

January 4, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of)
) Docket No. 50-400-LA
CAROLINA POWER & LIGHT)
COMPANY) ASLBP No. 99-762-02-LA
)
(Shearon Harris Nuclear Power Plant))
)

AFFIDAVIT OF ANTHONY P. ULSES IN SUPPORT OF
NRC STAFF BRIEF AND SUMMARY OF RELEVANT FACTS, DATA
AND ARGUMENTS UPON WHICH THE STAFF PROPOSES TO RELY
AT ORAL ARGUMENT ON TECHNICAL CONTENTION 2

Anthony P. Ulses, being duly sworn, does hereby state as follows:

1. I have been employed at the Nuclear Regulatory Commission since 1992. My current position is Nuclear Engineer in the Reactor Systems Branch, Division of Systems Safety and Analysis, Office of Nuclear Reactor Regulation. My responsibilities include performing reactor physics calculations. I have used the SCALE package of physics codes to perform reactor physics simulations to support staff safety evaluations since early 1997. I have a Masters Degree in Nuclear Engineering from the University of Maryland and a Bachelors of Science and Engineering degree in Nuclear Engineering from the University of Michigan. A statement of my professional qualifications is attached hereto (Exhibit 1).

2. The purpose of this testimony is to address the Atomic Safety and Licensing Board (Board) concerning the Board of Commissioners of Orange County 's (BCOC) Contention 2, Basis 2, as set forth in Orange County's Supplemental Petition to Intervene and

in the Atomic Safety and Licensing Board Memorandum and Order of July 12, 1999 (LBP-99-25).

3. In preparation for this testimony, I reviewed fuel and storage rack design information and criticality calculation results in the CP&L application for the proposed license amendment as well as the correspondence and technical documents identified below.

4. BCOC's Contention 2, as admitted by the Board, states :

Storage of pressurized water reactor ("PWR") spent fuel in pools C and D at the Harris plant, in the manner proposed in Carolina Power and Light's ("CP&L") license amendment application, would violate Criterion 62 of the General Design Criteria ("GDC") set forth in Part 50, Appendix A. GDC 62 requires that: "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations." In violation of GDC 62, CP&L proposes to prevent criticality of PWR fuel in pools C and D by employing administrative measures which limit the combination of burnup and enrichment for PWR fuel assemblies that are placed in those pools. This proposed reliance on administrative measures rather than physical systems or processes is inconsistent with GDC 62.

The Board utilized a two basis construct in admitting the contention, as follows:

Basis 1 -- CP&L's proposed use of credit for burnup to prevent criticality in pools C and D is unlawful because GDC 62 prohibits the use of administrative measures, and the use of credit for burnup is an administrative measure.

Basis 2 -- The use of credit for burnup is proscribed because Regulatory Guide 1.13 requires that criticality not occur without two independent failures, and one failure, misplacement of a fuel assembly, could cause a criticality if credit for burnup is used.

5. The purpose of this affidavit is to address Basis 2.¹

6. I have performed analyses to assess the possibility of unlikely misloading events leading to a criticality accident (Memorandum from Tony P. Ulses to Ralph Caruso, "Completion of Criticality Assessment of Misloading Error in Harris C and D Spent Fuel Pool," USNRC, November 2, 1999)(Exhibit 2). I assumed that the soluble boron concentration was 2000 ppm to be consistent with the nominal boron concentration present in the Shearon Harris spent fuel pool. I also assumed that the water at 4 degrees Celsius (40 degrees Fahrenheit) to be consistent with the calculations performed by the licensee contractor. I modeled the rack, fuel, and poison plate geometry using their nominal dimensions. I assumed that the worst conceivable misloading would involve a Westinghouse 15x15 assembly enriched to 5 weight percent U-235 without burnable poisons. This represents the highest allowed enrichment for commercial power reactor fuel and is, therefore, bounding. The use of the Westinghouse 15x15 assembly as the limiting assembly is based upon work performed by the licensee contractor (Letter from J. Scaraola (CP&L) to U.S. Nuclear Regulatory Commission, "Shearon Harris Nuclear Power Plant, Docket No. 50-400/License No. NPF-63, Request for License Amendment Spent Fuel Storage," Enclosure 6 (proprietary version), Section 4.0, Criticality Safety Evaluation, December 23, 1999)(Exhibit 3).²

¹ For a definition of criticality and related concepts, see Affidavit of Laurence I. Kopp.

² Exhibit 3 is proprietary and is identical to Exhibit 21 to the Affidavit of Laurence I. Kopp. It has been provided under separate cover to the Board and parties.

7. This analyses used validated 3-dimensional Monte Carlo methods from SCALE, which is a modular code system for performing standardized computer analyses for licensing evaluations. I assumed that the storage racks were filled entirely with misloaded assemblies. This effectively represents an infinite number of assemblies because reflective boundary conditions were used in the x and y plane. I consider that such misloading could result only from multiple unlikely events, requiring multiple errors. The double contingency principle ("Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983, American Nuclear Society, 1983)(Exhibit 4), refers to only two unlikely events. The maximum k_{inf} predicted for this configuration is 0.98. K_{inf} is the infinite multiplication factor which refers to the neutron multiplication of an infinite system and is always less than k_{eff} because it neglects system leakage. Since this is less than 1.0 (1.0 is a critical system) and the configuration studied represents the worst possible series of misloading events, I have concluded that the misloading of an entire rack of fresh fuel in spent fuel pools C or D, which would require multiple concurrent unlikely events, will not lead to a criticality. This bounds the situation where there is only one misloading event because an entire misloaded rack has a much higher reactivity than a rack misloaded with only one assembly.

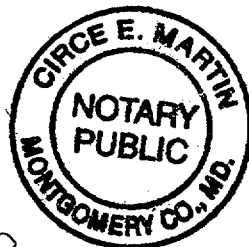
8. The exhibits attached hereto are true and correct copies of the documents relied upon in this affidavit.

9. I hereby certify that the foregoing is true and correct to the best of my knowledge, information and belief.

Anthony P. Ulses
Anthony P. Ulses

Subscribed and sworn to before me
this 4 day of January 2000

Circe E. Martin
Notary Public



My commission expires: March 1, 2003

Exhibits

1. Statement of Professional Qualifications of Anthony P. Ulses.
2. Memorandum from Tony P. Ulses to Ralph Caruso, "Completion of Criticality Assessment of Misloading Error in Harris C and D Spent Fuel Pool," USNRC, November 2, 1999.
3. Letter from J. Scaraola (CP&L) to U.S. Nuclear Regulatory Commission, "Shearon Harris Nuclear Power Plant, Docket No. 50-400/License No. NPF-63, Request for License Amendment Spent Fuel Storage," Enclosure 6, Section 4.0, Criticality Safety Evaluation, December 23, 1999. [PROPRIETARY - provided under separate cover to the Board and the parties. (Identical to Exhibit 21 to Affidavit of Laurence I. Kopp.)]
4. "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983, American Nuclear Society, 1983.

Anthony P. Uises

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Education

Master's of Science (MS) in Nuclear Engineering
University of Maryland at College Park, May 1999.

Bachelor's of Science and Engineering (BSE) in Nuclear Engineering, *Cum laude*
University of Michigan at Ann Arbor, August 1992.

Experience

United States Nuclear Regulatory Commission 5/93-present
Nuclear Engineer, Reactor Systems Branch, Division of Systems Safety and Analysis

United States Nuclear Regulatory Commission 9/92-5/93
Graduate Fellow, Advanced Reactors Project Directorate

Computer Code Development

- Maintained and upgraded legacy physics codes on UNIX workstations
- Developed VIKTORIA code for fuel channel analysis
- Coupled TRAC and NESTLE codes
- Assisted in NEWT development

Computer Code Analysis

- SCALE Fuel lattice criticality studies, depletion and collapsed cross section preparation
- MCBEND Reactor pressure vessel fluence studies
- NEWT Fuel power distributions, depletion and collapsed cross sections
- DOORS 3.2 3D Transport Calculations
- TRAC/NESTLE 3D BWR ATWS Studies
- DRAGON 3.2 Fuel power distributions, depletion and collapsed cross sections

Computer Codes

- DANTSYS 3.1, MONK

Computer Languages and Operating Systems

- UNIX, Fortran 90/95, Fortran 77, C, Windows, DOS, PVM

Regulatory Experience:

- License amendment evaluations
- Fuel manufacturer inspections
- Performing Audit Calculations of Licensee Submittals

General Experience:

- Managing High Performance Computer Networks
- Digital UNIX System's Administration

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UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

November 5, 1999

MEMORANDUM TO: Ralph Caruso, Chief
BWR Reactor Systems and Nuclear Performance Section
Reactor Systems Branch
Division of Systems Safety and Analysis

FROM: Tony P. Uises, Nuclear Engineer *TU*
BWR Reactor Systems and Nuclear Performance Section
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: COMPLETION OF CRITICALITY ASSESSMENT OF MISLOADING
ERROR IN HARRIS C AND D SPENT FUEL POOL

I have completed the analysis evaluating the potential for criticality from a misloading error if Shearon Harris begins to use high density storage racks in the currently inactive C and D spent fuel pools. The analysis discussed in the enclosed report assumes a worst case misloading error in which the entire rack is misloaded with fresh 5 w/o enriched Westinghouse 15x15 fuel which has been previously determined to be the most reactive PWR fuel type which could be loaded into the Harris pools. This analysis demonstrates that the multiplication factor will remain less than one (i.e. subcritical) for this postulated worst case scenario. The calculated eigenvalues are taken at upper 95/95 level and a manufacturing uncertainty of 1 percent has been added to the predicted value.

Enclosures:
As stated

**Evaluation of Postulated Worst Case Misloading Error for
Harris C and D Spent Fuel Pools**

**Tony P. Ulses
November 2, 1999**

1 Introduction

Carolina Power and Light (CP&L), the operator of the Shearon Harris nuclear power plant, requested a license amendment to activate the two unused spent fuel pools at the Harris site. The proposal is to use a "high density" storage configuration which requires the use of burnup credit racks. In the context of this report burnup credit racks refer to storage racks which require that the fuel has reached a pre-specified minimum burnup before it can be safely stored. The need for this burnup requirement is dictated by the fact that the inter-assembly spacing is reduced to achieve the desired "high density" configuration. Whenever one relies on a physical process such as burnup one needs to assess the impact of an assembly being inserted into the rack that has not reached the minimum acceptable burnup. Therefore, criticality analyses have been performed to assess the effect of an assembly misloading error in the Harris "C" or "D" spent fuel pool. In this analyses it was assumed that the entire rack was misloaded with UO_2 fuel enriched to 5 w/o U^{235} which is the highest enrichment allowed at commercial power plant's in the US. This would be the worst possible configuration.

