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Use of Burnup Credit When Performing Criticality Analyses for Spent Fuel Transportation Packages

NWTRB Briefing
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Agenda

- Background
- Criticality safety for spent fuel transportation packages
- Current NRC guidance on burnup credit
- Computer code validation for burnup credit criticality analyses
- Considerations for future guidance on burnup credit for transportation

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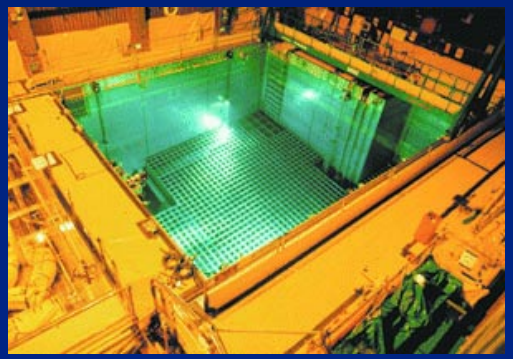
Background

- Burnup: the amount of energy released from a fuel assembly in a reactor in terms of Megawatt-Days per Metric Ton of initial Uranium (MWD/MTU), which results in an overall reduction of fuel assembly reactivity

Background (cont.)

- Burnup Credit: credit for the reduction in reactivity that occurs with fuel burn-up due to the net reduction of fissile nuclides and the production of actinide and fission-product neutron absorbers



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Goal:

- Higher capacity for dry storage and transportation, and the ability to transport entire inventory of commercial spent fuel when needed

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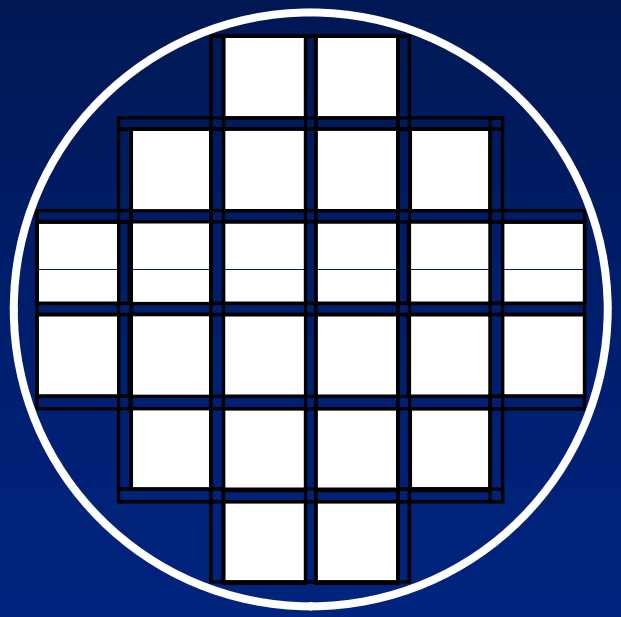
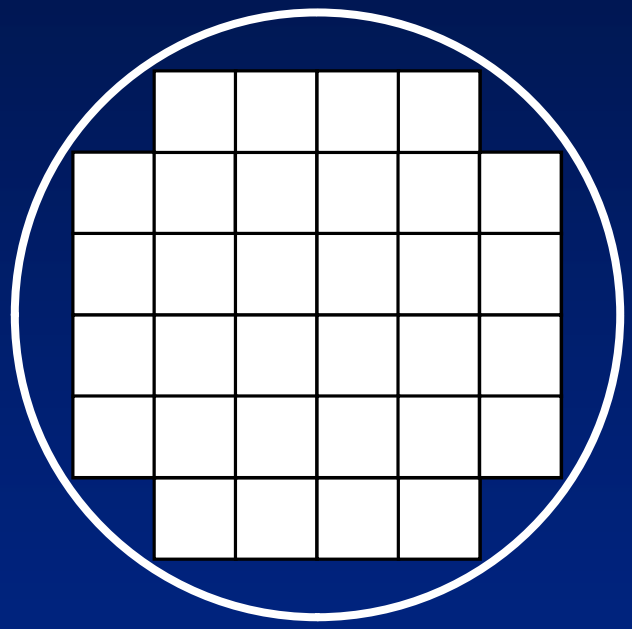
Transportation Cask Capacity



