MONK 8A
Validation and Verification

Eagle Rock Enrichment Facility

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List of Acronyms

NCS .... Nuclear Criticality Safety
NCSA .... Nuclear Criticality Safety Analysis
EREF .... Eagle Rock Enrichment Facility
ETC .... Enrichment Technology Company Limited
USL .... Upper Safety Limit
CFR .... Code of Federal Regulations
SAR .... Safety Analysis Report
JEF .... Joint European File
AOA .... Area of Applicability
IHECSBE .... International Handbook of Evaluated Criticality Safety Benchmark Experiments
AES .... AREVA Enrichment Services, LLC
PFPE .... Perfluoropolyether
$^w/_o$ .... Percent by weight

1.0 INTRODUCTION

1.1 PURPOSE

The purpose of this document is to present the process and results of the verification and validation of the MONK8A_RU1 (MONK8A) Monte Carlo code package (PC version) using JEF2.2 cross sections and the development of the Upper Safety Limit (USL) from Reference 2. The validated MONK8A code and the established USL will be used for verification of Nuclear Criticality Safety Analyses (NCSAs) performed by Enrichment Technology Company Limited (ETC) for support of the Eagle Rock Enrichment Facility (EREF).

1.2 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.

1.3 BACKGROUND INFORMATION AND REGULATORY REQUIREMENTS

1.3.1 BACKGROUND INFORMATION

The purpose of an enrichment facility is to separate a feed stream containing the naturally occurring proportions of uranium isotopes into a product stream (i.e., uranium enriched in the ^{235}U isotope) and a tails stream (i.e., uranium depleted in the ^{235}U isotope). The EREF will be constructed on the AREVA Enrichment Services, LLC (AES) selected site in Bonneville County, Idaho and licensed by the U.S. Nuclear Regulatory Commission under Title 10 CFR Part 70. The facility is designed to applicable U.S. codes and standards and will be operated by AES.

1.3.2 REGULATORY REQUIREMENTS

As required per 10 CFR Part 70.61 [1], “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, the EREF Safety Analysis Report (SAR), Section 5.2.1.5 [2] 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 AOAs and use of the code with 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 [3] Section 5.4.3.4.1(8), which is incorporated by reference in SAR Section 5 [2], 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-8.1-1998* [4], 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.

* - NUREG-1520 [3] references ANSI/ANS-8.1-1983. However, this original version of the standard was withdrawn in 1998 and replaced with the current version (ANSI/ANS-8.1-1998 [4]) and reaffirmed in 2007. The use of ANSI/ANS-8.1-1998 [4] is justified because the content of the standard was endorsed by the NRC (with exceptions) in Regulation Guide 3.71 (2005) [5].

In addition, SAR Section 5.2.1.1 [2] requires the validation report to meet AREVA's commitments to ANSI/ANS 8.1-1998 [4] 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.

2.0 ANALYTICAL METHODOLOGY

2.1 CALCULATION METHOD

The MONK8A Monte Carlo code package is used for EREF NCSAs. The code package, available through Serco Assurance, is installed and verified on an AREVA NP personal computer hardware platform (AREVA PC).

MONK8A 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 (k_{eff}). Complex configurations can be simply modeled and verified. Additionally, MONK8A has demonstrable accuracy over a wide range of applications. The EREF NCSAs are performed using MONK8A and the JEF2.2 data library. Specifically, the data library files listed in Table 2-1 were used for the MONK8A verification and validation runs. These files were provide by the computer code vendor, Serco, and are stored on the AREVA 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. Appendix A includes one input file for each of the 11 experiments.

Table 2-1: Data Libraries for Validation and Verification

Library Types :	**Library Names :**
MTCDB Database of composition of standard materials	monk_matdbv2.dat
DICE Point energy neutron library	dice96j2v2.dat
THERM Thermal library that must be used when H bound in water or polythene (polyethylene) is present.	therm96j2v2.dat

2.2 CRITICALITY CODE VALIDATION METHOD

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 EREF, the MONK8A code package was procured, installed and verified on the AREVA PC. 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 NCSA calculations, which are presented in separate documents. A summary of the results from the validation calculations is provided in Section 8.

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

- Identify general EREF 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 EREF design applications and key parameters associated with the normal and upset design conditions. Table 2-2 lists key parameters for the EREF.

Table 2-2: Characteristics / Key Parameters of the EREF System

Parameter	Fissile Material Physical / Chemical Form	Isotropic Composition of Fissile Material	Type of Moderation Materials	Anticipated Reflector Materials	Typical Geometry
	Uranyl Fluoride	≤ 5 $^{w}/_{o}$ ^{235}U	Hydrogen, PFPE Oil*, Carbon	Water, Concrete	Spheres, Cylinders, Slabs

* - Perfluoropolyether (PFPE)

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-8.1-1998 [4] 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 EREF 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 (IHECSBE) [6] and from NUREG/CR-1071 [7]. A description of all relevant experiments used is provided in Section 4.

The third step involves modeling the critical experiments and calculating the k_{eff} values of the selected critical benchmark experiments. Appendix C 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 NCS 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 3.3.

2.2.1 MONK8A CASES

ANSI/ANS-8.1-1998 [4] 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 unity. 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 [4]. 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 calculation uncertainties lie 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. Δβ is determined at the 95% confidence level for the EREF.

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 Δβ 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-uranium 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 [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 2.2.2 and 2.2.3 is taken from NUREG/CR-6698 [8].

2.2.2 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 as follows.

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

$K_{fit}(x)$ is the function derived in the trend analysis described in Section 2.2.5. Because a positive bias may be nonconservative, the following equation shall 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 (i.e., fit = 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 “ith” k_{eff} value
$\bar{x}$ = the weighted mean of the independent variables
z_{2P-1} = the systematic percentile of the Gaussian or normal distribution that contains the P fraction

$\gamma = \dfrac{1-p}{2}$

$\chi^2_{1-\gamma,n-2}$ = the upper Chi-square percentile.

For a weighted analysis:

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

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

$$S_{p_{fit}}=\sqrt{S_{fit}^2+\left(\overline{\sigma}\right)^2}$$

where

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

2.2.3 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 AOA. 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 2-3. 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 AOA beyond the limits of the validation data.

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

$$K_L = \overline{k}_{eff} - U(S_P)$$

If $\overline{k}_{eff} \geq 1$, then $K_L = 1 - U(S_P)$

where

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

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

where

Δ_{sm} = the margin of subcriticality

Δ_{AOA} = 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.

2.2.4 NONPARAMETRIC STATISTICAL TREATMENT

NUREG/CR-6698 [8] 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 = the non-parametric margin added to account for small sample size and it is obtained from Table 2-4

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

If the smallest k_{eff} value is > 1.0, then the non-parametric K_L becomes:

$K_L = 1 - S_P - NPM$

where, again,

S_P = square root of the pooled variance.

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

where

Δ_{sm} = the margin of subcriticality

Δ_{AOA} = 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

2.2.5 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 following equations, “x" is the independent variable representing some parameter (e.g., H/U). The variable "y" represents k_{eff} and 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 here.

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

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

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

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

2.2.6 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.

2.2.7 APPLICATION OF THE USL

For the EREF, 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 2.2.3. 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-8.1-1998 [4] 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 2.2.5 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 2-3: 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 2-4: 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)

3.0 ASSUMPTIONS

3.1 KEY ASSUMPTIONS

A key assumption is any assumption or limitation that must be verified prior to using the results and/or conclusions of a calculation for a safety-related task. There are no key assumptions in the present calculation.

3.2 JUSTIFIED ASSUMPTIONS

None.

3.3 MODELING SIMPLIFICATIONS AND CONSERVATISM

The EREF 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, while cold traps use a silicone based oil for a heat transfer medium. Mixtures of UF_6 and PFPE 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 [9]. Silicone oil is not included as a potential moderator or reflector because it is bounded by the water reflector considered in the centrifuge plant in the criticality analysis. The hydrogen content is less in silicone based oil than in water. Therefore, the moderator or reflector capabilities of silicone based oil need not be considered in the model.

A uranyl fluoride / water system is the worst combination of materials that can occur in an ETC supplied centrifuge enrichment facility with regard to nuclear criticality safety. In addition, uranium compounds with alumina (Al_2O_3), PFPE oil or active carbon are less reactive than a uranyl fluoride / water system. Alumina and PFPE 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 worse 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 4.2.

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 NCSAs.

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

4.0 DESIGN INPUTS

The EREF has only one AOA for the entire plant; it covers a uranyl fluoride / water mixture.

4.1 DESIGN APPLICATION URANYL FLUORIDE / WATER MIXTURE

As stated earlier, a uranyl fluoride / water system is the worst combination of materials that can occur in an ETC supplied centrifuge enrichment facility with regard to nuclear criticality safety. In addition, uranium compounds with alumina, PFPE oil or active carbon are less reactive than a uranyl fluoride / water system. Alumina and PFPE 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 worse 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 4.2.

Table 4-1 summarizes the anticipated characteristics for the design of the EREF 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 Separation Plant, except Contingency Dump System	Product Cylinder 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	$UO_2F_2 \cdot 3.5H_2O$	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 $UO_2F_2 \cdot 3.5H_2O$	5 $^w/_o$ ^{235}U	1 to 32	4.92E-8 to 2.7E-7

4.2 BENCHMARK EXPERIMENTS

Ten plant specific benchmark experiments, consisting of 83 critical configurations, with uranyl solutions and compounds are selected from the IHECSBE [6] to provide a good statistical base. An additional benchmark experiment, consisting of 10 critical configurations, was selected from NUREG/CR-1071 [7] for additional low enriched, low H/U ratio critical experiments. 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. Appendix A contains a sample MONK8A input for each of the 11 plant specific benchmark experiments. Appendix B is a listing of critical experiment parameters used in the benchmark.

