Nuclear Fuel Burnup Records: Generation And Accuracy

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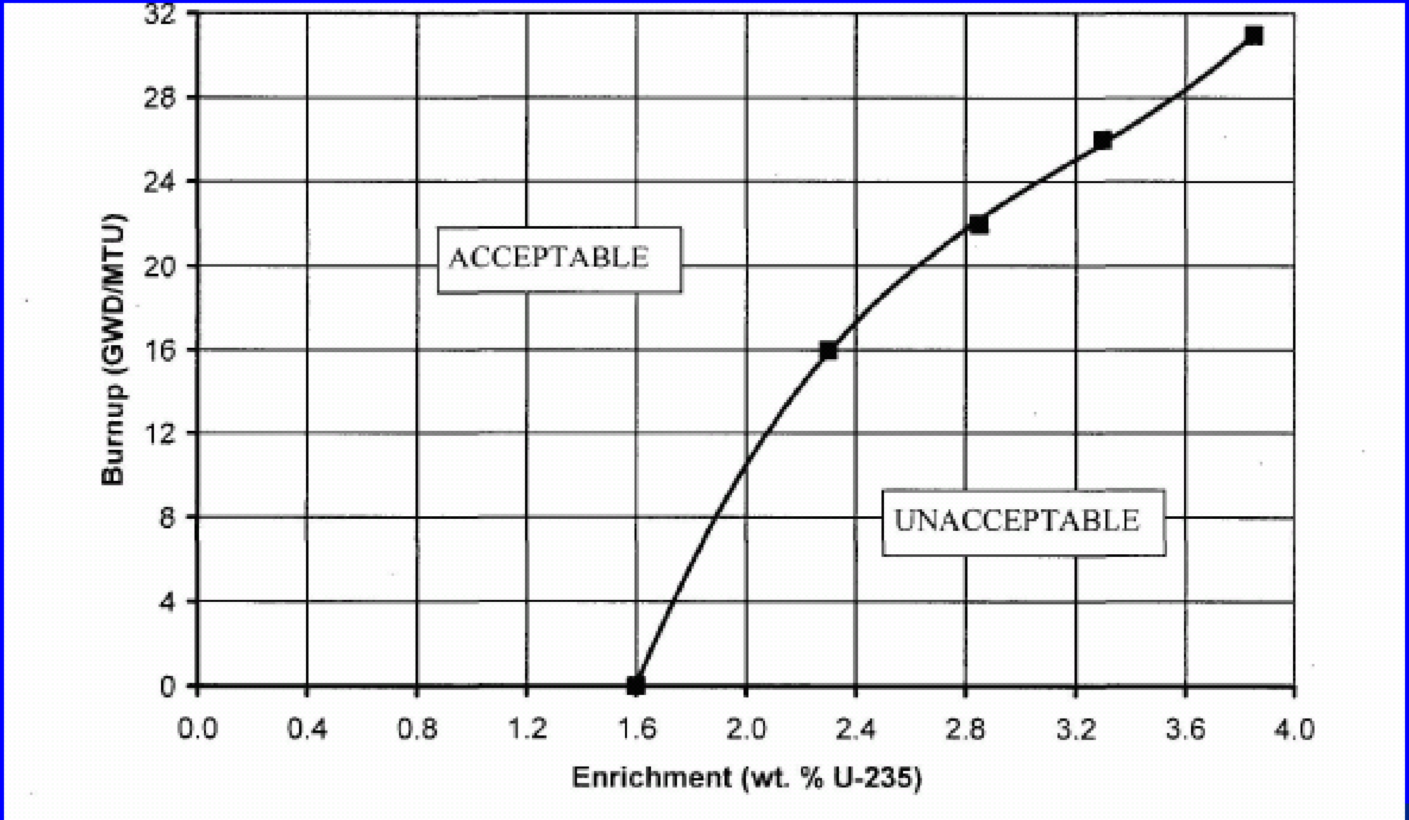
Background

- US NRC Interim Staff Guidance On Burnup Credit For Pressurized-water-reactor Spent Nuclear Fuel In Storage And Transportation Casks
  - Recommends out-of-core burnup measurement to confirm reactor records
    - Sampling measurement
  - Recommendation is intended to prevent unauthorized loading due to:
    - inaccuracies in reactor burnup records
    - human error resulting in improper assembly identification or in misplacement of assembly

