GENERAL ATOMIC

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LIST OF SYMBOLS

A	activity
A	experimentally determined constant in the sorption isotherm equations
A_i	total core fuel block activity of isotope i
$A_{i,cir}$	circulating activity of isotope i
$A_{i,plt}$	plateout activity of isotope i
A_k	$\log (a_k/\hat{a}_k)$ for specific plateout activity
a	kernel grain radius
a_i	$\partial \ln R / \partial \ln d_{si}$, proportional to sensitivity of release to systematic changes in the variable x_i
a_k	specific plateout activity at k^{th} location in the primary circuit
B	experimentally determined constant in the sorption isotherm equations
B_T	fission product birth rate in TRISO particles
C	fission product concentration
C_t	Langmuir transition concentration in sorption isotherm equations
D	diffusivity
D	experimentally determined constant in the sorption isotherm equations
d	hydraulic diameter
d_{ril}	random uncertainty factor for variable x_{il}
d_{si}	systematic uncertainty factor for variable x_{il}
E	experimentally determined constant in the sorption isotherm equations
F_B	fraction of BISO particles failed
F_T	fraction of TRISO particles failed
$\mathcal{F}$	factor by which release from failed TRISO particles is increased due to hydrolysis effects
f	fraction of failed TRISO fuel that is hydrolyzed
k	mass transfer coefficient
k_f	fuel rod thermal conductivity

k_g	graphite thermal conductivity
M	mass of total circulating helium
M_{ik}	$\partial A_k / \partial x_i$ for plateout activity, the slope of a plateout activity sensitivity curve
$\overline{M}_i$	average value of the slope of a sensitivity curve over a range of interest
m_c	primary coolant flow rate
m_s	purification system flow rate
n	exponent in R/B versus half-life relationship
P	reactor thermal power
p	fission product vapor pressure
Q	activation energy
R	total core fission product release of a particular nuclide
R	ideal gas constant
$\hat{R}$	nominal total core release of a particular nuclide
R_f	$\log (R/\hat{R})$ for metallic fission product release
R_ℓ	fission product release from a volume element at local point ℓ
R_m	median total core release of a particular nuclide
R_T	release rate from TRISO particles
Re	Reynolds number
R/B	steady-state release rate to birth rate ratio
$(R/B)_{WH}$	release fraction from hydrolyzed failed TRISO particles
$(R/B)_{WO}$	release fraction from unhydrolyzed failed TRISO particles
$(R/B)_f$	R/B for failed fuel particles
$(R/B)_m$	a fixed value of R/B at very low temperatures
$(R/B)_{min}$	minimum R/B value assumed in fission product release calculations
$(R/B)_o$	R/B at a reference temperature
$(\overline{R/B})_i$	core average release fraction for isotope i
r_p	plateout removal rate constant
r_s	purification removal rate constant
S_i^2	variance of i
Sc	Schmidt number
Sh	Sherwood number

T	temperature
T_{in}	core inlet helium temperature
T_o	reference temperature
TF	trapped fraction of cesium in thorium oxide kernels
t	fuel irradiation time
$t_{1/2}$	radioactive half-life
V	coolant velocity
x_i	$\log (y_i/\hat{y}_i)$ for metallic fission product release or specific plate-out activity
$x_{i\ell}$	value of i^{th} variable affecting release at local point ℓ
Y_i	cumulative fission yield of isotope i
y_i	i^{th} variable affecting total core release or specific plateout activity
$\alpha(T)$	temperature-dependent coefficient
$\beta(T)$	temperature-dependent coefficient
λ_i	radioactive decay constant for isotope i
μ	helium viscosity
μ_i	median of i
ξ_i	diffusion parameter for isotope i
π	permeation coefficient
ρ	helium density
$\rho_{x_i x_j}$	correlation coefficient between variables x_i and x_j
σ_i	standard deviation of i
σ_i^2	variance of i
τ	fractional plateout per pass of condensable species

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1. INTRODUCTION AND SUMMARY

1.1. INTRODUCTION

The primary objective of the Accident Initiation and Progression Analysis (AIPA) Program is to provide guidance for high-temperature gas-cooled reactor (HTGR) safety research and development. In the preliminary studies reported in Volume III of this report series, a first cut analysis of seventeen initiating events was performed to select for more detailed evaluation the events which are expected to contribute most to the attainment of the AIPA objective. In addition, several design basis events were selected for detailed consequence evaluation to provide a foundation for meaningful design option studies such as those presented in Volume III. The detailed consequence evaluations are described in Volume VI. They differ from the first cut analyses, which were based on data reported in various HTGR PSARs, in that a separate, realistic (i.e., less conservative) uncertainty assessment is made of the sequence of steps from fission product release from the PCRV to predicted doses at representative HTGR site boundaries.

Among the parameters considered in estimating the uncertainties in site boundary doses are uncertainties in fission product source terms generated under normal operating conditions, i.e., fuel body inventories, circulating coolant activity, total plateout activity in the primary circuit, and plateout distributions. This volume documents the analyses of these source term uncertainties. The results are used for the detailed consequence evaluations in Volume VI, and they provide the basis for evaluation of fission products important for HTGR maintenance and shielding reported in Volume VII.

Beginning with mathematical representations of fission product transport in the core and using fundamental fission product transport

properties and their uncertainties, a statistical method (the propagation of errors method) has been employed to obtain uncertainties in fission product release from a 3000-MW(t) HTGR core under normal operating conditions. Parameters which contribute the most to these uncertainties have been identified. This volume describes the methods used in these analyses and the approximations and limitations inherent in the methods. Source term activity uncertainty analyses and summaries of uncertainty analyses of fundamental fission product transport properties are described.

Figure 1-1 illustrates the analytical sequence used in this study.* Six groups of input data were supplied to various core design computer programs to obtain the fission product source terms of interest. The sections of this volume in which each result is described are shown in parentheses in Fig. 1-1. Uncertainty analyses were performed for the release of cesium-137 and strontium-90 (Section 3), followed by uncertainty analyses of fuel body activity, circulating gaseous activity, and total plateout activity, including iodine activity (Section 4). The results of these studies were then used to calculate uncertainties in fission product plateout distributions on the components of the primary circuit (Section 5).

In this study, uncertainties in fission product activity levels were determined by examining determinate uncertainties only in basic data and operating parameters. Indeterminate uncertainties induced by various modeling and design assumptions were not evaluated; in some cases, it was not possible within the framework of this study to replace conservative modeling approximations with more realistic assumptions. Examples of indeterminate uncertainties include uncertainties due to grouping of fission products into broad classifications, uncertainties inherent in transport modeling assumptions, and uncertainties in materials data introduced by effects which are known to occur but are presently not quantifiable. Thus, the resulting uncertainties are not purported to be all-inclusive or final. However, the general order of importance assigned to variables affecting the uncertainties is essentially correct.

*References, figures, and tables are located at the end of each section.

The accuracy with which the calculational models represent the fission product transport process will be evaluated in a fission product code validation program currently under way at General Atomic (GA) as a subtask of the ERDA-supported Fuels and Core Development Program, contract E(04-3)-167, PA-17. The validation program is described in Ref. 1-1. In this program, calculated and observed in-pile data will be compared, thereby yielding information on the importance of indeterminate uncertainties.

The statistical procedure used in this study yields median and upper 95% confidence activity levels which differ from current GASSAR activity level criteria (Ref. 1-2). In GASSAR, two activity level criteria are employed: expected and design. Current expected and design activity levels were established without the aid of any quantitative estimates of uncertainties in the inventories. Expected activity levels have been established by calculations which were anticipated to produce realistic but still conservative estimates of the inventories of fission product nuclides. Design activities have arbitrarily been set about a factor of ten higher than expected activities for most isotopes to provide assurance of an extremely low probability that the design activities will be exceeded during normal operation. Expected activity criteria are used for containment accessibility planning and in calculations for environmental impact reports. Design activity criteria are used as fission product source terms for design of certain critical plant components, safety analyses, equipment maintenance planning, and shielding design and are proposed in the technical specifications as the maximum allowable level for circulating activity. Therefore, in this report, comparisons are made between GASSAR activity criteria and the AIPA median and 95% confidence activity levels to assess the degree of conservatism inherent in the GASSAR activities.

1.2. SUMMARY

The results of the fission product source term studies are summarized in Table 1-1 and Fig. 1-2. Table 1-1 shows that a great deal more

confidence can presently be placed in the calculations of circulating activity than in any other release calculation. Fuel element activities are very certain because uncertainty in these values is controlled by uncertainty in total core power. On the other hand, plateout activity distributions are very uncertain.

Figure 1-2 qualitatively compares median and upper 95% confidence activities with current GASSAR activity criteria (Ref. 1-2). As shown in Fig. 1-2, 95% confidence activities are at or below GASSAR design activity criteria, with the exception of cesium and silver activities. Cesium and silver radionuclides are shown in Volume VII to contribute a major share of the total dose to personnel during certain maintenance operations. Clearly, if the uncertainties in the activities of these fission products are to be reduced, more work is required to characterize cesium and silver transport and release.

Most of the uncertainty in cesium-137 release was found to be due to uncertainties in cesium diffusivity in graphite and pyrocarbon (see Table 3-2). The TRAFIC code, used to calculate direct metallic fission product release from the HTGR core, assumes that diffusion of metallic species in core materials is described by the classical Fickian model. However, a number of investigators at GA and elsewhere have obtained experimental data which indicate that for relatively volatile species, such as cesium, a two-path diffusion model may be more appropriate. Thus, an effort to develop and validate a more advanced cesium transport model is recommended if the uncertainty in calculated cesium release is to be reduced. Continuation of experiments to characterize cesium migration in graphite and pyrocarbon having various structural properties and additional studies of cesium release from coated particle kernels are also necessary.

With regard to circulating and total plateout inventories, the calculated upper 95% confidence activity levels are, in general, distributed about the GASSAR expected activity criteria (see Tables 4-1 through 4-4). It was found that limited knowledge of the plateout behavior of condensible nuclides onto graphite and metal surfaces has a strong

influence on the uncertainty in circulating activity of those nuclides, particularly the isotopes of cesium and iodine, which are found in Volumes VII and VI, respectively, to contribute a major share of the total dose under certain maintenance or accident conditions. The uncertainty in plateout behavior of condensible species was also found (Section 5) to dominate the total uncertainty in plateout activity distributions in the primary coolant circuit (see Table 5-8). Extensive work is clearly required to improve characterization of the plateout process on primary circuit components and is accordingly recommended.

The uncertainty in the circulating activity of noble gas isotopes is dominated by the uncertainty in core average release fraction. This uncertainty is in turn dominated by uncertainty in R/B (steady-state release rate to birth rate ratio) for as-manufactured fuel, a fuel manufacturing parameter determined by the amount of thorium and uranium contamination in the fuel rod matrix. The second most important variable contributing to the total noble gas activity uncertainty is the increase in R/B for failed TRISO particles due to hydrolysis. The magnitude of this effect was estimated; no detailed calculational methods are presently available which couple the effect of moisture transport in graphite to steady-state fission gas release. Other important variables affecting the uncertainty in core average release fraction are shown in Table 4-8. The calculated uncertainty in gaseous fission product release activity is low relative to plateout inventory uncertainties. However, this relative comparison is subject to verification of the models used in fission gas release calculations.