24-Assembly Basket with Flux Traps



32-Assembly Basket without Flux Traps

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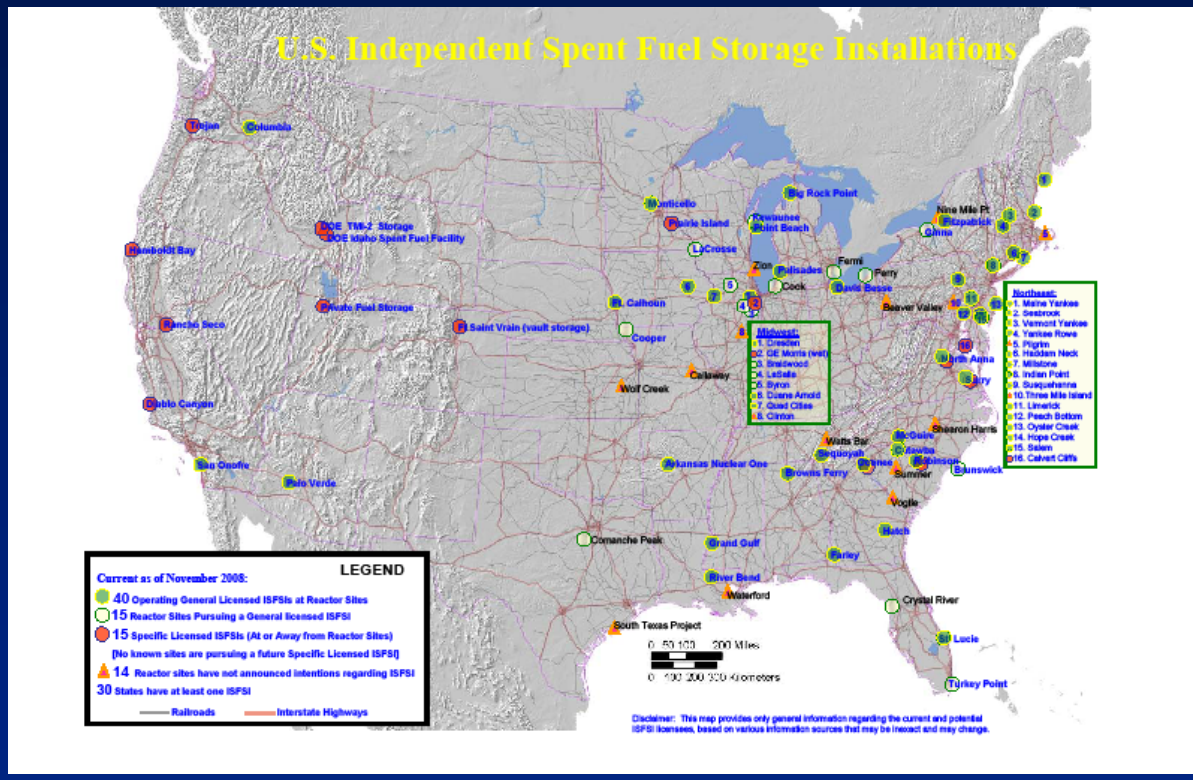
Storage Status

- 55 ISFSIs in 33 States
- 9-13 new ISFSIs in 2009-2012
- Over 1000 loaded storage casks



ISFSI Map available at www.nrc.gov, ADAMS# ML083020621

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Criticality Safety for Spent Fuel Casks

- 10 CFR 71.55 (b)
  - "… a package used for the shipment of fissile material must be so designed and constructed and its contents so limited that it would be subcritical if water were to leak into the containment system, or liquid contents were to leak out of the containment system so that, under following conditions, maximum reactivity of the fissile material would be attained…"

