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ATTN: Document Control Desk
Director
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Louisiana Energy Services, L. P.
National Enrichment Facility
NRC Docket No. 70-3103

Subject: Revised MONK 8A Validation and Verification Report

References:

1. Letter NEF#03-003 dated December 12, 2003, from E. J. Ferland (Louisiana Energy Services, L. P.) to Directors, Office of Nuclear Material Safety and Safeguards and the Division of Facilities and Security (NRC) regarding "Applications for a Material License Under 10 CFR 70, Domestic licensing of special nuclear material, 10 CFR 40, Domestic licensing of source material, and 10 CFR 30, Rules of general applicability to domestic licensing of byproduct material, and for a Facility Clearance Under 10 CFR 95, Facility security clearance and safeguarding of national security information and restricted data"

2. Letter NEF#04-002 dated February 27, 2004, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Revision 1 to Applications for a Material License Under 10 CFR 70, "Domestic licensing of special nuclear material," 10 CFR 40, "Domestic licensing of source material," and 10 CFR 30, "Rules of general applicability to domestic licensing of byproduct material"

3. Letter NEF#04-029 dated July 30, 2004, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Revision to Applications for a Material License Under 10 CFR 70, "Domestic licensing of special nuclear material," 10 CFR 40, "Domestic licensing of source material," and 10 CFR 30, "Rules of general applicability to domestic licensing of byproduct material"

LES Exhibit 126-M

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4. Letter NEF#04-037 dated September 30, 2004, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Revision to Applications for a Material License Under 10 CFR 70, "Domestic licensing of special nuclear material," 10 CFR 40, "Domestic licensing of source material," and 10 CFR 30, "Rules of general applicability to domestic licensing of byproduct material"

5. Letter NEF#05-021 dated April 22, 2005, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Revision to Applications for a Material License Under 10 CFR 70, "Domestic licensing of special nuclear material," 10 CFR 40, "Domestic licensing of source material," and 10 CFR 30, "Rules of general applicability to domestic licensing of byproduct material"

6. Letter NEF#05-022 dated April 29, 2005, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Revision to Applications for a Material License Under 10 CFR 70, "Domestic licensing of special nuclear material," 10 CFR 40, "Domestic licensing of source material," and 10 CFR 30, "Rules of general applicability to domestic licensing of byproduct material"

7. Letter NEF#05-025 dated May 25, 2005, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Revision to Applications for a Material License Under 10 CFR 70, "Domestic licensing of special nuclear material," 10 CFR 40, "Domestic licensing of source material," and 10 CFR 30, "Rules of general applicability to domestic licensing of byproduct material".

8. Letter NEF#05-029 dated June 10, 2005, from R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Revision to Applications for a Material License Under 10 CFR 70, "Domestic licensing of special nuclear material," 10 CFR 40, "Domestic licensing of source material," and 10 CFR 30, "Rules of general applicability to domestic licensing of byproduct material"

9. Letter NEF#04-008 dated May 7, 2004, R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "MONK 8A Validation and Verification"

10. Letter NEF#05-015 dated March 28, 2005, R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Clarifying Information Related to Criticality Computer Code Validation"

By letter dated December 12, 2003 (Reference 1), E. J. Ferland of Louisiana Energy Services (LES), L. P., submitted to the NRC applications for the licenses necessary to authorize construction and operation of a gas centrifuge uranium enrichment facility. Revision 1 to these applications was submitted to the NRC by letter dated February 27, 2004 (Reference 2). Subsequent revisions (i.e., revision 2, revision 3, revision 4, revision 5, revision 6, and revision 7) to these applications were submitted to the NRC by letters dated July 30, 2004 (Reference 3), September 30, 2004 (Reference 4), April 22, 2005 (Reference 5), April 29, 2005 (Reference 6), May 25, 2005 (Reference 7), and June 10, 2005 (Reference 8) respectively. In addition, the Reference 9 letter provided to the NRC the validation and verification report for the criticality code used for the NEF nuclear criticality safety analyses (i.e., Revision 0 of the MONK 8A Validation and Verification report).

In the Reference 10 letter, LES committed to provide to the NRC, by December 30, 2005, a revised validation report for the criticality computer code used for the NEF nuclear criticality safety analyses. To satisfy this commitment, this letter provides Revision 1 of the MONK 8A Validation and Verification report. This revision of the MONK 8A Validation and Verification report meets the LES commitment to ANSI/ANS-8.1-1998, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," and includes details of validation that state computer codes used, operations, recipes for choosing code options (where applicable), cross section sets, and any numerical parameters necessary to describe the input.

If you have any questions, please contact me at 630-657-2813.

Respectfully,

R. M. Krich
Vice President – Licensing, Safety, and Nuclear Engineering

Enclosure:
MONK 8A Validation and Verification, National Enrichment Facility, Revision 1

cc: T. C. Johnson, NRC Project Manager

ENCLOSURE

MONK 8A Validation and Verification
National Enrichment Facility
Revision 1

MONK 8A
Validation and Verification

National Enrichment Facility

Revision 1

December 20, 2005

AREVA

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DISCLAIMER

ABSTRACT

The objective of this report is the validation of the MONK 8A, Monte Carlo computer code package. The validated MONK 8A code is then used to verify the criticality calculations performed by Urenco for the National Enrichment Facility.

MONK 8A was validated against a set of 80 benchmark critical experiments. The average of the validation runs was 1.0017 ± 0.0005. This was in good agreement with the average of the corresponding MONK 8A benchmarks of 1.0016 ± 0.0005 performed by the computer code vendor.

Thirty Urenco criticality calculations were selected for verification. The average of the Urenco results documented for the thirty cases used for comparison in this report is 0.8764. The average of the verification runs is 0.8744 which is in good agreement with the Urenco results.

The purpose of Revision 1 of this report is to expand and reformat the report to add more detail to ensure that the report addresses all of the commitments made in Chapter 5 of the National Enrichment Facility Safety Analysis Report (Reference 11)

Two specific items that have been added to the report are the description of the Area of Applicability (AOA) and determination of the Upper Safety Limit (USL).

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

1 Introduction

1.1 Purpose

The purpose of this report is to validate the criticality codes and determine the Upper Safety Limit (USL) to be used for performing nuclear criticality safety calculations and analyses of the National Enrichment Facility (NEF).

1.2 Scope

The scope of this report is limited to the validation of the MONK8A Monte Carlo computer code and JEF 2.2 data library and the verification of criticality calculations performed for the NEF.

1.3 Applicability

The area of applicability (AOA) is identified to cover the entire range of activities in the plant. Any accumulation of uranium is taken to be in the form of a uranyl fluoride / water mixture.

1.4 Background

1.4.1 Overall NEF Design

The plant is designed to separate a feed stream containing the naturally occurring proportions of uranium isotopes into a product stream - enriched in the uranium-235 (^{235}U) isotope and a tails stream - depleted in the ^{235}U isotope. The NEF will be constructed on a LES site and licensed by the U.S. Nuclear Regulatory Commission (NRC) under Title 10 Code of Federal Regulations (CFR) Part 70. The facility is designed to applicable U.S. codes and standards and operated by LES.

1.4.2 Regulatory Requirements

10 CFR 70.61 requires that "under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety." In order to comply with this requirement, NEF Safety Analysis Report (SAR) Section 5.2.1.5 (Reference 11) requires a validation report that (1) demonstrates the adequacy of the margin of subcriticality for safety by assuring that the margin is large compared to the uncertainty in the calculated value of k_{eff}, (2) determines the areas of applicability (AOAs) and use of the code within the AOA such that calculations of k_{eff} are based on a set of variables whose values lie in a range for which the methodology used to determine k_{eff} has been validated, and (3) includes justification for extending the AOA by using trends in the bias, i.e., demonstrates that trends in the bias support the extension of the methodology to areas outside the AOAs.

NUREG 1520 (Reference 2) Section 5.4.3.4.1 (8), which is incorporated by reference in SAR Section 5.2.1.5, further states that the validation report should contain:

a) A description of the theory of the methodology that is sufficiently detailed and clear to allow understanding of the methodology and independent duplication of results.
b) A description of the area or areas of applicability that identifies the range of values for which valid results have been obtained for the parameters used in the methodology. In accordance with the provisions in ANSI/ANS-8.1-1983, any extrapolation beyond the area or areas of applicability should be supported by established mathematical methodology.
c) A description of the use of pertinent computer codes, assumptions, and techniques in the methodology.
d) A description of the proper functioning of the mathematical operations in the methodology (e.g., a description of mathematical testing).
e) A description of the data used in the methodology, showing that the data were based on reliable experimental measurements.
f) A description of the plant-specific benchmark experiments and the data derived there from that were used for validating the methodology.
g) A description of the bias, uncertainty in the bias, uncertainty in the methodology, uncertainty in the data, uncertainty in the benchmark experiments, and margin of subcriticality for safety, as well as the basis for these items, as they are used in the methodology. If the bias is determined to be advantageous to the applicant, the applicant shall use a bias of 0.0 (e.g., in a critical experiment where the k_{eff} is known to be 1.00 and the code calculates 1.02, the applicant cannot use a bias of 0.02 to allow calculations to be made above 1.00).
h) A description of the software and hardware that will use the methodology.
i) A description of the verification process and results.

In addition, SAR Section 5.2.1.1 requires the validation report to meet the LES commitments to ANSI/ANS 8.1-1998 and include details of validation that state computer codes used, operations, recipes for choosing code options (where applicable), cross section sets, and any numerical parameters necessary to describe the input.

These requirements are addressed in the following sections of this report.

2 Calculational Method

The MONK 8A code package is the computational code used for NEF criticality analyses. The code package is available through Serco Assurance. The MONK 8A code package is installed and verified on the Framatome-ANP Personal Computer (FANP PC) hardware platform.

MONK 8A is a powerful Monte Carlo tool for nuclear criticality safety analysis. The advanced geometry modeling capability and detailed continuous energy collision modeling treatments provide realistic three-dimensional models for an accurate simulation of neutronics behavior to provide the best estimate neutron multiplication factor, k-effective. Complex configurations can be simply modeled and verified. Additionally, Monk 8A has demonstrable accuracy over a wide range of applications. The NEF criticality analyses are performed using MONK 8A and the JEF 2.2 data library. Specifically, the data library files listed in Table 2-1 were used for the MONK 8A validation and verification runs. These files were provide by the computer code vendor, Serco, and are stored on the FANP PC. The MATCDB data file is used for material specification. This datafile is a database of composition of standard materials. The DICE datafile is used for determining cross sections. The datafile is a point energy neutron library. The THERM datafile is also used for determining cross sections. This datafile is the thermal library file that must be used with DICE when hydrogen bound in water or polythene is present.

Aside from the use of these data libraries no other code options need to be chosen. The rest of the input corresponds to building the proper geometry and material compositions to be used in the calculations. The input for the geometry and material composition is straight forward. Attachment 1A includes one input file for each of the 13 experiments.

Table 2-1 Data Libraries for Validation and Verification

Library Types	Library Names
MATCDB:	monk_matdbv2.dat
DICE:	dice96j2v5.dat
THERM:	therm96j2v2.dat

3 Criticality Code Validation Methodology

In order to establish that a system or process will be subcritical under all normal and credible abnormal conditions, it is necessary to establish acceptable subcritical limits for the operation and then show the proposed operation will not exceed those values.

The validation process involves three primary steps. The first step involves the procurement, installation, and verification of the criticality software on a specific computer platform. For the NEF, the MONK 8A code package was procured, installed and verified on the FANP PC hardware platform. A label is placed on the FANP PC indicating that it is a computer used for QA condition for Nuclear Safety related activities and that the configuration cannot be changed without authorization. This computer is a standalone computer where no automatic updates are allowed to occur to the operating system. This process ensures that the computer configuration

remains the same as used for the validation. This step is followed by the validation of the criticality software, which is the purpose of this report. The final step involves the Nuclear Criticality Safety Analyses (NCSA) calculations, which are presented in separate documents. A summary of the results from the NCSA calculations is provided in Section 7.

The criticality code validation methodology can be divided into four steps:

- Identify general NEF design applications
- Select applicable benchmark experiments for the AOA of interest.
- Model and calculate k_{eff} values of selected critical benchmark experiments
- Perform statistical analysis of results to determine computational bias and USL.

The first step is to identify the NEF design applications and key parameters associated with the normal and upset design conditions. Table 3-1 lists key parameters for the NEF.

The second step involves several sub steps. First, based on the key parameters, the AOA and expected range of the key parameter are identified. ANSI/ANS-8.1 defines the AOA as "the limiting range of material composition, geometric arrangements, neutron energy spectra, and other relevant parameters (such as heterogeneity, leakage interaction, absorption, etc.) within which the bias of a computational method is established." The NEF has only one AOA that covers a uranyl fluoride/water mixture. The AOA is presented in Section 4. After identifying the AOA, a set of critical benchmark experiments is selected. Benchmark experiments for the AOA are selected from the references listed in the International Handbook of Evaluated Criticality Safety Benchmark Experiments (Reference 4). A description of all relevant experiments used is provided in Section 5.

The third step involves modeling the critical experiments and calculating the k_{eff} values of the selected critical benchmark experiments. Attachment 1C presents the calculated results.

The final step involves the statistical analysis of the results in order to calculate the computational bias and USL. Section 6 presents the computational bias and USL results.

Another important piece of the validation methodology is the conservative assumptions used by the Nuclear Criticality Safety Engineer in performing NCSA. These conservative assumptions lead to added conservatism in the methodology. This conservatism is important when determining the proper amount of administrative margin that is required. These modeling conservatisms are discussed in Section 3.7.

3.1 MONK 8A Cases

ANSI/ANS-8.1-1998 requires a determination of the calculational bias by "correlating the results of critical and exponential experiments with results obtained for these same systems by the calculational method being validated." The correlation must be sufficient to determine if major changes in the bias can occur over the range of variables in the operation being analyzed. The standard permits the use of trends in the bias to justify extension of the AOA of the method outside the range of experimental conditions.

Calculational bias is the systematic difference between experimental data and calculated results. The simplest technique is to find the difference between the average value of the

calculated results of critical benchmark experiments and 1.0. This technique gives a constant bias over a defined range of applicability.

The recommended approach for establishing subcriticality based on numerical calculations of the neutron multiplication factor is prescribed in Appendix C of ANSI/ANS-8.1-1998. The criteria to establish subcriticality requires that for a design application (system or operation) to be considered as subcritical, the calculated multiplication factor for the system, k_s, must be less than or equal to an established maximum allowed multiplication factor based on benchmark calculations and uncertainty terms that is:

$$k_s \leq k_c - \Delta k_s - \Delta k_c - \Delta k_m$$

where:

k_s	= the calculated allowable maximum multiplication factor, (k_{eff}) of the design application (system)
k_c	= the mean k_{eff} value resulting from the calculation of benchmark critical experiments using a specific calculation method and data
Δk_s	= the uncertainty in the value of k_s
Δk_c	= the uncertainty in the value of k_c
Δk_m	= the administrative margin to ensure subcriticality.

Sources of uncertainty that determine Δk_s include:

- Statistical and/or convergence uncertainties
- Material and fabrication tolerances
- Limitations in the geometric and/or material representations used.

Sources of uncertainty that determine Δk_c include:

- Uncertainties in critical experiments
- Statistical and/or convergence uncertainties in the computation
- Extrapolation outside of the range of experimental data
- Limitations in the geometric and/or material representations used.

An assurance of subcriticality requires the determination of an acceptable margin based on known biases and uncertainties. The USL is defined as the upper bound for an acceptable calculation.

Critical benchmark experiments used to determine calculational bias (β) should be similar in composition, configuration, and nuclear characteristics to the system under examination. β is related to k_c as follows:

$$\beta = k_c - 1$$
$$\Delta\beta = \Delta k_c$$

Using this definition of bias, the condition for subcriticality is rewritten as:

$$k_s + \Delta k_s \leq 1 - \Delta k_m + \beta - \Delta\beta$$

A system is acceptably subcritical if a calculated k_{eff} plus calculational uncertainties lies at or below the USL.

$k_s + \Delta k_s \leq USL$

The USL can be written as:

$USL = 1 - \Delta k_m + \beta - \Delta\beta$

Bias is negative if $k_c < 1$ and positive if $k_c > 1$. For conservatism, a positive bias is set equal to zero for the purpose of defining the USL. $\Delta\beta$ is determined at the 95% confidence level for the NEF.

The USL takes into account bias, uncertainties, and administrative and/or statistical margins such that the calculated configuration will be subcritical with a high degree of confidence.

β is related to system parameters and may not be constant over the range of a parameter of interest. If k_{eff} values for benchmark experiments vary as a function of a system parameter, such as enrichment or degree of moderation, then β can be determined from a best fit as a function of the parameter upon which it is dependent. Extrapolation outside the range of validation must take into account trends in the bias.

Both $\Delta\beta$ and β can vary with a given parameter, and the USL is typically expressed as a function of the parameter. Normally, the most important system parameter that affects bias is the degree of moderation of the neutrons. This parameter can be expressed as moderator-to-fuel atomic ratio (H/U ratio).

In general, the bias can be broken down into components caused by system modeling error, code modeling inaccuracies, cross-sectional inaccuracies, etc. Bias associated with individual inaccuracies is usually combined into a total bias to represent the combined effect from all sources that prevent code and cross-sections from calculating the experimental value of k_{eff}.

One or two calculations are insufficient to determine calculational bias. In practice, it is necessary to determine the "average bias" for a group of experiments. A statistical analysis of the variation of biases around this average value is used to establish an uncertainty associated with the bias value when it is applied to a future calculation of a similar critical system. The lower limit of this band of uncertainty establishes an upper bound for which a future calculation of k_{eff} for a similar critical system can be considered subcritical with a high degree of confidence.

NUREG/CR-6698 (Reference 8) describes two statistical methods for the determination of an USL from the bias and uncertainty terms associated with the calculation of criticality. The first method is the single sided tolerance band and the second method is the single-sided tolerance limit. Both methods assume that the distribution of data points is normal. The following discussion of each method in Section 3.2 and 3.3 is taken from NUREG/CR-6698.

3.2 USL Method 1: Single-Sided Tolerance Band

When a relationship between a calculated k_{eff} and an independent variable can be determined, a one-sided lower tolerance band is used. This is a conservative method that provides a fitted curve above which the true population of k_{eff} is expected to lie. The tolerance band equation is actually a calibration curve relation. This was selected because it was anticipated that a given tolerance band would be used multiple times to predict bias. Other typical predictors, such as a single future value, can only be used for a single future prediction to ensure the degree of confidence desired.

The equation for the one-sided lower tolerance band is

$$K_L = K_{fit(x)} - S_{p_{fit}} \left\{ \sqrt{2F_a^{(2,n-2)} \left[\frac{1}{n} + \frac{(x-\bar{x})^2}{\sum (x_i - \bar{x})^2} \right]} + z_{2P-1} \sqrt{\frac{(n-2)}{\chi^2_{1-\gamma,n-2}}} \right\}$$

$K_{fit}(x)$ is the function derived in the trend analysis described in Section 3.5. Because a positive bias may be nonconservative, the equation below must be used for all values of x where $K_{fit}(x) > 1$.

$$K_L = 1 - S_{p_{fit}} \left\{ \sqrt{2F_a^{(2,n-2)} \left[\frac{1}{n} + \frac{(x-\bar{x})^2}{\sum (x_i - \bar{x})^2} \right]} + z_{2P-1} \sqrt{\frac{(n-2)}{\chi^2_{1-\gamma,n-2}}} \right\}$$

where:

p	=the desired confidence (0.95)
$F_a^{(fit,n-2)}$	=the F distribution percentile with degree of fit, n-2 degrees of freedom. The degree of fit is 2 for a linear fit.
n	=the number of critical experiments k_{eff} values
x	= the independent fit variable
x_i	=the independent parameter in the data set corresponding to the "i[th]" K_{eff} value
$\bar{x}$	= the weighted mean of the independent variables
z_{2P-1}	=the symmetric percentile of the Gaussian or normal distribution that contains the P fraction
γ	$= \frac{1-p}{2}$
$\chi^2_{1-\gamma,n-2}$	=the upper Chi-square percentile.

For a weighted analysis:

$$\sum (x_i - \bar{x})^2 = \frac{\sum \frac{1}{\sigma_i^2}(x_i - \bar{x})^2}{\frac{1}{n}\sum \frac{1}{\sigma_i^2}}$$

$$\bar{x} = \frac{\sum \frac{1}{\sigma_i^2} x_i}{\sum \frac{1}{\sigma_i^2}}$$

$$S_{P_{fit}} = \sqrt{s_{fit}^2 + \bar{\sigma}^2}$$

where:

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

and

$$S_{fit}^2 = \frac{\frac{1}{n-2}\sum \left\{ \frac{1}{\sigma_i^2} \left[k_{eff_i} - K_{fit}(x_i) \right]^2 \right\}}{\frac{1}{n}\sum \frac{1}{\sigma_i^2}}$$

3.3 USL Method 2: Single-Sided Tolerance Limit

A weighted single-sided lower tolerance limit (K_L) is a single lower limit above which a defined fraction of the true population of k_{eff} is expected to lie, with a prescribed confidence and within the area of applicability. The term "weighted" refers to a specific statistical technique where the uncertainties in the data are used to weight the data point. Data with high uncertainties will have less "weight" than data with small uncertainties.