2 Definition of Problem

In this analyses we will assess the impact of a worst case misloading accident by predicting the multiplication factor of the system. To this end, we will perform three base analyses and one sensitivity calculation. Two of the base analyses are intended to assess the staff's criticality calculations against the licensee calculations and the final analyses will assess the worst case misloading accident. The two comparative calculations are important because they will allow an assessment of the licensee method's and will serve to strengthen the staff's position with respect to these methods. A brief description of the problems will follow:

Typical Parameters

Fuel type:	Westinghouse 15x15 Assembly Enriched to 5 w/o U^{235}
Rack type:	Holtec High Density
Boundary Conditions:	Reflective in x, y, and z
# of Histories:	1000 groups of 3000 particles for a total of 3 million histories

Problem 1

This problem is extracted from reference 1. The rack should be assumed to be loaded with fresh fuel without soluble boron. All dimensions should be nominal.

Problem 2

This problem is the licensing basis for the storage racks. The rack should be loaded with fuel burned to 41.7 Mw_g/KgU . The depletion is to be performed assuming three cycles of operation with an average boron concentration of 900 ppm, a specific power of 42 kW/KgU , nominal fuel and clad temperature and slightly higher than expected moderator temperature. The criticality analyses should assume no soluble boron is present and credit will be taken for actinides and fission products. All dimensions should be nominal.

Problem 3

This problem assesses the effect of the worst case misloading accident. The rack should be loaded with fresh fuel and one should assume that the soluble boron is present. All dimensions should be nominal.

3 Description of Methods

The SCALE (ref. 2) system was chosen for both the criticality analyses and the burnup calculations. The SCALE system has been extensively assessed and validated for these types of calculations (refs. 3 - 5). The SAS2H sequence was used for the depletion calculations and the CSAS6 sequence was used for the criticality calculations. Both of these sequences use BONAMI and NITAWL-II to process cross sections into a problem specific AMPX working library. SAS2H uses XSDRN and ORIGEN to deplete the fuel and CSAS6 uses KENO-VI for criticality calculations. Both the 44 group and the 238 group ENDF/B-V based AMPX libraries were used in the criticality analyses and the 44 group AMPX library was used for depletion.

4 Presentation and Discussion of Results

The results for problems 1 and 2 are presented in table 1. For comparative purposes, we have included the results from the licensee's contractor (ref. 1). This comparison reveals that the licensee method seems to predict slightly higher multiplication factors (as much as 2% overall). However, given the differences in the methods the staff considers this to be excellent agreement and this gives us a great deal of confidence in the methods being used by both the staff and the licensee.

Table 1 Comparison of Results for Problem 1 and Problem 2

	CASMO	MCNP	SCALE ¹
Problem 1	1.2076	1.2056	1.19378
Problem 2	0.9126	N/A	0.8940

¹The SCALE results are the staff calculation.

The multiplication factor predicted for problem 3 is 0.978 at the upper 95/95 interval using the 44 group library and 0.979 using the 238 group library. The 238 group library was also used for this problem to ensure that collapsing spectrum used to generate the 44 group library from the 238 group library did not introduce any significant bias into the results. This demonstrates that even assuming the worst case misloading error (i.e. misloading an entire rack with fresh fuel) the rack will remain subcritical when one considers the soluble boron which will be present in the pool.

In order to assess the adequacy of multiplication factors predicted using Monte Carlo methods it is prudent to consider, in addition to the number of histories tracked, how well the spatial and energy domains of the problem were sampled. To this end, we have attached the spectrum

output for the global unit from KENO-VI in Appendix A and prepared several spectral plots. The information from the major edit indicates that all of the parts of the problem have been sampled. Note that the flux for region 1 in the global unit is zero because region 1 represents the hole containing the fuel which was inserted into the global unit. The flux should be zero in the global unit for this region.

The spectral plots are presented as Figures 1 and 2. The error bars represent one standard deviation and were extracted from the major edit (see Appendix A). From these plots we can ascertain that there are no unexpected trends in the results. For example, figure 1 shows a characteristic light water moderated reactor spectrum, but the thermal peak is smaller than it would be in the reactor. This reduction is caused by the additional absorption in the rack poison. Furthermore, we can see that we had complete coverage of the energy domain and that the sampling was significant enough to reduce the standard deviation to acceptable values.

5 Conclusions

Analyses have been performed to assess the effect of the worst case misloading scenario in the Harris "C" and "D" spent fuel pool. This analysis demonstrates that the maximum possible multiplication factor in the "C" and "D" spent fuel pools is 0.98 assuming that one credits the soluble boron present in the pool coolant. It should be noted that this analysis does not consider manufacturing tolerances, but the multiplication factor bias from manufacturing uncertainties is typically not larger than 1%. The staff has also been able to confirm that the methods used by the licensee contractor yield results that are consistent with the staff's results.

6 References

1. "Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools C and D," HI-971760, Holtec International, May 26, 1998. (Holtec International Proprietary)
2. "SCALE 4.3, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation," NUREG/CR-0200, Oak Ridge National Laboratory, 1995.
3. M.D. DeHart and S.M. Bowman, "Validation of the SCALE Broad Structure 44-Group ENDF/B-V Cross Section Library for use in Criticality Safety Analysis," NUREG/CR-6102, Oak Ridge National Laboratory, 1994.
4. O.W. Hermann, et. al., "Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, Oak Ridge National Laboratory, March 1995.
5. W. C. Jordan, et. al., "Validation of KENO.V.a, Comparison with Critical Experiments" ORNL/CSD/TM-238, Oak Ridge National Laboratory, 1986.

