Uncertainty and Target Accuracy Studies for the Very High Temperature Reactor (VHTR) Physics Parameters

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Uncertainty and Target Accuracy Studies for the Very High Temperature Reactor (VHTR) Physics Parameters

by

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ABSTRACT

The potential impact of nuclear data uncertainties on a number of performance parameters (core and fuel cycle) of the prismatic block-type Very High Temperature Reactor (VHTR) has been evaluated and results are presented in this report. An uncertainty analysis has been performed, based on sensitivity theory, which underlines what cross-sections, what energy range and what isotopes are responsible for the most significant uncertainties. In order to give guidelines on priorities for new evaluations or validation experiments, required accuracies on specific nuclear data have been derived, accounting for target accuracies on major design parameters.

Results of an extensive analysis indicate only a limited number of relevant parameters do not meet the target accuracies assumed in this work; this does not imply that the existing nuclear cross-section data cannot be used for the feasibility and pre-conceptual assessments of the VHTR. However, the results obtained depend on the uncertainty data used, and it is suggested to focus some future evaluation work on the production of consistent, as far as possible complete and user oriented covariance data.

1.0 INTRODUCTION

The Very High Temperature Reactor (VHTR) is a leading concept for the Next Generation Nuclear Plant (NGNP) being proposed by the USDOE as a demonstration advanced system for electricity and hydrogen production. A primary reason for this status is the high coolant exit temperature that makes the system quite attractive for the stated purposes. Two VHTR configurations using prismatic block and pebble fuel elements have been proposed. These are referred to as prismatic modular reactor (PMR) and pebble-bed reactor (PBR), respectively. [1] Both reactor types employ coated fuel particles (CFPs) dispersed in a graphite matrix. The fuel-graphite composite is contained either in a cylindrical fuel rod (fuel compact) in the prismatic block type or within a spherical graphite pebble in the pebble-bed type. In the PMR design, the fuel rods are contained in fuel holes in the hexagonal-prismatic fuel elements. Such fuel elements also have holes for coolant and control rods/materials passages, and fuel element handling. The fuel elements are stacked vertically to form a core column, with each core having multiple columns in an annular configuration. In the PBR design, the spherical graphite pebbles are contained in an annular core.

The physics of these reactors and their associated once-through fuel cycles are quite well understood. However, in order to comply more effectively with the system requirements and timely deployment, the optimization of the PMR and PBR requires directing research and development efforts to all areas, in particular to advanced fuel and material development, and also to the reactor physics area. In this last area, the role of nuclear data is quite significant. Most data are by and large available, but their accuracy and validation is still a major concern, particularly if nuclear data uncertainties lead to large errors in the prediction of core physics performance parameters.

In order to make a comprehensive assessment of these potential errors, sensitivity and uncertainty analysis tools are needed. These tools have been widely developed in the past, in particular for the assessment of the performance of fast reactors in the '70s and '80s. Recently, ANL performed a study on the impact of nuclear data uncertainties on the core physics performance parameters (criticality, reactivity coefficients, irradiated fuel isotopic composition, external source effectiveness, etc.) of a generic Accelerator Driven System (ADS), dedicated to

waste transmutation [2]. That study gave indications of the nuclear data uncertainties of relevance and allowed to quantify the requirements for their reduction. A similar study has now been performed for the block-type VHTR (PMR) and the results are reported here. While the fuel burnup and system operations for the pebble-bed type and block-type VHTR are different, it is however expected that the findings of this study would be indicative of the needed improvements of nuclear data for the two systems, because of their similar use of higher enrichment fuel and target of higher burnup compared to the current LWRs systems.

A major problem that is encountered in uncertainty and target accuracy studies is the unavailability of a complete set of nuclear data uncertainties and their correlations (covariance matrices) at the multigroup level for all the nuclides of interest (actinides, fission products and light elements). In this work, an “educated” guess of uncertainties was initially used for the pertinent nuclides, based on the nuclear data performance in the analysis of selected, clean integral experiments (irradiated fuel and sample analysis, criticality and fission rates in zero-power critical facilities) [3]. For the correlations between energy groups, this study used as a first guess a very crude hypothesis of partial energy correlations “by energy band” (PEC). We will call this set of uncertainties (diagonal values) and correlations “ANL Covariance Matrix” [4]. Quite recently, effort has been conducted at the OECD/NEA DataBank to extract relevant covariance data from current evaluations in major data files and to process them in the same multigroup structure used for the current sensitivity and uncertainty calculations [5]. The availability of this limited set of covariance data allowed us to indicate the crucial role of well-established covariance data, and the urgent need to develop a consistent, user-oriented set of covariance data to be associated with the evaluated data files.

The analysis has been applied to a number of integral parameters which characterize a reactor system and its associated fuel cycle: multiplication factor (k_{eff}), temperature reactivity defect, burnup reactivity swing, spent fuel nuclide concentrations, core peak power, spent fuel decay heat in a repository, neutron source of the spent fuel (at fuel fabrication), and dose (radiotoxicity) of the spent fuel at selected times after storage. Additionally, based on the results of the uncertainty analysis and a set of defined target accuracies for the various integral parameters, it has been possible to provide guidelines on potential priority requirements for data improvements.

In Section 2, the theoretical background of the approach used in this work is presented. The source and attributes of the reference VHTR core design utilized in the study are discussed in Section 3. The results of the uncertainty analyses using the cross-section variances only and the PEC model are summarized in Section 4. Section 5 contains the results of the target accuracy study. The conclusions and recommendations are presented in Section 6.

2.0 THEORETICAL APPROACHES FOR UNCERTAINTY AND TARGET ACCURACY ANALYSES

A systematic approach, similar to that used in an earlier ANL study, [2] has been utilized in this work. The approach consists of developing a reference configuration, defining target accuracies for relevant core and fuel-cycle parameters, evaluating the impact of nuclear data uncertainties on these parameters, and defining necessary re-evaluations or measurements for data contributing a large fraction of the total uncertainty.

The impact of the nuclear data uncertainties on the parameters of interest is obtained using uncertainty and target accuracy studies. The uncertainty study requires that sensitivity coefficients for the parameters (their variation with variations in the cross sections) and the covariance matrices for the cross sections be available. Generalized perturbation theory (GPT) is an approach that has been successfully used for calculating the sensitivity coefficients (particularly when many independent variables and a few integral parameters are involved). These coefficients are then folded with the covariance data to derive uncertainties in the values of the parameters arising from the nuclear data. Subsequently, a target accuracy study is performed to obtain information on the cross section uncertainties that would need improvement, including the magnitude of the required improvement.

The theoretical background and equations for deriving the pertinent quantities (sensitivity coefficients, parameter uncertainty, etc) and the code used in the study are summarized in the following paragraphs. Most of the discussion is taken directly from Reference 2.

2.1 Theoretical Background

The principles of uncertainty analysis and its applications to the fission reactor field are well documented (see, e.g., Ref. 6). We will simply recall here that we can represent a generic integral reactor parameter Q (such as K_{eff} , or a reactivity coefficient, or even a reaction rate like the power) as a function of cross-sections:

$$Q = f(\sigma_1, \sigma_2, \dots, \sigma_J) \tag{1}$$

where $\sigma_1, \sigma_2 \dots \sigma_J$ represent cross-sections by isotope, type of reaction and energy range (or energy group, in a multigroup representation). The uncertainties associated with the cross-section can be represented in the form of covariance matrix:

$$C_\sigma = \begin{pmatrix} c_{11} & c_{12} & \cdots & c_{1J} \\ c_{12} & c_{22} & \cdots & c_{2J} \\ \cdots & \cdots & \cdots & \cdots \\ c_{1J} & c_{2J} & \cdots & c_{JJ} \end{pmatrix} \quad (2)$$

where the elements c_{ij} represent the expected values related to the parameters σ_j , and σ_i .

The variations of the integral parameter Q due to variations of σ can be expressed using perturbation theories [7], to evaluate sensitivity coefficients S :

$$\delta Q/Q = \sum_j S_j \frac{\delta \sigma_j}{\sigma_j} \quad (3)$$

where the sensitivity coefficients S_j are formally given by:

$$S_j = \frac{\partial Q}{\partial \sigma_j} \cdot \frac{\sigma_j}{Q} \quad (4)$$

The variance of Q can then be obtained as:

$$var(Q) = \sum_{j,i} S_j S_i c_{ij} \quad (5)$$

To exploit Eq. (5), one needs to obtain explicitly the S_j coefficients and to establish an appropriate covariance matrix. For a set of integral parameters Q_n ($n = 1 \dots N$), the assessment of the variances as given by Eq. (5) is of course relevant, but it is also of relevance to assess the inverse problem, i.e., what are the required data uncertainties to meet specific target accuracies on the Q_n parameter.

The unknown uncertainty data requirements d_i can be obtained solving the following minimization problem [8]:

$$\sum_i \lambda_i / d_i^2 = \min \quad i=1 \dots I \quad (6)$$

with the following constraints:

$$\sum_i S_{ni}^2 d_i^2 < Q_n^T \quad n=1 \dots N \quad (7)$$

where S_{ni} are the sensitivity coefficients for the integral parameter Q_n , and Q_n^T are the target accuracies on the N integral parameters. The λ_i are cost parameters related to each σ_i and should give a relative figure of merit of the difficulty of improving that parameter (e.g., reducing uncertainties with an appropriate experiment).