The list of the experiments is provided in Table 4-2. Detailed descriptions of the criticality experiments were extracted from the IHECSBE [6] and from NUREG/CR-1071 [7] and are tabulated in Table 4-3. A description of the key parameters of these experiments is shown in Table 4-4 along side the key parameters used in the EREF NCSA.

Appendix A shows a sample MONK8A input for each of the 11 benchmark experiments. Also shown in Appendix B is key input parameters used in the benchmark.

As shown in Table 4-4, the resulting validated AOA contain the corresponding key parameters of the EREF NCSA for which the MONK8A 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 EREF uses the chemical form uranyl fluoride. In addition, the uranyl fluoride / water system is considered to be worse 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 ± 0.0017 for the uranyl nitrate cases compared to 0.9970 ± 0.0042 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 are therefore 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 EREF NCSA and ranges from 0.787 to 103 for the benchmark cases. Therefore the H/U ratio for the EREF NCSA is bounded by the benchmark cases. The EREF 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 EREF 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 EREF NCSAs for systems and components other than those associated with the Contingency Dump System are 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 EREF NCSA for which the MONK8A 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 MONK8A code to determine the reactivity of systems or components not associated with the Contingency Dump System. For use of the MONK8A 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 4-2: Uranium Experiments Used for Validation

MONK 8A Case Set	Case Description	Number of Experiments	Handbook Reference (Reference 4)
25	Low-enriched damp U_30_8 powder in cubic aluminum cans	10	NUREG/CR-1071 [7]
42	MARACAS Program: Polythene reflected critical configurations with low enriched and low moderated uranium dioxide powder $U(5)O_2$	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-0I0

Table 4-3: Expanded Descriptions of the Criticality Experiments

Handbook Reference	Title	Short Description
NUREG/CR-1071 [7]	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_30_8) is enriched to 4.46 $^w/_o$ ^{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 (5 $^w/_o$ 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, from 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 1950's at the Oak Ridge National Laboratory with low enriched uranium (4.9 $^w/_o$ ^{235}U). Critical experiment measurements were made with uranium oxyfluoride ($U0_2F_2$) solutions in a 27.3" inner-diameter (174 liter) sphere with an aluminum wall 1/16" 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.

Handbook Reference	Title	Short Description
LEU-SOL-THERM-004	STACY: Water-Reflected 10 $^{w}/_{o}$ -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 (JAERI). 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/l 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 $^{w}/_{o}$ ^{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 from 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 gU/l was pumped into the core or inner tank, a stainless steel cylindrical tank with an inner diameter of 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.

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 $^w/_o$ ^{235}U ("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 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 (NUCEF) in the Tokai Research Establishment of the Japan Atomic Energy Research Institute (JAERI). 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/l and the uranium enrichment was 10$^w/_o$. The core tank was unreflected.

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 4-4: Comparison of Key Parameters of the EREF NCSA and Benchmark Cases

	Chemical Form	Isotopics	Hydrogen / Uranium Ratio	Log Mean Energy of Neutron Causing Fission [MeV]
EREF NCSA, except Contingency Dump System	Uranyl Fluoride (UO_2F_2)	6 $^w/_o$ ^{235}U	1 to 32	4.92E-8 to 2.7E-7
EREF NCSA, Contingency Dump System	Uranyl Fluoride (UO_2F_2)	1.5 $^w/_o$ ^{235}U	7	4.92E-8 to 2.7E-7
Benchmark	Uranyl Nitrate ($UO_2(NO_3)_2$) Uranium Fluoride (UF_X) Uranium Oxyfluoride (UO_2F_2) UO_2 powder U_3O_8 powder	4.46 to 29.83 $^w/_o$ ^{235}U	0.787 to 102.613	3.739E-8 to 7.958E-6

5.0 SOFTWARE AND COMPUTER FILES

5.1 SOFTWARE

MONK8A – This software is a powerful Monte Carlo tool for NCS analyses. 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_{eff}. Complex configurations can be simply modeled and verified. Additionally, MONK8A has demonstrable accuracy over a wide range of applications. See Section 2.1 for additional information.

5.2 COMPUTER FILES

The validation of MONK8A requires the execution of various benchmark criticality safety experiments. These experiments are contained within the IHECSBE [6] and NUREG/CR-1071 [7]. See Section 4.2 for additional information.

6.0 RESULTS

6.1 URANYL FLUORIDE / WATER MIXTURE

Ninety three experiments are modeled with MONK8A 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., MONK8A Standard Deviation) are presented in Appendix C. The average and standard deviation calculated for the benchmark is 1.0017 + 0.0034. Figure 6-1 shows the distribution of the calculated k_{eff} values (without error). The results were analyzed statistically and 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: Fission Material Density, H/U ratio, ^{235}U Enrichment, and Mean Cord Length.

The fission material density goes from 1.3695 to 4.6 g/cm^3, the H/U ratio goes from 0.787 to 103, the ^{235}U enrichment goes from 4.46 to 29.83 $^{w}/_{o}$, the cord length goes from 6.97 to 72.57 cm, and the log mean energy of neutron causing fission goes from 3.739E-8 to 7.958E-6 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 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 [8]. As discussed in Section 4.2, the materials for these experiments are acceptable for use in this validation and as shown in Table 4-4 and Appendix C, experiments 25 and 42 cover the lower portion of the neutron energy range for the AOA.

Using the one-sided lower tolerance limit equation:

$$K_L = \overline{k}_{eff} - U(S_P)$$

where

$\overline{k}_{eff}$ is determined from the analysis to be 1.0010 (weighted average) and is set to 1.00

S_P is determined from the analysis to be 0.0044.

Since the sample size is 93, U is conservatively determined from Table 2-3 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:

$$K_L = 1.00 - 2.065 * 0.004434 = 0.9908$$

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 4.2, the benchmark experiments are similar to the actual applications
- As reflected in Section 4.2, the number and quality of benchmark experiments used is high
- The validation methodology described in Section 2.2 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 2.2.

For use of the MONK8A 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 AOA (Δ_{AOA}) is set to 0.0.

For the use of the MONK8A 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 2.2.5. NUREG/CR-6698 [8] allows for extrapolation outside the range bounded by the critical experiments and 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 2.412E-04 (k_{eff} per $^{w}/_{o}$ enrichment) for the low enrichment cases. The extrapolation penalty is then calculated to be:

(4.46-1.5) * 2.412E-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 [8], additional justification is provided for this extrapolation outside 10% of the range bounded by the critical experiments. The IHECSBE [6] 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 = 0.9908-0.05-0.0 = 0.9408

  USL = 0.9408

- For systems or components associated with the Contingency Dump System (i.e., systems or components with assumed enrichments of 1.5

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

  USL = 0.9908-0.05-0.0014 = 0.9394

  USL = 0.9394

Figure 6-1: MONK8A k_{eff} Histogram

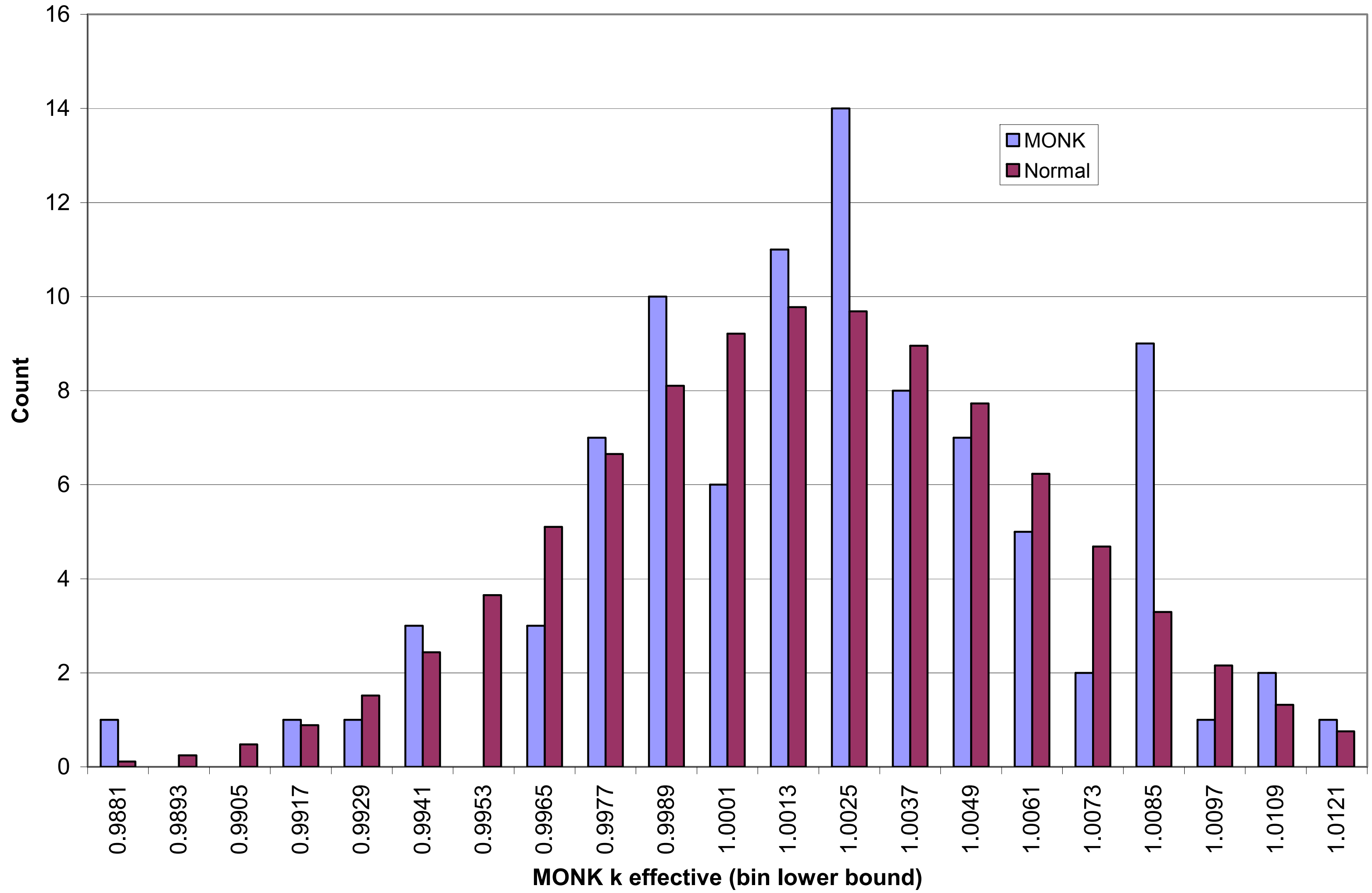

Figure 6-2: Plot of MONK8A k_{eff} vs. Fission Material Density [g/cm^3]

