Several Illustrative Examples of Nuclear Data Needs for Nuclear Energy Systems

Jasmina Vujic
Nuclear Science and Security Consortium (NSSC)
University of California, Berkeley

May 27, 2015
Research Project Title: Nuclear Science and Security Consortium

Award Institution: DOE NNSA

Lead Organization: University of California, Berkeley

Participating Universities: MSU, UCD, UCI, UCSD-IGCC, UNLV, WUSTL

Participating Laboratories: LBNL, LANL, LLNL, SNL

Lead PI: Prof. Jasmina Vujic, University of California, Berkeley

Executive Director: Prof. Karl van Bibber, University of California, Berkeley

Deputy Executive Director: Prof. Brad Sherrill, Michigan State University

External Advisory Board Chair: Dr. Jay Davis, President of the Hertz Foundations

Award Amount: $25 million for 5 years (2011 – 2016) +$1.5M for MSI

Lab Mentorship for NSSC Students: $125 k/per year/per lab
Nuclear Science and Security Consortium

Goals

• Support multiyear research projects which are of a basic or fundamental nature that do not necessarily align with programmatic missions of DOE/NNSA but are critical to maintaining the discipline of nuclear science and security.

• Enable collaborative research relationships between universities, the national laboratories, and other government agencies.

• Transition technology from universities to national laboratories.

• Motivate talented researchers toward careers in nuclear security applications.

• Recruit broadly, focusing on disciplinary excellence, not necessarily immediate relevance to specific NA-22 problems.

• Select those who combine (i) broad perspective, (ii) solid science & engineering foundation, (iii) highly developed specialization.
Nuclear Science and Security Consortium

PI: Jasmina Vujic (UCB)
Assoc. Dir.: Bethany Goldblum (UCB)
Exec. Dir.: Karl van Bibber (UCB)
Deputy ED: Brad Sherrill (MSU)
Dir. For Labs: Ed Hartouni (LLNL)
NNSA Liaison: Kai Vetter (UCB-LBNL)
A Science-Technology-Policy Interface

Nuclear & Particle Physics
Focus Area Lead: E.B. Norman

Nuclear Chemistry
Focus Area Lead: H. Nitsche

Nuclear Engineering
Focus Area Lead: R. Slaybaugh

Detection & Instrumentation
Focus Area Lead: K. Vetter
Nuclear Physics Focus Area

- Basic Nuclear Structure Physics with GRETINA
- Neutron Physics using a Modular Neutron Array (MoNa)
- Beta-Delayed Neutron Studies
- CUORE Double Beta Decay
- Anti-Neutrino Reactor Monitoring
- Low Background Measurements
- Nuclear Data

Eric Norman
Dept. Nuclear Engineering
UC Berkeley
Focus Area Lead
Radiochemistry Focus Area

Ken Czerwinski
Dept. of Chemistry
University of Nevada, Las Vegas
Focus Area Lead

- isotope ratio measurements
- actinides in soil samples
- radiochemical separations
- fallout sample characterization
- heavy and superheavy elements
- molecular nuclear forensics
Radiation Detection & Instrumentation
Focus Area

Kai Vetter
Dept. Nuclear Engineering
UC Berkeley
Focus Area Lead

- gamma-ray imaging systems
- position sensitive HPGe detectors
- image reconstruction and 3D data fusion
- coherent elastic neutrino-nucleus scattering with Ge
- background characterization with RadWatch and RadMap
Nuclear Engineering Focus Area

Rachel Slaybaugh
Dept. Nuclear Engineering
UC Berkeley
Focus Area Lead

- modeling and simulation
- high performance computing
- detector material characterization
- beta-delayed gamma ray analysis
- novel scintillators

APTS sol gel

Inverse Analysis of Delayed Gamma Spectra

WARP Results
Nuclear Security Policy Focus Area

- cross domain deterrence
- international cooperation on nuclear security
- network science for nonproliferation
- advanced detectors for international safeguards
- Nuclear Policy Working Group (NPWG) – New Chapters!

