February 10, 1997

MEMORANDUM TO: Charles J. Haughney, Acting Director
Spent Fuel Project Office, NMSS

FROM: Earl P. Easton, Section Chief
Transportation and Storage Safety Section
Transportation Safety and Inspection Branch
Spent Fuel Project Office, NMSS

THRU: Susan F. Shankman, Chief
Transportation Safety and Inspection Branch
Spent Fuel Project Office, NMSS

SUBJECT: MEETING WITH THE DEPARTMENT OF ENERGY REGARDING ACTINIDE ONLY BURNUP CREDIT

On January 23, 1997, a meeting was held between representatives of the U.S. Nuclear Regulatory Commission, the Department of Energy (DOE), and its contractor, to discuss DOE's "Topical Report on Actinide Only Burnup Credit for PWR Spent Nuclear Fuel Packages." Attachment 1 is a list of attendees. Attachment 2 is a copy of the meeting handout, which includes an agenda. A notice of public meeting was issued on January 9, 1997.

The DOE is currently responding to NRC's request for additional information (RAI) on the topical report, dated March 22, 1996. DOE plans to submit formal responses to the RAI in March 1997.

During the meeting, DOE staff and contractors made presentations on the planned revisions to the topical in the areas of criticality validation, reactor records, and burnup measurements. A second meeting, tentatively scheduled for February 1997, is planned to present the revisions to the isotopic validation and cask modeling chapters of the topical report.

Attachments: 1. List of Attendees
2. Meeting Handout

cc: See Attached

Distribution (w/ Attachments 1 & 2)
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Distribution (w/ Attachment 1)
NRC Attendees

OFC	SFPO	c	SFPO	SFPO				
NAME	RLewis/lmg		EEaston	SShankman				
DATE	1/ /97		2/7 /97	1/10 /97				

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PDR

97-29

Meeting Summary Distribution

cc: R. Loux, State of Nevada
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M. Murphy, Nye County, NV
M. Baughman, Lincoln County, NV
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D. Weigel, GAO
P. Niedzielski-Eichner, Nye County, NV
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R. Holden, NCAI
T. Burton, NIEC
S. Brocoum, YMPO
R. Arnold, Pahrump, NV
N. Stellavato, Nye County, NV
J. Lyznicky, AMA
R. Miller, YMPO
B. Russo, EPA
A. Gil

ATTENDANCE LIST
Meeting between NRC and DOE Staff
January 23, 1997

Charles Haughney	NRC/NMSS/SFPO
Marissa Bailey	NRC/NMSS/SFPO
Donald Carlson	NRC/NMSS/SFPO
Dennis Damon	NRC/NMSS/FCSS
Allen Howe	NRC/NMSS/SFPO
Eric Leeds	NRC/NMSS/SFPO
Robert Lewis	NRC/NMSS/SFPO
Charles Marotta	NRC/NMSS
Bruce Moran	NRC/NMSS/FCSS
Fritz Sturz	NRC/NMSS/SFPO
Dennis Vinson	NRC/NMSS/DWM
Bernie White	NRC/NMSS/SFPO
Carl Withee	NRC/NMSS/SFPO
Priscilla Bunton	DOE
Christopher Kouts	DOE
Bill Lake	DOE
Ram Murphey	DOE
Daniel Thomas	DOE/OCRWM/YMP-M&O
Emilio Fuentes	M&O/TRW
Chi Kang	M&O/TRW
Dale Lancaster	M&O/TRW
Meraj Rahimi	M&O/TRW
Mark Wisenburg	M&O/TRW
Andrew Cheslirman	BNFL Instruments
Archer Haskins	BNFL
Paul Highberger	JAI
Kyle Jones	Vectra
Robert Sweeney	IBEX
Gary Tjersland	Holtec Intl.
Altheia Wyche	SERCH Licensing/Bechtel
Al Zimmer	General Atomics

ATTACHMENT 1

Technical Exchange On PWR Actinide-Only Burnup Credit

January 23, 1997

Contents

ATTACHMENT 2

Agenda

8:00 am	Introductory Remarks	NRC/DOE
8:15 am	Overview of Technical Program	DOE
8:30 am	Status of Topical Revisions	DOE
9:00 am	Criticality Validation	DOE
10:00 am	Break	
10:15 am	Verification of Reactor Records	DOE
11:15 am	Open Discussion	NRC/DOE/Public
11:45 am	Recap and Closing Remarks	NRC/DOE/Public

U.S. Department of Energy
Office of Civilian Radioactive Waste Management

NRC/OCRWM TECHNICAL EXCHANGE ON BURNUP CREDIT

PRESENTED TO

U. S. NUCLEAR REGULATORY COMMISSION

PRESENTED BY

Bill Lake, USDOE
Office of Waste Acceptance, Storage and Transportation

January 23, 1997

TODAY'S TOPICS OF DISCUSSION

- TECHNICAL PROGRAM OVERVIEW
- STATUS OF DOE RESPONSE TO RAI
- CRITICALITY VALIDATION
- VERIFICATION OF REACTOR RECORDS

MILESTONES

- SUBMIT TOPICAL REPORT (5/31/95)
- RAI SENT TO OCRWM (3/22/96)
- TECHNICAL EXCHANGE MEETINGS
  - General Response Plan (5/31/96)
  - Criticality Validation, Measurements (1/23/97)
  - Isotopic Validation, Cask Modeling (2/97)
- SUBMIT REVISED TOPICAL REPORT (3/31/97)

MAJOR CHANGES

- ADDITIONAL ISOTOPIC DATA
- ADDITIONAL CRITICALS
- TREATMENT OF DATA
- MODELING EFFECTS
- VERIFICATION OF REACTOR RECORDS
- SAMPLE CASES

CRITICALITY SAFETY

- FRESH FUEL ASSUMPTION
  - Safety considerations
  - Inherent safety margin: Unquantified
- ACTINIDE-ONLY BURNUP CREDIT
  - Added safety considerations: Bounded
  - Ignored fission products
    - Inherent safety margin: Estimated

SAMPLE ANALYSES

- MAY 1995 TOPICAL REPORT
  - MPC CONCEPTUAL DESIGN
- REVISED TOPICAL REPORT
  - ACTUAL CASK DESIGNS
    - General Atomics: GA-4
    - Holtec: HI-STAR MPC-32
    - Nuclear Assurance Corporation: STC
    - Transnuclear: TN-40
    - VECTRA: NUHOMS-24P, NUHOMS-MP187

Overview of the Status of Actinide-Only Burnup Credit Topical Report Revisions

Dale Lancaster
January 23, 1997

Introduction

- Objective of the Meeting
- Changes to the Topical Report Since Last Meeting
- Schedule
- Summary

Objective of the Meeting

- Summarize the changes in the topical report due to the response to NRC questions.
- Present the changes in detail to Chapters 3, 5 and 6.
- Provide a revised schedule for response activities.
- Receive some feedback.

Key Changes to the Topical Report

- Isotopic Validation (Chapter 2)
  - Increasing the chemical assay data from 19 samples to greater than 45 samples.
  - Parametric fit of trends.
  - Uncertainties as a function of the parameters.
  - Am-241 is treated solely as a daughter product of Pu-241

Key Changes to the Topical Report

- Criticality Validation (Chapter 3)
  - Increasing the number of critical experiments used from 34 to 57
  - Changed the statistical approach to allow the uncertainty to vary as a function of spectrum. (This change makes the statistical approach consistent between Chapters 2 and 3)
  - Defined the Upper Safety Limit as the most limiting of that from either the UO_2 or MOX set of experiments.

Key Changes to the Topical Report

- Package Modeling Assumptions (Chapter 4)
  - Axial burnup profile data base is increased from 500 to 3135 shapes.
  - All qualifying axial shapes were analyzed and the most conservative shape selected.
  - For burnups beyond 25 GWD/MTU the method recommends a k bias unless analysis is performed with the worse shape.
  - An assembly horizontal burnup gradient database was formed and conservative tilts identified. The cask analysis must use these conservative horizontal tilts.

Key Changes to the Topical Report

- Loading Criteria (Chapter 5)
  - The loading curve will not be reduced to cover reactor record uncertainty.
  - The utility will demonstrate that the assembly reactor record burnup is sufficiently above the loading curve to cover the uncertainties.
- Burnup Verification (Chapter 6)
  - A measurement device specification is given.
  - An assembly rejection criteria is clearly defined.
  - Detector system designs are moved to the Appendix

Key Changes to the Topical Report

- Appendices
  - There will be sample calculations performed by several cask vendors on their casks. These casks will be casks for which a license or a license application exists with the NRC.
  - The sample calculations will be performed with the vendor's code system so the methodology will be demonstrated with more than just SCALE 4.2 and the 27BURNUPLIB.
  - Description of burnup measurement systems will be included as appendices.

Schedule

- Two technical reports
  - Validation for Actinide-Only Burnup Credit ~ March 15, 1997
  - Package Modeling Assumptions for Actinide-Only Burnup Credit ~ March 15, 1997
- Final responses and revision of the Topical March 31, 1997
- One or Two More Progress Meetings with the NRC February, 1997

Conservatism in Cask Analysis

All of the following must occur for the cask to go critical:

1. The cask floods.
2. The k_{eff} is greater than expected by 5%.
3. The calculations of k are as high as possible (not the average) when compared to the criticals.
4. The arrangement in the cask is most limiting.
5. The assemblies have no burnup.

If Burnup Is Assured, The Following Additional Items Are Required To Go Critical:

5. The error in the burnup reactivity change is equal to or greater than:

   a) The fission product worth, plus

   b) The worth of increasing *each* isotope's concentration in the most reactive direction by the largest error possible given the experiments.

6. The worst USL of UO_2 or MOX applies.

7. Each assembly in a cask has the worst axial profile.

If Burnup Is Assured, The Following Additional Items Are Required To Go Critical:

8. Each assembly has the worst horizontal tilt and is loaded in the most limiting direction.

9. Each assembly has the burnup loading limit value.

10. All parameters in burnup analysis are at their most limiting condition for each assembly.

11. Each assembly is at the lowest burnup allowed by the reactor record uncertainty.

Summary

- Added more data to address comments.
- Added significant amount of analysis.
- Continue to Use a conservative approach.
- Will complete on March 31, 1997.

Criticality Validation for

Actinide-Only Burnup Credit

(Chapter 3)

Emilio Fuentes
January 23, 1997

Objective

- To present a methodology for validating a computer code system to calculate the subcritical multiplication factor of a spent nuclear fuel package

Introduction

- (A) Acceptability of benchmark set of critical experiments
- (B) Equality analysis on UO_2-only/MOX experiments
- (C) Selection of appropriate trending analyses
- (D) Methodology to compute Calculational Uncertainty
- (E) Methodology to compute Upper Safety Limit
- (F) Results for SCALE 4.2

(A) Benchmark Set of Critical Experiments

- Selection
  - 34 from original Topical report
  - 23 additional experiments
  - High concentration of higher isotopes of Plutonium
  - Available experiments with similar conditions as light water reactor fuel assemblies

(A) Benchmark Set of Critical Experiments

Experiment Case	Enrichment U-235 (wt %)	Description	Original Reference
LWR UO_2 Fuel Pin Lattices - 19 Experiments			
Absorber plates:			
Experiment 1	2.35	Exp. 005: No Absorber (H_2O)	Ref. 3-5
Experiment 2	2.35	Exp. 017: Boral	Ref. 3-5
Experiment 3	2.35	Exp. 024: Aluminum plates	Ref. 3-5
Experiment 4	2.35	Exp. 028: Stainless steel	Ref. 3-5
Reflecting walls:			
Experiment 5	4.31	Urani[illegible] walls	Ref. 3-6
Experiment 6	4.31	[illegible] walls	Ref. 3-6
Experiment 7	4.31	Steel walls	Ref. 3-7
Soluble Boron:			
Experiment 8	4.31	Exp. 173: no boron	Ref. 3-8
Experiment 9	4.31	Exp. 177: 2550 ppmb	Ref. 3-8
Experiment 10	4.31	Exp. 178: no boron	Ref. 3-8
Experiment 11	4.31	Exp. 181: 2550 ppmb	Ref. 3-8
Flux traps:			
Experiment 12	4.31	Exp. 214R: flux traps (no voids)	Ref. 3-9
Experiment 13	4.31	Exp. 214V3: flux traps (with voids)	Ref. 3-9
UO_2 triangular lattices:			
Experiment 14	2.35	EPRI MOX Comparison	Ref. 3-12
Experiment 15	2.35	EPRI MOX Comparison	Ref. 3-12
UO_2 square lattices:			
Experiment 16	2.46/4.02	UO_2/Gd_2O_3 Comparison	Ref. 3-11
Experiment 17	5.74	SAXTON MOX Comparison	Ref. 3-13
Experiment 18	5.74	SAXTON MOX Comparison	Ref. 3-13
3x3 assy arrays w/absorbers:			
Experiment 19	2.46	Core IV: 84 B_4C pins 1 pitch between assemblies	Ref. 3-10