A lower tolerance limit should be used when there are no trends apparent in the critical experiment results. Use of this limit requires the critical experiment results to have a normal statistical distribution. If the data does not have a normal statistical distribution, a non-parametric statistical treatment must be used.

Lower tolerance limits, at a minimum, should be calculated with a 95% confidence that 95% of the data lies above K_L. This is quantified by using the single-sided lower tolerance factors (U) provided in Table 3-2. For cases where more than 50 data samples are available, the tolerance factor equivalent to 50 samples can be used as a conservative number.

This method cannot be used to extrapolate the area of applicability beyond the limits of the validation data.

The one-sided lower tolerance limit is defined by the equation:

$$K_L = \overline{k_{eff}} - US_P$$

If $\overline{k_{eff}} \geq 1$, then $K_L = 1 - US_P$

where:

Sp = square root (pooled variance)
U = one-sided lower tolerance factor

Then USL = $K_L - \Delta_{sm} - \Delta_{AOA}$

where, Δ_{sm} is the margin of subcriticality and Δ_{AOA} is an additional margin of subcriticality that may be necessary as a result of extrapolation of the area of applicability. If extrapolations are not made to the area of applicability, Δ_{AOA} is zero.

3.4 Nonparametric Statistical Treatment

NUREG/CR-6698 states that data that do not follow a normal distribution can be analyzed by non-parametric techniques. The analysis results in a determination of the degree of confidence that a fraction of the true population of data lies above the smallest observed value. The more data that is present in the sample, the higher the degree of confidence.

The following equation determines the percent confidence that a fraction of the population is above the lowest observed value:

$$\beta = 1 - \sum_{j=0}^{m-1} \frac{n!}{j!(n-j)!}(1-q)^j q^{n-j}$$

where:

q = the desired population fraction (normally 0.95)
n = the number of data in one data sample
m = the rank order indexing from the smallest sample to the largest (m=1 for the smallest sample; m=2 for the second smallest sample, etc.)

For a desired population fraction of 95% and a rank order of 1 (the smallest data sample), the equation reduces to:

$$\beta = 1 - q^n = 1 - 0.95^n$$

This information is used to determine K_L, the combination of bias and bias uncertainty.

For non-parametric data analysis, K_L is determined by:

K_L = Smallest k_{eff} value – Uncertainty for Smallest K_{eff} – Non-parametric Margin (NPM)

Where:

NPM = Non-parametric margin. This non-parametric margin is added to account for small sample size and it is obtained from Table 3-3 below.

Smallest k_{eff} value = the lowest calculated value in the data sample.

If the smallest k_{eff} value is greater than 1, then the non-parametric K_L becomes:

$K_L = 1 - S_P - NPM$

where:

S_P = Square root of the pooled variance

Then $USL = K_L - \Delta_{sm} - \Delta_{AOA}$

where, Δ_{sm} is the margin of subcriticality and Δ_{AOA} is an additional margin of subcriticality that may be necessary as a result of extrapolation of the AOA. If extrapolations are not made to the AOA, Δ_{AOA} is zero.

3.5 Trend Analysis

Trends are determined through the use of regression fits to the calculated results. In many instances a linear fit is sufficient to determine a trend in the bias. The use of weighted or unweighted least squares is a means for determining the fit of a function. In the equations below, "x" is the independent variable representing some parameter (e.g., $H/^{235}U$). The variable "y" represents k_{eff}. Variables "a" and "b" are coefficients for the function.

The equations used to produced a weighted fit of a straight line to a set of data are given below.

$$Y(x) = a + bx$$

$$a = \frac{1}{\Delta}\left(\sum\frac{x_i^2}{\sigma_i^2}\sum\frac{y_i}{\sigma_i^2} - \sum\frac{x_i^2}{\sigma_i^2}\sum\frac{x_i y}{\sigma_i^2}\right)$$

$$b = \frac{1}{\Delta}\left(\sum\frac{1}{\sigma_i^2}\sum\frac{x_i y}{\sigma_i^2} - \sum\frac{x_i}{\sigma_i^2}\sum\frac{y_i}{\sigma_i^2}\right)$$

$$\Delta = \sum\frac{1}{\sigma_i^2}\sum\frac{x_i^2}{\sigma_i^2} - \left(\sum\frac{x_i}{\sigma_i^2}\right)^2$$

3.6 Uncertainties

Uncertainties, as used in this report, refer to the uncertainty in k_{eff} associated with experimental unknowns or assumptions and the uncertainty values associated with Monte Carlo analyses.

Experimental uncertainty (σ_e) – Modeling of validation experiments frequently result in assumptions about experimental conditions. In addition, experimental uncertainties (such as measurements tolerances) influence the development of a computer model.

Statistical uncertainty (σ_s) – Monte Carlo calculation techniques result in a statistical uncertainty associated with the actual calculation. This type of uncertainty is dependent upon many factors, including number of neutron generations performed, variance reduction techniques employed, and problem geometry. For this document, σ_s refers to the statistical Monte Carlo uncertainty associated with the computer modeled validation experiment.

Total uncertainty – This is the total uncertainty associated with a calculated k_{eff} on a benchmark experiment. The total uncertainty for an individual benchmark is the combined error of the experimental and statistical uncertainties:

$$\sigma_t = ((\sigma_{e,i})^2 + (\sigma_{s,i})^2)^{1/2}$$

where the subscript (i) refers to an individual benchmark calculation.

3.7 Conservatism in the Calculational Models

The NEF NCSAs use several conservative assumptions in the modeling. These conservatisms are as follows.

For most components that form part of the centrifuge plant or are connected to it, any accumulation of uranium is taken to be in the form of a uranyl fluoride/water mixture at a maximum H/U atomic ratio of 7 (exceptions are product cylinders, vacuum pumps and UF_6 sample bottles.). This is based on the assumption that significant quantities of moderated uranium could accumulate by reaction between UF_6 and moisture in air leaking into the plant. Due to the high vacuum requirements of a centrifuge plant, inleakage is controlled at very low levels and thus the condition assumed above represents an abnormal condition. The H/U ratio of 7 assumption is conservative and the H/U ratio is not expected to be higher than 7. Higher H/U ratios due to excessive air in-leakage are precluded since the condition would cause a loss of vacuum which in turn would cause the affected centrifuges to crash and the enrichment process to stop. In case of oils, UF_6 pumps and vacuum pumps use a fully fluorinated PFPE (perfluorinated polyether) type lubricant. Mixtures of UF_6 and PFPE oil (also referred to as Fomblin oil) would be a less pessimistic case than the uranyl fluoride / water mixture considered since maximum hydrogen fluoride (HF) solubility in PFPE is only ~ 0.1% by weight (Reference 12).

A uranyl fluoride water system is the worst combination of materials that can occur in a Urenco enrichment plant with regard to criticality safety. In addition, uranium compounds with alumina, Fomblin oil or active carbon are less reactive than a uranyl fluoride water system. Alumina and Fomblin oil systems are less reactive because they contain no hydrogen to act as a moderating material, and active carbon systems are less reactive because carbon/graphite is a less efficient moderator than hydrogen. In addition, the uranyl fluoride water system is considered to be much worst than any normal non-moderated system. Therefore, the uranyl fluoride water system is the only system that needs to be included in the benchmark. Additional compounds are used in the benchmark experiments. The justification for using these additional compounds

is discussed in Section 5.1.

With exception of the product cylinders, where moderation is used as a control, either optimum moderation or worst case H/U ratio is assumed when performing criticality safety analysis.

Where appropriate, spurious reflection due to walls, fixtures, personnel, etc. has been accounted for by considering 2.5 cm of water reflection around vessels.

The NEF will operate with 5.0 $^{w}/_{o}$ ^{235}U enrichment limit. However, the nuclear criticality safety calculations used an enrichment of 6.0 $^{w}/_{o}$ ^{235}U. This assumption provides additional conservatism for plant design.

3.8 Application of the USL

For the NEF, the benchmark cases do not fall within a normal distribution. Therefore, it is appropriate to arrive at the USL using the non-parametric technique discussed in Section 3.4. The other statistical techniques are discussed in this report for completeness.

The USL is valid over the range of the parameters in the set of calculations used to determine the USL, with the exception of the enrichment value associated with the Contingency Dump System. ANSI/ANS-8.1 allows the range of applicability to be extended beyond this range by extrapolating the trends established for the bias. No precise guidelines are specified for the limits of extrapolation. Thus, engineering judgment should be applied when extrapolating beyond the range of the parameter bounds. For the Contingency Dump System, the trend analysis discussed in Section 3.5 is used to determine the equation of the line that is used to properly account for the additional uncertainty to be applied to the USL. This additional uncertainty is needed due to the enrichment value associated with the Contingency Dump System being beyond the range of the parameter bounds.

Table 3-1 Characteristics/Key Parameters of the NEF Systems

Parameter	Fissile Material Physical/Chemical Form	Isotopic Composition of Fissile Material	Type of Moderation Materials	Anticipated Reflector Materials	Typical Geometry
	Uranyl fluoride	$\leq 5\ ^{w}/_{o}\ ^{235}U$	Hydrogen Fomblin Oil Carbon	Water Concrete	Spheres Cylinders Slabs

Table 3-2 Single-Sided Lower Tolerance Factors

# Experiments (n)	U
10	2.911
11	2.815
12	2.736
13	2.670
14	2.614
15	2.566
16	2.523
17	2.486
18	2.453
19	2.423
20	2.396
21	2.371
22	2.350
23	2.329
24	2.309
25	2.292
30	2.220
35	2.166
40	2.126
45	2.092
50	2.065

Table 3-3 Non–Parametric Margins

Degree of Confidence for 95% of the Population	Non-parametric Margin (NPM)
>90%	0.0
>80%	0.01
>70%	0.02
>60%	0.03
>50%	0.04
>40%	0.05
≤40%	Additional data needed. (This corresponds to less than 10 data points)

4 NEF Design Application Classification

The NEF has only one area of applicability for the entire plant. The AOA covers a uranyl fluoride/water mixture.

4.1 Design Application – Uranyl Fluoride/Water Mixture

A uranyl fluoride water system is the worst combination of materials that can occur in a Urenco enrichment plant with regard to criticality safety. In addition, uranium compounds with alumina, Fomblin oil or active carbon are less reactive than a uranyl fluoride water system. Alumina and Fomblin oil systems are less reactive because they contain no hydrogen to act as a moderating material, and active carbon systems are less reactive because carbon/graphite is a less efficient moderator than hydrogen. In addition, the uranyl fluoride water system is considered to be much worst than any normal non-moderated system. Therefore, the uranyl fluoride water system is the only system that needs to be included in the benchmark. Additional compounds are used in the benchmark experiments. The justification for using these additional compounds is discussed in Section 5.1.

Table 4-1 summarizes the anticipated characteristics for the design of the NEF systems involving uranic material. The systems are assumed to contain a uranyl fluoride/water mixture. The table provides the relevant parameters (i.e., chemical form, isotopics, moderator to fuel atomic ratio) for the application.

Table 4-1 Anticipated Characteristics for the Design Application Involving Uranyl Fluoride

Room	Components Modeled	Chemical Form	Isotopics	Hydrogen/ Uranium Ratio
Main Separations Plant, except Contingency Dump System	Product Cylinders Product Cold Traps Pumps Pipe work Vacuum Cleaners	Uranyl fluoride water mixture UF_4/CH_2 (oil)	5 $^{w}/_{o}$ ^{235}U	7 to 21
Contingency Dump System	Sodium Fluoride Traps	$UO_2F_2.3.5H_2O$	1.5 $^{w}/_{o}$ ^{235}U	7
Technical Services Building	Waste Containers Product Traps Hex Bottles Pumps Vacuum Cleaner	UF_4/CH_2 UF_6/Carbon UF_6HF $UO_2F_2.3.5H_2O$	5 $^{w}/_{o}$ ^{235}U	1 to 32

5 Benchmark Experiments

5.1 Uranyl Fluoride/ Water Mixture

Thirteen plant specific benchmark experiments, consisting of 80 critical configurations, with uranyl solutions are selected from the International Handbook of Evaluated Criticality Safety Benchmark Experiments (Reference 4) to provide a good statistical base. All of the experiments have a k_{eff} =1, with experimental uncertainties from 0.0008 to 0.0063. Therefore, all experiments used are adequate and come from a reliable source. Attachment 1A contains a sample MONK 8A input for each of the thirteen plant specific benchmark experiments. Attachment 1B is a listing of critical experiment parameters used in the benchmark.

The list of the experiments is provided in Table 5-1. Detail descriptions of the criticality experiments were extracted from the International Handbook of Evaluated Criticality Safety Benchmark Experiments and are tabulated in Table 5-2. A description of the key parameters of these experiments is shown in Table 5-3 along side the key parameters used in the NEF NCSA.

Attachment 1A shows a sample MONK 8A input for each of the 13 benchmark experiments. Also shown in Attachment 1B are the key input parameters used in the benchmark.

As shown in Table 5-3, the resulting validated AOA contain the corresponding key parameters of the NEF NCSA for which the MONK 8A code will be used to determine reactivity, with the exception of the enrichment value for NCSA of the Contingency Dump System. The NCSA for the NEF uses the chemical form uranyl fluoride. In addition, the uranyl fluoride water system is considered to be much worst than any normal non-moderated system. Therefore, the uranyl fluoride water system is the only system that needs to be included in the benchmark. The chosen benchmark cases have uranyl nitrate and uranium oxyfluoride fuel solution cases. Uranyl fluoride and uranium oxyfluoride are both the chemical form UO_2F_2. Therefore, uranyl fluoride is adequately covered in the benchmark. The benchmark also includes many uranyl nitrate cases. The reason for including the uranyl nitrate cases is to include as many possible in-solution critical experiments as possible. The statistics for the uranyl nitrate cases were compared against the statistics for the uranyl oxyfluoride cases. The average and standard deviation of the cases are similar (i.e., 1.0016±0.0045 for the uranyl nitrate cases compared to 1.0018±0.0038 for the uranyl oxyfluoride cases). In addition, a uranyl nitrate case is responsible for lowest k_{eff} calculated in the benchmark. Since the benchmark results are non-normal, the uranyl nitrate case causes the USL to be lower which is conservative. Therefore, these benchmark cases were included. The H/U ratio varies from 1 to 32 for the NEF NCSA. and ranges from 0.103 to 1378 for the benchmark cases. Therefore the H/U ratio for the NEF NCSA is bounded by the benchmark cases. The NEF NCSA assumes that the enrichment is at 6 $^w/_o$, except for NCSA associated with the Contingency Dump System. For the Contingency Dump System, the NEF NCSA assumes that the enrichment is at 1.5 $^w/_o$. The benchmark cases range from 4.89 to 93.65 $^w/_o$. Therefore, the enrichment used in the NEF NSCA for systems and components other than those associated with the Contingency Dump System is also bounded by the benchmark cases. For the Contingency Dump System, extrapolation beyond the AOA is required (i.e., from 4.89 $^w/_o$ to 1.5 $^w/_o$). The high enrichment cases include a wide enrichment range and are included to provide as many in solution critical experiments as possible.

The resulting validated AOA contains the corresponding key parameters of the anticipated NEF NCSA for which the MONK 8A code will be used to determine reactivity, except for the

enrichment parameter associated with the Contingency Dump System. As such, no extrapolation beyond the AOA is required for use of the MONK 8A code to determine the reactivity of systems or components not associated with the Contingency Dump System. For use of the MONK 8A code to determine the reactivity for systems or components with an assumed enrichment of 1.5 $^{w}/_{o}$ (i.e., the Contingency Dump System), extrapolation beyond the AOA is required and additional AOA margin shall be assigned as reflected in Section 6.

If, in the future, a parameter value for design applications falls outside of the current validated AOA for systems or components not associated with the Contingency Dump System or falls outside the current extrapolated AOA associated with the Contingency Dump System, LES shall revise the validation report to identify additional AOA margin and provide a letter to the NRC describing the change prior to using results from calculations with a parameter value that falls outside the current validated AOA (or current extrapolated AOA in the case of the Contingency Dump System) in NCSAs.

Table 5-1 Uranium Solution Experiments Used for Validation

MONK 8A Case Set	Case Description	Number of Experiments	Handbook Reference (Reference 4)
13	High-enriched uranyl nitrate solutions at various H:U ratios (93.17 w/o ^{235}U)	12	HEU-SOL-THERM-002 HEU-SOL-THERM-003
23	Uranyl nitrate solution (~ 95 w/o enriched)	5	HEU-SOL-THERM-013 NS&E
35	High-enriched uranyl nitrate solutions (U concentration from 20-700 g/L)	11	HEU-SOL-THERM-009 - HEU-SOL-THERM-012
43	Low-enriched uranyl nitrate solutions	3	LEU-SOL-THERM-002
51	Low-enriched uranium solutions (new STACY experiments)	7	LEU-SOL-THERM-004
63	Boron carbide absorber rods in uranyl nitrate (5.6 w/o enriched)	3	LEU-SOL-THERM-005
67	Highly enriched uranyl nitrate solution with a concentration range between 59.65 and 334.66 g U/L	10	HEU-SOL-THERM-001
68	Highly enriched uranyl fluoride/heavy water solution with a concentration range between 60 and 679 g U/L and a heavy water reflector	6	HEU-SOL-THERM-004
71	STACY: 28 cm thick slabs of 10 w/o enriched uranyl nitrate solutions, water Reflected	7	LEU-SOL-THERM-016
80	STACY: Unreflected 10 w/o enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank	5	LEU-SOL-THERM-007
81	STACY: Concrete reflected 10 w/o enriched uranyl nitrate solution reflected by concrete	4	LEU-SOL-THERM-008
84	STACY: Borated concrete reflected 10 w/o enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank	3	LEU-SOL-THERM-009
85	STACY: Polyethylene reflected 10 w/o enriched uranyl nitrate solution in a 60 cm diameter cylindrical tank	4	LEU-SOL-THERM-010

Table 5-2 Expanded Descriptions of the Criticality Experiments

Handbook Reference	Title	Short Description
HEU-SOL-THERM-002	Concrete Reflected Cylinders of Highly Enriched Solutions of Uranyl Nitrate	Fourteen critical experiments, each involving a single reflected tank of highly enriched uranyl nitrate, were performed at the Rocky Flats Plant, which was operated at that time by Rockwell International. The critical height for each experiment was determined by linear interpolation between slightly supercritical and slightly subcritical states. The tanks were cylindrical in shape and placed at different locations in a concrete reflector. Critical configurations had height-to-diameter ratios less than 1.2. Uranium concentrations varied between 59.65 and 334.77 grams of uranium per liter (93.172 $^{w}/_{o}$ ^{235}U). [See NOTE 1]
HEU-SOL-THERM-003	Plexiglas Reflected Cylinders of Highly Enriched Solutions of Uranyl Nitrate	Nineteen critical experiments, each involving a single reflected tank of highly enriched uranyl nitrate, were performed at the Rocky Flats Plant, which was operated at that time by Rockwell International. The critical height for each experiment was determined by linear interpolation between slightly supercritical and slightly subcritical states. The tanks were cylindrical in shape and placed at different locations in a Plexiglas reflector. Critical configurations had height-to-diameter ratios less than 2.4. Uranium concentrations varied between 60.32 and 345.33 grams of uranium per liter (93.172 $^{w}/_{o}$ ^{235}U). [See NOTE 1]
HEU-SOL-THERM-013	Unreflected 174 Liter Spheres of Enriched Uranium Nitrate Solutions	The four measurements included in this evaluation are part of a series of experiments performed in the 1950's at the Oak Ridge National Laboratory with highly enriched (93.18 $^{w}/_{o}$ ^{235}U) uranium. Critical experiment measurements were made with uranyl nitrate solutions poisoned with boric acid in an unreflected 27.24-inch-diameter sphere (174 liters). The sphere was fabricated of 0.32-cm-thick 1100 aluminum. [See NOTE 2]
HEU-SOL-THERM-009	Water-Reflected 6.4-Liter Spheres of Enriched Uranium Oxyfluoride Solutions	The four water-reflected spheres included in this evaluation are part of a series of experiments performed in the 1950's at the Oak Ridge National Laboratory with highly enriched uranium. Critical experiment measurements were made with uranium oxyfluoride (UO_2F_2) solutions at various uranium concentrations (93.17-93.19 $^{w}/_{o}$ ^{235}U) in two water-reflected spheres nominally 9 inches in diameter (6.4 liters). Spherical reactors with nominal inner diameter of 9 inches were fabricated of aluminum and surrounded by an effectively infinite water reflector. The spheres were supported in the water reflector only by the top and bottom overflow and feed tubes, respectively.

