A Burnup Credit Approach for BWR Criticality Margin Estimation with UNF-ST&DARDS

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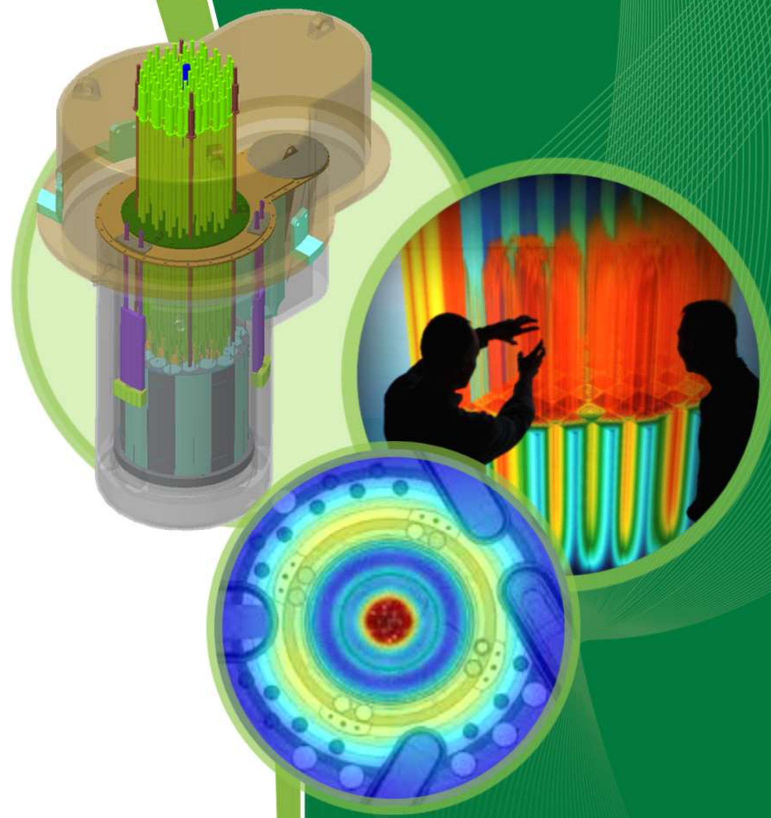


UNF-ST&DARDS has been used to determine criticality safety margin of 163 already loaded BWR canisters

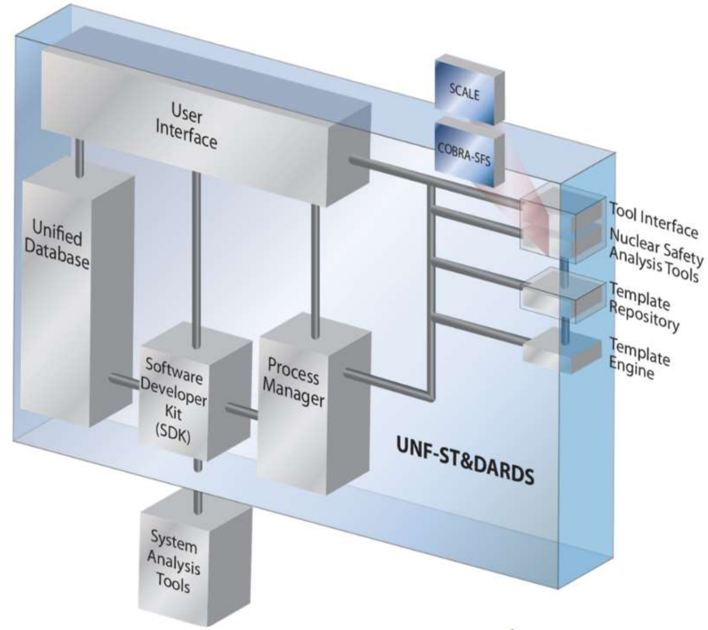
- A BWR burnup credit approach has been developed to support as-loaded criticality analysis of already loaded canisters
  – As-loaded criticality analysis is performed to determine inherent safety margin of the already loaded canisters
  – Inherent safety margin can be used to offset systems (e.g., canister, fuel assembly) integrity related uncertainties after extended storage and or during disposal time period
- The BWR burnup credit methodology consists of
  – Justification for BWR assembly modeling
  – Justification for using a uniform void profile
  – Justification for burnup profile selection