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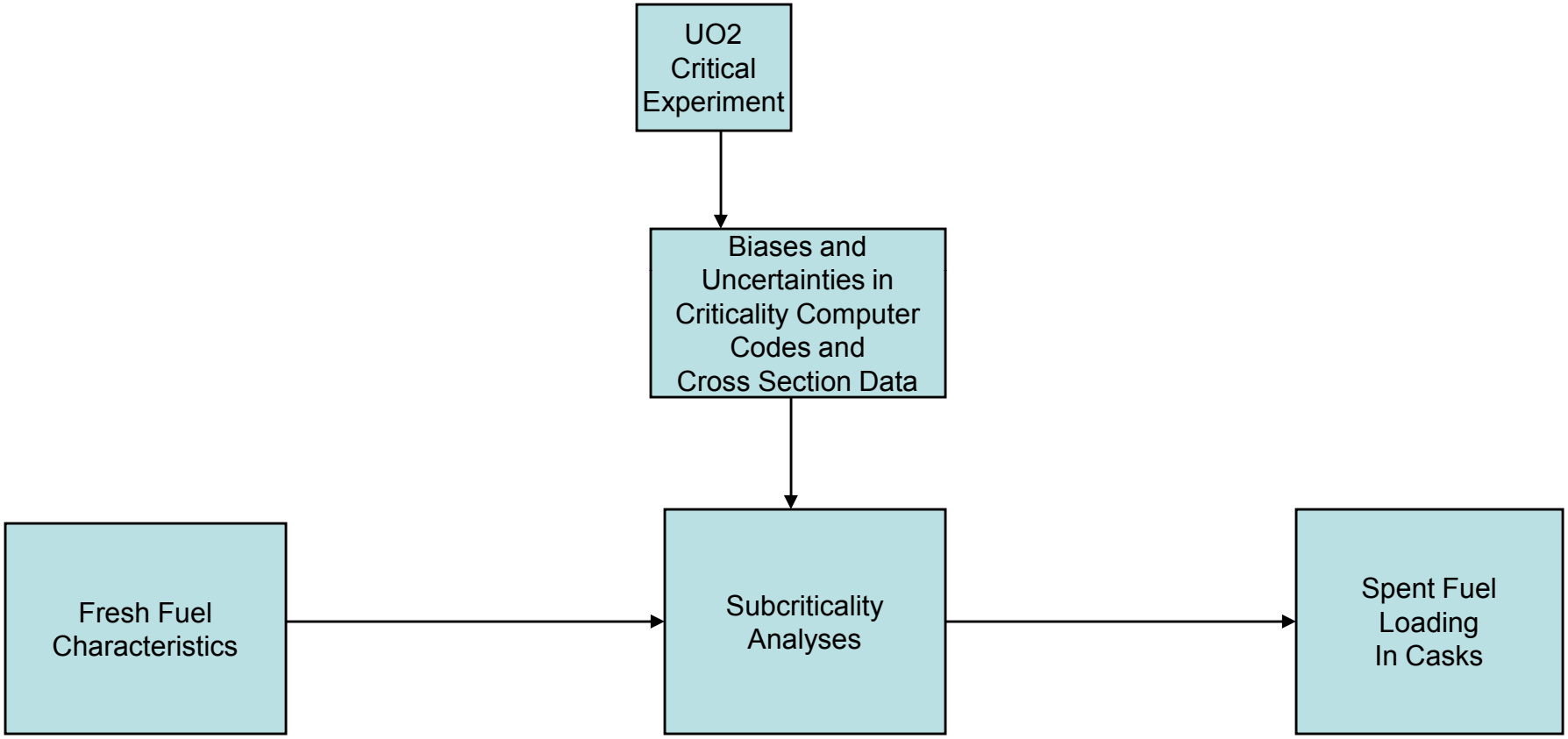
Criticality Safety for Spent Fuel Casks

- 10 CFR 71.83
  - "When the isotopic abundance, mass, concentration, degree of irradiation, degree of moderation, or other pertinent property of fissile material in any package is not known, the licensee shall package the fissile material as if the unknown properties have credible values that will cause the maximum neutron multiplication."

Criticality Safety for Spent Fuel Casks (fresh fuel assumption)



In fresh water environment

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Interim Staff Guidance 8

- "Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks"
  - Revision 2 published in 2002 **(http://www.nrc.gov/reading-rm/doc-collections/isg/isg-8R2.pdf)**
  - Actinide-only based on available validation data
  - Fission products to provide additional margin
  - Confirmatory burnup measurement to prevent misload, consistent with IAEA TS-R-1

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Interim Staff Guidance 8 (cont'd)

- Actinides represent roughly 75% of the reduction in reactivity due to burnup
  - Major actinides: ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu
- Fission products represent roughly the remaining 25% of the reduction in reactivity
  - Major fission products: ^{149}Sm, ^{143}Nd, ^{103}Rh, ^{151}Sm, ^{133}Cs, and ^{155}Gd