More realistic methods for calculating release activity levels for the special nuclides discussed in Section 4.4 should be used. The methods currently in use employ very conservative approximations which ultimately result in unnecessary plant cost increases.

In Sections 6, 7, and 8, limitations of the uncertainty analyses of fundamental fission product release and plateout data are discussed. Many of these limitations could be eliminated by a program of bench mark

measurements of certain fundamental data. In such a program, standard or common test specimens would be used in different laboratories with similar experimental techniques to measure parameters which contribute to the uncertainties in fission product activity levels. A consensus regarding the accuracy of measurements could substantially reduce calculational uncertainties.

The largest contribution to the uncertainty in strontium-90 release was found to be due to uncertainties in thermal design parameters (Table 3-2). However, these uncertainties are presently not crucial to the fuel or core design or the results in Volume VI since the calculated median and upper 95% confidence activities are well within the present expected and design release criteria of GASSAR. As discussed in Section 3, however, the low calculated strontium release activity is dependent upon the value of strontium diffusivity in thorium oxide kernels, for which only limited data exist. Hence, it is necessary to obtain more data on this material property to ensure that the strontium release has not been underestimated owing to indeterminate factors.

More information on uncertainties in calculated fission product source terms can be obtained by (1) further, more refined measurements of certain fundamental fission product transport parameters and (2) continued evaluation of in-pile experiments, as is being done for the fission product code validation program. The results obtained will guide further development of calculational methods and will eventually reduce source term uncertainties.

REFERENCES

- 1-1. Jensen, D. D., et al., "Planning Guide for Validation of Fission Product Transport Codes," ERDA Report GA-A13386, General Atomic, April 15, 1975.
- 1-2. "GASSAR-6," General Atomic Standard Safety Analysis Report, February 5, 1975 (NRC Docket STN 50-535), Section 11.1.

Fig. 1-1. AIPA fission product source term analysis sequence

Fig. 1-2. Comparison of GASSAR activities and AIPA calculations (schematic, not to scale)

TABLE 1-1
RATIOS OF UPPER 95% CONFIDENCE LEVELS TO MEDIAN ACTIVITY LEVELS

<u>Case</u>	<u>Ratio of 95% Level/Median Level</u>
1. Fuel element inventories (Table 4-5)	1.01
2. Circulating activity (Tables 4-1, 4-3)	
H-3	3.13
Kr-85m, Kr-87, Kr-88, Kr-89, Kr-90	1.63
Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-137, Xe-138	1.77
I-131, I-132, I-133, I-134, I-135, I-136	5.35
3. Plateout activity (Tables 4-2, 4-4)	
I-131, I-132, I-133, I-135	1.59
Sr-90	23.1
Cs-137	29.9
4. Plateout distributions (Tables 5-1 to 5-7)	
I-131 reheater	61
circulator	67
Sr-90 reheater	20
circulator	10
Cs-137 reheater	150
circulator	9980

2. METHOD OF ANALYSIS

In this section, the statistical method used in the AIPA analyses of fission product release and distribution is described. The limitations of the method are also discussed.

Calculated fission product release is uncertain owing to imperfect knowledge of the variables which govern release. For each important fission product nuclide, the uncertainty in release from the core can be quantified by a confidence statement of the following form:

There is a $100(1 - \alpha)\%$ confidence that core release of a given fission product nuclide will not exceed $R_{1-\alpha}$, where $0 < \alpha < 1$.

To make this confidence statement, i.e., to determine corresponding values for α and $R_{1-\alpha}$, the following information is required:

1. Median release value.
2. Uncertainty parameter for the release, e.g., the variance.
3. Distribution function for the release.

The method used to quantify the core release uncertainties for the AIPA study is a generalization of the propagation of errors method, in which random and systematic uncertainties are accounted for. To implement the method, uncertainty estimates are required for all important variables which affect release. The sensitivity of the release to these variables must also be determined. The application of the method and the assumptions inherent in the method as applied are described in this section.

2.1. RANDOM AND SYSTEMATIC UNCERTAINTIES

As previously stated, the calculated fission product release from the core is uncertain because many of the important contributing mechanisms and variables are not precisely known. Some of the uncertainties vary with spatial position in the core, and some vary with time. Other variables and their uncertainties are independent of space and time. This variation or lack of variation allows the uncertainties to be characterized as random or systematic, respectively. Such a distinction can be made for spatially varying uncertainties and time-varying uncertainties. For example, the uncertainty in power tilt is spatially random, whereas the uncertainty in the total core power is spatially systematic. The uncertainty in the power tilt is probably also random in time (because of control rod motions), whereas the uncertainty in a fission product diffusion coefficient is systematic in time (and also in space). The above characterization of types of uncertainties is summarized in the table below.

<u>Type Number</u>	<u>Spatial Variation</u>	<u>Time Variation</u>	<u>Example</u>
1	Random	Random	Power tilt
2	Random	Systematic	Block dimensions
3	Systematic	Random	Total core power
4	Systematic	Systematic	Diffusion coefficients

The distinction between random and systematic uncertainties in the variables governing release is important because the two types of uncertainty have different effects on the overall uncertainty in the core release. For example, in space, random uncertainties tend to average out, whereas systematic uncertainties do not. Thus, random uncertainties have a small effect on the uncertainty in the core release. However, random uncertainties cannot be completely neglected because the core release depends in a nonlinear manner on the input variables. As a result, the

random uncertainties increase the median value (50% confidence value) of the core release above that which is calculated from the median (nominal) values of the variables. On the other hand, the systematic uncertainties do not affect the median release. The above can be summarized as follows:

1. The uncertainty in the core release is determined by the systematic uncertainties.
2. The median value of the core release is determined by the median (nominal) values of the variables and by the random uncertainties.

It is particularly important to distinguish between the two types of uncertainty in space; the distinction in time is of less importance. Indeed, in the analysis for AIPA, all uncertainties in time are assumed to be time independent, i.e., systematic in time. Accordingly, in the remainder of this section, random and systematic refer to spatial variations. Elsewhere, the terms local and global are used.

The classification of an uncertainty as either random or systematic is not always easy to make for two reasons. First, it is necessary to decide over what spatial volume a variation is considered to be significant. For example, some variables vary from rod to rod, others vary from block to block, and still others vary from region to region. Strictly speaking, there should be rod uncertainties, block uncertainties, region uncertainties, and core uncertainties instead of just random and systematic uncertainties. In the following discussion, however, rod, block, and region uncertainties are considered to be random, and all are assumed to vary from one local calculational point to another. The core uncertainties are considered to be systematic.

A second complication in characterizing uncertainties is that some (e.g., uncertainty in power) have random and systematic components. The

random uncertainty in power is associated with the power tilt, the column peaking factor, and the region peaking factor. The systematic uncertainty is associated with the total core power. For most variables with two uncertainty components, one or the other clearly dominates. In any case, estimates of each should be made because their statistical treatment is different.

2.2. MEDIAN VERSUS NOMINAL RELEASE: THE LOG NORMAL DISTRIBUTION

The dependence of core release of a particular nuclide on spatial position and variables governing release can be indicated as follows:

$$R = \sum_{\ell} R_{\ell} (x_{1\ell}, x_{2\ell}, \dots x_{m\ell}) \quad , \quad (2-1)$$

where $R_{\ell} \equiv$ release from volume element associated with local point ℓ ,
 $x_{i\ell} \equiv$ value of i^{th} variable at local point ℓ .

To derive expressions for the median and variance of the release, the following assumptions are made regarding the uncertainties in the variables and the relationship between these uncertainties and the core release:

1. The uncertainties in the variables are of two types, random and systematic. These uncertainties may be mathematically introduced by letting

$$x_{i\ell} = d_{si} d_{ri\ell} \hat{x}_{i\ell} \quad , \quad (2-2)$$

where $d_{si} \equiv$ systematic uncertainty factor, which is independent of ℓ ,

$d_{ri\ell} \equiv$ random uncertainty factor, which depends on ℓ ,
 $\hat{x}_{i\ell} \equiv$ nominal (median) value of i^{th} variable at local point ℓ . (For each variable, the nominal value is taken to be the median value.)

2. The logarithms of the systematic uncertainty factors $\ln d_{si}$ have normal distributions, with means equal to 0 and variances equal to $\sigma_{\ln d_{si}}^2$.
3. The $\ln d_{ril}$ have normal distributions, with means equal to 0 and variances equal to $\sigma_{\ln d_{ri}}^2$, which are independent of ℓ .
4. The $\ln d_{si}$ and the $\ln d_{ril}$ are independent. (Assumption 1 is interpreted to mean that the $\ln d_{si}$ are independent of the $\ln d_{ril}$).
5. The release is a multiplicative function of the independent variables; i.e.,

$$R = \sum_{\ell} \left[\prod_i (d_{si} d_{ril})^{a_i} \right] \hat{R}_{\ell} = \left(\prod_i d_{si}^{a_i} \right) \sum_{\ell} \left(\prod_i d_{ril}^{a_i} \right) \hat{R}_{\ell} \quad , \quad (2-3)$$

where

$$a_i \equiv \frac{\partial \ln R}{\partial \ln d_{si}} \quad , \quad (2-4)$$

$\hat{R}_{\ell} \equiv$ nominal release from local point ℓ , i.e., the release calculated using nominal values for the variables.

6. The release is not highly localized.
7. The uncertainty in the local release is not dominated by random uncertainties.

Given these assumptions, the uncertainty in R , or equivalently in $\ln R$, can be quantified (Refs. 2-1, 2-2). The essential results are as follows:

1. The median (and mean) of $\ln R$ is

$$\mu_{\ln R} = \ln \hat{R} + \frac{1}{2} \sum_i a_i^2 \sigma_{\ln d_{ri}}^2 \quad . \quad (2-5)$$

2. The variance of $\ln R$ (see Section 2.3) is

$$\sigma_{\ln R}^2 = \sum_i a_i^2 \sigma_{\ln d_{si}}^2 \quad (2-6)$$

3. The distribution function of $\ln R$ is normal.

Equation 2-5 shows that the mean and median of $\ln R$ are increased above the nominal value only because of the random uncertainties, and Eq. 2-6 shows that the variance of $\ln R$ depends only on the systematic uncertainties. Because $\ln R$ has a normal distribution, it follows that R has a log normal distribution. The latter is skewed to high values of R .

The analogous properties for R are as follows:

1. The median of R is

$$\mu_R = \exp \left(\ln \hat{R} + \frac{1}{2} \sum_i a_i^2 \sigma_{\ln d_{ri}}^2 \right) \quad (2-7)$$

2. The variance of R is

$$\sigma_R^2 = \left[\exp \left(2\mu_{\ln R} + \sigma_{\ln R}^2 \right) \right] \left[\exp \left(\sigma_{\ln R}^2 \right) - 1 \right] \quad (2-8)$$

3. The distribution function of R is log normal.

These results are not without limitations. Probably the most restrictive limitations are associated with assumption 5, which concerns the functional form of the release. Two limitations are implicit in the functional form of Eq. 2-3:

1. With regard to systematic uncertainties, it is assumed that $\ln R$ is linear in $\ln d_{si}$.
2. With regard to random uncertainties, it is assumed that $\ln R_\ell$ is linear in $\ln d_{ril}$ with slope a_i independent of ℓ .

