NEI 12-16, "Guidance for Performing Criticality Analyses of Fuel Storage at Light-Water Reactor Power Plants"

NRC/NEI Meeting on SFP Criticality Guidance
January 17th 2014 • Rockville, MD

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Introduction and Meeting Objectives

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January 14th, 2014 • Rockville, MD

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Agenda

Time	Subject	Presenter
8:30 am	Welcome, Introductions and Meeting Purpose	Kris Cummings, NEI
8:45 am	Overview of EPRI Sensitivity Studies	Hatice Akkurt, EPRI
9:15 am	Monte Carlo Code Benchmarking/Validation	Dale Lancaster, NuclearConsultants.com
10:15 am	**Break**	
10:30 am	Monte Carlo Code Benchmarking/Validation	Dale Lancaster, NuclearConsultants.com
11:30 am	Alternate Code Validation	Kris Cummings, NEI
12:30 pm	**Lunch**	
1:30 pm	NVF/SFP Abnormal/Accident Conditions	Dave Phegley, Exelon
3:30 pm	**Break**	
3:45 pm	BWR Criticality Overview	Kris Cummings, NEI
4:15 pm	Summary of Actions Captured	Kris Cummings, NEI

Meeting Purpose

- Reach resolution on methods to be used in spent fuel pool criticality analysis
- Focus on issues around criticality code benchmarking and accident conditions.
- Identify areas needing additional description, justification or explanation in NEI 12-16.

Overview of Proposed EPRI Sensitivity Studies

Hatice Akkurt[1] and Albert Machiels[2]

[1]Senior Project Manager, EPRI

[2]Senior Technical Executive, EPRI

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Background and Objectives

- A set of sensitivity computations will be performed to determine the impact of certain parameters on the criticality analysis of spent fuel pools in support of NEI12-16 Guidance document.
- Determination is based on the discussion from the 2nd Public Meeting, conducted on October 31, 2013.
- Specific focus is on issues around modeling of the **fuel storage rack** and **neutron absorbers**
  1. Geometric model sensitivity
     - Infinite vs. finite
     - Eccentric positioning
     - Rack wrapper tolerances
  2. SFP temperature
  3. Material compositions

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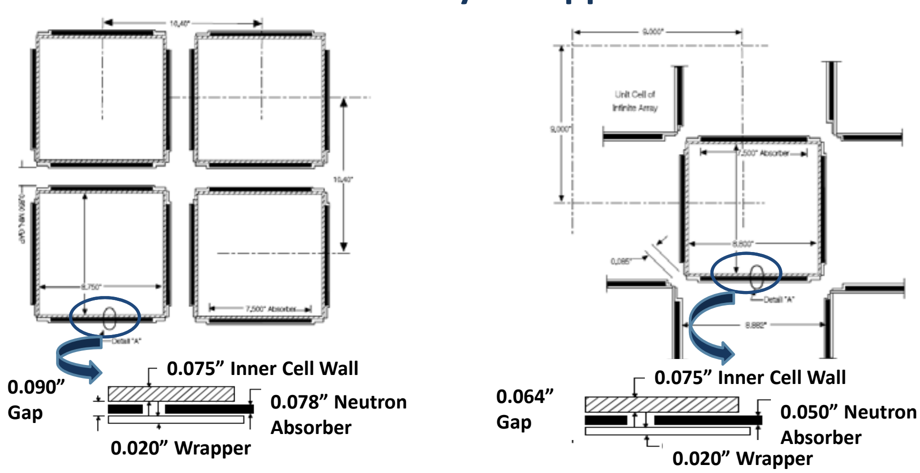
General Specifications for Sensitivity Analysis Computations

- Scale 6.1 with ENDF/B-VII 238-group cross-section library
- TRITON (t5-depl) module for depletion
- Depletion Parameters
  - Soluble Boron Concentration: 900 ppm
  - Fuel temperature: 1050 K
  - Moderator Temperature: 616 K
  - Moderator density: 0.60208 g/cm^3
- Analysis will be performed for
  - Westinghouse 17x17
  - Analysis will be performed using other fuel types, for a limited subset, to confirm the conclusions
- All the analysis will be performed at a single cooling time (100 hrs)

Geometric Model Sensitivity: Wrapper Plate Tolerances



Demonstrate the impact of wrapper plate tolerances on reactivity are negligible

Range of Parameters for Analysis

Region 1

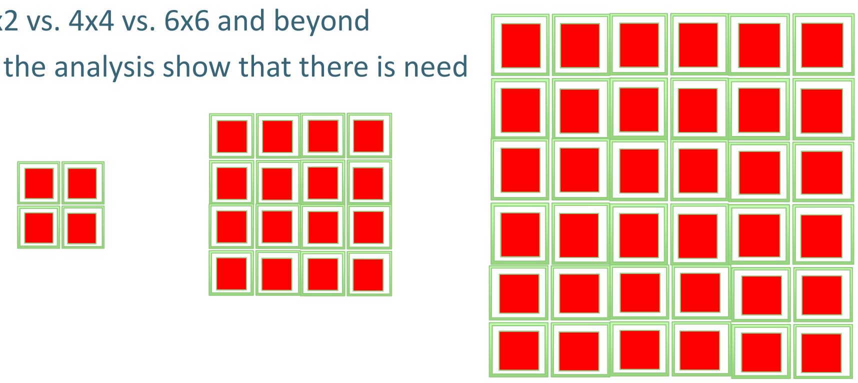
Areal Density (g/cm²)	Enrichment (wt%)	Burnup (GWd/T)
0.030	5	Fresh Fuel
0.015	3.5 and 5	0 and 10
None	2	0 and 20
	3.5 and 5	0; 20; 40