Table 5-2 Expanded Descriptions of the Criticality Experiments

Handbook Reference	Title	Short Description
HEU-SOL-THERM-010	Water-Reflected 9.7-Liter Spheres of Enriched Uranium Oxyfluoride Solutions	The four water-reflected spheres included in this evaluation are part of a series of experiments performed in the 1950's at the Oak Ridge National Laboratory with highly enriched uranium. Critical experiment measurements were made with uranium oxyfluoride solutions at temperatures and uranium concentrations (93.17-93.19 $^{w}/_{o}$ ^{235}U). A spherical reactor with nominal inner diameter of 26.4 cm (9.7 liters) was fabricated of aluminum and surrounded by an effectively infinite water reflector. The sphere was supported in the water reflector only by the top and bottom overflow and feed tubes, respectively.
HEU-SOL-THERM-011	Water-Reflected 17-Liter Spheres of Enriched Uranium Oxyfluoride Solutions	The two water-reflected spheres included in this evaluation are part of a series of measurements performed in the 1950's at the Oak Ridge National Laboratory with highly enriched uranium (93.2 $^{w}/_{o}$$^{235}U$). Critical experiment measurements were made with uranium oxyfluoride (UO_2F_2) solutions in a water-reflected 32-cm-inner-diameter (17-liter) sphere with an aluminum wall 1.27 mm thick. To provide 19 cm of water as an effectively infinite neutron reflector, the sphere was mounted in a cylinder of appropriate dimensions. The sphere was supported in the water reflector only by the top and bottom overflow and feed tubes, respectively.
HEU-SOL-THERM-012	Water-Reflected 91-Liter Sphere of Enriched Uranium Oxyfluoride Solution	This water-reflected sphere is part of a series of experiments performed in the 1950's at the Oak Ridge National Laboratory with highly enriched uranium (93.2 $^{w}/_{o}$ ^{235}U). This measurement was made with a uranium oxyfluoride (UO_2F_2) solution in a 27.9-cm inner radius (91 liters) water-reflected sphere. The sphere was fabricated of 0.20-cm-thick 1100 aluminum and surrounded by an effectively infinite water reflector.

Table 5-2 Expanded Descriptions of the Criticality Experiments

Handbook Reference	Title	Short Description
LEU-SOL-THERM-002	174 Liter Spheres of Low Enriched (4.9%) Uranium Oxyfluoride Solutions	The three experiments included in this evaluation are part of a series of measurements performed in the 1950s at the Oak Ridge National Laboratory with low-enriched uranium (4.9 $^{w}/_{o}$ ^{235}U). Critical experiment measurements were made with uranium oxyfluoride (UO_2F_2) solutions in a 27.3-in-inner-diameter (174-liter) sphere with an aluminum wall 1/16 in. thick. The sphere was supported only by the top and bottom overflow and feed tubes, respectively. Three experiments are evaluated. One measurement was made in an unreflected sphere and two measurements were water reflected. To provide an effectively infinite neutron reflector for these two measurements, the sphere was mounted in a cylinder of appropriate dimensions.
LEU-SOL-THERM-004	STACY: Water-Reflected 10%-Enriched Uranyl Nitrate Solution in a 60-Cm-Diameter Cylindrical Tank	Seven critical experiments included in this evaluation are part of a series of experiments with the Static Experiment Critical Facility (STACY) performed in 1995 at the Nuclear Fuel Cycle Safety Engineering Research Facility in the Tokai Research Establishment of the Japan Atomic Energy Research Institute. In the first series of experiments using the water-reflected 60-cm-diameter and 150-cm-high cylindrical tank, seven sets of critical data were obtained. The uranium concentration of the fuel solution ranged from 225 to 310 gU/liter and the uranium enrichment was 10 $^{w}/_{o}$ ^{235}U. On the bottom, side, and top of the core tank was a thick water reflector.

Table 5-2 Expanded Descriptions of the Criticality Experiments

Handbook Reference	Title	Short Description
LEU-SOL-THERM-005	Boron Carbide Absorber Rods in Uranium (5.64% ^{235}U) Nitrate Solution	A large number of critical experiments with absorber elements of different types in uranium nitrate solution of different enrichments and concentrations were performed in 1961 - 1963 at the Solution Physical Facility of the Institute of Physics and Power Engineering (IPPE), Obninsk, Russia. The purpose of these experiments was to determine the effects of enrichment, concentration, geometry, neutron reflection, and type, diameter, number, and arrangement of absorber rods on the critical mass of light-water-moderated homogeneous uranyl nitrate solutions. The experiments included ones with a central boron carbide or cadmium rod, clusters of boron carbide rods, and triangular lattices of boron carbide rods in cylindrical tanks of different dimensions filled with solutions of uranyl nitrate. The three experiments included in this evaluation were performed with uranium enriched to 5.64 $^{w}/_{o}$ ^{235}U. Uranium nitrate solution with uranium concentration of 400.2 g/l was pumped into the core or inner tank, a stainless steel cylindrical tank with inner diameter 110 cm. One experiment was performed without absorber rods, another one with a central rod, and another one with a cluster of seven absorber rods arranged at the corners and center of a hexagon with a pitch of 31.8 cm, inserted in the center of the core tank. There was a thick side and bottom water reflector in these experiments.
HEU-SOL-THERM-001	Minimally Reflected Cylinders of Highly Enriched Solutions of Uranyl Nitrate	Ten critical experiments, each involving a tank of highly enriched uranyl nitrate (93.172 $^{w}/_{o}$ ^{235}U), were performed at the Rocky Flats Plant, which was operated at that time by Rockwell International. The critical height for each experiment was determined by linear interpolation between reactor periods of slightly supercritical and slightly subcritical states. The tanks were cylindrical in shape and suspended in the approximate center of a large room. Critical configurations had height to diameter ratios less than 1.2. Uranium concentration varied between 50 and 360 grams of uranium per liter.

Table 5-2 Expanded Descriptions of the Criticality Experiments

Handbook Reference	Title	Short Description
HEU-SOL-THERM-004	Reflected Uranyl-Fluoride Solutions in Heavy Water	In the early 1950's, a series of experiments was performed at the Los Alamos Scientific Laboratory to investigate critical parameters of enriched (93.65 $^{w}/_{o}$ ^{235}U) uranyl-fluoride (UO_2F_2) heavy-water solutions over a wide range of deuterium to ^{235}U atomic ratios. A total of 10 experiments were performed. Six experiments consisted of heavy-water reflected spheres of uranyl fluoride in which the atomic ratio of deuterium to ^{235}U ranged from 34 to 430. The remaining four assemblies were bare cylinders with deuterium to ^{235}U ratios ranging from 230 to 2080.
LEU-SOL-THERM-016	STACY: 28-cm-Thick Slabs of 10%-Enriched Uranyl Nitrate Solutions, Water-Reflected	The seven critical configurations included in this evaluation are part of a series of experiments with the Static Experiment Critical Facility (STACY) performed from 1997 to the summer of 1998 at the Nuclear Fuel Cycle Safety Engineering Research Facility (NUCEF) at the Tokai Research Establishment of the Japan Atomic Energy Research Institute (JAERI). Employing the 28-cm thick, 69-cm-wide slab core tank, a 10 $^{w}/_{o}$ -enriched uranyl nitrate solution was used in these experiments. The uranium concentration was adjusted, in stages, to values in the range of approximately 464 gU/l to 300 gU/l. The free nitric acid concentration ranged from 0.8 mol/l to 1.0 mol/l, approximately.
LEU-SOL-THERM-007	STACY: Unreflected 10%-Enriched Uranyl Nitrate Solution in a 60-cm-Diameter Cylindrical Tank	Five critical experiments included in this evaluation are part of a series of experiments with the Static Experiment Critical Facility (STACY) performed in 1995 at the Nuclear Fuel Cycle Safety Engineering Research Facility in the Tokai Research Establishment of the Japan Atomic Energy Research Institute. In the first series of experiments using the unreflected 60-cm diameter and 150-cm-high cylindrical tank, five sets of critical data were obtained. The uranium concentration of the fuel solution ranged from 242 to 313 gU/liter and the uranium enrichment was 10 $^{w}/_{o}$. The core tank was unreflected.

Table 5-2 Expanded Descriptions of the Criticality Experiments

Handbook Reference	Title	Short Description
LEU-SOL-THERM-008	STACY: 60-cm-Diameter Cylinders of 10%-Enriched Uranyl Nitrate Solutions Reflected with Concrete	Four critical configurations included in this evaluation are part of a series of experiments with the Static Experiment Critical Facility (STACY) performed in 1996 at the Nuclear Fuel Cycle Safety Engineering Research Facility (NUCEF) in the Tokai Research Establishment of the Japan Atomic Energy Research Institute (JAERI). Employing the 60-cm-diameter cylindrical core tank, a 10 $^{w}/_{o}$-enriched uranyl nitrate solution was used in these experiments. The uranium concentration and the free nitric-acid concentration were adjusted to approximately 240 g/l and 2.1 mol/l, respectively. Four concrete reflectors of different thicknesses, packed in annular tube-shaped containers, were prepared and arranged against the outer wall of the core tank.
LEU-SOL-THERM-009	STACY: 60-cm-Diameter Cylinders of 10%-Enriched Uranyl Nitrate Solutions Reflected with Borated Concrete	Three critical configurations included in this evaluation are part of a series of experiments with the Static Experiment Critical Facility (STACY) performed in 1996 at the Nuclear Fuel Cycle Safety Engineering Research Facility (NUCEF) in the Tokai Research Establishment of the Japan Atomic Energy Research Institute (JAERI). Employing the 60-cm-diameter cylindrical core tank, a 10 $^{w}/_{o}$-enriched uranyl nitrate solution was used in these experiments. The uranium concentration and the free nitric-acid concentration were adjusted to approximately 240 g/l and 2.1 mol/l, respectively. Three borated-concrete reflectors of different boron content, packed in annular tube-shaped containers, were prepared and arranged against the outer wall of the core tank.
LEU-SOL-THERM-010	STACY: 60-cm-Diameter Cylinders of 10%-Enriched Uranyl Nitrate Solutions Reflected with Polyethylene	Four critical configurations included in this evaluation are part of a series of experiments with the Static Experiment Critical Facility (STACY) performed in 1996 at the Nuclear Fuel Cycle Safety Engineering Research Facility (NUCEF) in the Tokai Research Establishment of the Japan Atomic Energy Research Institute (JAERI). Employing the 60-cm-diameter cylindrical core tank, a 10 $^{w}/_{o}$-enriched uranyl nitrate solution was used in these experiments. The uranium concentration and the free nitric-acid concentration were adjusted to approximately 240 g/l and 2.1 mol/l, respectively. Four thicknesses of reflectors, polyethylene blocks packed in annular tube-shaped containers, were prepared and arranged next to the outer wall of the core tank.

NOTE 1: The SAR (Reference 11) lists HEU-SOL-THERM-002 as the Handbook document for case 13. The twelve case 13 experiments are not all documented in HEU-SOL-THERM-002 in the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (Reference 4). Six of the experiments in case 13 use concrete reflectors and the other six use plastic reflectors. HEU-SOL-THERM-002 is for concrete reflectors and specifically documents experiments 2, 3, 7, 10, and 11. HEU-SOL-THERM-003 is for plastic reflectors and documents experiments 1, 4, 5, 8, 9, and 12. Experiment 6 has a concrete reflector but it is not in HEU-SOL-THERM-002. However, the configuration details for experiment 6 are documented in two source documents (References 9 and 10) used by HEU-SOL-THERM-002.

NOTE 2: HEU-SOL-THERM-013, from the *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (Reference 4), lists four experiments. A fifth experiment from the original Nuclear Science & Engineering (Reference 6) was included by Serco Assurance.

Table 5-3 Comparison of Key Parameters of NEF NCSA and Benchmark

	Chemical Form	Isotopics	Hydrogen/ Uranium Ratio
NEF Nuclear Criticality Safety Analysis, except Contingency Dump System	Uranyl fluoride	6 w/o ^{235}U	1 to 32
NEF Nuclear Criticality Safety Analysis, Contingency Dump System	Uranyl fluoride	1.5 w/o ^{235}U	7
Benchmark	Uranyl Nitrate Uranium Oxyfluoride	4.89 to 93.65 w/o ^{235}U	0.103 to 1378

6 Analysis of Validation Results

6.1 Uranyl Fluoride/Water Mixture

Eighty experiments are modeled with MONK 8A using the JEF2.2 data library on a PC platform. These experiments include the following geometries:

- Water reflected slabs,
- Water reflected sphere,
- Water reflected cylinder
- Heavy Water reflected spheres,
- Concrete reflected cylinder,
- Borated concrete reflected cylinder,
- Plexiglas reflected cylinder,
- Polyethylene reflected cylinder,
- Bare (unreflected) cylinder
- Bare (unreflected) sphere.

The calculated k_{eff} values, experimental uncertainties and calculational uncertainties (i.e., Monk Standard Deviation) are presented in Attachment 1C. Figure 6-1 shows the distribution of the calculated k_{eff} values. The results were analyzed statistically and, due to the inclusion of a broad but distinct range of enrichments, the results have been shown to be a non-normal distribution. Therefore, the non-parametric technique is applied to the data. The results are analyzed statistically using four trending parameters: Solution Density, $H/^{235}U$ ratio, ^{235}U enrichment, and Mean Cord Length.

The solution density goes from 1.026 to 1.930 g/cc, the $H/^{235}U$ ratio goes from 0.103 to 1378, the ^{235}U enrichment goes from 4.89 to 93.65 $^{w}/_{o}$ and the cord length goes from 7.67 to 81.35 cm. Table 6-1 summarizes the statistical results. Figures 6-2 through Figure 6-5 show the results graphically.

The minimum k_{eff} is from case80.01, with a value of 0.9928 and a total uncertainty of 0.0013. Since the sample size is 80, the non-parametric margin is 0.0 and provides for a 95% confidence that 95% of the population lies above the smallest observed value. As a result, the lower tolerance limit is as follows:

$$K_L = 0.9928-0.0013-0.0 = 0.9915.$$

The value of the administrative margin (Δ_{SM}) is set to 0.05. This value is considered to be adequate due to the following considerations.

- As reflected in Section 5.1,the benchmark experiments are similar to the actual applications.
- As reflected in Section 5.1, the number and quality of benchmark experiments used is high.

- The validation methodology described in Sections 3.1 though 3.8 is consistent with regulatory requirements and guidance and is considered to be adequate.
- There is conservatism in the calculation of the bias and its uncertainty using the methods described in Sections 3.1 through 3.8.

For use of the MONK 8A code to determine the reactivity of systems or components NOT associated with the Contingency Dump System, the AOA is NOT being extrapolated past the range of applicability; therefore the margin required to extrapolate a parameter beyond the area of applicability (Δ_{AOA}) is set to 0.0.

For the use of the MONK 8A code to determine the reactivity of system or components associated with the Contingency Dump System (i.e., systems or components with assumed enrichment of 1.5 $^w/_o$), extrapolation of the AOA is required with respect to enrichment (i.e., from 4.89 $^w/_o$ to 1.5 $^w/_o$); therefore, the margin required to extrapolate beyond the AOA (Δ_{AOA}) is set to 0.004. This value is determined using trend analysis of the bias as described in Section 3.5. NUREG/CR-6698 (Reference 8) allows for extrapolation outside the range bounded by the critical experiments. Reference 8 allows for the use of trends in the bias to calculate the Δ_{AOA} for the extrapolated AOA. The bias versus enrichment from Table 6-1 is 5.796E-04 (k_{eff} per % enrichment) for the low enrichment cases. Only the low enrichment cases, i.e., 4.89 to 9.97 $^w/_o$, were used to determine the trend and the bias associated with an enrichment of 1.5 $^w/_o$. Using the low enrichment cases gives a more conservative bias value than using all of the case included in the plant specific benchmark. The extrapolation penalty is then calculated to be:

$$(4.89-1.5) \times 5.796E\text{-}04=0.002$$

The Contingency Dump System enrichment value of 1.5 $^w/_o$ falls outside of the 10% range of the critical experiments provided in the plant specific benchmark. Consistent with guidance in Reference 8, additional justification is provided for this extrapolation outside 10% of the range bounded by the critical experiments. Reference 4, the International Handbook of Evaluated Criticality Safety Benchmark Experiments, does not include any critical experiments in solution below 4.89 $^w/_o$. As such, the plant specific benchmark does not contain any critical experiments in solution for a 1.5 $^w/_o$ enrichment value. To account for extrapolating outside of the 10% range for the enrichment of the Contingency Dump System, the validation incorporates an additional penalty of 0.002 (in addition to the 0.002 penalty calculated above). The resultant Δ_{AOA} is the sum of these two penalties (i.e., 0.004).

Based on the above, the USL used in the determination of the reactivity of systems or components shall be as follows.

- For systems or components NOT associated with the Contingency Dump System (i.e., systems or components with assumed enrichments within the AOA):

  $USL = K_L - \Delta_{SM} - \Delta_{AOA}$

  USL = 0.9915-0.05-0.0

  USL = 0.9415

- For systems or components associated with the Contingency Dump System (i.e., systems or components with assumed enrichments of 1.5 $^{w}/_{o}$):