The assumption that $\ln R$ is linear in $\ln d_{si}$ can be relaxed if necessary. This is done by including the second partial derivatives ($\partial^2 \ln R / \partial \ln d_{si}^2$) and ($\partial^2 \ln R / \partial \ln d_{si} \partial \ln d_{sj}$) in Eq. 2-6. The extension of the derivation to include these derivatives is straightforward and results in a modification of the usual propagation of errors formula for the variance of $\ln R$. The unmixed partial derivatives ($\partial^2 \ln R / \partial \ln d_{si}^2$) are available from the sensitivity study performed to implement the linear version of the method. However, the mixed partial derivatives ($\partial^2 \ln R / \partial \ln d_{si} \partial \ln d_{sj}$) are not available and to determine them would require additional sensitivity calculations.

It should be noted that if $\ln R$ is concave downward in $\ln d_{si}$, then a conservatively large uncertainty in release is obtained by using only the first partial derivative. This is illustrated in Fig. 2-1. If the resulting conservatism is acceptable, then second partial derivatives need not be considered.

These limitations can be removed by a generalized method. However, this method entails more calculational effort than the simple method outlined above and would require some code development to implement.

2.3. PROPAGATION OF ERRORS: VARIANCE OF THE RELEASE

As stated in the introduction to Section 2, to make a confidence statement on the core release, the median release value, the variance of the release, and the release distribution function are needed. In this section, the method used to calculate the variance of the release is described in more detail.

The variance of the release is calculated using the propagation of errors method (Ref. 2-1). The total core release R can be written in terms of the important variables x_i that contain systematic uncertainties:

$$R = R(x_1, x_2, x_3, \dots x_n) \quad , \quad (2-9)$$

where the local point subscript ℓ has been dropped. As previously discussed, only variables with systematic uncertainties need be considered when calculating the variance of the release. Random uncertainties are assumed to average out over the core.

If propagation of errors is applied directly to Eq. 2-9, the following formula for the variance of R is obtained:

$$\sigma_R^2 = \sum_{i=1}^n \left(\frac{\partial R}{\partial x_i} \right)^2 \sigma_{x_i}^2 + 2 \sum_{i=1}^{n-1} \sum_{j>i}^n \rho_{x_i x_j} \left(\frac{\partial R}{\partial x_i} \right) \left(\frac{\partial R}{\partial x_j} \right) \sigma_{x_i} \sigma_{x_j} \quad , \quad (2-10)$$

where $\sigma_{x_i}^2$ and $\rho_{x_i x_j}$ are the variance and the correlation coefficient, respectively, of the appropriate variables. This formula is applicable if R is approximately linear in the variables $x_1, x_2, \dots x_n$ over the range of their uncertainties. In fact, fission product release is usually not a linear function of the governing variables. Thus, in order to apply the propagation of errors method, a linearizing transformation of the release function is required. In this case, since release is a multiplicative function of the x_i (Eq. 2-3), a logarithmic transformation is appropriate, and Eq. 2-9 can be rewritten as

$$\ln R = \ln R(\ln x_1, \ln x_2, \ln x_3, \dots \ln x_n) \quad . \quad (2-11)$$

Applying the propagation of errors method to this equation gives

$$\begin{aligned} \sigma_{\ln R}^2 &= \sum_{i=1}^n \left(\frac{\partial \ln R}{\partial \ln x_i} \right)^2 \sigma_{\ln x_i}^2 \\ &+ 2 \sum_{i=1}^n \sum_{j>i}^n \rho_{\ln x_i, \ln x_j} \left(\frac{\partial \ln R}{\partial \ln x_i} \right) \left(\frac{\partial \ln R}{\partial \ln x_j} \right) \sigma_{\ln x_i} \sigma_{\ln x_j} \quad . \quad (2-12) \end{aligned}$$

The assumption is made that $\ln R$ is approximately linear in $\ln x_1, \ln x_2, \dots \ln x_n$. Equation 2-12 provides the variance of $\ln R$, assuming all the quantities on the right side of the equation are known. It should be noted that Eq. 2-12 simplifies if all the important variables $x_1, x_2, \dots x_n$ are independent. In that case, the correlation terms contained in the double summation vanish, and only the first term in Eq. 2-12 remains.

The major limitations in the method are as follows:

1. If $\ln R$ is not linear in $\ln x_i$ over the range of uncertainty in x_i , terms containing higher partial derivatives in x_i may be needed in Eq. 2-12.
2. If the release is highly localized in the core, random uncertainties may not average out and may affect the uncertainty in core release.

To make a confidence statement on the core release, the release distribution function is needed; this information must be assumed. Given the form of Eq. 2-12, it is assumed that $\ln R$ has a normal distribution. Hence, R itself has a log normal distribution. Since the log normal distribution is skewed to higher values of R , the median value of the release is higher than the nominal value, as discussed in the previous section and shown by Eq. 2-5 (Ref. 2-2).

2.4. PROCEDURE

The procedure to be followed to implement the method can be summarized as follows:

1. Calculate the nominal release, i.e., the release in the absence of uncertainty, using median (nominal) values of the independent variables.

2. Identify the important variables that are uncertain. Determine whether the uncertainties are random or systematic.
3. Estimate the uncertainties in the important variables.
4. Calculate the relevant partial derivatives by sensitivity studies.
5. Calculate the median release from the nominal release, the random uncertainties, and the partial derivatives.
6. Calculate the variance of the release from the systematic uncertainties and the partial derivatives.
7. Make a confidence statement on the release.

The methods used to calculate nominal metallic release, gaseous release, and plateout distributions are described in the following sections. The identification of the important variables and the estimation of their uncertainties are also discussed.

Uncertainty estimates for each important variable are required. Specifically, for each x_i the variances $\sigma_{\ln d_{si}}^2$ and $\sigma_{\ln d_{ril}}^2$ must be estimated, where

$$x_{il} = d_{si} d_{ril} \hat{x}_{il} \quad , \quad (2-13)$$

and hence

$$\ln x_{il} = \ln \hat{x}_{il} + \ln d_{si} + \ln d_{ril} \quad . \quad (2-14)$$

Although the derivation of the method assumes that the distributions of $\ln d_{si}$ and $\ln d_{ril}$ are normal, i.e., d_{si} and d_{ril} are log normal, this is a

weak requirement. Even if the uncertainties in the variables are not log normal, the expressions derived for the median and variance of the release give reasonable results, provided appropriate values are used for $\sigma_{\ln d_{si}}^2$ and $\sigma_{\ln d_{ril}}^2$.

For some of the variables, uncertainty estimates are approximate because either the data for the variables are limited or the mechanisms giving rise to the variables are not well known. For the uncertainties in these variables, engineering judgments which encompass the limited state of knowledge are employed.

The relevant partial derivatives,

$$a_i = \left(\frac{\partial \ln R}{\partial \ln d_{si}} \right)_{d_{si}=1}, \quad (2-15)$$

can be calculated for each variable by performing a sensitivity study. This requires that a nominal calculation be available. The sensitivity calculations, which consist of perturbing each variable by about $\pm 2\sigma_{\ln d_{si}}$, are then performed, and $\partial \ln R / \partial \ln d_{si}$ and $\partial^2 \ln R / \partial \ln d_{si}^2$ are obtained. The latter is useful for checking the linearity of $\ln R$ with respect to $\ln d_{si}$ (and $\ln d_{ril}$) as assumed in the derivation.

Because computer calculations of the total core release are expensive, an expedient method may be adopted for calculating the partial derivatives. If the form of the release assumed in the derivation is correct, i.e., if

$$R = \sum_{\ell} R_{\ell} = \sum_{\ell} \left[\prod_i \left(d_{si} d_{ril} \right)^{a_i} \right] \hat{R}_{\ell}, \quad (2-16)$$

then the sensitivity of the release at each local point throughout the core to each variable is the same; i.e.,

$$a_i = \frac{\partial \ln R_{\ell}}{\partial \ln d_{ril}} \quad (2-17)$$

is independent of ℓ . Although this is not the case, it is reasonable to represent the whole core by a subset including a modest number of appropriately selected points which define the envelope of the values of a_i at which to perform the sensitivity calculations. The partial derivatives obtained for the partial core release are assumed to be the same as those for the entire core release. This results in a substantial reduction in the cost of the calculations without loss of the significance of the partial derivatives.

Various procedures for selecting the representative local points are possible. The approach usually adopted is to choose points from the highest releasing axial layer in the core. The motivation for this is that these points contribute the most to the release; hence, they will probably contribute the most to its uncertainty.

In practice, it is convenient to plot the results of the sensitivity calculations on log-log coordinates, as shown in Fig. 2-2. In this case, the logarithms are base 10, whereas in the preceding discussions, they are base e. The two are easily related, however, since

$$\ell n R = \frac{\log R}{\log e} = 2.30 \log R \quad , \quad (2-18)$$

and similarly

$$\ell n d_{si} = 2.30 \log d_{si} \quad . \quad (2-19)$$

Thus, the sensitivity coefficient

$$a_i = \frac{\partial \ell n R}{\partial \ell n d_{si}} = \frac{\partial \log R}{\partial \log d_{si}} \quad (2-20)$$

is the same regardless of the base of the logarithms used. However the variances of the uncertainties for use in Eqs. 2-5 and 2-6 must be adjusted accordingly. In particular,

$$\sigma_{\ln d_{si}}^2 = (2.30)^2 \sigma_{\log d_{si}}^2 = 5.30 \sigma_{\log d_{si}}^2, \quad (2-21)$$

and

$$\sigma_{\ln d_{ri}}^2 = 5.30 \sigma_{\log d_{ri}}^2. \quad (2-22)$$

From the above results, confidence statements on $\ln R$, and hence on R , are easily made. Thus, the median of R is $e^{\mu_{\ln R}}$, since

$$P \left\{ \ln R \leq \mu_{\ln R} \right\} = P \left\{ R \leq e^{\mu_{\ln R}} \right\} = 0.5. \quad (2-23)$$

Similarly, the upper one-sided 95% confidence limit on R is $e^{\mu_{\ln R} + 1.65\sigma_{\ln R}}$, since

$$P \left\{ \ln R \leq \mu_{\ln R} + 1.65\sigma_{\ln R} \right\} = P \left\{ R \leq e^{\mu_{\ln R} + 1.65\sigma_{\ln R}} \right\} = 0.95. \quad (2-24)$$

The major limitations of the procedure are as follows:

1. The calculational models contain some conservatisms that are difficult to remove. These are discussed in subsequent sections.
2. The uncertainty estimates for some of the variables are approximate owing to limited data.
3. Uncertainties due to modeling errors are neglected. The accuracy of fission product transport codes will be assessed under a subtask of the HTGR Base Program Plan, PA 17. The code validation program is discussed in Ref. 1-1.
4. The partial derivatives of release from the entire core are approximated by those of the release from selected points in the core.

The first three limitations can be remedied only by obtaining better data and models. The fourth limitation arises solely from attempts to limit the cost of performing the sensitivity calculations.

