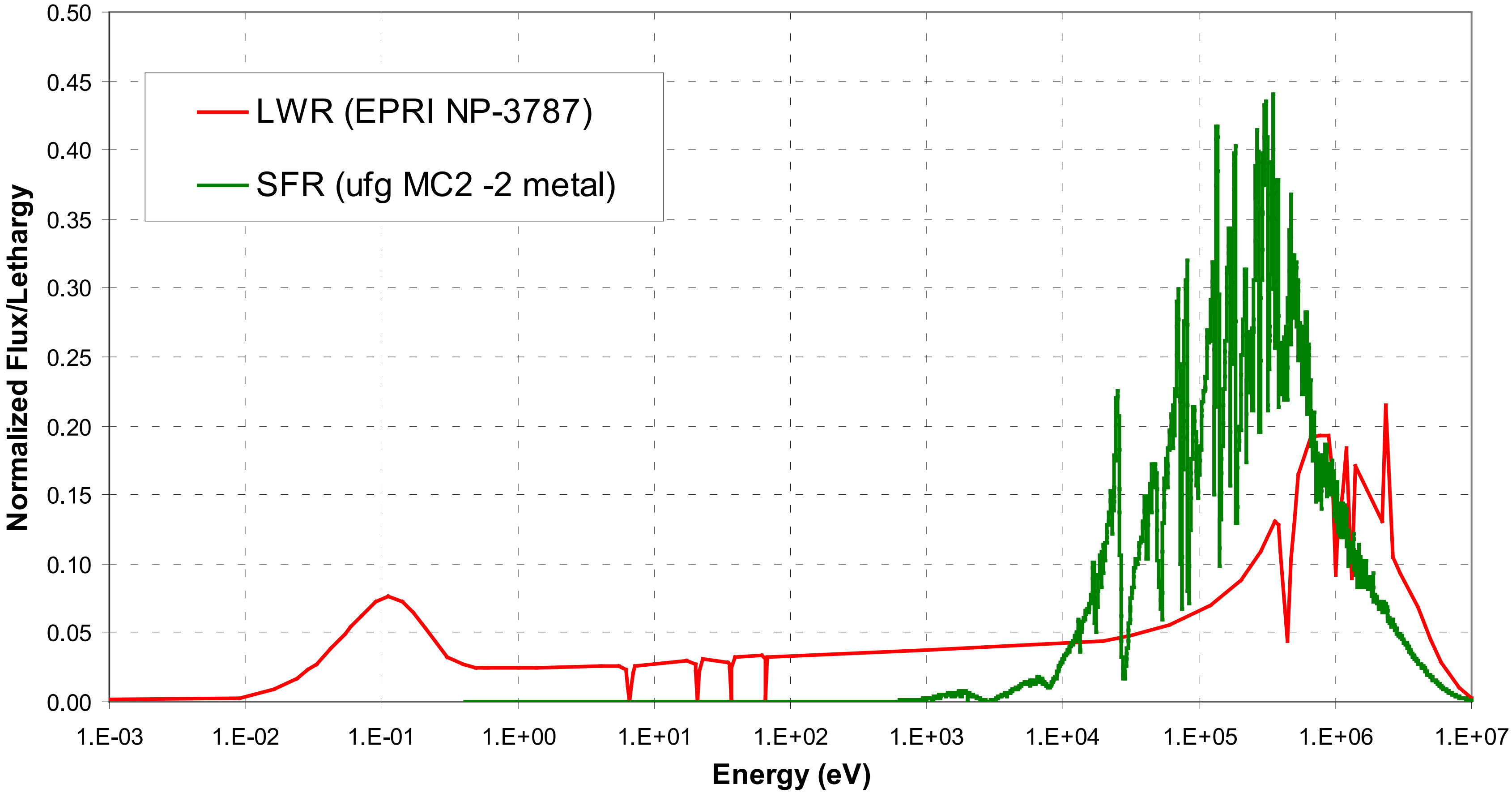
Outline

- Introduction to Fast Reactor Physics
  - Fast energy spectrum
    - Neutron moderation
  - Impacts of fast energy spectrum
    - Fission-to-capture, enrichment, etc.
  - Comparison to LWR neutron physics

- Nuclear Data and Validation
  - Typical Power Distributions
  - Anticipated uncertainties
  - Modern covariance and data adjustment

Comparison of LWR and SFR Spectra



- In LWR, most fissions occur in the 0.1 eV thermal “peak”
- In SFR, moderation is avoided – no thermal neutrons; most fissions occur at higher energies (>10 keV)

Neutron Moderation Comparison

- Significant elastic scattering of the neutrons after fission
- In FRs, neutron moderation is avoided by using high A materials
  - Sodium is most moderating
- In LWRs, neutrons are moderated primarily by hydrogen
- Oxygen in water and fuel also slows down the neutrons

	σ_s (barn)	N (#/barn·cm)	$\xi\Sigma_s$ (cm^{-1})
TRU	4.0	3.2E-03	1.1E-04
U	5.6	5.6E-03	2.7E-04
Zr	8.1	2.6E-03	4.6E-04
Fe	3.4	1.9E-02	2.3E-03
Na	3.8	8.2E-03	2.7E-03
H	11.9	2.9E-02	3.5E-01

- Slowing-down power in FR is ~1% that observed for typical LWR
- Thus, neutrons are either absorbed or leak from the reactor before they can reach thermal energies

Fast Reactor Moderating Materials

- Lead has highest scattering of the neutrons, but little moderation
- Oxide fuel in SFR leads to a significant moderation effect – resonances also observed
- LFR designs also utilize nitride or oxide fuel forms
- In modern GFR designs, SiC matrix for fuel is utilized, with most moderation of the FR cases