2.2 Sensitivity Coefficients and Perturbation Theories

For practical purposes, we will distinguish the explicit dependence from some cross-sections (e.g. σ_i^e) and the implicit dependence from some other cross-sections (e.g. σ_j^{im}) in the general expression of any integral parameter Q :

$$Q = f(\sigma_j^{im}, \sigma_i^e). \quad (8)$$

As an example, we consider a reaction rate:

$$R = \langle \underline{\sigma}^e, \underline{\Phi} \rangle \quad (9)$$

where brackets $\langle, \rangle$ indicate integration over the phase space. Note that in the present analysis $\underline{\Phi}$ is the homogeneous flux for the critical core. In Eq. (9), $\underline{\sigma}^e$ can be an energy dependent detector cross-section; R is explicitly dependent on the $\underline{\sigma}^e$ and implicitly dependent on the cross-sections which characterize the system, described by the flux $\underline{\Phi}$. In other terms, R depends on the system cross-sections via $\underline{\Phi}$. Equation (3) can be rewritten as follows:

$$\delta Q / Q = \sum_j S_j \frac{\delta \sigma_j^{im}}{\sigma_j^{im}} + \left(\frac{\partial Q}{\partial \sigma^e} \cdot \frac{\sigma^e}{Q} \right) \cdot \frac{\delta \sigma^e}{\sigma^e} \quad (10)$$

where we have the hypothesis of an explicit dependence of Q on only one σ^e . If we drop the index “ im ”:

$$\delta Q / Q = \sum_j S_j \frac{\delta \sigma_j}{\sigma_j} + \left(\frac{\partial Q}{\partial \sigma^e} \cdot \frac{\sigma^e}{Q} \right) \cdot \frac{\delta \sigma^e}{\sigma^e} = I + D \quad (11)$$

where the term I is generally called indirect effect, and the term D is called direct effect. While the direct effects can be obtained with explicit expressions of the derivatives of Q , the indirect effect (i.e. the sensitivity coefficients S), can be obtained with perturbation expression, most frequently at the first order [7].

In what follows, we will recall in a simplified way the formulations of the sensitivity coefficients at the first order for the indirect effects related to reactivity coefficients [9], reaction rates [7], and nuclide transmutation (i.e., evolution in time [10]). The case of reactivity loss during the irradiation will also be treated.

2.2.1 Reactivity Coefficients

A reactivity coefficient (e.g., the temperature reactivity defect) can be expressed as a variation of the reactivity of the unperturbed system (characterized by a value K of the multiplication factor, a Boltzmann operator M , a flux $\underline{\Phi}$ and an adjoint flux $\underline{\Phi}^*$):

$$\Delta \rho = \left(I - \frac{I}{K_p} \right) - \left(I - \frac{I}{K} \right) = \frac{I}{K} - \frac{I}{K_p} \quad (12)$$

where K_p corresponds to a variation of the Boltzmann operator such that:

$$\begin{aligned} M &\rightarrow M_p (= M + \delta M_p) & \underline{\Phi} &\rightarrow \underline{\Phi}_p (= \underline{\Phi} + \delta \underline{\Phi}_p) \\ \underline{\Phi}^* &\rightarrow \underline{\Phi}_p^* (= \underline{\Phi}^* + \delta \underline{\Phi}_p^*) & K &\rightarrow K_p (= K + \delta K_p) \end{aligned} \quad (13)$$

The sensitivity coefficients (at first order) for $\Delta \rho$ to variations of the σ_j are given as in Ref. 9:

$$S_j^{RO} = \frac{\partial(\Delta\rho)}{\partial\sigma_j} \cdot \frac{\sigma_j}{\Delta\rho} = \left\{ \frac{I}{I_f^p} \langle \underline{\Phi}^*, \sigma_j \underline{\Phi}_p \rangle - \frac{I}{I_f} \langle \underline{\Phi}^*, \sigma_j \underline{\Phi} \rangle \right\} \quad (14)$$

where $I_f = \langle \underline{\Phi}^*, F \underline{\Phi} \rangle$ and $I_f^p = \langle \underline{\Phi}^*, F \underline{\Phi}_p \rangle$, F being the neutron fission production part of the $M (= F - A)$ operator.

2.2.2 Reaction Rates

The classical formulations found in Ref. 7, for example, can be applied to the case of the power peak factor in the core:

$$R = \langle \underline{\Phi}, \underline{\Sigma}_R \rangle \quad (15)$$

The sensitivity coefficients are given by:

$$S_j^R = \langle \underline{\Psi}_R^*, \sigma_j \underline{\Phi} \rangle \quad (16)$$

where $\underline{\Phi}$ has been defined above, and $\underline{\Psi}_R^*$ is the solution of:

$$M^* \underline{\Psi}_R^* = \underline{\Sigma}_R \quad (17)$$

and M^* is the adjoint of the operator M .

2.2.3 Nuclide Transmutation

The generic nuclide K transmutation during irradiation can be represented as the nuclide density variation between time t_0 and t_F . If we denote n_F^K the “final” density, the appropriate sensitivity coefficient is given by:

$$S_j^K = \frac{\partial n_F^K}{\partial \sigma_j} \cdot \frac{\sigma_j}{n_F^K} = \frac{I}{n_F^K} \int_{t_0}^{t_F} \underline{n}^* \sigma_j \underline{n} \, dt \quad (18)$$

where the time dependent equations to obtain $\underline{n}^*$ and $\underline{n}$, together with their boundary conditions, are defined in Ref. 10.

The method previously described does not take into account the coupling with the flux field, [11,12] neglecting in this way the feedback from flux and spectrum changes during irradiation time.

2.2.4 Reactivity Loss during Irradiation, $\Delta\rho^{\text{cycle}}$

At the first order, and neglecting the multigroup nuclide cross-section variation during irradiation, we can write:

$$\Delta\rho^{\text{cycle}} = \sum_K \Delta n^K \rho_K \quad (19)$$

where:

$$\Delta n^K = n_F^K - n_0^K \quad (20)$$

and ρ_K is the reactivity per unit mass associated with the isotope K .

The related sensitivity coefficients S_j^{cycle} associated with the variation of a σ_j , are given by:

$$S_j^{\text{cycle}} = \frac{\sigma_j}{\Delta\rho^{\text{cycle}}} \frac{\partial \Delta\rho^{\text{cycle}}}{\partial \sigma_j} = \frac{\sigma_j}{\Delta\rho^{\text{cycle}}} \left(\sum_K \frac{\partial n^K}{\partial \sigma_j} \cdot \rho_K + \sum_K \Delta n^K \frac{\partial \rho_K}{\partial \sigma_j} \right) \quad (21)$$

Using the formulations of Sec. 2.2.1. and Sec. 2.2.3., we obtain:

$$S_j^{\text{cycle}} = \sum_K \frac{\rho_K}{\Delta\rho^{\text{cycle}}} \int_{t_0}^{t_F} \underline{n}^* \sigma_j \underline{n} \, dt + \left\{ \frac{I}{I_f^p} \langle \underline{\Phi}^*, \sigma_j \underline{\Phi}_p \rangle - \frac{I}{I_f} \langle \underline{\Phi}^*, \sigma_j \underline{\Phi} \rangle \right\} \quad (22)$$

where the index “ p ” refers to the core state at $t = t_F$.

Also in this case the time dependent variation of the flux spectrum during irradiation is assumed to be of negligible impact on the sensitivity coefficients for $\Delta\rho^{cycle}$.

2.2.5 Decay Heat

The decay heat is defined as:

$$H(t) = \sum_K \lambda_K Q_K n_K(t) \quad (23)$$

where for each isotope K , λ_K are the decay constants, Q_K is the heat released in decay reaction and $n_K(t)$ are the nuclide densities at time t . The equations for $n_K(t)$ are the classical ones:

$$\begin{aligned} \frac{dn_K(t)}{dt} = & \sum_F \gamma_{K,f} \tau_f + \sum_j n_K(t) \tau_j b_{j \rightarrow K} + \\ & + \sum_i n_i(t) \lambda_i b_{i \rightarrow K} - \tau_K n_K(t) - \lambda_K n_K(t) \end{aligned} \quad (24)$$

or in a more compact form:

$$\frac{dn_k(t)}{dt} = b_k + \sum_{j=1}^{K-1} C_{kj} n_j(t) - C_{kk} n_k(t) \quad (25)$$

where $\gamma_{K,f}$ are the fission yields for fissionable isotope f , τ are microscopic reaction rates and $b_{j \rightarrow k}$ are branching ratios.

This is an inhomogeneous Bateman-type equation that defines the appropriate nuclide field. The uncertainty on $H(t)$ is obtained by combining the appropriate derivatives of H with respect to λ , Q and n , and accounting for possible correlations. As far as variations of the n_K terms, they can be evaluated using the perturbation techniques indicated in Sec 2.2.3. A specific feature is represented by the variation of the fission yields γ , i.e., by the variation of the “source” term b_K in Eq. (25).