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Fig. 2-1. Comparison of confidence limits for $\ln R$ concave downward in $\ln d_{si}$

Fig. 2-2. Typical sensitivity plot

3. DIRECT METALLIC FISSION PRODUCT RELEASE

Metallic fission products released directly from the HTGR core are divided into three groups for calculational purposes. The first group consists of elements with release characteristics assumed to be like those of cesium. This group includes the rubidium isotopes. The second group consists of elements with release characteristics like those of strontium and includes barium, europium, and samarium. Most of the remaining metallic elements are less volatile and are assumed to have release characteristics corresponding to a minimum release fraction. These assumptions are necessitated by lack of data and are justified by the fact that these elements have a small impact on the health and safety of the public and the safe operation of the plant.

For the AIPA studies, uncertainties in direct release of strontium-90 and cesium-137 were calculated. The results of these calculations are employed in Section 4.4, where total plateout inventories and associated uncertainties are obtained for several metallic isotopes using the above assumptions. The calculational methods used are essentially those discussed in Section 2, with exceptions pointed out in the following sections. The most important results of the metallic fission product direct release AIPA studies are summarized in Tables 3-1 and 3-2.

It is clear in Table 3-1 that median (50% confidence) release for strontium-90 is well within the GASSAR expected release criterion, and the upper 95% one-sided release level is well within the GASSAR design release criterion. However, the median release for cesium-137 exceeds the GASSAR expected release criterion by about a factor of two. As discussed in Section 3.1, certain conservatisms contained in the TRAFIC model may result in higher than nominal, or median, calculated release. With these conservatisms removed, the median release is approximately equal to the GASSAR expected release criterion for cesium-137.

From Table 3-2, conclusions can be drawn concerning the contribution of different variables to the uncertainty in release. For strontium-90, the two largest contributions to the variance (74% of the total) are due to uncertainty in coolant flow rate and uncertainty in region mismatch temperature. Both these parameters are fundamental thermal design parameters. Eighty-one percent of the total variance in strontium-90 release is due to uncertainties in thermal design parameters. This result is not totally unexpected, since strontium vaporization rates and diffusion coefficients in core materials are characterized by relatively high activation energies and hence are quite sensitive to temperature.

The conclusions regarding strontium-90 release uncertainty must be qualified with one important point. As shown in Fig. 3-1, calculated strontium release is independent of diffusivity in the ThO_2 kernel over a wide range about the nominal diffusivity. Hence, uncertainty in this variable contributes nothing to the variance of the release. However, the data on strontium diffusion in oxide kernels consist of only five data points (Ref. 3-1). A mathematically correct statistical analysis of these data yields an uncertainty within the flat portion of the curve in Fig. 3-1. Intuitively, one does not feel comfortable with only five data points knowing that if the diffusivity in the kernel increases by more than four orders of magnitude, the calculated strontium release increases very sharply, as shown in Fig. 3-1. Clearly, materials properties considerations could become more important for strontium if the kernel diffusivity were higher. More data on diffusivity on strontium in oxide kernels should be obtained.

On the other hand, for cesium-137, the two largest contributors to the variance (83% of the total) are uncertainty of diffusivity in graphite and uncertainty of diffusivity in PyC. Such fundamental transport parameters contribute to 89% of the variance in cesium-137 release. This result is understandable in view of the uncertainty which exists regarding the assumed Fickian behavior of cesium transport in core materials.

In summary, uncertainty in calculated direct strontium-90 release can be reduced by reducing the uncertainties in core thermal calculations. Uncertainty in calculated direct cesium-137 release can be reduced by better characterization of fundamental cesium transport parameters in core materials. It is also essential to validate the models and the input data by comparison of predicted releases with values observed in representative in-pile experiments.

3.1. NOMINAL DIRECT METALLIC RELEASE

Nominal direct metallic fission product release at the end of plant life was calculated with the TRAFIC code, essentially following the method described in Ref. 3-2. The method employed in this study differed in two respects. First, thermal diffusion effects were not included in the nominal calculations but were included as an uncertain variable in the calculation of the variance of the release. Second, for cesium release calculations, data have been included which indicate that a fraction of the cesium produced in thorium oxide fuel kernels is trapped in the kernel during postirradiation annealing (see Section 6). Thus, release from the BISO particles is reduced by a factor of $(1 - TF)$, where TF is the trapped fraction of cesium. Uncertainty in the trapped fraction was taken into account in the calculation of the variance. These differences were employed in an attempt to remove some of the conservatisms discussed in Ref. 3-2 and make the nominal cesium-137 release calculation as representative as possible of actual in-core conditions.

As can be seen in Table 3-1, the calculated nominal cesium-137 release exceeds the GASSAR expected release criterion by about a factor of two (Ref. 3-3). However, there are still conservatisms contained in the TRAFIC model which were not removed in these calculations and are reflected in the values shown in the table. Such conservatisms make the nominal calculated release somewhat off-nominal. These conservatisms are discussed at length in Ref. 3-2 and are listed below.

1. The effectiveness of the kernel as a diffusive barrier to release from BISO particles is neglected.
2. The untrapped inventory in the kernel is immediately released to the fuel rod matrix upon particle failure.
3. The effectiveness of the fuel rod matrix as a diffusive barrier is neglected.
4. No kinetic limitation to transport across the fuel rod/graphite gap is taken into account.
5. Equilibrium vapor pressure is assumed in the fuel rod/graphite gap.
6. The coolant free stream concentration is equal to zero, thereby neglecting the effects of readsorption of fission products already released to the coolant.
7. No increase in graphite sorptivity under fast neutron irradiation is taken into account.
8. The kinetic limitation on evaporation into the coolant is neglected.

With the removal of these conservatisms and the improved data, a total calculated release reduction factor of two appears likely.

It should also be noted that the calculated median strontium-90 release is well below the GASSAR expected release criterion, and the upper 95% confidence level is representative of the expected release criterion.

3.2. VARIANCE OF DIRECT METALLIC FISSION PRODUCT RELEASE

Nineteen variables with potential systematic uncertainties were considered in the calculation of the variance of the logarithm of metallic fission product release. They are listed in Table 3-3. The metallic fission product uncertainty analysis does not include contributions arising from any variables other than those listed or from modeling errors. Justifiable quantitative assessments of these indeterminate effects cannot be made at this time.

The propagation of errors formulation, essentially as described in Section 2, was used to calculate the variance of the metallic release. Slight differences in notation were employed and are summarized below.

Calculated total core release R is defined:

$$R = R(y_1, y_2, y_3, \dots y_n) \quad , \quad (3-1)$$

where $y_i = i^{\text{th}}$ variable affecting total core release.

Nominal release $\hat{R}$ is defined:

$$\hat{R} = \hat{R}(\hat{y}_1, \hat{y}_2, \hat{y}_3, \dots \hat{y}_n) \quad , \quad (3-2)$$

where $\hat{y}_i =$ nominal value of i^{th} variable.

To make the linearizing transformation,

$$R_f = \log (R/\hat{R}) \quad (3-3)$$

is defined, and

$$x_i = \log (y_i/\hat{y}_i) \quad . \quad (3-4)$$

Then

$$R_f = R_f(x_1, x_2, x_3, \dots, x_n) \quad (3-5)$$

For metallic fission product release, the dependence of R_f on most of the nineteen x_i is found to be closely linear over the range of uncertainty in each x_i . This linearity is important when the propagation of errors formula is used to find the variance of R_f ; viz.,

$$\sigma_{R_f}^2 = \sum_{i=1}^n \left(\frac{\partial R_f}{\partial x_i} \right)^2 \sigma_{x_i}^2 + 2 \sum_{i=1}^{n-1} \sum_{j>1}^n \rho_{x_i x_j} \left(\frac{\partial R_f}{\partial x_i} \right) \left(\frac{\partial R_f}{\partial x_j} \right) \sigma_{x_i} \sigma_{x_j} \quad (3-6)$$

where $\sigma_{R_f}^2$ = variance of R_f ,
 $\sigma_{\log R}^2$ = variance of $\log R$,

$\sigma_{x_i}^2$ = variance of x_i ,

$\rho_{x_i x_j}$ = correlation coefficient between x_i and x_j .

Again, it is assumed that R_f has a normal distribution about a mean value of zero and hence R itself has a log normal distribution.

In the notation of Section 2,

$$x_i = \log(y_i/\hat{y}_i) = \log d_{si} \quad (3-7)$$

Sensitivity studies were performed to determine the values of the partial derivatives $\partial R_f / \partial x_i$. For metallic fission product studies, the FIPERS code was used (Refs. 3-4 through 3-8). This code was chosen over the TRAFIC code because variables 12 through 17 in Table 3-3 are directly input to the code; the mechanics of the sensitivity studies are thereby considerably simplified.

As discussed in Section 2.4, sensitivity calculations were performed over a number of representative core locations. It was assumed that the partial derivatives obtained for the partial core were the same as those for the whole core release.

For the metallic fission product release sensitivity studies, one axial layer of one reload segment (1/32 of the total core) was analyzed. The segment chosen for analysis was not the same for strontium and cesium. For strontium, the segment from which the calculated nominal release was equal to the total core average per segment was chosen. For cesium, the segment with the highest calculated nominal release was chosen to represent the total core. The difference in choice was the result of an arbitrary decision. In future studies of this type, the segment with the highest calculated nominal release should be chosen for both isotopes. As noted in Section 2.4, it is assumed that the segment contributing the most to the release will contribute the most to the uncertainty in the release.

The results of the sensitivity studies are shown in Figs. 3-1 through 3-37. Most of the results are plotted on a log-log scale. Note that the scale of the abscissa varies among the figures. It is clear in some of the figures that R_f is not a linear function of x_i in all cases. However, as discussed in Section 2.4, a conservatively large uncertainty in release is obtained by using only the first partial derivative if the sensitivity curve is concave downward. This is the case for variables which contribute the most to the variance of the release.

Tables 3-4 and 3-5 give the values of the partial derivatives obtained from Figs. 3-1 through 3-37 and the range over which the derivatives were calculated for cesium and strontium, respectively. The derivatives were obtained by calculating the average slope of the sensitivity curves over the ranges indicated for each variable in the tables. The average slope is given by

$$\bar{M}_i = \frac{R_f(x_{iu}) - R_f(x_{il})}{x_{iu} - x_{il}}, \quad (3-8)$$

where $\bar{M}_i$ = average slope of the sensitivity curve for the variable x_i over the range x_{il} to x_{iu} ,

$$\begin{aligned}
x_{iu} &= \text{upper limit of the range of } x_i \text{ over which } \bar{M}_i \text{ is} \\
&\quad \text{calculated,} \\
x_{il} &= \text{lower limit of the range of } x_i \text{ over which } \bar{M}_i \text{ is} \\
&\quad \text{calculated,} \\
R_f(x_{ij}) &= \log (R/\hat{R}) \text{ at the upper } (j = u) \text{ or lower } (j = l) \text{ value of} \\
&\quad x_i.
\end{aligned}$$

The range in x_i over which the calculation of $\bar{M}_i$ was done was usually chosen to be as narrow as possible while still encompassing the $\pm 1\sigma$ uncertainty range of each variable. In this way, errors resulting from any nonlinearity in the sensitivity curves were minimized.