Region 2

Areal Density (g/cm²)	Enrichment (wt%)	Burnup (GWd/T)
0.030	2	0
	3.5	0 and 30
	5	0; 30; 60
0.015	2	0
	3.5	0 and 30
	5	0; 30; 60
None	2	0
	3.5	0 and 30
	5	0; 30; 60

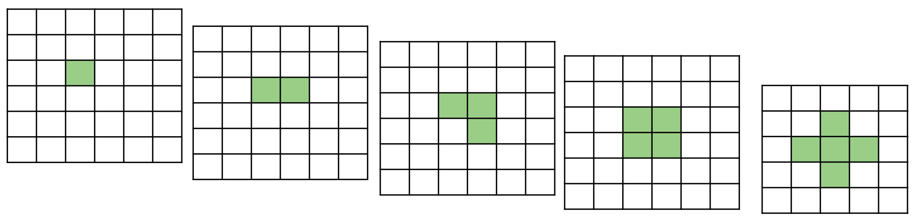
Geometric Model Sensitivity: Infinite vs. Finite

- How many cells/rows in a finite model make it equivalent to an infinite model for interface or accident delta k calculation?
- 2x2 vs. 4x4 vs. 6x6 and beyond
  if the analysis show that there is need



Geometric Model Sensitivity: Minimum Critical Volume

- How many co-located fuel assemblies are equivalent to an infinite model?
- Look at non-regular configurations of fuel assemblies to determine how many fuel assemblies, in what configuration are needed to achieve the minimum critical volume.
- Compare to infinite array (one cell with reflecting boundary)



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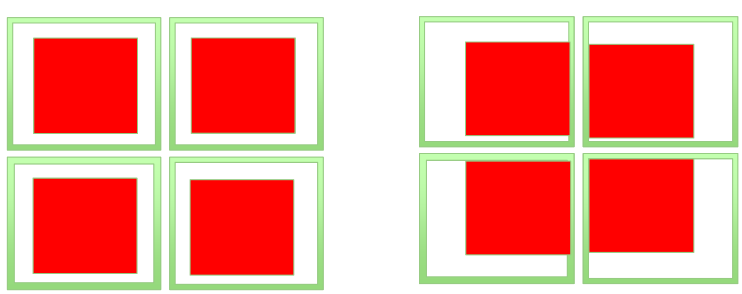
Geometric Model Sensitivity: Centered vs. Eccentric Positioning of Assemblies

- Assume 4 assemblies are eccentrically located and determine the reactivity difference between centered vs. eccentric for the set of rack designs.
- Determine if the results (Δk) sensitive to the number of cells used in the modeling/analysis, impact on 2x2 vs. 4x4 vs. 6x6



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Temperature, Composition, and Other Effects

- Confirm high and low temperature calculations are sufficient, no limiting conditions with intermediate temperatures
- Impact of concrete composition. Determine a conservative concrete composition (or set of compositions, if determined necessary) for SFP
- Provide justification, with analysis, to demonstrate that racks that are sufficiently separated (6 inches wall-to-wall) are neutronically decoupled and do not require further analysis

Summary

- Sensitivity analysis will be performed to determine the impact of certain parameters on criticality, including
  - Geometric Model sensitivity: Infinite array assumption; Eccentric positioning; Rack wrapper tolerances
  - SFP temperature
  - Concrete composition
- The proposed set of analysis was selected based on the discussions during the 2nd Public Meeting, October 31, 2013, for NEI Guidance Document
  - Feedback is requested to avoid any disconnect between the NRC and Industry expectations
- The scope of the sensitivity analysis will be expanded based on discussions at the subsequent meetings

Questions/Comments?

Monte Carlo Code Benchmarking/Validation

Dale Lancaster
NuclearConsultants.com

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Topics

- Overview
- Selection of Experiments
- Normality
- Trend Analysis
- Area of Applicability
- HTC/MOX criticals

Overview

- Covered in Section 3.2.1 and some of Section 3.2.2 of NEI 12-16
- Follows NUREG/CR-6698
- Five Steps from NUREG/CR-6698
  1. Identify range of parameters to be validated
  2. Select critical experiment data
  3. Model the experiments
  4. Analyze the data
  5. Define the area of applicability of the validation and limitations

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Identify range of parameters to be validated

- Isotopic Content
  - rack structure (e.g., stainless steel),
  - material for the cladding (e.g., zirconium),
  - fissile isotopes in the applicable enrichment range (e.g., U-235 for low enriched UO2, Pu-239 for MOX),
  - Water moderated
  - Absorber Materials (boron for the soluble boron and absorber plates, gadolinium for peak reactivity, and Ag/In/Cd if control rods are credited.

