SCALE 5 Analysis of GGU2 Sample

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Outline

- ORNL participation to high-burnup spent fuel programs
- SCALE 5 depletion analysis capabilities
  - 2-D depletion capabilities by TRITON / NEWT
  - Recent developments in ORIGEN-S
- Analysis of assay data for the GGU2 sample
  - Computational model
  - Preliminary results
- Key modeling issues in assay data analyses

ORNL Participation to High-Burnup Spent Fuel Programs

- **US Nuclear Regulatory Commission (NRC) initiated programs for investigating high-burnup spent fuel in order to:**
  - Evaluate uncertainties in isotopics for high burnup spent fuel and assess their potential impact on various applications
  - Extend the existing regulatory guidance related to transport and storage of spenf fuel, currently limited to lower burnups and enrichments
- **ORNL projects on high–burnup spent fuel**
  - Access, analyze, and evaluate new experimental assay data for high burnup spent fuel
  - Understand and quantify the uncertainties associated with computer code predictions for high burnup spent fuel isotopics
  - Model predictions in domains of increased enrichment, burnup, and cooling times
  - Evaluate the data to quantity the uncertainties in predicted decay heat, radiation source terms, and nuclide concentrations used in burnup credit
- **ORNL projects on burnup credit**
  - ORNL is leading an effort to establish a technical basis for fission product burnup credit in the United States.
  - Ongoing activities include:
    - Acquiring spent nuclear fuel assay data that include fission products of interest to burnup credit through international collaborations and domestic programs
    - Evaluating available experimental measurements with SCALE for a large range of burnups, enrichments, assembly designs, in order to assess the associated computational bias.

ORNL Participation to High-Burnup Spent Fuel Programs

- ORNL is involved in three international programs designed to acquire new experimental assay data (UO_2 fuel of primary interest):
  - ✓**MALIBU** program
    - ✓coordinated by Belgonucleaire
    - ✓samples from Gösgen reactor (Switzerland)
    - ✓measurement laboratories: SCK-CEN (Belgium), PSI (Switzerland), CEA (France)
  - ✓**REBUS** program
    - ✓coordinated by Belgonucleaire
    - ✓1 sample from Neckar GKN II reactor (Germany)
    - ✓measurement laboratory: SCK-CEN (Belgium)
  - ✓**ENUSA/CSN** program
    - ✓coordinated by ENUSA/CSN (Spain)
    - ✓samples from Vandellòs reactor (Spain)
    - ✓measurement laboratory: Studsvik Nuclear (Sweden)
    - ✓new set of measurements for new samples scheduled to start in Summer 2006 (includes fission products of interest to burnup credit such as ^{103}Rh, $^{103,106}Ru$, ^{99}Tc)

Experimental Data

Reactor (country)	Reactor type	Assembly design	Measurement laboratory	Enrichment (wt % ^{235}U)	No. of samples	Burnup (GWd/MTU)
Takahama (Japan)	PWR	17 x 17	JAERI (Japan)	4.11	10	14.3 – 47.3
Calvert Cliffs (USA)	PWR	15 x 15	PNNL (USA) Khlopin Institute (Russia)	2.45, 2.72 2.45, 3.04	5 4	18.7 – 46.7 27.4 – 44.3
Gösgen (Switzerland)	PWR	15 x 15	SCK-CEN/ PSI/ CEA (Belgium)	4.3 4.1	2 3	47.2, 68 29.1 – 59.7
Vandellos II (Spain)	PWR	17 x 17	Studsvik Nuclear (Sweden)	4.50	7	42.9 – 73.8
Neckar GKN II (Germany)	PWR	18 x 18	SCK-CEN (Belgium)	3.8	1	54.0
Novovoronezh (Russian Fed.)	VVER	Type 440	RIAR (Russia)	3.59	8	23 – 47
TMI-1 (USA)	PWR	15 x 15	ANL (USA)	4.013	11 (1 rod)	45.7 – 52.1
TMI-1 (USA)	PWR	15 x 15	GE-VNC (USA)	4.657	8 (3 rods)	25-3 – 31.4
H. B. Robinson (USA)	PWR	15 x 15	ANL (USA)	2.89	2	74 – 76