Cross Domain Deterrence

Network Science

Michael Nacht
Public Policy
UC Berkeley
Focus Area Lead

Nuclear Policy Working Group
**NSSC Lifetime Support**

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*To date (4.28.15). Final Year 4 numbers pending.*
## Affiliate Involvement/Impact

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*Numbers include Year 4 to date (5.15.15)
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*To date: 5.15.15. Year 4 numbers pending.
NSSC Fellows & Affiliates hired at National Laboratories

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*Includes both postdoctoral and staff positions at the labs for NSSC fellows and affiliates; Year 4 data included to date (5.13.15)
★ 7 affiliates
NSSC Status - Summary

• NSSC is running successfully at “full load” for four years
• More than 370 people engaged in NSSC supported research and activities
• 25 NSSC fellows hired at national laboratories to date
• NSSC undergraduate students are transitioning to NSSC graduate students
• Strong relationships between national laboratory scientists and students and post-docs working at national laboratories
  – NSSC PIs and students are collaborating with over 60 national laboratory scientists
• Successful summer schools held for three years in a row
  – 19 total summer schools delivered from 2012 - 2015
  – 6 NSSC supported summer schools planned for Summer 2015
• MSI process executed
  – 18 summer internship and scholarships for MSI students awarded to date
  – 29 research proposals received and reviewed
  – 5 MSI research proposals awarded
Nuclear Reactor Design and Analysis

• Variety of nuclear reactor designs:
  – Based on fast, epithermal, thermal neutron spectra
  – Nuclear fuel materials, and structural materials
  – Various coolants and moderators
  – Various operating temperatures

• Generation IV and Beyond – very different issues

• Nuclear Physics is typically incorporated into reactor simulation codes through nuclear data libraries

• There are a variety of Nuclear Data Libraries:
  – ENDF (USA), JENDL (Japan), JEFF (Europe), BRONDL (Russia)

• Regardless of many decades invested in the cross section library development, all those libraries contain approximations, inaccuracies, and produce discrepancies when compared.

• Nuclear data libraries could be further improved with improvements in nuclear theory and relevant experiments
What Parts of Fuel Cycle are Affected by Uncertainties in Nuclear Data?

- Reactor Core and Fuel Design
- Safety and safety margins
- Criticality safety
- Shielding
- Radiation damage in fuel and structural materials
- Decay heat produced in reactor shut-down
- Decay heat produced in the repository
- Long term spent nuclear fuel analysis
- Spent nuclear fuel reprocessing and recycling options
- Nuclear Materials detection

Uncertainties in nuclear data libraries propagate to uncertainties in calculated integral quantities, increasing safety margins and increasing costs in advanced nuclear reactor designs
Generations of Nuclear Reactor Designs

- **Generation I**: Early Prototype Reactors
  - Shippingport
  - Dresden, Fermi I
  - Magnox

- **Generation II**: Commercial Power Reactors
  - LWR-PWR, BWR
  - CANDU
  - AGR

- **Generation III**: Advanced LWRs
  - ABWR
  - System 80+

- **Generation III+**: Evolutionary Designs Offering Improved Economics for Near-Term Deployment

- **Generation IV**: Highly Economical
  - Enhanced Safety
  - Minimal Waste
  - Proliferation Resistant

**Timeline**:
- Gen I: 1950
- Gen III: 2000, 2010
- Gen III+: 2020
- Gen IV: 2030
## Existing Commercial Nuclear Reactors

<table>
<thead>
<tr>
<th>Врста</th>
<th>Описание</th>
<th>У раду</th>
<th>У изградњи</th>
<th>Угашени</th>
<th>У плану</th>
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<tr>
<td>PWR</td>
<td>Pressurized Water Reactor (light water cooled and moderated) и ВВЭР</td>
<td>270</td>
<td>53</td>
<td>38</td>
<td>101</td>
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<td>BWR</td>
<td>Boiling Water Reactor (light water cooled and moderated)</td>
<td>84</td>
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<td>PHWR</td>
<td>Pressurized Heavy Water moderated and cooled Reactor</td>
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<td>GCR</td>
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<td>LWGR</td>
<td>Light Water cooled, Graphite moderated Reactor (РБМК)</td>
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<td>HTGR</td>
<td>High Temperature Gas-cooled Reactor (graphite moderated)</td>
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<td>HWGCR</td>
<td>Heavy Water moderated, Gas Cooled Reactor</td>
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<td>HWLWR</td>
<td>Heavy Water moderated, boiling Light Water cooled Reactor</td>
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<td>SGHWR</td>
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<td>FBR</td>
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<td>Other</td>
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<td></td>
<td></td>
<td>2</td>
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<td>Укупно</td>
<td></td>
<td>435</td>
<td>65</td>
<td>143</td>
<td>114</td>
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The recognition of the major potential of fast neutron systems with closed fuel cycle for breeding (fissil regeneration) and waste minimization (minor actinide burning)
# Proposed Generation IV Nuclear Reactors