It can be seen from Tables 3-4 and 3-5 that for cesium and strontium, the greatest sensitivities of release are to variables directly related to calculated fuel temperatures such as inlet helium temperature, total core power, total coolant flow rate, etc. However, as shown below, a high sensitivity of release to a given variable does not necessarily result in a large contribution to the variance of the release because the sensitivities must be combined with the uncertainties in x_i to obtain the variance of the release.

Table 3-6 shows the values of the 1σ uncertainties obtained for each of the nineteen variables considered. References to sources of this information are included. The manner in which the values shown in the table were obtained is best illustrated with an example. Consider total core coolant flow rate. According to Ref. 3-10, the standard deviation for this variable is 3.52% of the mean value. Hence, $\sigma_{x_{11}} = \log (1.0352) = 0.015$. The 1σ uncertainty limits on x_i are marked by the vertical lines through the curves in Figs. 3-1 through 3-37.

The relative contributions of each of the nineteen uncertain variables to the variance of R_f is obtained by combining the partial derivatives (Tables 3-4 and 3-5) with the σ values (Table 3-6) to obtain the individual

terms in the propagation of errors formula Eq. 3-6. These terms, including correlation terms, are summarized in Table 3-2 for cesium and strontium. The numbers in parentheses to the left of each column rank each term from the largest contributor to the smallest.

The variances of the total core release are

Cs

$$\sigma_{R_f}^2 = 0.786 \text{ (base 10 logs)}$$

$$\sigma_{R_f}^2 = 4.17 \text{ (base e logs)}$$

Sr

$$\sigma_{R_f}^2 = 0.683 \text{ (base 10 logs)}$$

$$\sigma_{R_f}^2 = 3.62 \text{ (base e logs)}$$

Note that for variables which contribute the most to the variance, the sensitivity curves in Figs. 3-1 through 3-37 are concave downward or nearly linear over the range of uncertainty in x_i (see, for example, Figs. 3-6 and 3-7).

3.3. MEDIAN DIRECT METALLIC RELEASE

As discussed in Section 2, random uncertainties are assumed to have no effect on the variance of the release. However, they do increase the median release (50% confidence) to a value greater than the nominal in accordance with Eq. 2-7, written in this section as

$$R_m = \hat{R} \exp \left[\frac{1}{2} \sum_i \left(\frac{\partial R_f}{\partial x_i} \right)^2 \sigma_{x_{ir}}^2 \right], \quad (3-9)$$

where R_m = median release,
 $\sigma_{x_{ir}}^2$ = random variance (natural logarithmic) of the i^{th} variable,
 $(\partial R_f / \partial x_i)$ = sensitivity of release to variations in x_i , assumed independent of local point, as discussed in Section 2.2.

With the random variances expressed in base 10 logarithms, as was done for metallic release studies, Eq. 3-9 becomes

$$R_m = \hat{R} \exp \left[\frac{5.3}{2} \sum_i \left(\frac{\partial R_f}{\partial x_i} \right)^2 \sigma_{x_{ir}}^2 \right] . \quad (3-10)$$

Of the nineteen x_i listed in Table 3-3, only two were considered to contribute random uncertainties: core power and gap conductance. Table 3-7 summarizes the calculations. Random variances were obtained from Table 4.4-8 of Ref. 3-11. As shown in Table 3-7, median calculated cesium release exceeds the nominal calculated release by a factor of 1.17; median calculated strontium release exceeds the nominal value by a factor of 1.275. These results are reflected in Table 3-1.

3.4. CONFIDENCE STATEMENTS

The three pieces of information required to make confidence statements on metallic fission product release have been obtained:

1. Median release value (end of 40-yr plant life).

a. Cesium = 40,095 Ci.

b. Strontium = 256 Ci.

2. Variance of the logarithm of the release.

a. Cesium = $\sigma_{R_f}^2 = 4.17$ (base e).

b. Strontium = $\sigma_{R_f}^2 = 3.62$ (base e).

3. The distribution function for the release is assumed to be log normal.

Given these data, confidence statements on the release can be made using the following equation, which is similar to Eqs. 2-23 and 2-24:

$$R_c = R_m \exp (\xi_c \sigma_{R_f}) \quad , \quad (3-11)$$

where R_c = release at an upper one-sided confidence level,

ξ_c = number of standard deviations corresponding to the one-sided confidence level of interest.

For example, to obtain a release at the upper 95% confidence limit, $\xi_c = 1.65$; at the upper 80% level, $\xi_c = 0.842$; at the 50% confidence level, $\xi_c = 0$. The results for the upper 95% confidence limit are shown in Table 3-1.

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Fig. 3-1. Calculated strontium release vs diffusivity in BISO kernel at 1000°C

Fig. 3-2. Calculated cesium release vs TRISO failure

Fig. 3-3. Calculated strontium release vs TRISO failure

Fig. 3-4. Calculated cesium release vs BISO failure

Fig. 3-5. Calculated strontium release vs BISO failure

Fig. 3-6. Calculated cesium release vs PyC diffusivity at 1000°C

Fig. 3-7. Calculated cesium release vs diffusivity in graphite at 1000°C

Fig. 3-8. Calculated strontium release vs diffusivity in graphite at 1000°C

Fig. 3-9. Calculated cesium release vs diffusivity in helium

Fig. 3-10. Calculated strontium release vs diffusivity in helium

Fig. 3-11. Calculated cesium release vs matrix sorptivity

Fig. 3-12. Calculated strontium release vs matrix sorptivity

Fig. 3-13. Calculated cesium release vs graphite sorptivity

Fig. 3-14. Calculated strontium release vs graphite sorptivity

Fig. 3-15. Calculated cesium release vs activation energy for diffusivity in graphite

Fig. 3-16. Calculated strontium release vs activation energy for diffusivity in graphite

Fig. 3-17. Calculated cesium release vs activation energy for diffusivity in PyC

Fig. 3-18. Calculated strontium release vs activation energy for diffusion in BISO kernels

Fig. 3-19. Calculated cesium release vs core thermal power

Fig. 3-20. Calculated strontium release vs core thermal power

Fig. 3-21. Calculated cesium release vs total helium flow rate

Fig. 3-22. Calculated strontium release vs total helium flow rate

Fig. 3-23. Calculated cesium release vs convective heat transfer coefficient

Fig. 3-24. Calculated strontium release vs convective heat transfer coefficient

Fig. 3-25. Calculated cesium release vs graphite thermal conductivity

Fig. 3-26. Calculated strontium release vs graphite thermal conductivity

Fig. 3-27. Calculated cesium release vs fuel rod graphite gap conductance

Fig. 3-28. Calculated strontium release vs fuel rod graphite gap conductance

Fig. 3-29. Calculated cesium release vs fuel rod thermal conductivity

Fig. 3-30. Calculated strontium release vs fuel rod thermal conductivity

Fig. 3-31. Calculated cesium release vs core inlet helium temperature

Fig. 3-32. Calculated strontium release vs core inlet helium temperature

Fig. 3-33. Calculated cesium release vs region mismatch temperature

Fig. 3-34. Calculated strontium release vs region mismatch temperature

Fig. 3-35. Calculated cesium release vs trapped fraction in kernel

Fig. 3-36. Calculated cesium release vs contribution of thermal diffusion

Fig. 3-37. Calculated strontium release vs contribution of thermal diffusion

TABLE 3-1
FORTY-YEAR DIRECT METALLIC RELEASE INVENTORY (Ci)

	GASSAR (Ref. 3-3) (a)		AIPA Calculated		
	Expected	Design	Nominal	Median	Upper 95%
Sr-90	4,100	15,400	200	256	5,914
Cs-137	16,200	159,000	34,273	40,095	1,163,924

(a) Including contributions from gaseous precursors.

TABLE 3-2
SUMMARY OF TERMS IN PROPAGATION OF ERRORS FORMULA^(a)

i	Variable	$\left(\frac{\partial R_f}{\partial x_i}\right)^2 \sigma_{x_i}^2$	
		Cesium-137	Strontium-90
1.	TRISO failure fraction	(19) 0.00 ^(b)	(9) 2.48×10^{-3}
2.	BISO failure fraction	(16) 2.941×10^{-4}	0.00
3.	Diffusivity in particles	(2) 2.25×10^{-1}	0.00
4.	Diffusivity in graphite	(1) 4.26×10^{-1}	(5) 2.57×10^{-2}
5.	Diffusivity in helium	(17) 9.28×10^{-5}	(4) 2.771×10^{-2}
6.	Matrix sorptivity	(13) 8.034×10^{-4}	(13) 1.624×10^{-4}
7.	Graphite sorptivity	(18) 3.99×10^{-5}	(7) 2.245×10^{-2}
8.	Q-graphite	(11) 2.314×10^{-3}	(10) 1.219×10^{-3}
9.	Q-particles	(4) 4.21×10^{-2}	0.00
10.	Core power	(14) 3.477×10^{-4}	(11) 8.53×10^{-4}
11.	Flow rate	(6) 3.26×10^{-2}	(1) 3.08×10^{-1}
12.	Convective heat transfer	(8) 1.157×10^{-2}	(5) 2.473×10^{-2}
13.	Graphite conductivity	(12) 1.21×10^{-3}	(12) 1.81×10^{-4}
14.	Gap conductance	(7) 3.115×10^{-2}	0.00
15.	Fuel rod conductivity	(9) 5.914×10^{-3}	0.00
16.	Inlet helium temperature	(10) 4.42×10^{-3}	(6) 2.54×10^{-2}
17.	Region mismatch temperature	(5) 3.686×10^{-2}	(2) 1.964×10^{-1}
18.	Trapped fraction	(3) 5.69×10^{-2}	--
19.	Thermal diffusion contribution	(15) 3.23×10^{-4}	(3) 4.15×10^{-2}
	Correlation terms		
	$x_3 - x_9$	(21) -8.79×10^{-2}	0.00
	$x_4 - x_8$	(20) -4.62×10^{-3}	(8) 7.12×10^{-3}
	Total = $\sigma_{R_f}^2$	0.786	0.683

TABLE 3-2 (Continued)

(a)

$$\sigma_{R_f}^2 = \sum_{i=1}^{19} \left(\frac{\partial R_f}{\partial x_i} \right)^2 \sigma_{x_i}^2 + 2 \sum_{i=1}^{18} \sum_{j>i}^{19} \rho_{x_i x_j} \left(\frac{\partial R_f}{\partial x_i} \right) \left(\frac{\partial R_f}{\partial x_j} \right) \sigma_{x_i} \sigma_{x_j}$$

(b) The numbers in parentheses represent the ranking of each variable in order of importance of its contribution to the total variance of R_f for each isotope.