(A) Benchmark Set of Critical Experiments

UO_2-Gadolinium Lattice - 2 Experiments			
UO_2/Gd_2O_3 fuel rods:			
Experiment 20	1.94/2.46/4.02	Core 14: 12 Gd fuel rods	Ref. 3-11
Experiment 21	1.94/2.46/4.02	Core 16: 16 Gd fuel rods	Ref. 3-11
LWR Mixed Oxide Criticals - 36 Experiments			
EPRI UO_2/PuO_2 2wt%PuO_2:			
Experiment 22	0.71	0.700-in. pitch, 0 ppmb	Ref. 3-12
Experiment 23	0.71	0.700-in. pitch, 688 ppmb	Ref. 3-12
Experiment 24	0.71	0.870-in. pitch, 0 ppmb	Ref. 3-12
Experiment 25	0.71	0.870-in. pitch, 1090 ppmb	Ref. 3-12
Experiment 26	0.71	0.990-in. pitch, 0 ppmb	Ref. 3-12
Experiment 27	0.71	0.990-in. pitch, 767 ppmb	Ref. 3-12
SAXTON UO_2/PuO_2 6.6wt%PuO_2:			
Experiment 28	0.71	0.52-in. pitch	Ref. 3-13
Experiment 29	0.71	0.56-in. pitch	Ref. 3-13
Experiment 30	0.71	0.56-in. pitch, 337 ppmb	Ref. 3-13
Experiment 31	0.71	0.735-in. pitch	Ref. 3-13
Experiment 32	0.71	0.792-in. pitch	Ref. 3-13
Experiment 33	0.71	1.04-in. pitch	Ref. 3-13
PNL4976 MOX and UO_2 2wt%PuO_2:			
Experiment 34	MOX 0.71 UO_2 4.31	MOX and UO_2 rods in uniform pattern	Ref. 3-14

(A) Benchmark Set of Critical Experiments

PUP UO_2/PuO_2 triangular 2.0wt% PuO_2 8.0wt% Pu-240:			
Experiment 35	0.71	0.80-in lattice spacing	Ref. 3-15
Experiment 36	0.71	0.93-in lattice spacing	Ref. 3-15
Experiment 37	0.71	1.05-in lattice spacing	Ref. 3-15
Experiment 38	0.71	1.143-in lattice spacing	Ref. 3-15
Experiment 39	0.71	1.32-in lattice spacing	Ref. 3-15
Experiment 40	0.71	1.386-in lattice spacing	Ref. 3-15
PUP UO_2/PuO_2 triangular 2.0wt% PuO_2 16.0wt% Pu-240:			
Experiment 41	0.71	0.93-in lattice spacing	Ref. 3-15
Experiment 42	0.71	1.06-in lattice spacing	Ref. 3-15
Experiment 43	0.71	1.143-in lattice spacing	Ref. 3-15
Experiment 44	0.71	1.32-in lattice spacing	Ref. 3-15
PUP UO_2/PuO_2 triangular 2.0wt% PuO_2 24.0wt% Pu-240:			
Experiment 45	0.71	0.80-in lattice spacing	Ref. 3-15
Experiment 46	0.71	0.93-in lattice spacing	Ref. 3-15
Experiment 47	0.71	1.05-in lattice spacing	Ref. 3-15
Experiment 48	0.71	1.143-in lattice spacing	Ref. 3-15
Experiment 49	0.71	1.32-in lattice spacing	Ref. 3-15
Experiment 50	0.71	1.386-in lattice spacing	Ref. 3-15
PUP UO_2/PuO_2 triangular 4.0wt% PuO_2:			
Experiment 51	0.71	0.85-in lattice spacing	Ref. 3-15
Experiment 52	0.71	0.93-in lattice spacing	Ref. 3-15
Experiment 53	0.71	1.05-in lattice spacing	Ref. 3-15
Experiment 54	0.71	1.143-in lattice spacing	Ref. 3-15
Experiment 55	0.71	1.386-in lattice spacing	Ref. 3-15
Experiment 56	0.71	1.60-in lattice spacing	Ref. 3-15
Experiment 57	0.71	1.70-in lattice spacing	Ref. 3-15
Total Number of Experiments = 57			

(A) Benchmark Set of Critical Experiments

- Range of Applicability
  - Experiments span anticipated SNF package conditions

(A) Benchmark Set of Critical Experiments

Key Parameters	Anticipated SNF Package Conditions	57 Critical Experiments
Fuel Rod Parameters		
<u>Fuel Composition</u> Isotopic Composition	Spent Fuel	U-234 U-235 U-236 U-238 Pu-238 Pu-239 Pu-240 Pu-241 Pu-242 Am-241
Burnup	0 to 50 GWd/MTU	Unirradiated UO_2 and MOX fuel
Initial enrichment (wt % U-235)	0 71 to 5 83%	UO_2 2 35 to 5 74% $UO_2Gd_2O_3$ 1 94 to 4 02% MOX 0 71 to 4 31% (wt % U-235) 2 to 6 6% (wt% PuO_2)
Cooling Time	5 to 100 Years	N/A
<u>Fuel Material Nuclear Properties</u> Fuel Temperature	70°F	70°F
Fuel Material Form	Fuel Irradiated UO_2 Cladding Zircaloy Stainless Steel	Fuel UO_2 UO_2 - Gd_2O_3 MOX Cladding Zircaloy Aluminum Stainless Steel
Fuel Material Density	10 0 to 10.4 g/cm^3 (91% to 95% of theoretical density)	UO_2 9 2 to 10 4 g/cm^3 UO_2 - Gd_2O_3 9 5 to 10 2 g/cm^3 MOX 9 5 to 10 4 g/cm^3
Fuel Rod Geometry	Square lattice, heterogeneous 0 77 to 0 97 cm pellet dia 0 007 to 0 012 cm gap 0 91 to 1 12 cm cladding OD	Square and triangular lattices, heterogeneous 0 86 to 1 28 cm pellet dia 0 to 0 009 cm gap 0 99 to 1 44 cm cladding OD
Fuel rod spacing	1 20 to 1 47 cm pitch	UO_2 1 42 to 2 21 cm pitch UO_2 - Gd_2O_3 1 64 cm pitch MOX 1 32 to 4 32 cm pitch

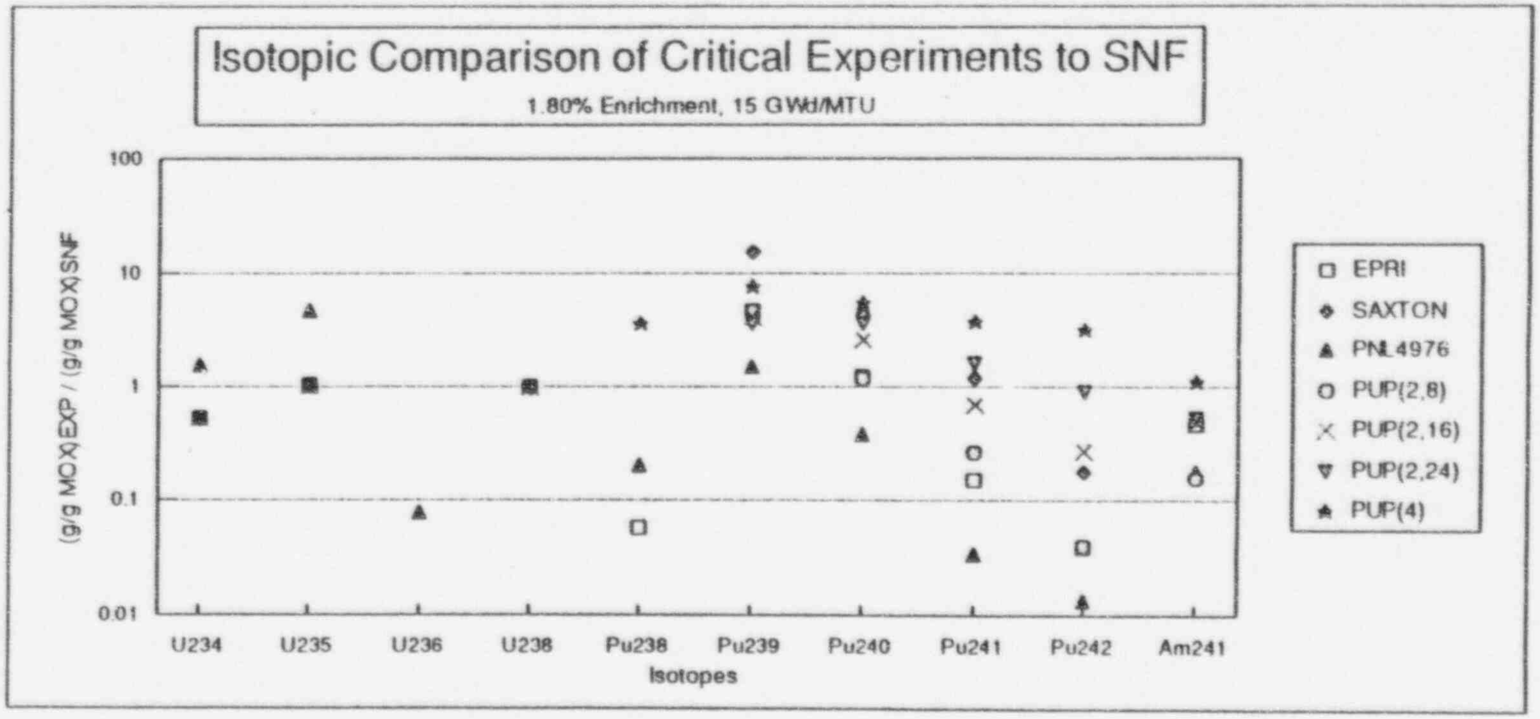
(A) Benchmark Set of Critical Experiments

Array Parameters		
Fixed Neutron absorbers	Borated materials (e.g., B_4C, borated stainless steel, boral, etc.)	External B_4C rods Boral, stainless steel, and aluminum plates
Materials of construction within array	Guide tubes	Water gaps
Moderator Conditions		
Water Density	$1 g/cm^3$	$1 g/cm^3$
Water Temperature	70°F	70°F
Moderator to Fuel Volume Ratio	1.5 to 2.3	UO_2: 1.09 to 5.07 UO_2 - Gd_2O_3: 1.88 MOX: 0.49 to 10.75
Soluble Boron Concentration	0 ppm	UO_2: 0 to 2550 ppm UO_2 - Gd_2O_3: 1579 to 1654 MOX: 0 to 1090 ppm
Reflector and Interaction Conditions		
Reflector Composition	Water, depleted uranium and stainless steel reflectors	Depleted uranium, lead, water, and stainless steel reflectors
Interaction with other fissile material	Fissile uranium and plutonium isotopes in SNF	Fissile uranium and plutonium isotopes in MOX fuel pins

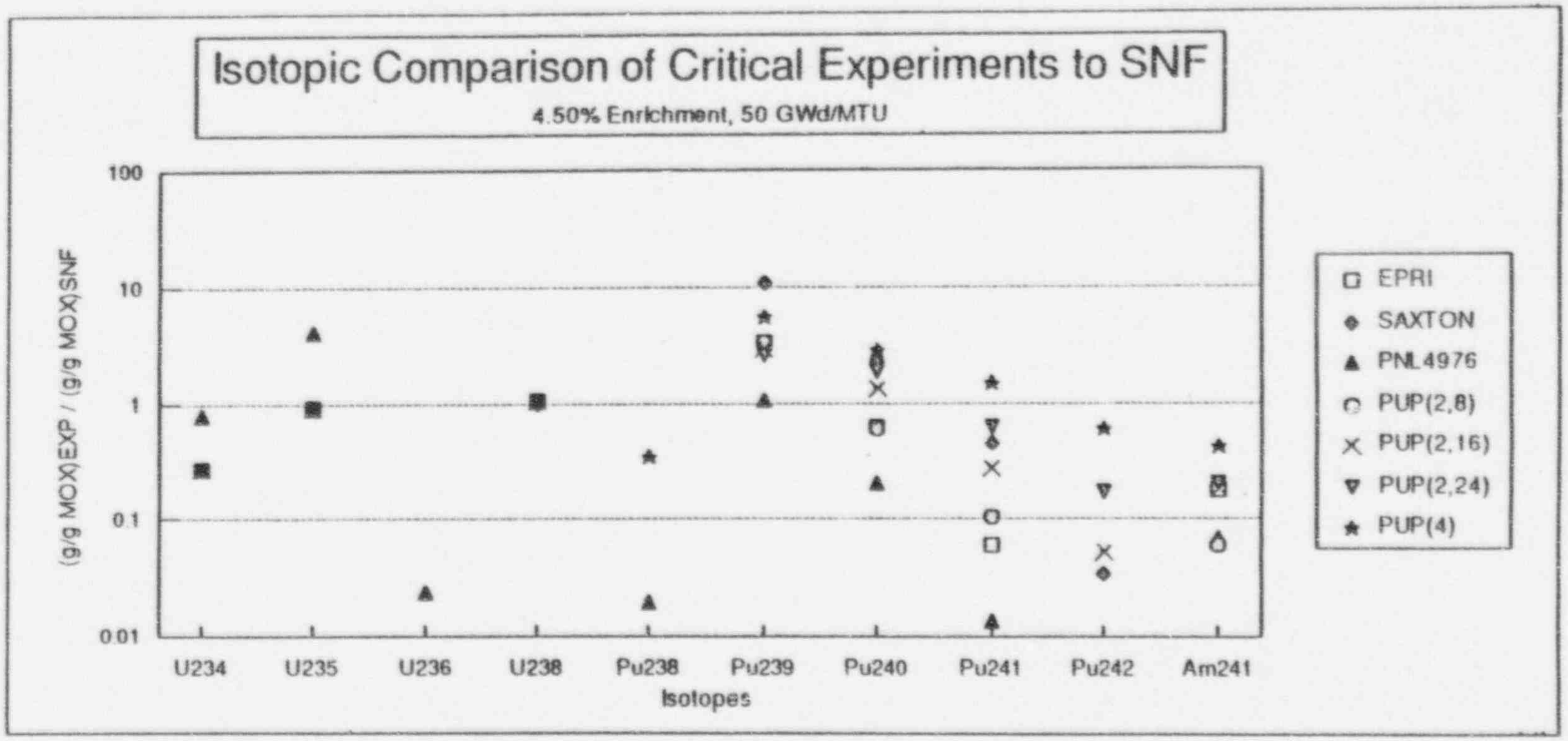
(A) Benchmark Set of Critical Experiments