Background



Illustrative Example of Loading Curve Generated by Burnup Credit Criticality Analysis

Background

- **Burnup: Amount of cumulative energy generated per weight of fuel, in MWD/MTU**

- **Burnup Credit: Allowing the inclusion of some neutron absorbing nuclides (actinides and fission products) in the criticality analysis. The concentration of these nuclides increases as U-235 depletes in the reactor.**

Assembly Burnup is an Important Input for:

- Safety Analysis
- Core design
- Fuel assembly mechanical integrity protection
- Special nuclear material (SNM)
- Spent fuel storage criticality analysis
- Spent fuel transportation criticality analysis

Burnup Determination

- **The cumulative burnup in Fuel Assembly i during reactor operating time Δt in days is determined as follows:**

$$\Delta B_i = [(RTP * \Delta t * CF) / MF] * P_i$$

Where:

ΔBi is the cumulative assembly burnup in MWD/MTU
RTP is reactor rated thermal power, in MW (constant)
*** CF is Capacity Factor (CF=1 if reactor operating at full power)**
MF is the mass of fuel in metric tons (MTU)
*** Pi is relative power in Fuel Assembly i**

*: uncertainty source

Parameters Important to Burnup Determination

The uncertainty associated with the burnup values of record come from two sources:

1. Actual Thermal Power: thermal output as continuously monitored and measured by plant instrumentation

2. Core Power Distribution: Generated by a reactor physics code on an on-line (continuous) or off-line (hourly / daily) basis. Necessary for verification of compliance with Technical Specifications.

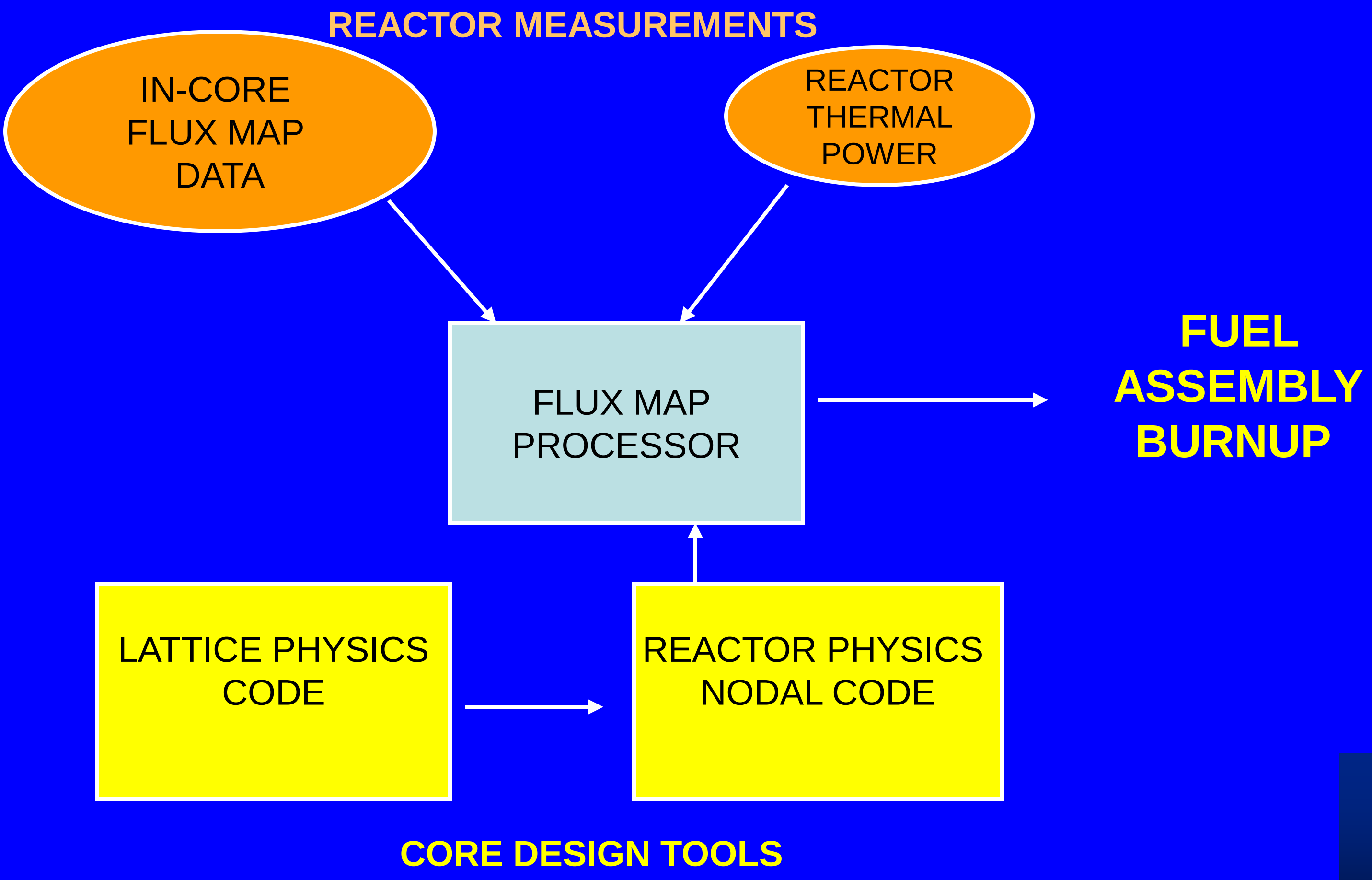
Parameters Important to Burnup Determination