	σ_s (barn)	N (#/barn·cm)	$\xi\Sigma_s$ (cm^{-1})
TRU	4.0	3.2E-03	1.1E-04
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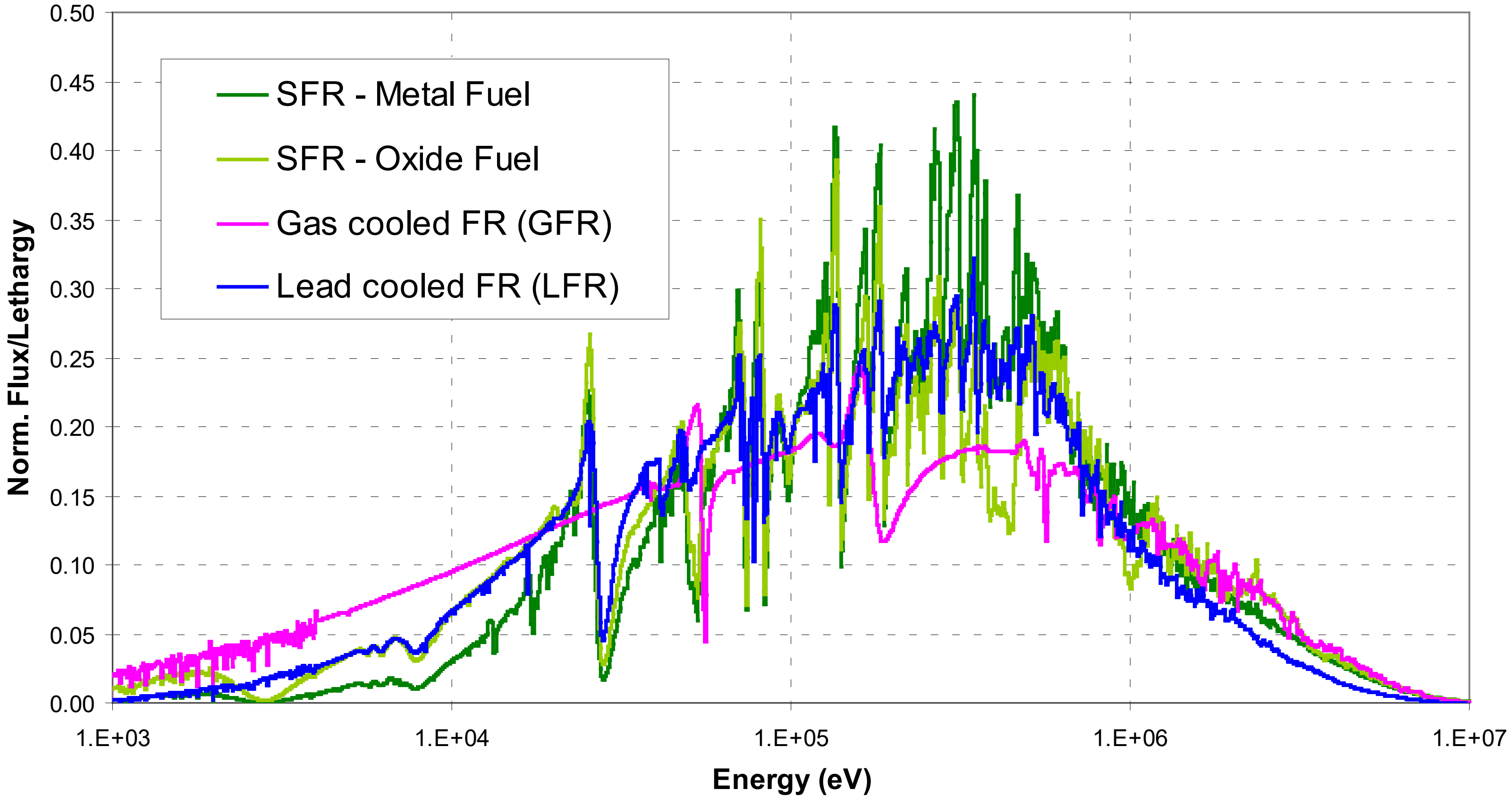
Zr	8.1	2.6E-03	4.6E-04
Fe	3.4	1.9E-02	2.3E-03
Na	3.8	8.2E-03	2.7E-03
O	3.6	1.4E-02	5.8E-03
Pb	8.6	1.6E-03	1.3E-04
C	3.9	1.6E-02	1.0E-02
He	1.7	3.1E-04	2.3E-04