<table>
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<tr>
<th>Reactor Type</th>
<th>Neutron Spectra//Coolant/Fuel</th>
<th>Inlet/Outlet Coolant Temp/Pressure</th>
<th>Fuel Cycle</th>
<th>Size/Power MWth</th>
<th>Applications</th>
<th>Research and Development</th>
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<td>Sodium-cooled Fast Reactor (SFR)</td>
<td>Fast Sodium Metal Alloy or Oxide</td>
<td>550°C outlet 1 atm</td>
<td>Closed</td>
<td>Med to Large 1000-5000</td>
<td>Electricity, Actinide Mgmt. (AM)</td>
<td>Advanced Recycle</td>
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<td>Lead-alloy Fast Reactor (LFR)</td>
<td>Fast Pb-Bi Metal alloy/ Nitride</td>
<td>550-800°C outlet 1 atm</td>
<td>Closed</td>
<td>Small to Large 125-3600</td>
<td>Electricity, Hydrogen Production</td>
<td>Fuels, Materials compatibility</td>
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<tr>
<td>Gas-Cooled Fast Reactor (GFR)</td>
<td>Fast Helium UPuC/SiC (70/30%)</td>
<td>490°C inlet 850°C outlet 90 bar</td>
<td>Closed</td>
<td>Med 600</td>
<td>Electricity, Hydrogen, AM</td>
<td>Fuels, Materials, Safety</td>
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<td>Supercritical Water Reactor (SCWR)</td>
<td>Thermal, Fast Water</td>
<td>280°C inlet 510-550°C outlet 25 MPa</td>
<td>Open, Closed</td>
<td>Large 1700MWe</td>
<td>Electricity</td>
<td>Materials, Safety</td>
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<td>Molten Salt Reactor (MSR)</td>
<td>Thermal Fluoride salts UF</td>
<td>565°C inlet 700-850°C outlet</td>
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<td>Large 1000MWe</td>
<td>Electricity, Hydrogen, AM</td>
<td>Fuel, Fuel treatment, Materials, Safety and Reliability</td>
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Some Studies Performed

• Nuclear Data needs for Gen-IV and other advanced reactor systems

• Sensitivity study for parameters affected the most by nuclear data uncertainties: Multiplication factor, Power peak, Burnup $\Delta k/k$, Coolant void reactivity coefficient, Doppler reactivity coefficient, Nuclide density at the end of cycle (transmutation potential), Neutron source at fuel fabrication, and Dose in a repository.


### Table 1. Fast Neutron Systems: Total Uncertainties (%).

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<th>Reactor</th>
<th>( k_{\text{eff}} )</th>
<th>Power Peak</th>
<th>Doppler</th>
<th>Void</th>
<th>Burnup [pcm]</th>
<th>Decay Heat</th>
<th>Dose</th>
<th>Neutronic Source</th>
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<td>EFR</td>
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(a) Partial Energy Correlation as used in ref. [1]
(b) BNL_ORNL_LANL_NRG_ANL
Salvatores et al. Results

- The contribution of the fission product uncertainty (due to “lumped” FPs) to the overall burnup reactivity is significant only in the case of a fast reactor with an extended burnup.

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<th>System</th>
<th>SFR</th>
<th>EFR</th>
<th>GFR</th>
<th>LFR</th>
<th>VHTR</th>
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<td>±381</td>
<td>±198</td>
<td>±530</td>
<td>±851</td>
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<td>Fission Products</td>
<td>±73</td>
<td>±755</td>
<td>±130</td>
<td>±76</td>
<td>±215</td>
<td>±244</td>
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<tr>
<td>Total</td>
<td>±282</td>
<td>±1153</td>
<td>±402</td>
<td>±212</td>
<td>±572</td>
<td>±885</td>
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Fuel Cycle Options*

*Base cases in red italics*

Once Through:

*Build ALWR/ Current Burnup (50 MWD/kg)*

Limited Thermal Reactor Recycle:

*PUREX-based one time recycling of U-Pu as mixed oxides (MOX) to LWRs*

Fast Reactor Recycle of all transuranics, TRU(metallic fueled reactors studied by ANL and GE):

*TRU to self-sustaining FR (Conversion Ratio =1)*

*TRU recycle in fast burner ABR (with low CR = 0.75)*

*TRU recycle in fast breeder FBR (with CR = 1.23)*

*M. Kazimi, MIT, 2010*
Many measurements have been identified in the fast reactor sensitivity calculations

- **Fission Cross Section Measurements**
  - Np237, Pu238, Pu239, Pu240, Pu241, Pu242, Am241, Am242m, Am243, Cm244, Cm245

- **Capture Cross Section Measurements**
  - Si28, Fe56, B10
  - Np237, U238, Pu239, Pu240, Pu242, Am241, Am242, Am243, Cm242, Cm244, Cm245

- **Inelastic Cross Section Measurements**
  - Na23, U238, Fe56

- **Fission neutron spectrum and multiplicity**
  - Pu238, Pu239, Pu240

The measurements and required accuracies are EXTREMELY challenging
Fig. 1. Analogy in thorium and uranium fertilization.
## Thorium-Based Fuel Cycle – Experience*

<table>
<thead>
<tr>
<th>Country</th>
<th>Reactor</th>
<th>Capacity</th>
<th>Fuel composition</th>
<th>Time</th>
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<tbody>
<tr>
<td>Germany</td>
<td>HTGR (Pebble bed)</td>
<td>15 MW(e)</td>
<td>Th + $^{235}$U, Coated Oxide &amp; carbides</td>
<td>1967–1988</td>
</tr>
<tr>
<td>Germany</td>
<td>The same</td>
<td>300 MW(e)</td>
<td>The same</td>
<td>1985–1989</td>
</tr>
<tr>
<td>Germany</td>
<td>BWR</td>
<td>60 MW(e)</td>
<td>Fuel (Th,Pu)O$_2$ pellets</td>
<td>1968–1973</td>
</tr>
<tr>
<td>UK, Sweden</td>
<td>HTGR (Pin-in-Block Design)</td>
<td>20 MW(t)</td>
<td>Th + $^{235}$U Driver, Coated fuel particles, Oxide &amp; Dicarbides</td>
<td>1966–1973</td>
</tr>
<tr>
<td>USA</td>
<td>HTGR (Prismatic Block)</td>
<td>40 MW(e)</td>
<td>The same</td>
<td>1966–1972</td>
</tr>
<tr>
<td>USA</td>
<td>MSBR</td>
<td>7.5 MWt</td>
<td>$^{233}$U Molten Fluorides</td>
<td>1964–1969</td>
</tr>
<tr>
<td>USA</td>
<td>BWR (Pins)</td>
<td>24 MW(e)</td>
<td>Th + $^{235}$U Fuel Oxide</td>
<td>1963–1968</td>
</tr>
<tr>
<td>USA</td>
<td>LWBR PWR (Pins)</td>
<td>100 MW(e)</td>
<td>Th + $^{233}$U Driver Fuel, Oxide Pellets</td>
<td>1977–1982</td>
</tr>
<tr>
<td>USA</td>
<td>The same</td>
<td>285 MW(e)</td>
<td>The same</td>
<td>1962–1980</td>
</tr>
<tr>
<td>Canada</td>
<td>MTR (Pins)</td>
<td>20-200 MW</td>
<td>Th + $^{235}$U, Test Fuel</td>
<td>1947–1957</td>
</tr>
<tr>
<td>India</td>
<td>MTR Thermal</td>
<td>40 MW(t); 100 MW(t)</td>
<td>Al + $^{233}$U Driver Fuel, Th &amp; ThO$_2$</td>
<td>1960–2010</td>
</tr>
<tr>
<td>India</td>
<td>PHWR (Pins)</td>
<td>220 MW(e)</td>
<td>ThO$_2$ Pellets</td>
<td>1980–pres.</td>
</tr>
<tr>
<td>India</td>
<td>LMFBR (Pins)</td>
<td>40 MW(t)</td>
<td>ThO$_2$ blanket</td>
<td>1985–pres.</td>
</tr>
</tbody>
</table>