The relative sensitivity coefficients corresponding to the decay heat at $t = t_x$ are given by:

$$S_K^\gamma = \tau_f \frac{\partial n_{t=t_x}^K}{\partial \gamma_{K,f}} \cdot \frac{\gamma_{K,f}}{n_{t=t_x}^K} = \frac{\tau_f}{n_{t=t_x}^K} \int_0^{t_x} \underline{n}^* \gamma_{K,f} dt \quad (31)$$

2.3 Calculational Tools and Basic Data Library

A collection of codes is available at Argonne for the calculation of the generalized importance function and the sensitivity coefficients required for this study. These include the European ERANOS code system, [13] and the depletion perturbation theory (DPT) and generalized perturbation theory (VARI3D) codes developed by ANL researchers. [14,15] The calculations for this work have, however, been done using the ERANOS code systems. This is because the VHTR sensitivity and uncertainty effort was leveraged with similar activities under the AFCI program that also used this tool.

It is noted that while these codes are robust and collectively allow the work to be done, they have limitations either in methodology or geometry options. For example, DPT contains models only for diffusion theory and R-Z geometry calculations. A major limitation of the ERANOS approach is that it employs an approximation for the coupling of the flux and nuclide fields inherent in the problem. Finally, the VARI3D capability is limited to finite-difference diffusion theory static problems (could be used approximately for depletion problems). Furthermore, the calculations for this study have focused only on the actinides and light elements contained in the core material compositions. The impact of the fission products have been accounted for implicitly in the target accuracy study (by using more stringent target accuracy for the burnup reactivity swing). However, in general, the non-representation of the fission products implies that the uncertainty values presented in Section 4 are lower than the expected valued. This discussion indicates the need for additional development of these tools as part of the VHTR methods development activities.

As aforementioned, all the sensitivity coefficient calculations have been performed with the ERANOS code system [13], which allows calculating the homogeneous solution of the Boltzmann equation, generalized importance functions, and performing perturbation and uncertainty analysis. The discrete ordinate module BISTRO [16] has been used to perform flux

and generalized importance function calculations. In the current study, the diffusion theory option of the code has been employed.

Decay heat calculations have been performed with the ORIGEN code [17]. Sensitivity coefficients for time-dependent quantities (Eq.18) have been obtained using the NUTS code. [21]

Because the ERANOS code does not contain models for representing the fuel particle heterogeneity effect, cross sections were first generated using a lattice code that provides the necessary representation. In this work, cross-section data have been processed to the selected multigroup model (15 groups, see Table 1) using the WIMS8 lattice code [18] to generate data for the different core regions. A 172-group library, based on the JEF2.2 data, was used for the WIMS8 spectrum and transport calculations. Since this is the first time ERANOS has been utilized in conjunction with the WIMS8 cross-section generation path, verification calculations were performed to ascertain that the WIMS8 and ERANOS codes (using the multigroup cross sections generated by WIMS8) give the same result.

Table 1. Multigroup Structure for Block-Type VHTR Study.

Group	Energy
1	19.6 MeV
2	6.07 MeV
3	2.23 MeV
4	1.35 MeV
5	498 KeV
6	183 KeV
7	67.4 KeV
8	24.8 KeV
9	9.12 KeV
10	2.03 KeV
11	454 eV
12	22.6 eV
13	4.00 eV
14	0.54 eV
15	0.10 eV

3.0 REFERENCE VHTR SYSTEM FOR THE UNCERTAINTY ANALYSIS

3.1 The Reference System

The methodology outlined in Section 2 has been applied to the analysis of the gas-cooled block-type VHTR design developed as part of a Next Generation Nuclear Plant (NGNP) fuel cycle study funded by the USDOE in FY 2004; the effort supported the fuels specification/testing program. [19] The goal of that study was to determine the feasibility of a once-through block-type VHTR to meet the cycle length and burnup goals within acceptable design ranges for the CFP kernel diameter, particle packing fraction in the matrix, and the fuel enrichment. A block-type VHTR based on the GT-MHR annular core design, [1,20] was used in the study. A two-batch core design was found to meet the stringent requirements defined in the study: cycle length greater than 1.5 years, burnup greater than 100 GWd/t, fuel kernel diameter of less than 425 μm , fuel enrichment less than 20% (constrained to 15% in the study), and fuel packing fraction constrained to 30%. The design has been used as the starting point for the current study.

Because of the limitation in the geometry option of the BISTRO code, noted in Section 2.3, a cylindrical R-Z model of the core has been defined. The FY 2004 study employed a whole-core Hex-Z model for core characterization and analysis. It is noted that the R-Z model is however quite sufficient for this type of study on nuclear data accuracy. Figure 1 shows the geometrical layout of the cylindrical R-Z model (dimensions are in cm). Table 2 contains the regional number densities used in the R-Z model. Since the R-Z model will primarily be utilized for cross section sensitivity and uncertainty studies, the 1-batch core loading case was selected. This allows information about the clean and depleted cores to be inferred systematically without ambiguity about the description of the core depletion zones in the R-Z model; it however resulted in a burnup of about 90 GWd/t. The required fuel uranium enrichment is 14%.

For assurance that the R-Z model is representative of the whole-core Hex-Z model used in FY 2004, a comparison of the two models was first done for the initial core. That comparison used the diffusion theory models of the DIF3D code for consistency. Results from this study are summarized in Table 3. At the time the study was done, additional cross-sections for the 6- and

23-group calculations were generated using the WIMS8 lattice code. The core k_{eff} (multiplication factor) is about 1.37 because burnable poisons have not been used in the core model. The results show that the R-Z model is representative for the k_{eff} and that the group effect (difference between the 172- and 6-group results) is about 630 pcm (Δk) for the initial core. This effect is lower when the 23-group case is compared to the 172-group case.

Figure 1. R-Z Model for Block-type VHTR Core.

Table 2. Prismatic Block-Type VHTR Regional Compositions [10^{24} at/cm³].

Inner, Central and Outer Fuel		
Isotope	BOL	EOL
U235	2.49199E-05	1.08813E-05
U238	1.51142E-04	1.41382E-04
Np237	-	1.47874E-07
Pu238	-	4.85573E-08
Pu239	-	2.51844E-06
Pu240	-	7.56531E-07
Pu241	-	8.90984E-07
Pu242	-	2.35550E-07
Am241	-	2.06991E-08
Am242	-	4.24398E-10
Am243	-	3.37888E-08
Cm242	-	7.58440E-09
Cm243	-	1.34310E-10
Cm244	-	8.06654E-09
Cm245	-	4.29784E-10
C	6.39955E-02	6.39955E-02
O	2.64092E-04	2.64092E-04
Si	5.22833E-04	5.22833E-04
Reflector		
C	8.73340E-02	

Table 3. Comparison of Eigenvalues from R-Z and Hex-Z Models of VHTR Core.

Core Models	Number of groups	Flux-Solver	k _{eff}
Hex-Z	6	DIF3D – Nodal	1.36904
R-Z	6	DIF3D – FDM	1.36872
R-Z	23	DIF3D – FDM	1.36585
R-Z	172	DIF3D – FDM	1.36244

3.2 Main Parameters of the Reference System

The main integral parameters of the reference system considered in this study are presented in Table 4. The beginning of life (BOL) and end of life (EOL) values for some of the parameters are provided. They were obtained with the calculation route indicated in Sections 2.3 and 3.1, using the 15-energy-group structure.

The beginning of life k_{eff} (1.378) is much greater than unity in Table 1 because the burnable poison used for achieving a critical core has not been represented in the model. The current model was purposefully developed to provide indication of the burnup reactivity swing in the system. The higher k_{eff} (compared to value in Section 3.1) is primarily due to the different spatial differentiation schemes used in the DIF3D and BISTRO codes (point versus “box” finite difference approaches). Another cause of the difference is the use of the 15-group data in the BISTRO calculation.

Table 4. Preliminary Uncertainties in VHTR Core Physics Integral Parameters.

Parameter	State ^a	Value
Core Integral Parameters		
k_{eff}	BOL	1.378
	EOL	1.016
Peak Power	BOL	2.0
	EOL	2.2
Temperature Reactivity Defect, pcm	BOL	2095
	EOL	3416
Burnup Reactivity Swing, pcm		-25829
Fuel Cycle Parameters		
Decay Heat, ^b W		16.7
Dose, ^c Sv		0.030
Neutron Source, ^d n/s		2.23×10^8

^a BOL and EOL are beginning of life and end of life, respectively.

^b Decay heat is for 100 years after discharge.

^c Dose is for 100,000 years after discharge.

^d Neutron source is at 2 years after discharge.

^e Values of the fuel cycle parameters are in arbitrary mass unit.