TABLE 3-3
VARIABLES EXAMINED IN AIPA STUDIES OF METALLIC FISSION PRODUCT RELEASE

1. TRISO failure fraction
2. BISO failure fraction
3. Diffusivity in particles at 1000°C
4. Diffusivity in graphite at 1000°C
5. Diffusivity in helium
6. Matrix sorptivity
7. Graphite sorptivity
8. Activation energy for diffusivity in graphite
9. Activation energy for diffusivity in patricles
10. Core thermal power
11. Total coolant flow rate
12. Convective heat transfer coefficient
13. Graphite thermal conductivity
14. Fuel rod/graphite gap conductance
15. Fuel rod thermal conductivity
16. Core inlet helium temperature
17. Region mismatch temperature
18. Trapped fraction in kernels (Cs only)
19. Thermal diffusion effects

TABLE 3-4
SUMMARY OF CESIUM RELEASE SENSITIVITIES

i	Variable	Range for Calculation of Partial Derivative	$\frac{\partial R_f}{\partial x_i}$
1.	TRISO failure	$-0.5 < x_1 < 0.5$	0.00
2.	BISO failure	$-1 < x_2 < 1$	0.101
3.	Diffusivity in PyC coatings	$-1 < x_3 < 1$	0.739
4.	Diffusivity in graphite	$-1 < x_4 < 1$	0.988
5.	Diffusivity in helium	$-1 < x_5 < 1$	0.032
6.	Matrix sorptivity	$-1 < x_6 < 1$	-0.115
7.	Graphite sorptivity	$-1 < x_7 < 1$	0.054
8.	Q-graphite	$-0.079 < x_8 < 0.079$	-1.266
9.	Q-particles	$-0.079 < x_9 < 0.079$	6.09
10.	Core power	$-0.021 < x_{10} < 0.021$	6.156
11.	Flow rate	$-0.097 < x_{11} < 0.097$	-12.04
12.	Convective heat transfer	$-0.41 < x_{12} < 0.41$	-5.12
13.	Graphite conductivity	$-0.097 < x_{13} < 0.097$	-0.84
14.	Gap conductance	$-0.097 < x_{14} < 0.097$	-1.82
15.	Fuel rod conductivity	$-0.477 < x_{15} < 0.477$	-1.267
16.	Inlet helium temperature	$-0.021 < x_{16} < 0.021$	8.88
17.	Region mismatch temperature ^(a)	$-30 < x_{17} < 30$	6.65×10^{-3}
18.	Trapped fraction ^(a)	$0.25 < x_{18} < 0.75$	-0.954
19.	Thermal diffusion ^(a)	$0 < x_{19} < 1$	0.03

^(a) Plotted on semilog scale; direct slope comparisons invalid,

TABLE 3-5
SUMMARY OF STRONTIUM RELEASE SENSITIVITIES

i	Variable	Range for Calculation of Partial Derivative	$\frac{\partial R_f}{\partial x_i}$
1.	TRISO failure	$-0.3 < x_1 < 0.3$	0.183
2.	BISO failure	$-1 < x_2 < 1$	0.00
3.	Diffusivity in kernel	$-2 < x_3 < 2$	0.00
4.	Diffusivity in graphite	$-0.447 < x_4 < 0.447$	1.303
5.	Diffusivity in helium	$-1 < x_5 < 1$	0.553
6.	Matrix sorptivity	$-1 < x_6 < 1$	-0.052
7.	Graphite sorptivity	$-1 < x_7 < 1$	-0.925
8.	Q-graphite	$-0.079 < x_8 < 0.079$	-1.499
9.	Q-particles	$-0.079 < x_9 < 0.079$	0.00
10.	Core power	$-0.021 < x_{10} < 0.021$	9.6415
11.	Flow rate	$-0.015 < x_{11} < 0.015$	-37.00
12.	Convective heat transfer	$-0.041 < x_{12} < 0.041$	-7.488
13.	Graphite conductivity	$-0.097 < x_{13} < 0.097$	-0.325
14.	Gap conductance	$-0.097 < x_{14} < 0.097$	0.00
15.	Fuel rod conductivity	$-0.477 < x_{15} < 0.477$	0.00
16.	Inlet helium temperature	$-0.021 < x_{16} < 0.021$	21.29
17.	Region mismatch temperature ^(a)	$-30 < x_{17} < 30$	1.535×10^{-2}
18.	Trapped fraction ^(a)	--	--
19.	Thermal diffusion ^(a)	$0 < x_{19} < 1$	0.337

(a) Plotted on semilog scale; direct slope comparisons invalid.

TABLE 3-6
SUMMARY OF VARIABLE UNCERTAINTIES, METALLIC RELEASE

i	Variable	σ_{x_i}	Ref.
1.	TRISO failure	0.271	3-9
2.	BISO failure	0.170	3-9
3.	Diffusivity in particles		
	Cesium in PyC	0.644	(a)
	Strontium in kernels	1.444	(a)
4.	Diffusivity in graphite		
	Cesium-137	0.661	(a)
	Strontium-90	0.123	(a)
5.	Diffusivity in helium	0.301	(b)
6.	Matrix sorptivity		
	Cesium-137	0.246	(a)
	Strontium-90	0.243	(a)
7.	Graphite sorptivity		
	Cesium-137	0.117	(a)
	Strontium-90	0.162	(a)
8.	Q-graphite		
	Cesium-137	0.038	(a)
	Strontium-90	0.0233	(a)
9.	Q-particles		
	Cesium in PyC	0.0337	(a)
	Strontium in kernels	0.0816	(a)
10.	Core power	3.029×10^{-3}	3-10
11.	Flow rate	0.015	3-10
12.	Convective heat transfer	0.021	3-11
13.	Graphite conductivity	0.041	3-12
14.	Gap conductance	0.097	3-11
15.	Fuel rod conductivity	0.061	3-13
16.	Inlet helium temperature	7.49×10^{-3}	3-10
17.	Region mismatch temperature ^(c)	28.87°F	(d)
18.	Trapped fraction ^(c)	0.25	(a)
19.	Thermal diffusion ^(c)	0.606	(e)

TABLE 3-6 (Continued)

- (a) See Section 6.
- (b) The σ value represents a conservative best guess based on limited unpublished data.
- (c) Uncertainty given over linear scale rather than logarithmic scale.
- (d) The standard deviation for a rectangular distribution of $\pm 50^\circ\text{F}$ about a nominal of 0°F is given by $\sigma = 50/\sqrt{3} = 28.87^\circ\text{F}$.
- (e) The σ value represents a best guess based on available data.

TABLE 3-7
RANDOM UNCERTAINTY CALCULATIONS, METALLIC RELEASE^(a)

i	Variable	$\sigma_{x_{ir}}$	$\left(\frac{\partial R_f}{\partial x_i}\right)^2 \sigma_{x_{ir}}^2$	
			Cesium	Strontium
10.	Core power	0.0314	3.74×10^{-2}	9.17×10^{-2}
14.	Gap conductance	0.083	2.28×10^{-2}	0.00

(a) The calculations are performed as follows:

$$R_m = \hat{R} \exp \left[\frac{5.3}{2} \sum_i \left(\frac{\partial R_f}{\partial x_i} \right)^2 \sigma_{x_{ir}}^2 \right] .$$

Cesium:

$$\begin{aligned} R_m &= \hat{R} \exp \left[\frac{5.3}{2} (0.0602) \right] , \\ &= 1.17 \hat{R} . \end{aligned}$$

Strontium:

$$\begin{aligned} R_m &= \hat{R} \exp \left[\frac{5.3}{2} (0.0917) \right] , \\ &= 1.275 \hat{R} . \end{aligned}$$

4. CIRCULATING, TOTAL PLATEOUT, AND FUEL BODY INVENTORIES

In this section, calculations of circulating, total plateout, and fuel body inventories are described. The results of the median direct release calculations for strontium-90 and cesium-137 discussed in Section 3 were used as input to calculations of total plateout inventory and uncertainties. Again, the method employed was essentially that described in Section 2, with exceptions as noted in the following sections.

The nominal values of the input parameters used in this section were not in all cases obtained from Section 7. Some nominal values were obtained from Ref. 4-1. These circumstances were imposed by time requirements during the AIPA study. The effect of these changes on the calculated nominal inventories was to reduce the inventory of xenon and xenon-like nuclides by about 20%. The uncertainties in the inventories were not substantially affected.

The final results of the calculations of circulating and total plateout inventories are summarized in Tables 4-1 through 4-5. The results have shown the GASSAR activity level criteria to be conservatively chosen. For most of the isotopes considered, the 95% confidence activities were distributed about the GASSAR expected activity levels rather than the design levels. The 95% confidence activities are considerably less than the GASSAR design level activity criteria.

Four major sources of uncertainty have been identified and should be given further consideration. First, under the nominal conditions used in these calculations, it was found that the contamination in the as-manufactured fuel is the source for slightly more than half the activity in the circulating coolant. Second, plateout characteristics have a strong

influence on the circulating activity uncertainty of condensible nuclides. This comparatively large uncertainty is the result of limited data on the plateout per pass for condensible nuclides. Third, uncertainties caused by assumptions in the RAD code and its use should be investigated and improved. Finally, more rigorous methods for analyzing the special nuclides discussed in Section 4.4 should be developed.

4.1. METHOD AND PROCEDURE

The methods described in Section 2 were applied with one modification: the circulating and plateout activities were calculated in two steps. The first step was to calculate core average release fractions ($\overline{R/B}$). The release fractions were then used to calculate the circulating and plateout activities. Two computer codes were used in the analyses. The SURVEY* code was used to calculate core average release fractions, and the RAD code (Ref. 4-2) was used to calculate the circulating and plateout activities.

The SURVEY code accepts as input local power distribution history (radial and axial), particle failure correlations (Ref. 4-4), core thermal parameters, and fuel burnup data. From these data, the core temperature distribution history is calculated, and subsequently, the particle failure fractions for TRISO and BISO particles and the total core birth rate and release rate for selected isotopes are calculated. The code performs these calculations at designated local mesh points and summarizes the results by block, segment, and core for particle types and contamination.

The RAD code (Ref. 4-2) accepts as input basic core average operating information and specific data on nuclides of interest for calculating circulating, plateout, and fuel body activity. This code considers the effects of coolant purification, condensible nuclide plateout, and radioactive decay in the determination of these activities. Purification and plateout are always removal terms, and radioactive decay is considered

*The SURVEY code is an improved and expanded version of the TREVER code, which is discussed in Ref. 4-3.

as a source term for evaluating precursor decay and as a removal term when evaluating the nuclide of interest.

The assumptions made in the RAD code or in its use are listed below.

1. The core release fraction of a nuclide is linear with the square root of the decay constant.
2. All gaseous fission products contained in a coated particle are released to the primary coolant at the instant of fuel particle failure. (This assumption is very conservative for short-lived fission products.)
3. The release fraction is constant with burnup.
4. There is no attenuation in the release incurred due to transit through the fuel rod matrix or the graphite web of the fuel element.
5. The release fractions of related chalcogens, halogens, and noble gases are equal for the same half-life (e.g., selenium, bromine, and krypton; tellurium, iodine, and xenon).

These assumptions and others employed in estimating HTGR primary coolant fission product inventories are discussed in detail in Refs. 4-1 and 4-2. As discussed in Section 1, uncertainties resulting from these assumptions were not quantified.

The procedure followed was first to calculate the nominal core average release fraction. Using this value as input, the nominal circulating and plateout activities were calculated, and the uncertain variables were identified and classified as random or systematic. Sensitivity studies were then performed with the SURVEY and RAD codes to determine partial derivatives for use in Eqs. 2-7 and 2-12.