- Comparisons with SNF using measured values and representative values from SAS2H (NITAWL, ORIGEN-S), CSAS (NITAWL, KENO) modules of SCALE 4.2
  - Isotopic concentrations
  - Actinide ratios
  - Neutronic behavior (flux spectra)

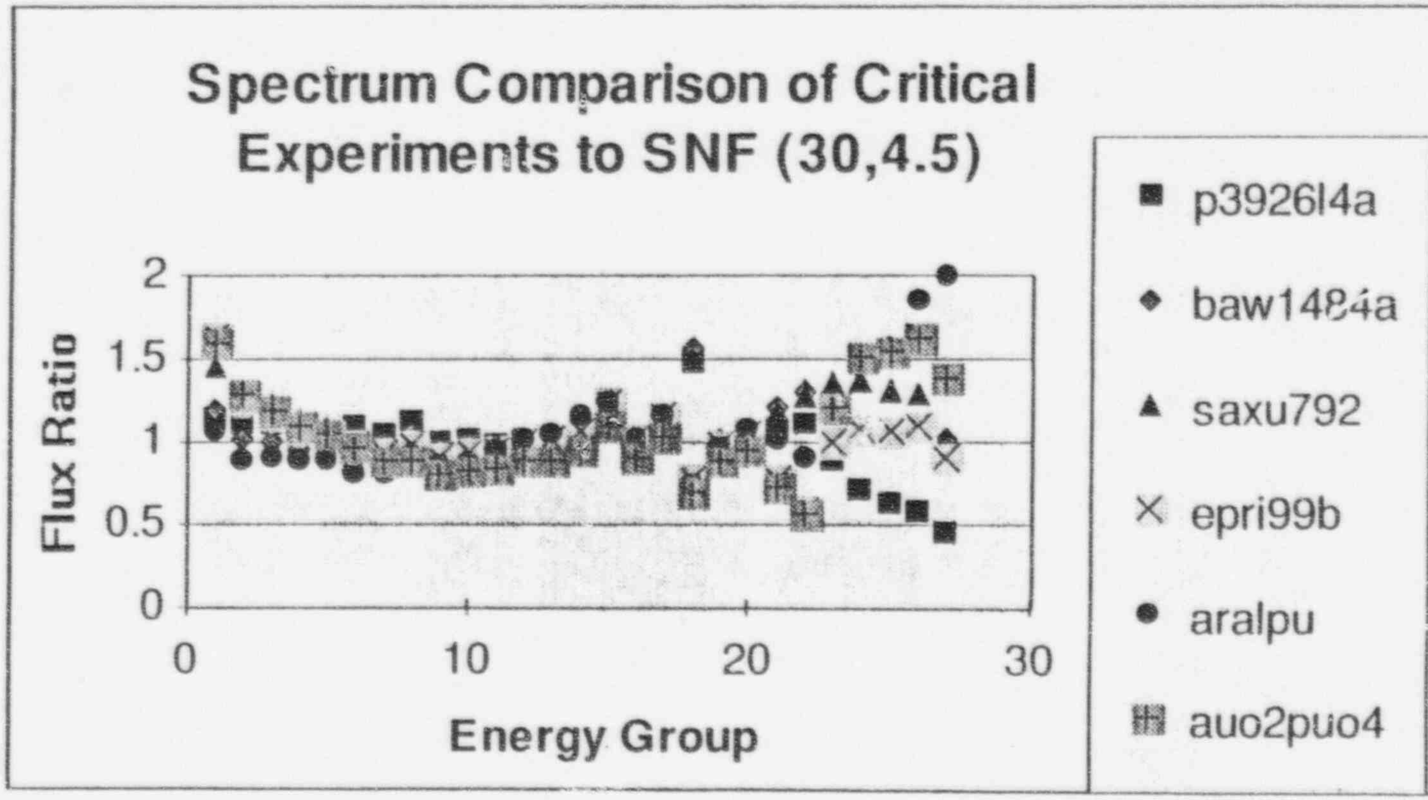
(A) Benchmark Set of Critical Experiments



(A) Benchmark Set of Critical Experiments



(A) Benchmark Set of Critical Experiments



(A) Benchmark Set of Critical Experiments

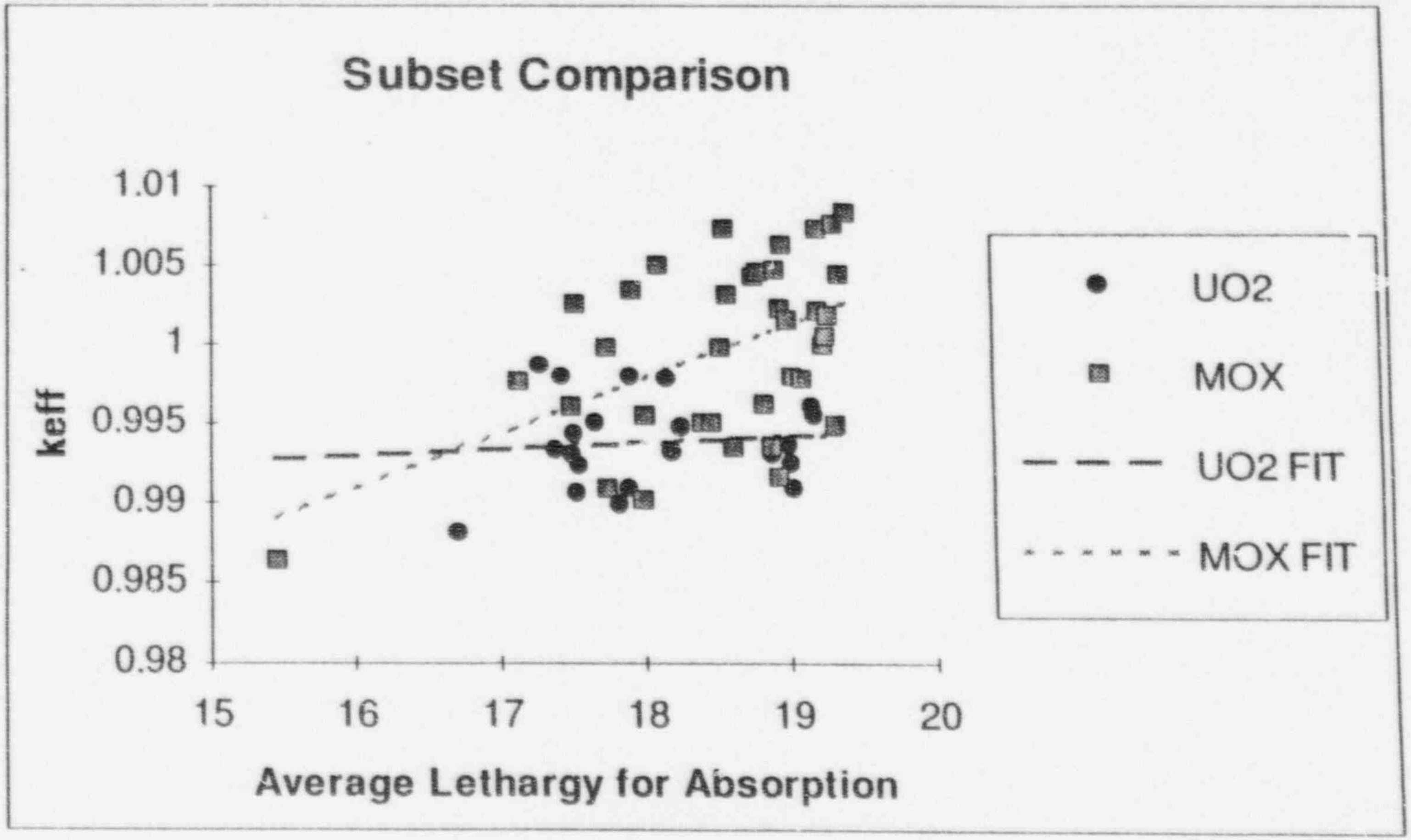
- Adequate and Appropriate for Criticality Validation

(B) Equality Analysis

- Benchmark set of experiments comprised of two subsets of data: 21 UO_2-only and 36 MOX experiments

- Can the two subsets be statistically combined?

(B) Equality Analysis



(B) Equality Analysis

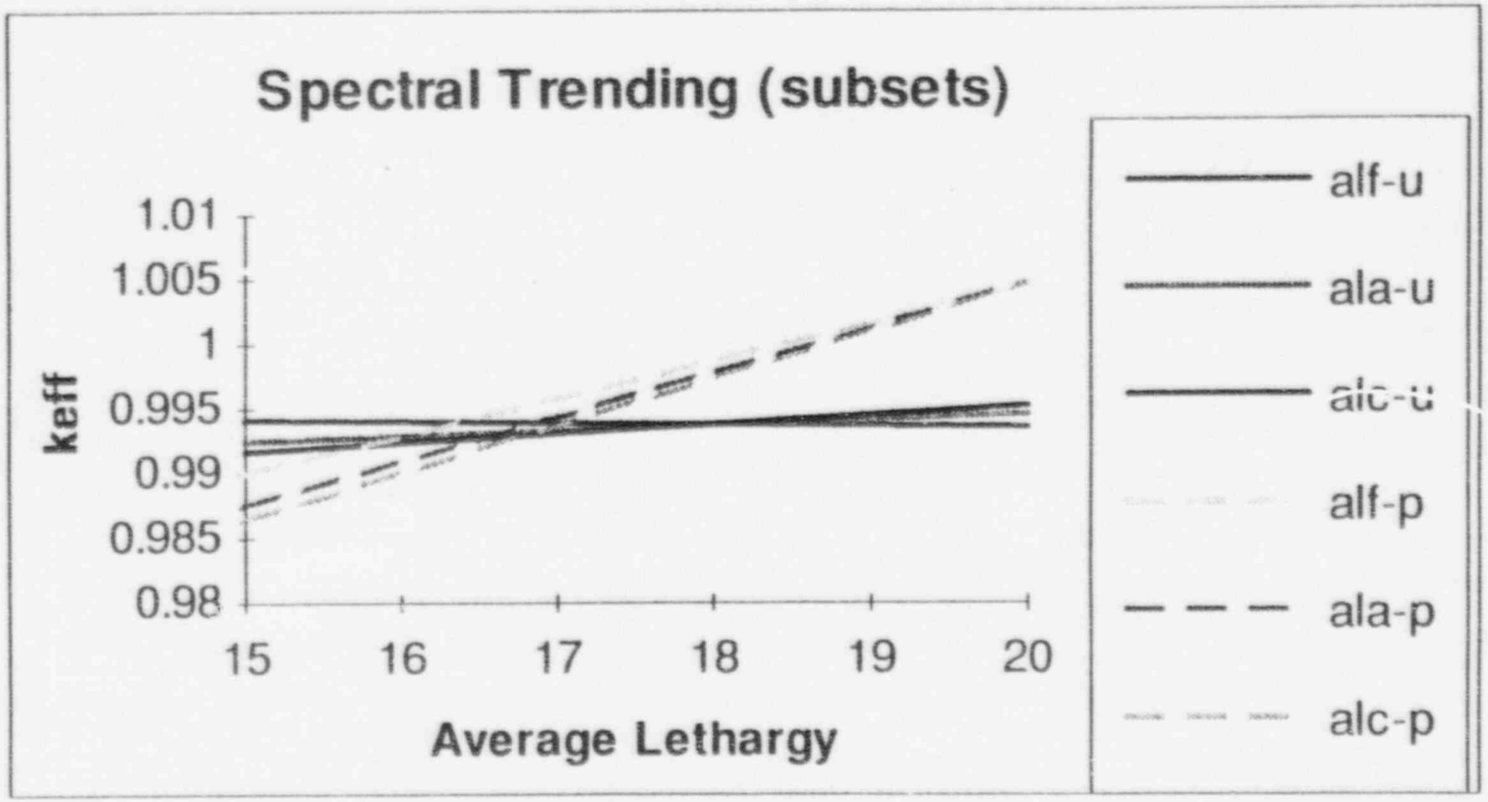
- From graph, data subsets seem independent

- Equality test on data supports hypothesis that the two subsets can NOT be statistically combined (95% confidence)

(C) Selection of Appropriate Trending Analyses

- Spectral trending
  - Average Energy Group for fission (AEG)
  - Average Lethargy for Fission (ALF)
  - Average Lethargy for Absorption (ALA)
  - Average Lethargy for Capture (ALC)