The peak power is defined as the point maximum power value normalized to the total power. In this model, it occurs at the core mid-plane axially and at the inner reflector-core interface radially, for both the BOL and EOL cases. The BOL and EOL values are nearly the same because a single burnup zone is used in the study. In the real core, the initial peak power location would likely be in the upper part of the core (cold region), if a uniform enrichment is used; compensation with different upper and lower reflector thicknesses would somewhat lower the location of the peak. Also the peak would be expected to decrease with burnup.

The temperature reactivity defect corresponds to the reactivity induced by changing the core state from a “hot zero power” condition (all core material temperatures at 773°K, the coolant inlet temperature) to a “hot full power” condition (fuel and moderator temperatures elevated to 1373°K and 1200°K, respectively).

Finally, the burnup reactivity swing represents the impact of core fuel depletion on the core reactivity. It accounts for the loss of fissile material, the accumulation and destruction of fertile materials, and the buildup of the fission products.

The other quantities in Table 4 are the decay heat of spent fuel in a repository (at 100 years after disposal), the neutron radiation source at fuel discharge, and the radiotoxicity at 100,000 years after disposal. These parameters are included in order to provide indications of the uncertainty in their values in the VHTR fuel cycle.

4.0 UNCERTAINTY ANALYSIS

Results of the uncertainty analysis are presented in this section. First, in Section 4.1, more detailed information on the covariance data used in the study is presented. Subsequently, the uncertainty results are presented in Section 4.2 for the core and fuel cycle parameters.

4.1 Covariance Matrix for Multigroup Data.

Covariance data are still scarce in all major nuclear data files, in particular for minor actinides, which are relatively important in our study because of the high burnup of the VHTR fuel (compared to PWRs). For the “home made” ANL covariance Matrix [4] used in this work, we started by updating the covariance matrix used in the ADS study, [2] taking into consideration the results of clean integral experiment analysis, in particular irradiated sample/fuel analysis (for valuable information on capture and some (n,2n) cross-sections) and fission rate measurements in critical assemblies [3].

The uncertainty values, have been given by “energy band”, consistent with multigroup energy structures used for deterministic calculations for the thermal and fast reactors considered in this study and the AFCI study. Fifteen-energy groups have been selected between 20 MeV and thermal energy. Two extra groups were added between 150 MeV and 20 MeV for ADS applications (not used in this study). The uncertainty values are given only for neutron cross-section data of actinides and structural materials. Fission products related uncertainties have not been considered in this study.

In the compilation of the covariance matrix we considered the work done for major actinides under the auspices of the JEF project, in order to produce “adjusted” cross-sections. For minor actinides, we used the analysis and recommendations of a working group of the WPEC updated, for selected isotopes and reactions, on the basis of integral experiment analysis. For most structural materials, we have often relied on the graphical intercomparison of different data files.

The covariance matrix diagonal values (variances) have been estimated on the performance of the most recent JEF files in the analysis of a large set of integral experiments in

different spectra. However, it was observed that the performance obtained using ENDF/B files was not substantially different. Therefore, the covariance matrix can be applied to both files.

At first, only “diagonal” values of the full covariance matrices were used. Their use implies neglecting all types of correlations (in energy, between different isotopes, among reactions, etc.) and results in the underestimation of the uncertainties.

In a second step, we introduced partial energy correlations. Our first guess has been to use the same correlations for all isotopes and reactions, under the form of full energy correlation in 5 energy bands. The idea was to single out:

1. the region above the threshold of fertile isotope fission cross-sections, and of many inelastic cross-sections, up to 20 MeV;
2. the slowing down region to the upper unresolved resonance energy limit;
3. the unresolved resonance energy region;
4. the resolved resonance region; and
5. the thermal range.

The correlations used are shown in Figure 2.

As indicated in the Introduction, available covariance data at the NEA DataBank [5] have been compared to the ANL Covariance Data [3]. The comparison cannot be complete, but limited to selected isotopes and reactions, since some data used in this analysis are missing in the available evaluated covariance data. The data selected (NEA-K Covariance Matrix) are shown in Table 5. Ideally, the covariance data should be taken from a single evaluation, i.e., the one used in the calculations, in order to assure consistency among the nuclear data used. At present no evaluation is complete enough to cover all the materials needed in this study and the NEA-K covariance matrices are therefore a selection (more or less consistent) of the data available in the ENDF/B-V, -VI, IRDF-2002, JENDL-3.3 and JEFF-3 evaluations.

Some comparisons of the ANL and NEA-K covariance matrices for selected isotopes and reactions are given in Figures 3 through 6.

Figure 2. Energy correlations for the ANL Correlation Matrix (PEC).

Table 5. NEA-K Covariance Matrix.

<i>Nuclide</i>	<i>Reaction Type</i>	<i>Library</i>
U235	(nu, fission, inelastic, elastic, capture, nxn)	JENDL3.3
U238	(nu, inelastic, elastic, nxn)	JENDL3.3
	(fission, capture)	IRDF-2002
Np237	(fission)	IRDF-2002
Pu239	(nu, inelastic, elastic, capture, nxn)	JENDL3.3
	(fission)	IRDF-2002
Pu240	(inelastic, elastic, fission, capture, nxn)	JENDL3.3
Pu241	(nu, inelastic, elastic, fission, capture, nxn)	JENDL3.3
Am241	(fission)	IRDF-2002
C	(elastic, capture)	ENDF/B-V
H	(elastic, capture)	ENDF/B-V
O	(inelastic, elastic)	ENDF/B-V
Cr52	(inelastic, elastic, capture, nxn)	ENDF/B-VI
Fe56	(inelastic, elastic, capture, nxn)	ENDF/B-VI
Na23	(inelastic, elastic, capture, nxn)	ENDF/B-VI
Pb206	(inelastic, elastic, capture, nxn)	ENDF/B-VI
Pb207	(inelastic, elastic, capture, nxn)	ENDF/B-VI
Pb208	(inelastic, elastic, capture, nxn)	ENDF/B-VI
Si	(inelastic, elastic)	ENDF/B-VI
Zr90	(inelastic, capture, nxn)	JENDL3.3
B10	(capture)	IRDF-2002
Ni58	(inelastic, elastic, capture, nxn)	JEFF3

Figure 3. Comparison of NEA-K and ANL Covariance Data for Pu-239(n,f).

Figure 4. Comparison of NEA-K and ANL Covariance Data for Pu-240(n, γ).

Figure 5. Comparison of NEA-K and ANL Covariance Data Pu-241(n,f).

Figure 6. Comparison of NEA-K and ANL Covariance Data Am-241(n,f).

4.2 Uncertainty in Core and Fuel Cycle Parameters

The results of the uncertainty analysis for the core and fuel cycle parameters are summarized in Table 6. Data obtained with the hypothesis of no correlation in energy, among reactions or isotopes (Total no Corr.) are presented in the first column under Uncertainty. Those obtained using the partial energy correlation model (Total PEC) are given in the second column.

Table 6. Uncertainties in VHTR Core Physics Integral Parameters.

Parameter	State	Uncertainty (%)	
		Total No Corr.	Total PEC
k_{eff}	BOL	± 0.41	± 0.58
	EOL	0.94	1.07
Peak Power	BOL	0.87	1.0
	EOL	0.93	1.1
Temperature Reactivity Defect	BOL	3.6	3.3
	EOL	5.8	5.5
Burnup Reactivity Swing (pcm)		1482	1574
Decay Heat, 100 years in Repository		2.5	3.1
Dose, 100,000 years in Repository		1.9	2.6
Neutron Source, 2-year Cooling		12.2	14.3

Generally, the uncertainty values on the integral parameters increase from the BOL to EOL states. This is due to the buildup of plutonium isotopes and minor actinides (Pu, Am, Np, and Cm) whose cross-sections have large uncertainties. Notably, the introduction of partial energy correlation leads to an increase in the uncertainty, except for the case of the temperature reactivity defect. This is due to opposite sign sensitivity coefficients in energy groups that are completely correlated.

Overall, the uncertainty values seem quite acceptable, except for the k_{eff} at EOL, which is about 1%, and the burnup reactivity swing (about 1500 pcm; equivalent to about 6% of nominal value). The uncertainties in the fuel cycle quantities appear to be acceptable, although that on the neutron source appears to be relatively large. In Table 7, the major contributors of the uncertainty in k_{eff} at EOL are presented. (A more complete list of the contributors, for all parameters, is given in Appendix A.) The major sources of uncertainties arise from fissions in U-

235, Pu-239, and Pu-241, and captures in U-238, Pu-239, Pu-240, and Pu-241. The capture in Pu-240 is the major contributor, arising from the low-energy lying Pu-240 capture resonance at ~1 eV. The energy distribution of the uncertainty is also shown in Table 7. In general, the largest values correspond to the major resonances of the actinides. (The total value reported at the bottom of the table for each reaction type is derived by taking the square root of the sum of the squares of each energy group value.)