Associated with the previous parameters are the following defined quantities:

Capacity Factor CF: The ratio of reactor actual thermal output to the rated thermal power. The goal of the plant is to operate with CF = 1.

Inferred Core Power Distribution: Generated from a combination of measurements (~1/3 of core bundles) and full-core predictions during the flux mapping process.

Assembly Burnup Determination Process



REACTOR MEASUREMENTS

IN-CORE FLUX MAP DATA

REACTOR THERMAL POWER

FLUX MAP PROCESSOR

FUEL ASSEMBLY BURNUP

LATTICE PHYSICS CODE

REACTOR PHYSICS NODAL CODE

CORE DESIGN TOOLS

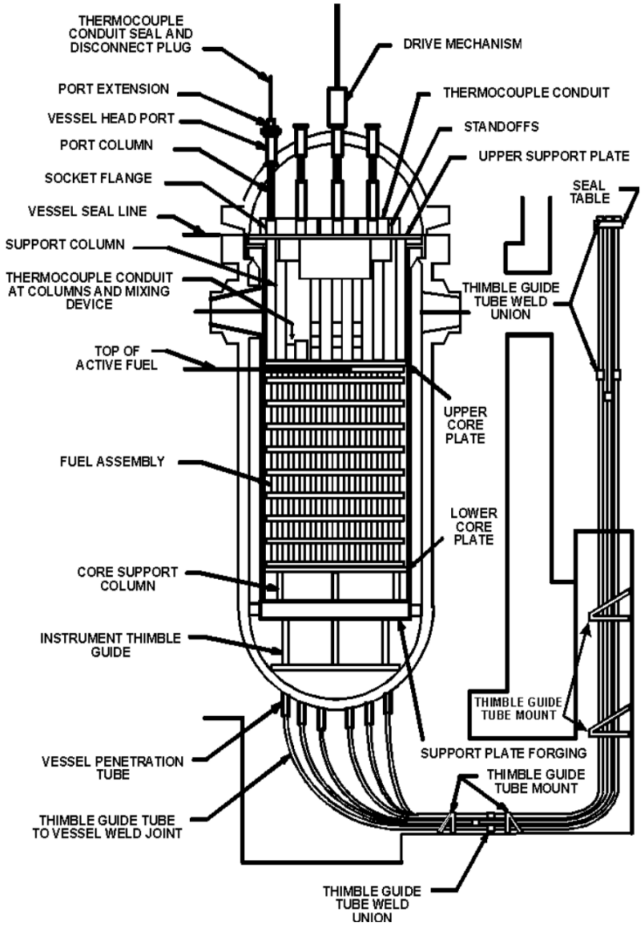
Core Thermal Power Measurement

- Measured continuously
- Periodic instrument calibration
- Redundant indications
- Determined from temperature and flow measurement of reactor cooling water
- ANSI Standard
- Measurement uncertainty ~ 1%