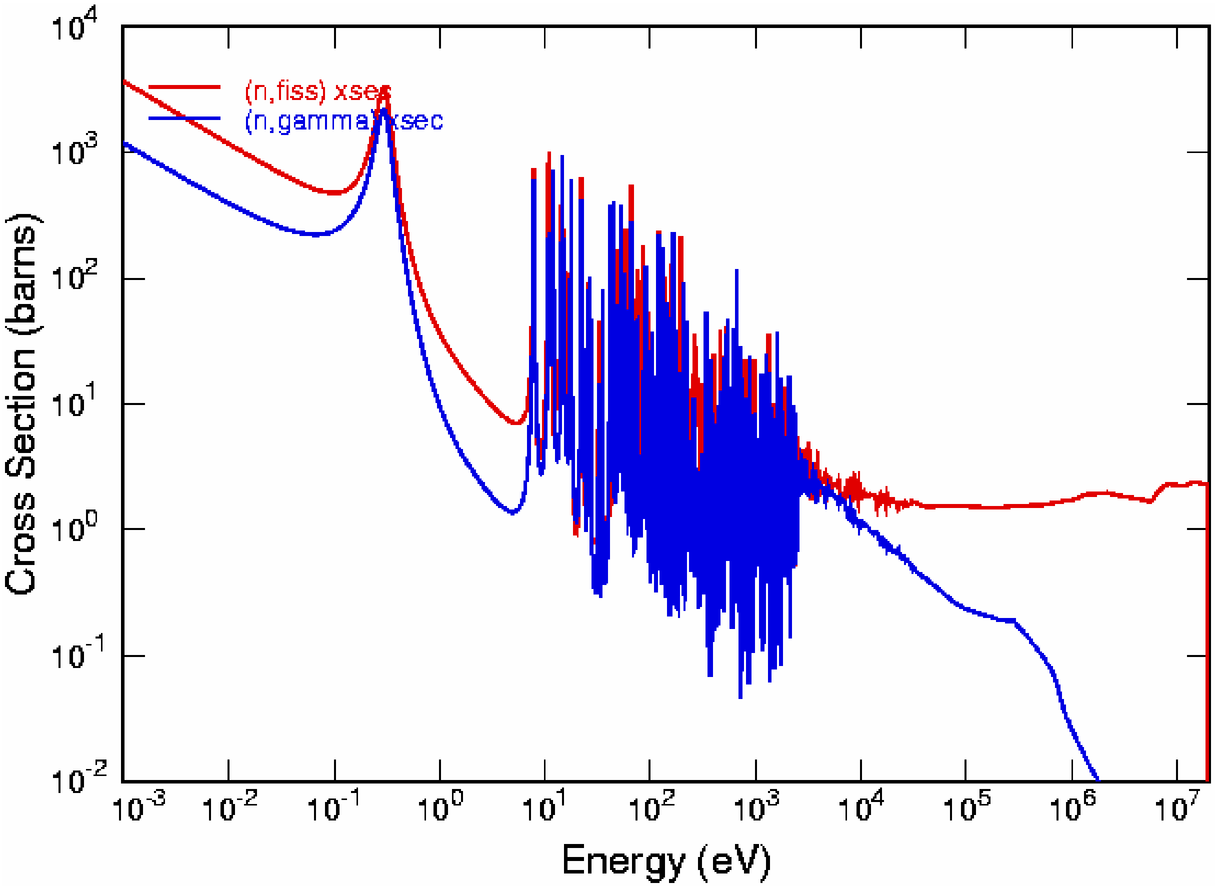
- Net result is that the SFR-metal has the hardest neutron spectrum
  - SFR-oxide and LFR similar with slightly more moderation
- GFR (with silicon-carbide matrix) has significant low energy tail

Comparison of Fast Reactor Spectra



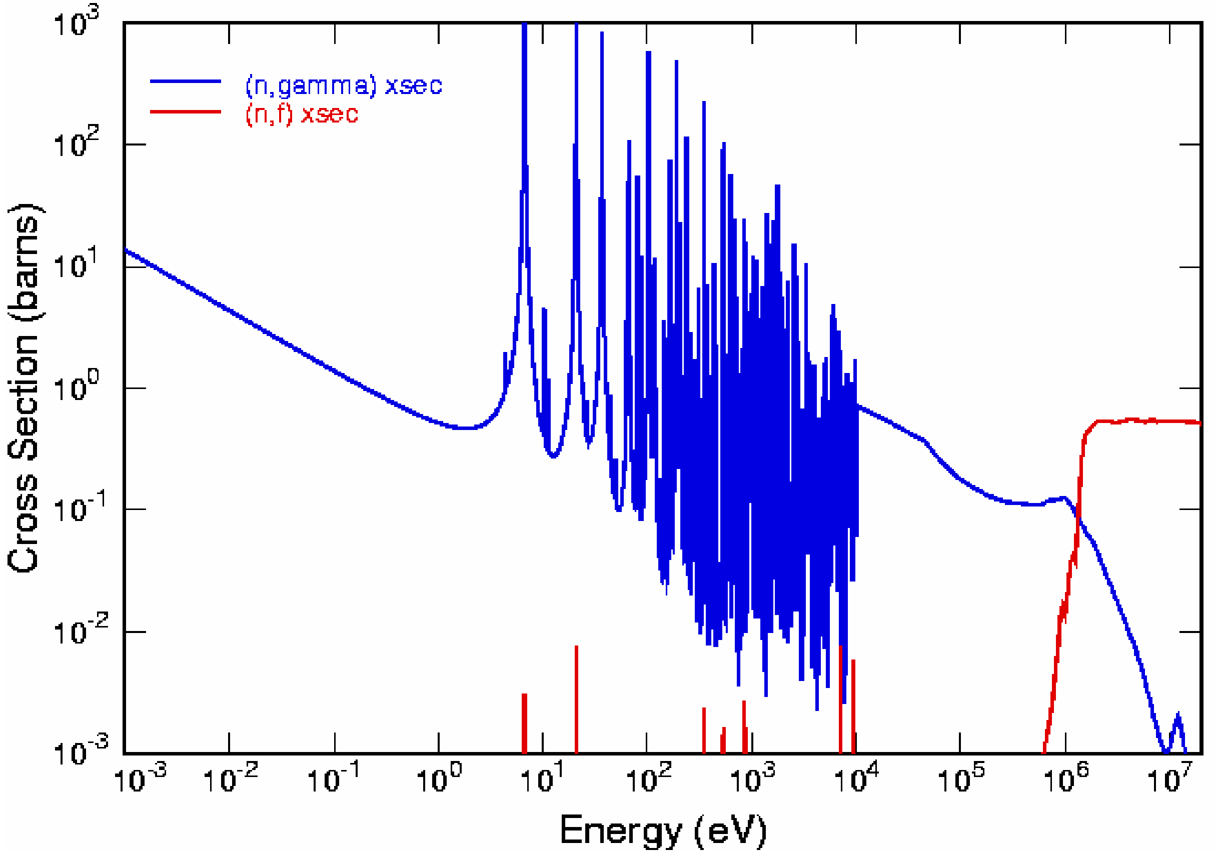
- Predictable spectral differences between fast reactor concepts
- Low energy tail caused by moderating materials (carbon in GFR)
- At high energy (>1 MeV) lead is effective inelastic scattering material
- Characteristic resonances from oxygen, iron are evident