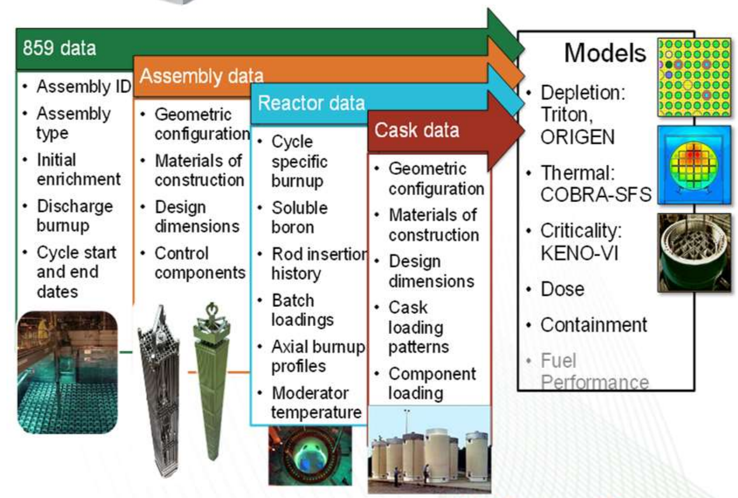
Overview

- Used Nuclear Fuel – Storage Transportation and Disposal Resource and Data System (UNF-ST&DARDS) Overview
- Differences between PWR and BWR Canister licensing
- Evolution of BWR fuel
- Fuel Depletion parameters for BWR Fuel
- Selection of BWR Axial Burnup Profiles
- As-loaded criticality calculations
- Results and Conclusions to date

UNF-ST&DARDS provides a means to integrate data and analysis tools to estimate safety margins

- A Unified tool that contains modeling information in a database integrated with analysis codes
- Database contains GC-859 collected fuel, irradiation information, and cask loading maps
- The tool also contains
  - templates for a number of canisters and fuel assemblies
  - ORIGEN libraries for a diverse set of fuel assembly types
- Criticality calculations are performed for storage /transportation and disposal





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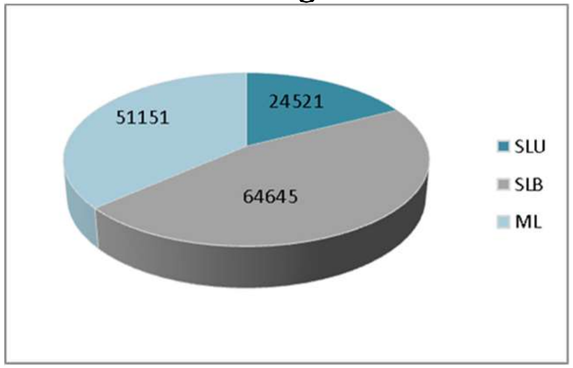
PWR and BWR licensing and information differences drive different burnup credit approaches

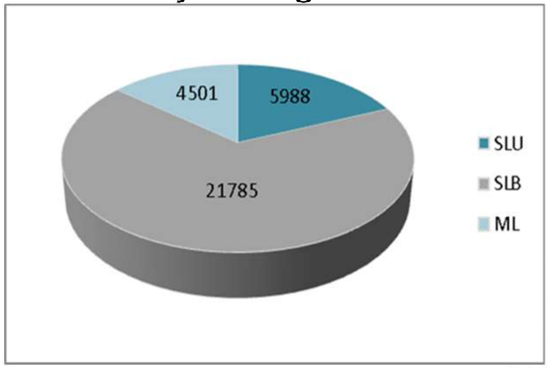
- Many PWR canisters in use today use burnup credit for transportation
- Soluble boron is typically credited for PWR canister loading (Storage)
- BWRs load under mostly the same conditions that are required for transport
- Disposal still a potential concern due to lack of credit for neutron absorber
- Potential for new licensing issues for transportation for extended storage timelines
- New higher capacity casks are being produced and licensed
- Limited previous application of BWR burnup credit

Evolution of BWR fuel informs depletion parameter selection

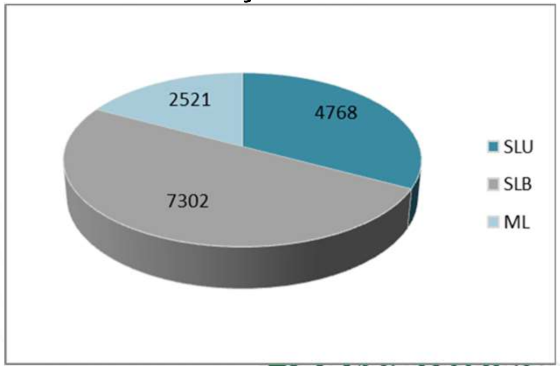
- Assemblies are classified into three categories mainly based on axial fuel features for applicability of burnup profiles
- The three categories are:
  - Single Lattice Unblanketed (SLU) – Beginning –Late 1970s
    - Mostly Early BWR-1 fuel, 7x7 fuel, and early 8x8
  - Single Lattice Blanketed (SLB) – Late 1970s until 1990s – Natural Blankets
    - Majority of the 8x8 fuel and Areva 9x9
  - Multi-Lattice (ML) – Late 1990s through today – Multiple lattices also Natural Blankets
    - Introduction of vanished lattice
    - All currently produced fuel GE 9x9, and 10x10 variants, Areva 10x10 and 11x11, SVEA series
- Distribution of fuel types as of mid 2013
  - Discharged