Flux Mapping Process

- Flux traces obtained using fission chamber radiation detectors inside fuel assemblies in the reactor
- Two types of systems:
  - Fixed detectors (B&W plants)
  - Movable detectors (Most other PWRs)
- Movable detectors discussed here
  - Processing of both types of detector traces is similar

Flux Mapping Process



Typical Westinghouse Movable In-Core Detector System

Core Instrumented Locations

Edit: Detector Number
Color: By Regioin Type

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							🟩	🟩 48	🟩						
2					🟩	🟦	🟪	🟪	🟪	🟦 39	🟩				
3				🟦 42	🟪	🟪	🟦 31	🟦 27	🟦	🟪	🟪	🟦 43			
4			🟦	🟪	🟪 26	🟦	🟪	🟦 15	🟪	🟦 22	🟪	🟪	🟦		
5		🟩	🟪 35	🟪	🟦 19	🟪	🟦 12	🟪	🟦	🟪	🟦 20	🟪 28	🟪	🟩 46	
6		🟦	🟪	🟦	🟪 13	🟦	🟪	🟦 4	🟪	🟦 8	🟪	🟦	🟪	🟦	
7	🟩	🟪	🟦 30	🟪	🟦	🟪	🟦 1	🟦	🟦 2	🟪	🟦	🟪 17	🟦	🟪 37	🟩
8	🟩 47	🟪	🟦 25	🟦	🟪 10	🟦	🟦	🟪	🟦	🟦 5	🟪	🟦	🟦 29	🟪 36	🟩
9	🟩	🟪	🟦	🟪	🟦 11	🟪	🟦	🟦	🟦 3	🟪 7	🟦	🟪	🟦	🟪	🟩 50
10		🟦	🟪 32	🟦	🟪	🟦	🟪 6	🟦	🟪	🟦	🟪	🟦 23	🟪	🟦 40	
11		🟩	🟪	🟪	🟦 18	🟪	🟦	🟪 9	🟦	🟪 14	🟦 21	🟪	🟪	🟩	
12			🟦 41	🟪	🟪	🟦	🟪 16	🟦	🟪	🟦	🟪	🟪 34	🟦 44		
13				🟦	🟪	🟪	🟦	🟦 24	🟦	🟪 33	🟪	🟦			
14					🟩 45	🟦	🟪	🟪	🟪 38	🟦	🟩				
15							🟩 49	🟩	🟩						

Shown is a typical 3-Loop W Configuration with 50 detector locations . Locations are numbered.

Reactor Physics Codes

- Lattice Physics Codes
  - Ex: CASMO, PHOENIX, TGBLA
  - Model fuel assembly axial region types or lattices
  - Multi-group, 2-D neutron flux solution
  - Generate cross sections to be used by nodal simulator codes

Reactor Physics Codes

- Nodal Simulator Codes
  - Ex: SIMULATE, MICROBURN, ANC
  - Two-group, 3-D neutron diffusion equation solution
  - Detailed modeling of core isotopic inventory and power distribution
  - Parameters are calculated at ~ 6” cubes (“nodes”)
    - Output include: Safety peaking factors, 3-D power distribution,, reactivity coefficients (rod worth, moderator temperature, power, etc.), critical boron concentration, K-effective.

Flux Map Processors

- Example: INCORE-3, INPAX, BEACON, FIDMS

- Approved per Licensing Topical Report (LTR) submittal.
  - Measurement uncertainty from actual plant data provided in LTR

- Provide inferred core power distribution from combination of:
  - In-core measurements at instrumented core locations
  - Nodal simulator core–wide power distribution predictions

Flux Map Processors

- Used for tech spec compliance with thermal limits
  - F_Q and $F_{\Delta h}$

- Other results:
  - Incore-Excore detector calibration
  - Axial Offset
  - Quadrant Power Tilt Ratio
  - Assembly burnup

Flux Map Processors

Typical Algorithm:

- **The “raw” measured traces are processed and normalized**
- **The assembly flux and/or power predictions are expanded from the nodal simulator number of axial nodes (~ 24) to that of the measured axial locations (~ 61)**
- **Measured reaction rates at the instrumented locations are compared to their predictions from the nodal simulator code**