Spectrum of W 15x15 Fuel in Poisoned Rack

KENO-VI Results

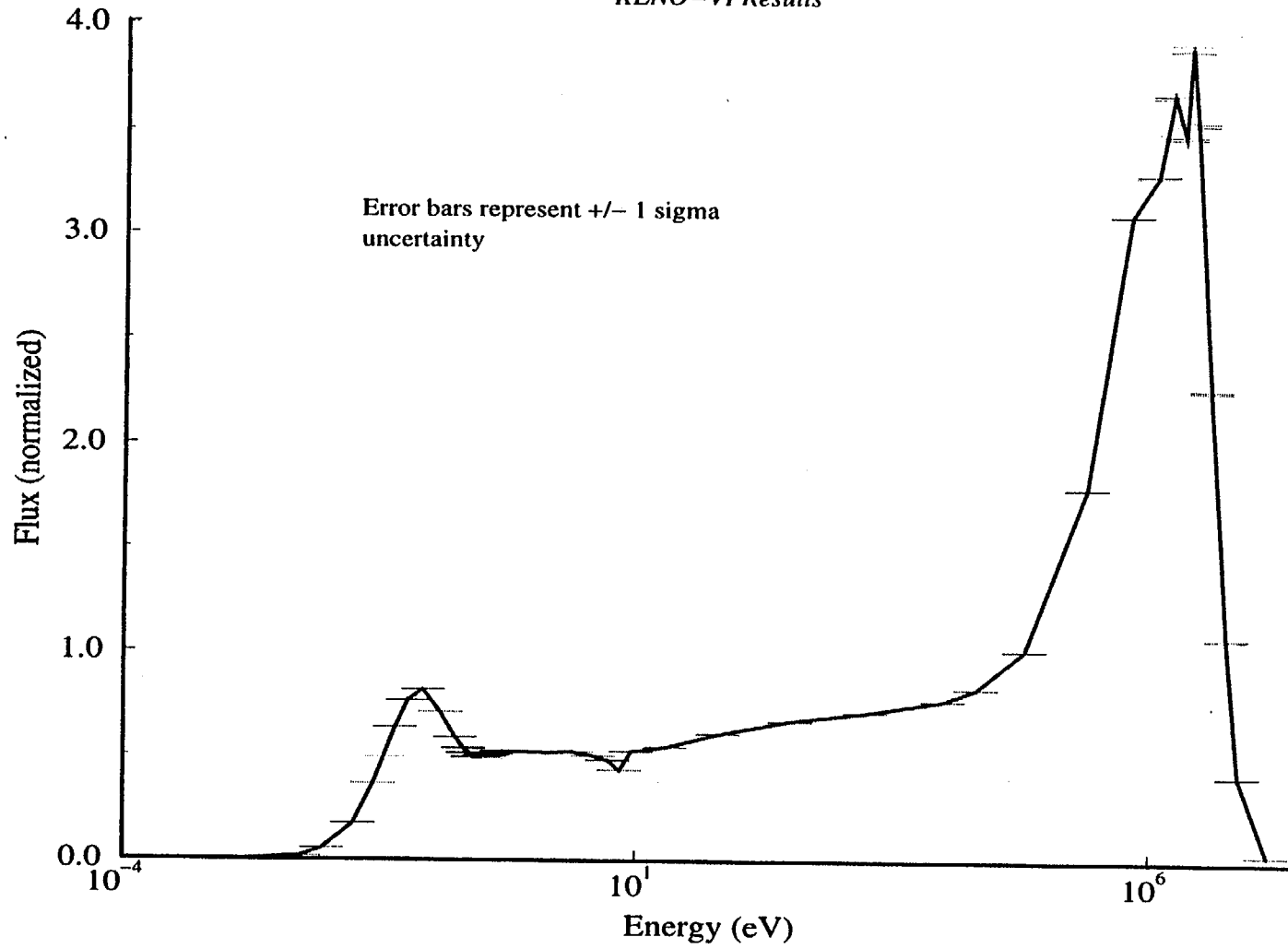


Figure 1 KENO Predicted Spectrum for W 15x15 Fuel Assembly

Spectrum in Outer Boral Sheeting

KENO VI Results

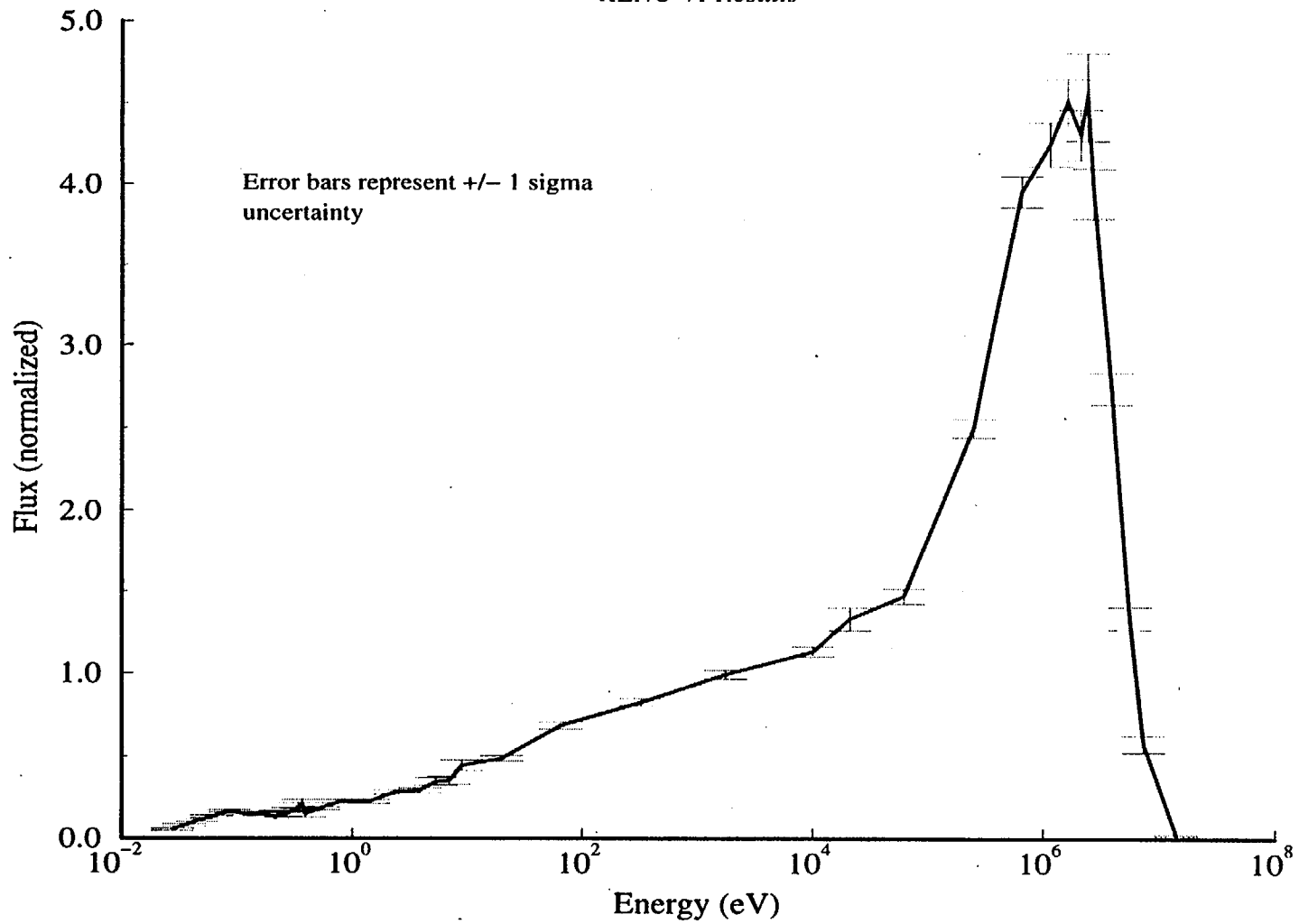


Figure 2 KENO Predicted Spectrum in Outer Boral Sheeting

Appendix A

Excerpt from KENO-VI Major Edit

1

keno-vi input for storage cell calc. for holtec rack w/ 15x15 w

Ofluxes for global unit

Ogroup	region 1		region 2		region 3		region 4		region 5		region 6	
	flux	percent deviation	flux	percent deviation	flux	percent deviation	flux	percent deviation	flux	percent deviation	flux	percent deviation
1	0.000E+00	0.00	1.376E-04	5.36	8.973E-06	18.11	1.932E-05	19.59	1.823E-05	18.58	2.150E-05	12.41
2	0.000E+00	0.00	4.190E-04	3.46	5.856E-05	8.68	5.653E-05	8.31	4.404E-05	9.44	4.438E-05	8.41
3	0.000E+00	0.00	1.267E-03	1.92	1.656E-04	5.23	1.544E-04	5.92	1.281E-04	5.35	1.475E-04	4.97
4	0.000E+00	0.00	4.204E-03	1.14	5.437E-04	3.46	5.072E-04	3.55	4.983E-04	3.51	4.957E-04	2.91
5	0.000E+00	0.00	2.834E-03	1.37	3.175E-04	3.97	3.393E-04	3.74	3.345E-04	4.01	3.336E-04	3.86
6	0.000E+00	0.00	8.974E-04	2.41	9.913E-05	5.97	1.171E-04	7.88	1.041E-04	6.83	1.040E-04	6.43
7	0.000E+00	0.00	3.574E-03	1.33	4.377E-04	3.59	4.251E-04	3.72	3.972E-04	4.34	4.402E-04	3.67
8	0.000E+00	0.00	4.386E-03	1.18	5.304E-04	3.18	5.120E-04	3.19	4.895E-04	3.44	5.272E-04	3.33
9	0.000E+00	0.00	6.307E-03	1.08	7.926E-04	3.15	7.232E-04	3.15	6.767E-04	3.19	7.520E-04	2.96
10	0.000E+00	0.00	1.103E-02	0.78	1.355E-03	2.44	1.291E-03	2.43	1.246E-03	2.54	1.271E-03	2.40
11	0.000E+00	0.00	1.178E-02	0.74	1.464E-03	2.26	1.336E-03	2.36	1.340E-03	2.54	1.391E-03	2.39
12	0.000E+00	0.00	7.178E-03	0.92	8.611E-04	3.00	8.099E-04	2.97	7.276E-04	3.03	7.950E-04	2.93
13	0.000E+00	0.00	1.595E-03	1.71	2.171E-04	5.22	1.834E-04	5.38	1.810E-04	5.70	1.772E-04	5.20
14	0.000E+00	0.00	7.130E-03	0.92	8.294E-04	2.75	7.465E-04	2.97	7.295E-04	3.31	8.029E-04	2.87
15	0.000E+00	0.00	6.261E-03	0.92	7.122E-04	2.72	6.722E-04	2.81	6.210E-04	2.98	6.581E-04	2.97
16	0.000E+00	0.00	5.505E-03	0.92	5.951E-04	2.66	5.580E-04	2.90	5.222E-04	2.90	5.567E-04	2.73
17	0.000E+00	0.00	3.273E-03	1.15	3.484E-04	3.02	3.119E-04	3.32	2.897E-04	3.31	3.040E-04	3.06
18	0.000E+00	0.00	2.444E-03	1.36	2.262E-04	3.42	2.102E-04	3.45	2.065E-04	3.42	2.197E-04	3.25
19	0.000E+00	0.00	4.374E-04	2.80	3.954E-05	7.46	3.452E-05	6.89	3.002E-05	7.71	3.751E-05	8.08
20	0.000E+00	0.00	5.568E-04	2.75	4.471E-05	6.49	4.308E-05	6.56	3.613E-05	6.99	4.229E-05	6.87
21	0.000E+00	0.00	4.168E-04	2.97	3.427E-05	6.93	2.959E-05	7.91	3.034E-05	7.48	3.174E-05	7.49
22	0.000E+00	0.00	7.767E-04	2.22	5.679E-05	5.29	6.221E-05	5.63	5.160E-05	5.67	5.866E-05	5.56
23	0.000E+00	0.00	8.810E-04	2.08	6.311E-05	4.74	6.017E-05	4.94	5.510E-05	4.97	6.113E-05	4.85
24	0.000E+00	0.00	9.433E-04	1.98	5.403E-05	5.21	5.279E-05	5.27	5.165E-05	5.02	5.716E-05	4.83
25	0.000E+00	0.00	7.081E-04	2.24	4.488E-05	5.09	3.932E-05	5.31	3.755E-05	5.45	3.815E-05	5.11
26	0.000E+00	0.00	6.778E-04	2.21	3.444E-05	5.51	3.357E-05	5.09	2.928E-05	5.60	3.339E-05	5.59
27	0.000E+00	0.00	8.796E-05	4.92	4.091E-06	13.38	5.436E-06	12.26	4.366E-06	15.39	4.106E-06	14.27
28	0.000E+00	0.00	9.516E-05	5.12	6.096E-06	12.92	4.348E-06	14.18	3.879E-06	14.43	4.893E-06	13.21
29	0.000E+00	0.00	1.080E-04	4.50	5.983E-06	13.01	5.313E-06	15.04	5.081E-06	12.86	6.356E-06	11.50
30	0.000E+00	0.00	2.454E-04	3.50	1.201E-05	8.66	1.019E-05	11.42	1.107E-05	8.96	1.019E-05	8.98
31	0.000E+00	0.00	1.288E-04	3.78	5.914E-06	11.15	6.818E-06	13.22	5.718E-06	12.02	5.699E-06	12.93
32	0.000E+00	0.00	1.513E-04	3.91	6.783E-06	10.57	6.677E-06	10.90	6.587E-06	11.27	7.080E-06	9.90
33	0.000E+00	0.00	1.739E-04	3.52	6.496E-06	9.67	6.806E-06	9.95	7.721E-06	10.56	6.509E-06	10.88
34	0.000E+00	0.00	4.281E-04	2.47	1.835E-05	6.12	1.563E-05	6.45	1.648E-05	6.63	1.735E-05	6.08
35	0.000E+00	0.00	6.916E-04	1.90	2.472E-05	5.58	2.395E-05	5.19	2.208E-05	5.36	2.412E-05	5.32
36	0.000E+00	0.00	6.888E-04	1.78	2.515E-05	4.84	2.490E-05	5.24	2.035E-05	6.30	2.428E-05	4.73
37	0.000E+00	0.00	5.795E-04	1.93	1.896E-05	5.63	1.813E-05	5.20	1.732E-05	5.52	1.871E-05	5.04
38	0.000E+00	0.00	3.240E-04	2.18	1.036E-05	7.29	8.607E-06	7.85	1.001E-05	7.77	9.824E-06	8.23
39	0.000E+00	0.00	3.261E-04	2.37	9.701E-06	8.07	8.411E-06	7.75	7.653E-06	7.96	9.549E-06	7.42
40	0.000E+00	0.00	1.468E-04	3.28	3.917E-06	10.32	4.053E-06	10.54	3.024E-06	12.74	3.141E-06	11.64
41	0.000E+00	0.00	3.566E-04	2.29	1.058E-05	6.57	9.020E-06	7.10	8.858E-06	7.00	9.430E-06	6.96
42	0.000E+00	0.00	3.604E-05	5.88	1.009E-06	27.70	8.304E-07	19.11	8.564E-07	25.72	8.251E-07	21.12
43	0.000E+00	0.00	3.968E-05	5.42	9.087E-07	18.02	1.018E-06	16.41	8.949E-07	30.82	6.629E-07	20.54
44	0.000E+00	0.00	6.744E-06	11.51	2.102E-07	37.98	1.685E-07	40.02	1.729E-08	70.77	2.139E-07	37.00