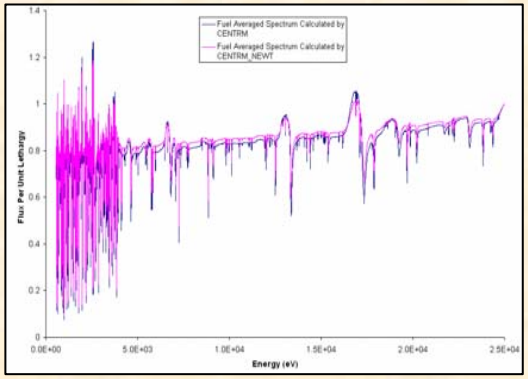
** Proprietary data

SCALE 5 Depletion Analysis Capabilities

- **ORIGEN-S (Oak Ridge Isotope GEneration code, SCALE version)**
  - Point depletion, decay, and radiation source-term analyses.
- **TRITON** (depletion module)
  - Allows 2-D depletion through coupling of the 2-D arbitrary polygonal mesh transport code NEWT with ORIGEN-S
  - Additional 3-D depletion capabilities using KENO V.a and KENO-VI Monte Carlo codes will be available with SCALE 5.1
  - TRITON/NEWT generates burnup-dependent cross sections, with a branch capability to provide cross-sections at each burnup step for perturbations in moderator density, fuel and moderator temperatures, boron concentration, and control rod insertion or removal
  - Performs pin-by pin depletion of different materials
  - Generates few-group cross-section data (including discontinuity factors) for use in subsequent nodal diffusion calculations

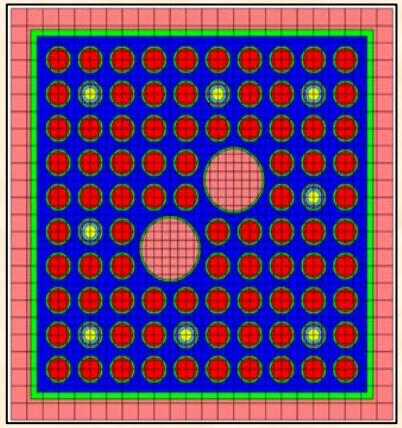
SCALE 5 Depletion Analysis Capabilities

TRITON - **T**ransport **R**igor **I**mplemented with **T**ime-dependent **O**peration for **N**eutronic depletion



CENTRM (Energy Detail)

+



NEWT (Spatial/Angular Detail)

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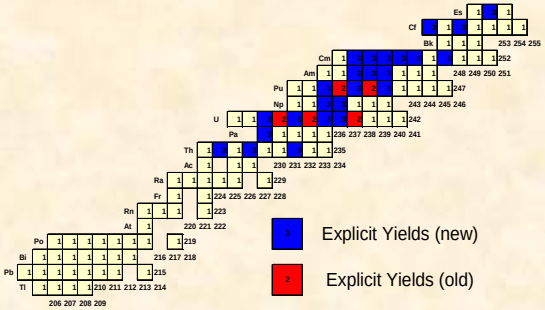