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Flux Map Processors

Typical Algorithm

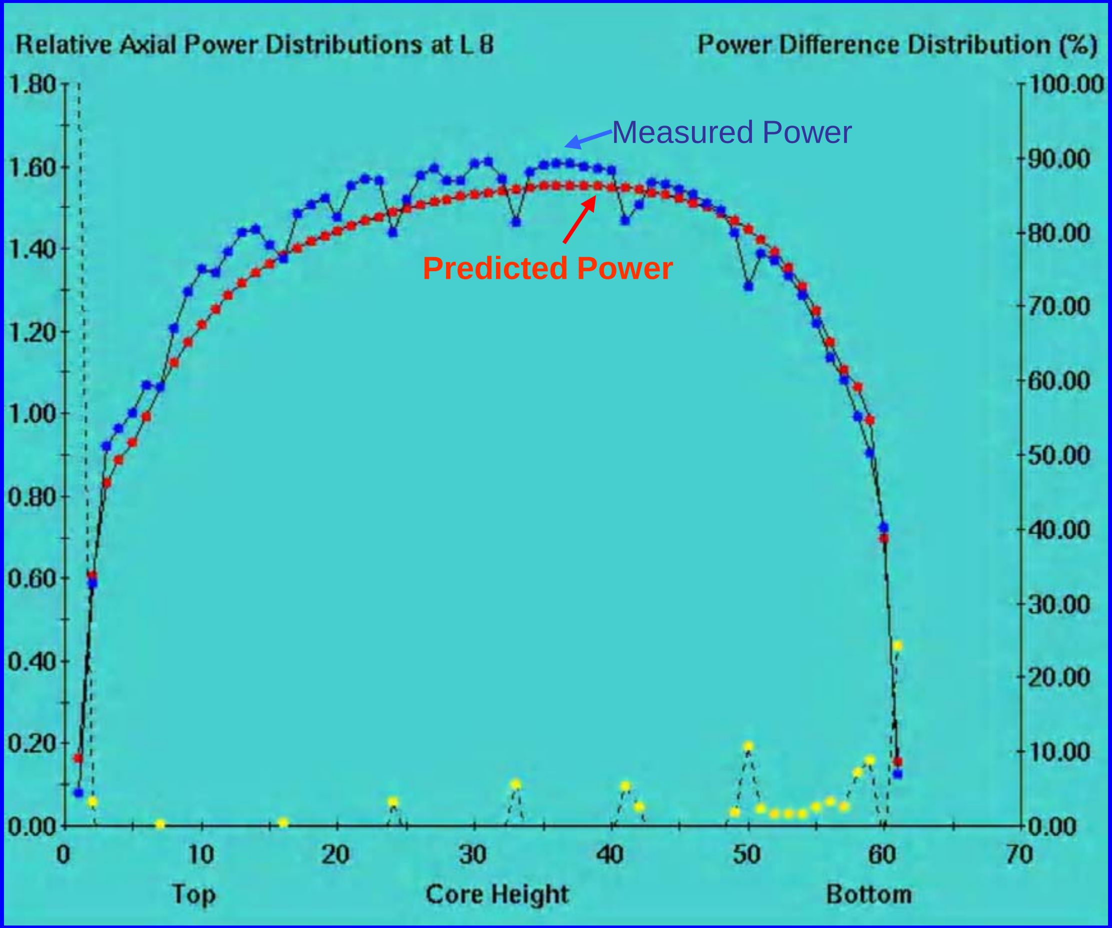
- Correction factors are generated from the ratios of the measured to predicted reaction rates at the instrumented locations and are used to adjust the predicted assembly power.
- Coupling or weighing coefficients for each non-instrumented location are derived using data from the nearest instrumented assemblies.

Flux Map Processors

- The power distributions at the non-instrumented locations are derived using the nodal simulator assembly powers along with the coupling coefficients.

