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February 17, 2006

FEDERAL EXPRESS

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DOCKETED
USNRC

February 21, 2006 (2:55pm)

OFFICE OF SECRETARY
RULEMAKINGS AND
ADJUDICATIONS STAFF

Re: LOUISIANA ENERGY SERVICES, L.P. (National Enrichment Facility)
Docket No. 70-3103-ML

Dear Administrative Judges:

As a follow-up to a discussion that occurred during the a teleconference with the Board on February 6, 2006, enclosed is a February 16, 2006 letter to the NRC staff (NEF#06-003) forwarding Revision 2 of the MONK 8A Validation and Verification Report, as well as certain conforming changes to the Safety Analysis Report. If you should have any questions regarding these documents, please contact me.

Yours sincerely,

James R. Curtiss

cc: Office of the Secretary (w/encls.)
Lisa Clark, Esq. (w/o encls.)

DC:456038.1

Template=SECY-049

SECY-02

February 16, 2006
NEF#06-003

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 and 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"

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"

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"

11. Letter NEF#05-034 dated December 22, 2005, R. M. Krich (Louisiana Energy Services, L. P.) to Director, Office of Nuclear Material Safety and Safeguards (NRC) regarding "Revised MONK 8A Validation and Verification"

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. The Reference 11 letter provided Revision 1 of the MONK 8A Validation and Verification report. In telephone calls between LES and NRC representatives, held on January 27, January 30, February 1, February 2, and February 8, 2006, and during an NRC in-house review held in LES offices on February 10, 2006, LES agreed to make certain revisions to the MONK 8A Validation and Verification report previously submitted in the Reference 11 letter. LES also agreed to update the License Application to reflect the results of the revised MONK 8A Validation and Verification report. Enclosure 1 provides Revision 2 of the MONK 8A Validation and Verification report. Enclosure 2 provides the updated Safety Analysis Report (SAR) pages (i.e., Revision 8). To facilitate the incorporation of the revised pages into the License Application, page removal and insertion instructions are also provided in Enclosure 2. No changes are made to the Integrated Safety Analysis (ISA) Summary, the Environmental Report, the Emergency Plan, the Physical Security Plan, the Safeguards Contingency Plan, the Guard Force Training and Qualification Plan, the Standard Practice Procedures Plan for the Protection of Classified Matter, or the Fundamental Nuclear Material Control Plan.

The License Application and ISA Summary, updated through Revision 8 of the SAR, continue to meet the applicable requirements of 10 CFR 70.22, "Contents of applications," 10 CFR 40.31, "Application for specific licenses," and 10 CFR 30.32, "Application for specific licenses," as described in the Reference 1 letter.

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

Respectfully,

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

Enclosures:

1. MONK 8A Validation and Verification, National Enrichment Facility, Revision 2
2. Updated Safety Analysis Report pages

cc: T. C. Johnson, NRC Project Manager

ENCLOSURE 1

MONK 8A Validation and Verification
National Enrichment Facility
Revision 2

MONK 8A
Validation and Verification

National Enrichment Facility

Revision 2

February 16, 2006

AREVA

Prepared by: Barbara Y. Hubbard 02/16/06
Barbara Y. Hubbard (Date)
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Prepared by: Glen Seeburger 2/16/06
Glen Seeburger (Date)
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Reviewed by: Grace M. Lam Feb 16, 2006
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George A. Harper (Date)
Manager, Regulatory Compliance

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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 93 benchmark critical experiments. The average of the validation runs was 1.0017 ± 0.0045. A subset of these experiments was selected to compare against the MONK 8A benchmark performed by the computer code vendor for the purpose of verification. The average of the verification runs was 1.0001 ± 0.0005. This was in good agreement with the average of the corresponding MONK 8A benchmarks of 1.0000 ± 0.0006 performed by the computer code vendor. Also, 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.

Revision 1 of this report expanded and reformatted the report to add more detail to ensure that the report addressed all of the commitments made in Chapter 5 of the National Enrichment Facility Safety Analysis Report (Reference 11).

Two specific items included in the report are the description of the Area of Applicability (AOA) and determination of the Upper Safety Limit (USL).

Revision 2 of this report removed the High Enriched Uranium benchmark critical experiments from the validation and added two additional Low Enriched Uranium critical experiments and one additional Intermediate Enriched Uranium critical experiment to the validation. This approach is more representative of the enrichments associated with the National Enrichment Facility and still maintains the range of the Hydrogen/Uranium ratio inside the area of applicability.

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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 11 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) and from NUREG/CR-1071 (Reference 13). 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- Δ_{sm}-Δ_{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 lie 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/U). The variable "y" represents k_{eff}. Variables "a" and "b" are coefficients for the function.

The equations used to produce 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, in-leakage 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 fall within a normal distribution. Therefore, it is appropriate to arrive at the USL using the Single-Sided Tolerance Limit technique discussed in Section 3.3. 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	Mean Log Energy of Neutron Causing Fission (MeV)
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	4.92E-8 to 2.7E-7
Contingency Dump System	Sodium Fluoride Traps	$UO_2F_2.3.5H_2O$	1.5 $^{w}/_{o}$ ^{235}U	7	4.92E-8 to 2.7E-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	4.92E-8 to 2.7E-7

5 Benchmark Experiments

5.1 Uranyl Fluoride/ Water Mixture

Ten plant specific benchmark experiments, consisting of 83 critical configurations, with uranyl solutions and compounds are selected from the International Handbook of Evaluated Criticality Safety Benchmark Experiments (Reference 4) to provide a good statistical base. In addition, an additional benchmark experiment, consisting of 10 critical configurations, was selected from the literature to add additional low enriched, low H/U ratio critical experiments (Reference 13). 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 twelve 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 from Reference 13 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 11 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., 1.0003±0.0027 for the uranyl nitrate cases compared to 0.9979±0.0022 for the uranyl oxyfluoride cases). Therefore, these benchmark cases were included. Also included were non solution cases involving UF_4, UO_2 and U_3O_8. Since oxygen is almost transparent to thermal neutrons UO_2 and U_3O_8 are similar to uranyl fluoride in its neutronic behavior and is therefore is appropriate to included in the benchmark. These cases are included because they expand the H/U ratio range down to 0.787. Uranium fluoride is also similar in its neutronic behavior to uranyl fluoride and therefore is appropriate to use. The H/U ratio varies from 1 to 32 for the NEF NCSA. and ranges from 0.787 to 103 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 4.46 to 29.83 %. 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 4.46 $^{w}/_{o}$ to 1.5 $^{w}/_{o}$).

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.

Table 5-1 Uranium Solution Experiments Used for Validation

MONK 8A Case Set	Case Description	Number of Experiments	Handbook Reference (Reference 4
25	Low-enriched damp U_3O_8 powder in cubic aluminum cans	10	NUREG/CR-1071
42	Low-enriched damp UO_2 powder reflected by polyethylene	18	LEU-COMP-THERM-049
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
69	Critical arrays of polyethylene-moderated $U(30)F_4$-Polytetrafluoroethylene one-inch cubes	29	IEU-COMP-THERM-001
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 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
NUREG/CR-1071	Critical Experiments with Interstitially-Moderated Arrays of Low-Enriched Uranium Oxide	The critical separation between two tables supporting arrays of cans containing low-enriched uranium oxide has been measured for twenty-one (21) reflected configurations having interstitial layers of moderating material between cans. The critical separation varied between 0.23 and 1.84 cm. The uranium oxide (U_3O_8) is enriched to 4.46%^{235}U, compacted to a density of 4.7 g/cm^3, and adjusted to an H/U atomic ratio of 0.77 by the addition of water. Each can weighs ~ 16 kg and is a 15.3 cm cube. Interstitial plastic moderator 1.0, 1.3, or 2.5 cm thick separates cans of the three-dimensional array.
LEU-COMP-THERM - 049	MARACAS Programme: Polythene-Reflected Critical Configurations with Low-Enriched and Low-Moderated Uranium Dioxide Powder, UO_2	The experiments considered in this program were low-water-moderated uranium dioxide (5wt.% enrichment) powder assemblies, with 'polythene' (polyethylene) reflection. Experiments were carried out using the split-table testing equipment called "MARACAS" in the experimental criticality facility at Valduc, near Dijon, France, in 1983-1987. Uranium dioxide powder was apportioned into boxes each containing 24 kg of dry oxide. The powder was moistened and the boxes were piled on a split table. The parallelepiped assembly was reflected by a 20-cm-thick polythene reflector. The subcritical approach parameter was the distance between the two half tables.
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.

Table 5-2 Expanded Descriptions of the Criticality Experiments

Handbook Reference	Title	Short Description
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.
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.

Table 5-2 Expanded Descriptions of the Criticality Experiments

Handbook Reference	Title	Short Description
IEU-COMP-THERM-001	Critical Arrays of Polyethylene-Moderated $U(30)F_4$-Polytetrafluoroethylene One-Inch Cubes	One-inch cubes of $U(30)F_4$-polytetrafluoroethylene[$(CF_2)_n$], 29.83 $^w/o$ U^{235} ("U-cubes") were stacked with one inch cubes and half-cubes of polyethylene ("H-cubes") into cuboid shapes on two aluminum platforms, one movable. Blocks were added until criticality was achieved when the two cuboids were brought together. Most critical cores were reflected by paraffin. Sheets of cadmium or boron surrounded the core in a few cases. Twenty-nine ratios and patterns of "U-cubes" and "H-cubes" were reported in sufficient detail to qualify as acceptable benchmark experiments.
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.

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

	Chemical Form	Isotopics	Hydrogen/Uranium Ratio	Mean Log Energy of Neutron Causing Fission (MeV)
NEF Nuclear Criticality Safety Analysis, except Contingency Dump System	Uranyl fluoride	6 $^{w}/_{o}$ ^{235}U	1 to 32	4.92E-8 to 2.7E-7
NEF Nuclear Criticality Safety Analysis, Contingency Dump System	Uranyl fluoride	1.5 $^{w}/_{o}$ ^{235}U	7	4.92E-8 to 2.7E-7
Benchmark	Uranyl Nitrate Uranium Fluoride Uranium Oxyfluoride UO2 powder, U3O8 powder	4.46 to 29.83 $^{w}/_{o}$ ^{235}U	0.787 to 103	3.78E-8 to 2.7E-1

6 Analysis of Validation Results

6.1 Uranyl Fluoride/Water Mixture

Ninety three 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
- Concrete reflected cylinder,
- Borated concrete reflected cylinder,
- Polyethylene reflected cylinder,
- Bare (unreflected) cylinder
- Bare (unreflected) sphere.
- Plexiglas Reflected array
- Polyethylene reflected array
- Bare slab
- Paraffin slab

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 the results have been shown to be a normal distribution. Therefore, the single-sided tolerance limit technique is applied to the data. The results are analyzed statistically using four trending parameters: Solution Density, H/U ratio, ^{235}U enrichment, and Mean Cord Length.

The solution density goes from 1.3695 to 4.6 g/cc, the H/U ratio goes from 0.787 to 103, the ^{235}U enrichment goes from 4.46 $^{w}/_{o}$ to 29.83 $^{w}/_{o}$, the cord length goes from 6.97 to 72.57 cm and the mean log energy of neutron causing fission goes from 3.78E-8 to 2.7E-1 MeV.

The cord length values for the array critical benchmark experiments, experiments 25 and 42, are not included. The geometry of the configuration for experiments 25 and 42 is different than the geometry of the configurations for the other experiments included in the validation (e.g., arrays versus a single solid object), as such, a comparison of cord length between experiments would not be a meaningful, therefore, the cord length values for these experiments are not calculated. Geometry is not considered as important as material specifications and neutron energy when determining the acceptability of critical experiments (Reference 8). As discussed in Section 5.1, the materials for these experiments are acceptable for use in this validation and as shown in Table 5-3 and Appendix 1.C, experiments 25 and 42 cover the lower portion of the neutron energy range for the AOA.

Table 6-1 summarizes the statistical results. Figures 6-2 through Figure 6-5 show the results graphically.

Using the one-sided lower tolerance limit equation:

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

Where, $\overline{k_{eff}}$ is determined from the analysis to be 1.0017 and set to 1.000 and Sp is determined from the analysis to be 0.0041.

Since the sample size is 93, U is conservatively determined from Table 3-2 to be 2.065 and provides for a 95% confidence that 95% of the population lies within this range. As a result, the lower tolerance limit is as follows:

K_L = 1.0- 2.065*0.0041 =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 4.46 $^w/_o$ to 1.5 $^w/_o$); therefore, the margin required to extrapolate beyond the AOA (Δ_{AOA}) is set to 0.0014. 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 2.495E-04 (k_{eff} per % enrichment) for the low enrichment cases. The extrapolation penalty is then calculated to be:

(4.46-1.5) x 2.495E-04=0.0007

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 within the AOA range for the 1.5 $^{w}/_{o}$ enrichment value. As such, the plant specific benchmark does not contain any critical experiments 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. 0007 (in addition to the 0.0007 penalty calculated above). The resultant Δ_{AOA} is the sum of these two penalties (i.e., 0.0014).