Spectral Variation of Neutron Cross Sections: Pu-239



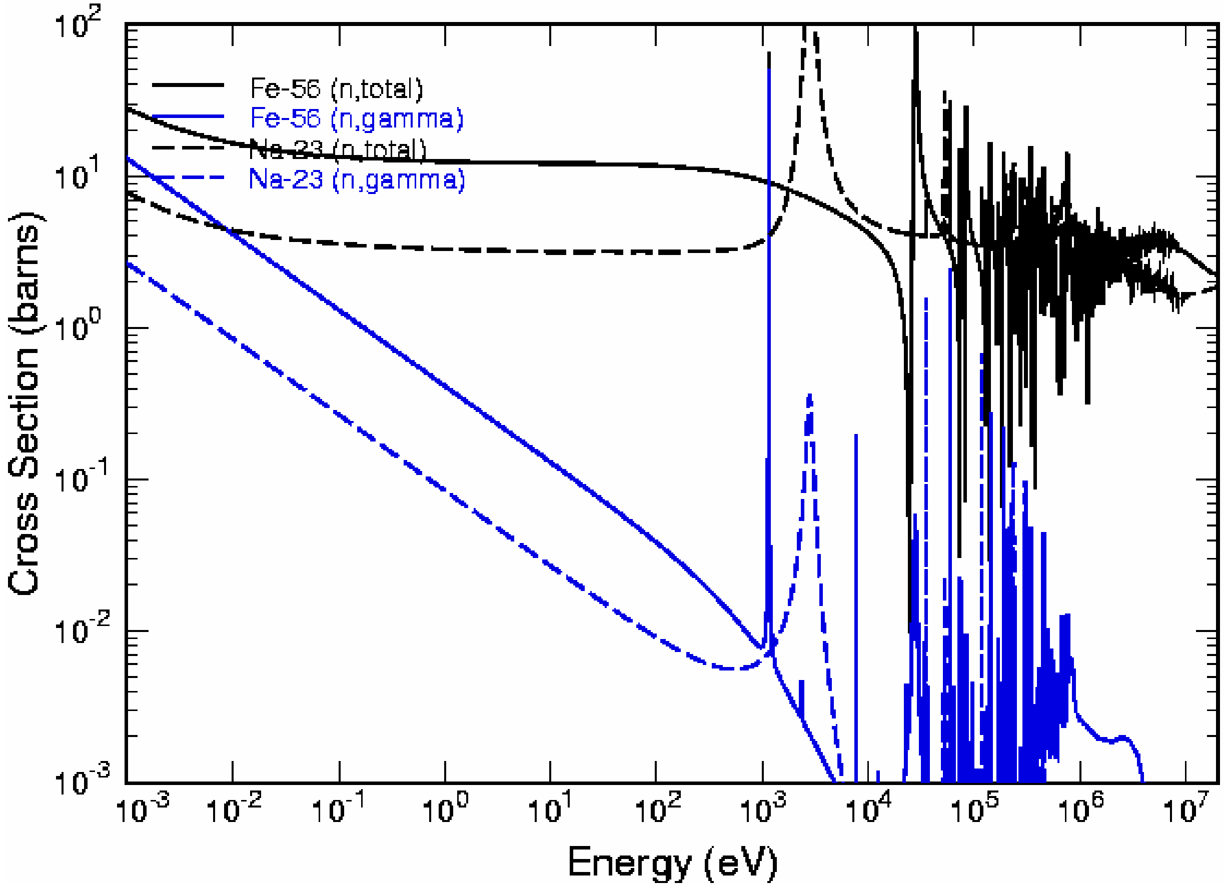
- Fission and capture cross section >100X higher in thermal range
- Sharp decrease in capture cross section at high energy

Spectral Variation of Neutron Cross Sections: U-238



- Much smaller thermal increase in capture (~10X)
- Unresolved resonance range begins at ~10 keV
- Threshold fission at ~1 MeV (~10% of total fission rate in a fast reactor)

Spectral Variation of Neutron Cross Sections: Fe and Na



- Capture cross sections much higher in thermal range
- Significant scattering resonance structure throughout fast range

Fission Probabilities (Fission/Absorption)

Isotope	Fast Reactor (Metal fuel)	Thermal Reactor
U235	0.80	0.81
U238	0.17	0.10
Np237	0.27	0.02
Pu238	0.70	0.08
Pu239	0.86	0.64
Pu240	0.55	0.01
Pu241	0.87	0.75
Pu242	0.52	0.02
Am241	0.21	0.01
Am243	0.23	0.01
Cm244	0.45	0.06