- From the inferred core power distribution the following are deduced:
  - safety factors and margins to the technical specifications thermal limits
  - Additional burnup cumulated for each fuel assembly

Measurement vs. Prediction at Instrumented Core Location



Actual Plant Flux Map Data for 3-Loop Westinghouse

Measured Full-Core Assembly Burnup

Edit: Assembly Exposure

Color: By Assembly Exposure

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							46.58	29.93	46.66						
2					37.54	29.13	6.70	6.94	6.72	29.15	37.87				
3				28.97	7.18	7.82	28.89	28.18	28.95	7.82	7.15	28.97			
4			28.97	6.90	8.34	32.59	8.45	29.38	8.47	32.59	8.33	6.89	28.97		
5		37.88	7.17	8.34	32.90	8.47	32.24	8.42	32.28	8.47	32.90	8.33	7.17	37.54	
6		29.16	7.83	32.60	8.48	32.71	7.94	32.79	7.96	32.71	8.47	32.59	7.82	29.14	
7	46.66	6.73	28.96	8.48	32.29	7.96	28.37	29.51	28.37	7.94	32.25	8.46	28.90	6.70	46.58
8	29.94	6.95	28.19	29.39	8.42	32.79	29.51	8.53	29.51	32.79	8.42	29.39	28.19	6.95	29.94
9	46.58	6.70	28.90	8.46	32.25	7.94	28.37	29.51	28.37	7.97	32.29	8.49	28.97	6.73	46.66
10		29.14	7.82	32.59	8.48	32.72	7.97	32.80	7.95	32.72	8.49	32.61	7.84	29.16	
11		37.54	7.18	8.34	32.91	8.49	32.30	8.43	32.26	8.49	32.92	8.36	7.18	37.88	
12			28.98	6.91	8.35	32.61	8.50	29.40	8.48	32.61	8.36	6.92	28.98		
13				28.98	7.18	7.84	28.98	28.20	28.92	7.84	7.20	28.98			
14					37.88	29.16	6.74	6.96	6.72	29.15	37.55				
15							46.66	29.95	46.59						

Color Index

Exposure

44.85 - 48.48
41.22 - 44.85
37.58 - 41.22
33.95 - 37.58
30.32 - 33.95
26.68 - 30.32
23.05 - 26.68
19.42 - 23.05
15.78 - 19.42
12.15 - 15.78
8.52 - 12.15
4.88 - 8.52

Exp. in GWD/MTU

Plant Confirmation of Physics Codes

- Cycle Startup Zero Power Physics Testing
    - Critical Boron Concentration
    - Control Rod Worth
    - Moderator Temperature Coefficient

- Power Escalation Testing
    - Operating Limits
    - Core Power Distribution

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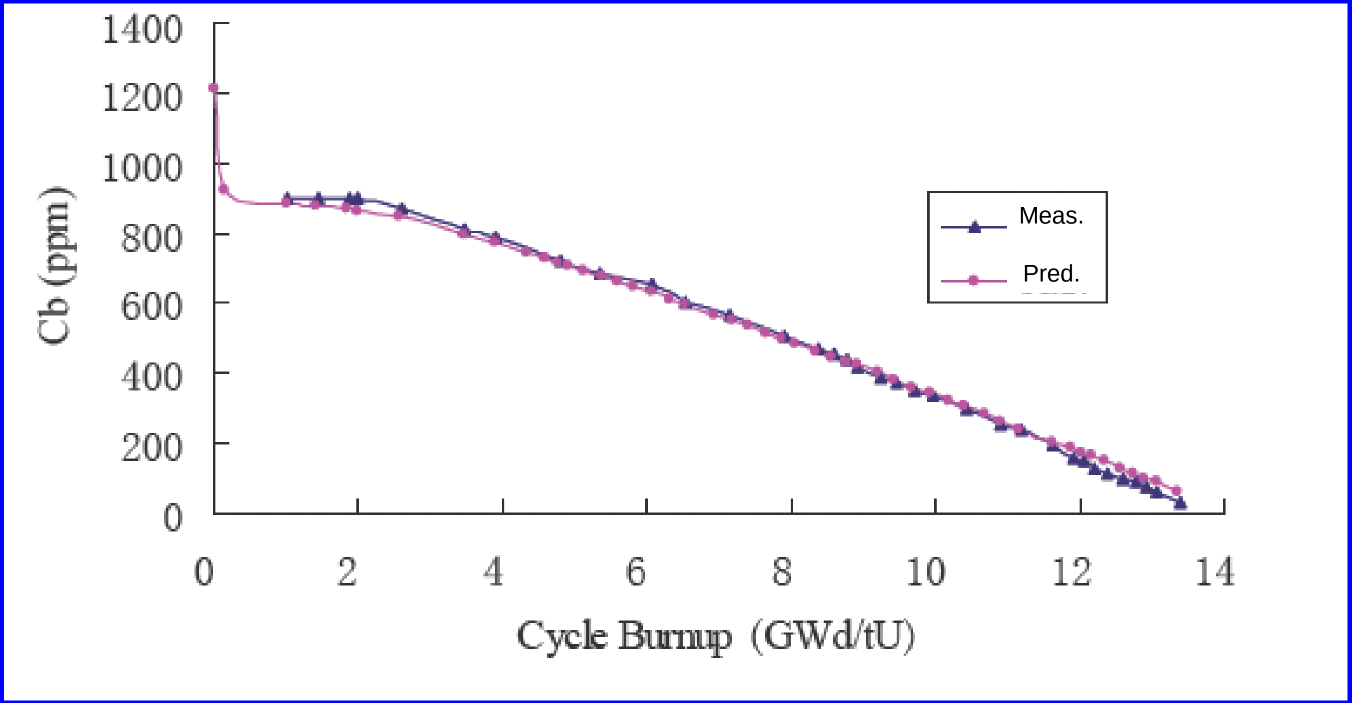
Plant Confirmation of Physics Codes