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 = 0.9915-0.05-0.0014

  USL = 0.9401

Figure 6-1 MONK k effective Histogram

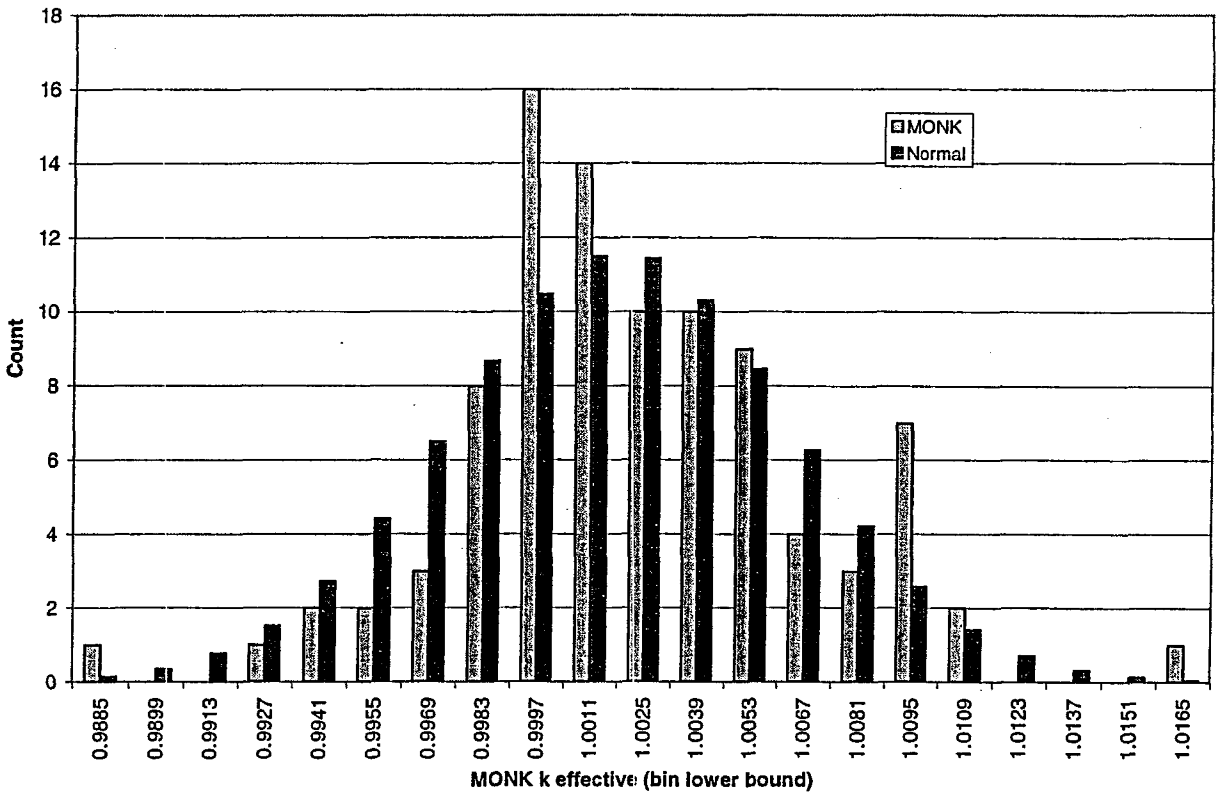

Figure 6-2 Plot of MONK k effective vs. Solution Density

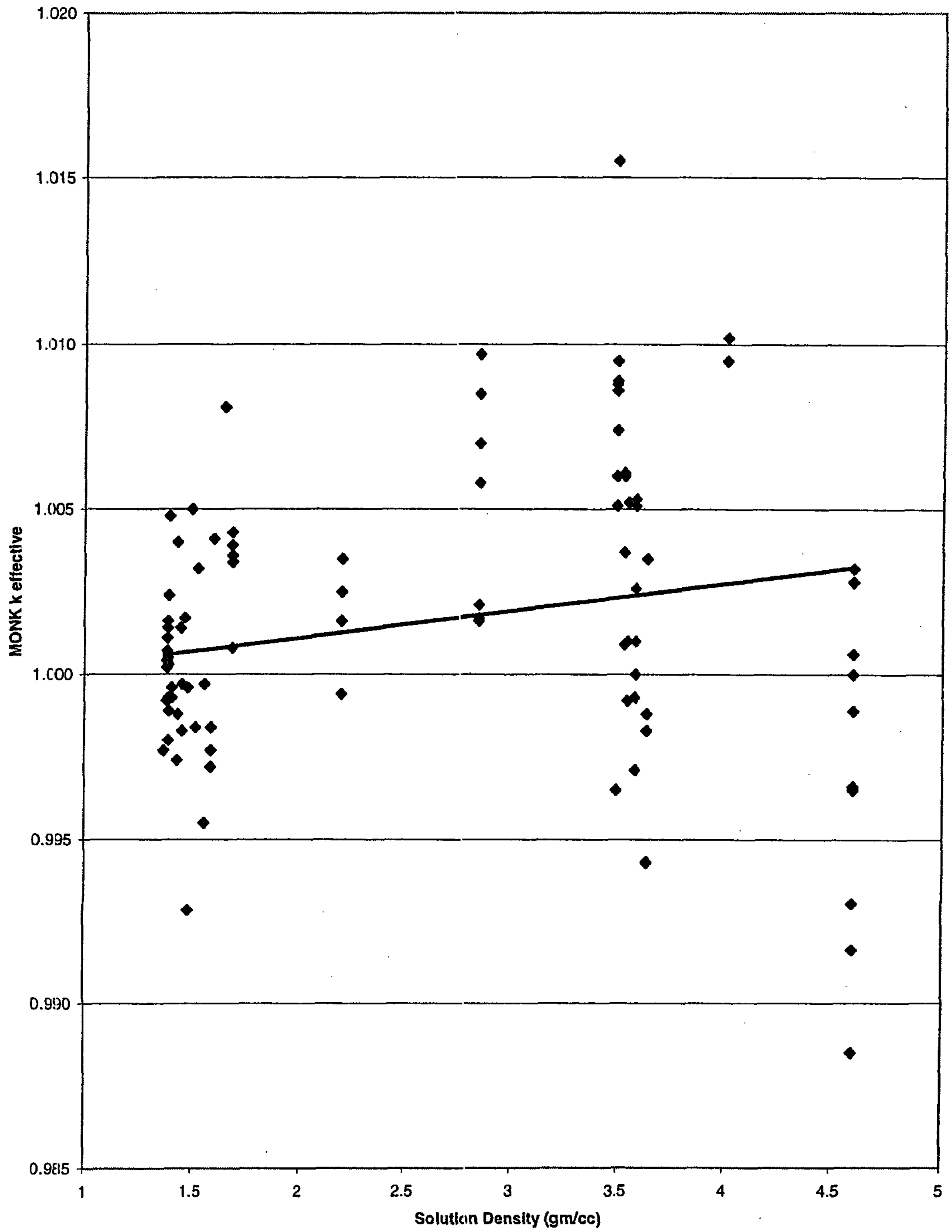

Figure 6-3 Plot of MONK k effective vs. H to U Number Ratio

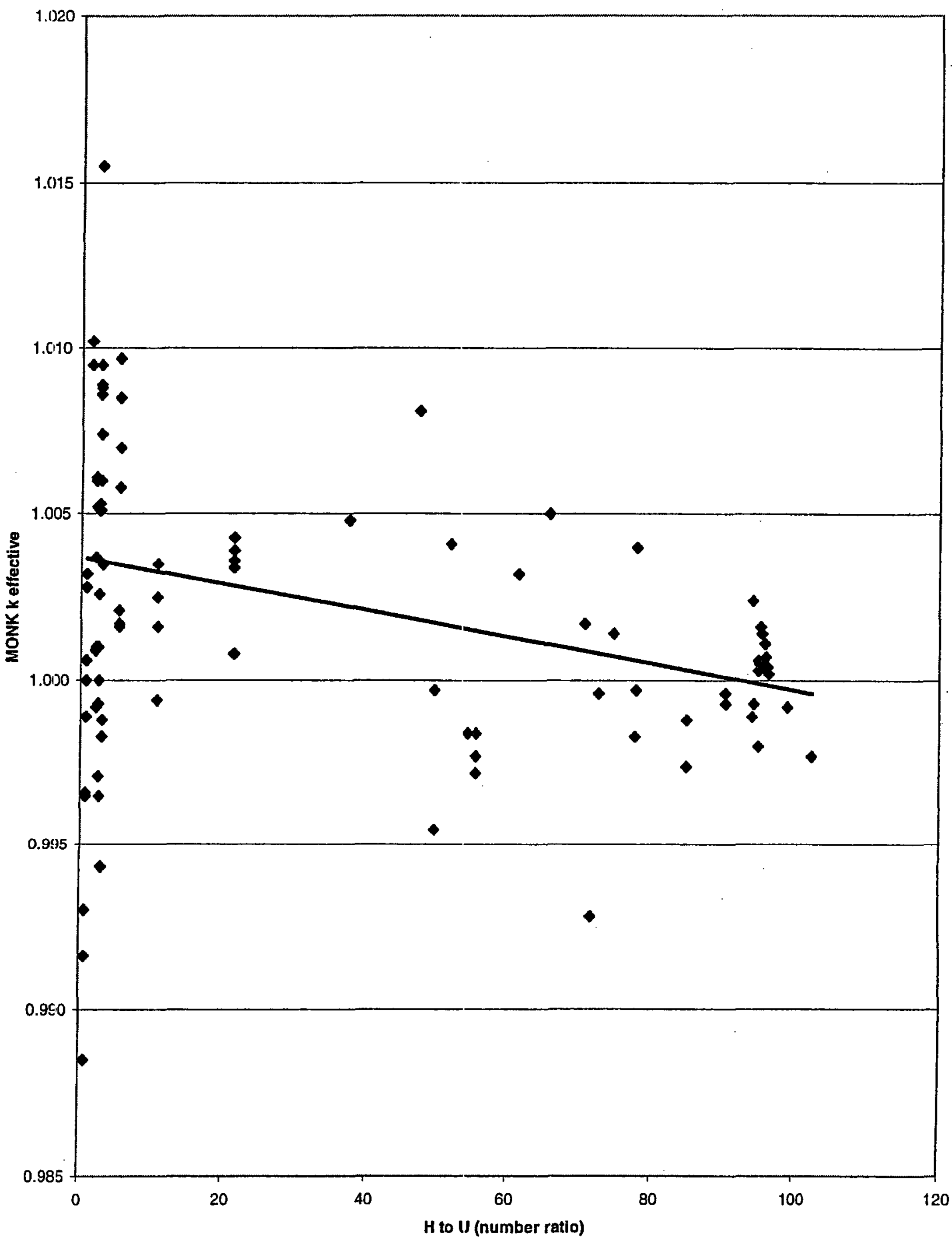

Figure 6-4 Plot of MONK k effective vs. ^{235}U Enrichment

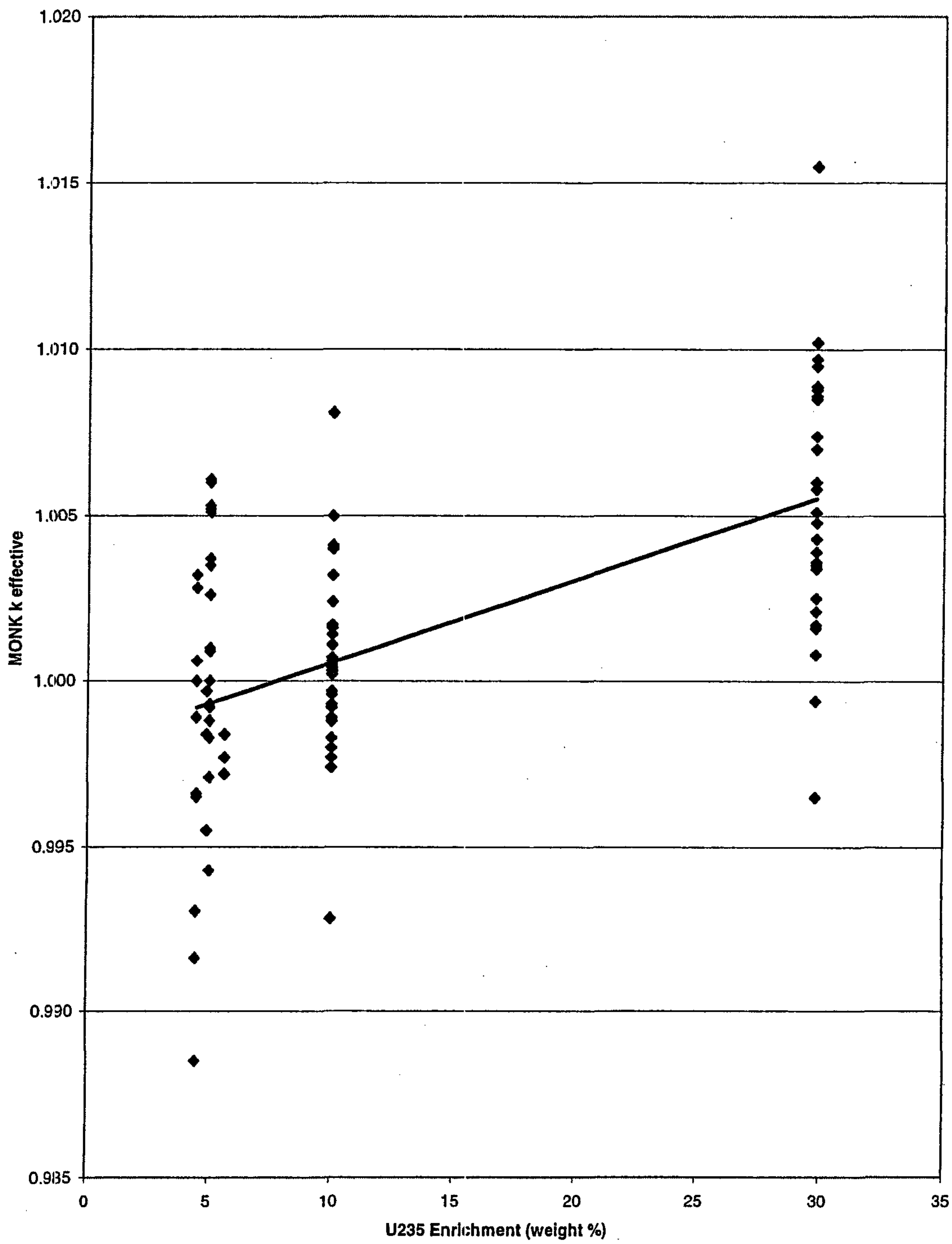

Figure 6-5 Plot of MONK k effective vs. Mean Cord Length

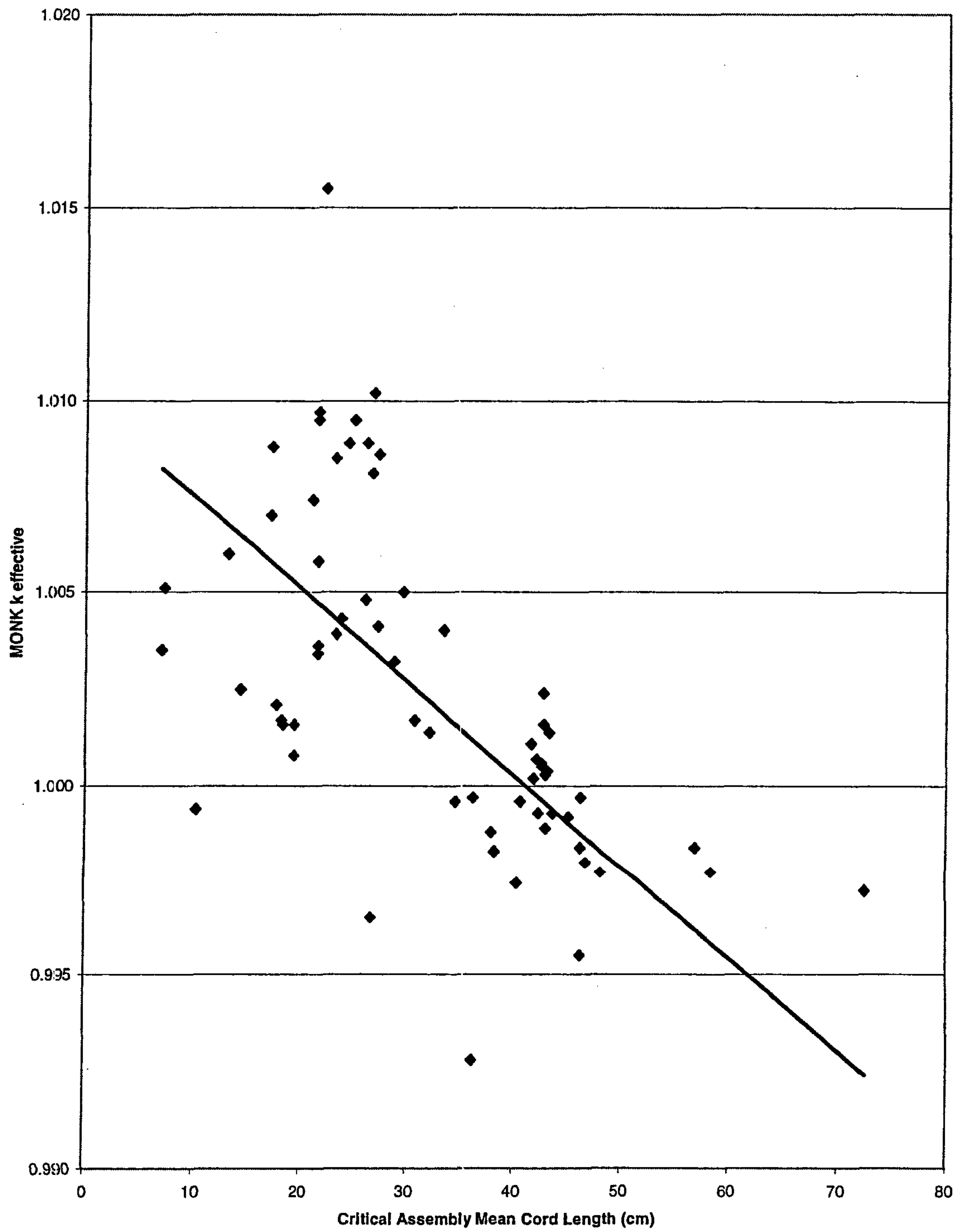

Figure 6-6 Plot of MONK k effective vs. Mean Log Energy of Neutron Causing Fission