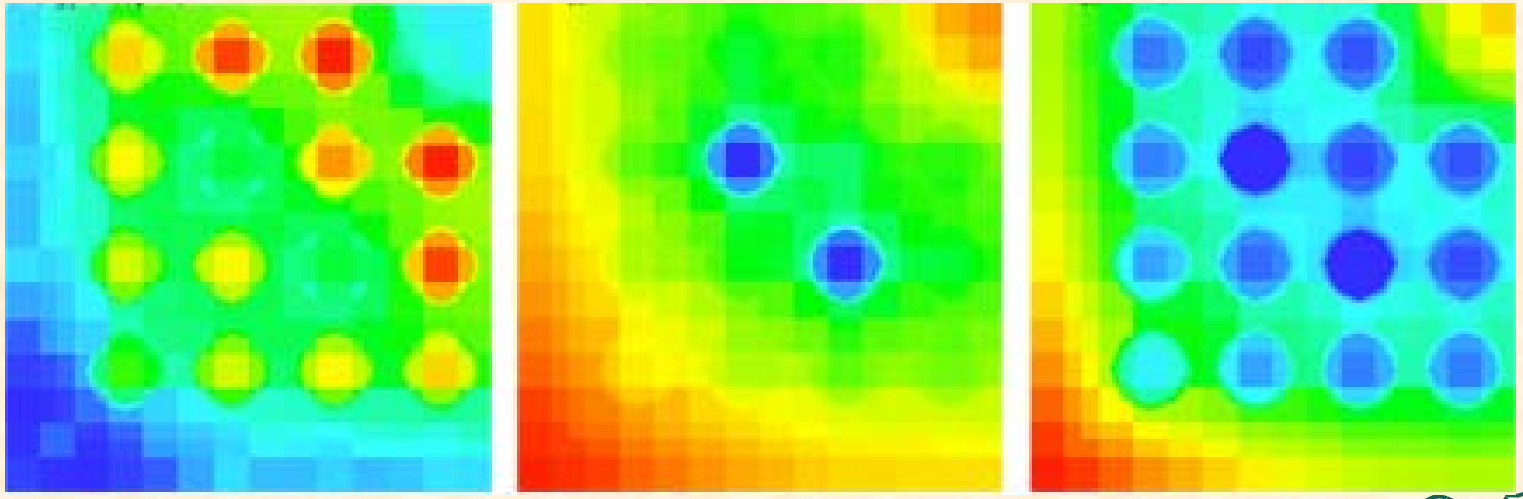
ORIGEN-S (Isotopic Detail, 1700 nuclides)

SCALE 5 Depletion Analysis Capabilities

- **CENTRM – Continuos ENergy TRansport Module**
  - Calculates a continuous energy neutron spectrum for each unit cell by solving the 1-D transport equation using a combination of pointwise and multigroup nuclear data
  - The resulting flux spectrum is employed in the PMC module as a problem-dependent weight function to generate multigroup cross sections for use with NEWT, for each nuclide in each material and in each unit cell
  - This approach allows explicit handling of resonance self-shielding effects and other spectral effects over a given energy domain

SCALE 5 Depletion Analysis Capabilities

- **NEWT - A <u>NEW</u> <u>T</u>ransport Algorithm**
  - Discrete ordinates solution based on Extended Step Characteristic (ESC) differencing scheme
  - Computational cells may be any arbitrary polygon
  - Automated grid generation based on specification of simple bodies
  - Provides eigenvalue/fixed source, forward/adjoint, and transverse (axial) leakage correction solution options



SCALE 5 Depletion Analysis Capabilities

- **Recent Developments in Nuclear Data for ORIGEN-S**
  - **Fission product yields**
    - ✓30 fissionable actinides
    - ✓ENDF/B-VI data
  - **Nuclear decay data**
    - ✓1119 fission products
    - ✓data and decay schemes primarily from ENDF/B-VI
    - ✓additional decay data obtained from ENSDF and JEF-2.2
  - **Gamma emission data**
    - ✓1132 nuclides with explicit photon yield data, 115,000 discrete photon lines
    - ✓data derived from of ENDF/B-VI, ENSDF, and JEF-2.2 evaluations
  - **Neutron emission data**
    - ✓primarily from data files distributed with the LANL SOURCES code
    - ✓spontaneous fission source, (α,n) source, delayed neutrons source
  - **Nuclear reaction cross section data**
    - ✓ENDF/B-VI, FENDL-2.0 and EAF-99 data
    - ✓include 698 activation nuclides, 129 actinides, 1119 fission products