*IAEA TECDOC-1450
Example from the 2012 Serpent User Group Meeting in Madrid

- Presentation by Manuele Aufiero, Politecnico di Milano
- Work on modifications of the Serpent Monte Carlo code to study the fuel isotopic evolution of molten salt reactors designed for continuous reprocessing
- Needed to determine conversion ratios (CR)
- Noticed a big discrepancy in the capture cross section for U-233 between JEFF 3.1 and ENDF/B 7.x
- Noticed also a discrepancy in the capture cross section for Pa-233 between JEFF 3.1 and ENDF/B 7.x above a few keV
- Both discrepancies lead to higher CRs when JEFF 3.1 is used as compared to the ENDF/B 7.x results
MSFR modelling in SERPENT

- Fuel salt initial composition: 
  \[ LiF - ThF_4 - UF_4 \] or 
  \[ LiF - ThF_4 - (Pu - MA)F_3 \]
- Blanket salt initial composition: \( LiF - ThF_4 \)
- \( Ni-based \) alloy for vessel and reflectors
- Gaseous & insoluble FP\(s \) extraction with time constants \( \sim \) tens of seconds (30s in the reference scenario)
- Few liters of salt reprocessed each day (40l in the reference scenario)

- \( 50 \cdot 10^6 \) neutron histories for equilibrium calc.
- \( 10 \cdot 10^6 \) neutron histories for transient calc.
- MPI = 6, OMP = 5, 1000+100 cycles
- URES activated only for main isotopes
- opti mode = 3

Thermal power: 3 GW
Example from the 2012 Serpent User Group Meeting in Madrid

**JEFF vs ENDF: \(^{233}U\) production**

Huge difference in the fuel cycle prediction between JEFF-3.1 and ENDF/B-VII.

Solid lines: JEFF-3.1 – Dashed line: ENDF/B-VII

Good agreement for the Uranium production only in the blanket.
Example from the 2012 Serpent User Group Meeting in Madrid

JEFF vs ENDF: equilibrium CR

From JEFF-3.1 to ENDF/B-VII, break-even reprocessing rate prediction doubles.

\(^{233}U\) Fission to capture ratio

Good agreement between the libraries almost everywhere...
Example from the 2012 Serpent User Group Meeting in Madrid

233\textsuperscript{U} Fission to capture ratio

...NOT in the MSFR energy spectrum region.
ENDF/B and JEFF disagree on U-233
Disagreement for Pu-239

Comparison of $^{239}$Pu $\alpha$ from different evaluations

- ENDF/B–VI.8 (2001)
- ENDF/B–VII.0 (2006)
- JEFF–3.1.1 (2009)
- ENDF/B–VII.1 (2011)
No Disagreement for U-235

Comparison of $^{235}\text{U} \alpha$ from different evaluations

- ENDF/B–VI.8 (2001)
- ENDF/B–VII.0 (2006)
- JEFF–3.1.1 (2009)
- ENDF/B–VII.1 (2011)

Capture to fission ratio vs. Neutron Energy [MeV]
UCB NE Advanced Reactor Design Projects - Current

- Collaborating Faculty: Ehud Greenspan, Max Fratoni, Jasmina Vujic

<table>
<thead>
<tr>
<th>Project</th>
<th>Researchers (no longer with project)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. NEUP: Thorium-based Fuel-self-sustaining RBWR &amp; TRU transmuting RBWR</td>
<td>Phillip Gorman, PhD, Sandra Bogetic, PhD, George Zhang, PhD, Jeff Seifried, Post Doc, Christopher Varela, MSc</td>
</tr>
<tr>
<td>2. NEUP: Seed-and-Blanket Liquid-Metal Reactors (S&amp;B SFR)</td>
<td>George Zhang, PhD, Staffan Qvist, Post Doc, Christian DiSanzo, PhD, Alejandra Jolodosky, MSc</td>
</tr>
<tr>
<td>3. NEUP: 3-D fuel shuffling in Breed and Burn (B&amp;B) reactors (Pebble-bed B&amp;B cores)</td>
<td>Phillip Gorman, MSc and NE-265 project team, Jason Hou, Post Doc, Staffan Qvist, Post Doc</td>
</tr>
<tr>
<td>4. Synergism between B&amp;B, S&amp;B and LWR fuel cycles</td>
<td>Christian DiSanzo, PhD (Graduated)</td>
</tr>
<tr>
<td>5. → 2-Tier and 3-Tier fuel cycles</td>
<td></td>
</tr>
<tr>
<td>6. Autonomous Reactivity Control (ARC) system for fast reactors (NEUP 2015?)</td>
<td>Staffan Qvist, Post Doc, Meg Suvdantsetseg, PhD, visiting from KTH</td>
</tr>
</tbody>
</table>
Concerns about Hitachi RBWR Design