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Identify range of parameters to be validated

- Spectrum
  - should cover a range of spectra.
  - quantified by an index (e.g., EALF or others)
- Geometry
  - fuel pin pitch,
  - pellet or clad diameter,
  - assembly separation, and
  - boron areal density.

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Select critical experiment data

- Recommend using OECD/NEA criticality benchmark handbook
- Low enriched (5 wt% U-235 or less) UO2 to cover the principle isotopes of concern.
- Fuel in rods to assure that the heterogeneous analysis is correct.
- Square lattices to assure the lattice features used in the rack analysis are verified. (Not needed for Continuous Energy XS)
- Presence of absorbers (soluble boron, borated steel, boron bearing rods, sheets of aluminum with boron, BoraflexTM, and Ag-In-Cd).
- No emphasis on a feature or material not of importance to the rack analysis (e.g. lead reflectors).
- From diverse evaluations
- From diverse labs

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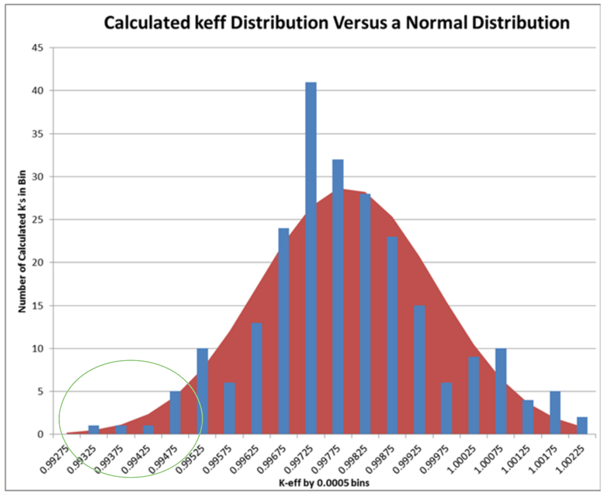
Issues

- Benchmark Quality:
  - Careful about Gd impurities which make some of the OECD/NEA benchmarks unacceptable
    - LEU-COMP-THERM-003, 004, 012
  - LEU-COMP-THERM-014 High Boron Uncertainty
- Moderator temperature does not need to be validated
- C_k analysis is not needed.
  - Trend analysis covers agreement between criticals and pools.
  - Critical experiments must cover the range of parameters of interest.

Normality

- NUREG/CR-6698 concentrated a validation using a small set (less than 50) for validation.
- References support that if using over 100 critical experiments normality assumption is acceptable unless strong non-normal distributions are seen.
- Nonparametric treatment is more appropriate for small sets. Using this for large set places too much emphasis on individual experiments and loses much of the information gained from the mean.
- Should show a histogram if using normality assumptions with poor normality results.

Typical Normality Histogram



Issues – Which Trends to Analyze

- Should cover materials: trend on enrichment and boron content
- Should cover cross section: trend on spectrum
- Should cover modeling approach: If using group cross-sections look for trends on pin diameter. Not needed for continuous energy models.
- Apparent trends may be caused by a parameter different than the trending parameter (e.g.: Trend on pin diameter could actually be caused by a spectral trend)

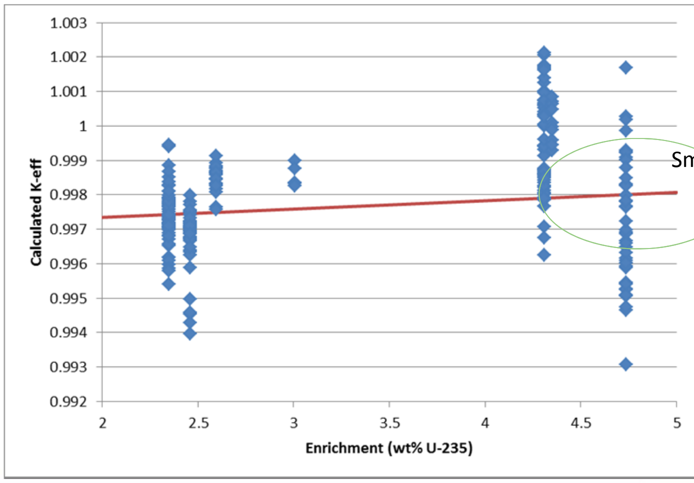
Issues: Statistically Significant

- NUREG/CR-6698 suggests a goodness of fit test but does not give an acceptance criteria
- Past reports have suggested other statistical tests where a trend is rejected unless there is a high confidence the slope of the trend is non-zero.
- Two options for evaluating trends:
  - If using the most limiting bias and uncertainty from all trends and from the set as a whole, no need to perform statistical significance test on trends.
  - Evaluate trends and eliminate trends that are not statistically significant.

Area of Applicability

- Area of Applicability should be defined.
- Small extrapolations should be acceptable using the trends
  - Enrichments lower than the data should be okay since low reactivity
  - Enrichments higher (4.7 to 5.0) okay due to small extrapolation.