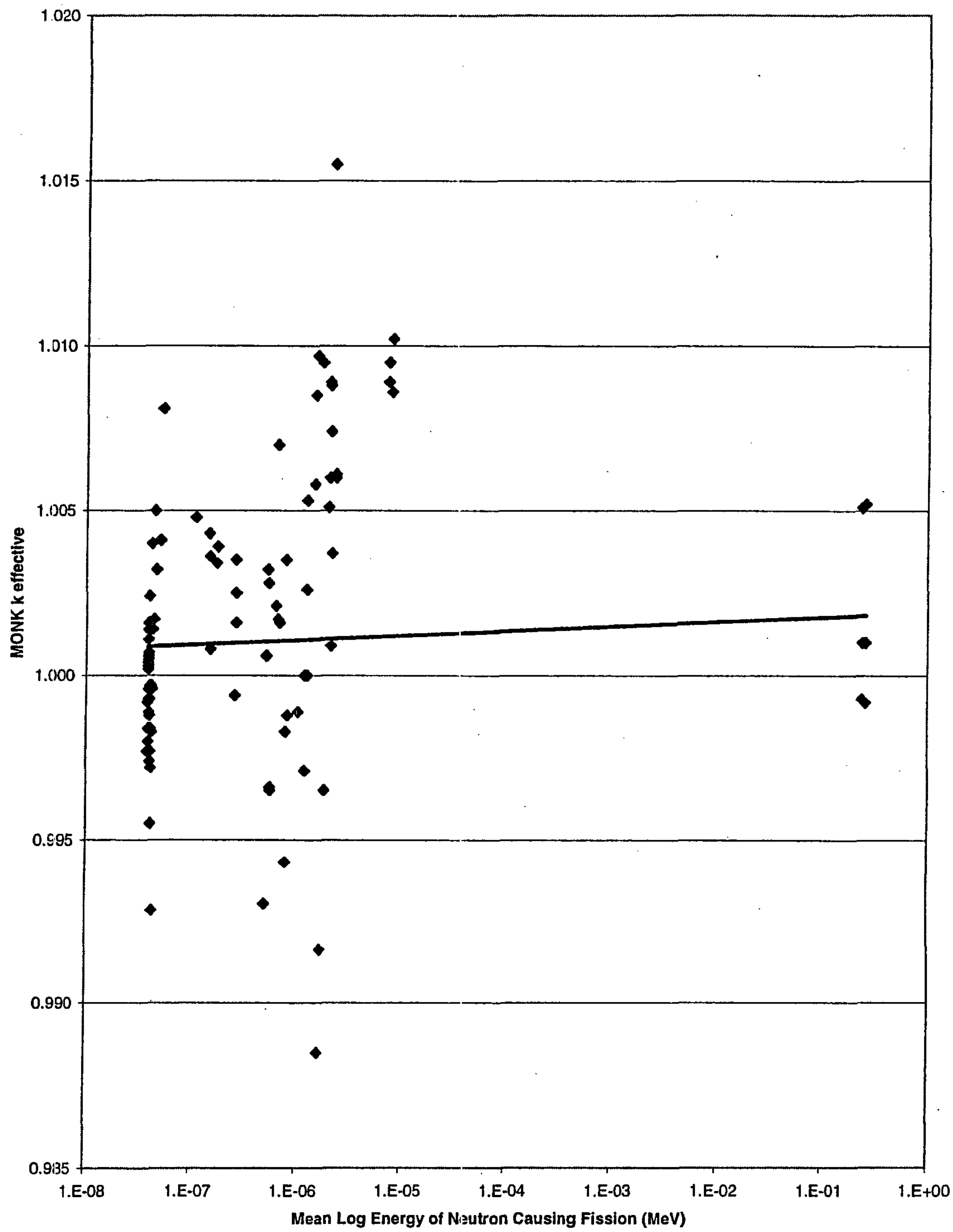

Table 6-1 Summary of Statistical Results

Fitted Parameter	Intercept	Slope	Correlation Coefficient (r^2)	Fitted Range	
				Min	Max
Solution Density (cm/cm^3)	0.9994	8.264E-04	0.037	1.370	4.60
H to U Number Ratio (unitless)	1.0037	-3.989E-05	0.142	0.787	102.61
^{235}U Enrichment ($^w/_o$)	0.9981	2.495E-04	0.190	4.460	29.83
Mean Cord Length (cm)	1.0099	-2. 412E-04	0.370	6.97	72.57
Mean Log Energy of Neutron Causing Fission	1.0009	3.613E-3	0.001	3.79E-8	0.26

* Excluded array cases from mean cord length fit

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 verification runs. Table 7-1 has the following definitions.

- "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 8 cases used in this verification is 1.0000 ± 0.0006

(Reference 6). The average of the AREVA verification runs was 1.0001 ± 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 43 listed in Table 7-1 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 0.9976 with a standard deviation of 0.0010. 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 Verification Runs

Experiment	Case	Serco Benchmark	AREVA Validation
43	1	0.995	0.9984
LEU	2	0.9921	0.9955
	3	0.9941	0.9997
51	1	1.0003	0.9996
LEU	2	1.0012	0.9997
	3	0.9958	0.9988
	4	1.0022	0.9996
	5	0.9996	1.0003
	6	1.0008	0.9992
	7	0.9991	0.9977
63	1	0.997	0.9984
LEU	2	0.9969	0.9977
	3	0.9972	0.9972
71	1	1.0083	1.0081
LEU	2	1.0072	1.0041
	3	1.0024	1.0032
	4	1.0034	1.005
	5	1.0044	1.0017
	6	1.0035	1.0014
	7	1.004	1.004
80	1	0.9997	0.9928
LEU	2	0.9991	0.9983
	3	0.9955	0.9974
	4	0.998	0.9993
	5	0.9981	0.998
81	1	1.002	1.0004
LEU	2	1.0003	1.0007
	3	1.0008	1.0011
	4	0.9996	1.0002
84	1	1.0013	0.9993
LEU	2	1.0011	1.0024
	3	0.9995	0.9989
85	1	0.9998	1.0014
LEU	2	0.9995	1.0016
	3	1.001	1.0005
	4	1.001	1.0006
Count	36	1.0000222	1.0000611
		0.0005549	0.0004525

Table 7-2 Results of Repeatability Sensitivity Study

Date	Time	Seed 1	Seed 2	k_{eff}
16/02/2004	16.31.41	32769	29133	0.9984
15/02/2006	15.55.59	26523	19135	0.9971
15/02/2006	16.17.14	32823	19135	0.9961
15/02/2006	16.28.05	35113	19135	0.9970
15/02/2006	16.38.47	36711	19135	0.9983
15/02/2006	16.38.47	36711	19135	0.9985
Count =	6		**Avg =**	0.9976
		Standard Deviation =		0.0010

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

*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 MONK 8A. This verification ensures that the cases run by Urenco are similar to the AREVA benchmark cases.

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 = 0.9401.

  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 0.0014.

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 8, February 2006. Chapter 5, "Nuclear Criticality Safety."

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

13. R. E. Rothe, I. Oh and G.R. Goebel, "Critical Experiments with Interstitially-Moderated Arrays of Low-Enriched Uranium Oxide," NUREG/CR-1071, September 1980.

Attachment 1A
Example MONK 8A Inputs

Input File case25.01

```
* MONK VALIDATION CALCULATIONS - EXPERIMENT 25.01
* ----------------------------------------------
* Calculations performed by N R Smith - July 1995
* Reported in ANSWERS/MONK/VAL/25
*

* Summary of experiment
* ---------------------
* Fissile Material: Low enriched Uranium oxide powder
* Geometry:         Homogeneous blocks in aluminium cans
* Moderator:        Plastic
* Neutron poison:   None
* Reflector:        Plastic
* Reference:        R E Rothe, I Oh and G R Goebel
*                       Critical Experiments with Intersitially-Moderated
*                       Arrays of Low-enriched Uranium Oxide
*                   NUREG/CR-1071
*                       September 1980

*
* Critical Parameter Data
* -----------------------

* Experiment 1 - Category O (optimum moderation)
* Configuration (b)
* Number of cans = 42
* Critical separation of north and south cores = 0.31cm

* Important Notes
* ---------------
* 1. Polythene bags assumed homogeneously mixed with powder
* 2. Average block composition data used
* 3. Powder impurities ignored
* 4. Miscellaneous tapes ignored
* 5. Curved can edges represented as square
* 6. Average plastic composition used
* 7. Filler percentage used to scale density (88%)
* 8. Average inner and outer reflector dimensions used
*******************************************************************************
*
BEGIN MATERIAL DATA
MONK
6 29 NUCNAMES

WGT  4.60   ! M1 - uranium oxide powder
  J2U234 3.8 J2U235 568.6 J2U236 10.2 J2U238 12165.4
  J2O16 2619.5 J2HINH2O 42.5 J2C 45

WGT  2.713  ! M2 - aluminium can
  J2AL27 99.36 J2SI 0.10 J2FE54 0.02 J2FE56 0.39 J2FE57 0.01
  J2CU 0.12

WGT  1.185  ! M3 - moderator plastic
  J2HINCH2 7.83 J2C 59.49 J2O16 32.48

WGT  1.110  ! M4 - filler plastic
  J2HINCH2 7.30 J2C 53.50 J2N14 0.13 J2O16 30.34 J2P31 0.82 J2CL 1.45
  JN3BR79 2.84 JN3BR81 2.84
```

```
WGT  1.261   ! M5 - reflector plastic
  J2HINCH2 7.30 J2C 53.50 J2N14 0.13 J2O16 30.34 J2P31 0.82 J2CL 1.45
  JN3BR79 2.84 JN3BR81 2.84

WGT  7.93    ! M6 - steel table top
  J2CR50 0.82 J2CR52 16.49 J2CR53 1.90 J2CR54 0.48
  J2NI58 6.94 J2NI60 2.74  J2NI61 0.12 J2NI62 0.39 J2NI64 0.10
  J2FE54 3.99 J2FE56 64.31 J2FE57 1.50 J2FE58 0.20

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

PART 1 NEST             ! North core assembly
BOX   BH3   0.0 0.0 0.0           33.0 68.44 50.72
BOX    M4   -5.87 0.0 -26.1       38.87 77.5 83.4
BOX    M5   -31.07 -25.6 -51.45   64.07 128.4 133.6
BOX    M3   -31.07 -25.6 -51.45   65.3 128.4 133.6

PART 2 NEST             ! South core assembly
BOX   BH9   0.0 0.0 0.0           33.0 68.44 50.72
BOX    M4   0.0 0.0 -26.1         46.77 77.5 83.4
BOX    M5   0.0 -25.6 -51.45      73.27 128.4 133.6
BOX    M3   -1.23 -25.6 -51.45    74.5 128.4 133.6

PART 3 CLUSTER          ! Complete assembly
BOX    P1   0.0 0.0 0.0        65.3 128.4 133.6
BOX    P2   65.61 0.0 0.0      74.5 128.4 133.6
BOX    M0   0.0 0.0 0.0        140.11 128.4 133.6

PART 4 NEST             ! Add steel table top
BOX    P3   0.0 0.0 0.0    140.11 128.4 133.6
BOX    M6   0.0 0.0 -1.3   140.11 128.4 134.9

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

POLY                    ! H1 - aluminium can and contents
  2 0
  1 8   0.15 0.15 0.15   0.15 15.13 0.15   15.13 15.13 0.15    15.13 0.15 0.15
        0.15 0.15 15.13  0.15 15.13 15.13  15.13 15.13 15.13  15.13 0.15 15.13
  -2 8  0.0 0.0 0.0      0.0 15.28 0.0     15.28 15.28 0.0     15.28 0.0 0.0
        0.0 0.0 15.28    0.0 15.28 15.28   15.28 15.28 15.28  15.28 0.0 15.28

LATTICE                 ! H2 - holes in can body
  DCOSINES -1 0 0 0 0 1
  7 4 RECT 2.18 3.82
  -1.09 -1.91 PINS 0.315 0.315 28*0  0 2

XYZMESH                 ! H3 - north assembly
  3   0.0 15.28 17.72 33.0
  7   0.0 15.28 17.72 33.0 35.44 50.72 53.16 68.44
  5   0.0 15.28 17.72 33.0 35.44 50.72
  -1  3 -1    3 3 3   -1 3 -1   3 3 3   -1 3 -1    3 3 3   -1  3 -1
   3  3  3    3 3 3    3 3  3   3 3 3    3 3  3    3 3 3    3  3  3
  -1  3 -1    3 3 3   -1 3 -1   3 3 3   -1 3 -1    3 3 3   -1  3 -1
  -4  3  3    3 3 3    3 3  3   3 3 3    3 3  3    3 3 3   -4  3  3
```

```
   4 -5 -1   -7 3 3   -1 3 -1   3 3 3   -1 3 -1   -8 3 3   4 -5 -1
  0

PLATE                    ! H4 - Filler/half-moderator Z segment
  0 0 1   1 1.23   4 3

PLATE                    ! H5 - Filler/half-moderator X+ segment
  1 0 0   1 1.21   3 4

PLATE                    ! H6 - Filler/half-moderator X- segment
  1 0 0   1 1.23   4 3

PLATE                    ! H7 - Filler/half-moderator Y+ segment
  0 1 0   1 1.21   3 4

PLATE                    ! H8 - Filler/half-moderator Y- segment
  0 1 0   1 1.23   4 3

XYZMESH                  ! H9 - south assembly
  3    0.0 15.28 17.72 33.0
  7    0.0 15.28 17.72 33.0 35.44 50.72 53.16 68.44
  5    0.0 15.28 17.72 33.0 35.44 50.72
  -1  3 -1    3  3  3   -1  3 -1    3  3  3   -1  3 -1    3  3  3   -1  3 -1
   3  3  3    3  3  3    3  3  3    3  3  3    3  3  3    3  3  3    3  3  3
  -1  3 -1    3  3  3   -1  3 -1    3  3  3   -1  3 -1    3  3  3   -1  3 -1
   3  3 -4    3  3 -4    3  3 -4    3  3 -4    3  3 -4    3  3 -4    3  3 -4
  -1 -6  4    3  3  4   -1 -6  4    3  3  4   -1 -6  4    3  3  4   -1 -6  4
  0

END
******************************************************************************
*
BEGIN CONTROL DATA
STAGES -1 100 1000 STDV 0.0010
END
******************************************************************************
*
BEGIN SOURCE GEOMETRY
ZONEMAT ZONE 1 PART 4 / MATERIAL 1
END
******************************************************************************
*
BEGIN ENERGY DATA

SCORING GROUPS 16
15.0 3.0 1.4 0.9 0.4 0.1 1.7E-2 3.0E-3 5.5E-4 1.0E-4 3.0E-5
1.0E-5 3.0E-6 1.0E-6 4.0E-7 1.0E-7 1.0E-20

END
```