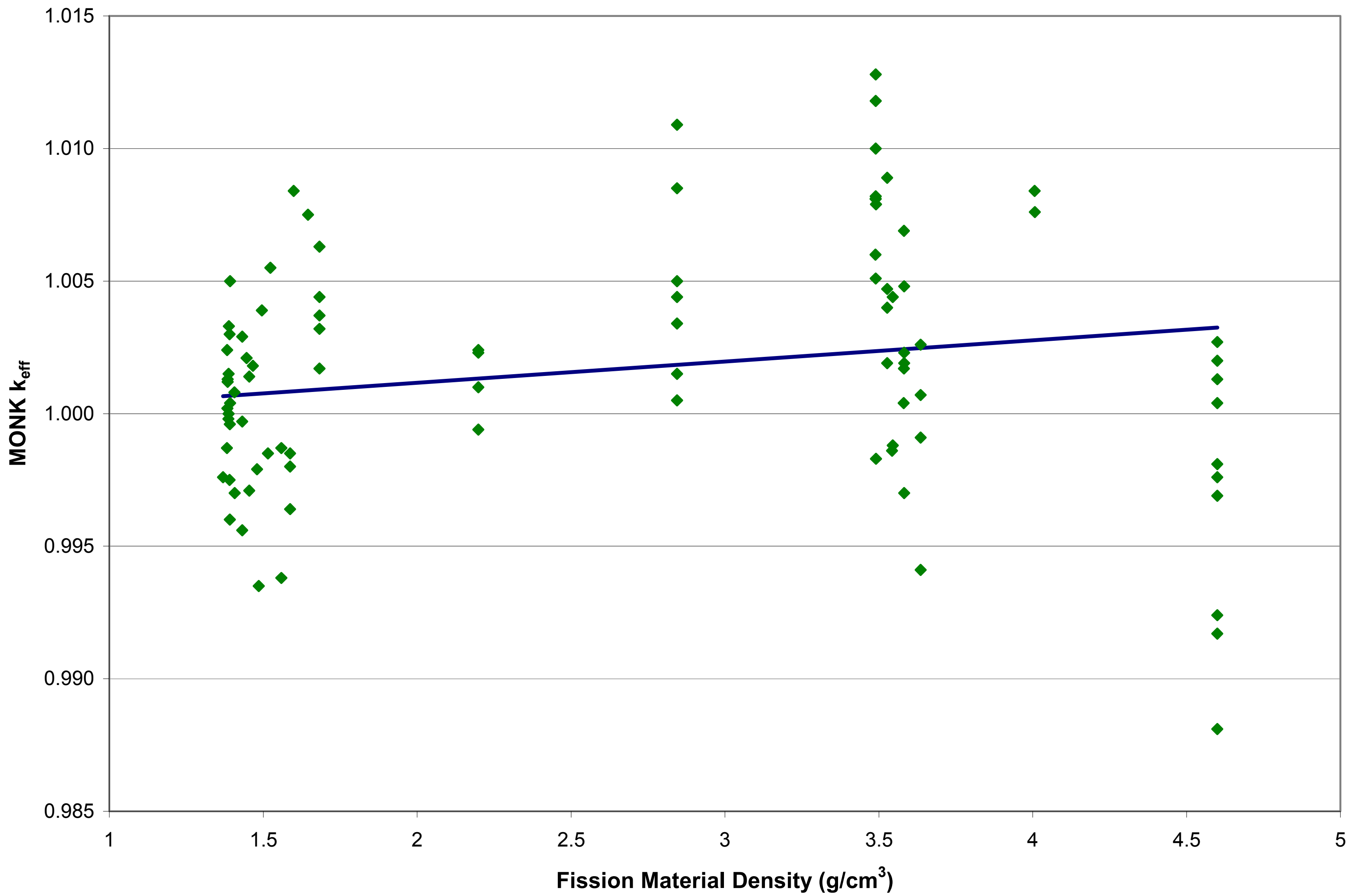

Figure 6-3: Plot of MONK8A k_{eff} vs. H/U Ratio

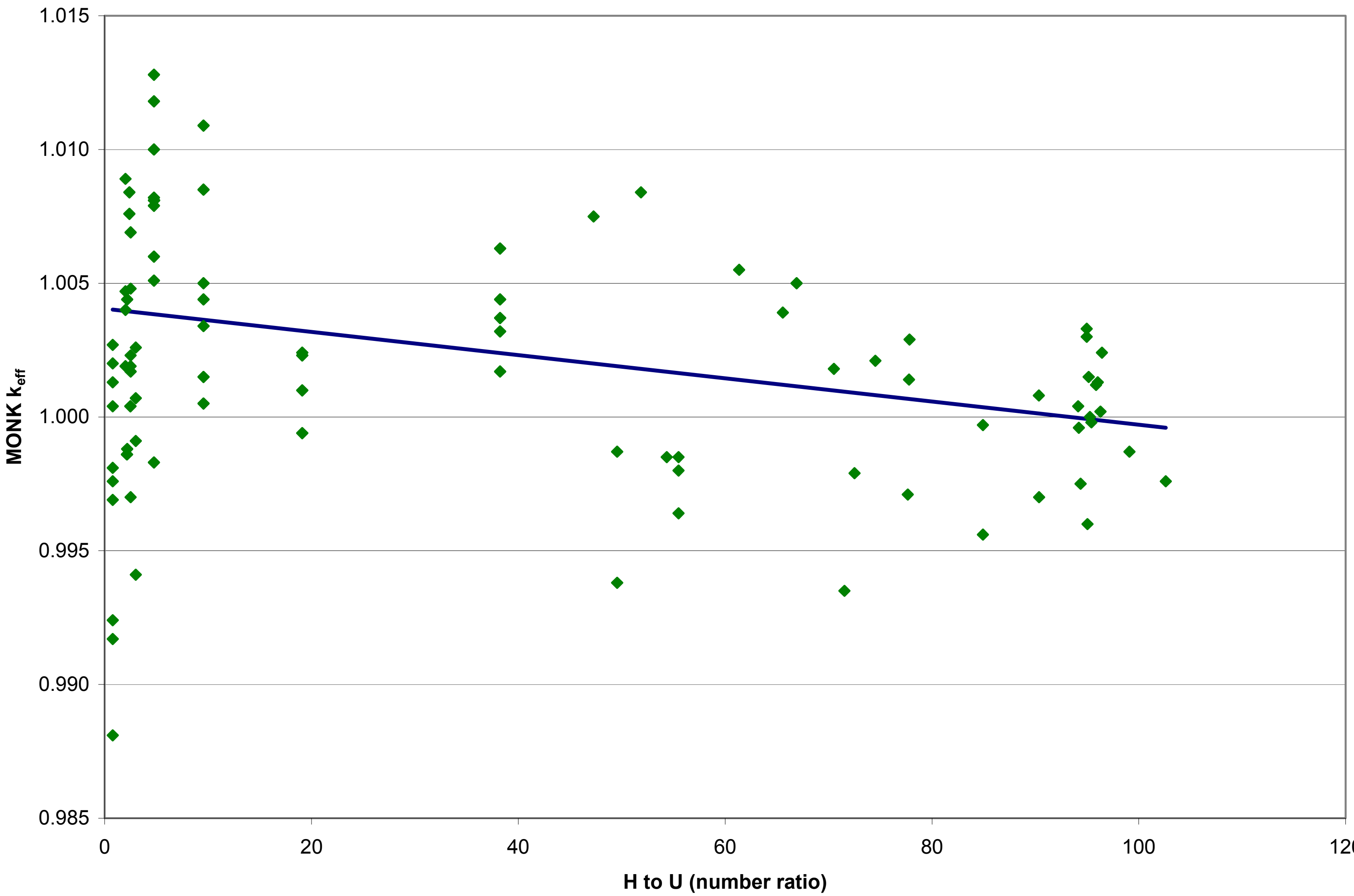

Figure 6-4: Plot of MONK8A k_{eff} vs. ^{235}U Enrichment