Dry Storage



Analyzed



BWR depletion parameters selected based on recent research

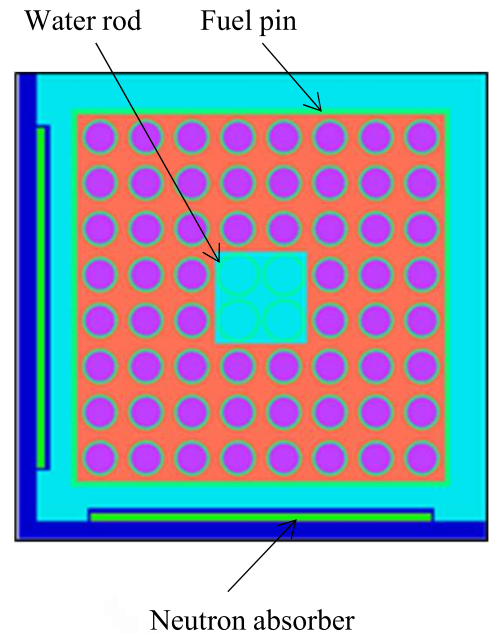
- Depletion parameters selection for PWRs and its impact are well understood
- Depletion conditions substantially different for BWR fuel than PWR fuel
  – Highly voided moderator
  – Extensive use of control blades during operation
  – More substantial variation in fuel design – multiple lattices axially and radial variations
- Research into BWR fuel depletion parameters investigated in NUREG-7224 for extended burnup credit
  – Single cycle of modern BWR operation using a variety of fuel designs
- Most important depletion parameters to discharge reactivity are void fraction, control blade insertion, and axial burnup profile

Void fraction and control blade insertion depletion parameter selected for simplicity and conservatism

- UNF-ST&DARDS uses point depletion ORIGEN libraries generated using TRITON to perform rapid depletion and decay for criticality, shielding, and thermal calculations
- Must select a set of depletion conditions that will be conservative but not overly penalizing if possible
  - Selection of conservative depletion parameters makes UNF-ST&DARDS as-loaded canister-specific analysis conservative
- 0.3 g/cm^3 uniform moderator density and full length control blade insertion used for BWR depletion analysis
  - 0.3 g/cm^3 represents average of representative axial void profile
  - Multiple voids fractions would increase the number of ORIGEN libraries – considered for future version of UNF-ST&DARDS
- In this work we investigated the effects of these depletion parameters compared to limiting values found in NUREG-7224

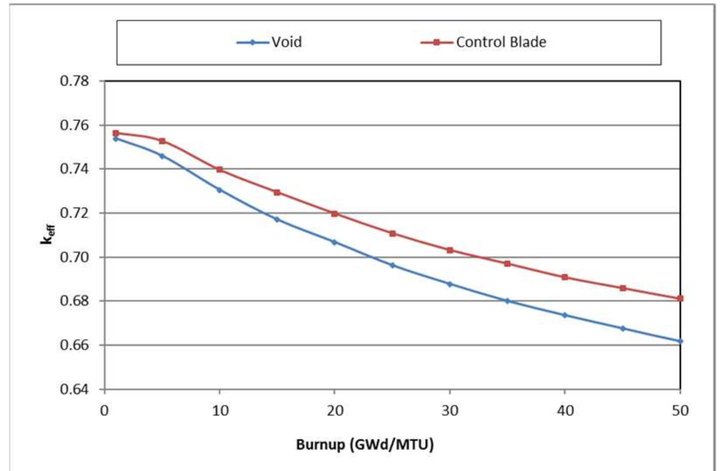
Void Fraction and Control Blade Insertion (cont'd)