Input File case42.01

```
* MONK VALIDATION CALCULATIONS - EXPERIMENT 42.01
* ----------------------------------------------
* Calculations performed by W Wright - October 1997
* Reported in ANSWERS/MONK/VAL/42
*

* Summary of experiment
* ---------------------
* Fissile Material:     Slightly Moderated Uranium Oxide Powder
*                       [U(U5=5wt%)O2 H/U 2.0, 2.5 and 3.0)
* Geometry:             Cuboidal
* Moderator:            Light Water
* Neutron poison:       None; Boron Steel (1.1wt% Boron)
* Reflector:            Polythene
* Reference:            G Poullot
*                       Poudre d'U(5)O2 faiblement moderee -
*                       BENCHMARK description
*                       SEC/T/0910/93.64/C.CEA
* Code Package:         MONK7B-JEF2

*
* Critical Parameter Data
* -----------------------
* H/U = 2.0 (After Mixing)
* Experiment Nomenclature: R2 - 2R (6,6)
* Table 1 contains 2 fuel box in X 6 in Y and 6 in Z
* Table 2 contains 2 fuel box in X 6 in Y and 6 in Z
* Tables are separated by 2.6 cm

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

BEGIN MATERIAL SPECIFICATION

NMATERIALS 4
ATOMS
* Material 1 - Fuel
MATERIAL 1 DENSITY 0.0
  U235 PROP  3.7095E-04
  U238 PROP  6.9590E-03
  O16  PROP  2.284091E-02
  H1   PROP  1.474922E-02

ATOMS
* Material 2 - Structural Material (AG3)
MATERIAL 2 DENSITY 0.0
  AL   PROP  5.8058E-02
  MG   PROP  1.9719E-03
  CU   PROP  1.0260E-05
  FE   PROP  1.0508E-04
  CR   PROP  6.4000E-06
  MN   PROP  1.0090E-04
  SI   PROP  6.9600E-05
  TI   PROP  6.8000E-06
  ZN64 PROP  4.9800E-06

ATOMS
* Material 3 - Seal
MATERIAL 3 DENSITY 0.0
  C    PROP  6.6131E-02
  H1   PROP  1.0844E-01
```

```
  O16   PROP  7.2484E-04
  N     PROP  3.5870E-04
  B     PROP  8.84E-08
  CD    PROP  8.5E-09

ATOMS
* Material 4 - Reflector (Polythene)
MATERIAL 4 DENSITY 0.0
  C     PROP  4.12149E-02
  H1    PROP  8.24290E-02

USE J2HINCH2 FOR H1 IN MATERIAL 4

END

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

BEGIN MATERIAL GEOMETRY

PART 1
BOX    1 -9.775 -9.775  0.55   19.55 19.55 17.85 ! Main Section of Fuel
BOX    2 -8.6   -8.6   18.4    17.2  17.2   1.2  ! Top Section of Fuel
BOX    3 -8.275 -8.275  0.15   16.55 16.55  0.4  ! Bottom Section of Fuel
BOX    4 -9.925 -9.925  0.4    19.85 19.85 18.15 ! Main Section of AG3 Box
BOX    5 -9.16  -9.16  18.55   18.32 18.32  0.9  ! Top Section of AG3 Box 1
BOX    6 -9.8   -9.8   19.45   19.6  19.6   0.15 ! Top Section of AG3 Box 2
BOX    7 -8.425 -8.425  0.0    16.85 16.85  0.4  ! Bottom Section of AG3 Box
BOX    8 -9.8   -9.8   19.6    19.6  19.6   0.1  ! Seal - Part 1
BOX    9 -8.6   -8.6   19.6    17.2  17.2   0.1  ! Seal - Part 2
BOX   10 -9.925 -9.925 19.7    19.85 19.85  0.3  ! Lid
YROD 11 -8.275 -8.275  0.0     0.55 23.40523     ! Cruciform Kink 1a
        VX COS 45 COS 225 COS 90 VZ COS 90 COS 90 COS 0
YROD 12 -8.275 -8.275  0.0     0.4  23.40523     ! Cruciform Kink 1b
        VX COS 45 COS 225 COS 90 VZ COS 90 COS 90 COS 0
XROD 13 -8.275  8.275  0.0     0.55 23.40523     ! Cruciform Kink 2a
        VX COS 45 COS 225 COS 90 VZ COS 90 COS 90 COS 0
XROD 14 -8.275  8.275  0.0     0.4  23.40523     ! Cruciform Kink 2b
        VX COS 45 COS 225 COS 90 VZ COS 90 COS 90 COS 0
BOX  15 -8.425 -8.425  0.0    16.85 16.85  0.55 ! Void Around Bottom Section of
AG3 Box
BOX  16 -9.925 -9.925  0.0    19.85 19.85 20.0  ! Void Surround
ZONES
/fuelmid/  M1  +1
/fueltop/  M1  +2
/fuelbot/  M1  +3 -11 -13
/cladmid/  M2  +4 -1 -2 -3 -11 -13
/cladtop1/ M2  +5 -2 -4 -6
/cladtop2/ M2  +6 -2 -4 -5
/cladbot/  M2  +7 -3 -4 -11 -13
/seal/     M3  +8 -9
/sealvoid/ M0  +9
/lid/      M2  +10
/kink1a/   M2  +11 -12 -13
/kink1b/   M0  +12 -13
/kink2a/   M2  +13 -14
/kink2b/   M0  +14
/kinkvoid/ M0  +15 -7 -3 -4
/void/     M0  +16 -15 -14 -13 -12 -11 -10 -9 -8 -7 -6 -5 -4 -3 -2 -1

PART 2 NEST ! FUEL BOX IN EGG-CRATE
BOX  P1   0.075 0.075 0.0  19.85 19.85 20.0
```

```
BOX  M0   -0.15 -0.15  0.0  20.3   20.3   20.0
BOX  M2   -0.25 -0.25  0.0  20.52  20.52  20.0

PART 3 NEST ! REFLECTOR IN EGG-CRATE
BOX  M4    0.0    0.0   0.0  20.0   20.0   20.0
BOX  M0   -0.15 -0.15  0.0  20.3   20.3   20.0
BOX  M2   -0.25 -0.25  0.0  20.52  20.52  20.0

PART 4 LIKE 3 M0 M0 M2 ! EMPTY EGG-CRATE

PART 5 ARRAY ! Table 1 (Movable) Arrangement
4 9 8
* Z-LAYER 1
4 4 4 4
4 3 3 3
4 3 3 3
4 3 3 3
4 3 3 3
4 3 3 3
4 3 3 3
4 3 3 3
4 3 3 3
* Z-LAYER 2
4 4 4 4
4 3 3 3
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 3 3
* Z-LAYER 3
4 4 4 4
4 3 3 3
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 3 3
* Z-LAYER 4
4 4 4 4
4 3 3 3
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 3 3
* Z-LAYER 5
4 4 4 4
4 3 3 3
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
```

```
4 3 3 3
* Z-LAYER 6
4 4 4 4
4 3 3 3
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 3 3
* Z-LAYER 7
4 4 4 4
4 3 3 3
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 2 2
4 3 3 3
* Z-LAYER 8
4 4 4 4
4 3 3 3
4 3 3 3
4 3 3 3
4 3 3 3
4 3 3 3
4 3 3 3
4 3 3 3
4 3 3 3

PART 6 ARRAY ! Table 2 (Fixed) Arrangement
5 9 8
* Z-LAYER 1
4 4 4 4 4
3 3 3 4 4
3 3 3 4 4
3 3 3 4 4
3 3 3 4 4
3 3 3 4 4
3 3 3 4 4
3 3 3 4 4
3 3 3 4 4
* Z-LAYER 2
4 4 4 4 4
3 3 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
3 3 3 4 4
* Z-LAYER 3
4 4 4 4 4
3 3 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
```

```
2 2 3 4 4
2 2 3 4 4
3 3 3 4 4
* Z-LAYER 4
4 4 4 4 4
3 3 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
3 3 3 4 4
* Z-LAYER 5
4 4 4 4 4
3 3 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
3 3 3 4 4
* Z-LAYER 6
4 4 4 4 4
3 3 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
3 3 3 4 4
* Z-LAYER 7
4 4 4 4 4
3 3 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
2 2 3 4 4
3 3 3 4 4
* Z-LAYER 8
4 4 4 4 4
3 3 3 4 4
3 3 3 4 4
3 3 3 4 4
3 3 3 4 4
3 3 3 4 4
3 3 3 4 4
3 3 3 4 4
3 3 3 4 4

PART 7 NEST ! Add AG3 to Lateral Exterior Faces and Base of Table 1
BOX P5     0.0    0.0    0.0  82.08 184.68 160.0
BOX M2    -1.1   -1.1   -2.5  84.28 186.88 162.5

PART 8 NEST ! Add AG3 to Lateral Exterior Faces and Base of Table 2
BOX P6     0.0    0.0    0.0 102.6  184.68 160.0
BOX M2    -1.1   -1.1   -2.5 104.8  186.88 162.5
```

```
PART 9 CLUSTER ! Complete Assembly
BOX  P7    0.0    0.0    0.0  84.28 186.88 162.5
BOX  P8  86.88    0.0    0.0 104.8  186.88 162.5
BOX  M0    0.0    0.0    0.0 191.68 186.88 162.5

END

BEGIN CONTROL DATA

STAGES -1 100 1000   STDV 0.0010

END

BEGIN SOURCE GEOMETRY

ZONEMAT
ZONE 1 PART 7 / MATERIAL 1
ZONE 1 PART 8 / MATERIAL 1

END

BEGIN ENERGY DATA

 SCORING GROUPS 16
  15.0 3.0 1.4 0.9 0.4 0.1 1.7E-2 3.0E-3 5.5E-4 1.0E-4 3.0E-5
  1.0E-5 3.0E-6 1.0E-6 4.0E-7 1.0E-7 1.0E-20

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 case69.01

```
* MONK VALIDATION EXPERIMENT NUMBER 69.01
* ----------------------------------------
*
* MONK VALIDATION CALCULATIONS - EXPERIMENT IEU-COMP-THERM-001 Case 1
* ----------------------------------------------------------------------
*
* Summary of experiment
* --------------------
* Fissile Material:   U(30)F4 -polytetrafluoroethylene [(CF2)n]
* Geometry:           Cubic
* Moderator:          Polyethylene
* Neutron poison:     None
* Reflector:          None; Paraffin; Cadmium; Boron
* Reference:          Virginia F. Dean
*            Critical Arrays Of Polyethylene-Moderated U(30)F4 -
*            Polytetrafluoroethylene One-Inch Cubes
*            IEU-COMP-THERM-001 (March 31, 1995)
* Code Package:       MONK8A
*
* Critical Parameter Data
* -----------------------
* H-cubes to U-cubes to Air ratio: 1:4:0
* Dimensions of complete layers:  15x14x14
* Total Number of H-cubes:     598
* Total Number of U-cubes:     2392
* Total Number of cubes:       2990
* Reflector:                   Paraffin

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

BEGIN MATERIAL SPECIFICATION

NMATERIALS 7

* Material 1 = Specified U Cube, UF4-(CF2)n
ATOMS
MATERIAL 1 DENSITY 0.0
U235  PROP 2.3690E-3
U238  PROP 5.5023E-3
F19   PROP 4.7049E-2
C     PROP 7.9574E-3
O16   PROP 1.8102E-4
AL27  PROP 7.5140E-4

* Material 2 = Specified H Cube, Polyethylene
ATOMS
MATERIAL 2 DENSITY 0.0
C     PROP 3.9232E-2
H1    PROP 7.5224E-2

* Material 3 = Aluminium 2S (given composition)
ATOMS
MATERIAL 3 DENSITY 0.0
AL27  PROP 5.9881E-2
SI    PROP 2.9054E-4
FE    PROP 1.4611E-4

* Material 4 = Paraffin (given composition)
```

```
ATOMS
MATERIAL 4 DENSITY 0.0
C    PROP 3.7138E-2
H1    PROP 7.7247E-2

* Material 5 = Cadmium (given composition)
ATOMS
MATERIAL 5 DENSITY 0.0
CD    PROP 4.6447E-2

* Material 6 = Boron (given composition)
ATOMS
MATERIAL 6 DENSITY 0.0
B10   PROP 3.2147E-3
B11   PROP 1.2939E-2

* Material 7 = Wood Table Top
ATOMS
MATERIAL 7 DENSITY 0.0
C     PROP 1.4659E-2
H1    PROP 2.7921E-2
O16   PROP 1.3960E-2

USE H1INCH2 FOR H1 IN MATERIAL 2
USE H1INCH2 FOR H1 IN MATERIAL 4
USE H1INCH2 FOR H1 IN MATERIAL 7

END

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

BEGIN MATERIAL GEOMETRY

* Part 1 - U Cube

PART 1

NEST

BOX   M1 0 0 0 2.5527 2.5527 2.5527

* Part 2 - H Cube

PART 2

NEST

BOX   M2 0 0 0 2.5527 2.5527 2.5527

* Part 3 - Paraffin Cube to Fill Top Layer

PART 3

NEST

BOX   M4 0 0 0 2.5527 2.5527 2.5527

* Part 4 - Layers 1, 6, 11

PART 4
```

ARRAY 15 14 1

(2 1 1 1 1)*3
(1 1 2 1 1)*3
(1 1 1 1 2)*3
(1 2 1 1 1)*3
(1 1 1 2 1)*3
(2 1 1 1 1)*3
(1 1 2 1 1)*3
(1 1 1 1 2)*3
(1 2 1 1 1)*3
(1 1 1 2 1)*3
(2 1 1 1 1)*3
(1 1 2 1 1)*3
(1 1 1 1 2)*3
(1 2 1 1 1)*3

* Part 5 - Wrap Layer Array

PART 5

NEST

BOX P4 0 0 0 38.2905 35.7378 2.5527

* Part 6 - Layers 2, 7, 12

PART 6

ARRAY 15 14 1

(1 1 1 1 2)*3
(1 2 1 1 1)*3
(1 1 1 2 1)*3
(2 1 1 1 1)*3
(1 1 2 1 1)*3
(1 1 1 1 2)*3
(1 2 1 1 1)*3
(1 1 1 2 1)*3
(2 1 1 1 1)*3
(1 1 2 1 1)*3
(1 1 1 1 2)*3
(1 2 1 1 1)*3
(1 1 1 2 1)*3
(2 1 1 1 1)*3

* Part 7 - Wrap Layer Array

PART 7

NEST

BOX P6 0 0 0 38.2905 35.7378 2.5527

* Part 8 - Layers 3, 8, 13

PART 8

ARRAY 15 14 1

```
(1 1 1 2 1)*3
(2 1 1 1 1)*3
(1 1 2 1 1)*3
(1 1 1 1 2)*3
(1 2 1 1 1)*3
(1 1 1 2 1)*3
(2 1 1 1 1)*3
(1 1 2 1 1)*3
(1 1 1 1 2)*3
(1 2 1 1 1)*3
(1 1 1 2 1)*3
(2 1 1 1 1)*3
(1 1 2 1 1)*3
(1 1 1 1 2)*3

* Part 9 - Wrap Layer Array

PART 9

NEST

BOX   P8 0 0 0 38.2905 35.7378 2.5527

* Part 10 - Layers 4, 9, 14

PART 10

ARRAY 15 14 1

(1 1 2 1 1)*3
(1 1 1 1 2)*3
(1 2 1 1 1)*3
(1 1 1 2 1)*3
(2 1 1 1 1)*3
(1 1 2 1 1)*3
(1 1 1 1 2)*3
(1 2 1 1 1)*3
(1 1 1 2 1)*3
(2 1 1 1 1)*3
(1 1 2 1 1)*3
(1 1 1 1 2)*3
(1 2 1 1 1)*3
(1 1 1 2 1)*3

* Part 11 - Wrap Layer Array

PART 11

NEST

BOX   P10 0 0 0 38.2905 35.7378 2.5527

* Part 12 - Layers 5, 10

PART 12

ARRAY 15 14 1

(1 2 1 1 1)*3
```

```
(1 1 1 2 1)*3
(2 1 1 1 1)*3
(1 1 2 1 1)*3
(1 1 1 1 2)*3
(1 2 1 1 1)*3
(1 1 1 2 1)*3
(2 1 1 1 1)*3
(1 1 2 1 1)*3
(1 1 1 1 2)*3
(1 2 1 1 1)*3
(1 1 1 2 1)*3
(2 1 1 1 1)*3
(1 1 2 1 1)*3

* Part 13 - Wrap Layer Array

PART 13

NEST

BOX P12 0 0 0 38.2905 35.7378 2.5527

* Part 14 - Partially Filled Top Layer 15

PART 14

ARRAY 15 14 1

3 3 3 3 3 3 3 3 3 3 3 3 3 3 3
3 3 3 3 3 3 3 3 3 3 3 3 3 3 3
3 3 3 3 3 3 3 3 3 3 3 3 3 3 3
3 3 3 3 1 1 1 2 1 1 1 3 3 3 3
3 3 3 3 2 1 1 1 1 2 1 3 3 3 3
3 3 3 3 1 1 2 1 1 1 1 3 3 3 3
3 3 3 3 1 1 1 1 2 1 1 3 3 3 3
3 3 3 3 1 2 1 1 1 1 2 3 3 3 3
3 3 3 3 1 1 1 2 1 1 1 3 3 3 3
3 3 3 3 2 1 1 1 1 2 1 3 3 3 3
3 3 3 3 3 3 3 1 3 3 3 3 3 3 3
3 3 3 3 3 3 3 3 3 3 3 3 3 3 3
3 3 3 3 3 3 3 3 3 3 3 3 3 3 3
3 3 3 3 3 3 3 3 3 3 3 3 3 3 3

* Part 15 - Wrap Layer Array

PART 15

NEST

BOX P14 0 0 0 38.2905 35.7378 2.5527

* Part 16 - Build Core of Cube Layers

PART 16

ARRAY 1 1 15

5 7 9 11 13 5 7 9 11 13 5 7 9 11 15

* Part 17 - Wrap Core with Paraffin Reflector
```

```
PART 17

NEST

BOX  P16 0 0 0 38.2905 35.7378 38.2905
BOX  M4 -17.78 -17.78 -17.78 73.8505 71.2978 73.8505

ALBEDO 0 0 0 0 0 0

END

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

BEGIN CONTROL DATA

STAGES -5   ! Start at stage number -5
    100  ! Finish at stage number 100
    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 17 /

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]
case25.01	NUREG/CR-1071	1	0.0042	U3O8	Plexiglas	array			0	N/A
case25.02	NUREG/CR-1071	2	0.0042	U3O8	Plexiglas	array			0	N/A
case25.03	NUREG/CR-1071	3	0.0042	U3O8	Plexiglas	array			0	N/A
case25.04	NUREG/CR-1071	4	0.0042	U3O8	Plexiglas	array			0	N/A
case25.05	NUREG/CR-1071	5	0.0042	U3O8	Plexiglas	array			0	N/A
case25.06	NUREG/CR-1071	6	0.0042	U3O8	Plexiglas	array			0	N/A
case25.07	NUREG/CR-1071	7	0.0042	U3O8	Plexiglas	array			steel plate	N/A
case25.08	NUREG/CR-1071	8	0.0042	U3O8	Plexiglas	array			steel plate	N/A
case25.09	NUREG/CR-1071	9	0.0042	U3O8	Plexiglas	array			steel plate	N/A
case25.10	NUREG/CR-1071	10	0.0042	U3O8	Plexiglas	array			steel plate	N/A
case42.01	LEU-COMP-THERM-049	1	0.0044	UO2	polyethylene	array			0	N/A
case42.02	LEU-COMP-THERM-049	2	0.0044	UO2	polyethylene	array			0	N/A
case42.03	LEU-COMP-THERM-049	3	0.0044	UO2	polyethylene	array			0	N/A
case42.04	LEU-COMP-THERM-049	4	0.0044	UO2	polyethylene	array			0	N/A
case42.05	LEU-COMP-THERM-049	5	0.0044	UO2	polyethylene	array			0	N/A
case42.06	LEU-COMP-THERM-049	6	0.0044	UO2	polyethylene	array			0	N/A
case42.07	LEU-COMP-THERM-049	7	0.0044	UO2	polyethylene	array			0	N/A
case42.08	LEU-COMP-THERM-049	8	0.0044	UO2	polyethylene	array			0	N/A
case42.09	LEU-COMP-THERM-049	9	0.0044	UO2	polyethylene	array			0	N/A
case42.10	LEU-COMP-THERM-049	10	0.0044	UO2	polyethylene	array			0	N/A
case42.11	LEU-COMP-THERM-049	11	0.0044	UO2	polyethylene	array			0	N/A
case42.12	LEU-COMP-THERM-049	12	0.0044	UO2	polyethylene	array			0	N/A
case42.13	LEU-COMP-THERM-049	13	0.0044	UO2	polyethylene	array			0	N/A
case42.14	LEU-COMP-THERM-049	14	0.0044	UO2	polyethylene	array			0	N/A
case42.15	LEU-COMP-THERM-049	15	0.0044	UO2	polyethylene	array			0	N/A
case42.16	LEU-COMP-THERM-049	16	0.0044	UO2	polyethylene	array			0	N/A