Analysis of Assay Data for GGU2 Sample

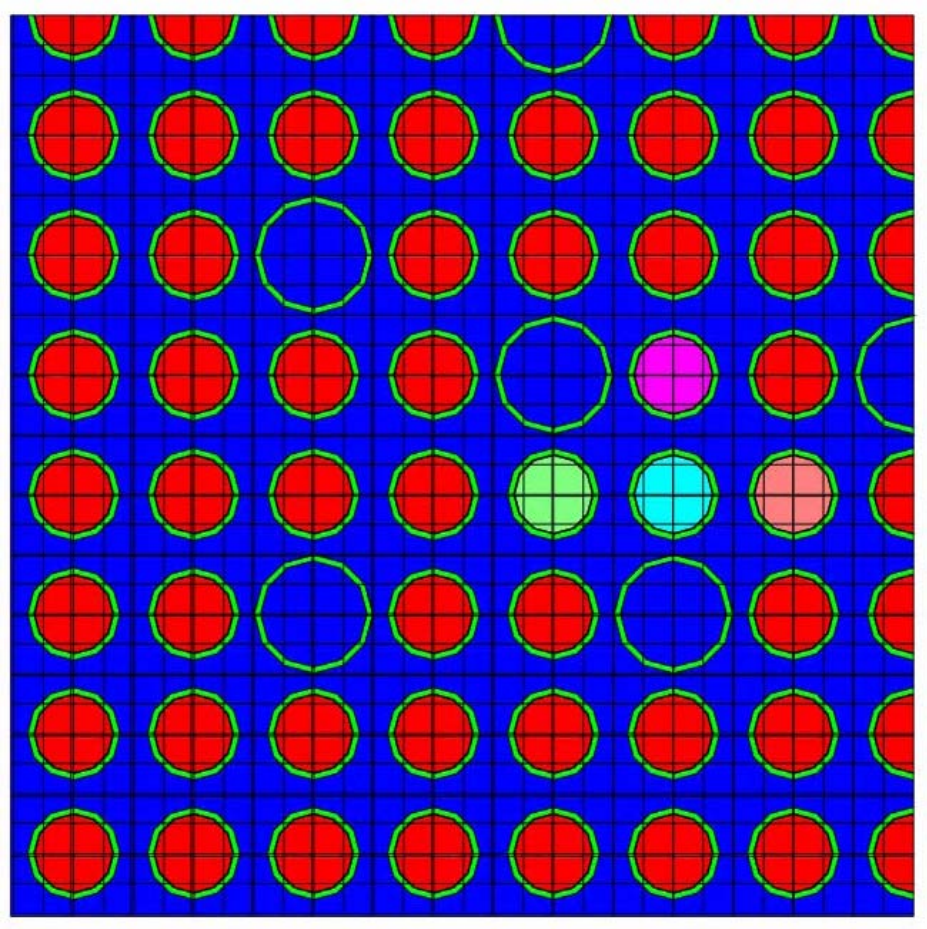
- **Assembly Data**
  - 15x15 design, with 205 fuel rods and 20 guide tubes
  - irradiated for 4 cycles in the Gosgen PWR reactor (cycle 19 to 22)
  - calculation of sample power was based on the load factor and sample burnup data provided in the Irradiation Data Report
  - each reactor cycle was split in four depletion steps for the convenience of specifying the variation of the boron content in the coolant and to account for the cycle power variation

- **Computational Model**
  - ¼ assembly modeled, given the symmetry
  - sample pin and surroundings pins depleted individually
  - sample power normalized to the ^{148}Nd content (~2% difference vs. nominal power)

- **Experimental Data**
  - comparison to preliminary data of measurements reported by SCK-CEN