Table 7. Uncertainty in VHTR EOL k_{eff} Arising from Nuclear Data Uncertainties (pcm).*

Group	Upper Energy	Leading Contributors					
		U-235 σ_f	U-238 σ_c	Pu-239 σ_c	Pu-239 σ_f	Pu-240 σ_c	Pu-241 σ_f
1	19.6 MeV	± 0	± 0	± 0	± 0	± 0	± 0
2	6.07 MeV	1	0	0	0	0	1
3	2.23 MeV	1	0	0	1	0	0
4	1.35 MeV	2	2	0	1	0	1
5	498 KeV	2	2	0	1	0	1
6	183 KeV	2	2	0	1	0	1
7	67.4 KeV	2	4	0	1	0	1
8	24.8 KeV	3	7	0	1	0	1
9	9.12 KeV	8	10	1	1	0	2
10	2.03 KeV	10	24	1	2	1	4
11	454 eV	51	185	11	13	11	26
12	22.6 eV	34	337	11	15	2	56
13	4.00 eV	38	8	5	12	620	6
14	0.54 eV	80	9	360	346	24	88
15	0.10 eV	46	5	21	23	6	23
Total		118	385	361	347	621	110

*Data presented are for the leading contributors. The overall or total uncertainty is obtained by taking the square root of the sum of the square of all contributors.

In Table 8 are summarized the uncertainty breakdown by major contributing nuclides for the k_{eff} , temperature reactivity defect, and the burnup reactivity swing, and the neutron source. Highlighted (yellow shading) are the largest components to the total uncertainties. Again, Pu-240 gives the greatest contribution to the EOL integral parameters. Am-243 is the leading contributor

to the uncertainty in the neutron source. This is due to the capture cross section component that produces Cm-244, a major contributor to the total neutron source.

Table 8. VHTR Uncertainties (%) PEC – Breakdown by Nuclides (Major Contributions).

Isotope	k_{eff}		Temperature Reactivity Coefficient		Burnup Reactivity Swing (pcm)	Neutron Source
	BOL	EOL	BOL	EOL		
U-235	±0.36	±0.25	±1.3	±0.6	±171	±0.02
U-238	0.43	0.55	2.7	2.2	150	2.61
Pu-239	0.00	0.57	0.0	3.0	624	2.26
Pu-240	0.00	0.63	0.0	3.9	1313	2.60
Pu-241	0.00	0.17	0.0	0.3	222	2.33
Pu-242	0.00	0.02	0.0	0.1	36	3.96
Am-243	0.00	0.02	0.0	0.1	27	12.60
Cm-244	0.00	0.00	0.0	0.0	3	2.30

Table 9 is a comparison of the uncertainties resulting from the use of the ANL and NEA-K covariance matrices, arising from the leading contributors. Comparable results are observed for Pu-240 capture. In the cases of U-238, Pu-239, and Pu-241, the NEA-K covariance matrix data give lower values (factor ~2 to 3). Inferences based on these results should be made with caution, since the NEA-K data does not represent a fully consistent and complete set of data. However, the results seem to indicate that the present covariance evaluations would tend to show significantly lower uncertainties on the integral parameters, which in principle indicate lower needs for data improvement.

Table 9. VHTR PEC/NEA-K Uncertainties (%) Comparison.

Nuclide	Data Source	BOL k_{eff}	EOL k_{eff}	Burnup Reactivity Swing (pcm)
U238	PEC	± 0.43	± 0.55	± 150
	NEA-K	0.19	0.24	60
Pu239	PEC	0.00	0.57	624
	NEA-K	0.00	0.18	211
Pu240	PEC	0.00	0.63	1313
	NEA-K	0.00	0.51	1017
Pu241	PEC	0.00	0.17	222
	NEA-K	0.00	0.05	73

4.3 Uncertainty in Final Nuclide Number Densities and the Contributors

The uncertainties in the final nuclide number densities have been calculated. The sources of these uncertainties have also been determined. The methodology contained in Section 2.2.3 was used for calculating the sensitivity coefficients. The resulting sensitivity coefficients were folded with the covariance matrices to give values of the nuclide density uncertainties resulting from uncertainty in nuclear data.

Table 10 gives the uncertainties for the major nuclide density variations ΔN during the cycle, using partial energy correlation (PEC) data. The first row identifies the nuclides of interest. Contributions to their nuclide density uncertainties arising from the reactions listed in the first two columns are also provided. Small uncertainties on nuclide density variations are observed, the largest values being related to the ΔN of Np-237 (due to U-236 capture), Pu-238 (also due to the U-236 capture), Pu-240 (due to the Pu-240 capture), Am-241 (due mostly to Am-241 capture), and Am-243 (due to Pu-242 capture and Am-243 capture).

Table 10. VHTR: $\delta(\Delta N)/\Delta N$ PEC Uncertainties (%).

Reaction Type		Np-237	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Am-241	Am-243
U-235	capture	±1.45	±1.31	±0.00	±0.00				
U-236	capture	6.97	6.31	0.02	0.01				
U-238	capture	0.00	0.28	2.52	2.55	2.58	2.62	2.60	2.63
Np-237	capture	1.40	2.75	0.01	0.01				
Pu-238	capture		1.26	0.02	0.01				
Pu-239	capture		0.23	1.02	1.83	1.91	2.05	1.98	2.13
	fission		0.07	1.07	1.02	0.94	0.80	0.86	0.72
Pu-240	capture		0.26		5.73	0.94	1.76	1.38	2.26
Pu-241	capture		0.04			0.56	2.34	0.50	2.39
	fission		0.08			1.10	0.88	0.98	0.76
Pu-242	capture						1.02		4.12
Am-241	capture		0.43				0.03	5.50	0.06
Am-243	capture								4.62
PEC(Total)		7.26	7.15	2.92	6.61	3.69	4.70	6.69	7.85

5.0 AN ASSESSMENT OF CROSS-SECTION TARGET ACCURACY

In the previous section, we presented an extensive uncertainty analysis for a number of relevant parameters of the block-type VHTR. The effect of the cross-section uncertainties we have adopted is relatively large for some parameters. These uncertainties can be tolerable in very preliminary design or scenario studies. The obtained uncertainty values do not indicate problems with using the existing nuclear data files for performing meaningful parametric and sensitivity studies and for evaluating the impact of technological choices. However, as soon as more precise information is needed, the margins to be taken on the nominal values (to provide acceptable conservatism in design or scenario studies) would introduce too many penalties. If, according to Section 2.1, one introduces target accuracies on the integral parameters, one can get significant quantitative indications of the cross-sections accuracies needed.

Regarding target accuracies on integral parameters, it was decided not to account for correlations between data. We then defined a tentative set of target accuracies for the multiplication factor k_{eff} ($\pm 0.5\%$), the power peak ($\pm 3\%$), the temperature reactivity defect ($\pm 10\%$), the reactivity loss during irradiation (500 pcm), and the nuclide concentrations at end of irradiation ($\pm 5\%$). These values are of course rather arbitrary, but they are consistent with standard requirements for reactor design in early phases of development. Considering Table 6, it can be observed that the BOL k_{eff} , BOL and EOL peak power, and the temperature reactivity defect meet the accuracy requirements on the integral parameters.

We have used the formulation discussed in Section 2.1, with the sensitivity coefficients obtained previously and assuming that the “cost” parameters λ are set equal to 1, to perform a target accuracy study in order to determine the required uncertainty in cross-sections to satisfy the accuracy requirement. To avoid the introduction of meaningless parameters, we have chosen as unknown “ d ” parameters (i.e., as cross-sections for which target accuracies are required), only those which globally account for 95% of the overall uncertainty for each integral parameter. Tables of contributors to the uncertainties, presented in Appendix B, have been used for this purpose.

The selected integral parameters used in the analysis are those that do not meet the target accuracies. Based on the results in Tables 6 and 10 and the target accuracies presented above,

these parameters are the EOL k_{eff} , the burnup reactivity swing, and the nuclide densities for Np-237, Pu-238, Pu-240, Am-241 and Am-243.

The cross sections uncertainties required for satisfying the target accuracies are then calculated by a minimization process that satisfy the nonlinear constraints with bounded parameters. Several optimization codes (including OPTIMA, KNITRO, SNOPT, etc.) were tested for this minimization process in order to verify that consistent answers were obtained and not local minima. At the end, the SNOPT code was selected because of the ease in using the FORTRAN interface. The product of this process is the required uncertainties that could allow the target accuracies to be met.

Table 11 is a summary of the current uncertainties and those required for the selected cross sections (including energy range) to meet the target accuracies for the k_{eff} , the burnup reactivity swing, and the nuclide densities. The *current* uncertainty is the value in the original covariance dataset (only diagonal elements) used for initially assessing the uncertainties in core parameters. The *required* uncertainty is that needed to meet the target accuracies in the core parameters. The unchanged cross section uncertainty data are not provided in this table.

From the results of Table 11, it can be seen that very stringent (<2%) requirements are needed for some cross sections (e.g., capture in Pu-240, Pu-239, and U-238, and fission in Pu-239) that might be very difficult to achieve by measurement and/or evaluation. The other major requirements are on the carbon (C) scattering cross section in the high-energy range and the Am-243 capture cross section (4 eV – 0.54 eV) for which the uncertainties need to be reduced by a factor of 2 to 3.