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]
case42.17	LEU-COMP-THERM-049	17	0.0044	UO2	polyethylene	array			0	N/A
case42.18	LEU-COMP-THERM-049	18	0.0044	UO2	polyethylene	array			borated steel plate	N/A
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
case69.01	IEU-COMP-THERM-001	1	0.004	UF4[CF2]	bare	slab	15x14x14		0	24.934

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]
case69.02	IEU-COMP-THERM-001	2	0.004	UF4[CF2]	bare	slab	12x12x11		0	21.01
case69.03	IEU-COMP-THERM-001	3	0.004	UF4[CF2]	bare	slab	10x10x9		0	17.60
case69.04	IEU-COMP-THERM-001	4	0.004	UF4[CF2]	bare	slab	10x10x8		0	18.23
case69.05	IEU-COMP-THERM-001	5	0.004	UF4[CF2]	bare	slab	16x14x14		0	25.99
case69.06	IEU-COMP-THERM-001	6	0.004	UF4[CF2]	bare	slab	10x10x10		0	17.02
case69.07	IEU-COMP-THERM-001	7	0.004	UF4[CF2]	bare	slab	11x10x10		0	18.12
case69.08	IEU-COMP-THERM-001	8	0.006	UF4[CF2]	bare	slab	11x11x10		0	19.30
case69.09	IEU-COMP-THERM-001	9	0.004	UF4[CF2]	bare	slab	13x12x12		0	21.56
case69.10	IEU-COMP-THERM-001	10	0.004	UF4[CF2]	bare	slab	11x11x14		0	17.16
case69.11	IEU-COMP-THERM-001	11	0.004	UF4[CF2]	bare	slab	10x10x19		0	13.09
case69.12	IEU-COMP-THERM-001	12	0.004	UF4[CF2]	bare	slab	9x9x39		0	7.26
case69.13	IEU-COMP-THERM-001	13	0.004	UF4[CF2]	bare	slab	9x9x11		0	14.26
case69.14	IEU-COMP-THERM-001	14	0.004	UF4[CF2]	bare	slab	8x8x16		0	10.21
case69.15	IEU-COMP-THERM-001	15	0.004	UF4[CF2]	bare	slab	8x7x26		0	6.97
case69.16	IEU-COMP-THERM-001	16	0.004	UF4[CF2]	paraffin	slab	11x11x10		0	19.30
case69.17	IEU-COMP-THERM-001	17	0.004	UF4[CF2]	bare	slab	16x16x16		0	27.23
case69.18	IEU-COMP-THERM-001	18	0.004	UF4[CF2]	bare	slab	13x14x13		0	23.23
case69.19	IEU-COMP-THERM-001	19	0.004	UF4[CF2]	bare	slab	12x13x12		0	21.53

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]
case69.20	IEU-COMP-THERM-001	20	0.004	UF4[CF2]	paraffin	slab	12x13x11		0	22.12
case69.21	IEU-COMP-THERM-001	21	0.004	UF4[CF2]	paraffin	slab	15x15x13		0	26.71
case69.22	IEU-COMP-THERM-001	22	0.004	UF4[CF2]	paraffin	slab	15x15x14		cadmium	26.11
case69.23	IEU-COMP-THERM-001.	23	0.004	UF4[CF2]	paraffin	slab	12x13x12		cadmium	21.53
case69.24	IEU-COMP-THERM-001	24	0.004	UF4[CF2]	paraffin	slab	12x13x12		boron	21.523
case69.25	IEU-COMP-THERM-001	25	0.004	UF4[CF2]	paraffin	slab	13x13x11		cadmium	23.32
case69.26	IEU-COMP-THERM-001	26	0.004	UF4[CF2]	paraffin	slab	12x13x12		0	21.53
case69.27	IEU-COMP-THERM-001	27	0.004	UF4[CF2]	paraffin	slab	14x13x12		0	23.83
case69.28	IEU-COMP-THERM-001	28	0.004	UF4[CF2]	paraffin	slab	14x14x13		0	24.41
case69.29	IEU-COMP-THERM-001	29	0.004	UF4[CF2]	paraffin	slab	16x15x15		0	26.64
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

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]
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
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/U (number ratio)	Density (gm/cm^3)	Reflector Material	Fuel Solution	Tank Shape	Mean Cord Length (cm)	Absorber	L MENCF	Monk K eff	Monk Std Dev	Total Uncertainty [1]
case25.01	0.0042	4.46	7.87E-01	4.60000	Plexiglas	U3O8	array	N/A	0	3.87408E-08	1.0006	0.0010	0.0043
case25.02	0.0042	4.46	7.87E-01	4.60000	Plexiglas	U3O8	array	N/A	0	4.09228E-08	0.9930	0.0010	0.0043
case25.03	0.0042	4.46	7.87E-01	4.60000	Plexiglas	U3O8	array	N/A	0	4.01048E-08	0.9916	0.0010	0.0043
case25.04	0.0042	4.46	7.87E-01	4.60000	Plexiglas	U3O8	array	N/A	0	4.18786E-08	0.9885	0.0010	0.0043
case25.05	0.0042	4.46	7.87E-01	4.60000	Plexiglas	U3O8	array	N/A	0	4.12788E-08	0.9989	0.0010	0.0043
case25.06	0.0042	4.46	7.87E-01	4.60000	Plexiglas	U3O8	array	N/A	0	3.9789E-08	1.0028	0.0010	0.0043
case25.07	0.0042	4.46	7.87E-01	4.60000	Plexiglas	U3O8	array	N/A	steel plate	3.92215E-08	0.9966	0.0010	0.0043
case25.08	0.0042	4.46	7.87E-01	4.60000	Plexiglas	U3O8	array	N/A	steel plate	3.8655E-08	0.9965	0.0010	0.0043
case25.09	0.0042	4.46	7.87E-01	4.60000	Plexiglas	U3O8	array	N/A	steel plate	3.81169E-08	1.0032	0.0010	0.0043
case25.10	0.0042	4.46	7.87E-01	4.60000	Plexiglas	U3O8	array	N/A	steel plate	3.78766E-08	1.0000	0.0010	0.0043
case42.01	0.0044	5.00	2.01E+00	3.52697	polyethylene	UO2	array	N/A	0	4.10755E-08	1.0009	0.0010	0.0045
case42.02	0.0044	5.00	2.52E+00	3.58222	polyethylene	UO2	array	N/A	0	4.09139E-08	0.9971	0.0010	0.0045
case42.03	0.0044	5.00	3.01E+00	3.63601	polyethylene	UO2	array	N/A	0	4.11523E-08	0.9943	0.0010	0.0045
case42.04	0.0044	5.00	2.01E+00	3.52697	polyethylene	UO2	array	N/A	0	5.17393E-08	1.0037	0.0010	0.0045
case42.05	0.0044	5.00	2.52E+00	3.58222	polyethylene	UO2	array	N/A	0	4.91982E-08	1.0053	0.0010	0.0045
case42.06	0.0044	5.00	3.01E+00	3.63601	polyethylene	UO2	array	N/A	0	4.57974E-08	1.0035	0.0010	0.0045
case42.07	0.0044	5.00	2.01E+00	3.52697	polyethylene	UO2	array	N/A	0	4.41634E-08	1.0060	0.0010	0.0045
case42.08	0.0044	5.00	2.52E+00	3.58222	polyethylene	UO2	array	N/A	0	4.32521E-08	1.0000	0.0010	0.0045
case42.09	0.0044	5.00	3.01E+00	3.63601	polyethylene	UO2	array	N/A	0	4.2193E-08	0.9983	0.0010	0.0045
case42.10	0.0044	5.00	2.01E+00	3.52697	polyethylene	UO2	array	N/A	0	4.12906E-08	1.0061	0.0010	0.0045
case42.11	0.0044	5.00	2.52E+00	3.58222	polyethylene	UO2	array	N/A	0	4.29012E-08	1.0026	0.0010	0.0045

Case	Experimental Uncertainty	Enrichment ($^w/_o$)	H/U (number ratio)	Density (gm/cm^3)	Reflector Material	Fuel Solution	Tank Shape	Mean Cord Length (cm)	Absorber	L MENCF	Monk K eff	Monk Std Dev	Total Uncertainty [1]
case42.12	0.0044	5.00	3.01E+00	3.63601	polyethylene	UO2	array	N/A	0	4.18668E-08	0.9988	0.0010	0.0045
case42.13	0.0044	5.00	2.14E+00	3.54333	polyethylene	UO2	array	N/A	0	4.02056E-08	0.9992	0.0010	0.0045
case42.14	0.0044	5.00	2.15E+00	3.54514	polyethylene	UO2	array	N/A	0	3.97593E-08	1.0010	0.0010	0.0045
case42.15	0.0044	5.00	2.15E+00	3.54514	polyethylene	UO2	array	N/A	0	3.88014E-08	1.0052	0.0010	0.0045
case42.16	0.0044	5.00	2.45E+00	3.58076	polyethylene	UO2	array	N/A	0	3.84555E-08	0.9993	0.0010	0.0045
case42.17	0.0044	5.00	2.46E+00	3.58149	polyethylene	UO2	array	N/A	0	3.89409E-08	1.0010	0.0010	0.0045
case42.18	0.0044	5.00	2.46E+00	3.58149	polyethylene	UO2	array	N/A	borated steel plate	3.8728E-08	1.0051	0.0010	0.0045
case43.01	0.0040	4.89	5.44E+01	1.51573	water	Uranium Oxyfluoride	sphere	46.20	0	3.85444E-08	0.9984	0.0010	0.0041
case43.02	0.0037	4.89	4.96E+01	1.55873	bare	Uranium Oxyfluoride	sphere	46.20	0	3.88597E-08	0.9955	0.0010	0.0038
case43.03	0.0044	4.89	4.96E+01	1.55873	water	Uranium Oxyfluoride	sphere	46.20	0	3.92136E-08	0.9997	0.0010	0.0045
case51.01	0.0008	9.97	7.25E+01	1.47998	water	Uranyl Nitrate	cylinder	34.50	0	3.93143E-08	0.9996	0.0010	0.0013
case51.02	0.0009	9.97	7.78E+01	1.45450	water	Uranyl Nitrate	cylinder	36.16	0	3.8481E-08	0.9997	0.0010	0.0013
case51.03	0.0009	9.97	8.49E+01	1.43209	water	Uranyl Nitrate	cylinder	37.89	0	3.86536E-08	0.9988	0.0010	0.0013
case51.04	0.0010	9.97	9.03E+01	1.40631	water	Uranyl Nitrate	cylinder	40.55	0	3.8785E-08	0.9996	0.0010	0.0014
case51.05	0.0010	9.97	9.50E+01	1.39092	water	Uranyl Nitrate	cylinder	42.89	0	3.86648E-08	1.0003	0.0010	0.0014
case51.06	0.0011	9.97	9.91E+01	1.38211	water	Uranyl Nitrate	cylinder	45.08	0	5.06277E-07	0.9992	0.0010	0.0015
case51.07	0.0011	9.97	1.03E+02	1.36952	water	Uranyl Nitrate	cylinder	48.11	0	5.04529E-07	0.9977	0.0010	0.0015
case63.01	0.0041	5.64	5.55E+01	1.58722	water	Uranyl Nitrate	cylinder	56.92	0	1.68649E-06	0.9984	0.0010	0.0042
case63.02	0.0050	5.64	5.55E+01	1.58722	water	Uranyl Nitrate	cylinder	58.40	1 B4C pin	1.63278E-06	0.9977	0.0010	0.0051
case63.03	0.0063	5.64	5.55E+01	1.58722	water	Uranyl Nitrate	cylinder	72.57	7 B4C pins	1.01275E-06	0.9972	0.0010	0.0064
case69.01	0.0040	29.83%	1.33E+00	4.00656	bare	UF4CF2	slab	24.93	0	8.1079E-07	1.0095	0.0010	0.0041
case69.02	0.0040	29.83%	2.66E+00	3.49019	bare	UF4CF2	slab	21.01	0	0.264123568	1.0074	0.0010	0.0041

Case	Experimental Uncertainty	Enrichment ($^{w}/_{o}$)	H/U (number ratio)	Density (gm/cm^3)	Reflector Material	Fuel Solution	Tank Shape	Mean Cord Length (cm)	Absorber	L MENCF	Monk K eff	Monk Std Dev	Total Uncertainty [1]
case69.03	0.0040	29.83%	5.33E+00	2.84474	bare	UF4CF2	slab	17.60	0	0.264712747	1.0021	0.0010	0.0041
case69.04	0.0040	29.83%	1.07E+01	2.19928	bare	UF4CF2	slab	18.23	0	0.264335874	1.0016	0.0010	0.0041
case69.05	0.0040	29.83%	3.73E+01	1.39246	bare	UF4CF2	slab	25.99	0	0.246428476	1.0048	0.0010	0.0041
case69.06	0.0040	29.83%	5.33E+00	2.84474	bare	UF4CF2	slab	17.02	0	0.249028498	1.0070	0.0010	0.0041
case69.07	0.0040	29.83%	5.33E+00	2.84474	bare	UF4CF2	slab	18.12	0	0.244893338	1.0017	0.0010	0.0041
case69.08	0.0060	29.83%	5.33E+00	2.84474	bare	UF4CF2	slab	19.30	0	7.22791E-06	1.0016	0.0010	0.0061
case69.09	0.0040	29.83%	2.66E+00	3.49019	bare	UF4CF2	slab	21.53	0	2.01864E-06	1.0095	0.0010	0.0041
case69.10	0.0040	29.83%	2.67E+00	3.48945	bare	UF4CF2	slab	17.16	0	6.20493E-07	1.0088	0.0010	0.0041
case69.11	0.0040	29.83%	2.67E+00	3.48953	bare	UF4CF2	slab	13.09	0	2.59273E-07	1.0060	0.0010	0.0041
case69.12	0.0040	29.83%	2.66E+00	3.49019	bare	UF4CF2	slab	7.26	0	1.07235E-07	1.0051	0.0010	0.0041
case69.13	0.0040	29.83%	1.07E+01	2.19928	bare	UF4CF2	slab	14.26	0	6.39598E-07	1.0025	0.0010	0.0041
case69.14	0.0040	29.83%	1.07E+01	2.19928	bare	UF4CF2	slab	10.21	0	6.48795E-07	0.9994	0.0010	0.0041
case69.15	0.0040	29.83%	1.07E+01	2.19928	bare	UF4CF2	slab	6.97	0	6.7632E-07	1.0035	0.0010	0.0041
case69.16	0.0040	29.83%	2.13E+01	1.68291	paraffin	UF4CF2	slab	19.30	0	1.69817E-06	1.0008	0.0010	0.0041
case69.17	0.0040	29.83%	2.66E+00	3.49019	bare	UF4CF2	slab	27.23	0	2.02832E-06	1.0086	0.0010	0.0041
case69.18	0.0040	29.83%	5.33E+00	2.84474	bare	UF4CF2	slab	23.23	0	1.98722E-06	1.0085	0.0010	0.0041
case69.19	0.0040	29.83%	2.13E+01	1.68291	bare	UF4CF2	slab	21.53	0	1.94106E-06	1.0034	0.0010	0.0041
case69.20	0.0040	29.83%	2.66E+00	3.49019	paraffin	UF4CF2	slab	22.12	0	2.58796E-07	1.0155	0.0010	0.0041
case69.21	0.0040	29.83%	1.33E+00	4.00707	paraffin	UF4CF2	slab	26.71	0	2.5422E-07	1.0102	0.0010	0.0041
case69.22	0.0040	29.83%	2.66E+00	3.49058	paraffin	UF4CF2	slab	26.11	cadmium	2.58348E-07	1.0089	0.0010	0.0041
case69.23	0.0040	29.83%	5.33E+00	2.84474	paraffin	UF4CF2	slab	21.53	cadmium	1.47446E-07	1.0058	0.0010	0.0041
case69.24	0.0040	29.83%	5.33E+00	2.84474	paraffin	UF4CF2	slab	21.53	boron	7.73337E-06	1.0097	0.0010	0.0041
case69.25	0.0040	29.83%	2.13E+01	1.68291	paraffin	UF4CF2	slab	23.32	cadmium	1.46437E-06	1.0039	0.0010	0.0041
case69.26	0.0040	29.83%	2.13E+01	1.68291	paraffin	UF4CF2	slab	21.53	0	1.69838E-07	1.0036	0.0010	0.0041