  $USL = K_L - \Delta_{SM} - \Delta_{AOA}$

  USL = 0.9915-0.05-0.004

  USL = 0.9375

Figure 6-1 MONK k effective Histogram

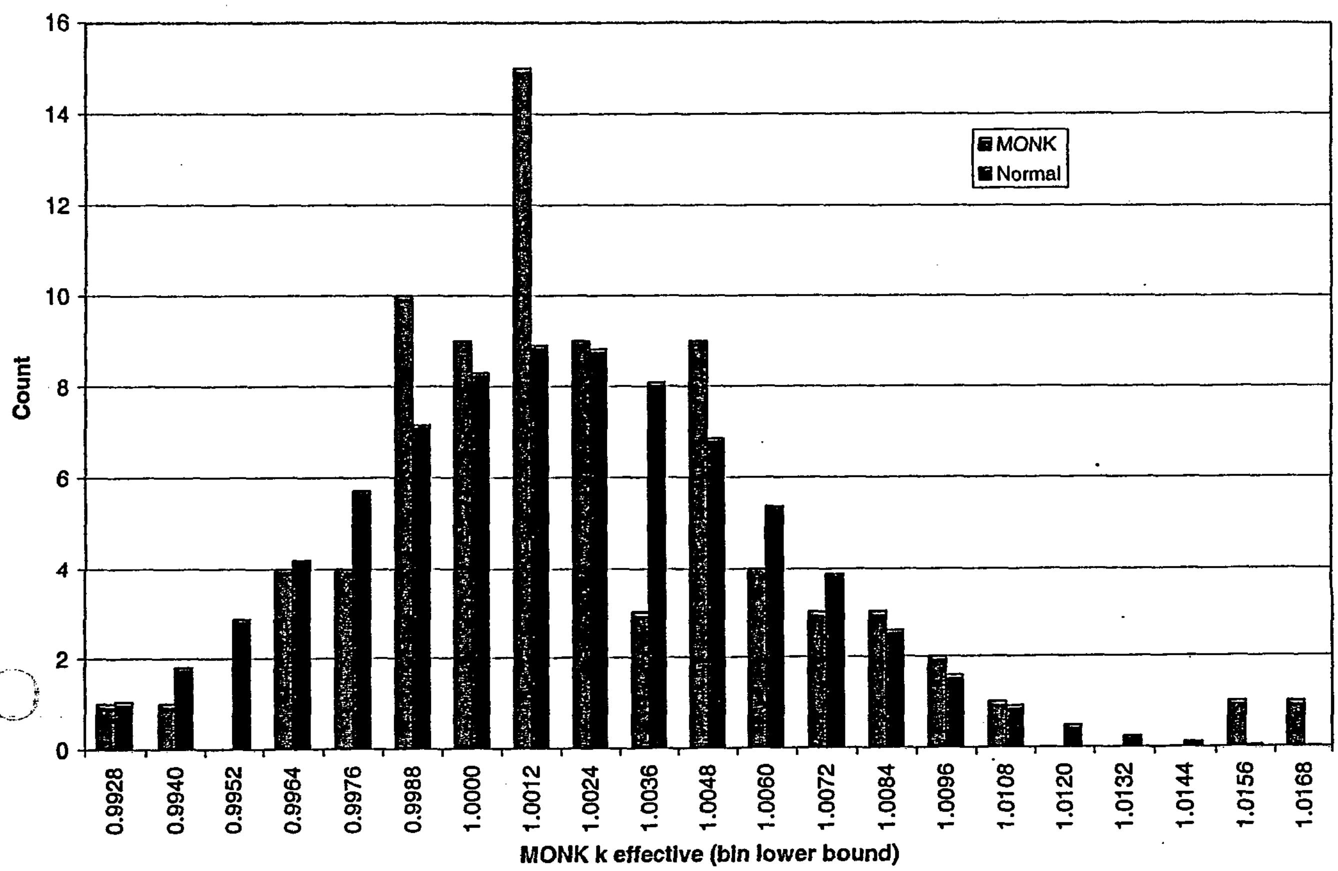

Figure 6-2 Plot of MONK k effective vs. Solution Density

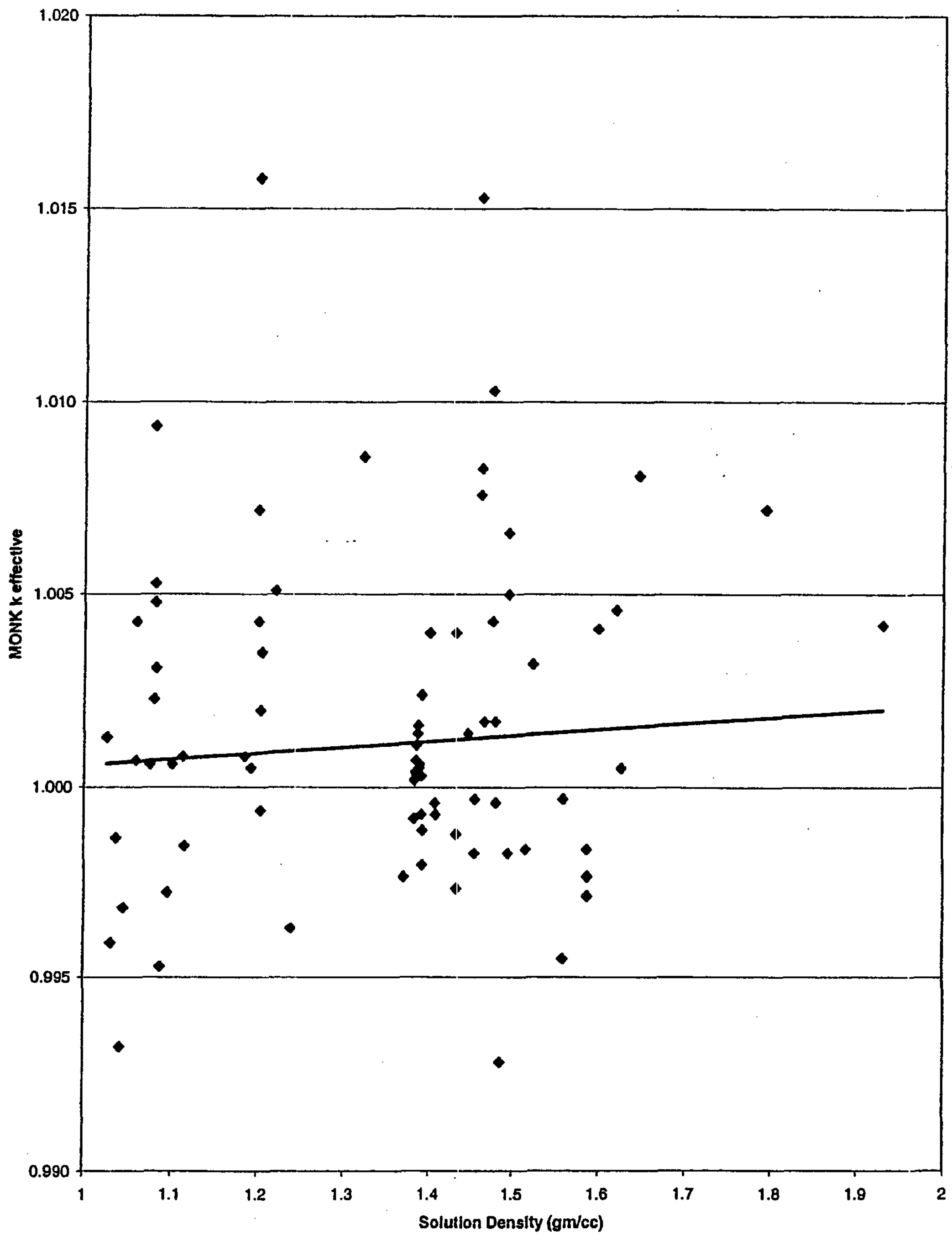

Figure 6-3 Plot of MONK k effective vs. H to ^{235}U Number Ratio

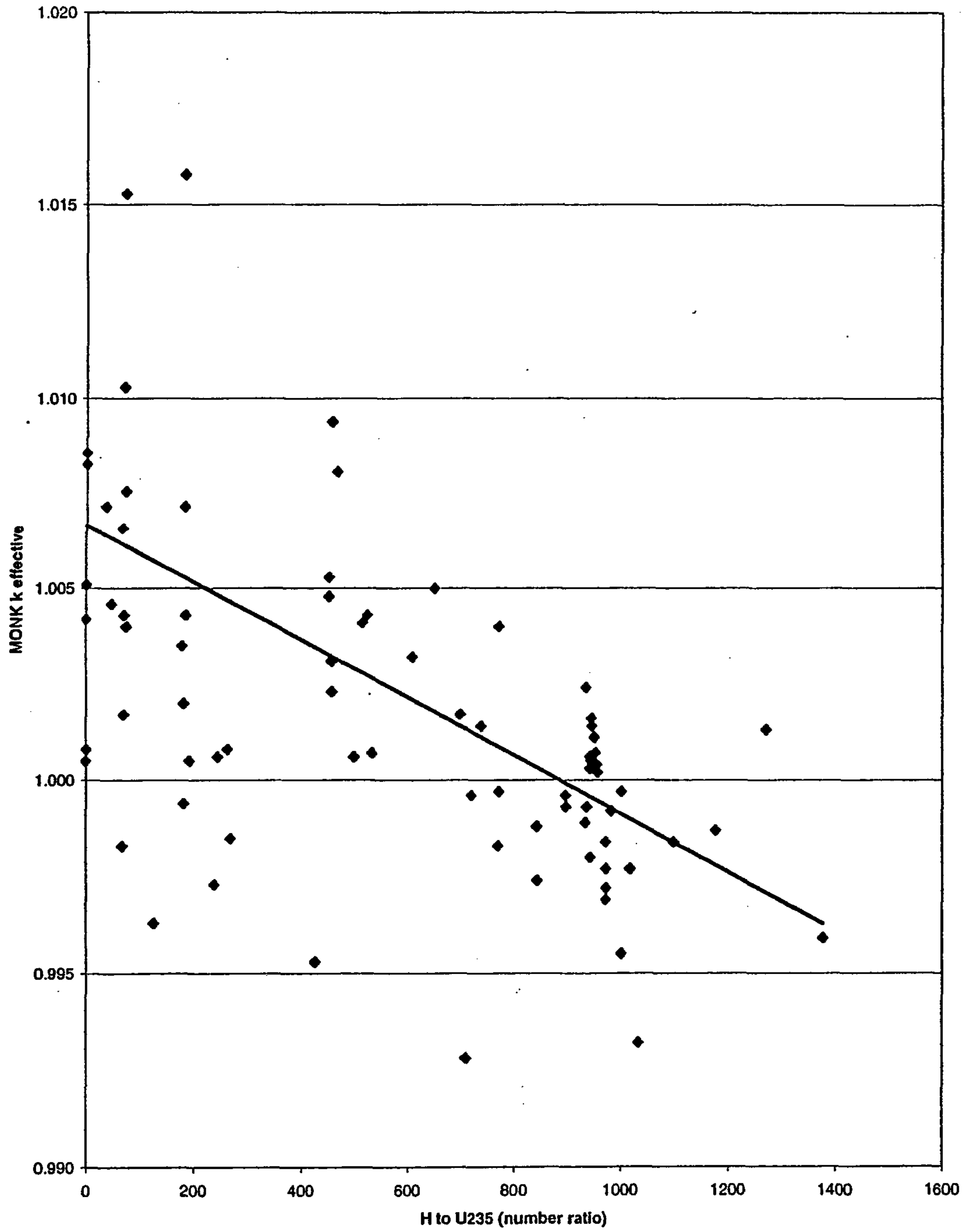

Figure 6-4 Plot of MONK k effective vs. ^{235}U Enrichment

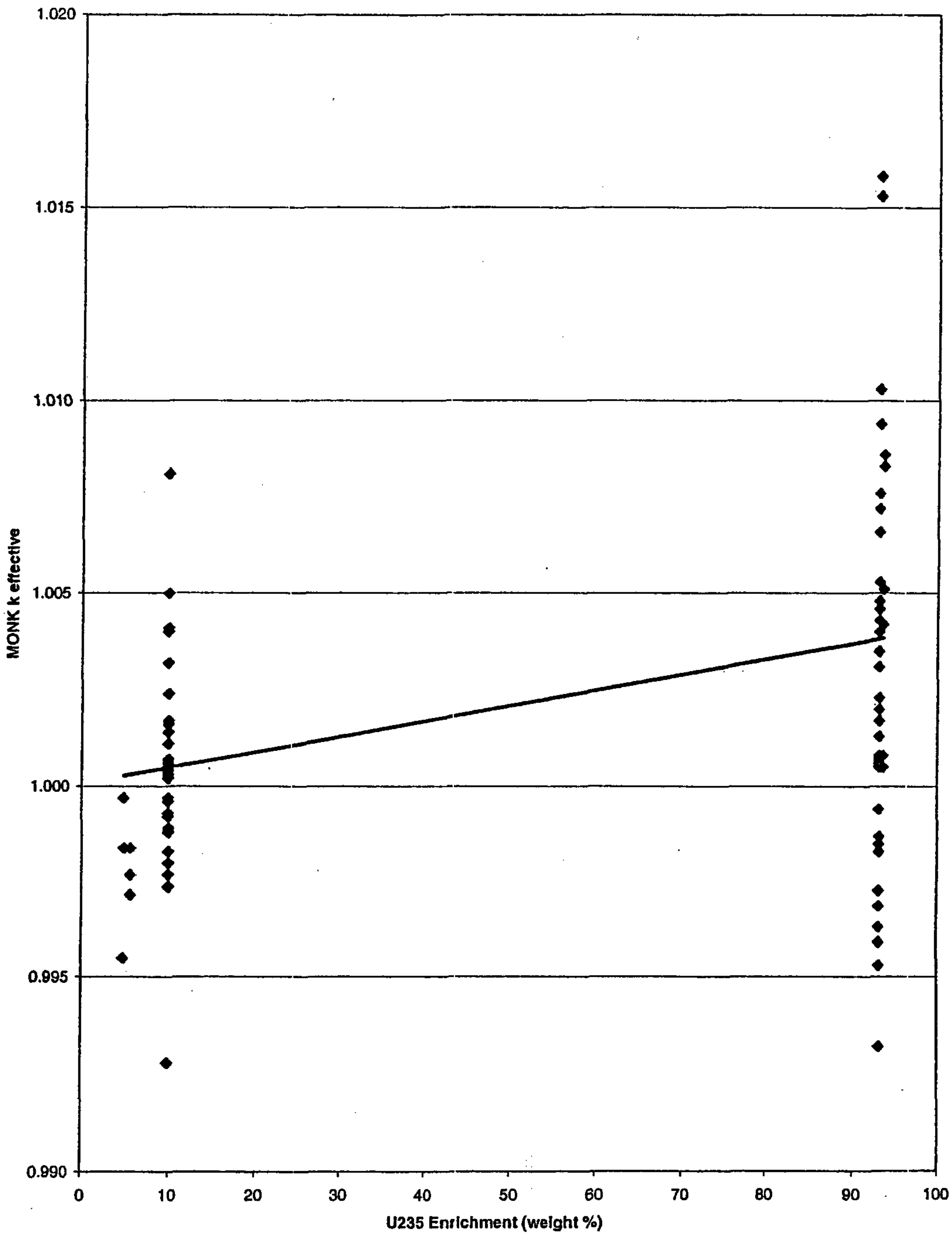

Figure 6-5 Plot of MONK k effective vs. Mean Cord Length

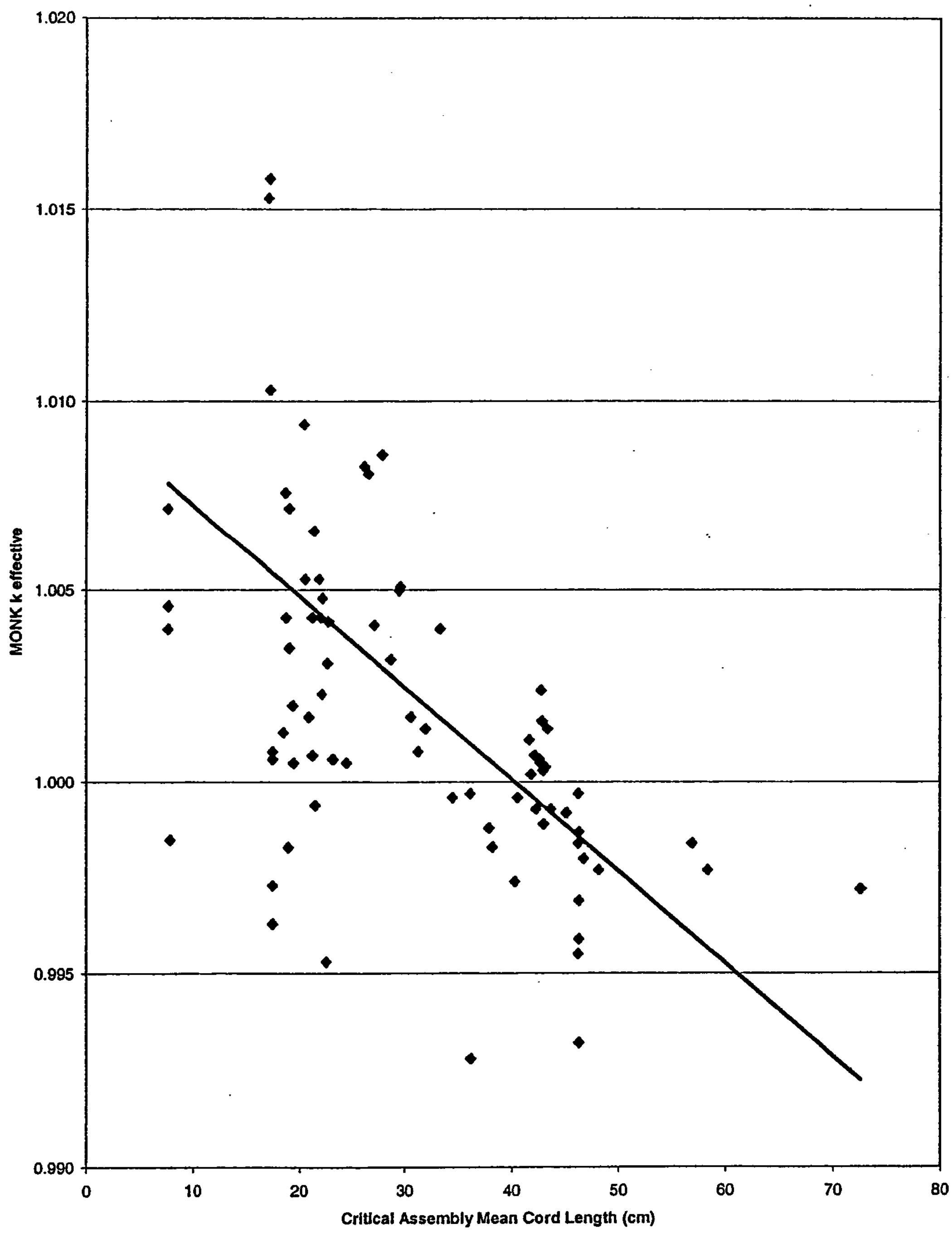

Table 6-1 Summary of Statistical Results

Fitted Parameter	Intercept	Slope	Correlation Coefficient (r^2)	Fitted Range	
				Min	Max
Solution Density (cm/cm^3)	0.99906	1.527E-03	0.003	1.026	1.930
H to ^{235}U Number Ratio (unitless)	1.00669	-7.558E-06	0.33	0.103	1378
^{235}U Enrichment ($^w/_o$)	1.00008	4.016E-05	0.13	4.89	93.65
^{235}U Enrichment ($^w/_o$)	0.99475	5.796E-04	0.02	4.89	9.97
Mean Cord Length (cm)	1.00969	-2.402E-04	0.36	7.67	81.35

7 Verification

NUREG 1520 requires a description of the verification process and results. In addition, NUREG 1520 requires a description of mathematical testing. In this report the verification and mathematical testing process is performed in three steps. The first step is to compare the results obtained in the AREVA benchmark to the computer code vendor, Serco, published results to show that MONK 8A was correctly installed and executed on the FANP PC. The second step is show that the results are repeatable if run at different times. This step is needed because MONK 8A uses the date time stamp to select a random seed value. Therefore, this step ensures that the results are similar if a different seed value is used. The final step is to repeat a subset of the MONK 8A criticality analysis cases run by Urenco. Urenco ran an extensive set of MONK 8A criticality calculations in support of their existing facilities and NEF. This step ensures that the cases run by Urenco are similar to the AREVA benchmark cases.

7.1 Benchmark Results Compared to Serco Results

The MONK 8A computer code vendor, Serco, provided a set of benchmarks identical to the benchmarks performed in this study to assure that the computer code had been installed correctly on the FANP PC and that the mathematical models are working correctly. Table 7-1 shows the results of the MONK 8A benchmark calculated by the computer code vendor and from the AREVA validation runs. Table 7-1 has the following definitions.

- "H/U" is the hydrogen to fissile atom ratios for each experiment (Reference 6).
- "Serco Benchmark" is the k_{eff} (Reference 6) values from the Serco benchmark report.
- "AREVA Validation" are the k_{eff} values from the validation runs.
- "Count" is the total number of experiments.
- "Average" is the average of all the Serco benchmark and AREVA validation k_{eff} values calculated using the Excel AVERAGE function.
- "Standard Deviation" is the standard deviation of the k_{eff} values from the Serco benchmark and AREVA validation. The standard deviation used the Excel STDEV function which uses the equation:

  $$\sigma = \sqrt{\frac{n\sum_{i=1}^{n} x_i^2 - \left(\sum_{i=1}^{n} x_i\right)^2}{n(n-1)}};$$

  where $x_i = k_{eff}$ of each experiment, n= number of experiments (80).
- "Standard Error" is the Standard Error of Measurement (Reference 7) of the k_{eff} values from the Serco benchmark and AREVA validation and uses the equation.

  $$\sigma_M = \frac{\sigma}{\sqrt{n}}.$$

Because the random number generator seed values were based on the MONK 8A default feature, the date and time of execution, the results of each experiment would not be expected to exactly match the Serco benchmark results. The average of the Serco

benchmark cases, for the 13 cases used in this project is 1.0016±0.0005 (Reference 6). The average of the AREVA validation runs was 1.0017±0.0005 as shown in Table 7-1. The agreement between the benchmark values and the validation runs is very good with the difference being attributed to the use of different seed values. This comparison shows that the computer code was installed on the FANP PC correctly.

7.2 Repeatability

As mentioned earlier, a fundamental feature of all Monte Carlo computer codes is the requirement of a random number to initiate the calculation. By default, MONK 8A utilizes the date and time of execution to derive the seed values for each case. It is of interest to evaluate the effect of the random number seed values for MONK 8A. Therefore, one validation case is chosen for a brief sensitivity study of this effect. The first case of experiment 23 listed in Table 5-1 was run on different dates and times to test the repeatability and reliability of MONK 8A. The results are summarized in Table 7-2.

The average k_{eff} of the six runs was 0.9966 with a standard deviation of 0.0011. Since the convergence criterion for the runs was a standard deviation of 0.0010; this demonstrates that MONK 8A calculates consistent results.

7.3 Verification of Urenco MONK 8A Cases

Urenco ran an extensive set of MONK 8A criticality calculations in support of their existing facilities and NEF. Thirty representative cases were selected for verification of the MONK 8A criticality analysis run by Urenco. As described in the validation section, the default seed values for the random number generator are used to make this verification independent of Urenco.

It is of interest to verify the reproducibility of the Monte Carlo solution. Therefore, the original random seed values were used in the first six cases in Table 7-3 to track the reproducibility of MONK 8A on the QA controlled computer. These six cases with the original seed values produced identical results to the Urenco cases.

The first six cases in Table 7-3 were also repeated with the default seed values. The results of all thirty cases chosen for verification are shown in Table 7-3. The average of the Urenco results for the thirty cases used in this report is 0.8764. The average of the verification runs is 0.8744 as shown on Table 7-3. The documented values and the verification runs are in good agreement.