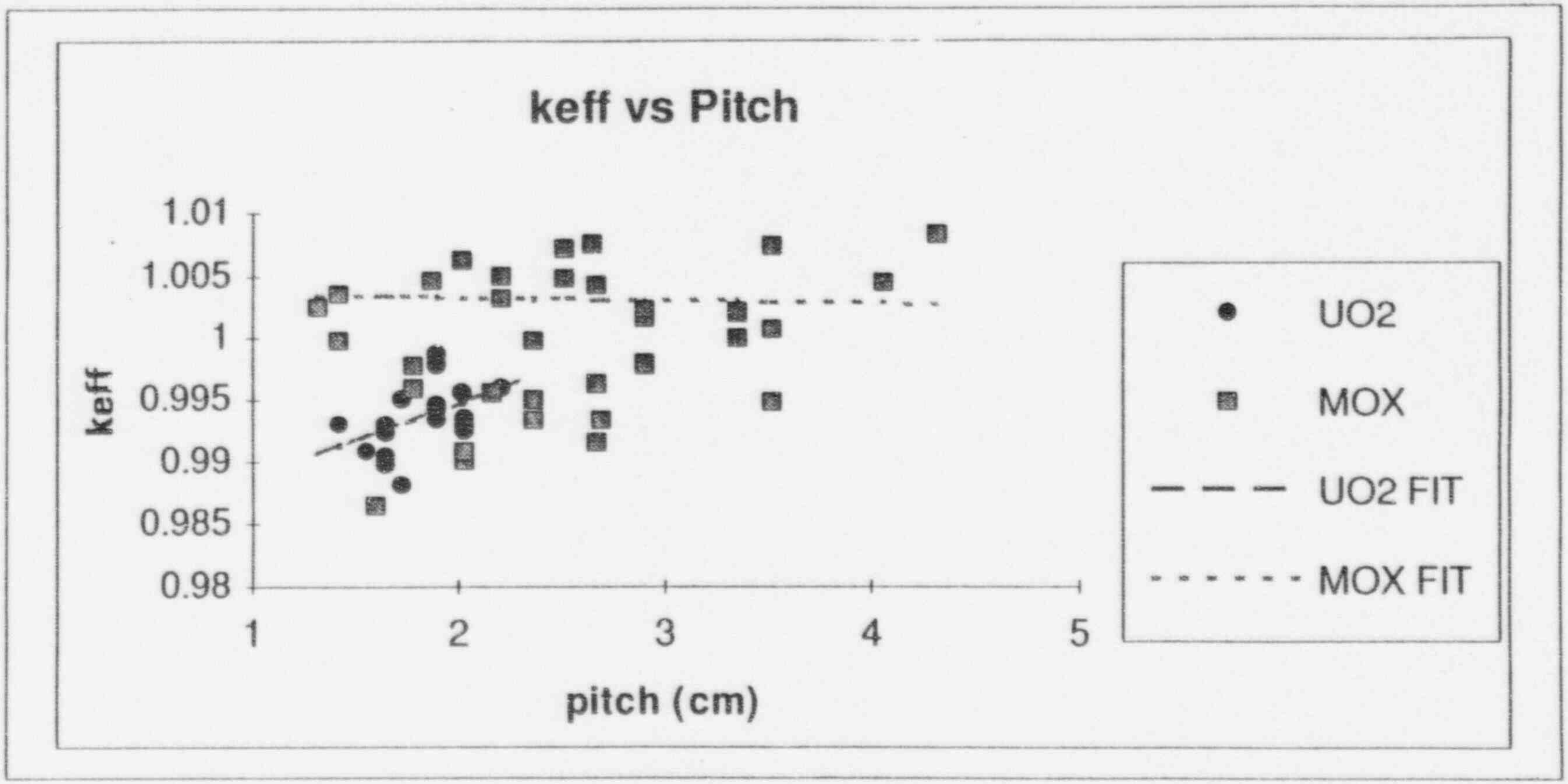
(C) Selection of Appropriate Trending Analyses



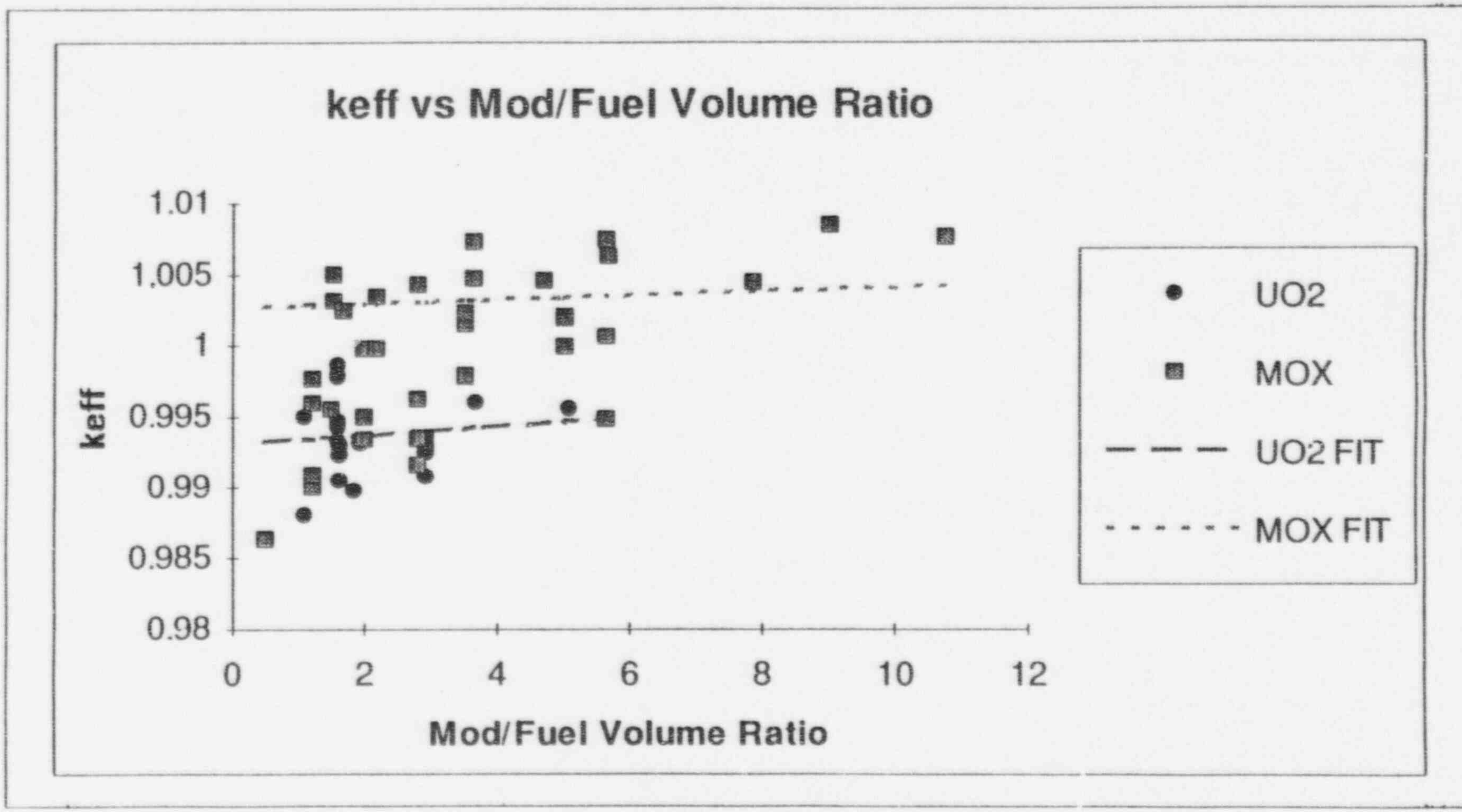
(C) Selection of Appropriate Trending Analyses

- General trending
  - Initial enrichment
  - Fuel density
  - Pellet diameter
  - Gap thickness
  - Clad outside diameter
  - Fuel rod spacing (pitch)
  - Moderator/Fuel volume ratio
  - Soluble boron concentration
- Can NOT show a trend with 95% confidence for any parameter, either subset

(C) Selection of Appropriate Trending Analyses



(C) Selection of Appropriate Trending Analyses



(C) Selection of Appropriate Trending Analyses

- Trending by nuclide
  - Reaction rate lethargies for each isotope
    - Absorption
    - Fission
    - Capture
- Can NOT show a trend with 95% confidence

(C) Selection of Appropriate Trending Analyses

- Plutonium trending
  - PuO_2 concentration
- Trend with 95% confidence

(C) Selection of Appropriate Trending Analyses

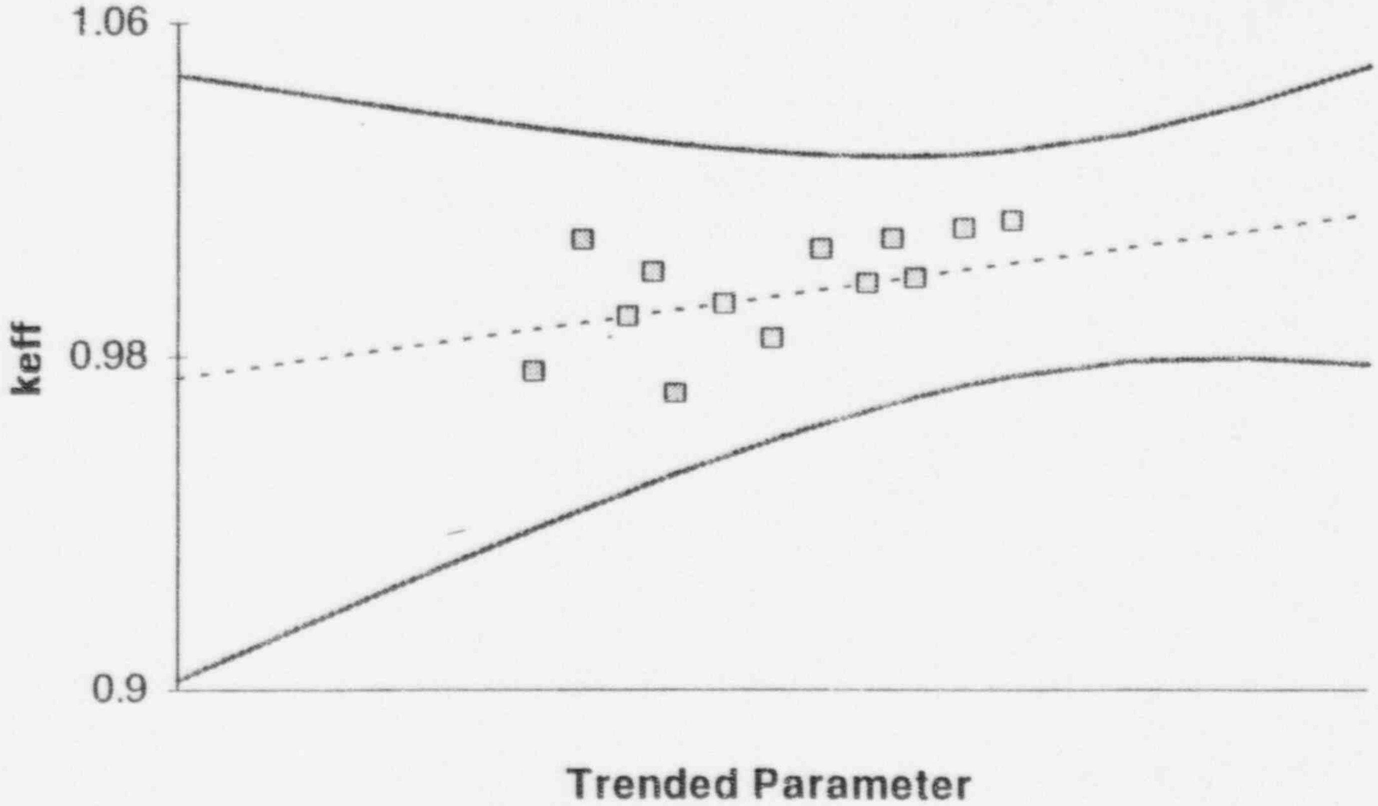
- Trend only with Average Lethargy for Absorption (ALA) for each subset

(D) Criticality Calculations Uncertainty

- Statistical method to compute uncertainty for multiple regression analysis
- Same method for chapters 2 and 3
- For one dimension:

$$\Delta k_c(ALA) = t_{\alpha,n-2}\sqrt{\left(\frac{n+1}{n}+\frac{(ALA-\overline{ALA})^2}{S_{xx}}\right)S_p^2}$$