Enrichment Range



Smaller Bias

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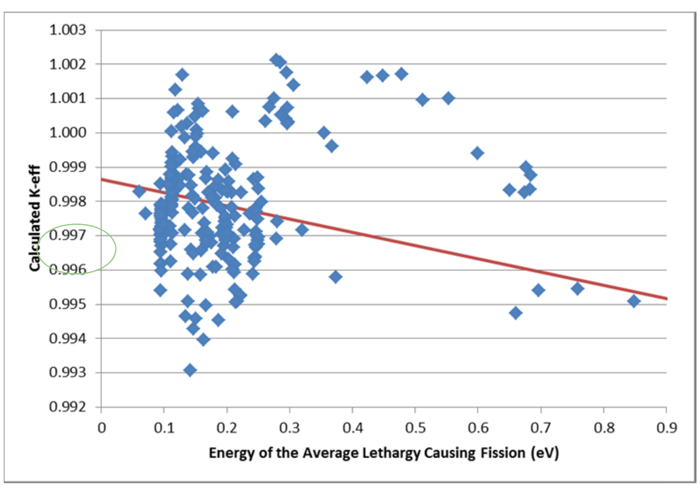
K-eff As A Function of EALF



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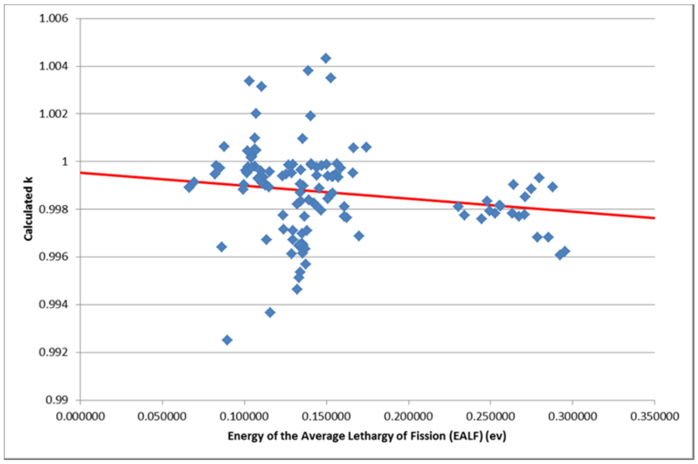
Major Actinide Validation For Used Fuel

- DSS-ISG-2010-001 includes discussion for validation of major actinides beyond U-235 and U-238. (Not required for EPRI depletion benchmark approach)
- HTC and MOX critical experiments include major actinides.

HTC Critical Experiments

- HTC criticals comprise over 100 critical experiments which model the Uranium and Plutonium content of 4.5 wt% fuel burned to 37.5 Gwd/T
- Do not have fission products
- Pu content of spent fuel can exceed HTC Pu content at higher burnups which Pu content in MOX criticals can address.

HTC Criticals as a Function of EALF



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MOX Critical Experiments

- 63 low enriched Pu (MOX criticals) lattice experiments in OECD Handbook
- ENDF/B-V through ENDF/B-VII predict higher k for Pu containing critical experiments than UO2 critical experiments. Therefore the UO2 data sets the limits.
- Mean k
  - **0.9978 for UO2 criticals**
  - **0.9988 for HTC criticals**
  - **0.9984 for MOX criticals**

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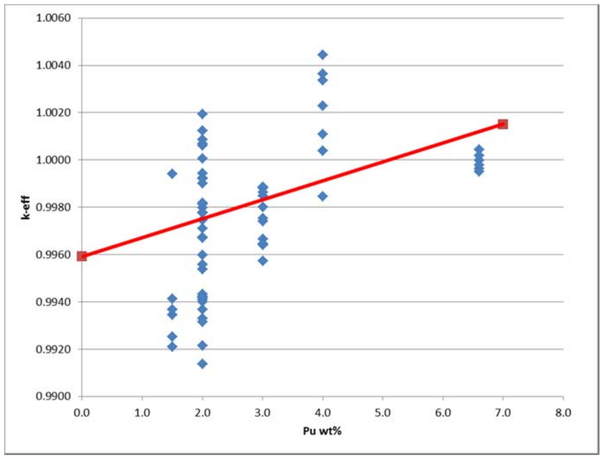
K as a function of Pu wt%



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Am-241

- Am-241 can be validated for all cooling times.
- MOX criticals with ENDF/B-VII show decreasing bias with increasing Am-241
- Crediting Am-241 with long cooling times is acceptable.