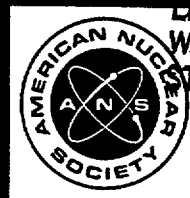
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**nuclear criticality safety in operations
with fissionable materials outside reactors**

(Reaffirmed in 1984)

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Foreword

(This Foreword is not a part of American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, ANSI/ANS-8.1-1983.)

This standard provides guidance for the prevention of criticality accidents in the handling, storing, processing, and transporting of fissionable material. It was first approved as American Standard N6.1-1964. A substantial revision that included the specification of subcritical limits applicable to process variables was approved as American National Standard N16.1-1969 and was reaffirmed, with minor revisions, as American National Standard N16.1-1975/ANS-8.1, under the prescribed five-year review. It was subsequently supplemented by American National Standard for Validation of Computational Methods for Nuclear Criticality Safety, ANSI N16.9-1975/ANS-8.11. To lessen the proliferation of nuclear criticality safety standards, the two standards have been consolidated in the present prescribed five-year review.

An important part of the present review was the examination of subcritical limits in the standard. In a few cases limits have been increased where the margin of subcriticality seemed unnecessarily large. In other cases, where subcriticality appeared doubtful, the limits have been reduced. Additional limits have been provided where they seemed likely to be useful. The limits make no allowance for operating contingencies (e.g., double batching) or for inaccurate knowledge of process variables (e.g., concentrations, masses, dimensions) and are "maximum subcritical limits." That is, under the stated conditions, the limits are close enough to critical to provide little incentive for attempting to justify slightly larger values, but, concomitantly, they are confidently expected actually to be subcritical. The stated conditions (infinitely long cylinders, absence of neutron-absorbing vessel walls, plutonium solutions without free nitric acid, etc.) are unlikely to be approached in practice; hence if a limit is reached, there will ordinarily be a larger margin of subcriticality than the minimal value used in its derivation. However, no account was taken of this unlikelihood in setting the limits. It is legitimate for the user of the standard, if he so chooses, (conservatively) to make adjustments in the limits to take advantage of the extent to which credible potential conditions may deviate from stated conditions, e.g., to increase a cylinder diameter limit to take advantage of a finite height and of neutron absorption in steel walls.

The prescribed five-year review of American National Standards N16.1-1975/ANS-8.1 and N16.9-1975/ANS-8.11 was performed by Subcommittee 8 of the Standards Committee of the American Nuclear Society, with Dr. H. K. Clark assuming principal responsibility for the revision. Limits were derived in accordance with the standard. The derivations have been reviewed by the subcommittee and have been published, largely in the open literature.

This revised standard was prepared under the guidance of ANS Subcommittee 8, Fissionable Materials Outside Reactors, which had the following membership at the time of its approval of this revision:

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Elizabeth B. Johnson, Secretary, *Oak Ridge National Laboratory*
F. M. Alcorn, *Babcock & Wilcox Company*
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E. D. Clayton, *Battelle Pacific Northwest Laboratories*

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American National Standards Committee N16, Nuclear Criticality Safety, which reviewed and approved this revision in 1982, had the following membership:

Dixon Callihan, Chairman
Elizabeth B. Johnson, Secretary

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Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors

1. Introduction

Operations with some fissionable materials introduce risks of a criticality accident resulting in a release of radiation that may be lethal to nearby personnel. However, experience has shown that extensive operations can be performed safely and economically when proper precautions are exercised. The few criticality accidents that have occurred show frequency and severity rates far below those typical of nonnuclear accidents. This favorable record can be maintained only by continued adherence to good operating practices such as are embodied in this standard; however, the standard, by itself, cannot establish safe processes in an absolute sense. Good safety practices must recognize economic considerations, but the protection of operating personnel¹ and the public must be the dominant consideration.

2. Scope

This standard is applicable to operations with fissionable materials outside nuclear reactors, except the assembly of these materials under controlled conditions, such as in critical experiments. Generalized basic criteria are presented and limits are specified for some single fissionable units of simple shape containing ^{233}U , ^{235}U , or ^{239}Pu , but not for multiunit arrays.² Requirements are stated for establishing the validity and areas of applicability of any calculational method used in assessing nuclear criticality safety. This standard does not include the details of administrative controls, the design of processes or equipment, the description of instrumentation for process control, or detailed criteria to be met in transporting fissionable materials.

¹Guidance for establishing an alarm system is contained in American National Standard Criticality Accident Alarm System, ANSI/ANS-8.3-1979.

²Limits for certain multiunit arrays are contained in American National Standard Guide for Nuclear Criticality Safety in the Storage of Fissile Materials, ANSI/ANS-8.7-1982.

3. Definitions

3.1 Limitations. The definitions given below are of a restricted nature for the purposes of this standard. Other specialized terms are defined in American National Standard Glossary of Terms in Nuclear Science and Technology, ANSI N1.1-1976/ANS-9 [1].³

3.2 Shall, Should, and May. The word "shall" is used to denote a requirement, the word "should" to denote a recommendation, and the word "may" to denote permission, neither a requirement nor a recommendation. In order to conform with this standard, all operations shall be performed in accordance with its requirements, but not necessarily with its recommendations.

3.3 Glossary of Terms

area(s) of applicability. The ranges of material compositions and geometric arrangements within which the bias of a calculational method is established.

areal density. The total mass of fissionable material per unit area projected perpendicularly onto a plane. (For an infinite, uniform slab, it is the product of the slab thickness and the concentration of fissionable material within the slab.)

bias. A measure of the systematic disagreement between the results calculated by a method and experimental data. The uncertainty in the bias is a measure of both the precision of the calculations and the accuracy of the experimental data.

calculational method (method). The mathematical equations, approximations, assumptions, associated numerical parameters (e.g., cross sections), and calculational procedures which yield the calculated results.

controlled parameter. A parameter that is kept within specified limits.

criticality accident. The release of energy as a result of accidentally producing a self-sustaining or divergent neutron chain reaction.

³Numbers in brackets refer to corresponding numbers in Section 7, References.

effective multiplication factor (k_{eff}). The ratio of the total number of neutrons produced during a time interval (excluding neutrons produced by sources whose strengths are not a function of fission rate) to the total number of neutrons lost by absorption and leakage during the same interval.

nuclear criticality safety. Protection against the consequences of an inadvertent nuclear chain reaction, preferably by prevention of the reaction.

subcritical limit (limit). The limiting value assigned to a controlled parameter that results in a subcritical system under specified conditions. The subcritical limit allows for uncertainties in the calculations and experimental data used in its derivation but not for contingencies; e.g., double batching or failure of analytical techniques to yield accurate values.

4. Nuclear Criticality Safety Practices

4.1 Administrative Practices

4.1.1 Responsibilities. Management shall clearly establish responsibility for nuclear criticality safety. Supervision should be made as responsible for nuclear criticality safety as for production, development, research, or other functions. Each individual, regardless of position, shall be made aware that nuclear criticality safety in his work area is ultimately his responsibility. This may be accomplished through training and periodic retraining of all operating and maintenance personnel. Nuclear criticality safety differs in no intrinsic way from industrial safety, and good managerial practices apply to both.

Management shall provide personnel skilled in the interpretation of data pertinent to nuclear criticality safety and familiar with operations to serve as advisors to supervision. These specialists should be, to the extent practicable, administratively independent of process supervision.

Management shall establish the criteria to be satisfied by nuclear criticality safety controls. Distinction may be made between shielded and unshielded facilities, and the criteria may be less stringent when adequate shielding and confinement assure the protection of personnel.⁴

⁴Guidance is provided in American National Standard Criteria for Nuclear Criticality Safety Controls in Operations with Shielding and Confinement, ANSI/ANS-8.10-1983.

4.1.2 Process Analysis. Before a new operation with fissionable materials is begun or before an existing operation is changed, it shall be determined that the entire process will be subcritical under both normal and credible abnormal conditions.⁵ Care shall be exercised to determine those conditions which result in the maximum effective multiplication factor (k_{eff}).

4.1.3 Written Procedures. Operations to which nuclear criticality safety is pertinent shall be governed by written procedures. All persons participating in these operations shall understand and be familiar with the procedures. The procedures shall specify all parameters they are intended to control. They should be such that no single, inadvertent departure from a procedure can cause a criticality accident.

4.1.4 Materials Control. The movement of fissionable materials shall be controlled. Appropriate materials labeling and area posting shall be maintained specifying material identification and all limits on parameters that are subjected to procedural control.

4.1.5 Operational Control. Deviations from procedures and unforeseen alterations in process conditions that affect nuclear criticality safety shall be reported to management and shall be investigated promptly. Action shall be taken to prevent a recurrence.

4.1.6 Operational Reviews. Operations shall be reviewed frequently (at least annually) to ascertain that procedures are being followed and that process conditions have not been altered so as to affect the nuclear criticality safety evaluation. These reviews shall be conducted, in consultation with operating personnel, by individuals who are knowledgeable in nuclear criticality safety and who, to the extent practicable, are not immediately responsible for the operation.