Table 7-1 Comparison of Serco Benchmark and AREVA Validation Runs

Experiment	Case	H/U	Serco Benchmark	AREVA Validation
13	1	453.74	1.0046	1.0053
HEU	2	73.50	1.0075	1.0076
	3	73.50	1.0151	1.0153
	4	70.94	1.0050	1.0043
	5	70.94	1.0078	1.0103
	6	458.77	1.0048	1.0026
	7	458.77	1.0096	1.0094
	8	453.74	1.0053	1.0048
	9	453.74	1.0031	1.0053
	10	183.78	1.0063	1.0072
	11	183.78	1.0158	1.0158
	12	179.55	1.0029	1.0035
23	1	1377.86	0.9963	0.9959
HEU	2	1176.89	0.9979	0.9987
	3	1033.25	0.9941	0.9932
	4	971.59	0.9966	0.9969
	5	1834.85	0.9966	1.0003
35	1	35.84	1.0067	1.0072
HEU	2	47.23	1.0052	1.0046
	3	76.08	1.0044	1.0040
	4	126.47	0.9953	0.9963
	5	269.97	1.0021	0.9985
	6	264.24	1.0016	1.0008
	7	245.70	0.9990	1.0006
	8	239.02	0.9973	0.9973
	9	523.41	1.0028	1.0043
	10	533.12	1.0020	1.0007
	11	1272.25	1.0006	1.0013
43	1	1098.33	0.9950	0.9984
LEU	2	1001.28	0.9921	0.9955
	3	1001.28	0.9941	0.9997
51	1	719.02	1.0003	0.9996
LEU	2	771.30	1.0012	0.9997
	3	842.18	0.9958	0.9988
	4	895.83	1.0022	0.9996
	5	941.69	0.9996	1.0003
	6	982.52	1.0008	0.9992
	7	1017.55	0.9991	0.9977
63	1	972.18	0.9970	0.9984
LEU	2	972.18	0.9969	0.9977
	3	972.18	0.9972	0.9972
67	1	181.79	1.0029	0.9994
HEU	2	70.60	1.0014	1.0017
	3	185.71	1.0027	1.0043

Experiment	Case	H/U	Serco Benchmark	AREVA Validation
	4	68.15	1.0044	1.0066
	5	499.44	0.9993	1.0006
	6	458.76	1.0050	1.0031
	7	193.28	1.0007	1.0005
	8	181.79	1.0023	1.0020
	9	68.15	0.9999	0.9983
	10	427.40	0.9941	0.9953
68	1	34.20	1.0040	1.0042
HEU	2	53.70	1.0011	1.0005
	3	81.20	1.0060	1.0083
	4	135.30	1.0088	1.0086
	5	243.00	1.0059	1.0051
	6	430.99	1.0016	1.0008
71	1	468.73	1.0083	1.0081
LEU	2	514.15	1.0072	1.0041
	3	608.43	1.0024	1.0032
	4	650.21	1.0034	1.0050
	5	699.14	1.0044	1.0017
	6	738.93	1.0035	1.0014
	7	771.79	1.0040	1.0040
80	1	709.25	0.9997	0.9928
LEU	2	769.97	0.9991	0.9983
	3	842.18	0.9955	0.9974
	4	896.05	0.9980	0.9993
	5	942.24	0.9981	0.9980
81	1	954.82	1.0020	1.0004
LEU	2	952.22	1.0003	1.0007
	3	950.69	1.0008	1.0011
	4	956.36	0.9996	1.0002
84	1	935.78	1.0013	0.9993
LEU	2	934.06	1.0011	1.0024
	3	933.49	0.9995	0.9989
85	1	946.20	0.9998	1.0014
LEU	2	944.81	0.9995	1.0016
	3	943.63	1.0010	1.0005
	4	941.67	1.0010	1.0006
Count	80	Average	1.0016	1.0017
		Standard Error	0.0005	0.0005

Table 7-2 Results of Repeatability Sensitivity Study

Date	Time	Date/Time	Seed 1	Seed 2	k_{eff}
02/16/04	14:47:44	2/16/04 14:47	16033	29133	0.9959
02/19/04	10:49:28	2/19/04 10:49	108785	59133	0.9967
02/19/04	16:13:43	2/19/04 16:13	31421	59133	0.9955
02/20/04	13:44:37	2/20/04 13:44	6751	59133	0.9957
02/20/04	14:29:47	2/20/04 14:29	14975	69133	0.9983
02/23/04	9:47:56	2/23/04 9:47	97327	99133	0.9972
Count =	6			**Avg =**	0.9966
			Standard Deviation =		0.0011

Table 7-3 Verification Results

Case	Brief Case Description	Urenco	AREVA
1	5 $^{w}/_{o}$ Critical Value- Mass 37kgU H/U=27	0.9992	0.9974
2	5 $^{w}/_{o}$ Critical Value- Volume 28.9L	0.9979	0.9998
3	5 $^{w}/_{o}$ Critical Value- Cylinder Diameter 26.2cm	0.9977	0.9959
4	6 $^{w}/_{o}$ Critical Value- Mass 27kgU H/U=32	0.9971	0.9958
5	6 $^{w}/_{o}$ Critical Value- Volume 24L	0.9952	0.9951
6	6 $^{w}/_{o}$ Critical Value- Cylinder Diameter 24.4cm	0.9951	0.9965
7	Cold trap, center-to-center separation 110 cm with 2.5 cm reflector	0.7985	0.8012
8	Cold trap, same as case 7 with two additional components in interaction	0.8184	0.8194
9	Cold trap, pump in contact and a 2.5 cm water reflector	0.8628	0.8685
10	Product Vent in contact with pump with vacuum cleaner at side. Aluminum trap walls	0.9282	0.9276
11	Product UF6 Pumps in isolation – H/U=12	0.7434	0.7435
12	Product UF6 Pumps touching at gearbox ends – H/U=12	0.8232	0.8222
13	Product UF6 Pumps touching with vacuum cleaner along side H/U=12	0.8399	0.8399
14	Product UF6 Pumps same as case 13 but with 2.5 cm water reflector	0.8698	0.8693
15	UF6 Product Pipe work, 52cm-150mm pipe – 6 $^{w}/_{o}$ H/U=12	0.9404	0.9399
16	UF6 Product Pipe work, 52cm-150mm pipe – 6 $^{w}/_{o}$ H/U=13	0.9379	0.9451
17	UF6 Product Pipe work, 52cm-150mm pipe – 6 $^{w}/_{o}$ H/U=14	0.9405	0.9357
18	UF6 Product Pipe work, 13.5cm-100mm pipe – 6 $^{w}/_{o}$ H/U=12	0.9399	0.9420
19	UF6 Product Pipe work, 13.5cm-100mm pipe – 6 $^{w}/_{o}$ H/U=13	0.9432	0.9414
20	UF6 Product Pipe work, 13.5cm-100mm pipe – 6 $^{w}/_{o}$ H/U=14	0.9396	0.9397
21	Contingency Dump Trap in isolation with 2.5 cm of water reflection	0.6421	0.6479
22	Vacuum Cleaners as isolated cylinder at optimum moderation with 2.5 cm reflector	0.7992	0.7924
23	TSB - isolated 12 liter containers at 60 cm containing contaminated charcoal	0.6980	0.6797
24	TSB – single isolated cylinder containing UF4/oil mixture	0.8495	0.8399
25	TSB – 5x5 array with a container in contact with a 2.5 cm water reflector	0.9236	0.9198
26	TSB Ventilation Room 7x7 array of chemical traps touching – H/U=12	0.9146	0.9124
27	TSB Ventilation Room 11x11 array of chemical traps 5 cm spacing – H/U=7	0.8620	0.8592
28	TSB Chemistry Laboratory 1S bottles in a 25x25 array with water flooding 1.5 cm spacing	0.6513	0.6397
29	TSB Decontamination Workshop – linear array of pairs of touching pumps 60 cm spacing	0.8507	0.8420
30	TSB Fomblin Oil Recovery System - optimum moderation H/U=14	0.7931	0.7842
	Average	0.8764	0.8744

8 Conclusions

The MONK 8A code package using the JEF 2.2 data library has been validated to perform criticality calculations for National Enrichment Facility. The validation covers all plant activities.

- For systems or components NOT associated with the Contingency Dump System (i.e., systems or components with assumed enrichments within the AOA), the USL = 0.9415.

  This USL accounts for the computational bias, uncertainties, and an administrative margin. The administrative margin is established at 0.05.

- For systems or components associated with the Contingency Dump System (i.e., systems or components with assumed enrichments of 1.5 $^{w}/_{o}$), the USL = 0.9375.

  This USL accounts for the computational bias, uncertainties, an administrative margin, and additional margin to account for the extrapolated AOA. The administrative margin is established at 0.05. The additional margin to account for the extrapolated AOA is established at 0.004

If, in the future, a parameter value for design applications falls outside of the current validated AOA for systems or components not associated with the Contingency Dump System or falls outside the current extrapolated AOA associated with the Contingency Dump System, LES shall revise the validation report to identify additional AOA margin and provide a letter to the NRC describing the change prior to using results from calculations with a parameter value that falls outside the current validated AOA (or current extrapolated AOA in the case of the Contingency Dump System) in NCSAs.

9 References

1. Serco Assurance (United Kingdom), ANSWERS(98)6, Issue 3, "MONK User Guide Version 8," © 1987-2001.

2. NRC (U.S. Nuclear Regulatory Commission), 2002. Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility, NUREG-1520, March 2002.

3. ANSI/ANS (American National Standards Institute/American Nuclear Society), 1998. American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, ANSI/ANS-8.1-1998, La Grange Park, IL.

4. Nuclear Energy Agency, NEA Nuclear Science Committee, NEA/NSC/DOC(95)03, "International Handbook of Evaluated Criticality Safety Benchmark Experiments," September 2002 Edition.

5. ANSI/ANS (American National Standards Institute/American Nuclear Society), 1984. Criticality Safety Criteria for Handling Storage, and Transportation of LWR Fuel Outside Reactors, ANSI/ANS-8.17, La Grange Park, IL.

6. Serco Assurance (United Kingdom), ANSWERS/MONK(99)8, Issue 3, "Benchmark Summary for MONK with a JEF2.2-based Nuclear Data Library," September 2002.

7. J. D. Brown, "Standard Error vs. Standard Error of Measurement," Shiken:JALT Testing & Evaluation SIG Newsletter, Vol. 3 No. 1 April 1999 (p. 15-17).

8. J.C. Dean, R.W. Tahoe, Jr., Guide to Validation of Nuclear Criticality Safety Calculation Methodology," NUREG/CR-6698, Science Applications International Corporation, Oak Ridge, TN, January 2001.

9. R. E. Rothe and I. Oh, "Benchmark Critical Experiments on High-Enriched Uranyl Nitrate Solution Systems", Nuclear Technology, vol. 41, pages 207-225, 1978.

10. I. Oh and R. E. Rothe, "A Calculational Study of Benchmark Critical Experiments on High-Enriched Uranyl Nitrate Solution Systems," Nuclear Technology, vol. 41, pages 226-243, 1978.

11. National Enrichment Facility, Safety Analysis Report, Revision 4, April 2005. Chapter 5, "Nuclear Criticality Analysis."

12. Del Pesco,T., Perfluoralkylpolyethers, 287-303, CRC Handbook of Lubrication and Tribology, Volume 111, 1994.

Attachment 1A
Example MONK 8A Inputs

Input File case13.01

```
columns 1 132
* MONK VALIDATION CALCULATIONS - EXPERIMENT NO. 13 (Version 2)
* Case 13.01
* Calculations performed by P Turner - July 2003

* Summary of Experiment
* ---------------------
* Fissile Material: High Enriched Uranyl Nitrate (93.17% U235)
* Geometry:         Cylindrical
* Moderator:        Nitrate Solution
* Reflector:        Plastic
* Reference:        Robert E. Rothe and Inki Oh
*                   Benchmark Critical Experiments on High-Enriched
*                   Uranyl Nitrate Solution Systems
*                   Nuclear Technology Volume 41
*                   December 1978.

* Experiment Critical Parameters
* ------------------------------

* Aluminium Tank Internal Diameter : 27.88 cm
* Aluminium Tank Internal Height   : 76.9 cm
* Uranium Concentration            : 60.32 g U/l
* Critical Height                  : 51.67 +/- 0.05 cm
* Position Of Tank                 : In Corner

* Important Notes
* ---------------
* 1. Assume Measured Internal Diameter/Height Was Before tank was painted
* 2. Tail Pipe Internal Surface Not Painted
* 3. No impurities in Fissile Solution Modelled
* 4. Temperature use 20degree room temp. Actual Reported 23degrees C.
* 5. Complete Reflector Modelled. Actual had bits missing from corners.
*******************************************************************************
BEGIN MATERIAL SPECIFICATION

NORMALISE
NMATERIALS 5

* Material  1 - Uranyl Nitrate Solution B
* Material  2 - Aluminium Tank
* Material  3 - Epoxy Paint (Phenoline 300)
* Material  4 - Plastic Reflector (Non-Fire Retardant)
* Material  5 - Plastic Reflector (Fire Retardant)

ATOMS
MATERIAL 1
DENSITY 0.0
U234 PROP 1.58648E-06
U235 PROP 1.44016E-04
U236 PROP 6.67987E-07
U238 PROP 8.19862E-06
O    PROP 3.40785E-02
H1   PROP 6.53452E-02
N    PROP 3.76998E-04

WEIGHT
MATERIAL 2
DENSITY 2.737
MG   PROP 0.0100
AL   PROP 0.9741
SI   PROP 0.0060
TI   PROP 0.0003
CR   PROP 0.0017
MN   PROP 0.0007
FE   PROP 0.0047
```

```
CU    PROP 0.0025

ATOMS
MATERIAL 3
DENSITY 0.0
C     PROP 0.0273170
O     PROP 0.0177320
TI    PROP 0.0029330
H1    PROP 0.0215810
N     PROP 0.0008412
SI    PROP 0.0017750
AL    PROP 0.0017804
K     PROP 0.0005795

MATERIAL 4
DENSITY 0.0
H1    PROP 0.0569020
C     PROP 0.0355140
O     PROP 0.0143480

MATERIAL 5
DENSITY 0.0
H1    PROP 0.0551690
C     PROP 0.0339690
O     PROP 0.0142320
N     PROP 0.0000553
P     PROP 0.0003851
CL    PROP 0.0003561

USE H1INCH2 FOR H1 IN MATERIAL 3
USE H1INCH2 FOR H1 IN MATERIAL 4
USE H1INCH2 FOR H1 IN MATERIAL 5

END
*******************************************************************************
BEGIN MATERIAL GEOMETRY

PART 1 ! Cylinder Surrounded by Walls and Roof.
NEST
ZROD  BH1  38.35  125.99  0.0    14.26   77.54
BOX   M0   20.6    20.6   0.0   122.9   122.9  122.9
BOX   M4    0.0     0.0   0.0   164.1   164.1  122.9
BOX   M5    0.0     0.0   0.0   164.1   164.1  143.5

PART 2 ! Floor Region Containing Tail Pipe
NEST
ZROD  BH6  38.35  125.99  0.0     1.27   20.6
BOX   M5    0.0     0.0   0.0   164.1   164.1   20.6

PART 3 ! Tail Pipe Below Reflector
NEST
ZROD  BH6  38.35  125.99  0.0     1.27    9.1
BOX   M0    0.0     0.0   0.0   164.1   164.1    9.1

PART 4 ! Complete Arrangement
CLUSTER
BOX   P1   0.0     0.0   29.7   164.1   164.1  143.5
BOX   P2   0.0     0.0    9.1   164.1   164.1   20.6
BOX   P3   0.0     0.0    0.0   164.1   164.1    9.1
BOX   M0   0.0     0.0    0.0   164.1   164.1  173.2

ALBEDO 0 0 0 0 0 0

END
*******************************************************************************
BEGIN HOLE DATA

* Hole 1 - Axial Description of Tank
PLATE
```

Input File case23.01

```
* MONK VALIDATION CALCULATIONS - EXPERIMENT 23.01
* ---------------------------------------------
* Calculations performed by L S Grindrod - July 1995
* Reported in ANSWERS/MONK/VAL/23
*

* Summary of experiment
* ---------------------
* Fissile Material:   Uranyl Nitrate Solution
* Geometry:           Spherical
* Neutron poison:     None
* Reflector:          None
* Reference:          R Gwin and D W Magnuson
*                       Eta of U233 and U235 for Critical
*                       Experiments. Nucl.Sci.Eng.12,364(1962)
*                     ORNL Spheres (1995)
* Code Package:              MONK7A-JEF2

*
* Critical Parameter Data
* -----------------------
* Fissile Solution Diameter   :  34.595 cm
* Vessel Wall Thickness              :  None
* Uranium Concentration              :  20.13  g/l
* NO3 Concentration           :  19.25  g/l
* Specific Gravity            :
**********************************************************************

BEGIN MATERIAL DATA
MONK 1 8 NUCNAMES

* material 1 ... uranyl nitrate

CONC  J2U234 5.38E-7 J2U235 4.8066E-5 J2U236 1.38E-7
      J2U238 2.807E-6 J2N14 1.862E-4  J2N15 0.007E-4
      J2HINH2O 0.066228 J2O16 0.033736

**********************************************************************
BEGIN MATERIAL GEOMETRY

PART 1 NEST
SPHERE M1  0.0 0.0 0.0 34.595 ! Uranyl nitrate sphere
END

**********************************************************************
BEGIN CONTROL DATA
STAGES -1 200 1000 STDV 0.0010
END

**********************************************************************
BEGIN SOURCE GEOMETRY
ZONEMAT
ZONE 1  PART 1 /
END

**********************************************************************
```

Input File case35.01

```
* MONK VALIDATION CALCULATIONS - EXPERIMENT 35.01
* ---------------------------------------------
* Calculations performed by W Wright - July 1996
* Reported in ANSWERS/MONK/VAL/35
*

* Summary of experiment
* ---------------------
* Fissile Material:          Uranium Oxyfluoride Solution
* Geometry:                  Spherical
* Neutron Poison:            None
* Reflector:                 Water
* Reference:                 M Pitts, F Rahnema, T G Williamsom
*                            Water-Reflected 6.4 Liter Spheres
*                            of Uranium Oxyfluoride Solutions
*                            HEU-SOL-THERM-009 (1995)
* Code Package:              MONK7A-JEF2.2

* Critical Parameter Data
* -----------------------
* Fissile Solution Diameter :  11.5177 cm
* Vessel Wall Thickness     :   0.1587 cm
* Uranium Concentration     : 696.42   g/l
* H/U235                    :  35.8
* Specific Gravity          :   1.7950 g/cc

***********************************************************

BEGIN MATERIAL DATA
MONK 3 12  NUCNAMES

* material 1 - Uranium Fluorine
* material 2 - Aluminium Vessel Wall
* material 3 - Water Reflector

CONC    J2U234    1.7561E-5  J2U235    1.6626E-3   J2U236  8.8837E-6
        J2U238    9.4079E-5  J2F19     3.5663E-3   J2O16   3.3360E-2
        J2HINH2O  5.9587E-2
CONC    J2AL27    5.9699E-2  J2SI      5.5202E-4   J2CU    5.1364E-5
        J2ZN64    2.4958E-5  J2MN55    1.4853E-5
CONC    J2HINH2O  6.6659E-2  J2O16     3.3329E-2
END

**********************************************************************

BEGIN MATERIAL GEOMETRY

PART 1 NEST
SPHERE M1 0.0 0.0 0.0 11.5177
SPHERE M2 0.0 0.0 0.0 11.6764
SPHERE M3 0.0 0.0 0.0 35.0
END

**********************************************************************

BEGIN CONTROL DATA
STAGES -1  200  1000  STDV 0.0010
END

**********************************************************************

BEGIN SOURCE GEOMETRY
ZONEMAT
ZONE 1  PART 1 /
END
```

Input File case43.01

```
*****************************************************************
* MONK VALIDATION CALCULATIONS - EXPERIMENT 43.01
* ----------------------------------------------
* Calculations performed by C J Bazell - June 1997

* Summary of experiment
* ---------------------
* Fissile Material:          Uranium Oxyfluoride Solution
* Geometry:                  Spherical
* Neutron Poison:            None
* Reflector:                 Water
* Reference:                 Pitts M., Rahnema F., Williamson T.G.
*                            174 Liter Spheres of Low Enriched (4.9%)
*                            Uranium Oxyfluoride Solutions
*                            LEU-SOL-THERM-002 (undated)
* Code Package:              MONK7B-JEF

* Critical Parameter Data
* -----------------------
* Fuel Region Radius        :  34.3990 cm
* Aluminium Wall Thickness  :   0.1588 cm
* Uranium Concentration     :   0.4522 g.cm-3
* H/U235                    :   1098
* Fuel Solution Density     :   1.5160 g.cm-3

* Notes
* -----
* The experiment temperature was assumed to be 25C and the
* atomic densities for the water reflector calculated accordingly.
* However, note that the MONK data temperature is 20C.

* Due to the unavailability of zinc cross-sections in the UKNDL database,
* the zinc concentration (atom/barn-cm) is combined with that of the aluminium.
*
*************************************************************

BEGIN MATERIAL SPECIFICATION

NMATERIALS 3

* material 1 - uranium oxyfluoride solution
* material 2 - 1100 aluminium
* material 3 - water

ATOMS
MATERIAL 1 DENSITY 0.0
U234 PROP 2.3271E-07
U235 PROP 5.6655E-05
U238 PROP 1.0878E-03
F19  PROP 2.2893E-03
O16  PROP 3.3402E-02
H1   PROP 6.2226E-02

ATOMS
MATERIAL 2 DENSITY 0.0
AL27 PROP 5.9724E-02
SI   PROP 5.5202E-04
CU   PROP 5.1364E-05
MN   PROP 1.4853E-05

ATOMS
MATERIAL 3 DENSITY 0.0
H1      PROP 6.6659E-02
O16     PROP 3.3329E-02

USE J2HINH2O FOR H1 IN ALL MATERIALS
```

```
END

************************************************************************

BEGIN MATERIAL GEOMETRY

PART 1 NEST
SPHERE M1 0.0 0.0 0.0 34.3990
SPHERE M2 0.0 0.0 0.0 34.5578
SPHERE M3 0.0 0.0 0.0 49.5578
END

************************************************************************

BEGIN CONTROL DATA
STAGES -1  200  1000  STDV 0.0010
END

************************************************************************

BEGIN SOURCE GEOMETRY
ZONEMAT
ZONE 1  PART 1 /
END
```