Analysis of Assay Data for GGU2 Sample

TRITON Model for GGU2 Sample



sample pin

sample pin neighbors

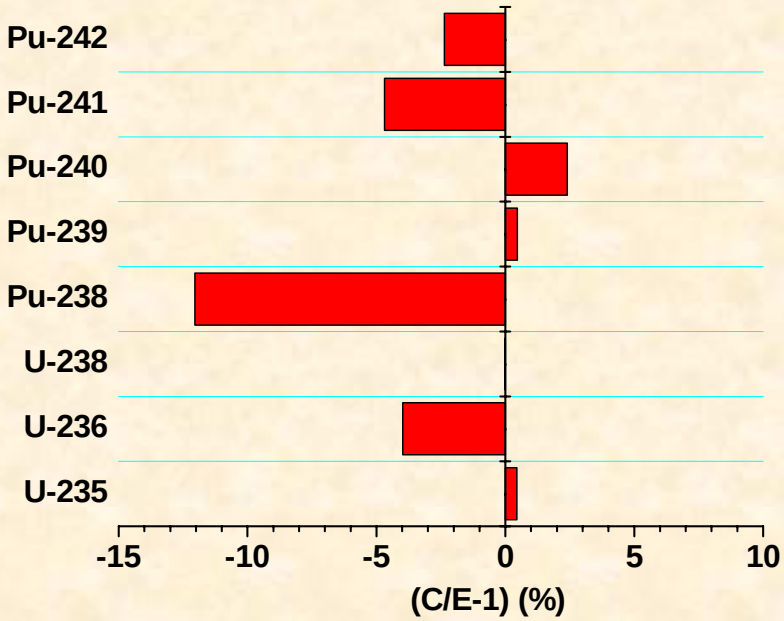
fuel rods in test assembly

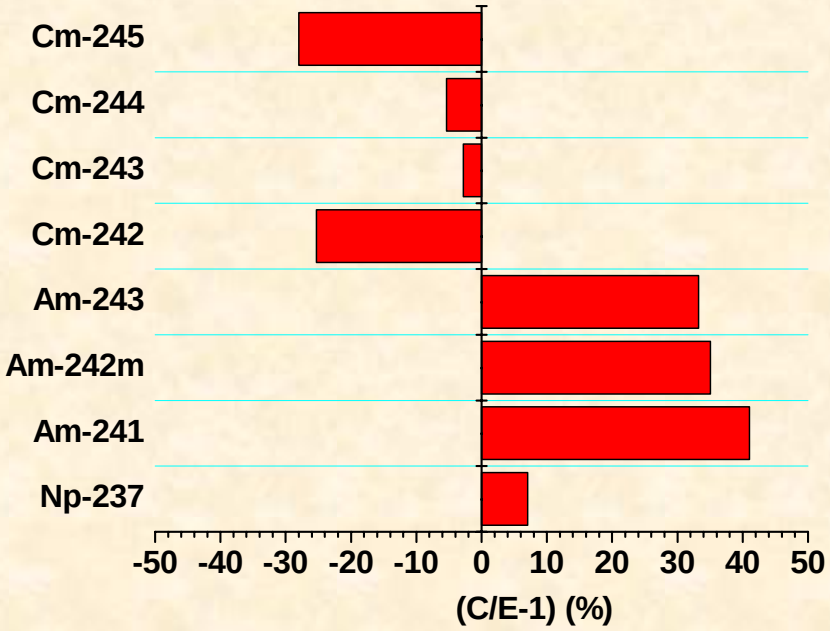
clad

moderator

Analysis of Assay Data for GGU2 Sample