4.1.7 Emergency Procedures. Emergency procedures shall be prepared and approved by management. Organizations, local and offsite, that are expected to respond to emergencies shall be made aware of conditions that might be encountered, and they should be assisted in preparing suitable procedures governing their responses.

⁵In some cases it may be necessary to resort to *in situ* neutron multiplication measurements to confirm the subcriticality of proposed configurations. Guidance for safety in performing such measurements is contained in American National Standard for Safety in Conducting Subcritical Neutron-Multiplication Measurements In Situ, ANSI/ANS-8.6-1983.

4.2 Technical Practices

4.2.1 Controlling Factors. The effective multiplication factor (k_{eff}) of a system containing fissionable material depends on:

- (1) The mass and distribution of all fissionable materials and
- (2) The mass, distribution, and nuclear properties of all other materials with which the fissionable materials are associated.

Nuclear criticality safety is achieved by controlling one or more parameters of the system within subcritical limits. Control may be exercised administratively through procedures (e.g., by requiring that a mass not exceed a posted limit), by physical restraints (e.g., by confining a solution to a cylindrical vessel with diameter no greater than the subcritical limit), through the use of instrumentation (e.g., by keeping a fissile concentration below a specific limit by devices that measure concentration and prevent its buildup through reflux in a chemical system), by chemical means (e.g., by prevention of conditions that allow precipitation, thereby maintaining concentration characteristic of an aqueous solution), by relying on the natural or credible course of events (e.g., by relying on the nature of a process to keep the density of uranium oxide less than a specified fraction of theoretical), or by other means. All controlled parameters and their limits shall be specified.

4.2.2 Double Contingency Principle. Process designs should, in general, incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

4.2.3 Geometry Control. Where practicable, reliance should be placed on equipment design in which dimensions are limited⁶ rather than on administrative controls. Full advantage may be taken of any nuclear characteristics of the process materials and equipment. All dimensions and nuclear properties on which reliance is placed shall be verified prior to beginning operations, and control shall be exercised to maintain them.

4.2.4 Neutron Absorbers. Reliance may be placed on neutron-absorbing materials, such as cadmium and boron, that are incorporated in

process materials or equipment, or both.⁷ Control shall be exercised to maintain their continued presence with the intended distributions and concentrations. Extraordinary care should be taken with solutions of absorbers because of the difficulty of exercising such control.

4.2.5 Subcritical Limits. Where applicable data are available, subcritical limits shall be established on bases derived from experiments, with adequate allowance for uncertainties in the data. In the absence of directly applicable experimental measurements, the limits may be derived from calculations made by a method shown by comparison with experimental data to be valid in accordance with 4.3.

4.3 Validation of a Calculational Method. There are many calculational methods suitable for determining the effective multiplication factor (k_{eff}) of a system or for deriving subcritical limits. The methods vary widely in basis and form, and each has its place in the broad spectrum of problems encountered in the nuclear criticality safety field. However, the general procedure to be followed in establishing validity is common to all.

4.3.1 Bias shall be established by correlating the results of criticality experiments with results obtained for these same systems by the method being validated. Commonly the correlation is expressed in terms of the values of k_{eff} calculated for the experimental systems, in which case the bias is the deviation of the calculated values of k_{eff} from unity. However, other parameters may be used. The bias serves to normalize a method over its area(s) of applicability so that it will predict critical conditions within the limits of the uncertainty in the bias. Generally neither the bias nor its uncertainty is constant; both should be expected to be functions of composition and other variables.

4.3.2 The area(s) of applicability of a calculational method may be extended beyond the range of experimental conditions over which the bias is established by making use of the trends in the bias. Where the extension is large, the method should be supplemented by other calculational methods to provide a better estimate of the bias in the extended area(s).

⁶Guidance for assessing the safety of piping systems for uranyl nitrate solutions is contained in American National Standard Nuclear Criticality Safety Guide for Pipe Intersections Containing Aqueous Solutions of Enriched Uranyl Nitrate, ANSI/ANS-8.9-1978.

⁷Guidance for the use of a particular absorber is contained in American National Standard Use of Borosilicate-Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material, ANSI/ANS-8.5-1979.

4.3.3 A margin in the correlating parameter, which margin may be a function of composition and other variables, shall be prescribed that is sufficient to ensure subcriticality. This margin of subcriticality shall include allowances for the uncertainty in the bias and for uncertainties due to any extensions of the area(s) of applicability.

4.3.4 If the method involves a computer program, checks shall be performed to confirm that the mathematical operations are performed as intended. Any changes in the computer program shall be followed by reconfirmation that the mathematical operations are performed as intended.

4.3.5 Nuclear properties such as cross sections should be consistent with experimental measurements of these properties.

4.3.6 A written report of the validation shall be prepared.⁸ This report shall:

(1) Describe the method with sufficient detail, clarity, and lack of ambiguity to allow independent duplication of results.

(2) State computer programs used, the options, recipes for choosing mesh points where applicable, the cross section sets, and any numerical parameters necessary to describe the input.

(3) Identify experimental data and list parameters derived therefrom for use in the validation of the method.

(4) State the area(s) of applicability.

(5) State the bias and the prescribed margin of subcriticality over the area(s) of applicability. State the basis for the margin.

5. Single-Parameter Limits for Fissile Nuclides

Operations with fissile materials may be performed safely by complying with any one of the limits given in 5.1, 5.2, 5.3, and 5.4 for single units provided the conditions under which the limit applies are maintained; these limits were calculated by methods satisfying the requirements of 4.3. A limit shall be applied only when surrounding materials, including other nearby fissionable materials, can be shown to increase the effective multiplication factor (k_{eff}) no more than does enclosing the unit by a contiguous layer of water of unlimited thickness. A limit

may be applied to a mixture of fissile nuclides by considering all components of the mixture to be the one with the most restrictive limit.

Process specifications shall incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.

5.1 Uniform Aqueous Solutions. Any one of the limits of Table 1 is applicable provided a uniform aqueous solution is maintained. It is therefore implied that the concentrations of the saturated solutions are not exceeded. The ²³⁹Pu limits apply to mixtures of plutonium isotopes provided the concentration of ²⁴⁰Pu exceeds that of ²⁴¹Pu and provided ²⁴¹Pu is considered to be ²³⁹Pu in computing mass or concentration. (Less restrictive limits are provided in 6.3 for plutonium isotopic compositions containing appreciable concentrations of ²⁴⁰Pu.) The limit on atomic ratio is equivalent to the limit on solution concentration, but the ratio limit may also be applied to non-aqueous solutions regardless of the chemical form of the fissile nuclide.

5.2 Aqueous Mixtures. The areal densities of Table 1 are independent of chemical compound and are valid for mixtures which may have density gradients provided the areal densities are uniform. The subcritical mass limits for ²³⁵U, ²³⁵U, and ²³⁹Pu in mixtures that may not be uniform are 0.50, 0.70, and 0.45 kg, respectively, and are likewise independent of compound [2-4].

5.2.1 Enrichment Limits. Table 2 contains ²³⁵U enrichment limits for uranium compounds mixed homogeneously⁹ with water with no limitations on mass or concentration.

⁹In the "homogeneous" mixtures to which calculations of these limits were normalized the average particle size of dry UO₃ was 60 microns [V. I. NEELEY and H. E. HANDLER, "Measurement of Multiplication Constant for Slightly Enriched Homogeneous UO₃-Water Mixtures and Minimum Enrichment for Criticality," HW-70310, Hanford Atomic Products Operations (August 1961)]. It seems likely that the average particle size of the dihydrate of UO₂(NO₃)₂ was approximately 100 microns [V. I. NEELEY, J. A. BERBERET and R. H. MASTERSON, " k_{∞} of Three Weight Per Cent ²³⁵U Enriched UO₃ and UO₂(NO₃)₂ Hydrogenous Systems," HW-66882, Hanford Atomic Products Operations (September 1961)]. Various H/U ratios in the nitrate mixtures were achieved with 1/8-inch spheres of polyethylene [S. R. BIERMAN and G. M. HESS, "Minimum Critical ²³⁵U Enrichment of Homogeneous Uranyl Nitrate," ORNL-CDC-5, Oak Ridge Criticality Data Center (June 1968)].

⁸Management may limit the distribution of the report to protect proprietary information.

5.3 Metallic Units. The enrichment limit for uranium and the mass limits given in Table 3 apply to a single piece having no concave surfaces. They may be extended to an assembly of pieces provided there is no interspersed moderation.

The ^{235}U and ^{235}U limits apply to mixtures of either isotope with ^{234}U , ^{236}U , or ^{238}U provided ^{234}U is considered to be ^{233}U or ^{235}U , respectively, in computing mass [3]. The ^{239}Pu limits apply to isotopic mixtures of plutonium provided the concentration of ^{240}Pu exceeds that of ^{241}Pu and all isotopes are considered to be ^{239}Pu in computing mass [4]. Density limits may be adjusted for isotopic composition.

5.4 Oxides. The limits in Tables 4 and 5 apply only if the oxide contains no more than 1.5% water by weight. The mass limits apply to a single piece having no concave surfaces. They may be extended to an assembly of pieces provided there is no additional interspersed moderation.

The mass limit is given equivalently as mass of nuclide and as mass of oxide (including moisture). It is emphasized that the limits in Tables 4 and 5 are valid only under the specified bulk density restrictions.¹⁰ With water content limited to 1.5% the enrichment limit of Table 2 for uranium oxides is increased to 3.2% ^{235}U [3].

¹⁰The user is cautioned that, particularly for UO_3 , material densities in excess of the full densities of Table 4 may be possible and hence that the limits of Table 4 may not be valid for highly compacted oxides. However, it is expected that oxides will generally be in the form of loose powders or, in the case of UO_2 , of accumulations of pellets and that the limits of Table 4 and perhaps Table 5 will be valid. Where other density limits are desired, where it is inconvenient to maintain the water content below 1.5% ($\text{H}/\text{U} \approx 0.47$), or where oxides are non-stoichiometric, the limits may be useful as points of departure in deriving more appropriate values.