Case	Experimental Uncertainty	Enrichment ($^{w}/_{o}$)	H/U (number ratio)	Density (gm/cm^3)	Reflector Material	Fuel Solution	Tank Shape	Mean Cord Length (cm)	Absorber	L MENCF	Monk K eff	Monk Std Dev	Total Uncertainty [1]
case69.27	0.0040	29.83%	2.13E+01	1.68291	paraffin	UF4CF2	slab	23.83	0	2.20464E-06	1.0043	0.0010	0.0041
case69.28	0.0040	29.83%	2.66E+00	3.49019	paraffin	UF4CF2	slab	24.41	0	7.81957E-06	1.0089	0.0010	0.0041
case69.29	0.0040	29.83%	2.66E+00	3.49067	paraffin	UF4CF2	slab	26.64	0	7.14527E-06	0.9965	0.0010	0.0041
case71.01	0.0008	9.97	4.73E+01	1.64592	water	Uranyl Nitrate	slab	26.61	0	1.43174E-06	1.0081	0.0010	0.0013
case71.02	0.0008	9.97	5.18E+01	1.59941	water	Uranyl Nitrate	slab	27.18	0	1.49962E-06	1.0041	0.0010	0.0013
case71.03	0.0009	9.97	6.14E+01	1.52341	water	Uranyl Nitrate	slab	28.71	0	1.73459E-07	1.0032	0.0010	0.0013
case71.04	0.0010	9.97	6.56E+01	1.49539	water	Uranyl Nitrate	slab	29.51	0	1.45802E-07	1.0050	0.0010	0.0014
case71.05	0.0010	9.97	7.05E+01	1.46621	water	Uranyl Nitrate	slab	30.64	0	1.43276E-07	1.0017	0.0010	0.0014
case71.06	0.0011	9.97	7.45E+01	1.44620	water	Uranyl Nitrate	slab	32.01	0	1.99974E-06	1.0014	0.0010	0.0015
case71.07	0.0012	9.97	7.78E+01	1.43151	water	Uranyl Nitrate	slab	33.35	0	1.8025E-06	1.0040	0.0010	0.0016
case80.01	0.0009	9.97	7.15E+01	1.48539	bare	Uranyl Nitrate	cylinder	36.20	0	3.87408E-08	0.9928	0.0010	0.0013
case80.02	0.0009	9.97	7.76E+01	1.45439	bare	Uranyl Nitrate	cylinder	38.21	0	4.09228E-08	0.9983	0.0010	0.0013
case80.03	0.0009	9.97	8.49E+01	1.43209	bare	Uranyl Nitrate	cylinder	40.30	0	4.01048E-08	0.9974	0.0010	0.0013
case80.04	0.0010	9.97	9.04E+01	1.40751	bare	Uranyl Nitrate	cylinder	43.60	0	4.18786E-08	0.9993	0.0010	0.0014
case80.05	0.0011	9.97	9.50E+01	1.39143	bare	Uranyl Nitrate	cylinder	46.72	0	4.12788E-08	0.9980	0.0010	0.0015
case81.01	0.0011	9.97	9.63E+01	1.38322	concrete	Uranyl Nitrate	cylinder	43.10	0	3.9789E-08	1.0004	0.0010	0.0015
case81.02	0.0010	9.97	9.60E+01	1.38404	concrete	Uranyl Nitrate	cylinder	42.10	0	3.92215E-08	1.0007	0.0010	0.0014
case81.03	0.0010	9.97	9.59E+01	1.38473	concrete	Uranyl Nitrate	cylinder	41.61	0	3.8655E-08	1.0011	0.0010	0.0014
case81.04	0.0010	9.97	9.64E+01	1.38253	concrete	Uranyl Nitrate	cylinder	41.80	0	3.81169E-08	1.0002	0.0010	0.0014
case84.01	0.0009	9.97	9.44E+01	1.39093	borated concrete	Uranyl Nitrate	cylinder	42.25	0	3.78766E-08	0.9993	0.0010	0.0013
case84.02	0.0009	9.97	9.42E+01	1.39142	borated concrete	Uranyl Nitrate	cylinder	42.70	0	4.10755E-08	1.0024	0.0010	0.0013
case84.03	0.0009	9.97	9.41E+01	1.39193	borated concrete	Uranyl Nitrate	cylinder	42.94	0	4.09139E-08	0.9989	0.0010	0.0013
case85.01	0.0011	9.97	9.54E+01	1.38644	polyethylene	Uranyl Nitrate	cylinder	43.29	0	4.11523E-08	1.0014	0.0010	0.0015

Case	Experimental Uncertainty	Enrichment ($^{w}/_{o}$)	H/U (number ratio)	Density (gm/cm^3)	Reflector Material	Fuel Solution	Tank Shape	Mean Cord Length (cm)	Absorber	L MENCF	Monk K eff	Monk Std Dev	Total Uncertainty [1]
case85.02	0.0010	9.97	9.53E+01	1.38722	polyethylene	Uranyl Nitrate	cylinder	42.78	0	5.17393E-08	1.0016	0.0010	0.0014
case85.03	0.0010	9.97	9.52E+01	1.38774	polyethylene	Uranyl Nitrate	cylinder	42.64	0	4.91982E-08	1.0005	0.0010	0.0014
case85.04	0.0010	9.97	9.50E+01	1.38853	polyethylene	Uranyl Nitrate	cylinder	42.57	0	4.57974E-08	1.0006	0.0010	0.0014

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

ENCLOSURE 2

Updated Safety Analysis Report Pages

Revision 8, February 2006
Including Page Removal and Insertion Instructions

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SAFETY ANALYSIS REPORT, REVISION 8
PAGE REMOVAL AND INSERTION INSTRUCTIONS

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The Nuclear Criticality Safety Program for the National Enrichment Facility (NEF) is in accordance with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 3.71, Nuclear Criticality Safety Standards for Fuels and Material Facilities (NRC, 1998). Regulatory Guide 3.71 (NRC, 1998) provides guidance on complying with the applicable portions of NRC regulations, including 10 CFR 70 (CFR, 2003a), by describing procedures for preventing nuclear criticality accidents in operations involving handling, processing, storing, and transporting special nuclear material (SNM) at fuel and material facilities. The facility is committed to following the guidelines in this regulatory guide for specific ANSI/ANS criticality safety standards with the exception of ANSI/ANS-8.9-1987, "Nuclear Criticality Safety Criteria for Steel-Pipe Intersections Containing Aqueous Solutions of Fissile Material." Piping configurations containing aqueous solutions of fissile material will be evaluated in accordance with ANSI/ANS-8.1-1998 (ANSI, 1998a), using validated methods to determine subcritical limits.

The information provided in this chapter, the corresponding regulatory requirements, and the section of NUREG-1520 (NRC, 2002), Chapter 5 in which the NRC acceptance criteria are presented is summarized below.

Information Category and Requirement	10 CFR 70 Citation	NUREG-1520 Chapter 5 Reference
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Management of the NCS Program	70.61(d) 70.64(a)	5.4.3.1
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Description of Safety Criteria	70.61	5.4.3.4.2
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Section 5.2 Methodologies and Technical Practices		
Methodology	70.61	5.4.3.4.1 5.4.3.4.4 5.4.3.4.6
Section 5.3 Criticality Accident Alarm System (CAAS)		
Criticality Accident Alarm System	70.24	5.4.3.4.3
Section 5.4 Reporting		
Reporting Requirements	Appendix A	5.4.3.4.7 (7)

5.1 THE NUCLEAR CRITICALITY SAFETY (NCS) PROGRAM

The facility has been designed and will be constructed and operated such that a nuclear criticality event is prevented, and to meet the regulatory requirements of 10 CFR 70 (CFR, 2003a). Nuclear criticality safety at the facility is assured by designing the facility, systems and components with safety margins such that safe conditions are maintained under normal and abnormal process conditions and any credible accident. Items Relied On For Safety (IROFS) identified to ensure subcriticality are discussed in the NEF Integrated Safety Analysis Summary.

5.1.1 Management of the Nuclear Criticality Safety (NCS) Program

The NCS criteria in Section 5.2, Methodologies and Technical Practices, are used for managing criticality safety and include adherence to the double contingency principle as stated in the ANSI/ANS-8.1-1998, Nuclear Criticality Safety In Operations with Fissionable Materials Outside Reactors (ANSI, 1998a). The adopted double contingency principle states "process design should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible." Each process that has accident sequences that could result in an inadvertent nuclear criticality at the NEF meets the double contingency principle. The NEF meets the double contingency principle in that process design incorporates sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

Using these NCS criteria, including the double contingency principle, low enriched uranium enrichment facilities have never had an accidental criticality. The plant will produce no greater than 5.0 $^{w}/_{o}$ enrichment. However, as additional conservatism, the nuclear criticality safety analyses are performed assuming a ^{235}U enrichment of 6.0 $^{w}/_{o}$, except for Contingency Dump System traps which are analyzed assuming a ^{235}U enrichment of 1.5 $^{w}/_{o}$, and include appropriate margins to safety. In accordance with 10 CFR 70.61(d) (CFR, 2003b), the general criticality safety philosophy is to prevent accidental uranium enrichment excesses, provide geometrical safety when practical, provide for moderation controls within the UF_6 processes and impose strict mass limits on containers of aqueous, solvent based, or acid solutions containing uranium. Interaction controls provide for safe movement and storage of components. Plant and equipment features assure prevention of excessive enrichment. The plant is divided into six distinctly separate Assay Units (called Cascade Halls) with no common UF_6 piping. UF_6 blending is done in a physically separate portion of the plant. Process piping, individual centrifuges and chemical traps other than the contingency dump chemical traps, are safe by limits placed on their diameters. Product cylinders rely upon uranium enrichment, moderation control and mass limits to protect against the possibility of a criticality event. Each of the liquid effluent collection tanks that hold uranium in solution is mass controlled, as none are geometrically safe. As required by 10 CFR 70.64(a) (CFR, 2003c), by observing the double contingency principle throughout the plant, a criticality accident is prevented. In addition to the double contingency principle, effective management of the NCS Program includes:

- An NCS program to meet the regulatory requirements of 10 CFR 70 (CFR, 2003a) will be developed, implemented, and maintained.

- Safety parameters and procedures will be established.
- The NCS program structure, including definition of the responsibilities and authorities of key program personnel will be provided.
- The NCS methodologies and technical practices will be kept applicable to current configuration by means of the configuration management function. The NCS program will be upgraded, as necessary, to reflect changes in the ISA or NCS methodologies and to modify operating and maintenance procedures in ways that could reduce the likelihood of occurrence of an inadvertent nuclear criticality.
- The NCS program will be used to establish and maintain NCS safety limits and NCS operating limits for IROFS in nuclear processes and a commitment to maintain adequate management measures to ensure the availability and reliability of the IROFS.
- NCS postings will be provided and maintained current.
- NCS emergency procedure training will be provided.
- The NCS baseline design criteria requirements in 10 CFR 70.64(a) (CFR, 2003c) will be adhered to.
- The NCS program will be used to evaluate modifications to operations, to recommend process parameter changes necessary to maintain the safe operation of the facility, and to select appropriate IROFS and management measures.
- The NCS program will be used to promptly detect NCS deficiencies by means of operational inspections, audits, and investigations. Deficiencies will be entered into the corrective action program so as to prevent recurrence of unacceptable performance deficiencies in IROFS, NCS function or management measures.
- NCS program records will be retained as described in Section 11.7, Records Management.

Training will be provided to individuals who handle nuclear material at the facility in criticality safety. The training is based upon the training program described in ANSI/ANS-8.20-1991, Nuclear Criticality Safety Training (ANSI, 1991). The training program is developed and implemented with input from the criticality safety staff, training staff, and management. The training focuses on the following:

- Appreciation of the physics of nuclear criticality safety.
- Analysis of jobs and tasks to determine what a worker must know to perform tasks efficiently.
- Design and development of learning objectives based upon the analysis of jobs and tasks that reflect the knowledge, skills, and abilities needed by the worker.
- Implementation of revised or temporary operating procedures.

Additional discussion of management measures is provided in Chapter 11, Management Measures.

5.1.2 Control Methods for Prevention of Criticality

The major controlling parameters used in the facility are enrichment control, geometry control, moderation control, and/or limitations on the mass as a function of enrichment. In addition, reflection, interaction, and heterogeneous effects are important parameters considered and applied where appropriate in nuclear criticality safety analyses. Nuclear Criticality Safety Evaluations and Analyses are used to identify the significant parameters affected within a particular system. All assumptions relating to process, equipment, material function, and operation, including credible abnormal conditions, are justified, documented, and independently reviewed. Where possible, passive engineered controls are used to ensure NCS. The determination of the safe values of the major controlling parameters used to control criticality in the facility is described below.

Moderation control is in accordance with ANSI/ANS-8.22-1997, Nuclear Criticality Safety Based on Limiting and Controlling Moderators (ANSI, 1997). However, for the purposes of the criticality analyses, it is assumed that UF_6 comes in contact with water to produce aqueous solutions of UO_2F_2 as described in Section 5.2.1.3.3, Uranium Accumulation and Moderation Assumption. A uniform aqueous solution of UO_2F_2, and a fixed enrichment are conservatively modeled using MONK8A (SA, 2001) and the JEF2.2 library. Criticality analyses were performed to determine the maximum value of a parameter to yield $k_{eff} = 1$. The criticality analyses were then repeated to determine the maximum value of the parameter to yield a $k_{eff} = 0.95$. Table 5.1-1, Safe Values for Uniform Aqueous Solution of Enriched UO_2F_2, shows both the critical and safe limits for 5.0 $^w/_o$ and 6.0 $^w/_o$.

Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, lists the safety criteria of Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2, which are used as control parameters to prevent a nuclear criticality event. Although the NEF will be limited to 5.0 $^w/_o$ enrichment, as additional conservatism, the values in Table 5.1-2, Safety Criteria for Buildings/Systems/ Components, represent the limits based on 6.0 $^w/_o$ enrichment except for the Contingency Dump System traps which are limited to 1.5 $^w/_o$ ^{235}U.

The values on Table 5.1-1 are chosen to be critically safe when optimum light water moderation exists and reflection is considered within isolated systems. The conservative modeling techniques provide for more conservative values than provided in ANSI/ANS-8.1 (ANSI, 1998a). The product cylinders are only safe under conditions of limited moderation and enrichment. In such cases, both design and operating procedures are used to assure that these limits are not exceeded.

All Separation Plant components, which handle enriched UF_6, other than the Type 30B and 48Y cylinders and the first stage UF_6 pumps and contingency dump chemical traps, are safe by geometry. Centrifuge array criticality is precluded by a probability argument with multiple operational procedure barriers. Total moderator or H/U ratio control as appropriate precludes product cylinder criticality.

In the Technical Services Building (TSB) criticality safety for uranium loaded liquids is ensured by limiting the mass of uranium in any single tank to less than or equal to 12.2 kg U (26.9 lb U). Individual liquid storage bottles are safe by volume. Interaction in storage arrays is accounted for.

Based on the criticality analyses, the control parameters applied to NEF are as follows:

Enrichment

Enrichment is controlled to limit the percent ^{235}U within any process, vessel, or container, except the contingency dump system, to a maximum enrichment of 5 $^{w}/_{o}$. The design of the contingency dump system controls enrichment to a limit of 1.5 $^{w}/_{o}$ ^{235}U. Although NEF is limited to a maximum enrichment of 5 $^{w}/_{o}$, as added conservatism nuclear criticality safety is analyzed using an enrichment of 6 $^{w}/_{o}$ ^{235}U.