- Fast fission fraction is a bit higher for Pu-239/241, similar for U-235
- However, for non-fissile isotopes in thermal reactor, almost always (>90%) neutron capture, with possible later fission as a higher A fissile isotope

Equilibrium Composition in Fast and Thermal Spectra

Isotope	Fast Reactor	Thermal Equil.	Thermal Once-thru
Np237	0.008	0.002	0.048
Pu238	0.014	0.046	0.024
Pu239	0.666	0.388	0.476
Pu240	0.243	0.197	0.225
Pu241	0.021	0.111	0.106
Pu242	0.018	0.085	0.066
Am241	0.021	0.019	0.034
Am242m	0.001	0.001	0.000
Am243	0.005	0.033	0.015
Cm242	0.000	0.002	0.000
Cm244	0.002	0.055	0.005
Cm245	0.000	0.018	0.000
Cm246	0.000	0.031	0
Cm247	0.000	0.004	0
Cm248	0.000	0.006	0

- Equilibrium higher actinide content much lower in fast spectrum system
- Generation of Pu-241 (key waste decay chain) is suppressed
- Once-through LWR composition has not reached thermal higher actinide (Am/Cm) equilibrium, but higher than fast reactor (U-238) equilibrium

Salvatores et al, "Fuel Cycle Analysis of TRU or MA Burner Fast Reactors with Variable Conversion Ratio, Nuclear Engineering and Design, **219,** 2160 (2009)

Impact of Energy Spectrum on Enrichment and Parasitic Absorption

Reaction	Thermal Concepts (barns)			Fast Concepts (barns)		
	PWR	VHTR	SCWR	SFR	LFR	GFR
U238c	0.91	**4.80**	0.95	0.20	0.26	0.32
Pu239f	89.2	164.5	138.8	1.65	1.69	1.90
P239f/U238c	97.7	34.3	146.6	**8.14**	**6.59**	**6.00**
Fec	0.4			**0.007**		
Fission Prod.c	90			**0.2**		

- One-group cross sections are significantly reduced in fast system
- Generation-IV fast systems have similar characteristics
- U-238 capture is much more prominent (low P239f/U238c) in fast reactors
  - *Therefore, a higher enrichment is required to achieve criticality*
- The parasitic capture cross section of fission products and conventional structural materials is much higher in a thermal spectrum

Neutron Balance

		PWR	SFR	
			CR=1.0	CR=0.5
U-235 or TRU enrichment, %		**4.2**	**13.9**	**33.3**
Source	fission	100.0%	99.8%	99.9%
	(n,2n)		0.2%	0.1%
Loss	leakage	3.5%	**22.9%**	**28.7%**
	radial	3.0%	12.3%	16.6%
	axial	0.4%	10.6%	12.1%
	absorption	96.5%	77.1%	71.3%
	fuel	76.7%	71.8%	62.2%
	(U-238 capture)	(27.2%)	(31.6%)	(17.1%)
	coolant	**3.4%**	0.1%	0.1%
	structure	0.6%	3.7%	3.7%
	fission product	**6.8%**	1.5%	2.4%
	control	9.0%	0.0%	2.9%

- Conversion ratio defined as ratio of TRU production/TRU destruction
  - Slightly different than traditional breeding ratio with fissile focus

Summary of Fast Spectrum Physics Distinctions

- Combination of increased fission/absorption and increased number of neutrons/fission yields more excess neutrons from Pu-239
  - Enables "breeding" of fissile material
- In a fast spectrum, U-238 capture is more prominent
  - Higher enrichment (TRU/HM) is required (neutron balance)
  - Enhances internal conversion
- Reduced parasitic capture and improved neutron balance
  - Allows the use of conventional stainless steel structures
  - Slow loss of reactivity with burnup
    - Less fission product capture and more internal conversion
- The lower absorption cross section of all materials leads to a much longer neutron diffusion length (10-20 cm, as compared to 2 cm in LWR)
  - Neutron leakage is increased (>20% in typical designs, reactivity coefficients)
  - Reflector effects are more important
  - Heterogeneity effects are relatively unimportant

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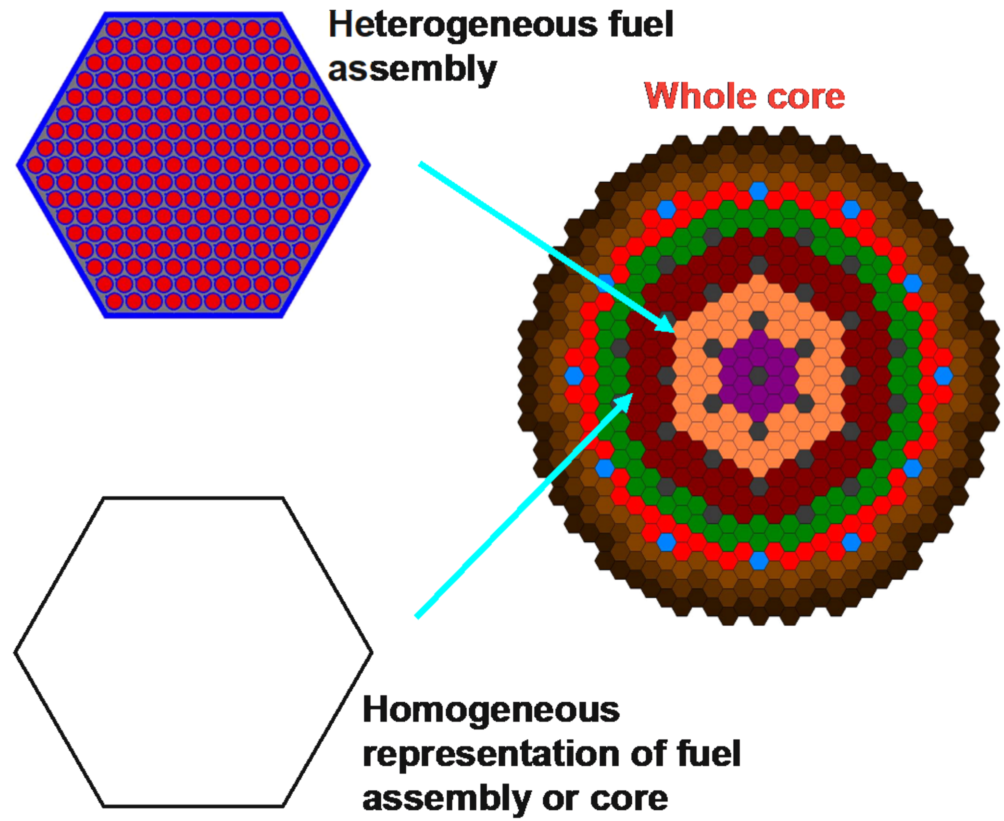
Typical Design Specifications of LWR and SFR