(D) Criticality Calculations Uncertainty

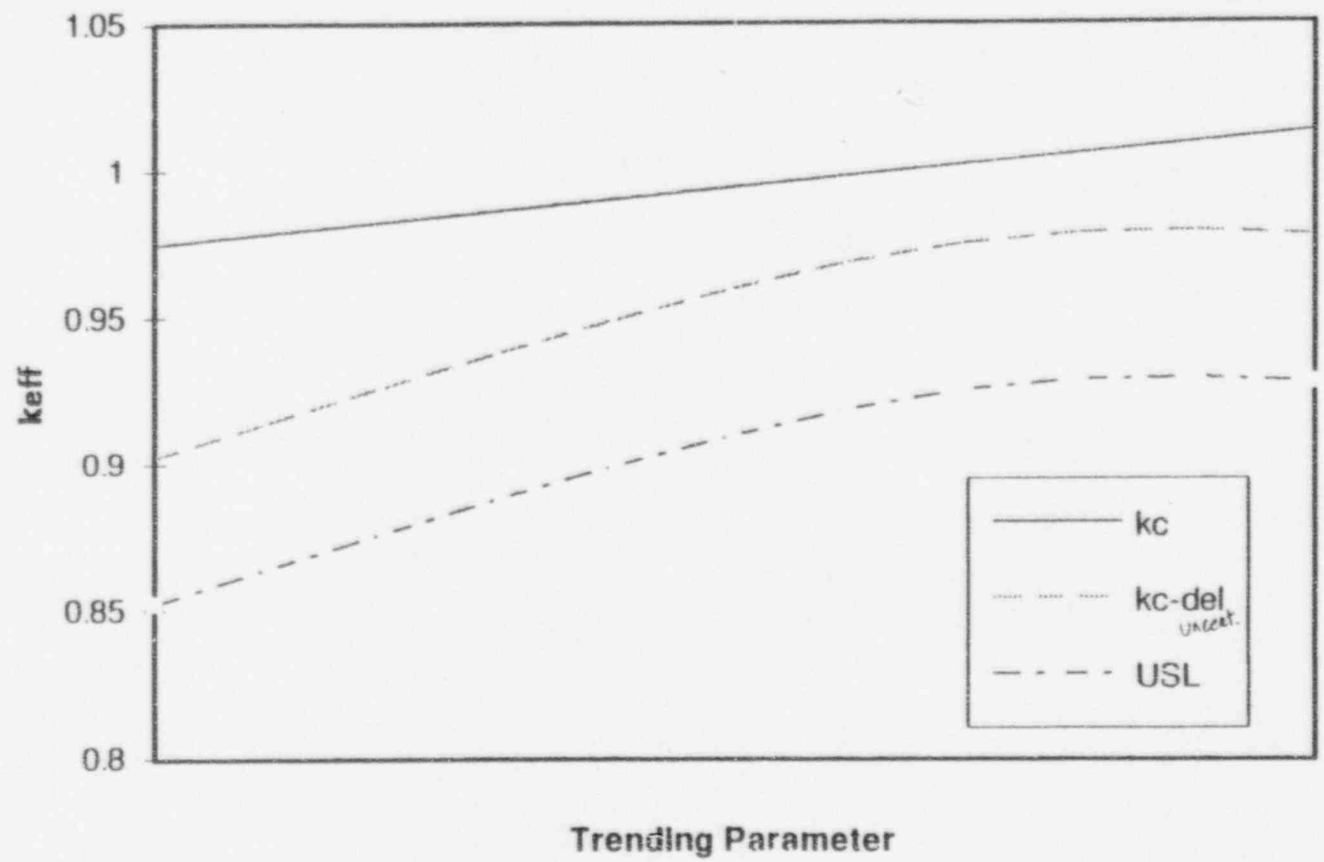


(E) Methodology to Compute Upper Safety Limit

- Upper Safety Limit (USL) is acceptance criterion for the calculated maximum value of k_{eff} (derived from ANSI/ANS 8.17)

$$USL = k_c(ALA) - \Delta k_c(ALA) - \Delta k_m$$

(E) Methodology to Compute Upper Safety Limit



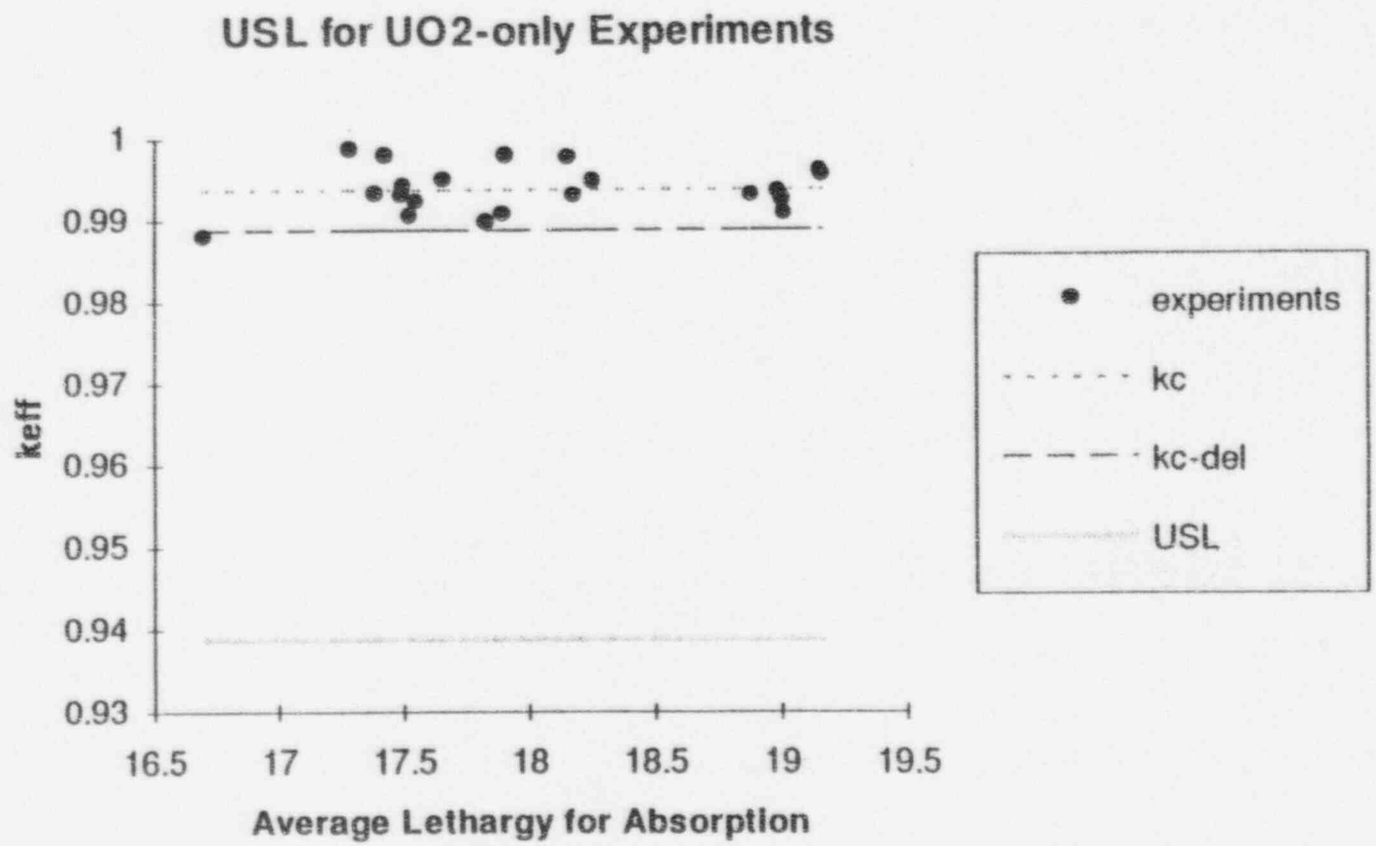
(E) Methodology to Compute Upper Safety Limit

- Perform independently on 21 UO_2, 36 MOX cases
- Test each subset for trend with Average Lethargy for Absorption (ALA)
- Determine USLs
- Conservatively combine

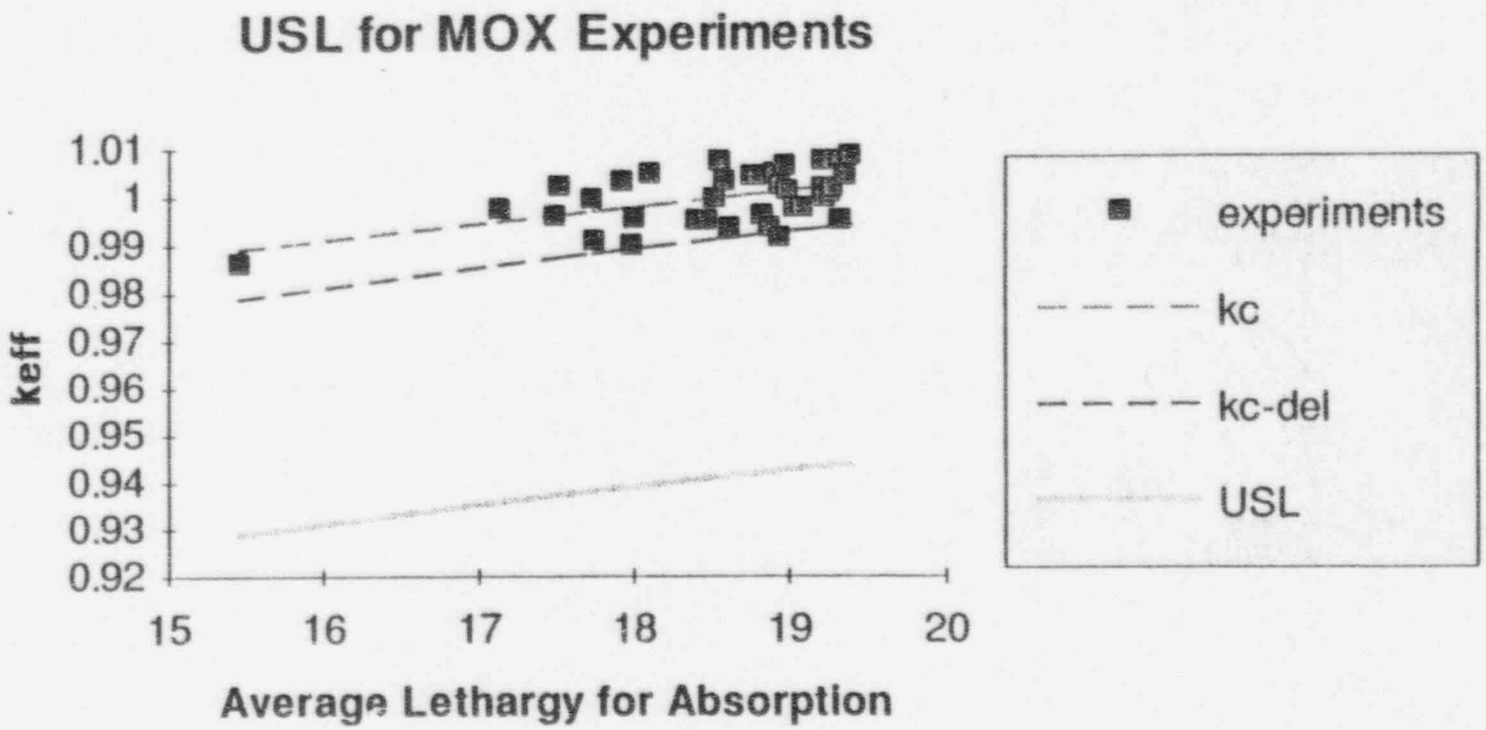
(F) USL- Results with SCALE 4.2

- UO_2-only subset does not have statistically significant trend
- MOX subset shows statistically significant trend