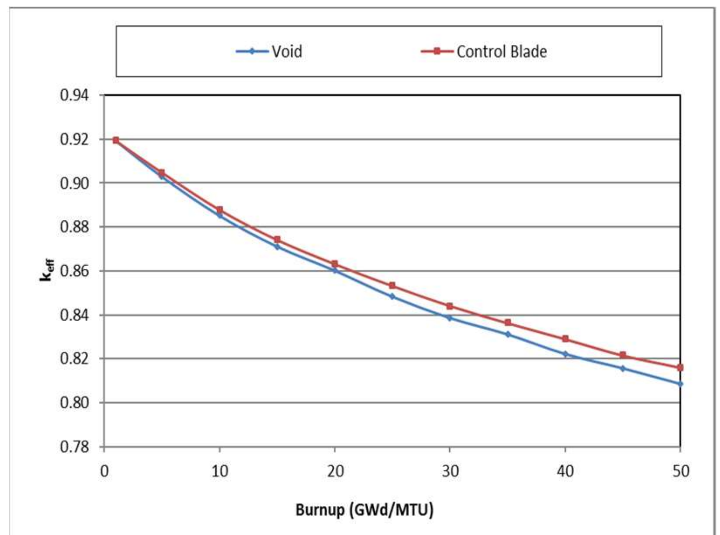
- NUREG identified a limiting void profile
  - Lowest exit moderator density ~0.12 g/cm^3 from NUREG vs ~0.22 g/cm^3 from commercial reactor critical state points (CRC)
- Result of limiting control blade insertion resulted in an increase of 0.012 Δk_{eff} compared to unrodded depletion identified in NUREG
- Investigated the effect of rodded depletion by g/cm^3 running calculations TRITION > ORIGAMI > KENO calculations
  - UNF-ST&DARDS case – 0.3 g/cm^3 with CB insertion
  - Limiting Moderator Distribution from NUREG with no CB insertion
- Used MPC-68 Model single basket cell model with radially reflected boundaries
  - Used limiting profile from 10-18 GWd/MTU to give high weight to void.



Depletion Parameter Study Results

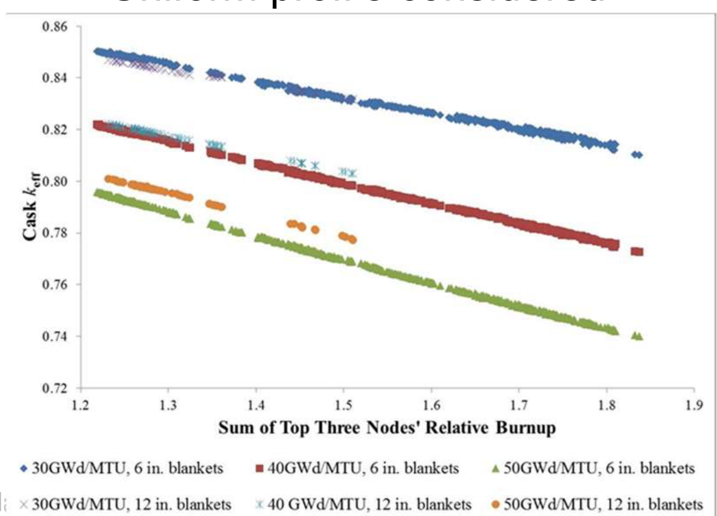
- Control blade insertion cases are more reactive
  – 2 w/o
    - 0.01545 Δk_{eff} more reactive @ 30 GWd/MTU
    - 0.01938 Δk_{eff} more reactive @ 50 GWd/MTU
  – 4 w/o
    - 0.00551 Δk_{eff} more reactive @ 30 GWd/MTU
    - 0.00739 Δk_{eff} more reactive @ 50 GWd/MTU
- Higher enrichment results show that the blades would not cover both the limiting void profile and maximum observed control blade insertion
  – Combination of limiting void profile and control blade insertion into the upper portion of the assembly unlikely
  – Work by Ade et al. Shows that there is significant relaxation in the axial burnup profile with control rod insertion
    - At least correlated if not caused