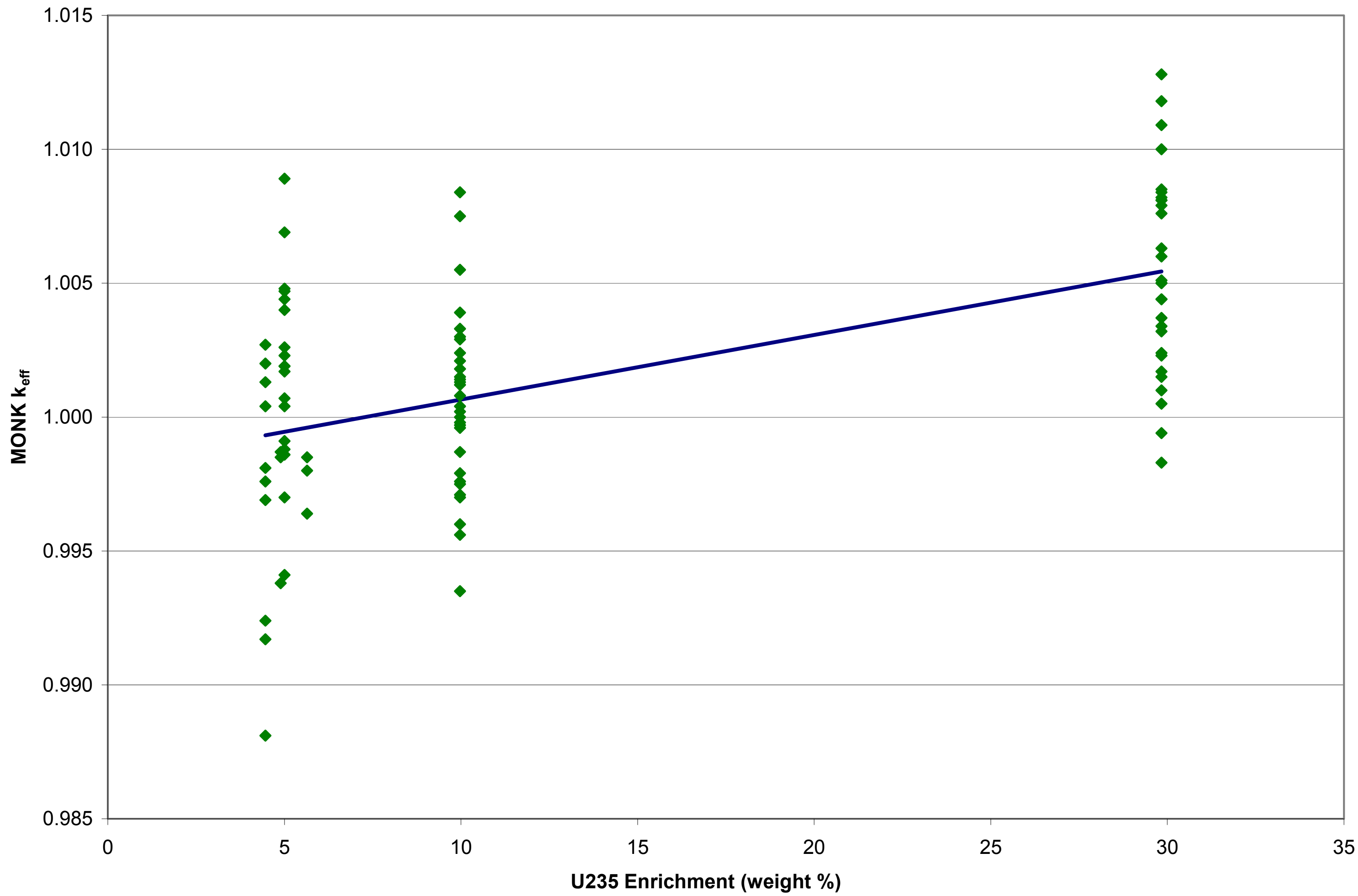

Figure 6-5: Plot of MONK8A k_{eff} vs. Mean Chord Length

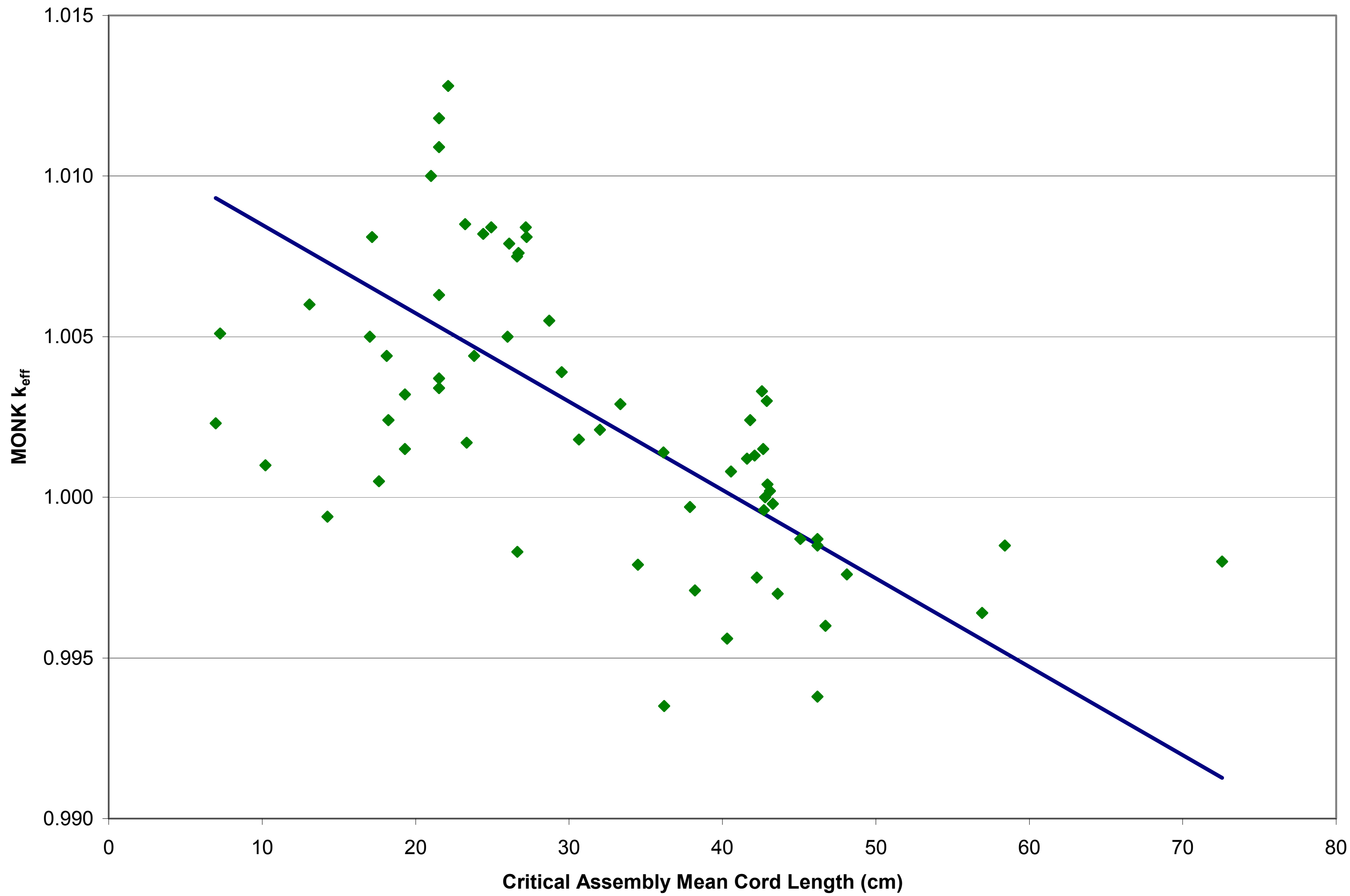

Figure 6-6: Plot of MONK8A k_{eff} vs. Mean Log Energy of Neutron Causing Fission (MLENCF)