The maximum bulk densities were derived from CRC Handbook values of 10.96, 8.3, 7.29, and 11.46 g/cm^3 for UO_2 , U_3O_8 , UO_3 , and PuO_2 together with the assumption of additive volumes of oxide and water. However, x-ray densities of UO_3 as high as 8.46 g/cm^3 have been reported. Moreover, the assumption of additive volumes may be incorrect; with H_2O assigned a density of unity, an effective UO_3 density of 10.47 g/cm^3 is required to produce a reported x-ray density of 6.71 g/cm^3 for $\alpha\text{-UO}_2(\text{OH})_2$.

6. Multiparameter Control

Although the single-parameter limits are adequate for many purposes, they are inconveniently and uneconomically small for many others. Simultaneous limitation of two or more parameters results in a less restrictive limit for the one of interest. A few particularly useful examples are given in 6.1 through 6.4. All were calculated by methods satisfying 4.3. These limits shall be applied only when surrounding materials can be shown to increase the effective multiplication factor (k_{eff}) no more than does enclosing the system by a contiguous layer of water of unlimited thickness. General guidance for multiparameter control may be found in the technical literature.¹¹⁻¹⁴

Process specifications shall incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.

6.1 Uranium Metal and Uranium Oxide-Water Mixtures at Low ^{235}U Enrichment. An application of multiparameter control is control of both the ^{235}U enrichment of uranium and one of the parameters of Section 5. Subcritical limits [5] applicable to aqueous systems containing uranium metal or uranium oxide (UO_2), regardless of the size and shape of metal or oxide pieces, are specified as functions of enrichment in Figs. 1 through 5 which give, respectively, the mass of ^{235}U , the cylinder diameter, the slab thickness, the volume, and the areal density.¹⁵

¹¹H. C. PAXTON, J. T. THOMAS, D. CALLIHAN, and E. B. JOHNSON, "Critical Dimensions of Systems Containing ^{235}U , ^{239}Pu , and ^{233}U ," TID-7028, U.S. Atomic Energy Commission (1964).

¹²J. T. THOMAS, "Nuclear Safety Guide, TID-7016, Rev. 2," NUREG/CR-0095 (ORNL/NUREG/CSD-6), Oak Ridge National Laboratory (1978).

¹³H. K. CLARK, "Handbook of Nuclear Safety," DP-532, Savannah River Laboratory (1961).

¹⁴R. D. CARTER, G. R. KEIL, K. R. RIDGWAY, "Criticality Handbook," ARH-600, Atlantic Richfield Hanford Company (1973).

¹⁵The data points through which the curves in Figs. 1-5 were drawn are the subcritical values listed in Tables VI-VIII of Ref. [5].

6.2 Aqueous Uranium Solutions at Low ^{235}U Enrichment. A similar application of multi-parameter control is control of both ^{235}U enrichment and one of the parameters of Table 1, together with the maintenance of a uniform aqueous solution. Table 6 lists subcritical limits for uniform aqueous solutions of uranium where the enrichment is controlled within the stated limit. Concentrations of saturated solutions, which are here taken to be 5 molar for UO_2F_2 solutions and 2.5 molar for $\text{UO}_2(\text{NO}_3)_2$ solutions, shall not be exceeded.

6.3 Uniform Aqueous Solutions of $\text{Pu}(\text{NO}_3)_4$ Containing ^{240}Pu . Reliance on, and hence control of, the isotopic concentration of ^{240}Pu in plutonium permits greater limits for $\text{Pu}(\text{NO}_3)_4$ solutions than are listed in Table 1.¹⁶ However, the amount of the increase is dependent on ^{241}Pu concentration. Table 7 contains limits for uniform aqueous solutions of $\text{Pu}(\text{NO}_3)_4$ as a function of isotopic composition. Any ^{238}Pu or ^{242}Pu present shall be omitted in computing the isotopic composition.

6.4 Aqueous Mixtures of Plutonium Containing ^{240}Pu . Subcritical mass limits for plutonium as PuO_2 in aqueous mixtures, which may be nonuniform, where ^{240}Pu and ^{241}Pu are subject

¹⁶Where plutonium, in addition, is intimately mixed with natural uranium, limits are even greater. Limits for this case are included in American National Standard for Nuclear Criticality Control and Safety of Homogeneous Plutonium-Uranium Fuel Mixtures Outside Reactors, ANSI/ANS-8.12-1978.

to the three pairs of restrictions on isotopic composition of Table 7, are, in increasing order of ^{240}Pu concentration, 0.53, 0.74, and 0.99 kg, respectively [4].

7. References

- [1] American National Standard Glossary of Terms in Nuclear Science and Technology, ANSI/N1.1-1976/ANS-9. American Nuclear Society, La Grange Park, Ill.
- [2] H. K. CLARK, "Subcritical Limits for ^{238}U Systems," *Nucl. Sci. Eng.* 81, 379-395 (1982). American Nuclear Society, La Grange Park, Ill.
- [3] H. K. CLARK, "Subcritical Limits for ^{235}U Systems," *Nucl. Sci. Eng.* 81, 351-378 (1982). American Nuclear Society, La Grange Park, Ill.
- [4] H. K. CLARK, "Subcritical Limits for Pu Systems," *Nucl. Sci. Eng.* 79, 65-84 (1981). American Nuclear Society, La Grange Park, Ill.
- [5] H. K. CLARK, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water," DP-1014, Savannah River Laboratory, Aiken, S. C., (1966).

When the preceding American National Standard referred to in this document is superseded by a revision approved by the American National Standards Institute, Inc., the revision shall apply.

Table 1

Single-Parameter Limits for Uniform Aqueous Solutions of Fissile Nuclides

Parameter	Subcritical Limit for Fissile Solute				
	$^{233}\text{UO}_2\text{F}_2$ [2]	$^{233}\text{UO}_2(\text{NO}_3)_2$ [2]	$^{235}\text{UO}_2\text{F}_2$ [3]	$^{235}\text{UO}_2(\text{NO}_3)_2$ [3]	$^{239}\text{Pu}(\text{NO}_3)_4$ [4]
Mass of fissile nuclide, kg	0.54	0.55	0.76	0.78	0.48
Diameter of cylinder of solution, cm	10.5	11.7	13.7	14.4	15.4
Thickness of slab of solution, cm	2.5	3.1	4.4	4.9	5.5
Volume of solution, l	2.8	3.6	5.5	6.2	7.3
Concentration of fissile nuclide, g/l	10.8	10.8	11.6	11.6	7.3
Atomic ratio of hydrogen to fissile nuclide ^(a)	2390	2390	2250	2250	3630
Areal density of fissile nuclide, g/cm ²	0.35	0.35	0.40	0.40	0.25

^(a) Lower limit

Table 2

 ^{235}U Enrichment Limits for Uranium Mixed Homogeneously with Water [3]

Compound	Subcritical Limit, wt% ^{235}U
Uranium metal	0.93
UO_2 , U_3O_8 , or UO_3	0.96
$\text{UO}_2(\text{NO}_3)_2$	1.96

Table 3

Single-Parameter Limits for Metal Units

Parameter	Subcritical Limit for		
	^{233}U [2]	^{235}U [3]	^{239}Pu [4]
Mass of fissile nuclide, kg	6.0	20.1	5.0
Cylinder diameter, cm	4.5	7.3	4.4
Slab thickness, cm	0.38	1.3	0.65
Uranium enrichment, wt% ^{235}U	—	5.0	—
Maximum density for which mass and dimension limits are valid, g/cm ³	18.65	18.81	19.82

Table 4

Single-Parameter Limits for Oxides Containing No More Than 1.5% Water By Weight at Full Density

Parameter	$^{233}\text{UO}_2$ [2]	$^{233}\text{U}_3\text{O}_8$ [2]	$^{233}\text{UO}_3$ [2]	$^{235}\text{UO}_2$ [3]	$^{235}\text{U}_3\text{O}_8$ [3]	$^{235}\text{UO}_3$ [3]	$^{239}\text{PuO}_2$ [4]
Mass of fissile nuclide, kg	10.1	13.4	15.2	32.3	44.0	51.2	10.2
Mass of oxide, ^(a) kg	11.7	16.0	18.7	37.2	52.8	62.6	11.5
Cylinder diameter, cm	7.2	9.0	9.9	11.6	14.6	16.2	7.2
Slab thickness, cm	0.8	1.1	1.3	2.9	4.0	4.6	1.4
Maximum bulk density ^(b) for which limits are valid, g/cm ³	$\frac{9.38}{1-0.085(1.5-w)}$	$\frac{7.36}{1-0.065(1.5-w)}$	$\frac{6.56}{1-0.056(1.5-w)}$	$\frac{9.44}{1-0.086(1.5-w)}$	$\frac{7.41}{1-0.065(1.5-w)}$	$\frac{6.60}{1-0.057(1.5-w)}$	$\frac{9.92}{1-0.091(1.5-w)}$

^(a) These values include the mass of any associated moisture up to the limiting value of 1.5% by weight.

^(b) w represents the quantity of water, in wt %, in the oxide.

Table 5

Single-Parameter Limits for Oxides Containing No More Than 1.5% Water By Weight at No More Than Half Density^(a)

Parameter	$^{233}\text{UO}_2$ [2]	$^{233}\text{U}_3\text{O}_8$ [2]	$^{233}\text{UO}_3$ [2]	$^{235}\text{UO}_2$ [3]	$^{235}\text{U}_3\text{O}_8$ [3]	$^{235}\text{UO}_3$ [3]	$^{239}\text{PuO}_2$ [4]
Mass of fissile nuclide, kg	23.4	30.5	34.7	88	122	142	27
Mass of oxide, ^(b) kg	27.0	36.6	42.4	102	146	174	30
Cylinder diameter, cm	11.9	14.8	16.3	20.4	26.0	28.8	12.6
Slab thickness, cm	1.6	2.2	2.6	5.8	8.0	9.3	2.8

^(a) These are half the maximum bulk densities of Table 4.

^(b) These values include the mass of any associated moisture up to the limiting value of 1.5% by weight.