Geometry/Volume

Geometry/volume control may be used to ensure criticality safety within specific process operations or vessels, and within storage containers.

The geometry/volume limits are chosen to ensure k_{eff} (k_{calc} + 3 σ_{calc}) $\leq$ 0.95.

The safe values of geometry/volume define the characteristic dimension of importance for a single unit such that nuclear criticality safety is not dependent on any other parameter assuming 6 $^{w}/_{o}$ ^{235}U for safety margin.

Moderation

Water and oil are the moderators considered in NEF. At NEF the only system where moderation is used as a control parameter is in the product cylinders. Moderation control is established consistent with the guidelines of ANSI/ANS-8.22-1997 (ANSI, 1997) and incorporates the criteria below:

- Controls are established to limit the amount of moderation entering the cylinders.
- When moderation is the only parameter used for criticality control, the following additional criteria are applied. These controls assure that at least two independent controls would have to fail before a criticality accident is possible.
    - Two independent controls are utilized to verify cylinder moderator content.
    - These controls are established to monitor and limit uncontrolled moderator prior to returning a cylinder to production thereby limiting the amount of uncontrolled moderator from entering a system to an acceptable limit.
    - The evaluation of the cylinders under moderation control includes the establishment of limits for the ratio of maximum moderator-to-fissile material for both normal operating and credible abnormal conditions. This analysis has been supported by parametric studies.
- When moderation is not considered a control parameter, either optimum moderation or worst case H/U ratio is assumed when performing criticality safety analysis.

Mass

Mass control may be utilized to limit the quantity of uranium within specific process operations, vessels, or storage containers. Mass control may be used on its own or in combination with

other control methods. Analysis or sampling is employed to verify the mass of the material. Conservative administrative limits for each operation are specified in the operating procedures.

Whenever mass control is established for a container, records are maintained for mass transfers into and out of the container. Establishment of mass limits for a container involves consideration of potential moderation, reflection, geometry, spacing, and enrichment. The evaluation considers normal operations and credible abnormal conditions for determination of the operating mass limit for the container and for the definition of subsequent controls necessary to prevent reaching the safety limits. When only administrative controls are used for mass controlled systems, double batching is conservatively assumed in the analysis.

Reflection

Reflection is considered when performing Nuclear Criticality Safety Evaluations and Analyses. The possibility of full water reflection is considered but the layout of the NEF is a very open design and it is highly unlikely that those vessels and plant components requiring criticality control could become flooded from a source of water within the plant. In addition, neither automatic sprinkler nor standpipe and hose systems are provided in the TSB, Separation Buildings, Blending and Liquid Sampling, CRDB, CAB, and Centrifuge Post Mortem areas. Therefore, full water reflection of vessels has therefore been discounted. However, some select analyses have been performed using full reflection for conservatism. Partial reflection of 2.5 cm (0.984 in) of water is assumed where limited moderating materials (including humans) may be present It is recognized that concrete can be a more efficient reflector than water; therefore, it is modeled in analyses where it is present. When moderation control is identified in the ISA Summary, it is established consistent with the guidelines of ANSI/ANS-8.22-1997 (ANSI, 1997).

Interaction

Nuclear criticality safety evaluations and analyses consider the potential effects of interaction. A non-interacting unit is defined as a unit that is spaced an approved distance from other units such that the multiplication of the subject unit is not increased. Units may be considered non-interacting when they are separated by more than 60 cm (23.6 inches).

If a unit is considered interacting, nuclear criticality safety analyses are performed. Individual unit multiplication and array interaction are evaluated using the Monte Carlo computer code MONK8A to ensure k_{eff} ($k_{calc} + 3\ \sigma_{calc}$) < 0.95.

Concentration, Density and Neutron Absorbers

NEF does not use mass concentration, density, or neutron absorbers as a criticality control parameter.

5.1.3 Safe Margins Against Criticality

Process operations require establishment of criticality safety limits. The facility UF_6 systems involve mostly gaseous operations. These operations are carried out under reduced atmospheric conditions (vacuum) or at slightly elevated pressures not exceeding three atmospheres. It is highly unlikely that any size changes of process piping, cylinders, cold traps, or chemical traps under these conditions, would lead to a criticality situation because a volume or mass limit may be exceeded.

Within the Separations Building, significant accumulations of enriched UF_6 reside only in the Product Low Temperature Take-off Stations, Product Liquid Sampling Autoclaves, Product Blending System or the UF_6 cold traps. All these, except the UF_6 cold traps, contain the UF_6 in 30B and 48Y cylinders. All these significant accumulations are within enclosures protecting them from water ingress. The facility design has minimized the possibility of accidental moderation by eliminating direct water contact with these cylinders of accumulated UF_6. In addition, the facility's stringent procedural controls for enriching the UF_6 assure that it does not become unacceptably hydrogen moderated while in process. The plant's UF_6 systems operating procedures contain safeguards against loss of moderation control (ANSI, 1997). No neutron poisons are relied upon to assure criticality safety.

5.1.4 Description of Safety Criteria

Each portion of the plant, system, or component that may possibly contain enriched uranium is designed with criticality safety as an objective. Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, shows how the safety criteria of Table 5.1-1, Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2, are applied to the facility to prevent a nuclear criticality event. Although the NEF will be limited to 5.0 $^w/_o$ enrichment, as additional conservatism, the values in Table 5.1-2, represent the limits based on 6.0 $^w/_o$ enrichment.

Where there are significant in-process accumulations of enriched uranium as UF_6, the plant design includes multiple features to minimize the possibilities for breakdown of the moderation control limits. These features eliminate direct ingress of water to product cylinders while in process.

5.1.5 Organization and Administration

The criticality safety organization is responsible for implementing the Nuclear Criticality Safety Program. During the design phase, the criticality safety function is performed within the design engineering organization. The criticality safety function for operations is described in the following section.

The criticality safety organization reports to the Health, Safety, and Environment (HS&E) Manager as described in Chapter 2, Organization and Administration. The HS&E Manager is accountable for overall criticality safety of the facility, is administratively independent of production responsibilities, and has the authority to shut down potentially unsafe operations.

Designated responsibilities of the criticality safety staff include the following:

- Establish the Nuclear Criticality Safety Program, including design criteria, procedures, and training
- Provide criticality safety support for integrated safety analyses and configuration control
- Assess normal and credible abnormal conditions
- Determine criticality safety limits for controlled parameters
- Develop and validate methods to support nuclear criticality safety evaluations (NCSEs) (i.e., non-calculation engineering judgments regarding whether existing criticality safety analyses bound the issue being evaluated or whether new or revised safety analyses are required)

- Perform NCS analyses (i.e., calculations), write NCS evaluations, and approve proposed changes in process conditions on equipment involving fissionable material
- Specify criticality safety control requirements and functionality
- Provide advice and counsel on criticality safety control measures, including review and approval of operating procedures
- Support emergency response planning and events
- Evaluate the effectiveness of the Nuclear Criticality Safety Program using audits and assessments
- Provide criticality safety postings that identify administrative controls for operators in applicable work areas.

The minimum qualifications for a criticality safety engineer are a Bachelor of Science (BS) or Bachelor of Arts (BA) degree in science or engineering with at least two years of nuclear industry experience in criticality safety. A criticality safety engineer must understand and have experience in the application and direction of criticality safety programs. The HS&E Manager has the authority and responsibility to assign and direct activities for the criticality safety staff. The criticality safety engineer is responsible for implementation of the NCS program. Criticality safety engineers will be provided in sufficient numbers to implement and support the operation of the NCS program.

The NEF implements the intent of the administrative practices for criticality safety, as contained in Section 4.1.1 of American National Standards Institute/American Nuclear Society (ANSI/ANS)-8.1-1998, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors (ANSI, 1998a). A policy will be established whereby personnel shall report defective NCS conditions and perform actions only in accordance with written, approved procedures. Unless a specific procedure deals with the situation, personnel shall report defective NCS conditions and take no action until the situation has been evaluated and recovery procedures provided.

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5.2 METHODOLOGIES AND TECHNICAL PRACTICES

This section describes the methodologies and technical practices used to perform the Nuclear Criticality Safety (NCS) analyses and NCS evaluations. The determination of the NCS controlled parameters and their application and the determination of the NCS limits on IROFS are also presented.

5.2.1 Methodology

MONK8A (SA, 2001) 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 3-dimensional models for an accurate simulation of neutronic behavior to provide the best estimate neutron multiplication factor, k-effective. Complex models can be simply set up and verified. Additionally, MONK8A (SA, 2001) has demonstrable accuracy over a wide range of applications and is distributed with a validation database comprising critical experiments covering uranium, plutonium and mixed systems over a wide range of moderation and reflection. The experiments selected are regarded as being representative of systems that are widely encountered in the nuclear industry, particularly with respect to chemical plant operations, transportation and storage. The validation database is subject to on-going review and enhancement. A categorization option is available in MONK8A (SA, 2001) to assist the criticality analyst in determining the type of system being assessed and provides a quick check that a calculation is adequately covered by validation cases.

5.2.1.1 Methods Validation

The validation process establishes method bias by comparing measured results from laboratory critical experiments to method-calculated results for the same systems. The verification and validation processes are controlled and documented. The validation establishes a method bias by correlating the results of critical experiments with results calculated for the same systems by the method being validated. Critical experiments are selected to be representative of the systems to be evaluated in specific design applications. The range of experimental conditions encompassed by a selected set of benchmark experiments establishes the area of applicability over which the calculated method bias is applicable. Benchmark experiments are selected that resemble as closely as practical the systems being evaluated in the design application.

The extensive validation database contains a number of experiments applicable to this application involving low and intermediate-enriched uranium. The MONK8A (SA, 2001) code with the JEF2.2 library was validated against these experiments which are provided in the International Handbook of Evaluated Criticality Safety Benchmark Experiments (NEA, 2002) and NUREG/CR-1071 (NRC, 1980). The experiments chosen are provided in Table 5.2-1, Uranium Experiments Used for Validation, along with a brief description. The overall mean calculated value from these 93 configurations is 1.0017 ± 0.0045 and the results are provided in the MONK8A Validation and Verification report (AREVA, 2006).

MONK8A is distributed in ready-to-run executable form. This approach provides the user with a level of quality assurance consistent with the needs of safety analysis. The traceability from source code to executable code is maintained by the code vendor.

In accordance with the guidance in NUREG-1520 (NRC, 2002), code validation for the specific application has been performed (AREVA, 2006). Specifically, the experiments provided in Table 5.2-1, Uranium Experiments Used for Validation, were calculated and documented in the MONK8A Validation and Verification report (AREVA, 2006) for the National Enrichment Facility. In addition, the MONK8A Validation and Verification report (AREVA, 2006) satisfies the commitment to ANSI/ANS-8.1-1998 (ANSI, 1998a) and includes details of computer codes used, operations, recipes for choosing code options (where applicable), cross sections sets, and any numerical parameters necessary to describe the input.

The MONK8A computer code and JEF2.2 library are within the scope of the Quality Assurance Program.

5.2.1.2 Limits on Control and Controlled Parameters

The validation process established a bias by comparing calculations to measured critical experiments. With the bias determined, an upper safety limit (USL) can be determined using the following equation from NUREG/CR-6698, Guide for Validation of Nuclear Criticality Safety Calculational Methodology (NRC, 2001):

$$USL = 1.0 + Bias - \sigma_{Bias} - \Delta_{SM} - \Delta_{AOA}$$

Where the critical experiments are assumed to have a k_{eff} of unity, and the bias was determined by comparison of calculation to experiment. From Section 5.2.1.1, Methods Validation, the bias is positive and since a positive bias may be non-conservative, the bias is set to zero. The σ_{Bias} from the MONK8A Validation and Verification (AREVA, 2006) is 0.0085 and a value of 0.05 is assigned to the subcritical margin, Δ_{SM}. The term Δ_{AOA} is an additional subcritical margin to account for extensions in the area of applicability. Since the experiments in the benchmark are representative of the application, the term Δ_{AOA} is set to zero for systems and components not associated with the Contingency Dump System. For the Contingency Dump System, it was necessary to extrapolate the area of applicability to include 1.5% enrichment and the term Δ_{AOA} is set to 0.0014 to account for this extrapolation. Thus, the USL becomes:

- USL = 1 + 0 – 0.0085 – 0.05 = 0.9415 (for systems and components NOT associated with the Contingency Dump System)
- USL = 1 + 0 – 0.0085 – 0.05 – 0.0014 = 0.9401 (for the Contingency Dump System)

NUREG/CR-6698 (NRC, 2001) indicates that the following condition be demonstrated for all normal and credible abnormal operating conditions:

$$k_{calc} + 2\,\sigma_{calc} < USL$$

The risk of an accidental criticality resulting from NEF operations is inherently low. The low risk warrants the use of an alternate approach.

At the low enrichment limits established for the NEF, sufficient mass of enriched uranic material cannot be accumulated to achieve criticality without moderation. Uranium in the centrifuge plant is inherently a very dry, unmoderated material. Centrifuge separation operations at NEF do not include solutions of enriched uranium. For most components that form part of the centrifuge plant or are connected to it, sufficient mass of moderated uranium can only accumulate by

reaction between UF_6 and moisture in air leaking into plant process systems, leading to the accumulation of uranic breakdown material. Due to the high vacuum requirements for the normal operation of the facility, air inleakage into the process systems is controlled to very low levels and thus the highly moderated condition assumed represents an abnormal condition. In addition, excessive air in-leakage would result in a loss of vacuum, which in turn would cause the affected centrifuges to crash (self destruct) and the enrichment process in the affected centrifuges to stop. As such, buildup of additional mass of moderated uranic breakdown material, such that component becomes filled with sufficient mass of enriched uranic material for criticality, is precluded. Even when accumulated in large UF_6 cylinders or cold traps, neither UF_6 nor UO_2F_2 can achieve criticality without moderation at the low enrichment limit established for the NEF.

Therefore, due to the low risk of accidental criticality associated with NEF operations and the margin that exists in the design and operation of the NEF with respect to nuclear criticality safety, any uncertainty in reactivity calculations associated with methodology, data, and bias is bounded and a margin of subcriticality for safety of 0.05 (i.e., $k_{eff} = k_{calc} + 3\sigma_{calc} < 0.95$) is adequate to ensure subcriticality is maintained under normal and abnormal credible conditions. As such, the NEF will be designed using the equation:

$$k_{eff} = k_{calc} + 3\,\sigma_{calc} < 0.95$$

5.2.1.3 General Nuclear Criticality Safety Methodology

The NCS analyses results provide values of k-effective (k_{eff}) to conservatively meet the upper safety limit. The following sections provide a description of the major assumptions used in the NCS analyses.

5.2.1.3.1 Reflection Assumption

The layout of the NEF is a very open design and it is not considered credible that those vessels and plant components requiring criticality control could become flooded from a source of water within the plant. Full water reflection of vessels has therefore been discounted. However, where appropriate, spurious reflection due to walls, fixtures, personnel, etc. has been accounted for by assuming 2.5 cm (0.984 in) of water reflection around vessels.

5.2.1.3.2 Enrichment Assumption

The NEF will operate with a 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.

5.2.1.3.3 Uranium Accumulation and Moderation Assumption

Most components that form part of the centrifuge plant or are connected to it assume that 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 discussed in the associated nuclear criticality safety analyses documentation). The ratio is based on the assumption that significant quantities of moderated uranium could only accumulate by reaction between UF_6 and moisture in air leaking into the plant. Due to the high vacuum requirements of a centrifuge plant, in-leakage is

controlled at very low levels and thus the H/U ratio of 7 represents an abnormal condition. The maximum H/U ratio of 7 for the uranyl fluoride-water mixture is derived as follows:

The stoichiometric reaction between UF_6 and water vapor in the presence of excess UF_6 can be represented by the equation:

$$UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF$$

Due to its hygroscopic nature, the resulting uranyl fluoride is likely to form a hydrate compound. Experimental studies (Lychev, 1990) suggest that solid hydrates of compositions $UO_2F_2{\cdot}1.5H_2O$ and $UO_2F_2{\cdot}2H_2O$ can form in the presence of water vapor, the former composition being the stable form on exposure to atmosphere.

It is assumed that the hydrate $UO_2F_2{\cdot}1.5H_2O$ is formed and, additionally, that the hydrogen fluoride (HF) produced by the UF_6/water vapor reaction is also retained in the uranic breakdown to give an overall reaction represented by:

$$UF_6 + 3.5H_2O \rightarrow UO_2F_2{\cdot}4HF{\cdot}1.5H_2O$$

For the MONK8A (SA, 2001) calculations, the composition of the breakdown product was simplified to $UO_2F_2{\cdot}3.5H_2O$ that gives the same H/U ratio of 7 as above.