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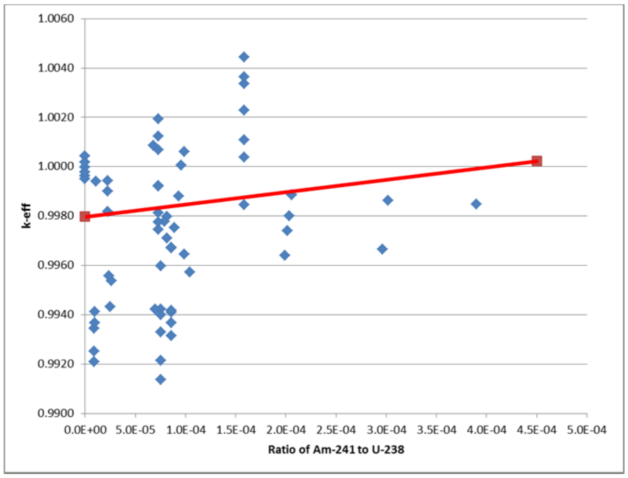
K as a function of Am-241 content



Conclusions

- Validation of fresh fuel is adequate using a large and diverse set of criticals from the OECD/NEA handbook
- Method from NUREG/CR-6698 should be followed.
- Normality is not a concern with large data sets but histogram is needed to check for concerns.
- Statistical test for significant trends is not required when using the most limiting bias and uncertainty from all trends and set as whole

Conclusions

- C_k analysis is not required if set of experiments cover the range of important parameters.
- For ENDF/B-V through ENDF/B-VII the bias for low enriched Pu systems is smaller than low enriched U systems so analysis of HTC or MOX criticals is not needed.

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Alternate Code Validation

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Value of Alternate Code Validation

- Allow more efficient, simpler codes to be used in certain cases
- Used if primary criticality code doesn't have functionality to model important benchmark experiments (most benchmarks)
- Primary code still needs to be able to model important neutronic and geometric aspects of storage configuration

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Method

- Primary code is used for criticality safety calculations
- Intermediary code is validated against appropriate benchmark experiments (discussed earlier).
- Primary code is validated against the intermediary code over a range of parameters (neutronic and geometric)

Important parameters

- Enrichment
- Burnup
- Absorber density
- Soluble boron content
- Storage rack geometry

Bias/Uncertainty Application

- Bias and uncertainty of primary code includes:
  - Bias and uncertainty of the intermediary code validation against the benchmark experiments
  - Bias and uncertainty of the primary code against the intermediary code
  - Biases are summed, negative biases conservatively set to zero
  - Uncertainties are combined statistically (sum of squares)

Abnormal / Accident Conditions
NFV & SFP

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Objectives

- Discuss proposed approaches to the analysis of abnormal and accident conditions for criticality safety analyses.
- Reach agreement on:
  - The approach to be used for assessing abnormal and accident condition impacts on criticality margin.
  - Actions to be taken by the group to close questions remaining for assessment options.

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Double Contingency Principle

- As stated in ANSI/ANS 8.1, Section 4.2.2 – "process designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible."
- Thus, the criticality safety analyses must demonstrate that criticality cannot occur without at least two incidents / abnormal occurrences that are:
  - Unlikely,
  - Independent, and
  - Concurrent
- No single occurrence can lead to a criticality event.

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Abnormal/Accident Conditions Analyzed

- All criticality safety analyses include:
  - Dropped / misplaced fuel bundle (SFP)
  - Bundle misloading (SFP)
  - Seismic event evaluation (SFP & NFV)
  - Loss of Cooling – abnormal pool water temperature (SFP)
  - Optimum moderation (moderator fog) event – unless physically precluded (NFV)
  - Flooded NFV conditions – unless physically precluded (NFV)
- PWR criticality safety analyses also include:
  - Boron dilution event (SFP)

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Dropped/Misplaced Fuel Bundle

- Dropped / misplaced assembly
  - The misplaced fuel assembly, adjacent to fuel rack(s) must be analyzed for locations that could possibly fit a fuel bundle (between rack/wall, between racks, etc.).
  - Especially important if no neutron absorber on exterior of rack
  - Placed adjacent to a fuel assembly in the rack at a distance based on the fuel assembly envelope and rack structure
  - A single bounding location may be utilized.

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Seismic Event

- Seismic event
  - Seismic analyses currently treat the SFP (including the neutron absorber) as if the materials are new (i.e., not degraded).
  - Neutron absorber is kept in place by sheathing, with little/no room for movement.
  - Neutron absorber loads during seismic event are only it's own weight (not a load bearing structure)

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Seismic Event (cont.)

- The seismic event analysis must account for any impact upon the specific neutron poison material present in the SFP racks (including localized poison material impacts).
- The seismic event must account for the level of degradation of neutron absorbers, based upon coupon and/or in-situ testing and industry experience.
- The results of the impact of the seismic event on the neutron absorber material must then be feed back into the criticality safety analysis to account for the impact on criticality margin.
- Impacts on the criticality safety margin must be reviewed with the minimum critical volume taken into account (i.e., averaged pool impacts are not sufficient to analyze as they may not be limiting vs. a smaller area impact that allows for a localized critical volume to form).

Seismic Event (cont.)

- Currently only the Boraflex neutron absorber material has exhibited sufficient degradation to have possible impacts from a seismic event. (Industry has moved away from crediting Boraflex)
- Boral's localized blistering and pitting will have minimal impact on the results of the seismic event.
- Future evaluation of Boral materials should provide confirmation of the strength of the Boral material after in-plant use.
- Similar tests for other material types could be considered to provide input information for seismic evaluations.

Misloaded Assembly

- Misloaded assembly
  - The analysis of single misloaded fresh fuel assembly is currently performed.
  - Expansion of the scope to cover multiple misloaded assemblies, based upon lessons learned from recent operating experience, and has been included in recent license amendment requests.