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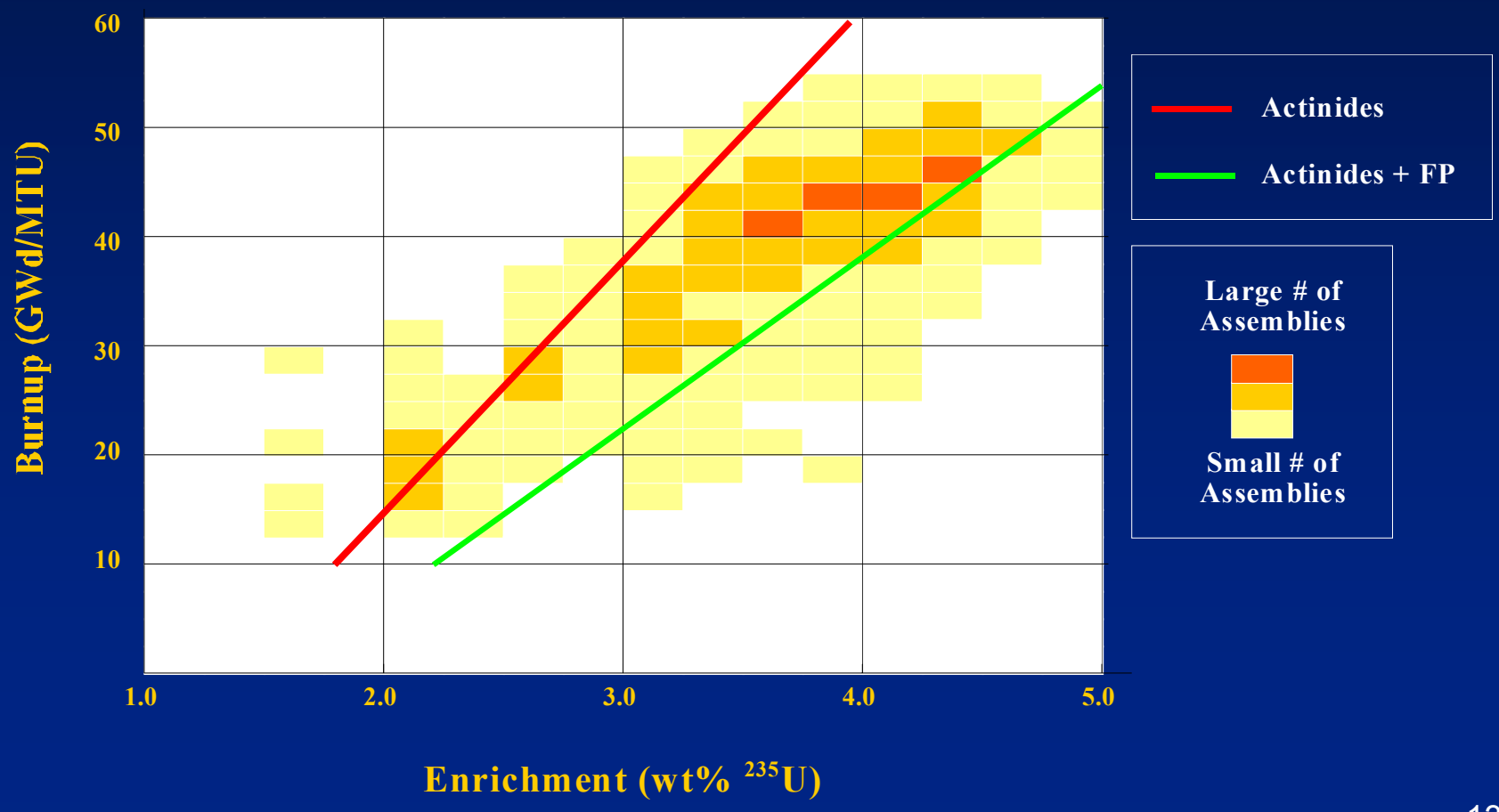
Discharged PWR Fuel Population



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Interim Staff Guidance 8 (cont'd)

- EPRI has concurred that experimental data for validation are a limiting factor for extending to full BUC:

  "ISG-8, Revision 2 can be viewed as providing as much burnup credit flexibility as can be currently expected (UO2 fuel irradiated in PWRs only, with no credit for fission products) based on the extent and range of the available data" (Source: EPRI 1002879)