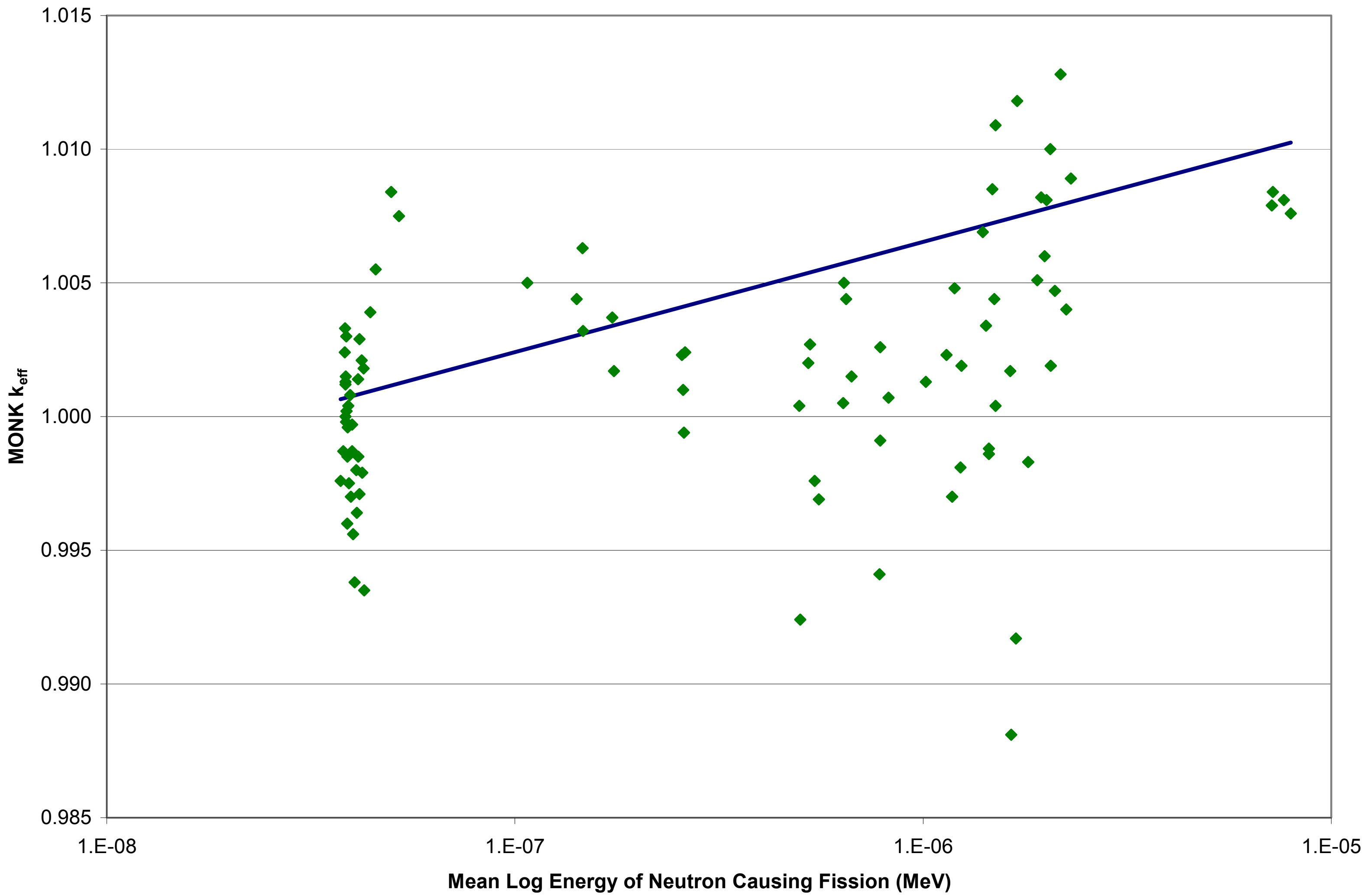

Table 6-1: Summary of Statistical Results

Fitted Parameter	Intercept	Slope	Correlation Coefficient[R^2]	Fitted Range Min	Fitted Range Max
Fission Material Density [g/cm^3]	0.9996	8.023E-04	0.028	1.370	4.6
H to U Number Ratio [unitless]	1.0040	-4.338E-05	0.126	0.787	102.613
^{235}U Enrichment [$^w/_o$]	0.9982	2.412E-04	0.142	4.46	29.83
Mean Cord Length [cm]	1.0112	-2.752E-04	0.373	6.97	72.57
Mean Log Energy of Neutron Causing Fission	1.0006	1.212E+3	0.088	3.739E-8	7.958E-6

* Excluded array cases from mean cord length fit.

7.0 VERIFICATION

NUREG-1520 [3] requires a description of the verification process and results. In addition, NUREG-1520 [3] 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 MONK8A was correctly installed and executed on the AREVA PC. The second step is show that the results are repeatable if run at different times. This step is needed because MONK8A 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 MONK8A criticality analysis cases run by ETC. ETC ran an extensive set of MONK8A criticality calculations in support of their existing facilities and EREF. This step ensures that the cases run by ETC are similar to the AREVA benchmark cases.

7.1 BENCHMARK RESULTS COMPARED TO SERCO RESULTS

The MONK8A 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 AREVA PC and that the mathematical models are working correctly. Table 7-1 shows the results of the MONK8A benchmark calculated by the computer code vendor and from the AREVA verification runs. Table 7-1 has the following definitions.

- "Filename" is the common filename for the benchmark and AREVA run names,
- "Serco Verification Files" - self explanatory,
    - "k_{eff}" is the k_{eff} value from the Serco benchmark case [10],
    - "STDV" is the standard deviation value from the Serco benchmark case [10],
- "AREVA Run-1 (R1-filename)" - list of results for AREVA verification runs using at-the-time-of-run random seed values,
    - "k_{eff}" is the k_{eff} value from the AREVA Run-1 case,
    - "STDV" is the standard deviation value from the AREVA Run-1 case,
- "AREVA Run-2 (R2-filename)"- list of results for AREVA verification runs using random seed values from the Serco Verification Files,
    - "k_{eff}" is the k_{eff} values from the AREVA Run-2 case,
    - "STDV" is the standard deviation value from the AREVA Run-2 case,
- "Count" is the total number of experiments.
- "Average" is the individual group average of the Serco benchmark cases, AREVA Run-1, and AREVA Run-2 validation k_{eff} values calculated using the Excel AVERAGE function.
- "STDEV" 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 (12).

- "Standard Error" is the Standard Error of Measurement [10] of the keff 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 for "AREVA Run-1" were based on the MONK8A 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 12 cases used in this verification is 0.8630 ± 0.1372 and the average of the "AREVA Run-1" verification runs was 0.8620 ± 0.1370, as shown in Table 7-1. Additionally, it is of interest to verify the reproducibility of the Monte Carlo solution. Therefore, all the original Serco random seed values were used for "AREVA Run-2" to track the reproducibility of MONK8A on the QA controlled computer. The results, listed in Table 7-1, are identical when compared to the Serco benchmark cases.

7.2 REPEATABILITY

As mentioned previously, a fundamental feature of all Monte Carlo computer codes is the requirement of a random number to initiate the calculation. By default, MONK8A 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 MONK8A. Therefore, one validation case is chosen for a brief sensitivity study of this effect. Case1.01 listed in Table 7-1 was run at different times to test the repeatability and reliability of MONK8A. The results are summarized in Table 7-2.

The average k_{eff} of the ten runs is 1.0031 with a standard deviation of 0.0013. This demonstrates that MONK8A calculates consistent results since the convergence criterion for the runs is a standard deviation of 0.0015.

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 AREVA PC correctly.

7.3 VERIFICATION OF ETC MONK8A CASES

ETC ran an extensive set of MONK 8A criticality calculations in support of their existing facilities and EREF. Twenty seven (27) representative cases were selected for verification of the

MONK8A criticality analysis run by ETC. As described in the validation section, the default seed values for the random number generator are used to make this verification independent of ETC.

The results of all 27 cases chosen for verification are shown in Table 7-3. The average of the ETC results for the 27 cases used in this report is 0.8753. The average of the verification runs is 0.8754 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

	Serco Verification Files		AREVA Run-1 (R1-*filename*)		AREVA Run-2 (R2-*filename*)	
Filename	**k_{eff}**	**STDV**	**k_{eff}**	**STDV**	**k_{eff}**	**STDV**
case1.01_out	1.0063	0.0015	1.0036	0.0015	1.0063	0.0015
case29.01_out	1.0017	0.0014	1.0012	0.0014	1.0017	0.0014
ex3jef_out	0.5949	0.0014	0.5927	0.0014	0.5949	0.0014
ex4uk_out	0.9358	0.0014	0.9346	0.0014	0.9358	0.0014
ex6jef_out	0.8036	0.0014	0.8056	0.0014	0.8036	0.0014
ex10uk_out	0.8774	0.0014	0.8788	0.0014	0.8774	0.0014
ex3.nm_out	0.5941	0.0014	0.5943	0.0014	0.5941	0.0014
ex10.nm_out	0.8717	0.0014	0.8691	0.0014	0.8717	0.0014
ris1_out	0.9246	0.0029	0.9230	0.0028	0.9246	0.0029
ris4_out	0.9124	0.0029	0.9136	0.0029	0.9124	0.0029
sel2_out	0.8839	0.0025	0.8796	0.0024	0.8839	0.0025
sel7_out	0.9493	0.0019	0.9480	0.0019	0.9493	0.0019
Average	*0.8630*		*0.8620*		*0.8630*	
STDEV	*0.1372*		*0.1370*		*0.1372*	
COUNT	*12*		*12*		*12*	

Table 7-2: Results of Repeatability Sensitivity Study

Output Filename	Time / seed value	Date(1) / seed value	k_{eff}
Rep01-case1.01.out	13.42.50 / 5627	10/08/2008 / 100809	1.0063
Rep02-case1.01.out	13.46.48 / 6621	10/08/2008 / 100809	1.0036
Rep03-case1.01.out	13.50.41 / 7469	10/08/2008 / 100809	1.0034
Rep04-case1.01.out	13.54.46 / 4551	10/08/2008 / 100809	1.0031
Rep05-case1.01.out	13.58.41 / 5461	10/08/2008 / 100809	1.0014
Rep06-case1.01.out	14.02.30 / 10311	10/08/2008 / 100809	1.0030
Rep07-case1.01.out	14.06.28 / 12521	10/08/2008 / 100809	1.0023
Rep08-case1.01.out	14.10.25 / 10933	10/08/2008 / 100809	1.0026
Rep09-case1.01.out	14.14.24 / 14433	10/08/2008 / 100809	1.0029
Rep10-case1.01.out	14.18.11 / 14611	10/08/2008 / 100809	1.0017
		Average:	1.0031
		Standard Deviation:	0.0013

(1) - European date format (dd/mm/yyyy) is used.

Table 7-3: Verification Results

Case	Brief Case Description	k_{eff} ETC	k_{eff} AREVA
1	UF6 Product Pipe work, 65mm pipe - 6w/o H/U=9	0.8075	0.8028
2	UF6 Product Pipe work, 65mm pipe - 6w/o H/U=10	0.8138	0.8144
3	UF6 Product Pipe work, 65mm pipe - 6w/o H/U=11	0.8176	0.8177
4	UF6 Product Pipe work, 65mm pipe - 6w/o H/U=12	0.8173	0.8194
5	UF6 Product Pipe work, 65mm pipe - 6w/o H/U=13	0.8206	0.8215
6	UF6 Product Pipe work, 65mm pipe - 6w/o H/U=14	0.8127	0.8171
7	UF6 Product Pipe work, 200mm pipe - 6w/o H/U=10	0.9407	0.9335
8	UF6 Product Pipe work, 200mm pipe - 6w/o H/U=11	0.9403	0.9403
9	UF6 Product Pipe work, 200mm pipe - 6w/o H/U=12	0.9426	0.9441
10	UF6 Product Pipe work, 200mm pipe - 6w/o H/U=13	0.9378	0.9399
11	UF6 Product Pipe work, 200mm pipe - 6w/o H/U=14	0.9382	0.9403
12	UF6 Product Pipe work, 150mm pipe - 6w/o H/U=11	0.9394	0.9423
13	UF6 Product Pipe work, 150mm pipe - 6w/o H/U=12	0.9399	0.9399
14	UF6 Product Pipe work, 150mm pipe - 6w/o H/U=13	0.9451	0.9451
15	UF6 Product Pipe work, 150mm pipe - 6w/o H/U=14	0.9357	0.9357
16	UF6 Product Pipe work, 100mm pipe - 6w/o H/U=12	0.942	0.942
17	UF6 Product Pipe work, 100mm pipe - 6w/o H/U=13	0.9414	0.9414
18	UF6 Product Pipe work, 100mm pipe - 6w/o H/U=14	0.9397	0.9397
19	6w/o Critical Value- Cylinder Diameter=24.4cm	0.9958	0.9958
20	6w/o Critical Value- Mass=27kgU, H/U=32cm	0.9951	0.9951
21	6w/o Critical Value- Volume=24L	0.9965	0.9965
22	Product UF6 Pump Arrays - 6 w/o H/U=8	0.7285	0.7285
23	Product UF6 Pump Arrays - 6 w/o H/U=10	0.7413	0.7413
24	Product UF6 Pump Arrays - 6 w/o H/U=12	0.7435	0.7435
25	Product UF6 Pump Arrays - 6 w/o H/U=14	0.7349	0.7349
26	Product UF6 Pump Arrays - 6 w/o H/U=16	0.7231	0.7231
27	Cold trap, center-to-center pitch110cm with 2.5cm H2O reflector	0.8012	0.8012
	Average	**0.8753**	**0.8754**

8.0 CONCLUSION

The MONK8A code package using the JEF2.2 data library has been validated to perform nuclear criticality safety calculations for the Eagle Rock Enrichment Facility. The validation covers all plant activities.