Misloaded Assembly (cont.)

- All criticality safety analyses must analyze for a single misloaded assembly – or justify why the normal conditions bound / represent the single misloaded assembly event. (i.e., qualified for fresh fuel)
- Need to consider misloaded assembly in different storage configurations/patterns. Can provide justification for a configuration to be bounding (i.e., located in empty cell)
- Multiple misload events have occurred rarely in the past:
  - For PWR pools where separate storage regions exist with differing storage limitations for fuel.
  - Typically the BWR SFP multiple misloading event does not need to be analyzed. (i.e., maximum reactivity)

Misloaded Assembly (cont.)

- In situations where the multiple misload event is possible, it must be analyzed, unless sufficiently robust administrative controls are in place to preclude a single mode failure leading to a multiple misload event.
  - The sufficiency of the administrative controls to preclude the single mode failure multiple misload event is dependent upon the complexity of the specific unit's SFP criticality loading restrictions.
  - In general, more convoluted SFP criticality loading restrictions require additional administrative controls to preclude multiple misload events from a single mode failure.
  - The criticality safety analysis must justify the scope of the multiple misload event evaluation based upon the unit's specific SFP loading restrictions and the specific administrative controls in place for SFP fuel moves.

Misloaded Assembly (cont.)

- In the event that a multiple-misload analysis is needed, it is necessary to define the conditions of the analysis.
  - Fresh, non-irradiated fuel assemblies can easily be distinguished from irradiated, dull assemblies. This eliminates the need to assume a fresh fuel assembly in every location. Additionally:
    - Most fresh fuel assemblies, both PWR and BWR, now contain a neutron absorber (Gd, IFBA, WABA, etc.,) which reduce the fresh fuel assembly reactivity.
    - Many PWR pools have a separate rack configuration for fresh fuel (flux-trap racks) that is visually distinct versus racks for spent fuel.
  - Assume a 100% misload and determine the minimum required burnup at maximum enrichment. Provide justification as to why this is acceptable.
  - Credit of the Tech Spec soluble boron content to demonstrate k_{eff} remains less than 0.95 (10 CFR 50.68).

Misloaded Assembly (cont.)

- In the event that a multiple-misload analysis is needed, it is necessary to define the conditions of the analysis.
  - Fresh, non-irradiated fuel assemblies can easily be distinguished from irradiated, dull assemblies. This eliminates the need to assume a fresh fuel assembly in every location. Additionally:
    - Most fresh fuel assemblies, both PWR and BWR, now contain a neutron absorber (Gd, IFBA, WABA, etc.,) which reduce the fresh fuel assembly reactivity.
    - Many PWR pools have a separate rack configuration for fresh fuel (flux-trap racks) that is visually distinct versus racks for spent fuel.
  - Credit of the Tech Spec soluble boron content to demonstrate k_{eff} remains less than 0.95 (10 CFR 50.68).
  - Need to discuss the specific details associated with analyzing the multiple misload event.

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Controls to Ensure Tech Spec Compliance

- With simple pool storage configurations (i.e., Region 1/Region 2 with a single uniform loading configuration in each rack), simple verification tools are sufficient to prevent multiple misloads.
- Existing computer codes for special nuclear material reporting requirements assist in ensuring Tech Spec Compliance.
  - Provides a report showing acceptable configuration for each fuel assembly
  - Provide plots/graphs to assist in identifying outliers
  - Color-coded maps allow visual verification of acceptability
  - Administrative procedural controls in place to ensure TS configuration is maintained.
  - Manual verification of acceptable fuel locations is acceptable
- Additionally, the loading requirements are an LCO in the Tech Spec, with actions to be taken in the event of a Tech Spec non-compliance (i.e., move fuel to be in compliance with the Tech Spec.).

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Controls to Ensure Tech Spec Compliance (cont.)

- For more numerous or complex TS loading configurations, additional controls should be instituted to ensure multiple misloads are prevented.
  - Validated software implementation to provide:
    - Surveillance reports to show acceptability of storage configurations
    - Graphical representation to augment manual verification
    - Pre-verify planned fuel moves and configuration
    - Visual (color coded) maps showing acceptability
  - Need detailed administrative procedures for implementation

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Conclusions

- The fuel bundle drop / mislocation event must account for all possible locations where a bundle may fit outside of a fuel rack (while only the limiting analysis may need to be specifically calculated).
- The seismic event impact on the unit specific neutron absorber materials must account for material degradation and must feed the impact back into the criticality safety analysis to account for criticality margin impacts.
- The misloaded assembly analysis must account for the multiple misload event unless the event can be shown to be not possible, bounded by existing analyses, or made sufficiently improbable through use of administrative fuel movement controls.

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Closing

- QUESTIONS
- COMMENTS
- DISCUSSION

BWR Criticality Primer

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Introduction

- Methodology for performing criticality safety evaluations for wet storage of BWR spent fuel in high-density racks.
- Assure that k_{eff} is less than or equal to 0.95 under all conditions for all assembly types present.
- Use of depletion code for in-core burnup calculations, k_{inf} in the SCCG, K_{inf} in the rack geometry.
- Monte-Carlo code is used to verify depletion code results and for 3-D geometries that are not possible in 2-D depletion code (i.e., eccentric positioning, accident conditions).