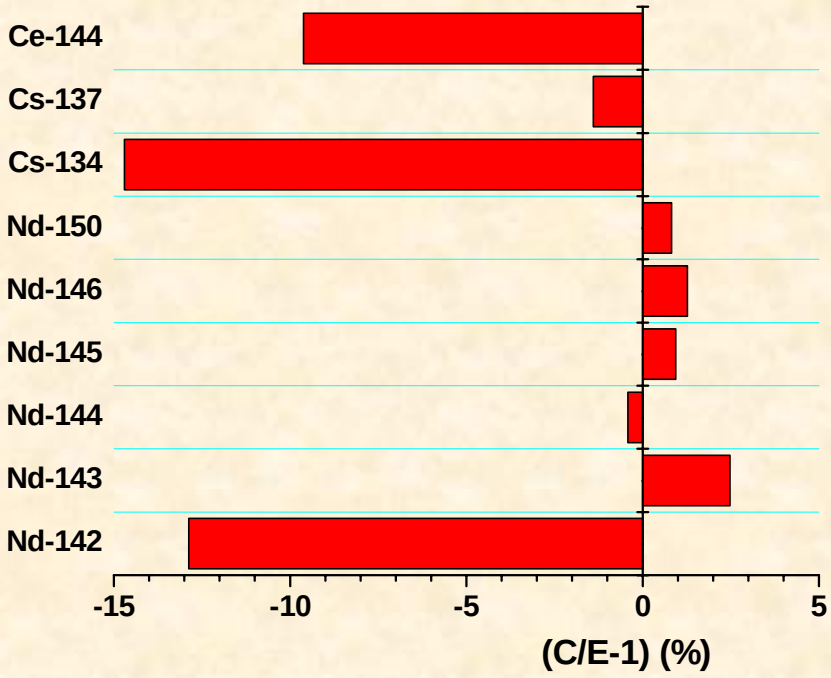
- **Preliminary Results - Actinides**

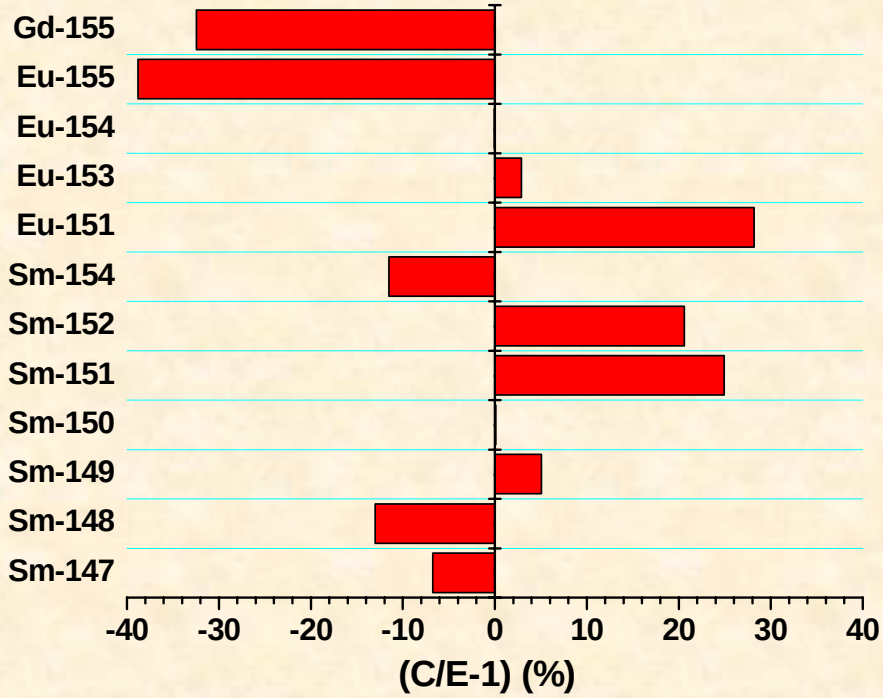




Analysis of Assay Data for GGU2 Sample

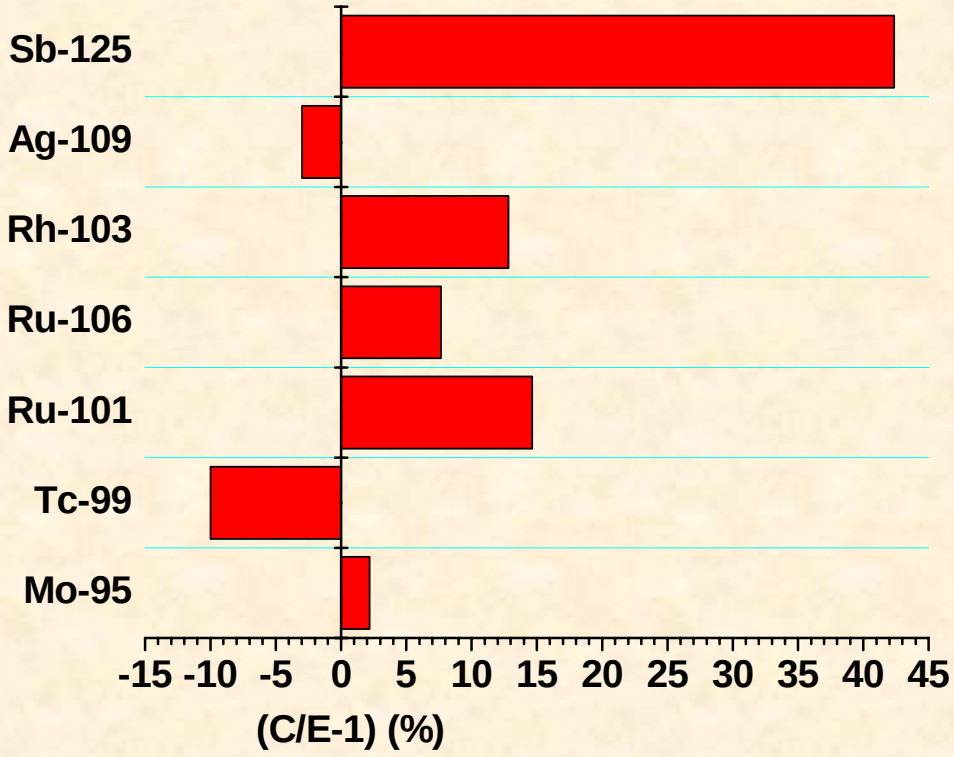
- **Preliminary Results – Fission Products**