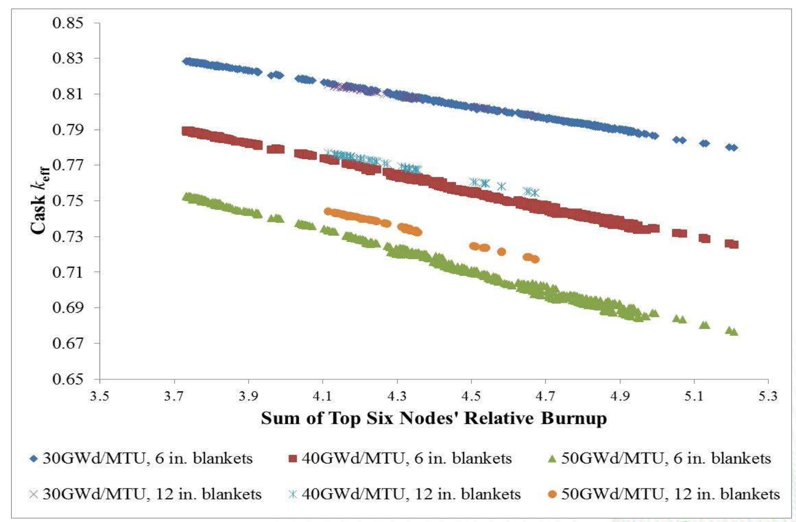




Axial burnup profiles selected from CRC data using recent research

- NUREG-7224 shows that sum of burnup in top 3 nodes or top 6 nodes out of 25 was most highly correlated with discharge reactivity for 30-50 GWd/MTU
- 2,312 profiles from 15 cycles of operation from 3 US BWRs – All SLB fuel
  - YMP grouped profile data from into bins of every 4 GWd/MTU up to 46 GWd/MTU
  - Took minimum burnup from top 3 and top 6 (same except for 4 instances) from YMP document and mapped to UNF-ST&DARDS burnup profile bins - 0–6, 6–10, 10–18, 18–34, and greater than 34 GWd/MTU
  - Ran ORIGAMI > KENO calculations for upper end of each bin to pick limiting profile – no blanket modeling
  - Uniform profile considered





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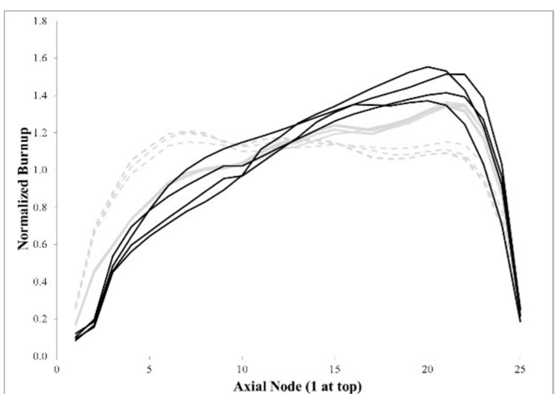
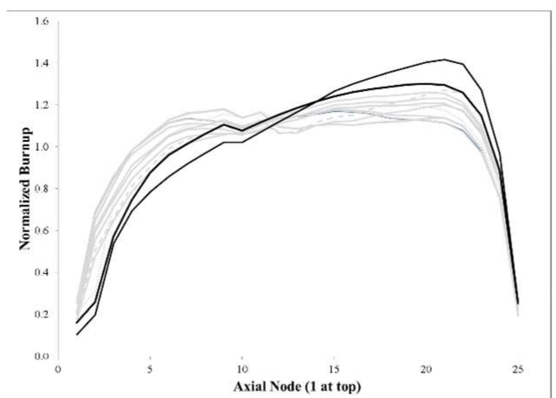
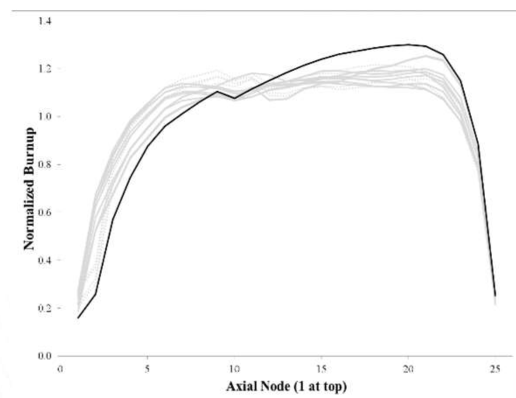
Applicability of Axial Burnup Profiles

- All profiles from SLB fuel – look at applicability to SLU and ML fuel
- SLB fuel has natural uranium blankets in at least upper 6 inches
  - Profiles should be more burned in the upper portion of fuel which contributes to end effect – observed with PWR blanketed and unblanketed fuel
- ML fuel less immediately clear due to change in lattice in upper portion of fuel
  - UNF-ST&DARDS profiles grouped into <25, 25-40, and >40 GWd/MTU corresponding to NUREG-7224 bins
  - Plotted along with NUREG limiting profiles – where bins intersect UNF-ST&DARDS profiles go in both.
  - SLB profiles substantially under burned in top portion of the compared to all profiles
  - Likely due to dryer upper lattice and many 12” blankets in CRC data