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

  the USL is 0.9408.

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 = 0.9394.

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 extrapolated AOA is established at 0.0014.

9.0 REFERENCES

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4. 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.

5. NRC, 2005. Regulatory Guide 3.71, “Standard Review Plan for the Review of a License Application for a Fuel Cycle Facility,” U.S. NRC, October 2005.

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

7. 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.

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

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

APPENDIX A: MONK8A SAMPLE INPUT FILES

A.1 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
```

A.2 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
```

A.3 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
```

A.4 INPUT FILE CASE51.01

```
* 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
```

A.5 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
```

A.6 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

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

A.7 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

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

A.8 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
```

A.9 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
```

A.10 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
```

A.11 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
```

APPENDIX B: LISTING OF CRITICAL EXPERIMENTAL PARAMETERS

Run or Experiment	Input File ID	Handbook ID	Experiment Uncertainty	Fuel Solution	Reflector Material	Tank Shape	Dimension [cm]	Critical Height [cm]	Absorber	4 V/S Mean Chord Length
1	case25.01	NUREG/CR-1071	0.0042	U3O8	plexiglas	array			0	N/A
2	case25.02	NUREG/CR-1071	0.0042	U3O8	plexiglas	array			0	N/A
3	case25.03	NUREG/CR-1071	0.0042	U3O8	plexiglas	array			0	N/A
4	case25.04	NUREG/CR-1071	0.0042	U3O8	plexiglas	array			0	N/A
5	case25.05	NUREG/CR-1071	0.0042	U3O8	plexiglas	array			0	N/A
6	case25.06	NUREG/CR-1071	0.0042	U3O8	plexiglas	array			0	N/A
7	case25.07	NUREG/CR-1071	0.0042	U3O8	plexiglas	array			steel plate	N/A
8	case25.08	NUREG/CR-1071	0.0042	U3O8	plexiglas	array			steel plate	N/A
9	case25.09	NUREG/CR-1071	0.0042	U3O8	plexiglas	array			steel plate	N/A
10	case25.10	NUREG/CR-1071	0.0042	U3O8	plexiglas	array			steel plate	N/A
1	case42.01	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
2	case42.02	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
3	case42.03	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
4	case42.04	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
5	case42.05	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
6	case42.06	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
7	case42.07	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
8	case42.08	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
9	case42.09	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
10	case42.10	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
11	case42.11	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
12	case42.12	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
13	case42.13	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
14	case42.14	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
15	case42.15	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
16	case42.16	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A
17	case42.17	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			0	N/A

Run or Experiment	Input File ID	Handbook ID	Experiment Uncertainty	Fuel Solution	Reflector Material	Tank Shape	Dimension [cm]	Critical Height [cm]	Absorber	4 V/S Mean Chord Length
18	case42.18	LEU-COMP-THERM-049	0.0044	UO2	polyethylene	array			Borated Steel Plate	N/A
1	case43.01	LEU-SOL-THERM-002	0.0040	UO2F2	water	sphere	69.3	62.5	0	46.20
2	case43.02	LEU-SOL-THERM-002	0.0037	UO2F2	bare	sphere	69.3	64.6	0	46.20
3	case43.03	LEU-SOL-THERM-002	0.0044	UO2F2	water	sphere	69.3	51.4	0	46.20
1	case51.01	LEU-SOL-THERM-004	0.0008	Uranyl Nitrate	water	cylinder	59	41.53	0	34.50
29	case51.02	LEU-SOL-THERM-004	0.0009	Uranyl Nitrate	water	cylinder	59	46.7	0	36.16
33	case51.03	LEU-SOL-THERM-004	0.0009	Uranyl Nitrate	water	cylinder	59	52.93	0	37.89
34	case51.04	LEU-SOL-THERM-004	0.0010	Uranyl Nitrate	water	cylinder	59	64.85	0	40.55
46	case51.05	LEU-SOL-THERM-004	0.0010	Uranyl Nitrate	water	cylinder	59	78.56	0	42.89
51	case51.06	LEU-SOL-THERM-004	0.0011	Uranyl Nitrate	water	cylinder	59	95.5	0	45.08
54	case51.07	LEU-SOL-THERM-004	0.0011	Uranyl Nitrate	water	cylinder	59	130.33	0	48.11
1	case63.01	LEU-SOL-THERM-005	0.0041	Uranyl Nitrate	water	cylinder	110	58.98	0	56.92
2	case63.02	LEU-SOL-THERM-005	0.0050	Uranyl Nitrate	water	cylinder	110	62.25	1 B4C pin	58.40
3	case63.03	LEU-SOL-THERM-005	0.0063	Uranyl Nitrate	water	cylinder	110	106.62	7 B4C pins	72.57
1	case69.01	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	15x14x14		0	24.93
2	case69.02	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	12x12x11		0	21.01
3	case69.03	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	10x10x9		0	17.60
4	case69.04	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	10x10x8		0	18.23
5	case69.05	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	16x14x14		0	25.99
6	case69.06	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	10x10x10		0	17.02
7	case69.07	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	11x10x10		0	18.12
8	case69.08	IEU-COMP-THERM-001	0.006	UF4[CF2]	bare	slab	11x11x10		0	19.30
9	case69.09	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	13x12x12		0	21.53
10	case69.10	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	11x11x14		0	17.16
11	case69.11	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	10x10x19		0	13.09
12	case69.12	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	9x9x39		0	7.26
13	case69.13	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	9x9x11		0	14.26

Run or Experiment	Input File ID	Handbook ID	Experiment Uncertainty	Fuel Solution	Reflector Material	Tank Shape	Dimension [cm]	Critical Height [cm]	Absorber	4 V/S Mean Chord Length
14	case69.14	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	8x8x16		0	10.21
15	case69.15	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	8x7x26		0	6.97
16	case69.16	IEU-COMP-THERM-001	0.004	UF4[CF2]	paraffin	slab	11x11x10		0	19.30
17	case69.17	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	16x16x16		0	27.23
18	case69.18	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	13x14x13		0	23.23
19	case69.19	IEU-COMP-THERM-001	0.004	UF4[CF2]	bare	slab	12x13x12		0	21.53
20	case69.20	IEU-COMP-THERM-001	0.004	UF4[CF2]	paraffin	slab	12x13x11		0	22.12
21	case69.21	IEU-COMP-THERM-001	0.004	UF4[CF2]	paraffin	slab	15x15x13		0	26.71
22	case69.22	IEU-COMP-THERM-001	0.004	UF4[CF2]	paraffin	slab	15x15x14		cadmium	26.11
23	case69.23	IEU-COMP-THERM-001	0.004	UF4[CF2]	paraffin	slab	12x13x12		cadmium	21.53
24	case69.24	IEU-COMP-THERM-001	0.004	UF4[CF2]	paraffin	slab	12x13x12		boron	21.53
25	case69.25	IEU-COMP-THERM-001	0.004	UF4[CF2]	paraffin	slab	13x13x11		cadmium	23.32
26	case69.26	IEU-COMP-THERM-001	0.004	UF4[CF2]	paraffin	slab	12x13x12		0	21.53
27	case69.27	IEU-COMP-THERM-001	0.004	UF4[CF2]	paraffin	slab	14x13x12		0	23.83
28	case69.28	IEU-COMP-THERM-001	0.004	UF4[CF2]	paraffin	slab	14x14x13		0	24.41
29	case69.29	IEU-COMP-THERM-001	0.004	UF4[CF2]	paraffin	slab	16x15x15		0	26.64
105	case71.01	LEU-SOL-THERM-016	0.0008	Uranyl Nitrate	water	slab	28 by 69	40.09	0	26.61
113	case71.02	LEU-SOL-THERM-016	0.0008	Uranyl Nitrate	water	slab	28 by 69	42.77	0	27.18
125	case71.03	LEU-SOL-THERM-016	0.0009	Uranyl Nitrate	water	slab	28 by 69	51.37	0	28.71
129	case71.04	LEU-SOL-THERM-016	0.0010	Uranyl Nitrate	water	slab	28 by 69	56.96	0	29.51
131	case71.05	LEU-SOL-THERM-016	0.0010	Uranyl Nitrate	water	slab	28 by 69	66.39	0	30.64
140	case71.06	LEU-SOL-THERM-016	0.0011	Uranyl Nitrate	water	slab	28 by 69	81.47	0	32.01
196	case71.07	LEU-SOL-THERM-016	0.0012	Uranyl Nitrate	water	slab	28 by 69	102.34	0	33.35
14	case80.01	LEU-SOL-THERM-007	0.0009	Uranyl Nitrate	bare	cylinder	59	46.83	0	36.20
30	case80.02	LEU-SOL-THERM-007	0.0009	Uranyl Nitrate	bare	cylinder	59	54.20	0	38.21
32	case80.03	LEU-SOL-THERM-007	0.0009	Uranyl Nitrate	bare	cylinder	59	63.55	0	40.30
36	case80.04	LEU-SOL-THERM-007	0.0010	Uranyl Nitrate	bare	cylinder	59	83.55	0	43.60
49	case80.05	LEU-SOL-THERM-007	0.0011	Uranyl Nitrate	bare	cylinder	59	112.27	0	46.72
74	case81.01	LEU-SOL-THERM-008	0.0011	Uranyl Nitrate	concrete	cylinder	59	79.99	0	43.10
76	case81.02	LEU-SOL-THERM-008	0.0010	Uranyl Nitrate	concrete	cylinder	59	73.50	0	42.10