It should be noted that only krypton-85m and xenon-138 were analyzed in detail in the release fraction calculations owing to assumptions 1 and 5 above. These two groupings represent all major isotopes contributing to the circulating activity.

4.2. CORE AVERAGE RELEASE FRACTION: NOMINAL VALUE AND VARIANCE

Fifteen variables with potential uncertainties were considered in the calculation of the nominal core average release fractions of krypton-85m and xenon-138 and their variances; these are listed in Table 4-6. This uncertainty analysis does not include contributions arising from any variables other than those listed or from calculational model uncertainties. Justifiable assessments of these indeterminate effects cannot be made at this time. In this study, only uncertainties in the basic input parameters used in the analysis were used to determine the uncertainty in fission product inventories.

4.2.1. Nominal Core Average Release Fraction

Calculations with SURVEY for core average release fractions were performed at the middle of the fourth, fifth, sixth, and seventh years of operation. The midyear release and birth rates were used because they adequately approximate the release fraction for annual averaged core release calculations. The release rate and birth rate for the two nuclides of interest were summed for each axial layer at each core depletion time point for release from TRISO, BISO, and contamination sources. The average release and birth rates for the equilibrium cycle were obtained by averaging over all layers and time points. From these rates, the steady-state release fractions for krypton-85m and xenon-138 were calculated: release rate was divided by birth rate. The results of the calculations are shown in Table 4-7, which includes the relative distribution of the release fraction among TRISO, BISO, and contamination sources. Especially noteworthy is the large relative contribution of contamination sources to the release fraction for krypton-85m and xenon-138.

4.2.2. Variance of Core Average Release Fraction

To obtain the variance of the core average release fraction, the fifteen variables in Table 4-6 were independently varied in the SURVEY code over the range of $-1.65 \sigma_i$ to $+1.65 \sigma_i$, where σ_i is the standard deviation of $\ln(y_i/\hat{y}_i)$ (according to the notation of Section 3.1). The resulting sensitivity curves were used to graphically determine the derivative of the release fraction with respect to each variable.

To limit computing costs, the SURVEY sensitivity studies were done only for reload segment 2, layer 6. The reload segment 2 release fraction follows the time behavior of the core average release fraction most closely. Layer 6 is the layer of highest fission gas release and is therefore assumed to contribute the most to the uncertainty in the total core release.

The results of the core average release fraction sensitivity studies are shown in Figs. 4-1 through 4-12. Only the curves for those variables which contribute to the variance of the release fraction are shown. The standard deviations for each variable are shown in the figures and are summarized with references in Table 4-8. In addition, the value of the slope of the curve about the nominal point is given in each figure by the variable a .

The method used to obtain the standard deviations in Table 4-8 can be illustrated with the following example. Reference 3-13 gives the standard deviation of fuel rod thermal conductivity as 15% of the mean value. Hence, the standard deviation in Table 4-8 is given by $\sigma_{11} = \ln(1.15) = 0.14$.

The above method was not used in the case of TRISO kernel hydrolysis. Fuel kernel hydrolysis occurs primarily in low-burnup, failed TRISO particles, and hydrolysis affects only exposed carbide kernels. The oxide

BISO kernels and the intact TRISO particles, whose SiC coatings are impervious to moisture, are not affected.

The following relationships provide a correction for the effects of hydrolysis on release from failed TRISO particles:

$$R_T = (1 - f) F_T B_T (R/B)_{WO} + f F_T B_T (R/B)_{WH} \quad , \quad (4-1)$$

or

$$R_T = F_T B_T (R/B)_{WO} \left[1 + f \left(\frac{(R/B)_{WH}}{(R/B)_{WO}} - 1 \right) \right] = F_T B_T (R/B)_{WO} \mathcal{F} \quad , \quad (4-2)$$

where R_T = release rate from failed TRISO particles,
 f = fraction of failed TRISO that is hydrolyzed,
 F_T = fraction of TRISO failed,
 B_T = birth rate in TRISO,
 $(R/B)_{WO}$ = release fraction in failed TRISO not hydrolyzed,
 $(R/B)_{WH}$ = release fraction in hydrolyzed failed TRISO,
 $\mathcal{F}$ = factor by which release from TRISO is increased to account for hydrolysis (defined in Eq. 4-2).

The product $F_T B_T (R/B)_{WO}$ is the release from TRISO calculated by SURVEY.

It has been shown (Ref. 4-5) that the ratio of the release fraction for hydrolyzed failed fuel $(R/B)_{WH}$ to the release fraction for unhydrolyzed failed fuel $(R/B)_{WO}$ is 30 ± 15 based on data from unirradiated particles. This value is thought to be conservative when applied to reactor operation. The uncertainty results in $\sigma_{13} = 0.095$, as shown in Table 4-8. Uncertainty in the fraction of failed TRISO that is hydrolyzed, f , was not considered.

The contributions of each of the fifteen variables to the variance of the core average release fraction for krypton-85m and xenon-138 are given

in Table 4-9. The numbers in parentheses to the left of each column rank each term from the largest contributor to the smallest. As can be seen in the table, the uncertainty in R/B for as-manufactured fuel contributes to slightly less than half the variance in core average release fraction.

None of the fifteen variables governing core average release fraction were determined to be correlated. All uncertainties were classified as systematic since the core average release fraction is defined for the total core, and on that scale the random uncertainties are assumed to average out.

In summary, the following results were obtained for core average release fraction:

<u>Isotope</u>	<u>Nominal $\overline{R/B}$</u>	<u>$\sigma^2 \ln [(\overline{R/B}) / (\hat{R/B})]$</u>
Krypton-85m	2.78×10^{-5}	0.087
Xenon-138	3.07×10^{-6}	0.120

Confidence statements can be made on core average release fraction by using equations of the form of Eq. 3-10.

4.3. CIRCULATING AND PLATEOUT ACTIVITIES: NOMINAL VALUES AND VARIANCE

4.3.1. Nominal Circulating Activities

Nominal circulating and plateout activities were calculated using the RAD code. The method was essentially that described in Ref. 4-1. Seven variables with potential uncertainties were considered in the calculations; these are listed in Table 4-10 along with the nominal values assumed for the calculations and the uncertainty for each variable. Three of the RAD input variables listed in Table 4-10 require further explanation. Minimum R/B is the minimum release fraction employed in the release calculations.

This parameter adds conservatism since it is used for all isotopes with release fractions less than the stated value and therefore increases their release.

The plateout removal rate constant translates the plateout effect into a form which can be handled exponentially. It is defined by (Ref. 4-1):

$$r_p = - \frac{m_c}{M} \ln (1 - \tau) \quad , \quad (4-3)$$

where r_p = plateout removal rate constant (sec^{-1}),
 m_c = primary coolant flow rate (lb/sec),
 M = mass of total circulating helium (lb),
 τ = fractional plateout per pass.

The purification removal rate constant is defined by (Ref. 4-1):

$$r_s = \frac{m_s}{M} \quad , \quad (4-4)$$

where r_s = purification removal rate constant (sec^{-1}),
 m_s = purification system flow rate (lb/sec),
 M = mass of total circulating helium (lb).

The influence of these parameters is handled by RAD through the solution of the following expressions for the circulating activity and the plateout activity (Ref. 4-1):

$$A_{i_{\text{cir}}} = \frac{\frac{3.15 \times 10^{10} \text{ fission}}{\text{W-sec}}}{\frac{3.7 \times 10^{10} \text{ disintegrations}}{\text{sec-Ci}}} \frac{(\overline{R/B})_i \text{ PY}_i}{1 + \frac{r_s + r_p}{\lambda_i}} \left[1 - e^{-(\lambda_i + r_s + r_p)t} \right] \quad , (4-5)$$

where $A_{i_{\text{cir}}}$ = circulating activity (Ci) of isotope i ,
 $(\overline{R/B})_i$ = core average release fraction of isotope i ,

P = reactor thermal power (W),
 Y_i = cumulative fission yield of isotope i ,
 λ_i = radioactive decay constant (sec^{-1}) of isotope i .

$$A_{i\text{plt}} = \frac{3.15 \times 10^{10} \frac{\text{fission}}{\text{W-sec}}}{3.7 \times 10^{10} \frac{\text{disintegrations}}{\text{sec-Ci}}} (\overline{R/B})_i PY_i (1 - e^{-\lambda_i t}) \quad , \quad (4-6)$$

where $A_{i\text{plt}}$ = plateout activity of isotope i .

Equation 4-6 has the added conservatism that all condensible fission products which are released are assumed to plate out.

Nominal values for the plateout removal rate constant and the purification removal rate constant are calculated from the nominal parameter values listed in Table 4-10. The results of the nominal calculations of circulating activity are given in Table 4-1 and are compared with GASSAR criteria limits. The calculations for additional isotopes which required special treatment are described in Section 4.4.

4.3.2. Variance of Circulating Activity

Uncertainty in circulating activity was determined from uncertainties in the variables shown in Table 4-10. The uncertainty in these variables is systematic, and no correlation terms were considered in the propagation of errors formula, Eq. 2-12.

The uncertainty results for each nuclide listed in Table 4-1 are given in Table 4-11, which shows the variance of the activity and the contribution of each uncertainty to the variance. Contributions to the variance are calculated for core average R/B , minimum R/B , purification removal rate, and plateout removal rate. The results of the sensitivity

studies performed are shown in Figs. 4-13 through 4-17. A more detailed discussion of the methods and assumptions used to calculate each contribution follows.

The release characteristics of most of the significant circulating nuclides are assumed to be like those of krypton or xenon. The nuclides in the krypton group are the other isotopes of krypton as well as selenium and bromine. The xenon group consists of the other isotopes of xenon as well as the isotopes of tellurium and iodine. For short-lived isotopes, the release fractions of the nuclides in a group have been shown to vary with their decay constants according to (Ref. 4-7)

$$R/B_{i,j} \cong 3 \sqrt{\frac{\xi_i}{\lambda_{i,j}}} , \quad (4-7)$$

where $R/B_{i,j}$ = release fraction of isotope j of element i ,
 ξ_i = diffusion parameter for element i (sec^{-1}),
 $\lambda_{i,j}$ = decay constant for isotope j of element i .

Since the release fractions for krypton-85m and xenon-138 have been calculated and the decay constant for these nuclides is known, a representative diffusion parameter for each group can be determined. From this, the various release fractions within a group can be evaluated.

The krypton-85m and xenon-138 release fraction uncertainties are assumed to be representative for their groups. On this basis, the uncertainty in circulating activity due to core average release fraction was calculated for the two groups (Fig. 4-13). The variance contribution to the circulating activity uncertainty is 0.065 for the krypton-85m group and 0.079 for the xenon-138 group.