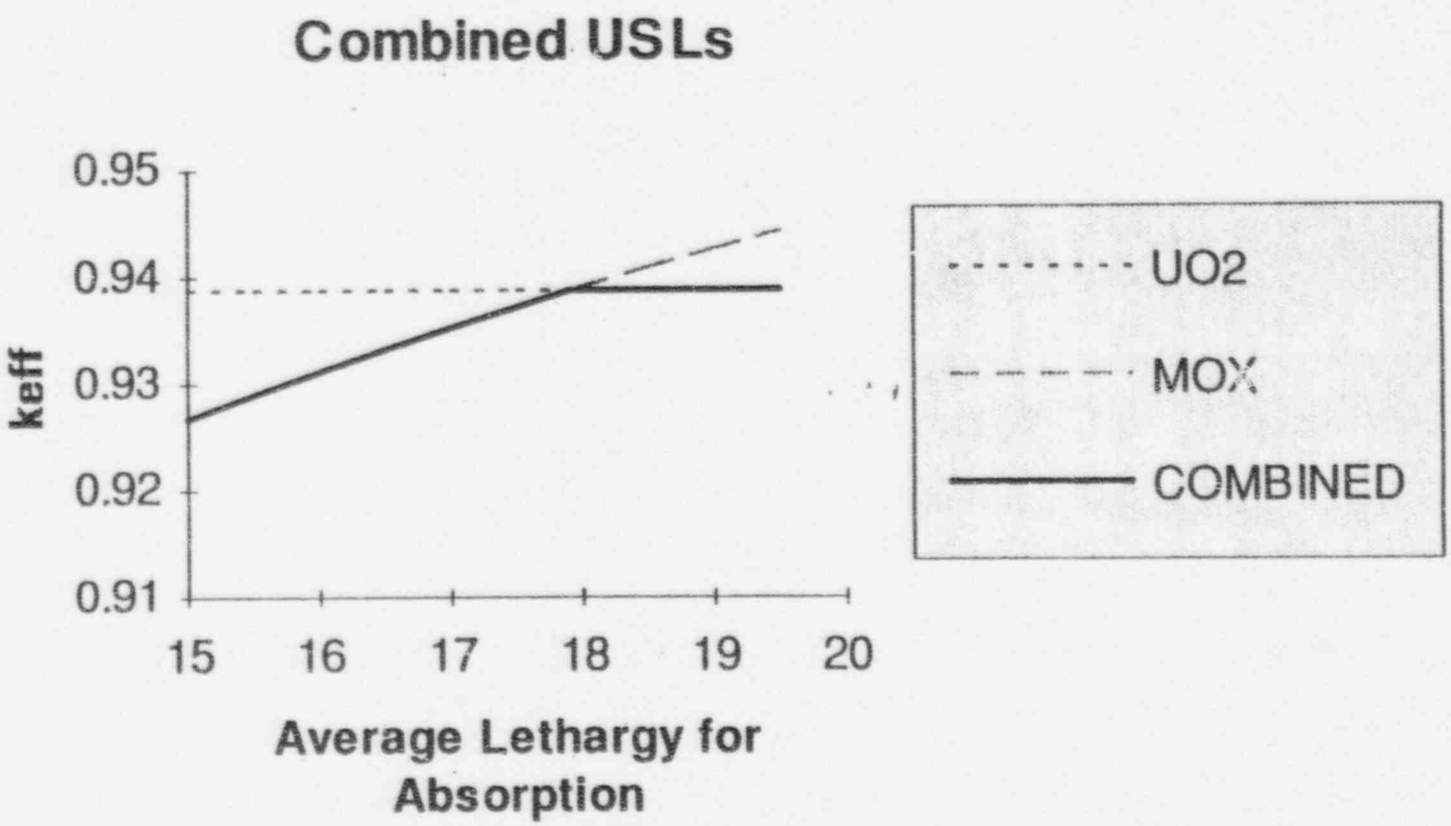
(F) USL- Results with SCALE 4.2



(F) USL- Results with SCALE 4.2



(F) USL- Results with SCALE 4.2



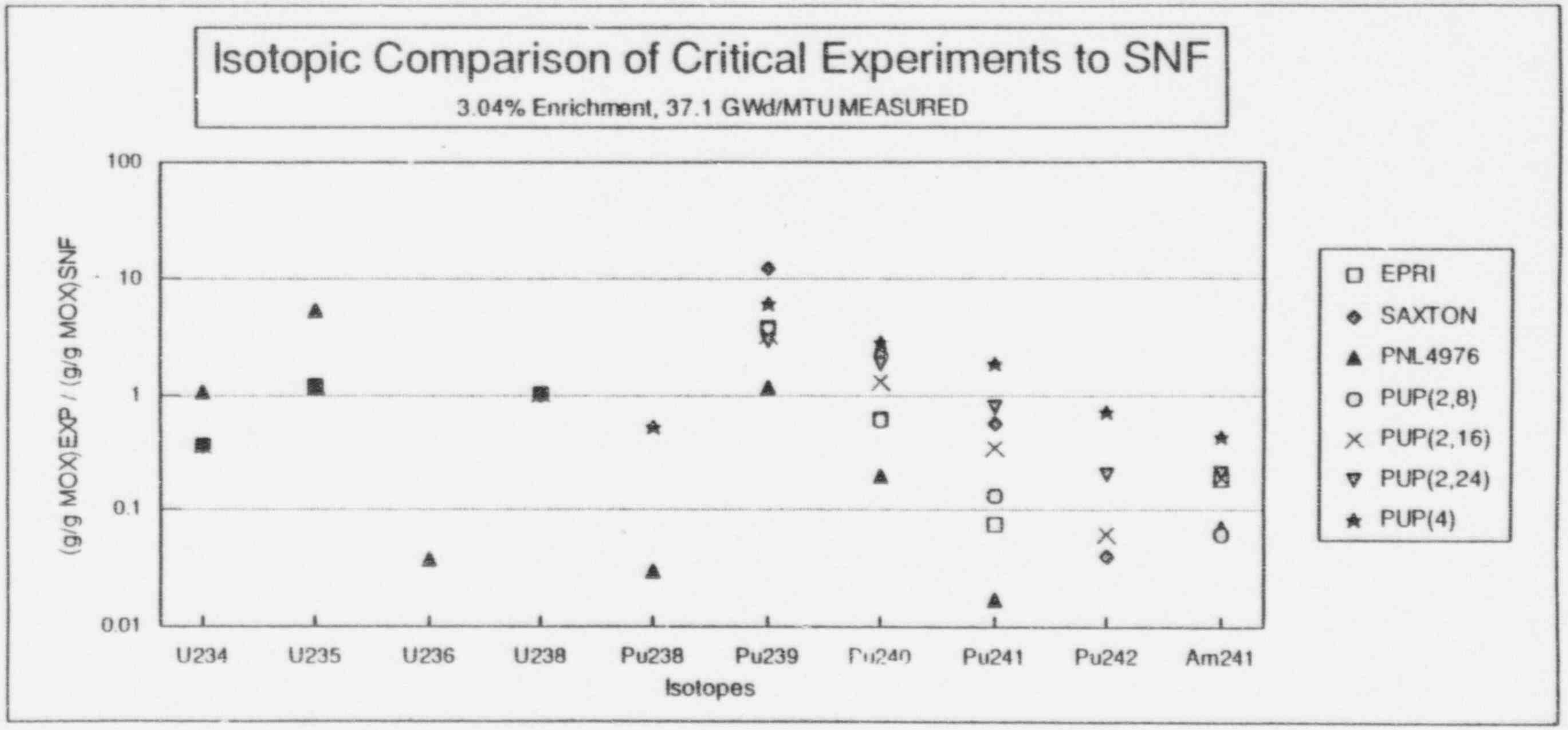
Summary

- Benchmark set of critical experiments is adequate set for criticality validation
- Need to separate UO_2-only and MOX subsets
- Use Average Lethargy for Absorption (ALA) as trending parameter
- Modified statistical method to compute uncertainty and determine Upper Safety Limit (USL)

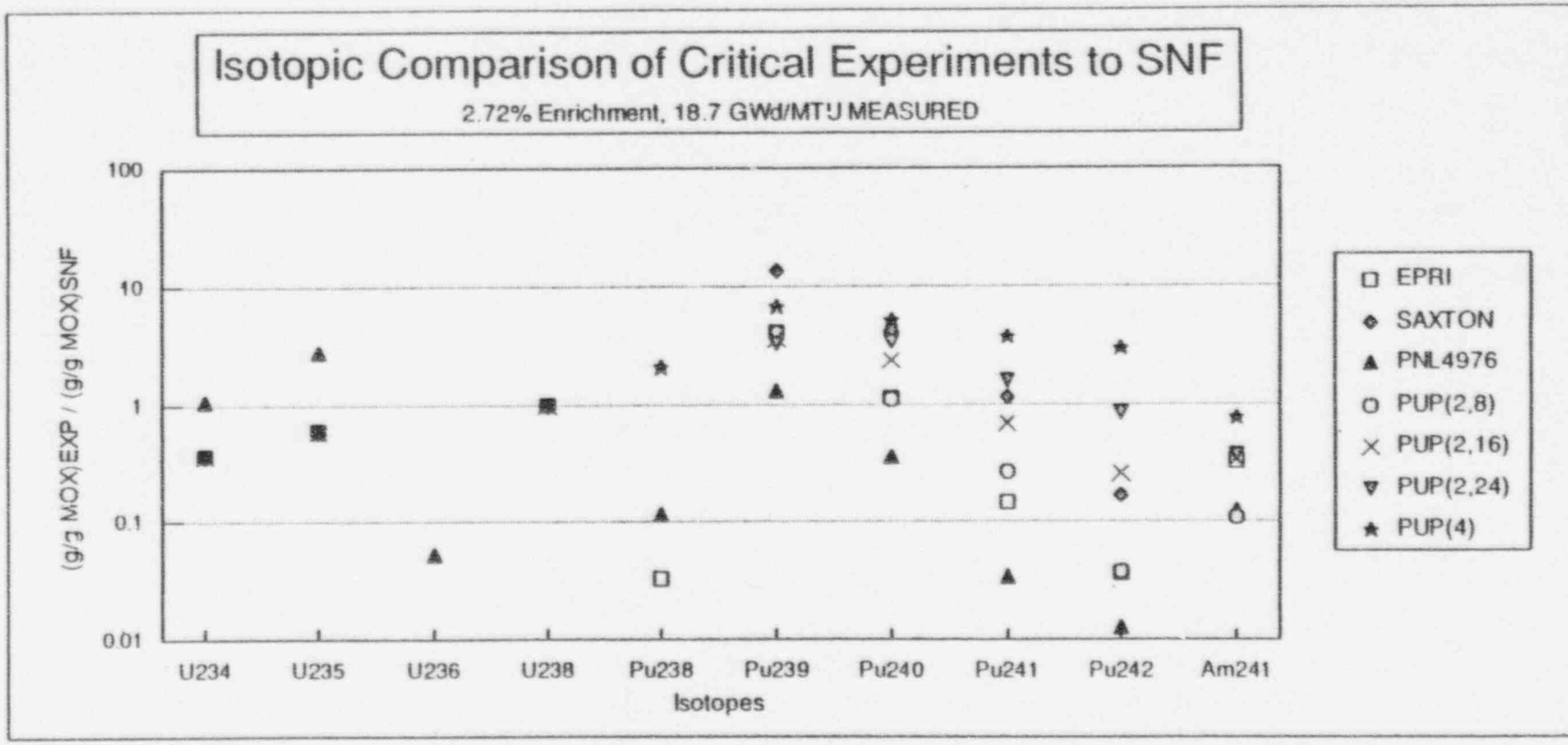
Backup Slides for Criticality Validation for Actinide-Only Burnup Credit (Chapter 3)

Emilio Fuentes
January 23, 1997

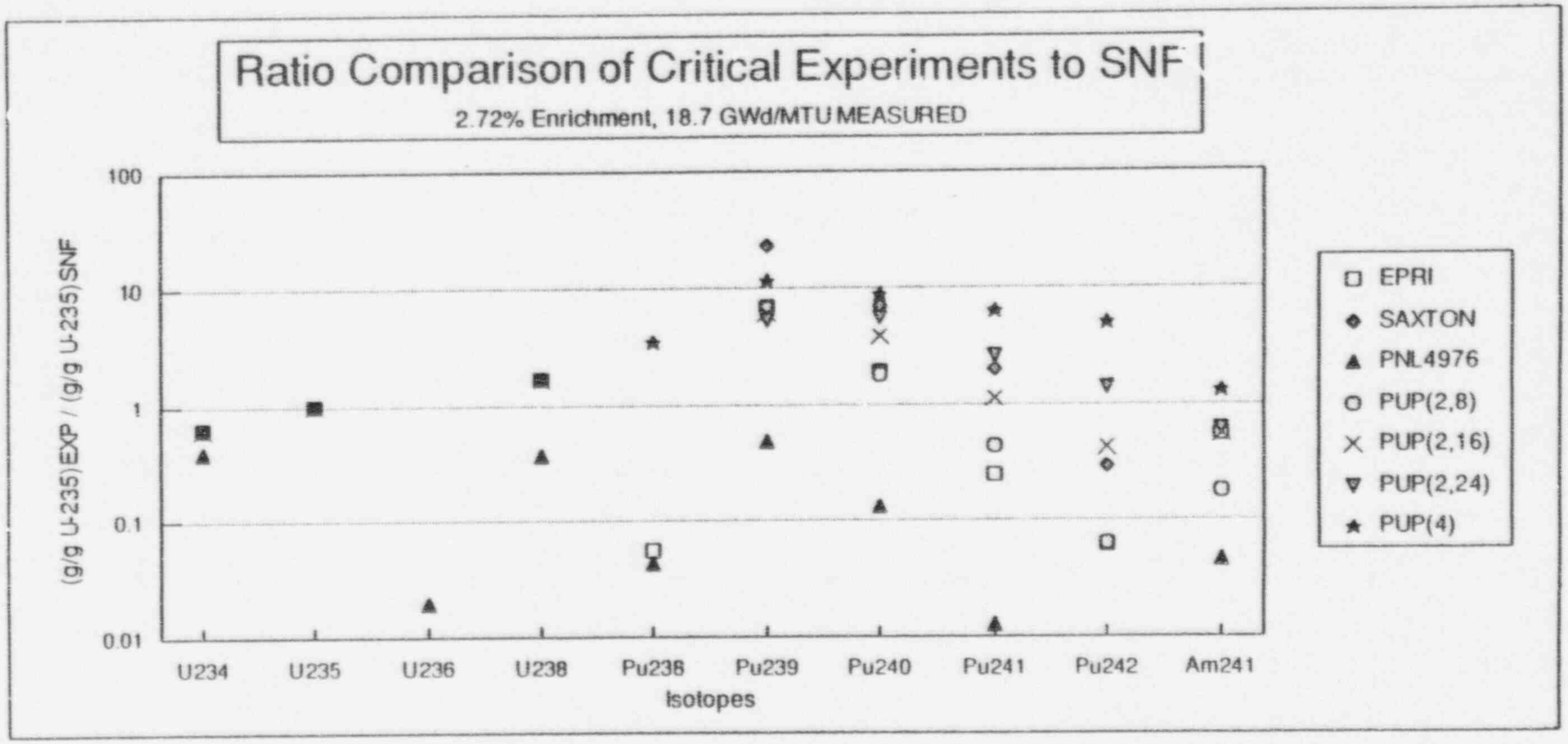
(A) Benchmark Set of Critical Experiments