Run or Experiment	Input File ID	Handbook ID	Experiment Uncertainty	Fuel Solution	Reflector Material	Tank Shape	Dimension [cm]	Critical Height [cm]	Absorber	4 V/S Mean Chord Length
78	case81.03	LEU-SOL-THERM-008	0.0010	Uranyl Nitrate	concrete	cylinder	59	70.58	0	41.61
72	case81.04	LEU-SOL-THERM-008	0.0010	Uranyl Nitrate	concrete	cylinder	59	71.71	0	41.80
92	case84.01	LEU-SOL-THERM-009	0.0009	Uranyl Nitrate	borated concrete	cylinder	59	74.38	0	42.25
93	case84.02	LEU-SOL-THERM-009	0.0009	Uranyl Nitrate	borated concrete	cylinder	59	77.29	0	42.70
94	case84.03	LEU-SOL-THERM-009	0.0009	Uranyl Nitrate	borated concrete	cylinder	59	78.88	0	42.94
83	case85.01	LEU-SOL-THERM-010	0.0011	Uranyl Nitrate	polyethylene	cylinder	59	81.26	0	43.29
85	case85.02	LEU-SOL-THERM-010	0.0010	Uranyl Nitrate	polyethylene	cylinder	59	77.81	0	42.78
86	case85.03	LEU-SOL-THERM-010	0.0010	Uranyl Nitrate	polyethylene	cylinder	59	76.92	0	42.64
88	case85.04	LEU-SOL-THERM-010	0.0010	Uranyl Nitrate	polyethylene	cylinder	59	76.42	0	42.57

APPENDIX C: TABLE OF VALIDATION RESULTS

Input File ID	Experiment Uncertainty	Enrichment [w/o]	H ratio	Density [g/cm3]	Reflector Material	Fuel Solution	Tank Shape	Mean Chord [cm]	Absorber	LMENCF	Monk keff	Monk Std. Dev.	Total Uncertainty(1)
case25.01	0.0042	4.46	7.87E-01	4.6	plexiglas	U3O8	array	N/A	0	4.974E-07	1.0004	0.0010	0.0043
case25.02	0.0042	4.46	7.87E-01	4.6	plexiglas	U3O8	array	N/A	0	5.003E-07	0.9924	0.0010	0.0043
case25.03	0.0042	4.46	7.87E-01	4.6	plexiglas	U3O8	array	N/A	0	1.688E-06	0.9917	0.0010	0.0043
case25.04	0.0042	4.46	7.87E-01	4.6	plexiglas	U3O8	array	N/A	0	1.646E-06	0.9881	0.0010	0.0043
case25.05	0.0042	4.46	7.87E-01	4.6	plexiglas	U3O8	array	N/A	0	1.016E-06	1.0013	0.0010	0.0043
case25.06	0.0042	4.46	7.87E-01	4.6	plexiglas	U3O8	array	N/A	0	5.286E-07	1.0027	0.0010	0.0043
case25.07	0.0042	4.46	7.87E-01	4.6	plexiglas	U3O8	array	N/A	steel plate	5.428E-07	0.9976	0.0010	0.0043
case25.08	0.0042	4.46	7.87E-01	4.6	plexiglas	U3O8	array	N/A	steel plate	5.553E-07	0.9969	0.0010	0.0043
case25.09	0.0042	4.46	7.87E-01	4.6	plexiglas	U3O8	array	N/A	steel plate	5.234E-07	1.0020	0.0010	0.0043
case25.10	0.0042	4.46	7.87E-01	4.6	plexiglas	U3O8	array	N/A	steel plate	1.235E-06	0.9981	0.0010	0.0043
case42.01	0.0044	5.00	2.01E+00	3.52697	polyethylene	UO2	array	N/A	0	2.055E-06	1.0019	0.0010	0.0045
case42.02	0.0044	5.00	2.52E+00	3.58222	polyethylene	UO2	array	N/A	0	1.178E-06	0.9970	0.0010	0.0045
case42.03	0.0044	5.00	3.01E+00	3.63601	polyethylene	UO2	array	N/A	0	7.818E-07	0.9941	0.0010	0.0045
case42.04	0.0044	5.00	2.01E+00	3.52697	polyethylene	UO2	array	N/A	0	2.103E-06	1.0047	0.0010	0.0045
case42.05	0.0044	5.00	2.52E+00	3.58222	polyethylene	UO2	array	N/A	0	1.194E-06	1.0048	0.0010	0.0045
case42.06	0.0044	5.00	3.01E+00	3.63601	polyethylene	UO2	array	N/A	0	7.849E-07	1.0026	0.0010	0.0045
case42.07	0.0044	5.00	2.01E+00	3.52697	polyethylene	UO2	array	N/A	0	2.241E-06	1.0040	0.0010	0.0045
case42.08	0.0044	5.00	2.52E+00	3.58222	polyethylene	UO2	array	N/A	0	1.141E-06	1.0023	0.0010	0.0045
case42.09	0.0044	5.00	3.01E+00	3.63601	polyethylene	UO2	array	N/A	0	7.848E-07	0.9991	0.0010	0.0045
case42.10	0.0044	5.00	2.01E+00	3.52697	polyethylene	UO2	array	N/A	0	2.303E-06	1.0089	0.0010	0.0045
case42.11	0.0044	5.00	2.52E+00	3.58222	polyethylene	UO2	array	N/A	0	1.242E-06	1.0019	0.0010	0.0045
case42.12	0.0044	5.00	3.01E+00	3.63601	polyethylene	UO2	array	N/A	0	8.233E-07	1.0007	0.0010	0.0045
case42.13	0.0044	5.00	2.16E+00	3.54333	polyethylene	UO2	array	N/A	0	1.450E-06	0.9986	0.0010	0.0045
case42.14	0.0044	5.00	2.18E+00	3.54514	polyethylene	UO2	array	N/A	0	1.450E-06	0.9988	0.0010	0.0045
case42.15	0.0044	5.00	2.18E+00	3.54514	polyethylene	UO2	array	N/A	0	1.494E-06	1.0044	0.0010	0.0045
case42.16	0.0044	5.00	2.50E+00	3.58076	polyethylene	UO2	array	N/A	0	1.505E-06	1.0004	0.0010	0.0045
case42.17	0.0044	5.00	2.51E+00	3.58149	polyethylene	UO2	array	N/A	0	1.635E-06	1.0017	0.0010	0.0045

Input File ID	Experiment Uncertainty	Enrichment [w/o]	H ratio	Density [g/cm3]	Reflector Material	Fuel Solution	Tank Shape	Mean Chord [cm]	Absorber	LMENCF	Monk keff	Monk Std. Dev.	Total Uncertainty(1)
case42.18	0.0044	5.00	2.51E+00	3.58149	polyethylene	UO2	array	N/A	Borated Steel Plate	1.402E-06	1.0069	0.0010	0.0045
case43.01	0.0040	4.89	5.44E+01	1.51573	water	UO2F2	sphere	46.20	0	3.890E-08	0.9985	0.0010	0.0041
case43.02	0.0037	4.89	4.96E+01	1.55873	bare	UO2F2	sphere	46.20	0	4.047E-08	0.9938	0.0010	0.0038
case43.03	0.0044	4.89	4.96E+01	1.55873	water	UO2F2	sphere	46.20	0	3.993E-08	0.9987	0.0010	0.0045
case51.01	0.0008	9.97	7.25E+01	1.47998	water	Uranyl Nitrate	cylinder	34.50	0	4.223E-08	0.9979	0.0010	0.0013
case51.02	0.0009	9.97	7.78E+01	1.4545	water	Uranyl Nitrate	cylinder	36.16	0	4.128E-08	1.0014	0.0010	0.0013
case51.03	0.0009	9.97	8.49E+01	1.43209	water	Uranyl Nitrate	cylinder	37.89	0	3.995E-08	0.9997	0.0010	0.0013
case51.04	0.0010	9.97	9.03E+01	1.40631	water	Uranyl Nitrate	cylinder	40.55	0	3.947E-08	1.0008	0.0010	0.0014
case51.05	0.0010	9.97	9.50E+01	1.39092	water	Uranyl Nitrate	cylinder	42.89	0	3.863E-08	1.0030	0.0010	0.0014
case51.06	0.0011	9.97	9.91E+01	1.38211	water	Uranyl Nitrate	cylinder	45.08	0	3.797E-08	0.9987	0.0010	0.0015
case51.07	0.0011	9.97	1.03E+02	1.36952	water	Uranyl Nitrate	cylinder	48.11	0	3.739E-08	0.9976	0.0010	0.0015
case63.01	0.0041	5.64	5.55E+01	1.58722	water	Uranyl Nitrate	cylinder	56.92	0	4.096E-08	0.9964	0.0010	0.0042
case63.02	0.0050	5.64	5.55E+01	1.58722	water	Uranyl Nitrate	cylinder	58.40	1 B4C pin	4.131E-08	0.9985	0.0010	0.0051
case63.03	0.0063	5.64	5.55E+01	1.58722	water	Uranyl Nitrate	cylinder	72.57	7 B4C pins	4.089E-08	0.9980	0.0010	0.0064
case69.01	0.004	29.83	2.389E+00	4.00656	bare	UF4[CF2]	slab	24.93	0	7.196E-06	1.0084	0.0010	0.0041
case69.02	0.004	29.83	4.778E+00	3.49019	bare	UF4[CF2]	slab	21.01	0	2.050E-06	1.0100	0.0011	0.0041
case69.03	0.004	29.83	9.557E+00	2.84474	bare	UF4[CF2]	slab	17.60	0	6.374E-07	1.0005	0.0011	0.0041
case69.04	0.004	29.83	1.9113E+01	2.19928	bare	UF4[CF2]	slab	18.23	0	2.614E-07	1.0024	0.0011	0.0041
case69.05	0.004	29.83	6.6897E+01	1.39246	bare	UF4[CF2]	slab	25.99	0	1.072E-07	1.0050	0.0010	0.0041
case69.06	0.004	29.83	9.557E+00	2.84474	bare	UF4[CF2]	slab	17.02	0	6.396E-07	1.0050	0.0011	0.0041
case69.07	0.004	29.83	9.557E+00	2.84474	bare	UF4[CF2]	slab	18.12	0	6.474E-07	1.0044	0.0011	0.0041
case69.08	0.006	29.83	9.557E+00	2.84474	bare	UF4[CF2]	slab	19.30	0	6.674E-07	1.0015	0.0011	0.0061
case69.09	0.004	29.83	4.778E+00	3.49019	bare	UF4[CF2]	slab	21.53	0	1.700E-06	1.0118	0.0011	0.0041