Concerns:
- Positive void coefficient (turned out to be negative)
- Very high LHGR – too low safety margins
- Axial power instabilities
- Very high peak burnup and high fast neutron fluence

Our (NEUP) Approach:
- Use thorium instead of depleted uranium as the primary fertile fuel
  - Greatly reduce positive spectral component of void reactivity
  - Do not have to rely on enhanced neutron leakage probability
- Use longer seeds and eliminate internal blanket

Collaborators: UoM (Downar), MIT (Kazimi), BNL (Todosow)
Sensitivity and uncertainty analysis for nuclear energy systems

Jeffrey Seifried
Postdoctoral Researcher
Department of Nuclear Engineering
University of California, Berkeley

Sather workshop on the thorium fuel cycle and nuclear data
November 20, 2013
Three Parts to Sensitivity and Uncertainty Analysis

1. Quantify the sensitivity of results \( R \) to nuclear data \( p \)

\[
S_{R,p} \equiv \frac{\partial R}{\partial p} \frac{p}{R} \approx \frac{\delta R}{R} \frac{p}{\delta p}
\]

2. Estimate the covariance of those nuclear data \( \text{cov}[p] \)

\[
\text{ENDF6 MF=33}
\]

3. Collapse to estimate result uncertainty

\[
\text{var} [R] = \sum_p \langle S_{R,p} \mid \text{cov} [p] \mid S_{R,p} \rangle
\]
Direct sampling is the obvious, but expensive approach

- Its procedure is straightforward ...
  1. Directly perturb an input (-5%, 0%, +5%)
  2. Perform an entire perturbed simulation (depletion)
  3. Extract perturbed results
  4. Quantify result sensitivities

- ... but it is extremely expensive
  - 12 [isotopes]
    × 8 [reactions/isotope]
    × 50 [energy regions/reaction]
    = ~5,000 [uncertainty inputs]
    = ~5,000 [depletion calculations]!
  - Monte Carlo counting uncertainties must not swamp nuclear data defects
Two tools for adjoint-based S/U analysis

- **SCALE/TSUNAMI** is more convenient and mature
  - Built-in covariance libraries
  - Automated inner products
  - Many extraneous tools for analysis
  - Dancoff factors
  - Multi-group Monte Carlo (forward and adjoint) transport
  - Very slow!

- **MCNP6/KSEN** is much faster (but still slow)
  - Continuous-energy Monte Carlo (forward only) transport
  - Efficient forward estimator for the adjoint distribution
  - Parallelized transport
  - Matrix operations must be done by hand
The S/U analysis example for static multiplication factors for the PBFHR

- ... for a single pebble unit cell ...
  - TRISO pebble fuel
  - 20% enriched $^{235}\text{U}$ oxy-carbide
  - Immersed in $^7\text{Li}$-enriched flibe
  - Infinitely hexagonal lattice

- ... on my work desktop ...
  - Intel i7-2600 @ 3.4 GHz
  - 4 GB RAM

- ... took 1 week!
Results

Nuclear Reaction Uncertainty Contribution to $k_{\text{eff}}$ @ BOC (1.48070 ± 0.00816)

- Easy
- Important
- Obvious
Issues with Lead Cross Sections

• UCB NE worked on the design of two lead-bismuth cooled reactors:
  • The Encapsulated Nuclear Heat Source (ENHS) is a new conceptual designs of small lead-bismuth or lead cooled reactors with natural circulation.
  • The International Atomic Energy Agency (IAEA) proposed a Coordinated Research Programme (CRP) on ”Development of Small Reactors without On-site Refuelling”. The RBEC-M lead-bismuth cooled fast reactor benchmark is suggested for this purpose.
  • The depletion benchmark problem was prepared based on the RBEC-M core, which is a 900 MW(th) lead-bismuth cooled fast reactor conceptual design developed by the Russian Research Centre, ”Kurchatov Institute” (RRC KI).
The MCNP and ORIGEN2.1 utility codes interfaced by the MOCUP driver were used.