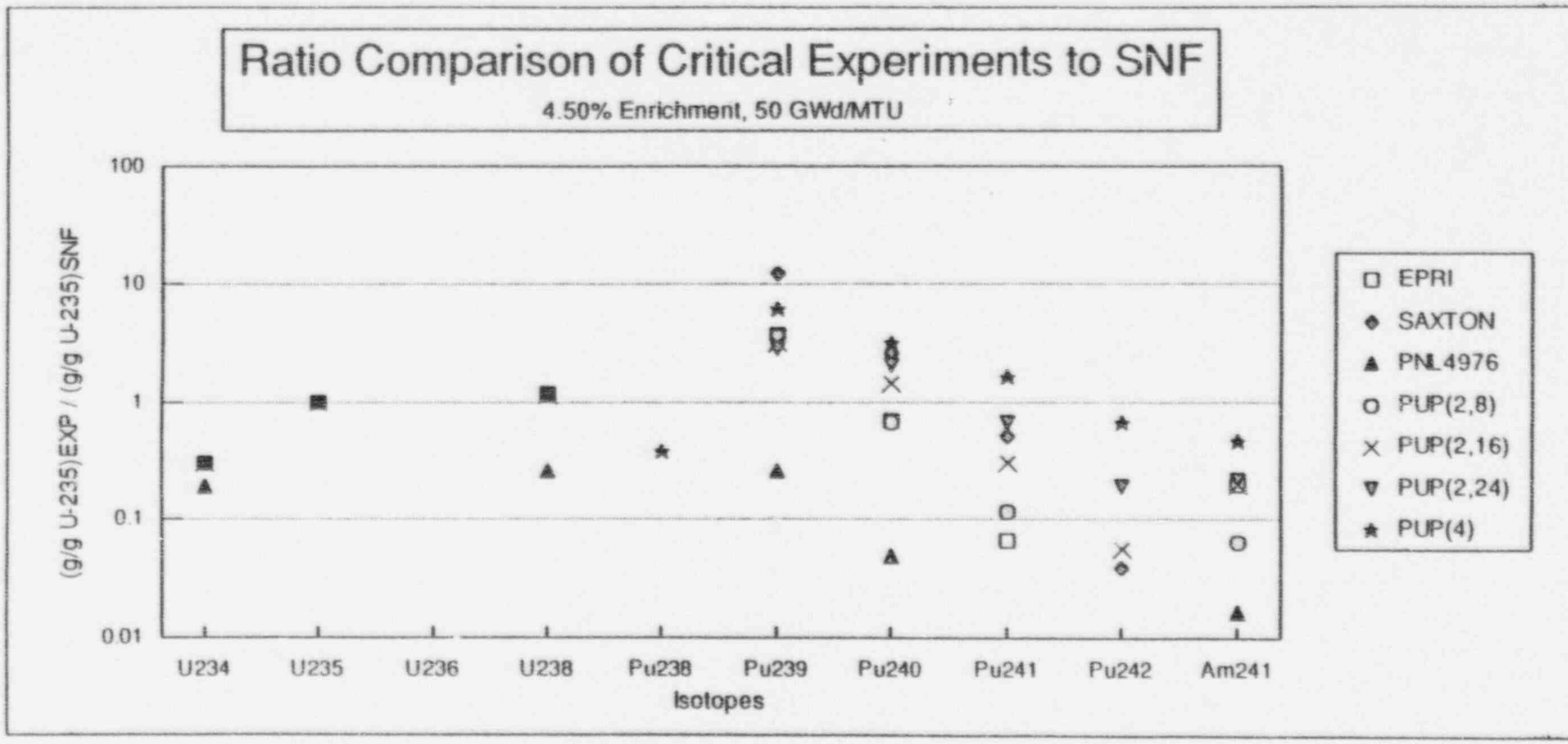
(A) Benchmark Set of Critical Experiments



(A) Benchmark Set of Critical Experiments



(A) Benchmark Set of Critical Experiments

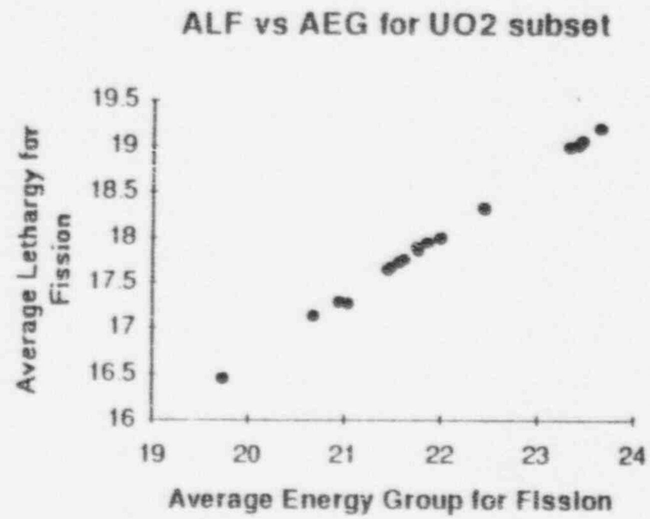


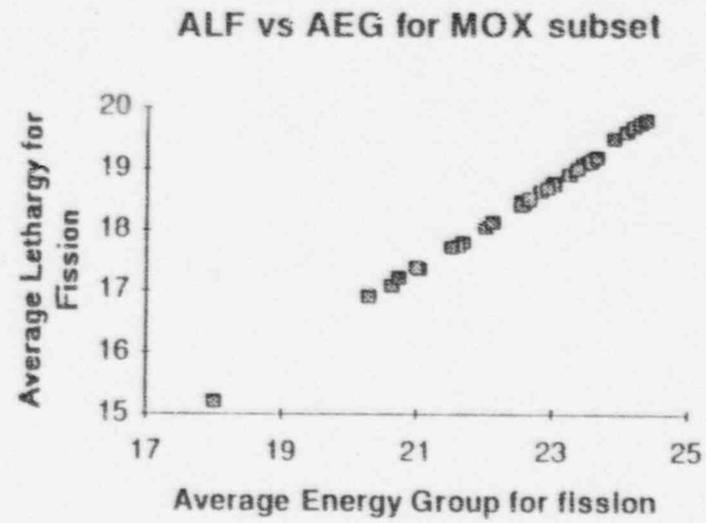
(B) Equality Analysis

$$T = \frac{\overline{Y}_1 - \overline{Y}_2}{\sqrt{\frac{(n-1)S_{y1}^2 + (m-1)S_{y2}^2}{n+m-2}}\sqrt{\frac{1}{n} + \frac{1}{m}}}$$

$$S_{y1}^2 = \frac{\sum (Y_i - \overline{Y}_1)^2}{n-1}$$

(C) Selection of Appropriate Trending Analyses





(C) Selection of Appropriate Trending Analyses

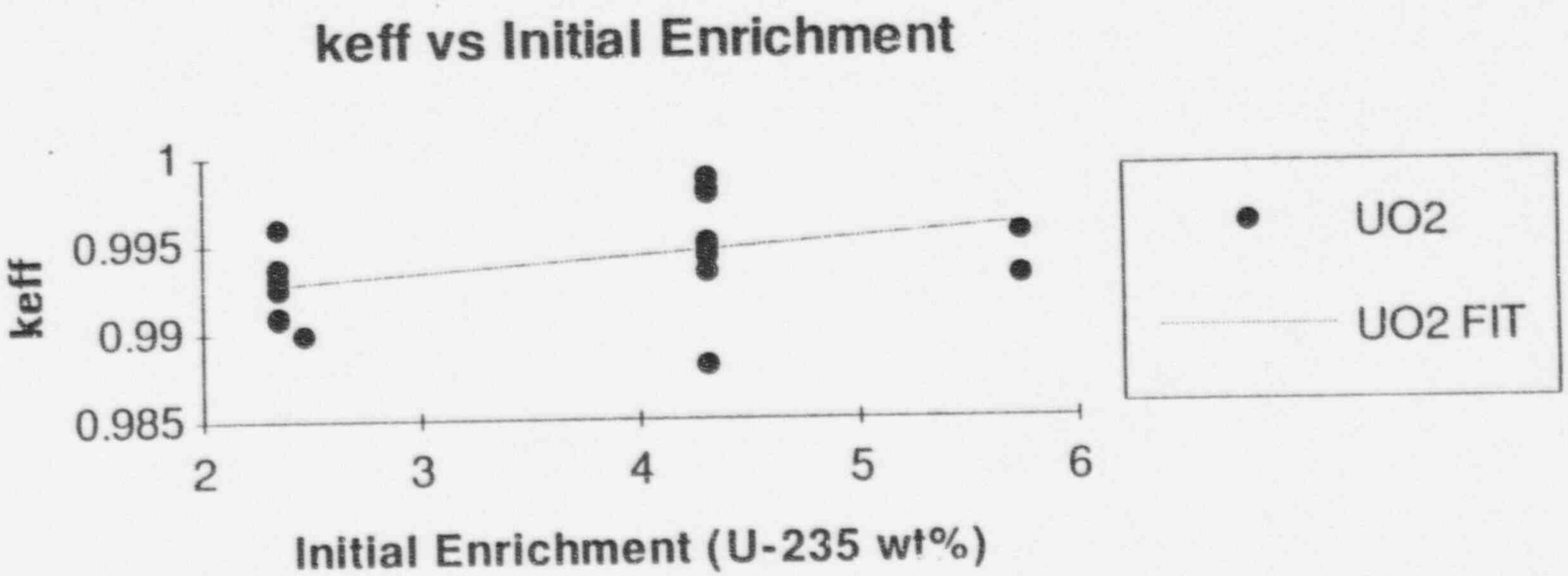
$$y = a + bx$$

$$T_{crit} = b\sqrt{\frac{(n-2)S_{xx}}{SS_R}}$$

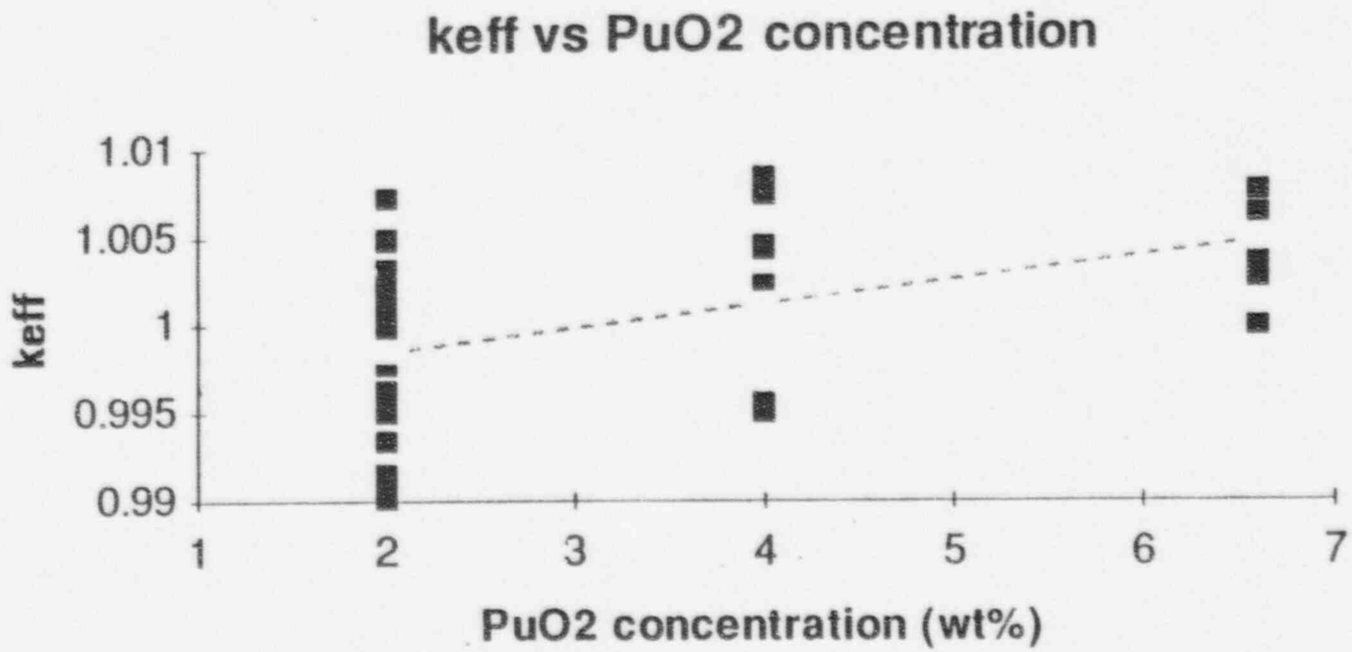
$$S_{xx} = \sum (x_i - \bar{x})^2$$

$$SS_R = \sum (y_i - a - bx_i)^2$$

(C) Selection of Appropriate Trending Analyses



(C) Selection of Appropriate Trending Analyses



(D) Criticality Calculations Uncertainty

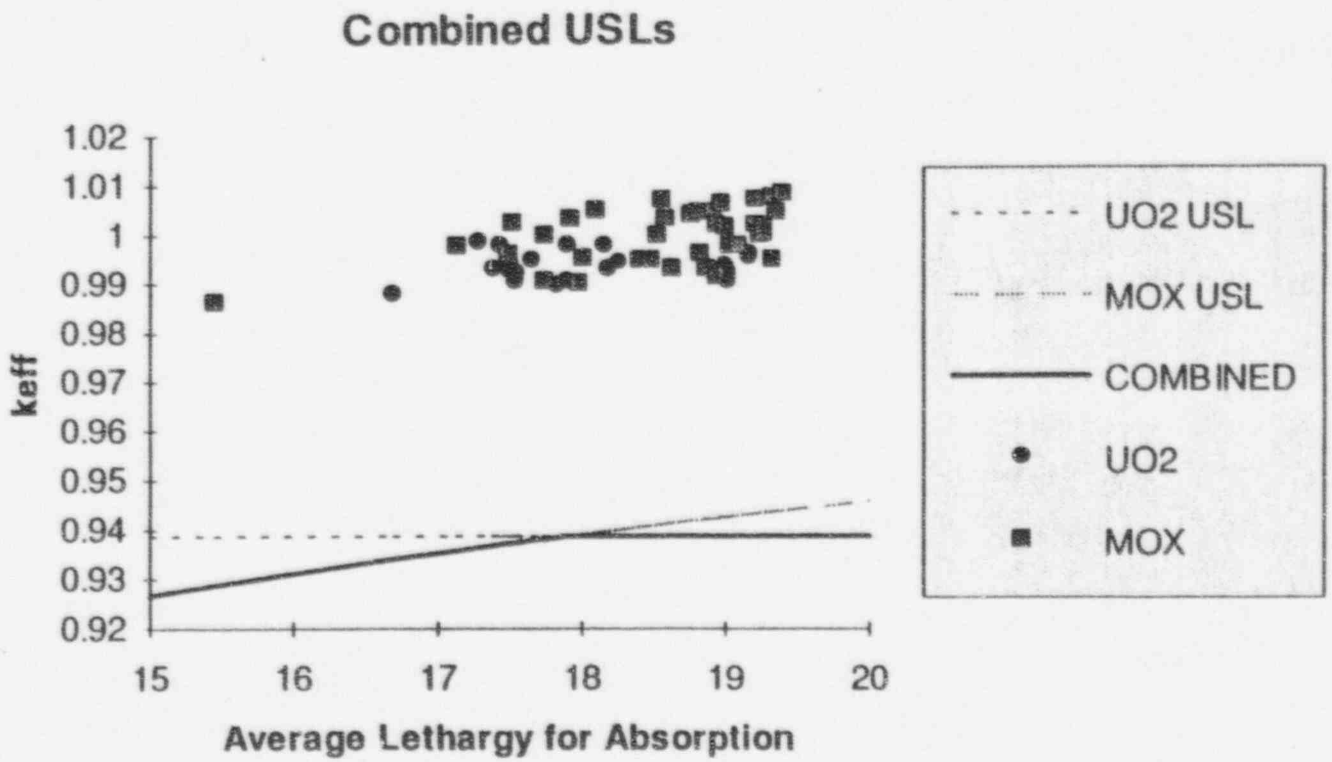
$$S_{xx} = \sum (x_i - \bar{x})^2$$

$$S_p^2 = \frac{SS_R}{n-2} + S_w^2$$

$$SS_R = \sum (y_i - a - bx_i)^2$$

$$S_w^2 = \frac{1}{n} \sum \sigma_i^{\ 2}$$

(F) USL- Results with SCALE 4.2



Civilian Radioactive Waste
Management System

Management & Operating
Contractor

TRW
TRW Environmental Safety
Systems Inc.