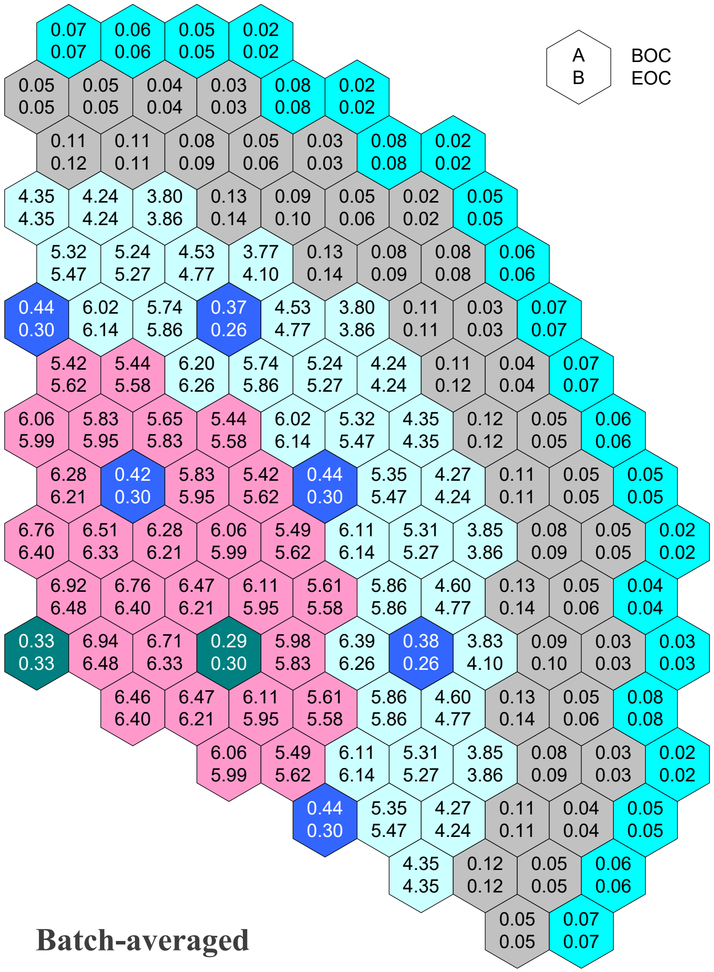
			PWR	SFR
General	Specific power (kWt/kgHM)		786 (U-235)	556 (Pu fissile)
	Power density (MWt/m^3)		102	300
Fuel	Rod outer diameter (mm)		9.5	7.9
	Clad thickness (mm)		0.57	0.36
	Rod pitch-to-diameter ratio		1.33	1.15
	Enrichment (%)		~4.0	~20 Pu/(Pu+U)
	Average burnup (MWd/kg)		40	100
Thermal Hydraulic	Coolant	pressure (MPa)	15.5	0.1
		inlet temp. (°C)	293	332
		outlet temp. (°C)	329	499
		reactor Δp (MPa)	0.345	0.827
	Rod surface heat flux	average (MW/m^2)	0.584	1.1
		maximum MW/m^2)	1.46	1.8
	Average linear heat rate (kW/m)		17.5	27.1
	Steam	pressure (MPa)	7.58	15.2
		temperature (°C)	296	455