Tables 12 and 13 show the uncertainties in the integral parameters and the final nuclide densities using current values (initial) and those obtained using the required uncertainties (final). The results are very encouraging, since all the pertinent integral parameters uncertainties can be brought within the target accuracy; for some of the parameters not included in the optimization process, uncertainty values larger than the targeted accuracy values were found. Obtaining these results, however, implies the need for more stringent requirements on some of the cross sections (as discussed above). It is unlikely these required neutron cross-section uncertainties (e.g., ~1-

2% for Pu-240, Pu-239 and U-238 reactions) can be achieved with differential measurements or sophisticated evaluations. Two potential items for future activities (beyond 2005) could include (1) the use of existing integral experimental data to improve the cross section uncertainties, and (2) the use of “cost” factors that are better informed by input from experimentalist, in the optimization process. The cost factors give a relative figure of merit of the difficulty of improving the knowledge of a cross section (e.g., its associated uncertainty).

Because of the large value of the burnup reactivity swing and the imposed target accuracy, it was speculated that this might be the reason for the stringent cross section uncertainty requirements. Consequently, an additional calculation was performed using target accuracy of 1000 pcm (instead of 500 pcm) in the optimization calculation. The results showed that the required accuracies on the cross sections were not significantly different (increase less than 1%). This is because most of these stringent requirements arise from the uncertainties on other parameters such as the k_{eff} .

Table 11. Current and Required Cross Section Uncertainties (%) for Target Accuracies.

Isotope	Cross Section	Energy Range	Uncertainty	
			Initial	Required
U236	σ_{capt}	22.6 eV-4.00 eV	8	7.1
U238	σ_{capt}	454 eV-22.6 eV	3	1.9
		22.6 eV-4.00 eV	3	1.4
Pu239	σ_{capt}	0.54 eV-0.10 eV	3	1.1
	σ_{fiss}	0.54 eV-0.10 eV	2	1
Pu240	σ_{capt}	454 eV-22.6 eV	10	9.6
		4.00 eV-0.54 eV	7	1.1
Pu241	σ_{fiss}	454 eV-22.6 eV	10	8.1
		22.6 eV-4.00 eV	10	5.5
		0.54 eV-0.10 eV	2	1.9
Am241	σ_{capt}	0.54 eV-0.10 eV	10	9.4
Am243	σ_{capt}	4.00 eV-0.54 eV	20	12.4
C	σ_{scatt}	6.07 MeV-2.23 MeV	35	12.3

Table 12. Initial and Final Uncertainties in VHTR Integral Parameters.

		Uncertainty (%)	
Parameter		Current	Final
k_{eff}	BOL	± 0.41	± 0.32
	EOL	0.94	0.43
Peak Power	BOL	0.87	0.52
	EOL	0.93	0.52
Temperature Reactivity Defect	BOL	3.6	3.2
	EOL	5.8	2.8
Burnup Reactivity Swing (pcm)		1482	500
Decay Heat, 100 years in Repository		2.5	1.4
Dose, 100,000 years in Repository		1.9	1.2
Neutron Source, 2-year Cooling		12.2	8.3

Table 13. Current and Final Uncertainties (%) in Nuclide Densities.*

Parameter	Current Uncertainty	Final Uncertainty
U-235	0.7	0.7
U-238	0.1	0.1
Np-237	5.5	5.0
Pu-238	5.4	5.0
Pu-239	2.2	1.1
Pu-240	6.2	1.5
Pu-241	3.0	1.6
Pu-242	4.0	2.6
Am-241	5.9	4.9
Am-242m	6.9	6.2
Am-243	6.8	5.1
Cm-242	4.5	3.1
Cm-243	21.5	21.2
Cm-244	13.1	9.0
Cm-245	34.5	33.0

*The uncertainties presented are for the no correlation case. The PEC values would be higher.

6.0 CONCLUSIONS

Sensitivity and uncertainty analysis have been performed for the block-type VHTR design. The uncertainty results obtained for integral parameters show that the assumed covariance matrices for the nuclide cross sections can result in significant uncertainties in the values of the integral parameters. However, the results do not indicate that the current cross section data (in nuclear data files) are insufficient for evaluating the feasibility of the VHTR concept or for performing pre-conceptual design studies. This conclusion is obviously dependent on the assumed cross-section uncertainty data. Better and more consistent covariance data would reinforce this conclusion. However, the timescale for the production of a full set of covariance matrices should be 2-3 years, in order for the data to have an impact on the feasibility assessment of the VHTR concepts and to provide the basis for potential, high priority new cross-section measurements or nuclear data oriented integral experiments. Furthermore, the data should be easy to process in a user oriented format (i.e. by energy group), particularly including covariance data for the resonances of some major uranium and plutonium isotopes.

In order to propose a credible program for new cross-section measurements, one should show the impact of the existing cross section uncertainties on the relevant parameters taking into account the target accuracies on the design parameters. For this purpose, we have considered design target accuracies which could be relevant in successive design phases and have evaluated nuclear data improvement requirements. The resulting requirements indicated that a careful analysis is needed in order to define the most appropriate and effective strategy for data uncertainty reduction.

For uncertainty reduction purposes, it is noted that integral experiments and statistical data adjustments are a powerful tool to overcome most difficulties, since they provide a global validation of data, and allow developing improved evaluations for selected isotopes, reaction types and energy domains.

Although statistical data adjustments can also be used for design calculations, a number of conditions should be satisfied:

- The integral experiments used in the adjustment procedure should be “clean”, in the sense that the associated experimental uncertainties are small and well understood;
- The calculation results for the integral experiments should not be affected by any significant modeling uncertainty (e.g. geometrical description, number of energy group used in the analysis, etc.), in order to avoid the introduction of systematic errors;
- The covariance data should be reliable, complete, and consistent;
- Data adjustments should as much as possible relate to the physics parameters which describe the cross-sections, to make adjustments independent from the energy collapsing procedures.

In summary, better uncertainty data will play an essential role both in the assessment of potential new data needs with reduced uncertainties and in design oriented statistical data adjustments.

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Appendix A. VHTR Uncertainty Breakdown

Table A.1. Uncertainty Breakdown by Isotope.

ISOTOPE	K _{eff}		Peak		Doppler		Burnup		
	BOL	EOL	BOL	EOL	BOL	EOL	XS	Delta N	Total
U235	0.26	0.17	0.22	0.09	2.70	1.08	0.42	0.07	0.49
U238	0.30	0.39	0.02	0.02	1.91	1.58	0.34	0.07	0.41
Np237	0.00	0.01	0.00	0.00	0.00	0.08	0.05	0.07	0.12
Pu238	0.00	0.00	0.00	0.00	0.00	0.05	0.02	0.01	0.03
Pu239	0.00	0.53	0.00	0.22	0.00	3.52	2.06	0.17	2.23
Pu240	0.00	0.62	0.00	0.10	0.00	3.88	2.40	2.46	4.87
Pu241	0.00	0.13	0.00	0.07	0.00	0.67	0.51	0.17	0.68
Pu242	0.00	0.02	0.00	0.00	0.00	0.07	0.08	0.05	0.13
Am241	0.00	0.01	0.00	0.00	0.00	0.10	0.06	0.05	0.10
Am242m	0.00	0.00	0.00	0.00	0.00	0.01	0.01	0.00	0.01
Am243	0.00	0.02	0.00	0.00	0.00	0.08	0.07	0.02	0.09
Cm242	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Cm243	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Cm244	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.01
Cm245	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.01
O	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Si	0.05	0.05	0.02	0.02	0.55	0.39	0.01	0.00	0.01
C	0.09	0.11	0.84	0.89	1.35	1.15	0.17	0.00	0.17
Total	0.41	0.94	0.87	0.93	3.62	5.76	3.26	2.48	5.74

Table A.2. Uncertainty PEC - Breakdown by Isotope.

ISOTOPE	K _{eff}		Peak		Doppler		Burnup		
	BOL	EOL	BOL	EOL	BOL	EOL	XS	Delta N	Total
U235	0.36	0.25	0.32	0.14	1.32	0.64	0.56	0.10	0.66
U238	0.43	0.55	0.03	0.03	2.70	2.22	0.47	0.11	0.58
Np237	0.00	0.02	0.00	0.00	0.00	0.06	0.06	0.09	0.16
Pu238	0.00	0.01	0.00	0.00	0.00	0.05	0.02	0.02	0.04
Pu239	0.00	0.57	0.00	0.23	0.00	3.00	2.20	0.22	2.42
Pu240	0.00	0.63	0.00	0.10	0.00	3.90	2.46	2.63	5.08
Pu241	0.00	0.17	0.00	0.09	0.00	0.25	0.65	0.21	0.86
Pu242	0.00	0.02	0.00	0.00	0.00	0.07	0.08	0.06	0.14
Am241	0.00	0.02	0.00	0.00	0.00	0.08	0.06	0.05	0.11
Am242m	0.00	0.00	0.00	0.00	0.00	0.01	0.01	0.00	0.01
Am243	0.00	0.02	0.00	0.00	0.00	0.08	0.08	0.03	0.10
Cm242	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Cm243	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Cm244	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.01
Cm245	0.00	0.00	0.00	0.00	0.00	0.01	0.01	0.00	0.01
O	0.00	0.00	0.00	0.00	0.00	0.00	0.00		0.00
Si	0.07	0.06	0.03	0.03	0.10	0.27	0.01		0.01
C	0.12	0.13	0.99	1.04	1.26	1.09	0.15		0.15
Total	0.58	1.07	1.04	1.09	3.26	5.55	3.44	2.65	6.10

Table A.3. Uncertainty NEA-K - Breakdown by Isotope.