Burnup Verification for Actinide-Only Burnup Credit

(Chapters 5 and 6)

Dale Lancaster
January 23, 1997

Introduction

- Philosophy
- Reactor Record Uncertainty
- Measurement System Design Requirements
- Summary

Philosophy

- The Reactor Records on Assembly Average Burnup are of adequate quality and use the most pieces of independent data and hence are the record of choice.
  - Monthly flux maps.
  - Calibrate the reactor power daily.
  - Large errors in records would be detected in power distributions.

Reactor Record Uncertainty

- Define reactor record assembly average burnup:

1. Must use the in-core detector system to establish relative assembly-average power (RelAssm).

2. Establish core power (P) using calibrated instrumentation with known uncertainties (e.g., flowmeters, thermocouples, etc.)

Reactor Record Uncertainty

3. **Determine the nominal Reactor Record Assembly Average burnup ($RRAA_{Burnup}$) from:**

$$RRAA_{Burnup}(MWD/MTU_{assem}) =$$

$$\int_{assem.\ lifetime} [RelAssm(t)*P(t)]dt\ ]/MTU_{assem}$$

or

$$RRAA_{Burnup}(MWD/\ MTU_{assem}) =$$

$$sum_i\ [RelAssm_i(t) * P_i(t) * \Delta t_i]/MTU_{assem}$$

Reactor Record Uncertainty

- The Utilities then establish the reactor record uncertainty associated with each assembly by use of power distribution uncertainty, core power uncertainty, and MTU uncertainty.

- The expected value is around 5%.

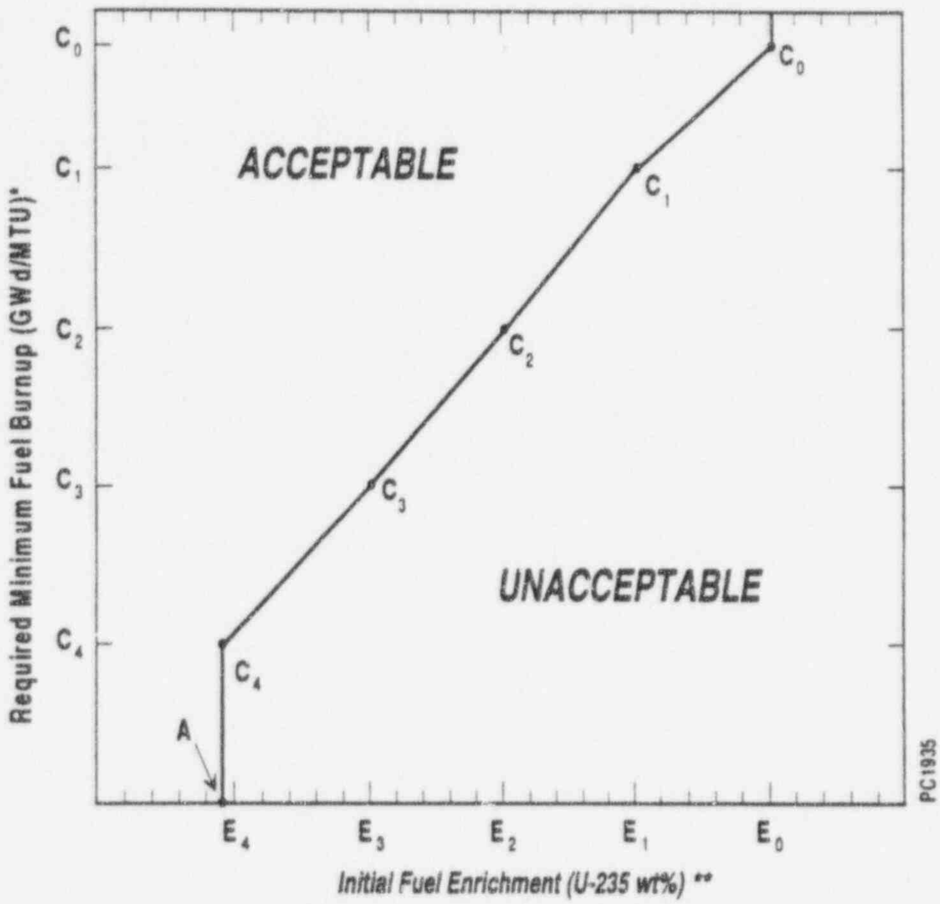
Reactor Record Uncertainty

- The loading curve will use the Minimum Assembly Averaged Burnup ($MinAA_{burnup}$)

- The Minimum Assembly Averaged Burnup must obey the following equation:

$$MinAA_{burnup} < RRAA_{burnup} - \text{Uncertainties}$$

- The loading curve is not reduced by the uncertainties (rather $RRAA_{burnup}$ is reduced).



Assembly Design: ____________

Minimum Cooling Time: ____________

Maximum Number of Removable Burnable Absorber Rods: ____________

Note: This loading curve was generated with the following generic assumptions: Maximum Cycle Average ppm Boron of _____, Maximum Core Outlet Temperature of _____, and the Maximum Pellet Average Temperature of _____

* The nominal burnup must be reduced by the utility so there is a 95% confidence level of meeting the Required Minimum Fuel Burnup.

** If the assembly has more than one enrichment, the highest enrichment must be used.

Figure 5-1. Development of Burnup Credit Loading Curve for a Specific SNF Package

Measurement System Design Philosophy

- The measurement system will be used to confirm the reactor record assembly average burnup.

- The level of confirmation should be consistent with the reactivity margin of cask systems.

Measurement System Design Requirements

- Seven Areas of Requirements for the Measurement System are established:
  - Rejection criterion
  - Measurement Accuracy
  - Correct Horizontal Average
  - Operational Considerations
  - Characteristics of Assemblies to be Measured
  - Analysis Tools
  - Pool Compatibility

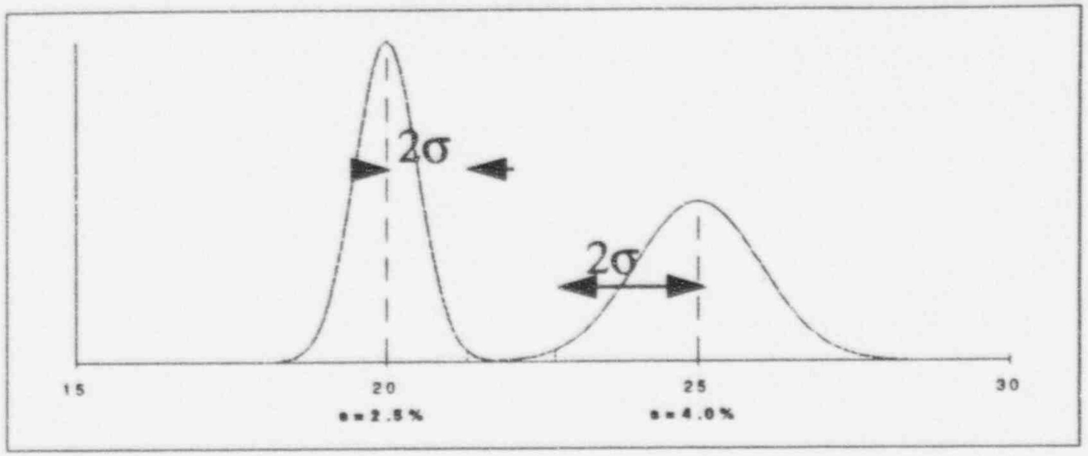
Rejection criterion

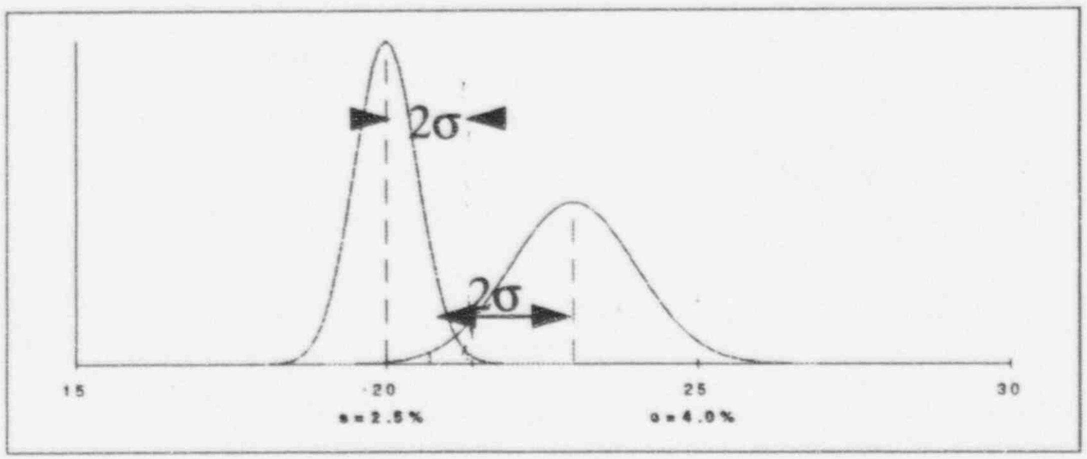
- **An assembly should be rejected if the reactor record and verification measurement are in disagreement.**

- **Disagreement is defined as outside the 95% prediction band of both the reactor records and the measurement system; i.e.:**

$$2^{*}\sigma_{\text{reactor records}} + 2^{*}\sigma_{\text{measurement system}}$$

The Problem Graphically





Rejection criterion

- An assembly will be rejected if its measured burnup deviates from the Reactor Record Assembly Average burnup ($RRAA_{Burnup}$) by greater than 10%.

- The 10% value was selected since the estimated uncertainties in the reactor records and measurement system are each 5%, and

- The reactivity effect of a burnup error of:

  10% - reactor record uncertainty

  is small compared to the conservatism in the method.

Measurement Accuracy - Dependent Systems

- For systems that use the reactor records for calibration, the calibration curve includes the uncertainties in the reactor records as well as the measurement system.

- Using similar statistics as used in chapters 2 and 3, an expected burnup uncertainty, CBU, around the calibration curve can be established. This uncertainty must be less than 10%.

Measurement Accuracy - Dependent Systems

$$CBU = t_{\alpha/2,n-2}\sqrt{\left(\frac{n+1}{n}+\frac{(x-\bar{x})^2}{S_{xx}}\right)\frac{SS_R}{n-2}}$$

Where:

$$S_{xx} \equiv \sum_{i=1}^{n}(x_i - \bar{x})^2$$

$$SS_R \equiv \sum_{i=1}^{n}(y_i - y_{fit})^2$$

Measurement Accuracy - Dependent Systems

- The minimum sample size for the calibration curve is set by the requirement that the burnup uncertainty, CBU, is 10% or less.

- The calibration curve must be segmented to no greater than 10 GWD/MTU segments to account for possible errors in the slope of the calibration curve.

Measurement Accuracy - Independent Systems

- Systems that do not depend on the reactor record burnup must meet the following minimum requirement.

$$\Delta BU > 2*(\sigma_{\text{reactor records}}^2 + \sigma_{\text{measurement system}}^2)^{.5}$$

for

$$\Delta BU = 5\% + \sigma_{\text{reactor records}}$$

Correct Horizontal Average

- **The measurement system must account for possible radial burnup tilts across assemblies by measuring at least two opposing sides of each assembly.**

Operational Considerations

- The measurement must account for the following:
  - Detector electronic drift
  - Detector positioning
  - Pool boron concentration, temperature, and water purity
  - Fuel characteristics differences
  - Counting time

Characteristics of Assemblies to be Measured

- **The range of parameters for which the measurement system is qualified must be clearly stated. These include:**
  - Burnup Range
  - Initial Enrichment Range
  - Cooldown Range
  - Nominal Dimensions
  - Assembly Design Type

Analysis Tools And Pool Compatibility

- The analysis tools used to convert the signals to measured burnup must be benchmarked and qualified.

- The process and equipment used for the verification measurement should be compatible with normal operations in spent fuel pools.

Summary

- The burnup is determined from the Reactor Records. The measurement is done to confirm the Reactor Records.
- Determination of the uncertainties for reactor records and detector will be the responsibility of those that are in the best position to do so.
  - Utilities will determine reactor record uncertainties
  - Detector vendors will determine their measurement uncertainties
- Vendors and utilities have been contacted and agree to this position.

Summary

- Specifications have been established to clarify the requirements of the utilities and measurement vendors.

- These specifications have been tested on current measurement systems and are achievable.