Boron letdown curve (Daily)

Plant Confirmation of Physics Codes

- Flux Map (monthly):
  - Axial and integral bundle power at core instrumented locations
- Mid-Cycle Startup Criticality
- End of Cycle Moderator Temperature Coefficient measurement

Plant Confirmation of Physics Codes

- Cycle Planning – Fleet Outage Schedule
  - 2/3 of Assemblies are reinserted in the upcoming cycle – Their BU values are important to cycle design and safety analysis
  - Cycle end of full power capability must be predicted within a narrow operating window - Ex: if core burnup missed by ~ 5%, the shutdown date would be missed by ~ 30 days.
  - Significant cost if window missed (cycle redesign, operation at lower power, premature discharge of fuel)
  - Plants have excellent record of cycle planning

Burnup Uncertainty Evaluation Studies

- Very large database
- Easily accessible
- Of interest:
    - EPRI
    - DOE
    - Duke Power
    - TVA

EPRI Study

- "*Determination of the Accuracy of Utility Spent Fuel Burnup Records*" – July 1999
  - Three consecutive cycles flux map data
  - Run SIMULATE at only the instrumented locations
  - One-sided tolerance toward burnup under-prediction (more conservative)
  - Uncertainty expressed as difference between measured and predicted assembly average BU at end of each cycle
    - Cycle "1": 2.49%
    - Cycle "2": 1.67%
    - Cycle "3": 1.99%

DOE Study

- “*Reactor Record Uncertainty Determination*” - 2004
- In support of Yucca Mountain Project
- Nine plants
- 5447 fuel assemblies
- Uncertainty between 2 and 4% with 95/95 Confidence

Duke Study

- 1900 assemblies
- Observed measured error ~ 4%
- Smaller error with higher discharge burnup

TVA Study

- To validate the spent fuel database at its three sites (2004)
  - Identify any transcription, data entry and other human factor errors
- Sample size: 1117 assemblies
- Compare flux map generated burnup against design code values
    - Most BU errors were < 1GWD/MTU
    - Largest error 1.51 GWD/MTU on a 41.8 GWD/MTU assembly (or 3.6%)
    - Only 7 assemblies burnup values required adjustment

Plant Records Discussion

- Significant amount of data support the finding that the uncertainty of spent fuel burnup records are < 5%

- Accuracy increases with increased irradiation

- Burnup measurement improved over time
  - Improved instrumentation
  - Improved codes

Plant Records Discussion

- Inconsistency in burnup measurement and record keeping among plants
    - Differences in:
        - Procedures
        - Codes
        - QA requirements

    - Transfer to new record systems
        - Vulnerability: Corrupted/missing record for under-burned discharged fuel

Plant Records Discussion

- Some plants – like TMI-1 – had “batch average” burnup, NOT individual assembly burnup
    - One TMI assembly was found to be significantly more under-burned (by 16 GWD/MTU)

- Even within the same plant, the process may have changed with time: older records may need revising
    - Some utilities have conducted spent fuel inventory verifications; However most have not.

Moving Forward

- Utilities need to demonstrate that their records are accurate and independently verifiable:
  - Consistent reporting process needs to be developed
  - Detailed description of burnup measurement methods and their accuracy