Input File ID	Experiment Uncertainty	Enrichment [w/o]	H ratio	Density [g/cm3]	Reflector Material	Fuel Solution	Tank Shape	Mean Chord [cm]	Absorber	LMENCF	Monk keff	Monk Std. Dev.	Total Uncertainty(1)
case69.10	0.004	29.83	4.783E+00	3.48945	bare	UF4[CF2]	slab	17.16	0	2.006E-06	1.0081	0.0010	0.0041
case69.11	0.004	29.83	4.782E+00	3.48953	bare	UF4[CF2]	slab	13.09	0	1.985E-06	1.0060	0.0011	0.0041
case69.12	0.004	29.83	4.778E+00	3.49019	bare	UF4[CF2]	slab	7.26	0	1.905E-06	1.0051	0.0010	0.0041
case69.13	0.004	29.83	1.9113E+01	2.19928	bare	UF4[CF2]	slab	14.26	0	2.596E-07	0.9994	0.0011	0.0041
case69.14	0.004	29.83	1.9113E+01	2.19928	bare	UF4[CF2]	slab	10.21	0	2.584E-07	1.0010	0.0011	0.0041
case69.15	0.004	29.83	1.9113E+01	2.19928	bare	UF4[CF2]	slab	6.97	0	2.565E-07	1.0023	0.0011	0.0041
case69.16	0.004	29.83	3.8227E+01	1.68291	paraffin	UF4[CF2]	slab	19.30	0	1.468E-07	1.0032	0.0011	0.0041
case69.17	0.004	29.83	4.778E+00	3.49019	bare	UF4[CF2]	slab	27.23	0	7.659E-06	1.0081	0.0010	0.0041
case69.18	0.004	29.83	9.557E+00	2.84474	bare	UF4[CF2]	slab	23.23	0	1.478E-06	1.0085	0.0011	0.0041
case69.19	0.004	29.83	3.8223E+01	1.68291	bare	UF4[CF2]	slab	21.53	0	1.732E-07	1.0037	0.0010	0.0041
case69.20	0.004	29.83	4.778E+00	3.49019	paraffin	UF4[CF2]	slab	22.12	0	2.171E-06	1.0128	0.0011	0.0041
case69.21	0.004	29.83	2.387E+00	4.00707	paraffin	UF4[CF2]	slab	26.71	0	7.958E-06	1.0076	0.0010	0.0041
case69.22	0.004	29.83	4.776E+00	3.49058	paraffin	UF4[CF2]	slab	26.11	cadmium	7.154E-06	1.0079	0.0011	0.0041
case69.23	0.004	29.83	9.557E+00	2.84474	paraffin	UF4[CF2]	slab	21.53	cadmium	1.426E-06	1.0034	0.0011	0.0041
case69.24	0.004	29.83	9.557E+00	2.84474	paraffin	UF4[CF2]	slab	21.53	boron	1.505E-06	1.0109	0.0011	0.0041
case69.25	0.004	29.83	3.8227E+01	1.68291	paraffin	UF4[CF2]	slab	23.32	cadmium	1.748E-07	1.0017	0.0011	0.0041
case69.26	0.004	29.83	3.8227E+01	1.68291	paraffin	UF4[CF2]	slab	21.53	0	1.465E-07	1.0063	0.0011	0.0041
case69.27	0.004	29.83	3.8227E+01	1.68291	paraffin	UF4[CF2]	slab	23.83	0	1.417E-07	1.0044	0.0011	0.0041
case69.28	0.004	29.83	4.778E+00	3.49019	paraffin	UF4[CF2]	slab	24.41	0	1.947E-06	1.0082	0.0010	0.0041
case69.29	0.004	29.83	4.776E+00	3.49067	paraffin	UF4[CF2]	slab	26.64	0	1.809E-06	0.9983	0.0010	0.0041
case71.01	0.0008	9.97	4.73E+01	1.64592	water	Uranyl Nitrate	slab	26.61	0	5.203E-08	1.0075	0.0010	0.0013
case71.02	0.0008	9.97	5.18E+01	1.59941	water	Uranyl Nitrate	slab	27.18	0	4.974E-08	1.0084	0.0010	0.0013
case71.03	0.0009	9.97	6.14E+01	1.52341	water	Uranyl Nitrate	slab	28.71	0	4.562E-08	1.0055	0.0010	0.0013
case71.04	0.0010	9.97	6.56E+01	1.49539	water	Uranyl Nitrate	slab	29.51	0	4.425E-08	1.0039	0.0010	0.0014
case71.05	0.0010	9.97	7.05E+01	1.46621	water	Uranyl Nitrate	slab	30.64	0	4.260E-08	1.0018	0.0010	0.0014
case71.06	0.0011	9.97	7.45E+01	1.4462	water	Uranyl Nitrate	slab	32.01	0	4.214E-08	1.0021	0.0010	0.0015
case71.07	0.0012	9.97	7.78E+01	1.43151	water	Uranyl Nitrate	slab	33.35	0	4.159E-08	1.0029	0.0010	0.0016

Input File ID	Experiment Uncertainty	Enrichment [w/o]	H ratio	Density [g/cm3]	Reflector Material	Fuel Solution	Tank Shape	Mean Chord [cm]	Absorber	LMENCF	Monk keff	Monk Std. Dev.	Total Uncertainty(1)
case80.01	0.0009	9.97	7.15E+01	1.48539	bare	Uranyl Nitrate	cylinder	36.20	0	4.273E-08	0.9935	0.0010	0.0013
case80.02	0.0009	9.97	7.76E+01	1.45439	bare	Uranyl Nitrate	cylinder	38.21	0	4.163E-08	0.9971	0.0010	0.0013
case80.03	0.0009	9.97	8.49E+01	1.43209	bare	Uranyl Nitrate	cylinder	40.30	0	4.014E-08	0.9956	0.0010	0.0013
case80.04	0.0010	9.97	9.04E+01	1.40751	bare	Uranyl Nitrate	cylinder	43.60	0	3.963E-08	0.9970	0.0010	0.0014
case80.05	0.0011	9.97	9.50E+01	1.39143	bare	Uranyl Nitrate	cylinder	46.72	0	3.881E-08	0.9960	0.0010	0.0015
case81.01	0.0011	9.97	9.63E+01	1.38322	concrete	Uranyl Nitrate	cylinder	43.10	0	3.866E-08	1.0002	0.0010	0.0015
case81.02	0.0010	9.97	9.60E+01	1.38404	concrete	Uranyl Nitrate	cylinder	42.10	0	3.846E-08	1.0013	0.0010	0.0014
case81.03	0.0010	9.97	9.59E+01	1.38473	concrete	Uranyl Nitrate	cylinder	41.61	0	3.848E-08	1.0012	0.0010	0.0014
case81.04	0.0010	9.97	9.64E+01	1.38253	concrete	Uranyl Nitrate	cylinder	41.80	0	3.831E-08	1.0024	0.0010	0.0014
case84.01	0.0009	9.97	9.44E+01	1.39093	borated concrete	Uranyl Nitrate	cylinder	42.25	0	3.922E-08	0.9975	0.0010	0.0013
case84.02	0.0009	9.97	9.42E+01	1.39142	borated concrete	Uranyl Nitrate	cylinder	42.70	0	3.895E-08	0.9996	0.0010	0.0013
case84.03	0.0009	9.97	9.41E+01	1.39193	borated concrete	Uranyl Nitrate	cylinder	42.94	0	3.903E-08	1.0004	0.0010	0.0013
case85.01	0.0011	9.97	9.95E+01	1.38644	polyethylene	Uranyl Nitrate	cylinder	43.29	0	3.855E-08	0.9998	0.0010	0.0015
case85.02	0.0010	9.97	9.53E+01	1.38722	polyethylene	Uranyl Nitrate	cylinder	42.78	0	3.844E-08	1.0000	0.0010	0.0014
case85.03	0.0010	9.97	9.52E+01	1.38774	polyethylene	Uranyl Nitrate	cylinder	42.64	0	3.849E-08	1.0015	0.0010	0.0014
case85.04	0.0010	9.97	9.50E+01	1.38853	polyethylene	Uranyl Nitrate	cylinder	42.57	0	3.835E-08	1.0033	0.0010	0.0014

(1) - Total Uncertainty is the statistical combination of the Experimental Uncertainty (i.e., σ_e) and the MONK8A Standard Deviation (i.e., σ_s).