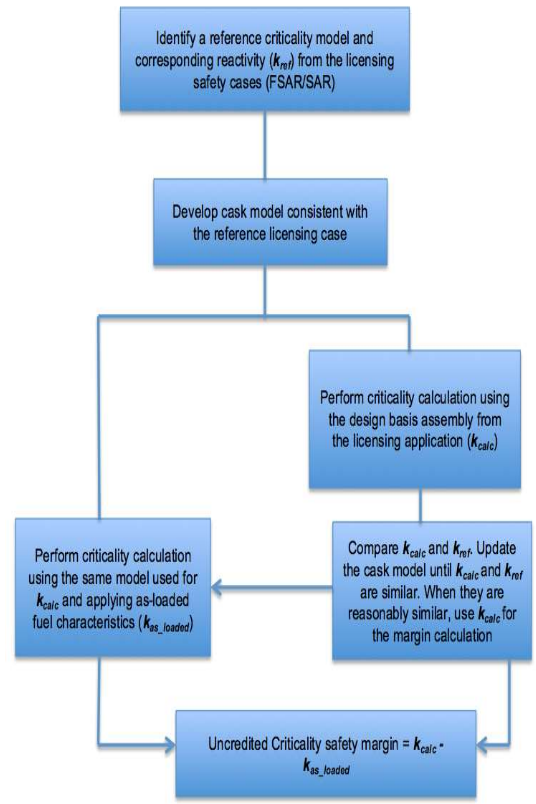






Criticality Analysis

- Model Individual assemblies with nominal assembly enrichments, burnups and cooling times for desired analysis date from GC-859
  - Transportation: In-service date to 2100
  - Disposal: In-service date to year 10,000
- Analysis performed for 163 BWR canisters at 8 sites
  - 3 canisters – one each for two legacy sites, MPC-68 for operating sites
- Transportation
  - Calculation of k_{eff} values for each date
  - Calculation of margin to the licensing basis
- Disposal
  - Degraded Neutron Absorber
    - Neutron absorber was not credited in the analysis
    - Assumed survival of basket structure for SS components but failure of CS components
    - Subcritical limit of 0.98 assumed



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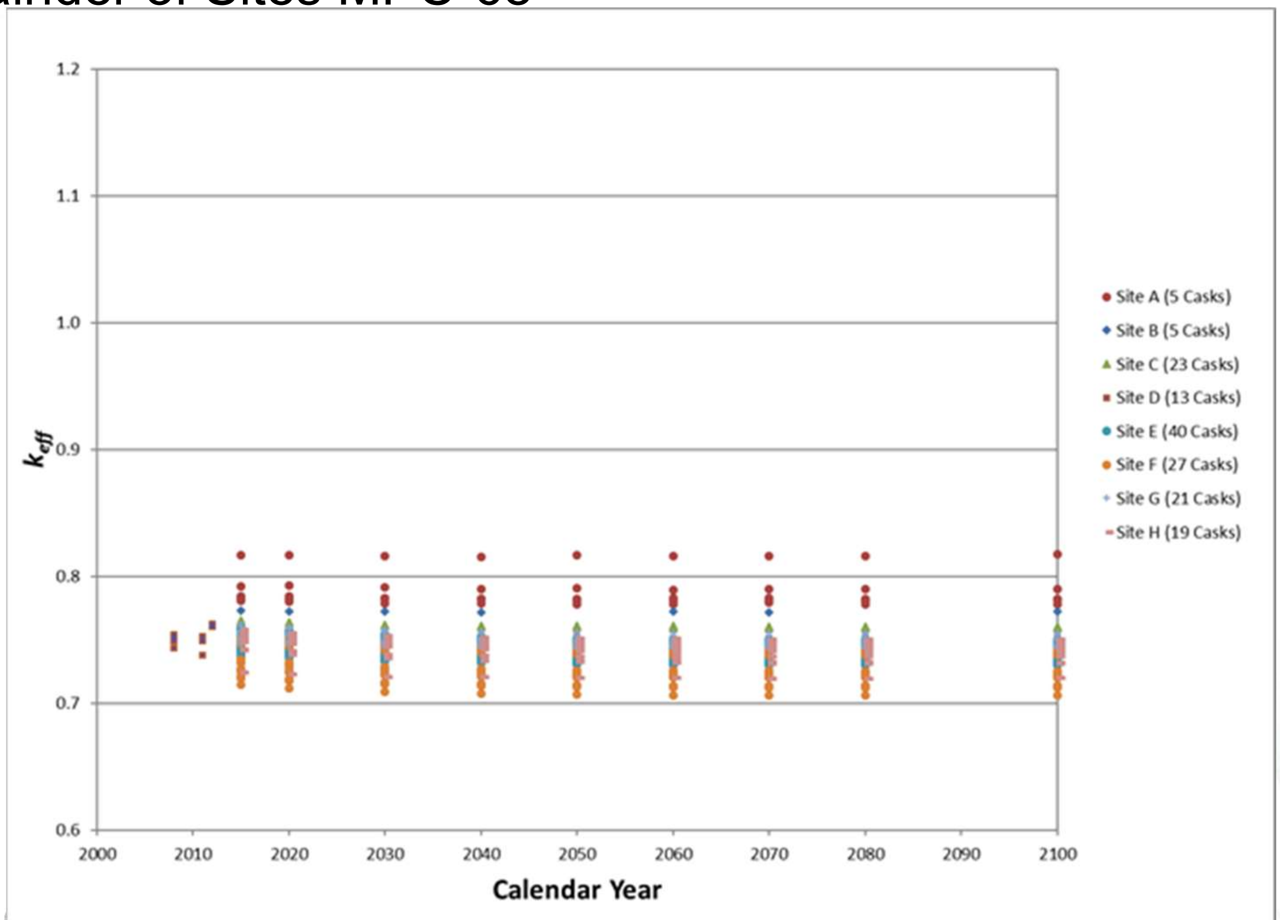
Transportation Results - k_{eff}