BWR Rack Design (Example)

- Stainless Steel Egg Crate Structure
- Box I.D.: 5.85” – 6”
- Pitch: 6” – 6.2”
- Absorber Loading: ~ 0.0200 g/cm^2 ^{10}B minimum
- Identical to PWR Region 2 racks (except dimensions)

Assumptions

- Moderator:
  - Pure water at temperature corresponding to highest reactivity (4° C).
- Axial, radial neutron loss neglected (k_{inf}).
- Minor structural members neglected.
- Conservative in-core operating conditions.
- Uniform planar average enrichments instead of distributed.
- Most reactive fuel assembly in each cell.

Reactivity Effect of Temperature/Voids (Example)

Temperature/Voids	Reactivity Change
4 ºC	Reference
20 ºC	-0.0025
50 ºC	-0.0074
110 ºC	-0.0178
120 ºC	-0.0233
120 ºC + 20% void	-0.0452

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Acceptance Criteria

- Three mutually exclusive acceptance criteria can be specified to ensure $k_{eff} \leq 0.95$.
  1. Maximum allowable enrichment.
  2. Maximum k_{inf} in the standard cold-core geometry (SCCG) for max. planar enrichment.
  3. Minimum Gadolinium loading in a specified number of rods for max. planar enrichment.

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Methodology

- Maximum Allowable Enrichment
  - Storage racks are shown to allow storage of lower enriched assemblies without consideration for burnup, gadolinium or the k_{inf} in the SCCG.
  - Includes applicable biases and uncertainties
  - Typical maximum enrichment allowed is ~3.0-3.3 wt% ^{235}U, depending on rack dimensions, ^{10}B loading, etc.

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Standard Cold-Core Geometry (SCCG)

- SCCG – defined as an infinite array of fuel assemblies on a 6-inch lattice spacing at 20°C without any voids or control absorbers.
- k_{inf} in the SCCG is typically provided by the fuel vendor for each unique planar (axial) region in a fuel assembly.
- Max k_{inf} in the SCCG (over entire range of burnups) for recently manufactured fuel is ~ 1.24 - 1.28.

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Methodology

- Maximum k_{inf} in the SCCG
  - Perform in-core depletion calculations for all assemblies present in spent fuel pool (No Gadolinium).
  - Restart in-core calculations with assemblies in the SCCG geometry and the rack geometry.
  - These calcs provide k_{inf} in the SCCG and k_{inf} in the rack geometry as a function of the burnup.
  - k_{inf} in the rack geometry is correlated to the k_{inf} in the SCCG for the entire range of burnups (Figure 1).

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Methodology

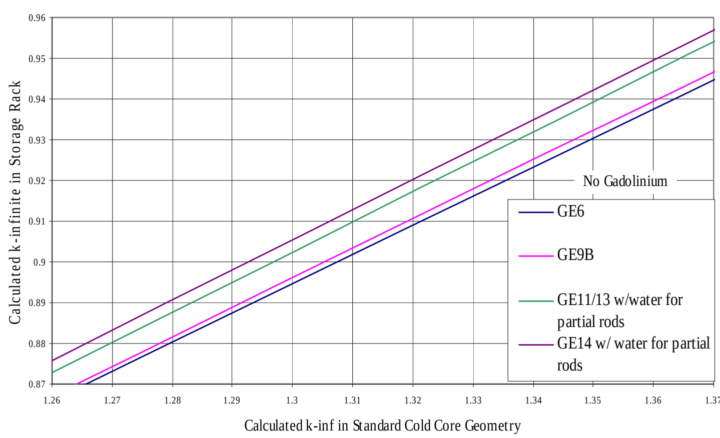

Figure 1: Correlation of the k_{inf} in the SCCG to the k_{inf} in the Fuel Storage Rack

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Methodology

- Maximum k_{inf} in the SCCG (cont.)
  - From this correlation, the max k_{inf} in the SCCG may be determined to ensure that the k_{eff} in the storage racks remains less than or equal to 0.95.
    - k_{eff} is determined from the calculated k_{inf} plus the allowances for manufacturing tolerances, uncertainty in depletion calculations and other correction factors (i.e., applicable biases and uncertainties).

Methodology

- Reference Fuel Assembly – for misc. calculations such as reactivity effect of manufacturing tolerances and accident/abnormal conditions.
  - Choose the assembly that gives the highest k_{inf} in the storage rack for a given k_{inf} in the SCCG (Fig. 1).
  - Assemblies with partial rods must be addressed dually due to 2D limitation of depletion codes; once with partial rod locations modeled as water, secondly with partial rod locations modeled as fuel rods.