- One burnup credit application approved
- Three others under consideration

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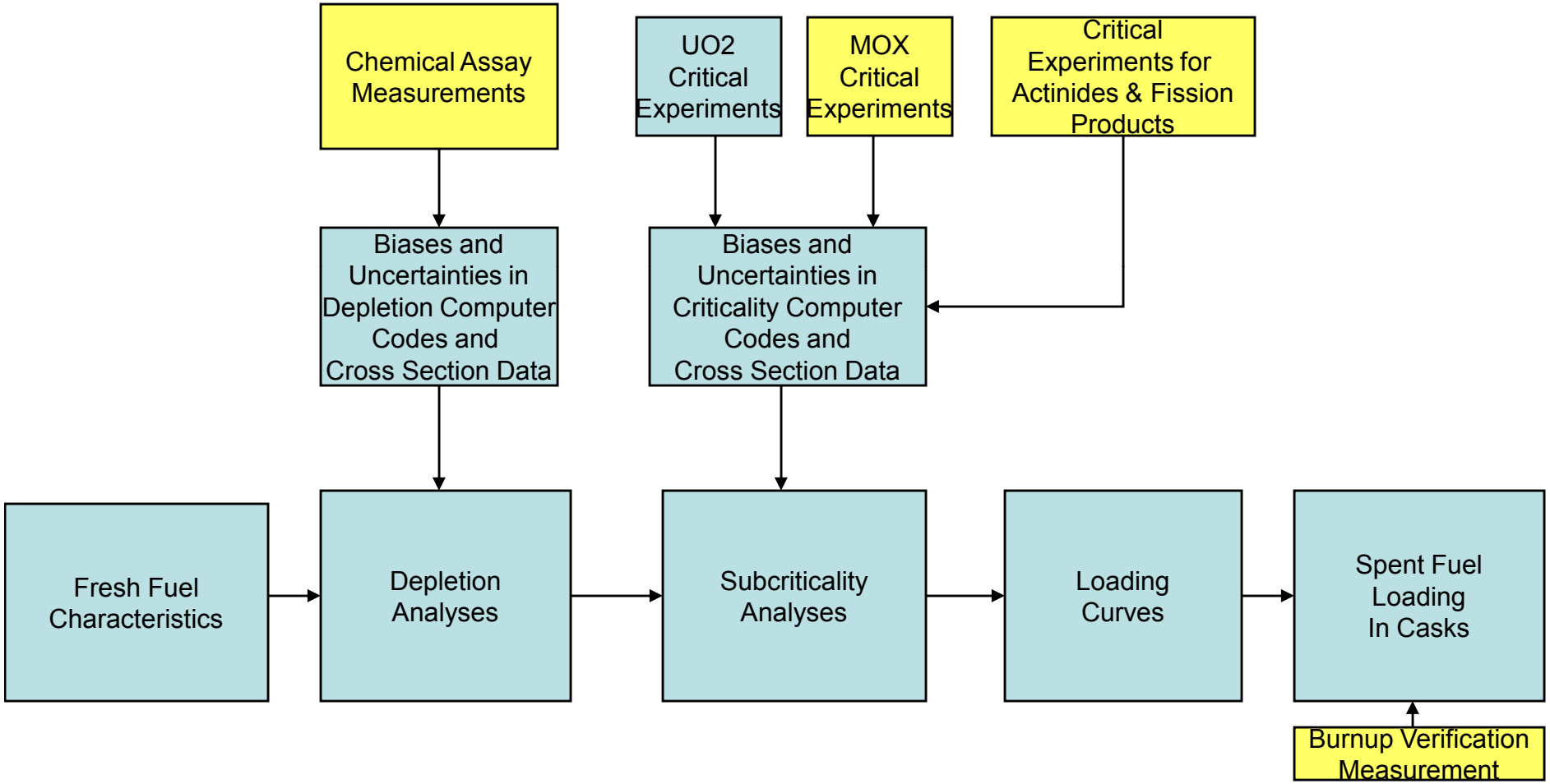
Code Validation for Burnup Credit

- NRC expects software validation for burnup credit in spent fuel transportation to be consistent with well-established domestic (ANSI/ANS) and international (ISO) practice for out-of-reactor criticality safety analyses

- Standards require comparison of predicted vs. experimental data to obtain bias and bias uncertainty
  - Radiochemical assay data for depletion code validation
  - Critical experiments for criticality code validation

Criticality Safety for Spent Fuel Casks (with burnup credit)



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Expanding Technical Basis for Burnup Credit

- NUREG/CR-6979, “Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data”
- NUREG/CR-6951, “Sensitivity and Uncertainty Analysis of Commercial Reactor Criticals for Burnup Credit”
- Currently reviewing new high burnup radiochemical assay data for depletion code validation
- Investigating possible acquisition of French fission product critical experiment data for fission product criticality validation

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Considerations for Revision 3 – Code Validation

- Availability of French HTC actinide data gives greater degree of confidence in actinide criticality validation than existed at the time ISG-8, Rev. 2 was published
- New chemical assay data for fission product depletion validation
- Potential use of French fission product critical experiment data
- Potential use of CRC data to augment validation process

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Considerations for Revision 3 – Burnup Measurements

- NUREG/CR-6955, "Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask"
- Draft NUREG on information related to spent fuel burnup confirmation
- NRC Office of Research evaluating misload probability
- Potential for misload analysis in lieu of measurement

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Summary

- Burnup credit increasingly sought by industry to maximize the capacity of spent fuel transportation packages
- Burnup credit criticality analysis and validation much more complicated than for fresh fuel assumption
- NRC working to expand the technical basis for burnup credit to allow some credit for fission products and provide alternatives to confirmatory burnup measurements
- Crediting fission products can increase the fraction of the discharged fuel population that can be transported in high capacity transportation packages

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Questions?