Input File case 51.01

```
* MONK VALIDATION CALCULATION 51.01
* ----------------------------------------------
* Calculation performed by W V Wright - January 1999

* Summary of experiment
* ---------------------
* Fissile Material:          10% enriched uranyl nitrate solution
* Geometry:                  Cylindrical
* Neutron Poison:            None
* Reflector:                 Water
* Reference:                 T Yamamoto, Y Miyoshi
*                            STACY: Water-Reflected 10%-Enriched Uranyl
*                            Nitrate Solution in a 60cm Diameter
*                            Cylindrical tank
*                            LEU-SOL-THERM-004 (30/09/98)
* Code Package:              MONK8A-JEF2.2

* Critical Parameters Data -

* Uranium Concentration     : 310.1  gU/l
* Solution Height           :  41.53 cm

* Additional Notes -

* The experimental temperature was assumed to be 25 degrees C (298 K)
* MONK nuclear data temperature is at 20 degrees C.

* Keyword Parameters -
*
* solution height (height of solution above tank inner base)
*
BEGIN MATERIAL SPECIFICATION

NMATERIALS 4

* material 1 - uranyl nitrate solution
* material 2 - stainless steel
* material 3 - water
* material 4 - air

ATOMS
MATERIAL 1 DENSITY 0.0
U234   PROP 6.3833E-07
U235   PROP 7.9213E-05
U236   PROP 7.9114E-08
U238   PROP 7.0556E-04
H1     PROP 5.6956E-02
N      PROP 2.8778E-03
O      PROP 3.8029E-02

ATOMS
MATERIAL 2 DENSITY 0.0
C      PROP 4.3736E-05
SI     PROP 1.0627E-03
MN     PROP 1.1561E-03
P      PROP 4.3170E-05
S      PROP 2.9782E-06
NI     PROP 8.3403E-03
CR     PROP 1.6775E-02
FE     PROP 5.9421E-02

ATOMS
MATERIAL 3 DENSITY 0.0
H1     PROP 6.6658E-02
O      PROP 3.3329E-02
```

```
ATOMS
MATERIAL 4 DENSITY 0.0
N     PROP 3.9016E-05
O     PROP 1.0409E-05

USE H1INH2O FOR H1 IN ALL MATERIALS

END

*****************************************************************

BEGIN MATERIAL GEOMETRY

PART 1 NEST

ZROD M1  3*0.0         29.5 41.53      ! fuel solution
ZROD M4  3*0.0         29.5 150.0      ! inside tank
ZROD M2  2*0.0 -2.0    29.8 154.5      ! tank wall
ZROD M3  2*0.0 -32.0   59.8 204.5      ! water reflector
END

******************************************************************

BEGIN CONTROL DATA
STAGES -1  200  1000  STDV 0.0010
END

*************************************************************************

BEGIN SOURCE GEOMETRY
ZONEMAT
ZONE 1  PART 1 /
MATERIAL 1
END
```

Input File case63.01

```
* MONK VALIDATION EXPERIMENT NUMBER 63.01
* --------------------------------------
*
* MONK VALIDATION CALCULATIONS - EXPERIMENT LEU-SOL-THERM-005 Case 1
* ------------------------------------------------------------------
*
* Summary of experiment
* ---------------------
* Fissile Material:     Uranium (5.64% U235) Nitrate Solution
* Geometry:             Cylindrical
* Neutron poison:       None; Boron Carbide
* Reflector:            Water
* Moderator:            Uranium Nitrate Solution
* Reference:            A Tsiboulia, Y Rozhikhin, V Gurin
*                       Boron Carbide Absorber Rods in Uranium
*                       (5.64% 235U) Nitrate Solution
*                       LEU-SOL-THERM-005 (September 30, 1998)
* Code Package:         MONK8A
*
* Critical Parameter Data
* -----------------------
* Number of absorber rods      =  0
* Critical Height of solution  = 58.9839 cm

*******************************************************
BEGIN MATERIAL SPECIFICATION
 NMATERIALS 4

ATOMS                        ! Uranium Nitrate Solution
 MATERIAL 1 DENSITY 0.0
 U234    PROP 3.0893E-7
 U235    PROP 5.7830E-5
 U236    PROP 5.1050E-7
 U238    PROP 9.5450E-4
 N       PROP 2.9898E-3
 O       PROP 3.8624E-2
 H1      PROP 5.6221E-2

ATOMS                        ! Boron Carbide
 MATERIAL 2 DENSITY 0.0
 B10     PROP 1.0844E-2
 B11     PROP 4.3648E-2
 C       PROP 1.3623E-2

ATOMS                        ! Water
MATERIAL 3 DENSITY 0.0
 H1      PROP 6.6742E-02
 O       PROP 3.3371E-02

ATOMS                        ! Stainless Steel
 MATERIAL 4 DENSITY 0.0
 Fe      PROP 5.9088E-2
 Cr      PROP 1.6532E-2
 Ni      PROP 8.1369E-3
 Mn      PROP 1.3039E-3
 Si      PROP 1.3603E-3
 Ti      PROP 5.9844E-4

USE H1INH2O FOR H1 IN ALL MATERIALS

END
*******************************************************
BEGIN MATERIAL GEOMETRY
PART 1                            ! Inner Tank
 NEST
 zrod    BH1   3*0.0   54.8 1.7          ! lattice plate
```

```
 zrod   M1   3*0.0   55.0 58.9839    ! uranium solution
 zrod   M0   3*0.0   55.0 248.5      ! inside, inner tank

PART 2                          ! Outer Tank
 zrod   1  2*0.0 38.5    55.0 248.5  ! inner tank, inner wall
 zrod   2  2*0.0 37.0    55.6 250.0  ! inner tank, outer wall
 zrod   3  2*0.0 1.0     99.2 286.0  ! outer tank, outer wall
 zrod   4  3*0.0        100.0 287.0  ! outer tank, outer wall
 zp     5  146.5                     ! void over water
 zones
 /1innertank/  P1    +1             ! inside inner tank
 /2intankwal/  M4    -1 +2          ! inner tank wall
 /3water/      M3    -2 +3 -5       ! water in tank
 /4voidover/   M0    -2 +3 +5       ! water in tank
 /5outertank/  M4    -3 +4          ! outer tank wall

END
*******************************************************
BEGIN HOLE DATA
*  Hole 1,Lattice Plate

 TRIANGLE  10.6  2.775 2.8
 WRAP 6  100.0 100.1  OMIT 6
 1 4 4  4 4
END
*******************************************************
BEGIN CONTROL DATA
 STAGES -1 200 1000  STDV 0.0010
END
*******************************************************
BEGIN SOURCE GEOMETRY
 ZONEMAT
 ZONE 1 PART 2 / MATERIAL 1
END
```

Input File case67.01

```
* MONK VALIDATION EXPERIMENT NUMBER 67.01
* --------------------------------------
*
* MONK VALIDATION CALCULATIONS - EXPERIMENT HEU-SOL-THERM-001 Case 1
* -------------------------------------------------------------------
*
* Summary of experiment

* ---------------------
* Fissile Material:      Uranyl Nitrate [93.17wt.% 235U, 50 - 350 g(U)/l]
* Geometry:              Cylinder
* Neutron poison:        None
* Reflector:             None
* Reference:             Brian Palmer
*                        Minimally Reflected Cylinders Of Highly Enriched Solutions Of
*                        Uranyl Nitrate
*                        HEU-SOL-THERM-001 (September 30, 1997)
* Code Package:          MONK8A
*
* Critical Parameter Data
* -----------------------

* Solution Height (cm):         31.20
* Tank Inside Diameter (cm):    27.92
* Tank Inside Height (cm):      41.6
* Side Wall Thickness (cm):      0.32
* Bottom Thickness (cm):         0.64
* Tank Material:                 Stainless Steel

* Solution Data
* -------------
* Uranium Concentration (gU/l):       145.68
* Excess Nitric Acid (moles/liter):     0.294
* Solution Density (g/cc):              1.2038
***************************************************************

BEGIN MATERIAL SPECIFICATION

NMATERIALS 2

* Material 1 = Specified UN solution
ATOMS
MATERIAL 1 DENSITY 0.0
U235     PROP 3.4777E-4
U234     PROP 3.8310E-6
U236     PROP 1.6130E-6
U238     PROP 1.9798E-5
O16      PROP 3.5037E-2
N        PROP 9.2307E-4
H1       PROP 6.3220E-2

* Material 2 = S/S (given composition)
ATOMS
MATERIAL 2 DENSITY 0.0
C        PROP 2.6231E-4
SI       PROP 1.3768E-3
P        PROP 3.8530E-5
S        PROP 2.8282E-5
CR       PROP 1.6985E-2
MN       PROP 1.1209E-3
FE       PROP 5.9852E-2
NI       PROP 7.4500E-3
MO       PROP 8.9563E-6

END
```

```
***********************************************

BEGIN MATERIAL GEOMETRY

* Part 1 - S/S Tank of UN Solution

PART 1

NEST

ZROD M1 0.0 0.0  0.0  13.960 31.20
ZROD M0 0.0 0.0  0.0  13.960 41.6
ZROD M2 0.0 0.0 -0.64 14.280 42.24

ALBEDO 0 0 0

END

***********************************************

BEGIN CONTROL DATA

STAGES -1     ! Start at stage number -1
       200    ! Finish at stage number 200
       1000   ! 100 superhistories (neutrons)
              ! (10 generations per superhistory)
STDV 0.0010   ! Finish when Standard Deviation reaches 0.0010

END

***********************************************

BEGIN SOURCE GEOMETRY

ZONEMAT

ZONE 1 IN PART 1 /

END

***********************************************
```

Input File case68.01

```
* MONK VALIDATION EXPERIMENT NUMBER 68.01
* --------------------------------------
*
* MONK VALIDATION CALCULATIONS - EXPERIMENT HEU-SOL-THERM-004 Case 1
* ------------------------------------------------------------------
*
* Summary of experiment

* ---------------------
* Fissile Material:     Uranyl Fluoride/Heavy Water Solution [93.65wt.% 235U]
* Geometry:             Spherical
* Neutron poison:       None
* Reflector:            Heavy Water
* Reference:            Joseph L. Sapir
*                       Reflected Uranyl-fluoride Solutions In Heavy Water
*                       HEU-SOL-THERM-004 (March 31, 1995)
* Code Package:         MONK8A
*
* Critical Parameter Data
* -----------------------
* Solution Radius (cm):          17.088
* Solution Tank Radius (cm):     17.189
* Reflector Radius (cm):         44.367
* Reflector Tank Radius (cm):    44.621

* Solution Data
* -------------
* Deuterium/235U Atomic Ratio:  34.2
* U235 Density (g/cc):           0.679
***************************************************************

BEGIN MATERIAL SPECIFICATION

NMATERIALS 3

* Material 1 = Specified UO2F2/D2O solution
ATOMS
MATERIAL 1 DENSITY 0.0
U234    PROP 1.9029E-5
U235    PROP 1.7397E-3
U238    PROP 9.7761E-5
F19     PROP 3.7129E-3
O16     PROP 3.3461E-2
H2      PROP 5.9318E-2
H1      PROP 1.7849E-4

* Material 2 = Type 321 Stainless Steel (given composition)
ATOMS
MATERIAL 2 DENSITY 0.0
FE      PROP 5.9355E-2
CR      PROP 1.6511E-2
NI      PROP 7.7203E-3
MN      PROP 1.7363E-3
SI      PROP 1.6982E-3

* Material 3 = D2O (given composition)
ATOMS
MATERIAL 3 DENSITY 0.0
H2      PROP 6.6078E-2
H1      PROP 3.9886E-4
O16     PROP 3.3238E-2

END

**********************************************
```

```
BEGIN MATERIAL GEOMETRY

* Part 1 - Water Reflected Al Sphere of UO2F2 Solution

PART 1

NEST

SPHERE M1 0 0 0 17.088
SPHERE M2 0 0 0 17.189
SPHERE M3 0 0 0 44.367
SPHERE M2 0 0 0 44.621

ALBEDO 0

END

**********************************************

BEGIN CONTROL DATA

STAGES -5     ! Start at stage number -5
       200    ! Finish at stage number 200
       1000   ! 1000 superhistories (neutrons)
              ! (10 generations per superhistory)
STDV   0.0010! Stop Calculation when Standard Deviation = 0.0010

END

**********************************************

BEGIN SOURCE GEOMETRY

ZONEMAT

ZONE 1 IN PART 1 /

END

**********************************************
```

Input File case71.01

```
* MONK VALIDATION EXPERIMENT NUMBER 71.01
* --------------------------------------
*
* MONK VALIDATION CALCULATIONS - EXPERIMENT LEU-SOL-THERM-016 Case  1
* -----------------------------------------------------------------
*
* Summary of experiment
* ---------------------
* Fissile Material:     10%-enriched Uranyl Nitrate (U conc. range 300-464gU/l)
* Geometry:             Slab
* Moderator:            Nitrate Solution
* Neutron poison:       None
* Reflector:            Light Water
* Reference:            Shouichi Watanabe and Tsukasa Kikuchi
*                       STACY: 28-cm-thick Slabs of 10%-enriched
*                       Uranyl Nitrate Solutions, Water-Reflected
*                       LEU-SOL-THERM-016 (September 30, 1999)
* Code Package:         MONK8A
*
* Critical Parameter Data
* -----------------------
* Experiment Run No.            : 105
* U conc. (gU/l)                : 464.2    +/- 0.8
* Free nitric acid conc. (mol/l) :   0.852  +/- 0.018
* Solution Density (g/cc)       :   1.6462 +/- 0.0005
* Critical Height (cm)          :  40.09   +/- 0.02
* Experiment Temperature        :  23.8
* Benchmark k-effective         :   0.9996 +/- 0.0013
****************************************************************

BEGIN MATERIAL SPECIFICATION

NMATERIALS 4

* Material 1 = Uranyl Nitrate
ATOMS
MATERIAL 1 DENSITY 0.0
U234    PROP 9.5555E-7
U235    PROP 1.1858E-4
U236    PROP 1.1843E-7
U238    PROP 1.0562E-3
H1      PROP 5.5582E-2
N       PROP 2.8647E-3
O16     PROP 3.8481E-2

* Material 2 = Water
ATOMS
MATERIAL 2 DENSITY 0.0
H1      PROP 6.6658E-2
O16     PROP 3.3329E-2

* Material 3 = Stainless Steel (304L) Tank
ATOMS
MATERIAL 3 DENSITY 0.0
C       PROP 7.1567E-5
SI      PROP 7.1415E-4
MN      PROP 9.9095E-4
P       PROP 5.0879E-5
S       PROP 1.0424E-5
NI      PROP 8.5600E-3
CR      PROP 1.6725E-2
FE      PROP 5.9560E-2

* Material 4 = Air
ATOMS
MATERIAL 4 DENSITY 0.0
```

```
N      PROP 3.9016E-5
O16    PROP 1.0409E-5

END

**********************************************

BEGIN MATERIAL GEOMETRY

* Part 1 - Water Reflected Uranyl Nitrate System

PART 1

NEST
BOX M1 0.0 0.0 0.0 28.08 69.03 40.09
BOX M4 0.0 0.0 0.0 28.08 69.03 149.75
BOX M3 -2.53 -2.53 -2.04 33.14 74.09 154.67
BOX M2 -32.53 -32.53 -32.04 93.14 134.09 204.67

ALBEDO 0 0 0 0 0 0

END

**********************************************

BEGIN CONTROL DATA

STAGES -5     ! Start at stage number -5
       200    ! Finish at stage number 200
       1000   ! 1000 superhistories (neutrons)
              ! (10 generations per superhistory)
STDV 0.0010   ! Stop Calculation when Standard Deviation <=0.0010

END

**********************************************

BEGIN SOURCE GEOMETRY

ZONEMAT

ZONE 1 IN PART 1 /

END

**********************************************
```

Input File case80.01

```
* MONK VALIDATION CALCULATION 80.01
* --------------------------------
* ICSBEP EXPERIMENT: LEU-SOL-THERM-007 Case 1

* Calculation performed by D Hanlon - December 2001

* Summary of experiment
* --------------------
* Fissile Material:        10% enriched uranyl nitrate solution
* Geometry:                Cylindrical
* Neutron Poison:          None
* Reflector:               None
* Reference:               T Yamamoto, Y Miyoshi
*                          STACY: Unreflected 10%-Enriched Uranyl
*                          Nitrate Solution in a 60cm Diameter
*                          Cylindrical tank
*                          LEU-SOL-THERM-007 (30/09/99)
* Code Package:            MONK8B

* Critical Parameters Data -

* Uranium Concentration     : 313.0  gU/l
* Solution Height           :  46.83 cm

* Additional Notes -

* The experimental temperature was assumed to be 25 degrees C (298 K)
* MONK nuclear data temperature is at 20 degrees C.

* Keyword Parameters -
*
* solution height (height of solution above tank inner base)
*

@sol_ht=46.83

BEGIN MATERIAL SPECIFICATION

NMATERIALS 3

* material 1 - uranyl nitrate solution
* material 2 - stainless steel
* material 3 - air

ATOMS
MATERIAL 1 DENSITY 0.0
U234   PROP 6.4430E-07
U235   PROP 7.9954E-05
U236   PROP 7.9854E-08
U238   PROP 7.1216E-04
H1     PROP 5.6707E-02
N      PROP 2.9406E-03
O      PROP 3.8084E-02

ATOMS
MATERIAL 2 DENSITY 0.0
C      PROP 4.3736E-05
SI     PROP 1.0627E-03
MN     PROP 1.1561E-03
P      PROP 4.3170E-05
S      PROP 2.9782E-06
NI     PROP 8.3403E-03
CR     PROP 1.6775E-02
FE     PROP 5.9421E-02

ATOMS
```

```
MATERIAL 3 DENSITY 0.0
N      PROP 3.9016E-05
O      PROP 1.0409E-05

END

****************************************************************

BEGIN MATERIAL GEOMETRY

PART 1 NEST

ZROD M1  0.0 0.0   0.0  29.5 @sol_ht    ! fuel solution
ZROD M3  0.0 0.0   0.0  29.5 150.0      ! inside tank
ZROD M2  0.0 0.0  -2.0  29.8 154.5      ! tank wall

END

*****************************************************************

BEGIN CONTROL DATA
STAGES -1  200  1000  STDV 0.0010
END

************************************************************************

BEGIN SOURCE GEOMETRY
ZONEMAT
ZONE 1  PART 1 /
MATERIAL 1
END
```