ISOTOPE	K _{eff}		Peak		Doppler		Burnup		
	BOL	EOL	BOL	EOL	BOL	EOL	XS	Delta N	Total
U235	0.24	0.15	0.19	0.07	1.18	0.45	0.41	0.06	0.47
U238	0.19	0.24	0.01	0.01	1.11	0.84	0.18	0.05	0.23
Np237	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.01
Pu238	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Pu239	0.00	0.18	0.00	0.15	0.00	1.18	0.71	0.11	0.82
Pu240	0.00	0.51	0.00	0.10	0.00	2.65	1.99	1.95	3.94
Pu241	0.00	0.05	0.00	0.03	0.00	0.16	0.18	0.11	0.28
Pu242	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.03	0.03
Am241	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.02	0.02
Am242m	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Am243	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.01
Cm242	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Cm243	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Cm244	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Cm245	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
O	0.00	0.00	0.00	0.00	0.00	0.00	0.00		0.00
Si	0.00	0.00	0.00	0.00	0.00	0.00	0.00		0.00
C	0.20	0.25	0.21	0.20	0.17	0.65	0.20		0.20
Total	0.37	0.66	0.28	0.28	1.63	3.12	2.17	1.96	4.13

Table A.4. Uncertainty Breakdown by Isotope.

	Decay Heat	Dose	Neutron Source
U234	0.00	0.00	0.00
U235	0.14	0.13	0.01
U236	0.82	0.77	0.06
U238	1.51	1.46	1.79
Np237	0.34	0.26	0.02
Pu238	0.14	0.11	0.01
Pu239	1.64	0.80	2.14
Pu240	0.57	0.36	2.55
Pu241	0.26	0.26	1.96
Pu242	0.06	0.02	3.60
Am241	0.06	0.10	0.21
Am242m	0.01	0.00	0.18
Am243	0.12	0.11	10.64
Cm242	0.01	0.00	0.00
Cm243	0.00	0.00	0.00
Cm244	0.03	0.00	2.05
Cm245	0.00	0.00	0.06
NC	2.49	1.92	12.19
PEC	3.12	2.60	14.27
NEA-K	1.44	1.12	3.00

Table A.5. VHTR Isotope Contribution.

Isotope	Decay Heat (%)	Dose (%)	Neutron Source (%)
Pb210	0.00	8.80	0.00
Ra226	0.00	3.38	0.00
Ac227	0.00	0.74	0.00
Th229	0.00	6.17	0.00
Th230	0.00	2.50	0.00
Pa231	0.00	0.58	0.00
U233	0.00	0.62	0.00
U234	0.01	1.02	0.00
U235	0.00	0.05	0.00
U236	0.00	0.54	0.00
U238	0.00	0.11	0.00
Np237	0.01	3.04	0.00
Pu238	18.77	0.00	1.32
Pu239	3.06	63.00	0.08
Pu240	4.11	0.04	0.38
Pu241	0.02	0.00	0.00
Pu242	0.03	9.42	0.30
Am241	72.45	0.00	0.29
Am243	0.35	0.01	0.01
Cm242	0.09	0.00	5.23
Cm243	0.03	0.00	0.01
Cm244	1.06	0.00	92.14
Cm245	0.01	0.00	0.00
Cm246	0.00	0.00	0.23
Total	100.00	100.00	100.00

Table A.6. Detailed Breakdown of Nuclide Density Uncertainty.

Nominal Values		U235	U238	Np237	Pu238	Pu239	Pu240	Pu241	Pu242	Am241	Am242m	Am243	Cm242	Cm243	Cm244	Cm245
$N_f - N_i$		-1.40E-5	-9.76E-6	1.48E-7	4.86E-8	2.52E-6	7.57E-7	8.91E-7	2.35E-7	2.07E-8	4.24E-10	3.38E-8	7.58E-9	1.34E-10	8.07E-9	4.30E-10
$(N_f - N_i) / N_f$		-1.29	-0.07	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
ΔN Uncert. (%)		U235	U238	Np237	Pu238	Pu239	Pu240	Pu241	Pu242	Am241	Am242m	Am243	Cm242	Cm243	Cm244	Cm245
U235	capture	0.16		0.90	0.82	0.00	0.00									
	fission	0.48		0.19	0.13	0.00	0.00									
	N,2N	0.00		0.00	0.00	0.00	0.00									
U236	capture			5.36	4.86	0.01	0.01									
	fission			0.01	0.01	0.00	0.00									
	N,2N			0.00	0.00	0.00	0.00									
U238	capture		1.77	0.00	0.19	1.73	1.75	1.77	1.79	1.79	1.79	1.80	1.80	1.81	1.81	1.81
	fission		0.03	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	N,2N		0.01	0.21	0.24	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Np237	capture			1.03	2.02	0.01	0.00									
	fission			0.01	0.00	0.00	0.00									
	N,2N			0.00	0.00	0.00	0.00									
Pu238	capture				0.89	0.01	0.01									
	fission				0.09	0.00	0.00									
	N,2N				0.00	0.00	0.00									
Pu239	capture				0.22	0.97	1.74	1.81	1.94	1.88	1.90	2.02	1.95	2.03	2.09	2.13
	fission				0.07	1.00	0.96	0.89	0.75	0.81	0.79	0.67	0.74	0.66	0.60	0.56
	N,2N				0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Pu240	capture				0.26		5.61	0.92	1.72	1.35	1.47	2.22	1.79	2.27	2.66	2.93
	fission				0.00		0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
	N,2N				0.00		0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Pu241	capture				0.03			0.47	1.97	0.42	0.41	2.01	0.37	0.33	2.06	2.08
	fission				0.06			0.88	0.71	0.78	0.76	0.61	0.69	0.60	0.54	0.49
	N,2N				0.00			0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Pu242	capture								0.93			3.75			3.90	3.98
	fission								0.01			0.00			0.00	0.00
	N,2N								0.00			0.00			0.00	0.00
Am241	capture				0.39				0.03	4.97	2.45	0.05	2.89	3.49	0.06	0.06
	fission				0.01				0.00	0.08	0.07	0.00	0.07	0.06	0.00	0.00
	N,2N				0.00				0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
Am242m	capture										4.83	0.18			0.19	0.20
	fission										2.70	0.02			0.02	0.02
	N,2N										0.00	0.00			0.00	0.00
Am243	capture											3.90			11.51	11.93
	fission											0.01			0.00	0.00
	N,2N											0.00			0.00	0.00
Cm242	capture				0.03								0.32	19.91		
	fission				0.00								0.05	0.04		
	N,2N				0.00								0.00	0.00		
Cm243	capture													1.06		
	fission													6.33		
	N,2N													0.00		
Cm244	capture														2.31	31.05
	fission														0.04	0.04
	N,2N														0.00	0.00
Cm245	capture															0.96
	fission															6.64
	N,2N															0.00
No Correlation		0.50	1.77	5.54	5.43	2.22	6.20	3.01	3.97	5.89	6.86	6.81	4.46	21.52	13.14	34.47
PEC		0.73	2.58	7.26	7.15	2.92	6.61	3.69	4.70	6.69	8.71	7.85	5.15	31.11	15.39	39.32
Kodeli		0.44	1.12	0.91	0.86	1.41	4.91	1.88	2.64	2.03	2.09	2.89	2.24	2.51	3.14	3.30

Appendix B. Major Contributors to Uncertainties

Legend to be used for this appendix:

SC: Sensitivity Coefficient (%);

UC: Uncertainty Coefficient (%);

IU: Incremented Uncertainty (%);

CSSC: Sensitivity Coefficient (%) from Cross-Sections (for burnup uncertainty determination);

NSC: Sensitivity Coefficient (%) from ΔN component (for burnup uncertainty determination).

Table B.1. VHTR K_{eff} BOL: Total Uncertainty = 0.41%.

Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU
U238	σ_{capt}	12	-8.66	3	0.26	0.26	U235	σ_{fiss}	14	8.92	1	0.09	0.37	U235	σ_{capt}	14	-5.58	1	0.06	0.38
U235	ν	14	38.78	0.5	0.19	0.32	U235	ν	15	21.58	0.3	0.06	0.37	U235	σ_{capt}	11	-1.01	5	0.05	0.38
U238	σ_{capt}	11	-4.83	3	0.14	0.36	U235	σ_{capt}	12	-1.23	5	0.06	0.38							

Table B.2. VHTR K_{eff} EOL: Total Uncertainty = 0.94%.

Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU
Pu240	σ_{capt}	13	-8.86	7	0.62	0.62	Pu239	σ_{fiss}	14	17.29	2	0.35	0.80	U238	σ_{capt}	11	-6.17	3	0.18	0.88
Pu239	σ_{capt}	14	-12.00	3	0.36	0.72	U238	σ_{capt}	12	-11.23	3	0.34	0.86	Pu239	ν	14	35.93	0.5	0.18	0.90

Table B.3. VHTR Peak Power BOL: Total Uncertainty = 0.87%.

Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU
C	σ_{inel}	2	-2.13	35	0.75	0.75	U235	σ_{fiss}	14	17.06	1	0.17	0.80	C	σ_{capt}	15	-5.18	2	0.10	0.81
C	σ_{el}	14	23.32	1	0.23	0.78	C	σ_{el}	4	-2.5	5	0.12	0.81	C	σ_{capt}	13	-0.22	20	0.04	0.81
														C	σ_{capt}	14	-2.01	2	0.04	0.82

Table B.4. VHTR Peak Power EOL: Total Uncertainty = 0.93%.

Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU
C	σ_{inel}	2	-2.26	35	0.79	0.79	C	σ_{el}	4	-3.2	5	0.16	0.87	C	σ_{capt}	15	-5.52	2	0.11	0.89
C	σ_{el}	14	26.52	1	0.27	0.84	C	σ_{el}	13	6.15	2	0.12	0.88	C	σ_{capt}	13	-0.25	20	0.05	0.89
Pu239	σ_{fiss}	14	9.12	2	0.18	0.85	Pu239	σ_{capt}	14	3.79	3	0.11	0.88	C	σ_{capt}	14	-1.89	2	0.04	0.89

Table B.5. VHTR Temperature Reactivity Defect BOL: Total Uncertainty = 3.62%.

Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU
U235	ν	14	-	385.44	0.5	1.93	U238	σ_{capt}	11	30.48	3	0.91	3.09	U235	σ_{capt}	14	53.77	1	0.54	3.26
U238	σ_{capt}	12	55.62	3	1.67	2.55	U235	σ_{fiss}	15	68.37	1	0.68	3.16	C	σ_{capt}	13	2.54	20	0.51	3.30
U235	ν	15	360.00	0.3	1.08	2.77	C	σ_{inel}	2	1.56	35	0.55	3.21	U235	σ_{capt}	15	-48.63	1	0.49	3.33
U235	σ_{fiss}	14	-	102.22	1	1.02														

Table B.6. VHTR Temperature Reactivity Defect EOL: Total Uncertainty = 5.76%.

Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU
Pu240	σ_{capt}	13	55.46	7	3.88	3.88	Pu239	σ_{capt}	14	70.36	3	2.11	5.09	Pu239	ν	14	-232.42	0.5	1.16	5.40
Pu239	σ_{fiss}	14	-125.94	2	2.52	4.63	U238	σ_{capt}	12	46.47	3	1.39	5.27	U238	σ_{capt}	11	24.82	3	0.74	5.45

Table B.7. VHTR Burnup Reactivity Swing: Total Uncertainty = 5.74%.

Isotope	Cross Section	Gr.	CSSC	NSC	σ unc	UC	IU	Isotope	Cross Section	Gr.	CSSC	NSC	σ unc	UC	IU
Pu240	σ_{capt}	13	34.31	31.80	7	4.63	4.63	Pu239	σ_{fiss}	14	-66.94	19.55	2	-0.95	5.45
Pu239	σ_{capt}	14	46.47	23.34	3	2.09	5.11								

Table B.8. VHTR N_f .

Uncertainty N_f (U235): 0.65							Uncertainty N_f (Pu241): 3.01							Uncertainty N_f (Am243): 6.81						
Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU	Isotope	Cross Section	Gr.	SC	σ unc	UC	IU
U235	σ_{fiss}	14	-49.78	1.00	0.50	0.50	Pu239	σ_{capt}	14	60.16	3	1.80	1.80	Am243	σ_{capt}	13	19.15	20	3.83	3.83
U235	σ_{fiss}	15	-27.80	1.00	0.28	0.57	U238	σ_{capt}	12	50.28	3	1.51	2.35	Pu242	σ_{capt}	13	74.67	5	3.73	5.35
U235	σ_{fiss}	13	-5.27	3.00	0.16	0.59	Pu240	σ_{capt}	13	13.10	7	0.92	2.52	Pu240	σ_{capt}	13	31.63	7	2.21	5.79
U235	σ_{fiss}	11	-4.58	3.00	0.14	0.61	U238	σ_{capt}	11	30.45	3	0.91	2.68	Pu239	σ_{capt}	14	67.11	3	2.01	6.13
U235	σ_{capt}	12	-2.55	5.00	0.13	0.62	Pu239	σ_{fiss}	14	44.28	2	0.89	2.83	Pu241	σ_{capt}	14	65.30	3	1.96	6.43
Uncertainty N_f (U238): 0.12							Pu241	σ_{fiss}	14	40.49	2	0.81	2.94	U238	σ_{capt}	12	51.26	3	1.54	6.62
U238	σ_{capt}	12	-3.47	3.00	0.10	0.10	Uncertainty N_f (Pu242): 3.97							Uncertainty N_f (Cm242): 4.46						
U238	σ_{capt}	11	-2.10	3.00	0.06	0.12	Pu239	σ_{capt}	14	64.50	3	1.94	1.94	Am241	σ_{capt}	14	27.68	10	2.77	2.77
Uncertainty N_f (Pu237): 5.54							Pu241	σ_{capt}	14	64.03	3	1.92	2.73	Pu239	σ_{capt}	14	64.94	3	1.95	3.38
U236	σ_{capt}	12	63.86	8	5.11	5.11	Pu240	σ_{capt}	13	24.52	7	1.72	3.22	Pu240	σ_{capt}	13	25.61	7	1.79	3.83
U236	σ_{capt}	11	20.07	8	1.61	5.35	U238	σ_{capt}	12	50.96	3	1.53	3.57	U238	σ_{capt}	12	51.05	3	1.53	4.13
Uncertainty N_f (Pu238): 5.43							U238	σ_{capt}	11	30.86	3	0.93	3.68	U238	σ_{capt}	11	30.92	3	0.93	4.23
U236	σ_{capt}	12	57.83	8	4.63	4.63	Pu242	σ_{capt}	13	18.42	5	0.92	3.80	Am241	σ_{capt}	13	7.61	10	0.76	4.30
Np237	σ_{capt}	14	39.20	4	1.57	4.88	Uncertainty N_f (Am241): 5.89							Uncertainty N_f (Cm243): 21.52						
U236	σ_{capt}	11	18.17	8	1.45	5.10	Am241	σ_{capt}	14	47.60	10	4.76	4.76	Cm242	σ_{capt}	11	40.39	40	16.16	16.16
Np237	σ_{capt}	13	10.55	10	1.06	5.20	Pu239	σ_{capt}	14	62.56	3	1.88	5.12	Cm242	σ_{capt}	12	21.01	40	8.40	18.21
Uncertainty N_f (Pu239): 2.22							U238	σ_{capt}	12	50.73	3	1.52	5.34	Cm242	σ_{capt}	14	16.45	40	6.58	19.36
U238	σ_{capt}	12	49.01	3	1.47	1.47	Pu240	σ_{capt}	13	19.27	7	1.35	5.51	Cm243	σ_{fiss}	13	11.78	40	4.71	19.93
Pu239	σ_{fiss}	14	50.05	2	1.00	1.78	Am241	σ_{capt}	13	13.08	10	1.31	5.66	Cm242	σ_{capt}	15	9.91	40	3.96	20.32
Pu239	σ_{capt}	14	32.19	3	0.97	2.02	Uncertainty N_f (Am242m): 6.86							Cm243	σ_{fiss}	14	9.14	40	3.65	20.64
U238	σ_{capt}	11	29.68	3	0.89	2.21	Am242 m	σ_{capt}	14	10.96	40	4.38	4.38	Uncertainty N_f (Cm244): 13.14						
Uncertainty N_f (Pu240): 6.20							Am242 m	σ_{fiss}	14	49.89	5	2.49	5.04	Am243	σ_{capt}	13	56.52	20	11.30	11.30
Pu240	σ_{capt}	13	80.02	7	5.60	5.60	Am241	σ_{capt}	14	23.45	10	2.34	5.56	Pu242	σ_{capt}	13	77.65	5	3.88	11.95
Pu239	σ_{capt}	14	57.77	3	1.73	5.86	Am242 m	σ_{capt}	15	5.04	40	2.02	5.92	Pu240	σ_{capt}	13	38.00	7	2.66	12.24
U238	σ_{capt}	12	49.65	3	1.49	6.05	Pu239	σ_{capt}	14	63.25	3	1.90	6.21	Cm244	σ_{capt}	12	5.73	40	2.29	12.46
							U238	σ_{capt}	12	50.85	3	1.53	6.40	Pu239	σ_{capt}	14	69.39	3	2.08	12.63
							Pu240	σ_{capt}	13	21.05	7	1.47	6.57	Uncertainty N_f (Cm245): 34.47						
														Cm244	σ_{capt}	12	77.09	40	30.84	30.84
														Am243	σ_{capt}	13	58.56	20	11.71	32.99