Conventional Approximations

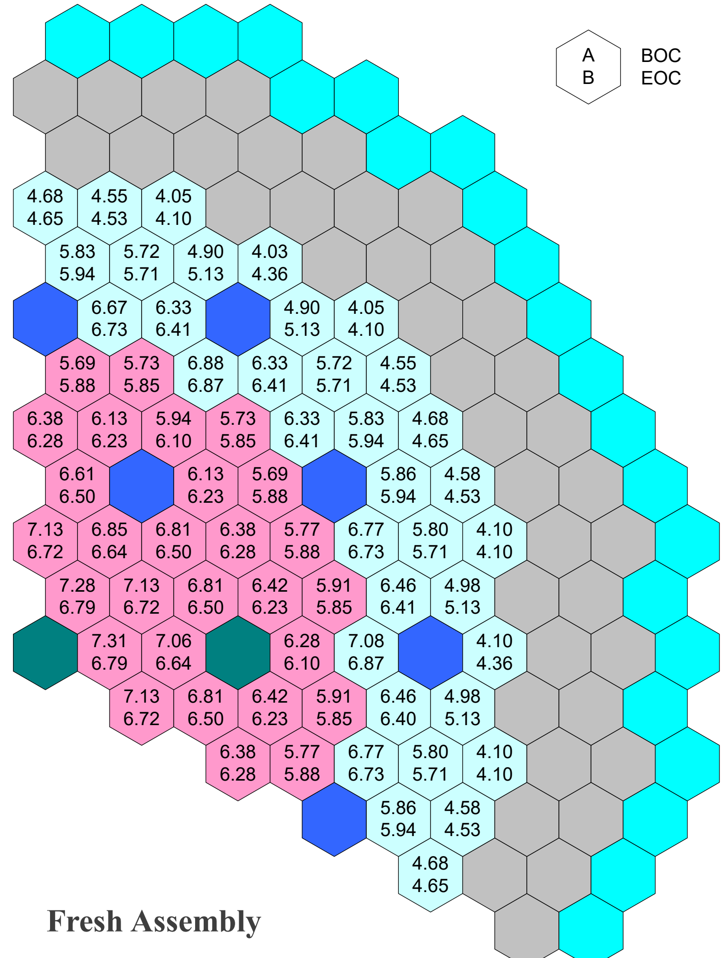
- **Detailed space-energy-direction analysis performed for a repeated portion of the geometric domain (lattice physics)**
  - Lower order approximation of Boltzmann equation for whole-core analysis
  - Assumed/approximate boundary conditions
  - Condensed space/energy cross sections
  - Detailed information recovered by de-homogenization methods
- **Nuclide depletion and buildup using quasi-steady model**
  - Depletion steps are ~ hours – days
- **Faster transients modeled with condensed (often single point) model for time-dependent amplitude**
  - Accuracy depends on frequency of kinetic-parameter and space-energy-direction flux shape recalculation