Input File case81.01

```
columns 1 132
* MONK VALIDATION CALCULATION 81.01
* --------------------------------
* ICSBEP EXPERIMENT: LEU-SOL-THERM-008 Run 74

* Calculation performed by T Dean - January 2002

* Summary of experiment
* ---------------------
* Fissile Material:        10% enriched uranyl nitrate solution
* Geometry:                Cylindrical
* Neutron Poison:          None
* Reflector:               Concrete
* Reference:               T Kikuchi, Y Miyoshi
*                          STACY: 60-cm-Diameter Cylinders of
*                         10%-Enriched Uranyl Nitrate Solutions
*                          Reflected with Concrete
*                          LEU-SOL-THERM-008 (30/09/99)
* Code Package:            MONK8B

* Additional Notes -

* The experimental temperature was assumed to be 25 degrees C (298 K)
* MONK nuclear data temperature is at 20 degrees C.

* Keyword Parameters -
*
* @sol_ht = solution height (height of solution above tank inner base)
* @inngap =  inner gap (gap between core tank and concrete reflector)
* @outwall = outer wall thickness
* @reflthk = concrete reflector thickness

@sol_ht=79.99
@inngap=0.50
@outwall=0.80
@reflthk=4.94
******************************************************************************
BEGIN MATERIAL SPECIFICATION

NMATERIALS 7

* material 1 - uranyl nitrate solution
* material 2 - stainless steel (core tank)
* material 3 - air
* material 4 - aluminium (inner and outer reflector walls and lower reflector plate)
* material 5 - concrete
* material 6 - stainless steel (upper reflector plate)
* material 7 - stainless steel (reflector support disk)

ATOMS
MATERIAL 1 DENSITY 0.0
U234  PROP 4.9445E-07
U235  PROP 6.1357E-05
U236  PROP 6.1281E-08
U238  PROP 5.4652E-04
H1    PROP 5.8585E-02
N     PROP 2.4634E-03
O     PROP 3.7276E-02

ATOMS
MATERIAL 2 DENSITY 0.0
C     PROP 4.3736E-05
SI    PROP 1.0627E-03
MN    PROP 1.1561E-03
P     PROP 4.3170E-05
S     PROP 2.9782E-06
```

```
NI     PROP 8.3403E-03
CR     PROP 1.6775E-02
FE     PROP 5.9421E-02

ATOMS
MATERIAL 3 DENSITY 0.0
N      PROP 3.9016E-05
O      PROP 1.0409E-05

ATOMS
MATERIAL 4 DENSITY 0.0
AL     PROP 5.9523E-02
SI     PROP 5.7679E-05
TI     PROP 6.7667E-06
MN     PROP 2.9487E-06
FE     PROP 1.7114E-04

CU     PROP 3.5689E-05

ATOMS
MATERIAL 5 DENSITY 0.0
H1      PROP 1.6908E-02
O       PROP 4.5713E-02
NA      PROP 8.4727E-04
MG      PROP 4.9008E-04
AL      PROP 1.5864E-03
SI      PROP 1.5305E-02
S       PROP 9.1007E-05
CL      PROP 1.5797E-06
K       PROP 5.4725E-04
CA      PROP 2.2133E-03
FE      PROP 3.9747E-04

ATOMS
MATERIAL 6 DENSITY 0.0
C      PROP 1.9880E-04
SI     PROP 9.1819E-04
MN     PROP 1.0518E-03
P      PROP 4.0087E-05
S      PROP 5.9564E-06
NI     PROP 6.7699E-03
CR     PROP 1.6716E-02
FE     PROP 6.1269E-02

ATOMS
MATERIAL 7 DENSITY 0.0
C      PROP 1.5904E-04
SI     PROP 9.3519E-04
MN     PROP 1.1213E-03
P      PROP 4.4712E-05
S      PROP 2.9782E-06
NI     PROP 6.8512E-03
CR     PROP 1.6890E-02
FE     PROP 6.0951E-02

END

*****************************************************************

BEGIN MATERIAL GEOMETRY

PART 1 NEST

ZROD M1  0.0 0.0   0.0   29.5  @sol_ht      ! fuel solution
ZROD M3  0.0 0.0   0.0   29.5  149.86       ! inside tank
ZROD M2  0.0 0.0  -2.02  29.82 154.82       ! tank wall

PART 2 NEST
ZROD P1  0.0 0.0 1.98     29.82 154.82
```

```
ZROD BH1 0.0 0.0 0.0      68.5  156.8

END

*****************************************************************
BEGIN HOLE DATA

RZMESH
6
 [29.82+@inngap]             ! Tank Radius + inner gap
 [29.82+0.31+@inngap]        ! Tank Radius + inner gap + inner wall
 31.7                        ! Support plate hole radius
 [29.82+0.31+@inngap+@reflthk]         ! Hole radius + reflector thickness
 [29.82+0.31+@inngap+@reflthk+@outwall]  ! Hole radius + reflector thickness + outer
wall
 68.5                        ! Support plate radius
4
 0
 2.5                         ! Support plate
 [2.5+1.5]                   ! Support plate + reflector base
 [2.5+1.5+142.0]             ! Support plate + reflector base + reflector
 [2.5+1.5+142.0+0.6]         ! Support plate + reflector base + reflector + reflector
top
* Materials
0 0 0 7 7 7
0 4 4 4 4 0
0 4 5 5 4 0
0 6 6 6 6 0
0

END
*****************************************************************
BEGIN CONTROL DATA
STAGES -1  200  1000  STDV 0.0010
END

*****************************************************************

BEGIN SOURCE GEOMETRY
ZONEMAT
ZONE 1  PART 1 /
MATERIAL 1
END
```

Input File case84.01

```
columns 1 132
* MONK VALIDATION CALCULATION 84.01
* --------------------------------
* ICSBEP EXPERIMENT: LEU-SOL-THERM-009 Run 92

* Calculation performed by T Dean - March 2002

* Summary of experiment
* ---------------------
* Fissile Material:        10% enriched uranyl nitrate solution
* Geometry:                Cylindrical
* Neutron Poison:          None
* Reflector:               Concrete
* Reference:               T Kikuchi, Y Miyoshi
*                          STACY: 60-cm-Diameter Cylinders of
*                        10%-Enriched Uranyl Nitrate Solutions
*                          Reflected with Borated Concrete
*                          LEU-SOL-THERM-009 (30/09/99)
* Code Package:            MONK8B

* Additional Notes -

* The experimental temperature was assumed to be 25 degrees C (298 K)
* MONK nuclear data temperature is at 20 degrees C.

* Keyword Parameters -
*
* @sol_ht = solution height (height of solution above tank inner base)
* @inngap =  inner gap (gap between core tank and concrete reflector)
* @outwall = outer wall thickness
* @reflthk = concrete reflector thickness

@sol_ht=74.38
@inngap=0.47
@outwall=0.80
@reflthk=20.04
********************************************************************************
BEGIN MATERIAL SPECIFICATION

NMATERIALS 7

* material 1 - uranyl nitrate solution
* material 2 - stainless steel (core tank)
* material 3 - air
* material 4 - aluminium (inner and outer reflector walls and lower reflector plate)
* material 5 - borated concrete (B010)
* material 6 - stainless steel (upper reflector plate)
* material 7 - stainless steel (reflector support disk)

ATOMS
MATERIAL 1 DENSITY 0.0
U234  PROP 5.0371E-07
U235  PROP 6.2507E-05
U236  PROP 6.2429E-08
U238  PROP 5.5676E-04
H1    PROP 5.8493E-02
N     PROP 2.5043E-03
O     PROP 3.7367E-02

ATOMS
MATERIAL 2 DENSITY 0.0
C     PROP 4.3736E-05
SI    PROP 1.0627E-03
MN    PROP 1.1561E-03
P     PROP 4.3170E-05
S     PROP 2.9782E-06
```

```
NI      PROP 8.3403E-03
CR      PROP 1.6775E-02
FE      PROP 5.9421E-02

ATOMS
MATERIAL 3 DENSITY 0.0
N       PROP 3.9016E-05
O       PROP 1.0409E-05

ATOMS
MATERIAL 4 DENSITY 0.0
AL      PROP 5.9523E-02
SI      PROP 5.7679E-05
TI      PROP 6.7667E-06
MN      PROP 2.9487E-06
FE      PROP 1.7114E-04
CU      PROP 3.5689E-05

ATOMS
MATERIAL 5 DENSITY 0.0
H1       PROP 1.9421E-02
O        PROP 4.4070E-02
B10      PROP 1.1085E-04
B11      PROP 4.4618E-04
C        PROP 1.4039E-04
NA       PROP 2.4291E-04
MG       PROP 3.2722E-04
AL       PROP 6.7331E-04
SI       PROP 1.3594E-02
S        PROP 1.9104E-04
CL       PROP 1.2060E-06
K        PROP 1.7773E-04
CA       PROP 4.8293E-03
FE       PROP 2.0741E-04

ATOMS
MATERIAL 6 DENSITY 0.0
C       PROP 1.9880E-04
SI      PROP 9.1819E-04
MN      PROP 1.0518E-03
P       PROP 4.0087E-05
S       PROP 5.9564E-06
NI      PROP 6.7699E-03
CR      PROP 1.6716E-02
FE      PROP 6.1269E-02

ATOMS
MATERIAL 7 DENSITY 0.0
C       PROP 1.5904E-04
SI      PROP 9.3519E-04
MN      PROP 1.1213E-03
P       PROP 4.4712E-05
S       PROP 2.9782E-06
NI      PROP 6.8512E-03
CR      PROP 1.6890E-02
FE      PROP 6.0951E-02

END

*****************************************************************

BEGIN MATERIAL GEOMETRY

PART 1 NEST

ZROD M1  0.0 0.0    0.0    29.5  @sol_ht      ! fuel solution
ZROD M3  0.0 0.0    0.0    29.5  149.86       ! inside tank
ZROD M2  0.0 0.0   -2.02   29.82 154.82       ! tank wall
```

```
PART 2 NEST
ZROD P1  0.0 0.0 1.98     29.82 154.82
ZROD BH1 0.0 0.0 0.0      68.5  156.8

END

*****************************************************************
BEGIN HOLE DATA

RZMESH
6
 [29.82+@inngap]               ! Tank Radius + inner gap
 [29.82+0.31+@inngap]          ! Tank Radius + inner gap + inner wall
 31.7                          ! Support plate hole radius
 [29.82+0.31+@inngap+@reflthk]           ! Hole radius + reflector thickness
 [29.82+0.31+@inngap+@reflthk+@outwall]  ! Hole radius + reflector thickness + outer
wall
 68.5                          ! Support plate radius
4
 0
 2.5                           ! Support plate
 [2.5+1.5]                     ! Support plate + reflector base
 [2.5+1.5+142.0]               ! Support plate + reflector base + reflector
 [2.5+1.5+142.0+0.6]           ! Support plate + reflector base + reflector + reflector
top
* Materials
0 0 0 7 7 7
0 4 4 4 4 0
0 4 5 5 4 0
0 6 6 6 6 0
0

END
*****************************************************************
BEGIN CONTROL DATA
STAGES -1  200  1000
STDV 0.0010
END

*****************************************************************

BEGIN SOURCE GEOMETRY
ZONEMAT
ZONE 1  PART 1 /
MATERIAL 1
END
```

Input File case85.01

```
columns 1 132
* MONK VALIDATION CALCULATION 85.01
* --------------------------------
* ICSBEP EXPERIMENT: LEU-SOL-THERM-010 Run 83

* Calculation performed by T Dean - March 2002

* Summary of experiment
* ---------------------
* Fissile Material:         10% enriched uranyl nitrate solution
* Geometry:                 Cylindrical
* Neutron Poison:           None
* Reflector:                Polyethylene
* Reference:                T Kikuchi, Y Miyoshi
*                           STACY: 60-cm-Diameter Cylinders of
*                         10%-Enriched Uranyl Nitrate Solutions
*                           Reflected with Polyethylene
*                           LEU-SOL-THERM-010 (30/09/99)
* Code Package:             MONK8B

* Additional Notes -

* The experimental temperature was assumed to be 25 degrees C (298 K)
* MONK nuclear data temperature is at 20 degrees C.

* Keyword Parameters -
*
* @sol_ht = solution height (height of solution above tank inner base)
* @inngap =  inner gap (gap between core tank and concrete reflector)
* @outwall = outer wall thickness
* @reflthk = concrete reflector thickness

@sol_ht=81.26
@inngap=2.13
@innwall=0.30
@outwall=0.81
@reflthk=3.15
*****************************************************************************
BEGIN MATERIAL SPECIFICATION

NMATERIALS 7

* material 1 - uranyl nitrate solution
* material 2 - stainless steel (core tank)
* material 3 - air
* material 4 - aluminium (inner and outer reflector walls and lower reflector plate)
* material 5 - polyethylene (P30)
* material 6 - stainless steel (upper reflector plate)
* material 7 - stainless steel (reflector support disk)

ATOMS
MATERIAL 1 DENSITY 0.0
U234   PROP 4.9836E-07
U235   PROP 6.1843E-05
U236   PROP 6.1766E-08
U238   PROP 5.5084E-04
H1     PROP 5.8516E-02
N      PROP 2.4851E-03
O      PROP 3.7311E-02

ATOMS
MATERIAL 2 DENSITY 0.0
C      PROP 4.3736E-05
SI     PROP 1.0627E-03
MN     PROP 1.1561E-03
P      PROP 4.3170E-05
```

```
S       PROP 2.9782E-06
NI      PROP 8.3403E-03
CR      PROP 1.6775E-02
FE      PROP 5.9421E-02

ATOMS
MATERIAL 3 DENSITY 0.0
N       PROP 3.9016E-05
O       PROP 1.0409E-05

ATOMS
MATERIAL 4 DENSITY 0.0
AL      PROP 5.9523E-02
SI      PROP 5.7679E-05
TI      PROP 6.7667E-06
MN      PROP 2.9487E-06
FE      PROP 1.7114E-04
CU      PROP 3.5689E-05

ATOMS
MATERIAL 5 DENSITY 0.0
H1       PROP 7.8360E-02
C        PROP 3.9316E-02

ATOMS
MATERIAL 6 DENSITY 0.0
C       PROP 1.9880E-04
SI      PROP 9.1819E-04
MN      PROP 1.0518E-03
P       PROP 4.0087E-05
S       PROP 5.9564E-06
NI      PROP 6.7699E-03
CR      PROP 1.6716E-02
FE      PROP 6.1269E-02

ATOMS
MATERIAL 7 DENSITY 0.0
C       PROP 1.5904E-04
SI      PROP 9.3519E-04
MN      PROP 1.1213E-03
P       PROP 4.4712E-05
S       PROP 2.9782E-06
NI      PROP 6.8512E-03
CR      PROP 1.6890E-02
FE      PROP 6.0951E-02

USE DFN 370293 FOR H1 IN MATERIAL 5

END
****************************************************************

BEGIN MATERIAL GEOMETRY

PART 1 NEST

ZROD M1  0.0 0.0   0.0   29.5  @sol_ht      ! fuel solution
ZROD M3  0.0 0.0   0.0   29.5  149.86       ! inside tank
ZROD M2  0.0 0.0  -2.02  29.82 154.82       ! tank wall

PART 2 NEST
ZROD P1  0.0 0.0 1.98     29.82 154.82
ZROD BH1 0.0 0.0 0.0      68.5  156.8

END
****************************************************************
BEGIN HOLE DATA
```

```
RZMESH
6
 31.7                          ! Support plate hole radius
 [29.82+@inngap]               ! Tank Radius + inner gap
 [29.82+@innwall+@inngap]         ! Tank Radius + inner gap + inner wall
 [29.82+@innwall+@inngap+@reflthk]          ! Hole radius + reflector thickness
 [29.82+@innwall+@inngap+@reflthk+@outwall]  ! Hole radius + reflector thickness + outer
wall
 68.5                          ! Support plate radius
4
 0
 2.5                           ! Support plate
 [2.5+1.5]                     ! Support plate + reflector base
 [2.5+1.5+142.0]               ! Support plate + reflector base + reflector
 [2.5+1.5+142.0+0.6]           ! Support plate + reflector base + reflector + reflector
top
* Materials
0 7 7 7 7 7
0 0 4 4 4 0
0 0 4 5 4 0
0 0 6 6 6 0
0

END
****************************************************************
BEGIN CONTROL DATA
STAGES -1  200  1000
STDV 0.0010
END

****************************************************************

BEGIN SOURCE GEOMETRY
ZONEMAT
ZONE 1  PART 1 /
MATERIAL 1
END
```

Attachment 1B
Critical Experiment Parameters

Critical Experiment Parameters

Input file	Handbook ID	Run or Experiment number	Experimental Uncertainty	Fuel Solution	Reflector material	Tank shape	Dimension (cm)[1]	Critical height (cm)	Absorber	4V/S (mean cord length)[2]
case13.01	HEU-SOL-THERM-003	7	0.0049	Uranyl Nitrate	Plexiglas	cylinder	27.88	51.67	0	21.96
case13.02	HEU-SOL-THERM-002	7	0.0020	Uranyl Nitrate	concrete	cylinder	28.01	28.6	0	18.80
case13.03	HEU-SOL-THERM-002	8	0.0020	Uranyl Nitrate	concrete	cylinder	28.01	22.33	0	17.21
case13.04	HEU-SOL-THERM-003	10	0.0049	Uranyl Nitrate	Plexiglas	cylinder	28.01	28.84	0	18.85
case13.05	HEU-SOL-THERM-003	11	0.0049	Uranyl Nitrate	Plexiglas	cylinder	28.01	22.87	0	17.37
case13.06	HEU-SOL-THERM-002	9	0.0020	Uranyl Nitrate	concrete	cylinder	33.01	34.1	0	22.24
case13.07	HEU-SOL-THERM-002	10	0.0020	Uranyl Nitrate	concrete	cylinder	33.01	27.27	0	20.56
case13.08	HEU-SOL-THERM-003	12	0.0049	Uranyl Nitrate	Plexiglas	cylinder	33.01	34.33	0	22.29
case13.09	HEU-SOL-THERM-003	13	0.0049	Uranyl Nitrate	Plexiglas	cylinder	33.01	27.7	0	20.68
case13.10	HEU-SOL-THERM-002	11	0.0020	Uranyl Nitrate	concrete	cylinder	33.01	22.85	0	19.17
case13.11	HEU-SOL-THERM-002	12	0.0020	Uranyl Nitrate	concrete	cylinder	33.01	18.24	0	17.33
case13.12	HEU-SOL-THERM-003	16	0.0049	Uranyl Nitrate	Plexiglas	cylinder	33.01	22.78	0	19.14
case23.01	HEU-SOL-THERM-013	1	0.0026	Uranyl Nitrate	bare	sphere	69.42		0	46.28
case23.02	HEU-SOL-THERM-013	2	0.0036	Uranyl Nitrate	bare	sphere	69.42		boric acid	46.28

1. For a cylinder tank, the dimension represents the cylinder diameter; for a sphere, the sphere diameter; for a slab, the length and width.
2. Mean cord length is calculated as 4 times the volume divided by the surface area.

Input file	Handbook ID	Run or Experiment number	Experimental Uncertainty	Fuel Solution	Reflector material	Tank shape	Dimension (cm)[1]	Critical height (cm)	Absorber	4V/S (mean cord length)[2]
case23.03	HEU-SOL-THERM-013	3	0.0036	Uranyl Nitrate	bare	sphere	69.42		boric acid	46.28
case23.04	HEU-SOL-THERM-013	4	0.0036	Uranyl Nitrate	bare	sphere	69.42		boric acid	46.28
case23.05	NS&E 12,364 (1965)	10	0.0036	Uranyl Nitrate	bare	sphere	122.02		0	81.35
case35.01	HEU-SOL-THERM-009	1	0.0056	Uranium Oxyfluoride	water	sphere	11.52		0	7.68
case35.02	HEU-SOL-THERM-009	2	0.0056	Uranium Oxyfluoride	water	sphere	11.52		0	7.68
case35.03	HEU-SOL-THERM-009	3	0.0056	Uranium Oxyfluoride	water	sphere	11.5		0	7.67
case35.04	HEU-SOL-THERM-009	4	0.0056	Uranium Oxyfluoride	water	sphere	11.8		0	7.87
case35.05[3]	HEU-SOL-THERM-010	1	0.0056	Uranium Oxyfluoride	water	sphere	26.4		0	17.60
case35.06	HEU-SOL-THERM-010	2	0.0056	Uranium Oxyfluoride	water	sphere	26.4		0	17.60
case35.07	HEU-SOL-THERM-010	3	0.0056	Uranium Oxyfluoride	water	sphere	26.4		0	17.60
case35.08	HEU-SOL-THERM-010	4	0.0056	Uranium Oxyfluoride	water	sphere	26.4		0	17.60
case35.09	HEU-SOL-THERM-011	1	0.0018	Uranium Oxyfluoride	water	sphere	32		0	21.33
case35.10	HEU-SOL-THERM-011	2	0.0018	Uranium Oxyfluoride	water	sphere	32		0	21.33
case35.11	HEU-SOL-THERM-012	1	0.0058	Uranium Oxyfluoride	water	sphere	27.9		0	18.60

3. The report for this experiment states that not all of the typical contributors to the experimental uncertainty were reported. Therefore the uncertainty for a similar experiment (HEU-SOL-THERM-009) was substituted for case35.05 through case35.08.