Analysis of Assay Data for GGU2 Sample

- **Preliminary Results – Fission Products (Metallic) – Main + Residue Solution Data**



Key Modeling Issues in Assay Data Analyses

- Complete and accurate design and operating data are needed.

- Critical details may include
  - poison rods – location and content of absorber material
  - moderator – density, temperature, soluble boron content
  - fuel enrichment zoning

- Edge rods may require
  - information on nearest assemblies (enrichment, burnup)
  - higher order computational methods to solve accurately

- Protected, commercially sensitive information is sometimes needed to perform accurate simulations; this may not be disclosed or may have distribution restrictions.

- High accuracy measurements needed for key fission product burnup indicators, such as ^{148}Nd, ^{137}Cs, ^{139}La, to normalize power to compensate for inaccuracies in operating data.

Key Modeling Issues in Assay Data Analyses

- **Example – Edge Rod**

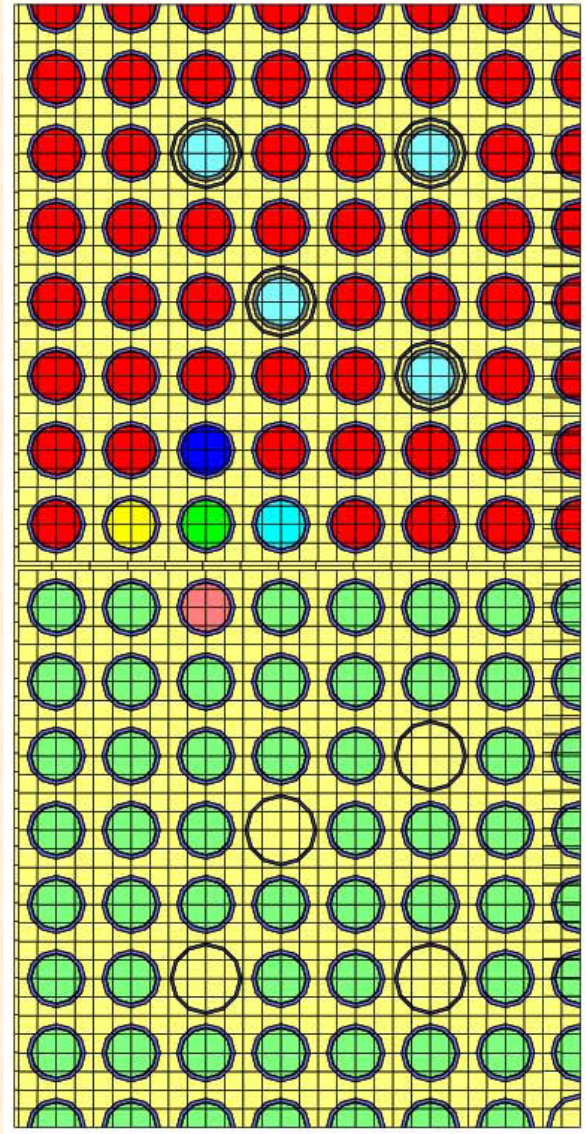
Use of detailed data on assembly pitch, poison rod location, and nearby assembly burnup and enrichment led to an improvement in calculated to experimental agreement for ^{239}Pu, from 10% to 2%.

test rod

neighbors of test rod

absorber rod

fuel rods in test assembly

fuel rods in nearby assembly

moderator