The continuous energy MCNP library based on the ENDF/B-VI.8, ENDF/B-VII.0, JEFF-3.1 and JENDL-3.3 evaluations was prepared for all lead nuclides.

The largest differences between various evaluations were observed for $^{208}\text{Pb}$ cross section data. For other lead nuclides ($^{204}\text{Pb}$, $^{206}\text{Pb}$ and $^{207}\text{Pb}$) the modern evaluations converge to the JEFF-3.1 evaluation.

It was also found our hat ENDF/B-VI did not have data for Pb-204.

Comparison for Lead

Effective multiplication factor ($k_{eff}$) vs. Fuel burnup [Days]

- Red: Lead ENDF/B-VI.8
- Purple: Lead ENDF/B-VII.0
- Cyan: Lead ENDF/B-V.2
- Blue: Lead JEFF-3.1
- Brown: Lead JENDL-3.1
- Yellow: Lead JENDL-3.3
- Green: Lead EXFOR (Vinca-1.0)

MCNP-4C/ORIGEN-2.1 with ENDF/B-VI.8 based library.
Three core zones model, and time steps equal to 200 days.
Results of this analysis, given in Figure 5 show:

- a good agreement between calculations based on the ENDF/B-VI.8, ENDF/B-VII.0, JEFF-3.1 and evaluation [13] founded on the EXFOR data for lead;
- a notable difference (about -600 pcm) between calculations based on the ENDF/B-VI.8 and older evaluated cross section data for lead (ENDF/B-V.2 and JENDL-3.1); and
- a notable difference (about 1000 pcm) between calculations based on the ENDF/B-VI.8 and JENDL-3.3 evaluated cross section data for lead (due to slightly higher values of $^{207}$Pb elastic cross section data in the JENDL-3.3 evaluation in comparison with the ENDF/B-VI.8, ENDF/B-VII.0 and JEFF-3.1 evaluation).
Comparison for Bismuth

MCNP-4C/ORIGEN-2.1 with ENDF/B-VI.8 and JEFF-3.1 (Lead) based library. Three core zones model, and time steps equal to 200 days.
The ENHS Benchmark


• The ENHS is a lead-bismuth or lead cooled novel reactor concept that is fuelled with metallic alloy of Pu, U and Zr, and is designed to operate for 20 effective full power years without refuelling and with very small burnup reactivity swing.

• A significant difference (about 1500 pcm) was found in k-eff between the ENDF/B-VI.8 and JENDL-3.3 evaluations due to a slightly higher values in JENDL-3.3 evaluation for elastic cross section of $^{207}\text{Pb}$. 
The ENHS Benchmark –Pb-206

One fuel zone 2D r-z model, $^{206}$Pb coolant

Effective multiplication factor

Burnup time [Year]

- ENDF/B-VI.8
- JEFF-3.1
- JENDL-3.3
The ENHS Benchmark –Pb-207

One fuel zone 2D r-z model, $^{207}$Pb coolant

Effective multiplication factor

Burnup time [Year]

ENDF/B-VI.8
JEFF-3.1
JENDL-3.3
The ENHS Benchmark –Pb-207

One fuel zone 2D r-z model, $^{208}$Pb coolant

Effective multiplication factor

Burnup time [Year]
Bauge* highlighted the uncertainty in reaction databases for \((n, n_{el})\) and \((n, n')\) in a prompt fission neutron spectrum

*E. Bauge et al.,
SUMMARY

• Although the quality of the main evaluated data libraries mentioned in this presentation is high, there is still a lot of work to be done.

• There examples (particularly in nuclear criticality experiments) that good results are obtained mainly due to compensation of errors, as shown in recently presented uncertainty analysis of Jezebel criticality experiment.

• The CIELO paper (Chadwick et al., Nuclear Data Sheets 118, 1-24, 2014), lists and analyses some important nuclides
  – Light elements (H-1, O-16),
  – Structural materials (Fe-56), inelastic scattering, (n,xn), (N,xp), (n,alpha)
  – U-235, Pu-239
  – U-238

• Thorium and its isotopes also very important
Acknowledgement

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