Reactivity Allowances/Effects

- Manufacturing Tolerances
- Channel Distortion
- Uncertainty in Depletion Calculations
- Vendor Comparison

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Typical Manufacturing Tolerances (Example)

Tolerance	Variation	Δk
UO_2 density	± 0.20 g/cc	~ 0.0027
Enrichment	± 0.05 wt%	~ 0.0036
Cell Pitch	± ~ 1%	~ 0.0035
Cell Wall Thickness	± ~ 10%	~ 0.0009
Boral Width	± 1/16”	~ 0.0018
^{10}B loading	± 8%	~ 0.0045
Statistical Sum		~ 0.0075

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Reactivity Effect of Zircaloy Channel (Example)

Case	Reactivity Effect (Δk)
Reference Case – Thickest Channel	Reference
Fuel Channel Removed	~ -0.0060
Thinner Fuel Channel	~ -0.0007

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Reactivity Allowances/Effects

- Uncertainty in Depletion Calculations
  - Since not fully possible to benchmark depletion codes against critical experiments, it is necessary to include an allowance for uncertainty in the depletion calculations.
  - Uncertainty is assumed to be less than 5% of the reactivity decrement from zero burnup to the burnup corresponding to the maximum allowable k_{inf} in the SCCG.

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Reactivity Allowances/Effects

- Vendor Comparison
  - Fuel vendor calculates k_{inf} in the SCCG with different analytical tools.
  - A reactivity allowance of 0.01 Δk is conservatively added to the calculated rack k_{inf} (i.e., not statistically combined).

Reactivity Allowances/Effects (Example)

Summary	
Manufacturing Tolerances	~ 0.0075
Depletion Uncertainty	~ 0.0050
Vendor Comparison	~ 0.0100

Methodology

- Gadolinia Effects and Burnup
  - Gadolinia (Gd_2O_3) is used in all BWR fuel designs for augmenting reactivity control during in-core operations
  - Gd_2O_3 causes an assembly's reactivity to increase with burnup, reaching a max. when the Gd_2O_3 is virtually depleted (Figure 2).
  - The peak reactivity for each of the Gd_2O_3 loadings in Figure 2 is plotted as a function of the Gd_2O_3 content (Figure 3).
  - Linear interpolation is used to define the minimum acceptable loading for each rod configuration.

Methodology

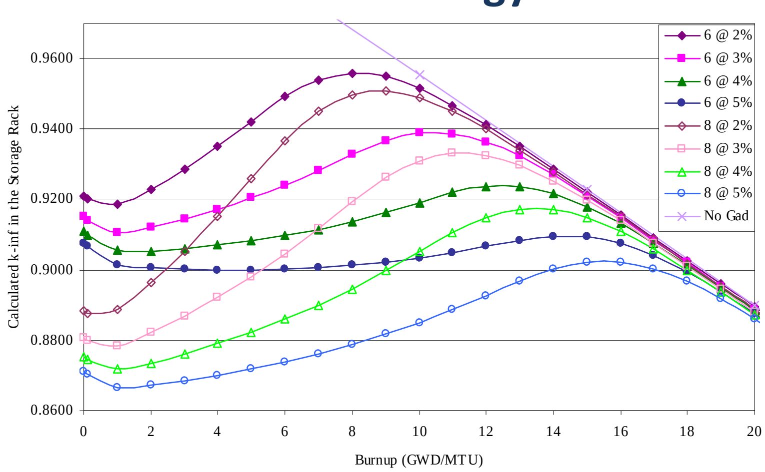

Figure 2: K_{inf} in the Storage Rack at Various Fuel Burnups and Gadolinium Loadings

Methodology

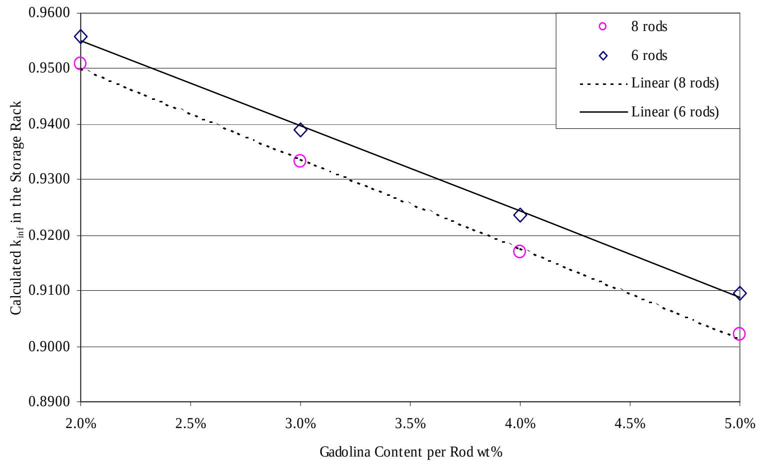

Figure 3: K_{inf} in Storage Rack as a Function of Gadolinia Concentration

Conclusion/Summary

- Three mutually exclusive acceptance criteria are detailed to ensure $k_{eff} \leq 0.95$.
  1. Maximum allowable average enrichment. (~3.2 wt% ^{235}U)
  2. Maximum allowable planar k_{inf} in the standard cold-core geometry (SCCG) for 5.0 wt% ^{235}U fuel. (~ 1.32)
  3. Minimum Gadolinium loading in a specified number of rods for 5.0 wt% ^{235}U fuel. (i.e., 3.5 wt% Gd_2O_3 in at least 6 rods)
- Each unique planar (axial) region must meet one of the three criteria above.