Assembly Power (MW) of 1000 MWt ABR – Startup Core



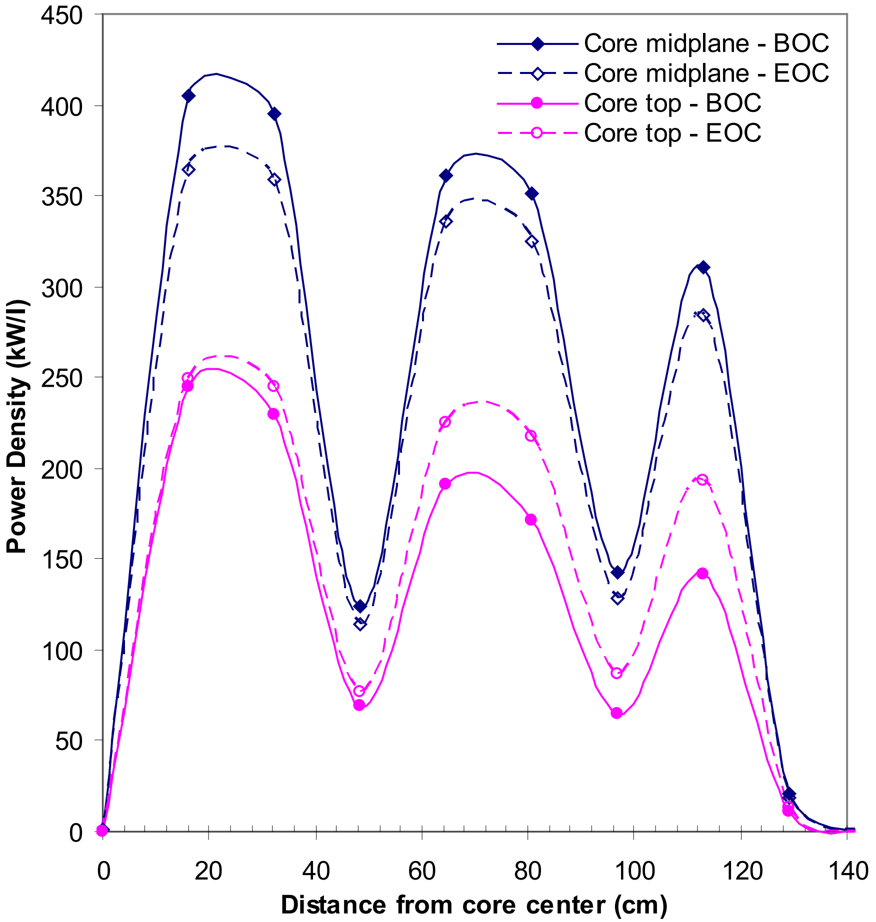
Batch-averaged Assembly Power

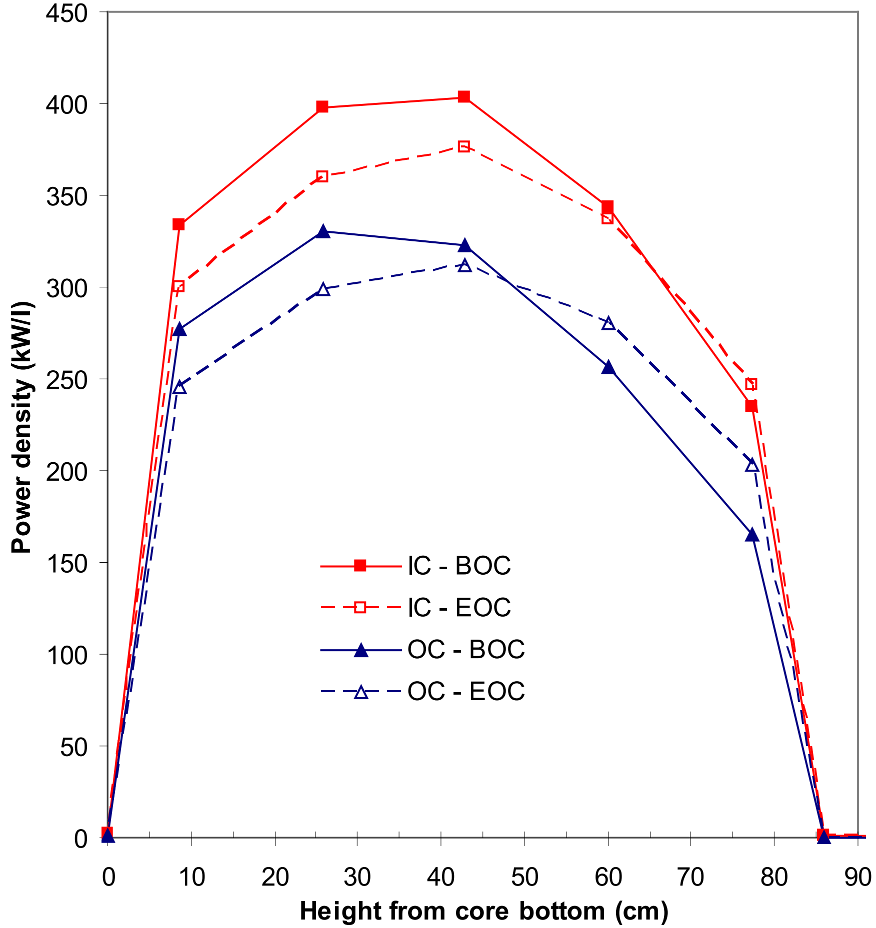


Fresh Assembly Power

- Power peaking factor (BOC/EOC) = 1.43 /1.39

Power Profiles of 1000 MWt ABR - Startup Core





- Bottom skewed axial distribution (control assembly tip position at BOC = 57 cm)
- Axial power peaking factor at BOC/EOC = 1.20/1.16

Estimated Target Accuracies for Fast Reactors

Parameter	Current	Desired
Multiplication factor, k_{eff}	~1%	<0.2-0.3%
Relative Power density		
Peak	~2%	~1%
Distribution	5-7%	2%
Control rod worth		
Element	10%	5%
Total	5%	1-2%
Burnup reactivity swing (of reactivity value)	3% or 0.5%Δk	<1% or 0.3%Δk
Breeding gain	0.05-0.06	0.02-0.04
Reactivity coefficients		
Large effects	10%	5%
Small effects	20%	10%
Kinetics parameters	5-10%	2-5%
Local nuclide densities		
Major constituents	5%	2%
Minor constituents	10-20%	10%

- Two main sources of uncertainties
  - *Physical data*: Cross sections, fabrication data, etc.
  - *Modeling*: Approximations in computational methodology of design process
- High-fidelity simulation can reduce modeling uncertainties
- Modern data and analysis techniques will allow:
  - Reliable propagation of uncertainties
  - Correct evaluation of impacts of uncertainty from input data

Covariance Data Status

Idaho National Laboratory

- Despite significant efforts in generating new high quality neutron cross section data and in producing associated covariance matrices, the state of affairs is not yet fully satisfactory. The user is puzzled by many inconsistencies among evaluated cross sections and corresponding covariance data that in many cases fail to explain discrepancies between measurements and calculations for integral experiments.
- As an example, the K_{eff} of the well known and simple system JEZEBEL, calculated by the three major libraries (ENDF/B, JEFF, and JENDL) is equal to 1 within experimental uncertainties. However, an analysis by component finds that there are huge (thousands of pcm) compensations among reactions (inelastic, fission, χ) and energy range. These differences are not covered, at the one sigma level, by the uncertainty analysis using the associated covariance data.
- Noticeable differences have been observed among the covariance matrices in use not only on individual cross section uncertainties, but also on correlation among data.
- Some covariance data are still missing and in particular: secondary energy distribution for inelastic cross sections (multigroup transfer matrix), cross correlations (reactions and isotopes), delayed data (nubar and fission spectra).
- However, serious and reliable efforts are still going on for improving covaraiance data and a lot of progress has been made in the last 15 years by the nuclear data community.

ZPPRs C/E: ENDF/B-VIII vs. JEFF-3.3

EXPERIMENT	JEFF-3.3 (C-E)/E	ENDF/B-VIII (C-E)/E	ENDF/B-VIII Uncert. %	JEFF-3.3 Uncert. %
ZPPR-9 K_{eff}	0.717	-0.178	1.220	1.183
ZPPR-9 F28/F25	-5.443	-0.266	8.017	3.285
ZPPR-9 C28/F25	-2.757	-0.322	1.546	1.399
ZPPR-9 VOID STEP 3	8.347	4.325	7.638	6.065
ZPPR-9 VOID STEP 5	4.646	0.802	9.881	8.053
ZPPR-10 K_{eff}	0.781	-0.105	1.135	1.171
ZPPR-10 VOID STEP 2	24.222	19.221	7.006	6.189
ZPPR-10 VOID STEP 3	13.359	8.798	7.070	6.324
ZPPR-10 VOID STEP 6	11.881	7.358	7.952	7.146
ZPPR-10 VOID STEP 9	9.551	5.087	9.058	8.218
ZPPR-10 Central Control Rod reactivity	6.085	6.166	1.611	1.948
ZPPR-15 K_{eff}	1.221	-0.004	0.985	1.242

Questions?