Input file	Handbook ID	Run or Experiment number	Experimental Uncertainty	Fuel Solution	Reflector material	Tank shape	Dimension (cm)[1]	Critical height (cm)	Absorber	4V/S (mean cord length)[2]
case43.01	LEU-SOL-THERM-002	1	0.0040	Uranium Oxyfluoride	water	sphere	69.3	62.5	0	46.20
case43.02	LEU-SOL-THERM-002	2	0.0037	Uranium Oxyfluoride	bare	sphere	69.3	64.6	0	46.20
case43.03	LEU-SOL-THERM-002	3	0.0044	Uranium Oxyfluoride	water	sphere	69.3	51.4	0	46.20
case51.01	LEU-SOL-THERM-004	1	0.0008	Uranyl Nitrate	water	cylinder	59	41.53	0	34.50
case51.02	LEU-SOL-THERM-004	29	0.0009	Uranyl Nitrate	water	cylinder	59	46.7	0	36.16
case51.03	LEU-SOL-THERM-004	33	0.0009	Uranyl Nitrate	water	cylinder	59	52.93	0	37.89
case51.04	LEU-SOL-THERM-004	34	0.0010	Uranyl Nitrate	water	cylinder	59	64.85	0	40.55
case51.05	LEU-SOL-THERM-004	46	0.0010	Uranyl Nitrate	water	cylinder	59	78.56	0	42.89
case51.06	LEU-SOL-THERM-004	51	0.0011	Uranyl Nitrate	water	cylinder	59	95.5	0	45.08
case51.07	LEU-SOL-THERM-004	54	0.0011	Uranyl Nitrate	water	cylinder	59	130.33	0	48.11
case63.01	LEU-SOL-THERM-005	1	0.0041	Uranyl Nitrate	water	cylinder	110	58.98	0	56.92
case63.02	LEU-SOL-THERM-005	2	0.0050	Uranyl Nitrate	water	cylinder	110	62.25	1 B4C pin	58.40
case63.03	LEU-SOL-THERM-005	3	0.0063	Uranyl Nitrate	water	cylinder	110	106.62	7 B4C Pins	72.57
case67.01	HEU-SOL-THERM-001	1	0.0025	Uranyl Nitrate	bare	cylinder	33.01	31.2	0	21.59
case67.02	HEU-SOL-THERM-001	2	0.0025	Uranyl Nitrate	bare	cylinder	33.01	28.93	0	21.02
case67.03	HEU-SOL-THERM-001	3	0.0025	Uranyl Nitrate	bare	cylinder	33.01	33.55	0	22.13

Input file	Handbook ID	Run or Experiment number	Experimental Uncertainty	Fuel Solution	Reflector material	Tank shape	Dimension (cm)[1]	Critical height (cm)	Absorber	4V/S (mean cord length)[2]
case67.04	HEU-SOL-THERM-001	4	0.0025	Uranyl Nitrate	bare	cylinder	33.01	30.91	0	21.52
case67.05	HEU-SOL-THERM-001	5	0.0025	Uranyl Nitrate	bare	cylinder	33.01	39.48	0	23.28
case67.06	HEU-SOL-THERM-001	6	0.0025	Uranyl Nitrate	bare	cylinder	33.01	36.67	0	22.76
case67.07	HEU-SOL-THERM-001	7	0.0025	Uranyl Nitrate	bare	cylinder	33.01	23.96	0	19.55
case67.08	HEU-SOL-THERM-001	8	0.0025	Uranyl Nitrate	bare	cylinder	33.01	23.67	0	19.45
case67.09	HEU-SOL-THERM-001	9	0.0025	Uranyl Nitrate	bare	cylinder	33.01	22.53	0	19.05
case67.10	HEU-SOL-THERM-001	10	0.0025	Uranyl Nitrate	bare	cylinder	50.69	20.48	0	22.65
case68.01	HEU-SOL-THERM-004	1	0.0033	Uranium Oxyfluoride (heavy water)	heavy water	sphere	34.29		0	22.86
case68.02	HEU-SOL-THERM-004	2	0.0036	Uranium Oxyfluoride (heavy water)	heavy water	sphere	36.83		0	24.55
case68.03	HEU-SOL-THERM-004	3	0.0039	Uranium Oxyfluoride (heavy water)	heavy water	sphere	39.37		0	26.25
case68.04	HEU-SOL-THERM-004	4	0.0046	Uranium Oxyfluoride (heavy water)	heavy water	sphere	41.91		0	27.94

Input file	Handbook ID	Run or Experiment number	Experimental Uncertainty	Fuel Solution	Reflector material	Tank shape	Dimension (cm)[1]	Critical height (cm)	Absorber	4V/S (mean cord length)[2]
case68.05	HEU-SOL-THERM-004	5	0.0052	Uranium Oxyfluoride (heavy water)	heavy water	sphere	44.45		0	29.63
case68.06	HEU-SOL-THERM-004	6	0.0059	Uranium Oxyfluoride (heavy water)	heavy water	sphere	46.99		0	31.33
case71.01	LEU-SOL-THERM-016	105	0.0008	Uranyl Nitrate	water	slab	28 by 69	40.09	0	26.61
case71.02	LEU-SOL-THERM-016	113	0.0008	Uranyl Nitrate	water	slab	28 by 69	42.77	0	27.18
case71.03	LEU-SOL-THERM-016	125	0.0009	Uranyl Nitrate	water	slab	28 by 69	51.37	0	28.71
case71.04	LEU-SOL-THERM-016	129	0.0010	Uranyl Nitrate	water	slab	28 by 69	56.96	0	29.51
case71.05	LEU-SOL-THERM-016	131	0.0010	Uranyl Nitrate	water	slab	28 by 69	66.39	0	30.64
case71.06	LEU-SOL-THERM-016	140	0.0011	Uranyl Nitrate	water	slab	28 by 69	81.47	0	32.01
case71.07	LEU-SOL-THERM-016	196	0.0012	Uranyl Nitrate	water	slab	28 by 69	102.34	0	33.35
case80.01	LEU-SOL-THERM-007	14	0.0009	Uranyl Nitrate	bare	cylinder	59	46.83	0	36.20
case80.02	LEU-SOL-THERM-007	30	0.0009	Uranyl Nitrate	bare	cylinder	59	54.2	0	38.21
case80.03	LEU-SOL-THERM-007	32	0.0009	Uranyl Nitrate	bare	cylinder	59	63.55	0	40.30
case80.04	LEU-SOL-THERM-007	36	0.0010	Uranyl Nitrate	bare	cylinder	59	83.55	0	43.60
case80.05	LEU-SOL-THERM-007	49	0.0011	Uranyl Nitrate	bare	cylinder	59	112.27	0	46.72

Input file	Handbook ID	Run or Experiment number	Experimental Uncertainty	Fuel Solution	Reflector material	Tank shape	Dimension (cm)[1]	Critical height (cm)	Absorber	4V/S (mean cord length)[2]
case81.01	LEU-SOL-THERM-008	74	0.0011	Uranyl Nitrate	concrete	cylinder	59	79.99	0	43.10
case81.02	LEU-SOL-THERM-008	76	0.0010	Uranyl Nitrate	concrete	cylinder	59	73.5	0	42.10
case81.03	LEU-SOL-THERM-008	78	0.0010	Uranyl Nitrate	concrete	cylinder	59	70.58	0	41.61
case81.04	LEU-SOL-THERM-008	72	0.0010	Uranyl Nitrate	concrete	cylinder	59	71.71	0	41.80
case84.01	LEU-SOL-THERM-009	92	0.0009	Uranyl Nitrate	borated concrete	cylinder	59	74.38	0	42.25
case84.02	LEU-SOL-THERM-009	93	0.0009	Uranyl Nitrate	borated concrete	cylinder	59	77.29	0	42.70
case84.03	LEU-SOL-THERM-009	94	0.0009	Uranyl Nitrate	borated concrete	cylinder	59	78.88	0	42.94
case85.01	LEU-SOL-THERM-010	83	0.0011	Uranyl Nitrate	polyethylene	cylinder	59	81.26	0	43.29
case85.02	LEU-SOL-THERM-010	85	0.0010	Uranyl Nitrate	polyethylene	cylinder	59	77.81	0	42.78
case85.03	LEU-SOL-THERM-010	86	0.0010	Uranyl Nitrate	polyethylene	cylinder	59	76.92	0	42.64
case85.04	LEU-SOL-THERM-010	88	0.0010	Uranyl Nitrate	polyethylene	cylinder	59	76.42	0	42.57

Attachment 1C
Table of Key Results

Table of Key Results

Case	Experimental Uncertainty	Enrichment ($^w/_o$)	$H/^{235}U$ (number ratio)	Density (gm/cm^3)	Reflector Material	Fuel Solution	Tank Shape	Mean Cord Length (cm)	Absorber	Monk K eff	Monk Std Dev	Total Uncertainty [1]
case13.01	0.0049	93.17	4.54E+02	1.08141	Plexiglas	Uranyl Nitrate	cylinder	21.96	0	1.0053	0.0010	0.0050
case13.02	0.0020	93.17	7.35E+01	1.46116	concrete	Uranyl Nitrate	cylinder	18.80	0	1.0076	0.0010	0.0022
case13.03	0.0020	93.17	7.35E+01	1.46116	concrete	Uranyl Nitrate	cylinder	17.21	0	1.0153	0.0010	0.0022
case13.04	0.0049	93.17	7.09E+01	1.47545	Plexiglas	Uranyl Nitrate	cylinder	18.85	0	1.0043	0.0010	0.0050
case13.05	0.0049	93.17	7.09E+01	1.47545	Plexiglas	Uranyl Nitrate	cylinder	17.37	0	1.0103	0.0010	0.0050
case13.06	0.0020	93.17	4.59E+02	1.08021	concrete	Uranyl Nitrate	cylinder	22.24	0	1.0023	0.0010	0.0022
case13.07	0.0020	93.17	4.59E+02	1.08021	concrete	Uranyl Nitrate	cylinder	20.56	0	1.0094	0.0010	0.0022
case13.08	0.0049	93.17	4.54E+02	1.08141	Plexiglas	Uranyl Nitrate	cylinder	22.29	0	1.0048	0.0010	0.0050
case13.09	0.0049	93.17	4.54E+02	1.08141	Plexiglas	Uranyl Nitrate	cylinder	20.68	0	1.0053	0.0010	0.0050
case13.10	0.0020	93.17	1.84E+02	1.19996	concrete	Uranyl Nitrate	cylinder	19.17	0	1.0072	0.0010	0.0022
case13.11	0.0020	93.17	1.84E+02	1.19996	concrete	Uranyl Nitrate	cylinder	17.33	0	1.0158	0.0010	0.0022
case13.12	0.0049	93.17	1.80E+02	1.20456	Plexiglas	Uranyl Nitrate	cylinder	19.14	0	1.0035	0.0010	0.0050
case23.01	0.0026	93.18	1.38E+03	1.03112	bare	Uranyl Nitrate	sphere	46.28	0	0.9959	0.0010	0.0028
case23.02	0.0036	93.18	1.18E+03	1.03672	bare	Uranyl Nitrate	sphere	46.28	boric acid	0.9987	0.0010	0.0037
case23.03	0.0036	93.18	1.03E+03	1.04218	bare	Uranyl Nitrate	sphere	46.28	boric acid	0.9932	0.0010	0.0037
case23.04	0.0036	93.18	9.72E+02	1.04515	bare	Uranyl Nitrate	sphere	46.28	boric acid	0.9969	0.0010	0.0037
case23.05	0.0036	93.20	1.83E+03	1.02160	bare	Uranyl Nitrate	sphere	81.35	0	1.0003	0.0010	0.0037
case35.01	0.0056	93.18	3.58E+01	1.79447	water	Uranium Oxyfluoride	sphere	7.68	0	1.0072	0.0010	0.0057
case35.02	0.0056	93.18	4.72E+01	1.62004	water	Uranium Oxyfluoride	sphere	7.68	0	1.0046	0.0010	0.0057
case35.03	0.0056	93.18	7.61E+01	1.39990	water	Uranium Oxyfluoride	sphere	7.67	0	1.0040	0.0010	0.0057
case35.04	0.0056	93.13	2.70E+02	1.11539	water	Uranium Oxyfluoride	sphere	7.87	0	0.9985	0.0010	0.0057
case35.05	0.0056	93.18	1.26E+02	1.23901	water	Uranium Oxyfluoride	sphere	17.60	0	0.9963	0.0010	0.0057
case35.06	0.0056	93.13	2.64E+02	1.11313	water	Uranium Oxyfluoride	sphere	17.60	0	1.0008	0.0010	0.0057
case35.07	0.0056	93.13	2.46E+02	1.10106	water	Uranium Oxyfluoride	sphere	17.60	0	1.0006	0.0010	0.0057
case35.08	0.0056	93.13	2.39E+02	1.09553	water	Uranium Oxyfluoride	sphere	17.60	0	0.9973	0.0010	0.0057

1. Total Uncertainty is the statistical combination of the Experimental Uncertainty (σ_e) and the Monk Standard Deviation (i.e., σ_s)

Case	Experimental Uncertainty	Enrichment ($^{w}/_{o}$)	$H/^{235}U$ (number ratio)	Density (gm/cm^3)	Reflector Material	Fuel Solution	Tank Shape	Mean Cord Length (cm)	Absorber	Monk K eff	Monk Std Dev	Total Uncertainty [1]
case35.09	0.0018	93.18	5.23E+02	1.05923	water	Uranium Oxyfluoride	sphere	21.33	0	1.0043	0.0010	0.0021
case35.10	0.0018	93.18	5.33E+02	1.05911	water	Uranium Oxyfluoride	sphere	21.33	0	1.0007	0.0010	0.0021
case35.11	0.0058	93.18	1.27E+03	1.02600	water	Uranium Oxyfluoride	sphere	18.60	0	1.0013	0.0010	0.0059
case43.01	0.0040	4.89	1.10E+03	1.51573	water	Uranium Oxyfluoride	sphere	46.20	0	0.9984	0.0010	0.0041
case43.02	0.0037	4.89	1.00E+03	1.55873	bare	Uranium Oxyfluoride	sphere	46.20	0	0.9955	0.0010	0.0038
case43.03	0.0044	4.89	1.00E+03	1.55873	water	Uranium Oxyfluoride	sphere	46.20	0	0.9997	0.0010	0.0045
case51.01	0.0008	9.97	7.19E+02	1.47998	water	Uranyl Nitrate	cylinder	34.50	0	0.9996	0.0010	0.0013
case51.02	0.0009	9.97	7.71E+02	1.45450	water	Uranyl Nitrate	cylinder	36.16	0	0.9997	0.0010	0.0013
case51.03	0.0009	9.97	8.42E+02	1.43209	water	Uranyl Nitrate	cylinder	37.89	0	0.9988	0.0010	0.0013
case51.04	0.0010	9.97	8.96E+02	1.40631	water	Uranyl Nitrate	cylinder	40.55	0	0.9996	0.0010	0.0014
case51.05	0.0010	9.97	9.42E+02	1.39092	water	Uranyl Nitrate	cylinder	42.89	0	1.0003	0.0010	0.0014
case51.06	0.0011	9.97	9.83E+02	1.38211	water	Uranyl Nitrate	cylinder	45.08	0	0.9992	0.0010	0.0015
case51.07	0.0011	9.97	1.02E+03	1.36952	water	Uranyl Nitrate	cylinder	48.11	0	0.9977	0.0010	0.0015
case63.01	0.0041	5.64	9.72E+02	1.58722	water	Uranyl Nitrate	cylinder	56.92	0	0.9984	0.0010	0.0042
case63.02	0.0050	5.64	9.72E+02	1.58722	water	Uranyl Nitrate	cylinder	58.40	1 B4C pin	0.9977	0.0010	0.0051
case63.03	0.0063	5.64	9.72E+02	1.58722	water	Uranyl Nitrate	cylinder	72.57	7 B4C pins	0.9972	0.0010	0.0064
case67.01	0.0025	93.17	1.82E+02	1.20354	bare	Uranyl Nitrate	cylinder	21.59	0	0.9994	0.0010	0.0027
case67.02	0.0025	93.17	7.06E+01	1.47972	bare	Uranyl Nitrate	cylinder	21.02	0	1.0017	0.0010	0.0027
case67.03	0.0025	93.17	1.86E+02	1.20042	bare	Uranyl Nitrate	cylinder	22.13	0	1.0043	0.0010	0.0027
case67.04	0.0025	93.17	6.82E+01	1.49482	bare	Uranyl Nitrate	cylinder	21.52	0	1.0066	0.0010	0.0027
case67.05	0.0025	93.17	4.99E+02	1.07554	bare	Uranyl Nitrate	cylinder	23.28	0	1.0006	0.0010	0.0027
case67.06	0.0025	93.17	4.59E+02	1.08224	bare	Uranyl Nitrate	cylinder	22.76	0	1.0031	0.0010	0.0027
case67.07	0.0025	93.17	1.93E+02	1.19203	bare	Uranyl Nitrate	cylinder	19.55	0	1.0005	0.0010	0.0027
case67.08	0.0025	93.17	1.82E+02	1.20354	bare	Uranyl Nitrate	cylinder	19.45	0	1.0020	0.0010	0.0027
case67.09	0.0025	93.17	6.82E+01	1.49482	bare	Uranyl Nitrate	cylinder	19.05	0	0.9983	0.0010	0.0027
case67.10	0.0025	93.17	4.27E+02	1.08805	bare	Uranyl Nitrate	cylinder	22.65	0	0.9953	0.0010	0.0027
case68.01	0.0033	93.65	1.03E-01	1.92960	heavy water	(heavy water) Uranium Oxyfluoride	sphere	22.86	0	1.0042	0.0010	0.0034

Case	Experimental Uncertainty	Enrichment ($^{w}/_{o}$)	$H/^{235}U$ (number ratio)	Density (gm/cm^3)	Reflector Material	Fuel Solution	Tank Shape	Mean Cord Length (cm)	Absorber	Monk K eff	Monk Std Dev	Total Uncertainty [1]
case68.02	0.0036	93.65	1.61E-01	1.62677	heavy water	(heavy water) Uranium Oxyfluoride	sphere	24.55	0	1.0005	0.0010	0.0037
case68.03	0.0039	93.65	2.44E-01	1.46263	heavy water	(heavy water) Uranium Oxyfluoride	sphere	26.25	0	1.0083	0.0010	0.0040
case68.04	0.0046	93.65	4.06E-01	1.32215	heavy water	(heavy water) Uranium Oxyfluoride	sphere	27.94	0	1.0086	0.0010	0.0047
case68.05	0.0052	93.65	7.29E-01	1.22022	heavy water	(heavy water) Uranium Oxyfluoride	sphere	29.63	0	1.0051	0.0010	0.0053
case68.06	0.0059	93.65	1.29E+00	1.18428	heavy water	(heavy water) Uranium Oxyfluoride	sphere	31.33	0	1.0008	0.0010	0.0059
case71.01	0.0008	9.97	4.69E+02	1.64592	water	Uranyl Nitrate	slab	26.61	0	1.0081	0.0010	0.0013
case71.02	0.0008	9.97	5.14E+02	1.59941	water	Uranyl Nitrate	slab	27.18	0	1.0041	0.0010	0.0013
case71.03	0.0009	9.97	6.08E+02	1.52341	water	Uranyl Nitrate	slab	28.71	0	1.0032	0.0010	0.0013
case71.04	0.0010	9.97	6.50E+02	1.49539	water	Uranyl Nitrate	slab	29.51	0	1.0050	0.0010	0.0014
case71.05	0.0010	9.97	6.99E+02	1.46621	water	Uranyl Nitrate	slab	30.64	0	1.0017	0.0010	0.0014
case71.06	0.0011	9.97	7.39E+02	1.44620	water	Uranyl Nitrate	slab	32.01	0	1.0014	0.0010	0.0015
case71.07	0.0012	9.97	7.72E+02	1.43151	water	Uranyl Nitrate	slab	33.35	0	1.0040	0.0010	0.0016
case80.01	0.0009	9.97	7.09E+02	1.48539	bare	Uranyl Nitrate	cylinder	36.20	0	0.9928	0.0010	0.0013
case80.02	0.0009	9.97	7.70E+02	1.45439	bare	Uranyl Nitrate	cylinder	38.21	0	0.9983	0.0010	0.0013
case80.03	0.0009	9.97	8.42E+02	1.43209	bare	Uranyl Nitrate	cylinder	40.30	0	0.9974	0.0010	0.0013
case80.04	0.0010	9.97	8.96E+02	1.40751	bare	Uranyl Nitrate	cylinder	43.60	0	0.9993	0.0010	0.0014
case80.05	0.0011	9.97	9.42E+02	1.39143	bare	Uranyl Nitrate	cylinder	46.72	0	0.9980	0.0010	0.0015
case81.01	0.0011	9.97	9.55E+02	1.38322	concrete	Uranyl Nitrate	cylinder	43.10	0	1.0004	0.0010	0.0015
case81.02	0.0010	9.97	9.52E+02	1.38404	concrete	Uranyl Nitrate	cylinder	42.10	0	1.0007	0.0010	0.0014
case81.03	0.0010	9.97	9.51E+02	1.38473	concrete	Uranyl Nitrate	cylinder	41.61	0	1.0011	0.0010	0.0014
case81.04	0.0010	9.97	9.56E+02	1.38253	concrete	Uranyl Nitrate	cylinder	41.80	0	1.0002	0.0010	0.0014
case84.01	0.0009	9.97	9.36E+02	1.39093	borated concrete	Uranyl Nitrate	cylinder	42.25	0	0.9993	0.0010	0.0013
case84.02	0.0009	9.97	9.34E+02	1.39142	borated concrete	Uranyl Nitrate	cylinder	42.70	0	1.0024	0.0010	0.0013
case84.03	0.0009	9.97	9.33E+02	1.39193	borated concrete	Uranyl Nitrate	cylinder	42.94	0	0.9989	0.0010	0.0013

Case	Experimental Uncertainty	Enrichment ($^{w}/_{o}$)	$H/^{235}U$ (number ratio)	Density (gm/cm^3)	Reflector Material	Fuel Solution	Tank Shape	Mean Cord Length (cm)	Absorber	Monk K eff	Monk Std Dev	Total Uncertainty [1]
case85.01	0.0011	9.97	9.46E+02	1.38644	polyethylene	Uranyl Nitrate	cylinder	43.29	0	1.0014	0.0010	0.0015
case85.02	0.0010	9.97	9.45E+02	1.38722	polyethylene	Uranyl Nitrate	cylinder	42.78	0	1.0016	0.0010	0.0014
case85.03	0.0010	9.97	9.44E+02	1.38774	polyethylene	Uranyl Nitrate	cylinder	42.64	0	1.0005	0.0010	0.0014
case85.04	0.0010	9.97	9.42E+02	1.38853	polyethylene	Uranyl Nitrate	cylinder	42.57	0	1.0006	0.0010	0.0014