- Sites A and B legacy sites
  - Substantial fraction of location occupied by damaged fuel
  - Damaged fuel model as fresh fuel of nominal enrichment with optimum pitch permitted by storage cell
- Remainder of Sites MPC-68



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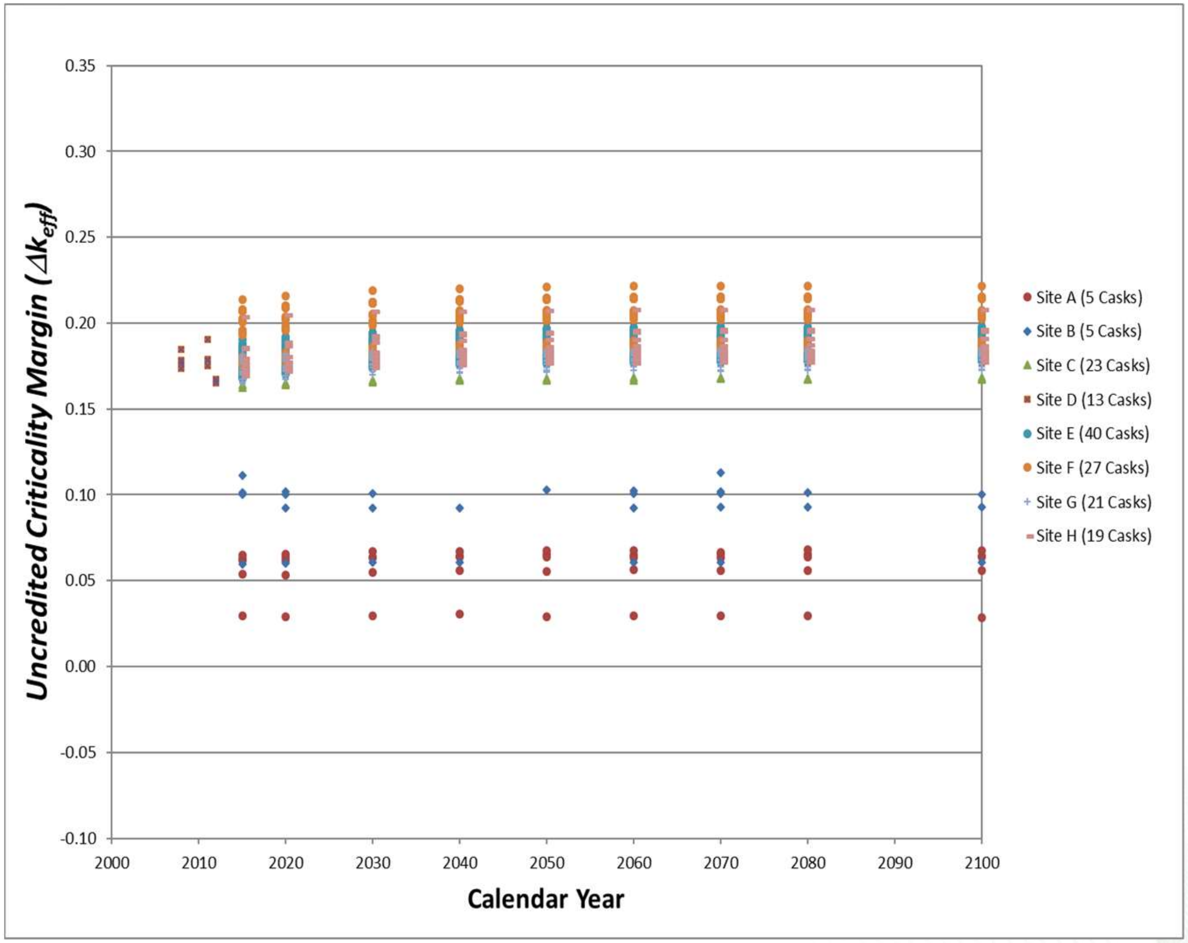
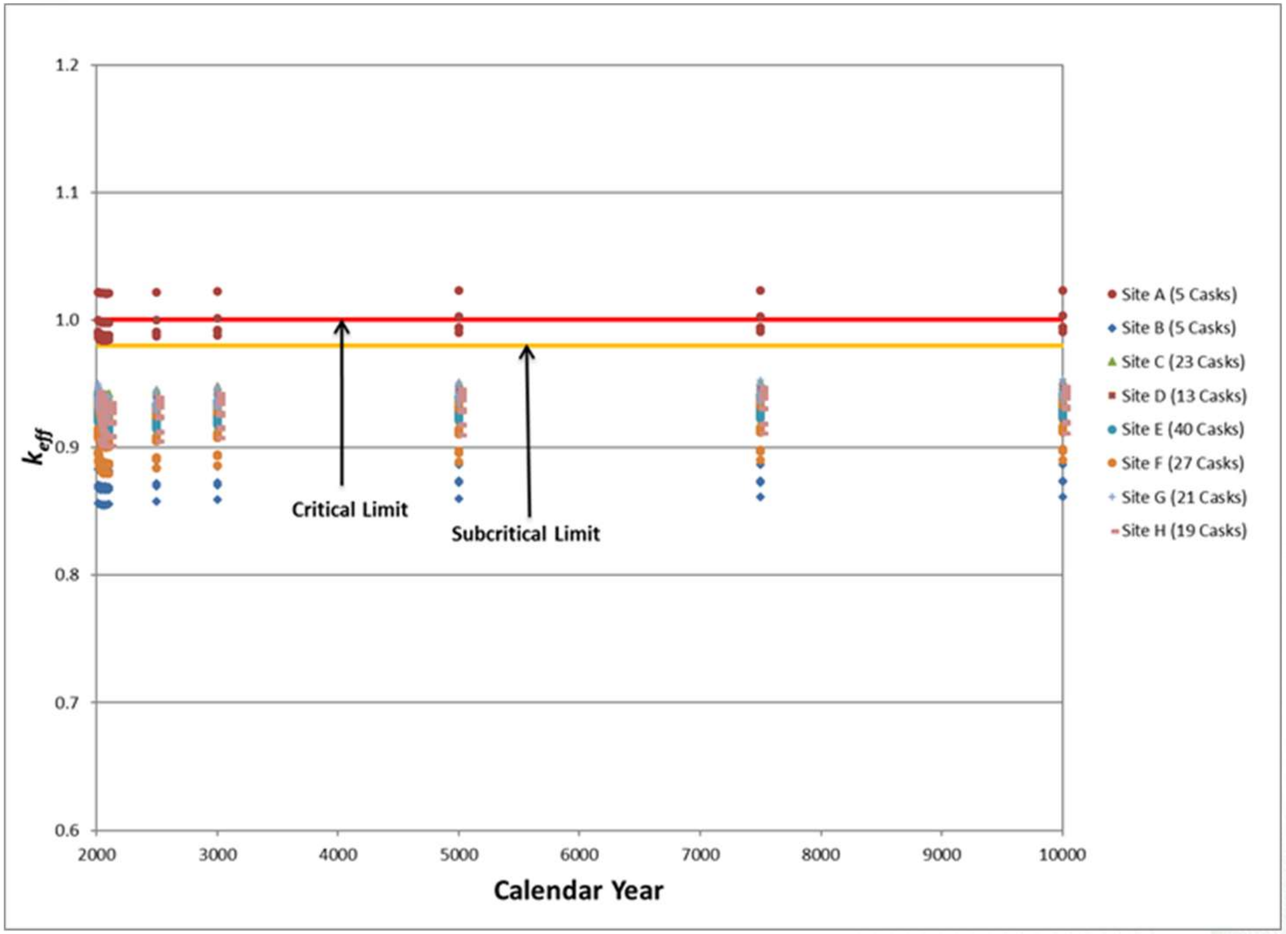
Transportation Results – Margin to Basis

- Site A and B canisters have $k_{ref} \approx 0.83$ in addition to damaged fuel
- MPC-68 $k_{ref} \approx 0.92$



Disposal Results

- Only canisters to exceed limit are the Site A canisters with Damaged fuel



Conclusions

- Used recently published research to establish burnup credit approach for as-loaded BWR casks
  - Bounding control rod insertion with assembly average moderator density
  - Selected axial burnup profiles from available CRC data
- Approach is overall very conservative owing to using blanketed profiles on unblanketed models
  - Still seems to be sufficient especially with more realistic damaged fuel treatment
- We are collecting more fuel, operational, and cask data for refining and expanding the approach