Table 6

Subcritical Limits for Uniform Aqueous Solutions of Low-Enriched Uranium [3]

Parameter	Enrichment, wt% ^{235}U	Subcritical Limit	
		UO_2F_2	$\text{UO}_2(\text{NO}_3)_2$
Mass, kg ^{235}U	10.0	1.07	1.47
	5.0	1.64	3.30
	4.0	1.98	6.50
	3.0	2.75	—
	2.0	8.00	—
Cylinder diameter, cm	10.0	20.1	25.2
	5.0	26.6	42.7
	4.0	30.2	58.6
	3.0	37.4	—
	2.0	63.0	—
Slab thickness, cm	10.0	8.3	11.9
	5.0	12.6	23.4
	4.0	15.1	33.7
	3.0	20.0	—
	2.0	36.5	—
Volume, <i>l</i>	10.0	14.8	26.7
	5.0	30.6	111.0
	4.0	42.7	273.0
	3.0	77.0	—
	2.0	340.0	—
Concentration, g U/ <i>l</i>	10.0	123.0	128.0
	5.0	261.0	283.0
	4.0	335.0	375.0
	3.0	470.0	—
	2.88	—	594.9 ^(a)
	2.0	770.0	—
	1.45	1190.0 ^(a)	—

^(a) Saturated solution.

Table 7

Subcritical Limits for Uniform Aqueous Solutions of $\text{Pu}(\text{NO}_3)_4$ Containing ^{240}Pu [4]

Parameter	Subcritical Limit		
	≥ 5 wt% ^{240}Pu ≤ 1 wt% ^{241}Pu	≥ 15 wt% ^{240}Pu ≤ 6 wt% ^{241}Pu	≥ 25 wt% ^{240}Pu ≤ 15 wt% ^{241}Pu
Mass, kg Pu	0.57	0.78	1.02
Cylinder diameter, cm	17.4	19.5	21.3
Slab thickness, cm	6.7	8.0	9.2
Volume, <i>l</i>	10.0	13.6	17.2
Concentration, g Pu/ <i>l</i>	7.8	8.9	10.2
H/Pu	3400	2980	2600
Areal density, g Pu/cm ²	0.28	0.34	0.4

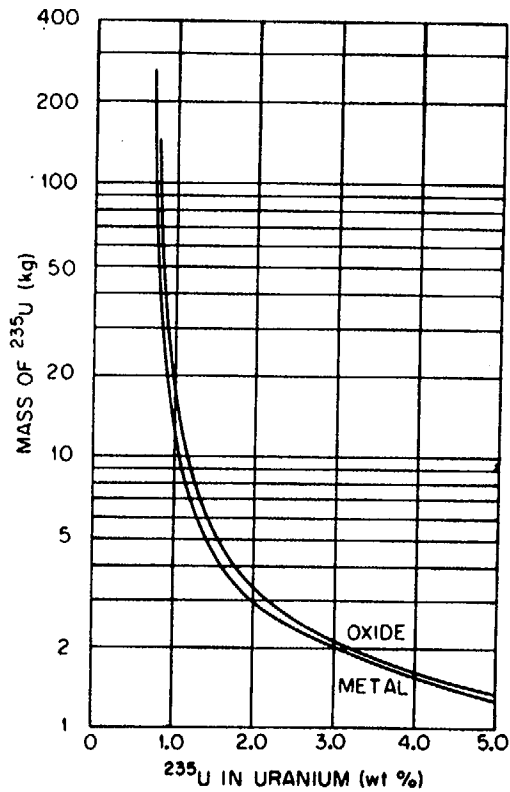


Fig. 1 Mass limit for uranium-water lattices.

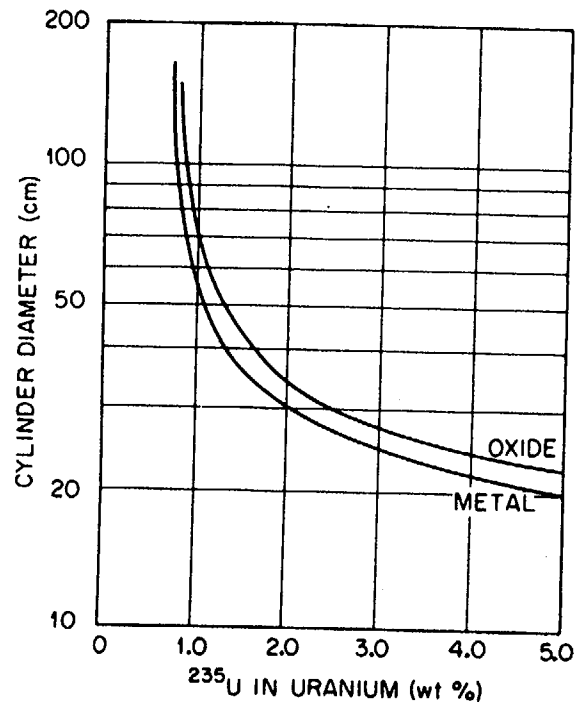


Fig. 2 Cylinder diameter limit for uranium-water lattices.

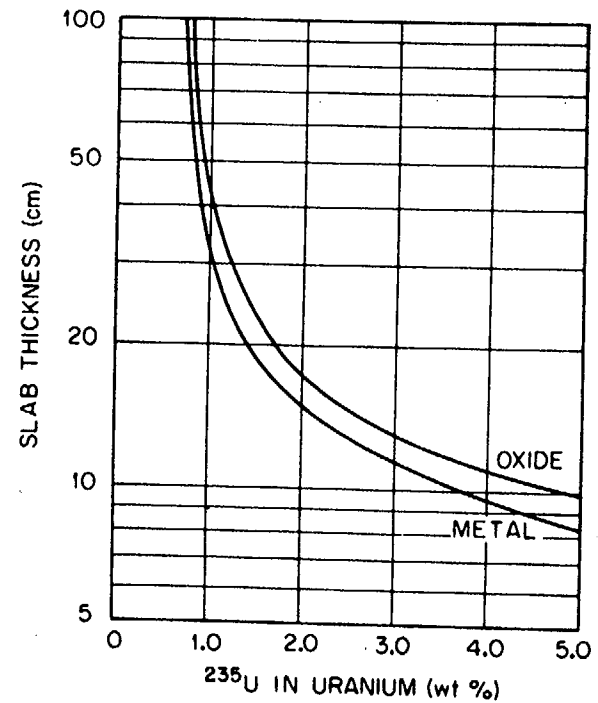


Fig. 3 Slab thickness limit for uranium-water lattices.

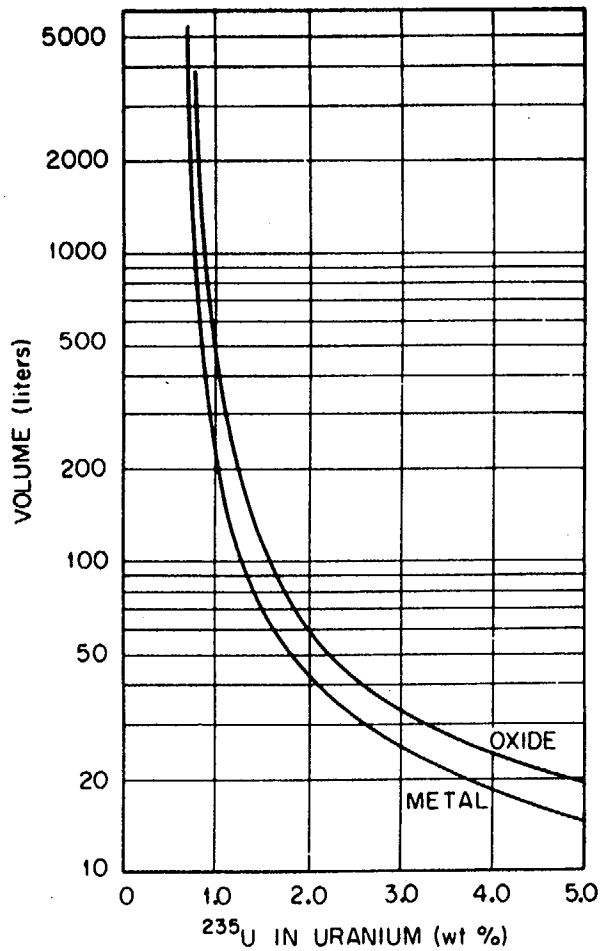


Fig. 4 Volume limit for uranium-water lattices.

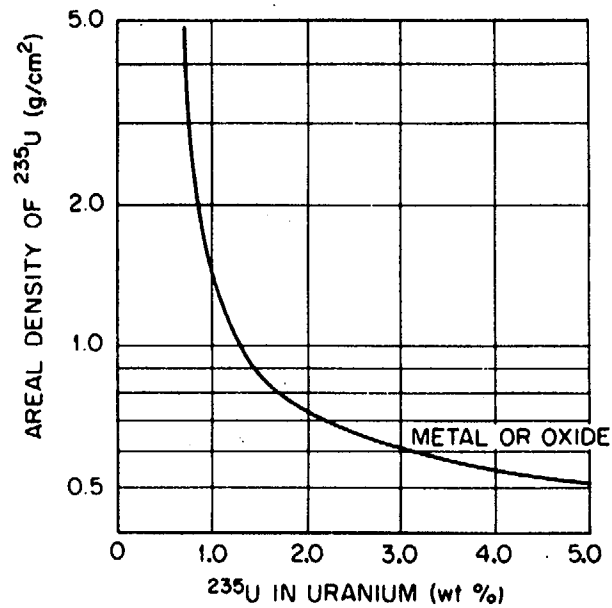


Fig. 5 Areal density limit for uranium-water lattices.

Appendix A

(This Appendix is not a part of American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, ANSI/ANS-8.1-1983, but is included for information purposes only.)