In the case of oils, UF_6 pumps and vacuum pumps use a fully fluorinated perfluorinated polyether (PFPE) type lubricant, often referred to by the trade name "Fomblin." Mixtures of UF_6 and PFPE oil would be a less conservative case than a uranyl fluoride/water mixture, since the maximum HF solubility in PFPE is only about 0.1 $^{w}/_{o}$. Therefore, the uranyl fluoride/water mixture assumption provides additional conservatism in this case.

5.2.1.3.4 Vessel Movement Assumption

The interaction controls placed on movement of vessels containing enriched uranium are specified in the facility procedures. In general, any item in movement (an item being either an individual vessel or a specified batch of vessels) must be maintained at 60 cm (23.6 in) edge separation from any other enriched uranium, and that only one item of each type, e.g., one trap and one pump, may be in movement at one time. These spacing restrictions are relaxed for vessels being removed from fixed positions. In this situation, one vessel may approach an adjacent fixed plant vessel/component without spacing restrictions.

5.2.1.3.5 Pump Free Volume Assumption

There are two types of pumps used in product and dump systems of the plant:

- The vacuum pumps (product and dump) are rotary vane pumps. In the enrichment plant fixed equipment, these are assumed to have a free volume of 14 L (3.7 gal) and are modeled as a cylinder in MONK8A (SA, 2001). This adequately covers all models likely to be purchased.
- The UF_6 pumping units are a combination unit of two pumps, one 500 m^3/hr (17,656 ft^3/hr) pump with a free volume of 8.52 L (2.25 gal) modeled as a cylinder, and a larger 2000 m^3/hr (70,626 ft^3/hr) pump which is modeled explicitly according to manufacturer's drawings.

5.2.1.4 Nuclear Criticality Safety Analyses

Nuclear criticality safety is analyzed for the design features of the plant system or component and for the operating practices that relate to maintaining criticality safety. The analysis of individual systems or components and their interaction with other systems or components containing enriched uranium is performed to assure the criticality safety criteria are met. The nuclear criticality safety analyses and the safe values in Table 5.1-1, Safe Values for Uniform Aqueous Solution of Enriched UO_2F_2, provide a basis for the plant design and criticality hazards identification performed as part of the Integrated Safety Analysis.

Each portion of the plant, system, or component that may possibly contain enriched uranium is designed with criticality safety as an objective. Table 5.1-2, Safety Criteria for Buildings/ Systems/Components, shows how the safe values of Table 5.1-1, are applied to the facility design to prevent a nuclear criticality event. The NEF is designed and operated in accordance with the parameters provided in Table 5.1-2. The Integrated Safety Analysis reviewed the facility design and operation and identified Items Relied On For Safety to ensure that criticality does not pose an unacceptable risk.

Where there are significant in-process accumulations of enriched uranium as UF_6 the plant design includes multiple features to minimize the possibilities for breakdown of the moderation control limits. These features eliminate direct ingress of water to product cylinders while in process.

Each NCS analysis includes, as a minimum, the following information.

- A discussion of the scope of the analysis and a description of the system(s)/process(es) being analyzed.
- A discussion of the methodology used in the criticality calculations, which includes the validated computer codes and cross section library used and the k_{eff} limit used (0.95).
- A discussion of assumptions (e.g. reflection, enrichment, uranium accumulation, moderation, movement of vessels, component dimensions) and the details concerning the assumptions applicable to the analysis.
- A discussion on the system(s)/process(es) analyzed and the analysis performed, including a description of the accident or abnormal conditions assumed.
- A discussion of the analysis results, including identification of required limits and controls.

During the design phase of NEF, the NCS analysis is performed by a criticality safety engineer and independently reviewed by a second criticality safety engineer. During the operation of NEF, the NCS analysis is performed by criticality safety engineer, independently reviewed by a second criticality safety engineer and approved by the HS&E Manager. Only qualified criticality safety engineers can perform NCS analyses and associated independent review.

5.2.1.5 Additional Nuclear Criticality Safety Analyses Commitments

The NEF NCS analyses were performed using the above methodologies and assumptions. NCS analyses also meet the following:

- NCS analyses are performed using acceptable methodologies.

- Methods are validated and used only within demonstrated acceptable ranges.
- The analyses adhere to ANSI/ANS-8.1-1998 (ANSI, 1998a) as it relates to methodologies.
- The validation report statement in Regulatory Guide 3.71 (NRC, 1998) is as follows: LES has demonstrated (1) the adequacy of the margin of safety for subcriticality by assuring that the margin is large compared to the uncertainty in the calculated value of $k_{eff,}$ (2) that the calculation of k_{eff} is 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) that trends in the bias support the extension of the methodology to areas outside the area or areas of applicability.
- A specific reference to (including the date and revision number) and summary description of either a manual or a documented, reviewed, and approved validation report for each methodology are included. Any change in the reference manual or validation report will be reported to the NRC by letter.
- The reference manual and documented reviewed validation report will be kept at the facility.
- The reference manual and validation report are incorporated into the configuration management program.
- The NCS analyses are performed in accordance with the methods specified and incorporated in the configuration management program.
- The NCS methodologies and technical practices in NUREG-1520 (NRC, 2002), Section 5.4.3.4, are used to analyze NCS accident sequences in operations and processes.
- The acceptance criteria in NUREG-1520 (NRC, 2002), Section 3.4, as they relate to: identification of NCS accident sequences, consequences of NCS accident sequences, likelihood of NCS accident sequences, and descriptions of IROFS for NCS accident sequences are met.
- NCS controls and controlled parameters to assure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety are used.
- As stated in ANSI/ANS-8.1-1998 (ANSI, 1998a), process specifications incorporate margins to protect against uncertainties in process variables and against a limit being accidentally exceeded.
- ANSI/ANS-8.7-1998 (ANSI, 1998b), as it relates to the requirements for subcriticality of operations, the margin of subcriticality for safety, and the selection of controls required by 10 CFR 70.61(d) (CFR, 2003b), is used.
- ANSI/ANS-8.10-1983 (ANSI, 1983b), as modified by Regulatory Guide 3.71 (NRC, 1998), as it relates to the determination of consequences of NCS accident sequences, is used.
- If administrative k_{eff} margins for normal and credible abnormal conditions are used, NRC pre-approval of the administrative margins will be sought.

- Subcritical limits for k_{eff} calculations such that: k_{eff} subcritical = 1.0 - bias - margin, where the margin includes adequate allowance for uncertainty in the methodology, data, and bias to assure subcriticality are used.
- Studies to correlate the change in a value of a controlled parameter and its k_{eff} value are performed. The studies include changing the value of one controlled parameter and determining its effect on another controlled parameter and k_{eff}.
- The double contingency principle is met. The double contingency principle is used in determining NCS controls and IROFS.
- The acceptance criteria in NUREG-1520 (NRC, 2002) Section 3.4, as they relate to subcriticality of operations and margin of subcriticality for safety, are met.

5.2.1.6 Nuclear Criticality Safety Evaluations (NCSE)

For any change (i.e., new design or operation, or modification to the facility or to activities of personnel, e.g., site structures, systems, components, computer programs, processes, operating procedures, management measures), that involves or could affect uranium, a NCSE shall be prepared and approved. Prior to implementing the change, it shall be determined that the entire process will be subcritical (with approved margin for safety) under both normal and credible abnormal conditions. If this condition cannot be shown with the NCSE, either a new or revised NCS analysis will be generated that meets the criteria, or the change will not be made.

The NCSE shall determine and explicitly identify the controlled parameters and associated limits upon which NCS depends, assuring that no single inadvertent departure from a procedure could cause an inadvertent nuclear criticality and that the safety basis of the facility will be maintained during the lifetime of the facility. The evaluation ensures that all potentially affected uranic processes are evaluated to determine the effect of the change on the safety basis of the process, including the effect on bounding process assumptions, on the reliability and availability of NCS controls, and on the NCS of connected processes.

The NCSE process involves a review of the proposed change, discussions with the subject matter experts to determine the processes which need to be considered, development of the controls necessary to meet the double contingency principle, and identification of the assumptions and equipment (e.g., physical controls and/or management measures) needed to ensure criticality safety.

Engineering judgment of the criticality safety engineer is used to ascertain the criticality impact of the proposed change. The basis for this judgment is documented with sufficient detail in the NCSE to allow the independent review by a second criticality safety engineer to confirm the conclusions of the judgment of results. Each NCSE includes, as a minimum, the following information.

- A discussion of the scope of the evaluation, a description of the system(s)/process(es) being evaluated, and identification of the applicable nuclear criticality safety analysis.
- A discussion to demonstrate the applicable nuclear criticality safety analysis is bounding for the condition evaluated.

- A discussion of the impact on the facility criticality safety basis, including effect on bounding process assumptions, on reliability and availability NCS controls, and on the nuclear criticality safety of connected system(s)/process(es).
- A discussion of the evaluation results, including (1) identification of assumptions and equipment needed to ensure nuclear criticality safety is maintained and (2) identification of limits and controls necessary to ensure the double contingency principle is maintained.

The NCSE is performed and documented by a criticality safety engineer. Once the NCSE is completed and the independent review by a criticality safety engineer is performed and documented, the HS&E Manager approves the NCSE. Only criticality safety engineers who have successfully met the requirements specified in the qualification procedure can perform NCSEs and associated independent review.

The above process for NCSEs is in accordance with ANSI/ANS-8.19-1996 (ANSI, 1996).

5.2.1.7 Additional Nuclear Criticality Safety Evaluations Commitments

NCSEs also meet the following:

- The NCSEs are performed in accordance with the procedures specified and incorporated in the configuration management program.
- The NCS methodologies and technical practices in NUREG-1520 (NRC, 2002), Sections 5.4.3.4.1(10)(a), (b), (d) and (e), are used to evaluate NCS accident sequences in operations and processes.
- The acceptance criteria in NUREG-1520 (NRC, 2002), Section 3.4, as they relate to: identification of NCS accident sequences, consequences of NCS accident sequences, likelihood of NCS accident sequences, and descriptions of IROFS for NCS accident sequences are met.
- NCS controls and controlled parameters to assure that under normal and credible abnormal conditions, all nuclear processes are subcritical, including use of an approved margin of subcriticality for safety are used.
- The double contingency principle is met. The double contingency principle is used in determining NCS controls and IROFS.
- The acceptance criteria in NUREG-1520 (NRC, 2002) Section 3.4, as they relate to subcriticality of operations and margin of subcriticality for safety, are met.

5.3 CRITICALITY ACCIDENT ALARM SYSTEM (CAAS)

The facility is provided with a Criticality Accident Alarm System (CAAS) as required by 10 CFR 70.24, (CFR, 2003d). Areas where Special Nuclear Material (SNM) is handled, used, or stored in amounts at or above the 10 CFR 70.24 (CFR, 2003d) mass limits are provided with CAAS coverage. Emergency management measures are covered in the facility Emergency Plan.

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5.4 REPORTING

The following are NCS Program commitments related to event reporting:

- A program for evaluating the criticality significance of NCS events will be provided and an apparatus will be in place for making the required notification to the NRC Operations Center. Qualified individuals will make the determination of significance of NCS events. The determination of loss or degradation of IROFS or double contingency principle compliance will be made against the license and 10 CFR 70 Appendix A (CFR, 2003f).

- The reporting criteria of 10 CFR 70 Appendix A and the report content requirements of 10 CFR 70.50 (CFR, 2003g) will be incorporated into the facility emergency procedures.

- The necessary report based on whether the IROFS credited were lost, irrespective of whether the safety limits of the associated parameters were actually exceeded will be issued.

- If it cannot be ascertained within one hour of whether the criteria of 10 CFR 70 Appendix A (CFR, 2003f) Paragraph (a) or (b) apply, the event will be treated as a one-hour reportable event.

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5.5 REFERENCES

ANSI, 1998a. Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors, ANSI/ANS-8.1-1998, American National Standards Institute/American Nuclear Society, 1998.

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CFR, 2003c. Title 10, Code of Federal Regulations, Section 70.64, Requirements for new facilities or new processes at existing facilities, 2003.

CFR, 2003d. Title 10, Code of Federal Regulations, Section 70.24, Criticality accident requirements, 2003.

CFR, 2003e. Title 10, Code of Federal Regulations, Section 70.72, Facility changes and change process, 2003.

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

SA, 2001. Serco Assurance, ANSWERS Software Service, "Users Guide for Version 8 ANSWERS/MONK(98) 6," 1987-2001.

TABLES

Table 5.1-1 Safe Values for Uniform Aqueous Solutions of Enriched UO_2F_2

Parameter	Critical Value $k_{eff} = 1.0$	Safe Value $k_{eff} = 0.95$	Safety Factor
Values for 5.0 $^{w}/_{o}$ enrichment			
Volume	28.9 L (7.6 gal)	21.6 L (5.7 gal)	0.75
Cylinder Diameter	26.2 cm(10.3 in)	23.6 cm (9.3 in)	0.90
Slab Thickness	12.6 cm (5.0 in)	10.7 cm (4.2 in)	0.85
Water Mass	17.3 kg H_2O (38.1 lb H_2O)	12.7 kg H_2O (28.0 lb H_2O)	0.73
Areal Density	11.9 g/cm^2 (24.4 lb/ft^2)	9.8 g/cm^2 (20.1 lb/ft^2)	0.82
Uranium Mass	37 kg U (81.6 lb U)		
- no double batching		26.6 kg U (58.6 lb U)	0.72
- double batching		16.6 kg U (36.6 lb U)	0.45
Values for 6.0 $^{w}/_{o}$ enrichment			
Volume	24 L (6.3 gal)	18 L (4.8 gal)	0.75
Cylinder Diameter	24.4 cm (9.6 in)	21.9 cm (8.6 in)	0.90
Slab Thickness	11.5 cm (4.5 in)	9.9 cm (3.9 in)	0.86
Water Mass	15.4 kg H_2O (34.0 lb H_2O)	11.5 kg H_2O (25.4 lb H_2O)	0.75
Areal Density	9.5 g/cm^2 (19.5 lb/ft^2)	7.5 g/cm^2 (15.4 lb/ft^2)	0.79
Uranium Mass	27 kg U (59.5 lb U)		
- no double batching		19.5 kg U (43.0 lb U)	0.72
- double batching		12.2 kg U (26.9 lb U)	0.45

Table 5.1-2 Safety Criteria for Buildings/Systems/Components
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Building/System/Component	Control Mechanism	Safety Criteria
Enrichment	Enrichment	5.0 $^{w}/_{o}$ (6 $^{w}/_{o}$ ^{235}U used in NCS)
Centrifuges	Diameter	< 21.9 cm (8.6 in)
Product Cylinders (30B)	Moderation	H < 0.95 kg (2.09 lb)
Product Cylinders (48Y)	Moderation	H < 1.05 kg (2.31 lb)
UF_6 Piping	Diameter	< 21.9 cm (8.6 in)
Chemical Traps	Diameter	< 21.9 cm (8.6 in)
Product Cold Trap	Diameter	< 21.9 cm (8.6 in)
Contingency Dump System Traps	Enrichment	1.5 $^{w}/_{o}$ ^{235}U
Tanks	Mass	< 12.2 kg U (26.9 lb U)
Feed Cylinders	Enrichment	< 0.72 $^{w}/_{o}$ ^{235}U
Uranium Byproduct Cylinders	Enrichment	< 0.72 $^{w}/_{o}$ ^{235}U
UF_6 Pumps (first stage)	N/A	Safe by explicit calculation
UF_6 Pumps (second stage)	Volume	< 18.0 L (4.8 gal)
Individual Uranic Liquid Containers, e.g., Fomblin Oil Bottle, Laboratory Flask, Mop Bucket	Volume	< 18.0 L (4.8 gal)
Vacuum Cleaners Oil Containers	Volume	<18.0 L (4.8 gal)

Table 5.2-1 Uranium Experiments Used for Validation

MONK8A Case	Case Description	Number of Experiments	Handbook Reference
25	Low-enriched damp U_3O_8 powder in cubic aluminum cans	10	NUREG/CR-1071
42	Low-enriched damp UO_2 powder reflected by polyethylene	18	LEU-COMP-THERM-049
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
69	Critical arrays of polyethylene-moderated $U(30)F_4$-Polytetrafluoroethylene one-inch cubes	29	IEU-COMP-THERM-001
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

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