The determination, required by 4.2.1, that a process will be subcritical under normal and credible abnormal conditions and the determination of those conditions resulting in the maximum effective multiplication factor (k_{eff}) require careful study. The few criticality accidents that have occurred in industrial operations have resulted from failure to anticipate conditions that might arise; none has resulted from a faulty calculation¹⁷ of k_{eff} . The following are typical examples of variations in process conditions that should be considered:

- (1) A change in intended shape or dimensions resulting from bulging, corrosion, or bursting of a container, or failure to meet specifications in fabrication;
- (2) An increase in the mass of fissionable material in a location as the result of operational error, improper labeling, equipment failure, or failure of analytical techniques;
- (3) A change in the ratio of moderator to fissionable material resulting from:
 - (a) Inaccuracies in instruments or chemical analyses,
 - (b) Flooding, spraying, or otherwise supplying units or groups of units with water, oil, snow (i.e., low-density water), cardboard, wood, or other moderating material,
 - (c) Evaporating or displacing moderator,
 - (d) Precipitating fissionable material from solutions,
 - (e) Diluting concentrated solutions with additional moderator,
 - (f) Introducing air bubbles between rows of fuel assemblies in a storage basin;
- (4) A change in the fraction of the neutron population lost by absorption resulting from:
 - (a) Loss of solid absorber by corrosion or by leaching,
 - (b) Loss of moderator,
 - (c) Redistribution of absorber and fissionable material by precipitation of one but not the other from a solution,
 - (d) Redistribution of solid absorber within a matrix of moderator or solution by clumping,
 - (e) Failure to add the intended amount of absorber to a solution or failure to add it with the intended distribution,
 - (f) Failure of analytical techniques to yield correct amounts of concentrations;
- (5) A change in the amount of neutron reflection resulting from:
 - (a) An increase in reflector thickness by adding additional material (e.g., water or personnel),
 - (b) A change in reflector composition such as loss of absorber (e.g., by corrosion of an outer casing of absorber);
- (6) A change in the interaction between units and reflectors resulting from:
 - (a) The introduction of additional units or reflectors (e.g., personnel),
 - (b) Improper placing of units,
 - (c) Loss of moderator and absorber between units,
 - (d) Collapse of a framework used to space units;
- (7) An increase in the density of fissionable material.

¹⁷See H. C. PAXTON, "Criticality Control in Operations with Fissile Material," LA-3366 (Rev.), Los Alamos National Laboratory (1972).

Appendix B

(This Appendix is not a part of American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, ANSI/ANS-8.1-1983, but is included for information purposes only.)

The following fictitious example is presented to illustrate the application of the requirements of 4.3 for validating a calculational method.¹⁸

B1. Problem. Validate a method for calculating subcritical mass limits for water-reflected solutions of fissionable material (Fm) ranging in concentration from 2 to 32 g ²⁹²Fm/l with no restriction on shape.

B2. Method. The method selected consists of the XYZ computer code for spherical systems using Smith's approximations as described in NIL-3638 and Jones' 3-group cross sections (NIL-5000).

Available Data from Criticality Experiments - (NIL-2867)

Concentration, ρ (g ²⁹² Fm/l)	Critical Radius (cm)
2	19.9
8	10.7
16	10.2

The solution was contained in thin water-reflected spherical shells having nuclear properties differing insignificantly from those of water.

B3. Validation.

B3.1 The XYZ code was operational on the local computer. Sample problems distributed with the code were run. A comparison with results obtained from the code author for the sample problems indicated the code was operating correctly for multi-region spherical systems.

B3.2 Computations were made for the three experimental points and produced the following results:

Concentration, ρ (g ²⁹² Fm/l)	k_{eff}
2	1.0046 ± 0.0057
8	0.9864 ± 0.0041
16	0.9696 ± 0.0041

The quoted errors represent those introduced by the quoted experimental data errors. The calculations were converged to a computational error in k_{eff} of ±0.0001, which is small compared with the experimental error. Within the area of applicability covered by experimental data (2 to 16 g/l), the computed value of k_{eff} is a nearly linear function of concentration and there appears to be no reason to expect deviations from smooth behavior. The area of applicability, however, must be extended to include concentrations as great as 32 g/l. Between 2 and 16 g/l, k_{eff} as a function of concentration is slightly concave upward (see Fig. B1); hence linear extrapolation of the values at 8 and 16 might be expected to give an estimate of k_{eff} which is too low at 32 g/l. The linearly extrapolated result, which is shown in Fig. B1, is $k_{eff} = 0.936$. The large extrapolation, however, should receive further support.

B3.3 In view of the downward drift of k_{eff} with an increase in concentration, a study was made to determine the cause. The result of this study was that the epithermal capture cross section of ²⁹²Fm

¹⁸The literature contains other, more complex examples of validations generally meeting the requirements of 4.3. In particular, the subcritical limits in the standard were calculated by methods meeting these requirements.

appeared to have the greatest uncertainty and was the likely cause of the discrepancy. A reduction of 20% in the epithermal capture cross section was made. Calculations made with the modified method (epithermal cross sections reduced 20%) yielded:

Concentration, ρ (g $^{292}\text{Ft/l}$)	k_{eff}
2	1.0109
8	1.0084
16	1.0106

This modification produced results which appear to minimize the drift with concentration variation and which may be expected to produce a k_{eff} of approximately 1.01 at 32 g/l.

B3.4 The following calculations were made at a concentration of 32 g/l:

Radius (cm)	k_{eff} (original ^{292}Ft cross sections)	k_{eff} (80% epithermal ^{292}Ft cross sections)
12.429	1.0000	1.0708
11.274	0.9343	1.0000

This shows that the reactivity difference or relative bias between the two calculations is $\delta k_{\text{eff}}/k_{\text{eff}} = -0.068 \pm 0.002$.

B3.5 Based on the assumption that the modified method would yield a $k_{\text{eff}} = 1.01$ for a critical system, it can be determined by linear interpolation of the data shown in the table of B3.4 that the unmodified method should give a $k_{\text{eff}} = 0.9443$ for a critical water-reflected solution containing 32 g $^{292}\text{Ft/l}$, when using the XYZ code with the unmodified Jones cross sections.

B3.6 The bias for the XYZ code using unmodified Jones cross sections, over the concentration range $2 \leq \rho \leq 32$, is thus estimated to be:

Concentration, ρ (g $^{292}\text{Ft/l}$)	Bias
2	+0.0046
8	-0.0136
16	-0.0304
32	-0.0557

B3.7 The uncertainty in the bias in the range of 2 to 16 g $^{292}\text{Ft/l}$ is mainly due to experimental error. (Some uncertainty is associated with interpolation.) The uncertainty at 32 g $^{292}\text{Ft/l}$ also must cover all errors introduced by extrapolation. A margin in k_{eff} ample to compensate for uncertainty in the bias and to assure subcriticality was judged to be 0.03 in the 2 to 16 g $^{292}\text{Ft/l}$ range and 0.05 at 32 g $^{292}\text{Ft/l}$. Any system with k_{eff} , computed by this method, no greater than 0.9746, 0.9564, 0.9396, or 0.8943 for concentrations of 2, 8, 16, and 32 g $^{292}\text{Ft/l}$, respectively, is confidently expected to be subcritical.

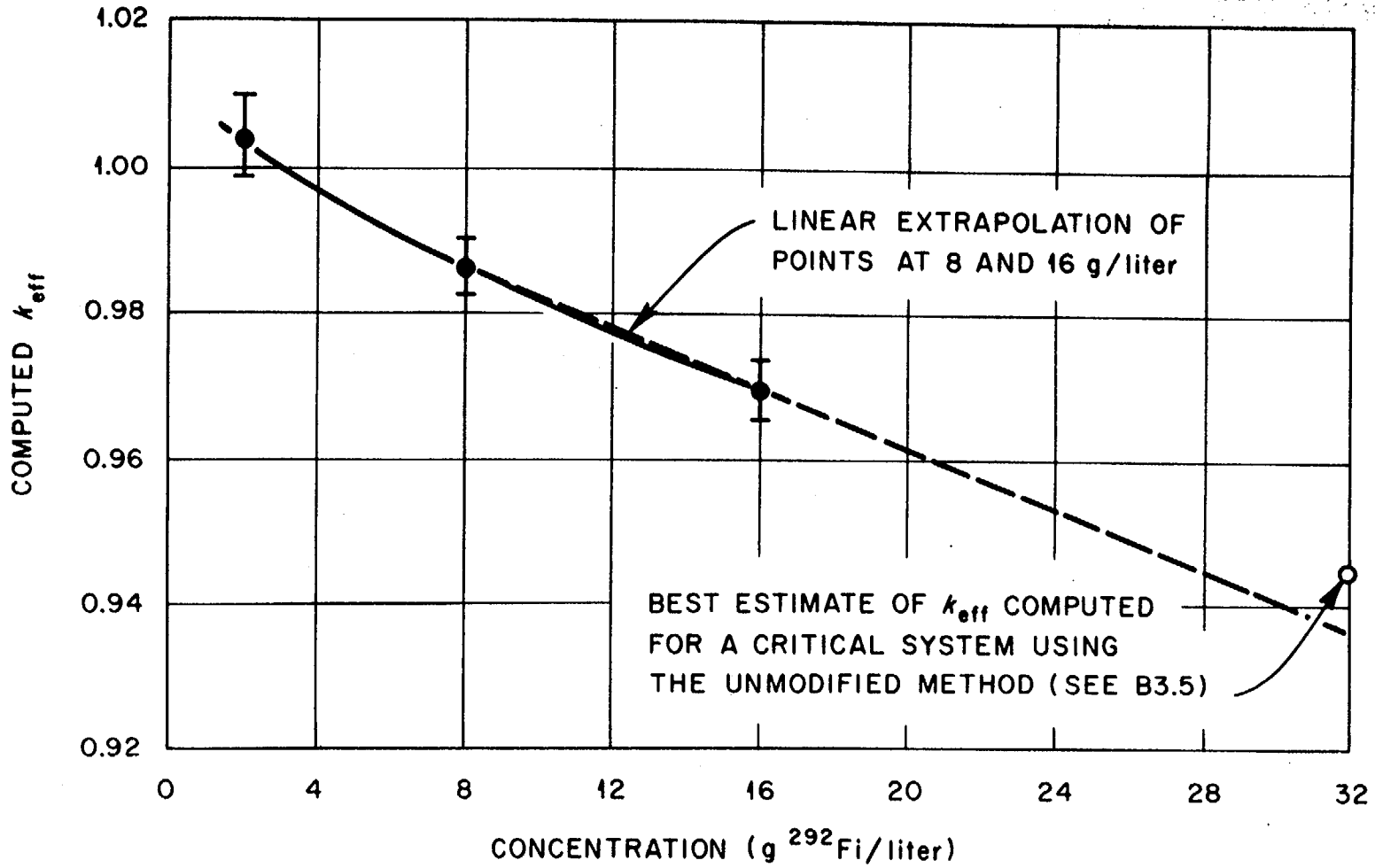


Fig. B1 Computed k_{eff} as a function of ^{292}Fi density for experimentally critical systems.