Based on HTGR fuel specification limits on heavy metal contamination, it is conservatively assumed that the minimum release fraction is 1×10^{-7}

(see Section 7). Some of the shorter-lived nuclides have lower release fractions according to Eq. 4-7, but in this situation their values are conservatively set equal to the minimum value. The sensitivity of the circulating activity to this variable is illustrated in Fig. 4-14. For nuclides whose release fraction is below the lower uncertainty limit (e.g., strontium-91), the minimum release fraction is always used and the curve is linear. There are some nuclides whose calculated release fraction falls within the range of the uncertainty on the minimum release fraction (e.g., cesium-140 and cesium-138). These nuclides show a linear dependence on the minimum release fraction up to the point where the calculated value equals the minimum value. At this point, the minimum release fraction ceases to influence the result. Nuclides not listed in Table 4-12 or shown in Fig. 4-14 are not affected by the minimum release fraction.

The purification system removal rate constant is defined in Eq. 4-4. The uncertainty in this variable is determined by performing the required differentiation on Eq. 4-4 to obtain the variance formula

$$\sigma_{r_s}^2 = \left(\frac{1}{M}\right)^2 \sigma_{m_s}^2 + \left(\frac{\frac{m_s}{M^2}}{\frac{m_s}{M^2}}\right)^2 \sigma_M^2 \quad (4-8)$$

(See Eq. 4-4 for definition of terms.)

All terms in Eq. 4-8 are given in Table 4-10, and therefore the variance in the purification system removal rate constant can be evaluated.

From this information, the circulating activity uncertainty due to uncertainty in r_s can be calculated. The results of these calculations are illustrated in Fig. 4-15. From the curves, it can be seen that an increase in purification decreases the circulating activity. The degree to which the circulating activity is affected is determined by the relative magnitudes of the decay constant of the various nuclides and the purification removal rate constant. The purification system has little if

any influence on nuclides with decay constants substantially larger than the purification removal rate constant ($\lambda \gg r_s$). Table 4-13 contains a summary of the results of the r_s uncertainty study.

Plateout removes nuclides from the circulating coolant and acts like a decay constant in this study. Owing to the length of the time intervals evaluated (40-yr plant life), the exponential term in Eq. 4-5 is approximately equal to 1, and the circulating activity is controlled by the term

$$\frac{\lambda_i}{r_p + r_s} \quad (4-9)$$

The resulting relation is illustrated in Figs. 4-16 and 4-17. For reasons of mathematical convenience, the circulating activity has been plotted as a transformed function of the plateout per pass (τ). The transformation is necessary to obtain a log normal distribution. As the plateout per pass is reduced, the circulating activity increases. Owing to the behavior of the above term, the slope becomes very steep as τ approaches zero.

The plateout per pass for bromine and iodine after 40 yr of plant operation has a nominal value of 0.35, which leads to a plateout variance term of 0.96 for these elements. The remaining elements evaluated (strontium, yttrium, zirconium, niobium, antimony, tellurium, and cesium) have a nominal plateout per pass of 0.90, which leads to a plateout variance term of 0.38.

4.3.3. Plateout Activity

As previously noted, the RAD code assumes that for plateout calculations all condensible fission products released from the core are plated out. Hence, only the uncertainties in core average release fraction and minimum release fraction are evaluated. The treatment is the same as

for circulating activity, so only the results are reported. A summary of these results is presented in Table 4-2, which includes median and upper 95% confidence levels on plateout activity and GASSAR criteria.

A substantial fraction of the plateout activity consists of tellurium and iodine isotopes, which are assumed to have release fractions equal to the release fraction of xenon isotopes. From Eq. 4-6, one would expect to see a linear dependence of plateout activity on the core average release fraction, as was seen in the circulating activity study. This expectation is verified by the results illustrated in Fig. 4-18. The contribution to the plateout variance from the release fraction uncertainty is 0.079 for all tellurium and iodine isotopes listed in Table 4-2.

The minimum release fraction uncertainty influences all the nuclides in Table 4-2, but is most strongly influences strontium-91, zirconium-95, and niobium-95. The plateout of zirconium-95 and niobium-95 depends directly on the minimum release fraction. The strontium behavior is irregular since it has a precursor, krypton-91, whose release fraction falls within the minimum release fraction uncertainty range. The strontium-91 is therefore only affected over a part of the range of the minimum release fraction. The remaining nuclides are only insignificantly affected, as shown in Fig. 4-19 and Table 4-14.

4.4. UNCERTAINTY FOR NUCLIDES REQUIRING SPECIAL ANALYSIS

Several nuclides are not directly treated by the methods discussed thus far. This section discusses the techniques used to evaluate these special nuclides, which are listed in Tables 4-3 and 4-4.

4.4.1. Uncertainty in Krypton-85 Activity

The release fractions used in the preceding calculations are equilibrium values which are only applicable for short-lived nuclides. For

long-lived and stable nuclides, the following expression is appropriate (Ref. 4-9):

$$R/B_i \cong 6 \sqrt{\frac{\xi_i t}{\pi}} - 3\xi_i t \quad , \quad (4-10)$$

where ξ_i = diffusion parameter for element i (sec^{-1}),
 t = irradiation time (sec).

Using this expression, the krypton-85 release fraction is calculated to be 2.20×10^{-3} for a nominal krypton diffusion parameter of 4.18×10^{-15} for a 4-yr irradiation period. The variance σ^2 of this result is taken to be equal to that for the other krypton isotopes evaluated, i.e., 0.065.

As demonstrated earlier for other krypton isotopes, the uncertainty in the circulating inventory is governed solely by the release fraction uncertainty (Table 4-11), so the above variance is also the total krypton-85 variance.

4.4.2. Uncertainty in Silver-110m Activity

In these calculations, the release of silver-110m from the core was taken from experimental measurements of the release rather than mechanistic calculations. The nominal silver-110m circulating and plateout activity is calculated using a core release fraction of 0.06. This is the experimentally observed silver-110m release fraction in the Peach Bottom reactor. The 95% confidence level for silver-110m activity is arbitrarily defined to be that corresponding to a 100% release from BISO particles. This corresponds to a total core release fraction of 0.60. Based on these definitions, the silver-110m release variance is given by $\sigma_{\ln(A/\hat{A})}^2 = 0.90$.

4.4.3. Uncertainty in Cesium- and Strontium-Like Nuclides

The circulating and plateout activity of an isotope is derived from direct yield from fission and precursor decay. The directly released activities of strontium-90 and cesium/barium-137m and their variances are evaluated in Section 3. The release fractions for the precursors of these nuclides, krypton-90 and xenon-137, are discussed in Section 4.3. In RAD, the direct and indirect release of strontium-90 and cesium/barium-137m are calculated to determine the total circulating and plateout activities.

When analyzing the uncertainty in these results, the source of the activity must be considered. For strontium-90 and cesium/barium-137m, the source was found to be predominately direct release, so the variances determined in the metallic study of Section 3 are used; i.e., strontium-90 $\sigma_{\ln(A/\hat{A})}^2 = 3.62$, and cesium/barium-137m $\sigma_{\ln(A/\hat{A})}^2 = 4.17$.

A detailed study of the direct release of cesium-134, cesium-140, and cerium/praseodymium-144 has not been done, but these isotopes are assumed to have the same direct release fraction and uncertainty as cesium/barium-137m owing to similar chemical behavior. All the cesium-134 and cerium/praseodymium-144 comes from direct release, so they are assumed to have the same uncertainty as cesium/barium-137m [$\sigma_{\ln(A/\hat{A})}^2 = 4.17$].

The cesium-140 source is only 63.8% direct release as calculated by the RAD code. The remainder is from xenon-140 precursor decay. The uncertainty in cesium-140 activity must therefore be determined from the direct and indirect sources. It is assumed that the sum of the 95% confidence activity from direct and indirect sources is the 95% confidence value for the total. This yields a variance [$\sigma_{\ln(A/\hat{A})}^2$] of 2.23 for cesium-140, which is greater than the variance of 0.179 for xenon-140 but less than the variance of 4.17 for directly released cesium-140.

In a similar manner, barium/lanthanum-140 is primarily derived from direct release, and it behaves chemically like strontium-90. Hence, it is

assumed that its uncertainty is the same as that of strontium-90; i.e.,
 $\sigma_{\ln(A/\hat{A})}^2 = 3.62$.

The isotopes of rubidium also come from direct and indirect sources. The rubidium from direct release behaves chemically like cesium, and the rubidium from indirect release acts like its krypton precursor. The 95% confidence levels associated with the rubidium isotopes are therefore calculated in the same manner as those for cesium-140. The results of these calculations are summarized in Tables 4-3 and 4-4. Similar calculations have been performed for circulating and plateout activity.

4.4.4. Uncertainty in Tritium Release

Tritium release is defined in terms of Peach Bottom operating experience. The nominal value of the source of tritium is taken to be the tritium derived from the neutron activation of helium-3 in the coolant observed during Peach Bottom operation. For commercial HTGRs, the coolant level of helium-3 is specified to be a fraction of 10^{-7} , as observed in the Peach Bottom reactor. The nominal value plus the tritium from 100% activation of tritium producing H-451 graphite impurities is defined to be the value at $+1\sigma$. From these assumptions, the tritium variance of $\sigma_{\ln(A/\hat{A})}^2 = 0.48$ was determined.

4.5. FUEL BODY INVENTORY

The total fuel body activity is given by the following expression:

$$A_i = \frac{3.15 \times 10^{10} \frac{\text{fissions}}{\text{W-sec}}}{3.7 \times 10^{10} \frac{\text{disintegrations}}{\text{sec-Ci}}} P Y_i [1 - (R/B)_i] (1 - e^{-\lambda_i t}) \quad , \quad (4-11)$$

where A_i = total core activity in the fuel blocks (Ci),
 P = thermal power of the reactor (W),

Y_i = cumulative fission yield of isotope i,
 $(R/B)_i$ = core average release fraction for isotope i,
 λ_i = decay constant for isotope i (sec^{-1}),
 t = irradiation time (sec).

The activity calculated from this relation is the total core inventory. The average fuel element inventory is obtained by dividing this result by the total number of fuel blocks in the system (3944). This relation, with precursor decay accounted for, is contained in the RAD code. The nominal fuel body activity was calculated using the nominal values of the variables listed in Table 4-10.

In the AIPA study, the fission yields and decay constants were assumed to be well known. Therefore, for the average fuel element inventory, the uncertainty is solely due to uncertainty in the total reactor thermal power level and the release fraction of the nuclides involved. Since the release fraction for all nuclides except silver-110m is sufficiently small so that the term $(1 - R/B)$ does not significantly change, the uncertainty in the average fuel element inventory is essentially due to the total reactor thermal power uncertainty. The fuel element inventory of silver-110m is controlled by its release fraction since the release fraction uncertainty is large compared with the power uncertainty.

The thermal power probability distribution is normal, and consequently the fuel element activity probability distribution is normal. There is a linear relation between fuel element activities and thermal power, and the remaining variables can be reasonably approximated by constants. The standard deviation in total core thermal power is 0.70% of the nominal value (Ref. 3-10). This is the standard deviation in average fuel body activity for all isotopes considered except the following:

Krypton-85	$\sigma = 0.90\%$	} of the nominal values.
Silver-110m	$\sigma = 3.0\%$	

The results of the fuel body inventory analysis are summarized in Table 4-5.

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Fig. 4-1. Core average release fraction vs TRISO failure fraction

Fig. 4-2. Core average release fraction vs BISO failure fraction

Fig. 4-3. Core average release fraction vs R/B failed fuel

Fig. 4-4. Core average release fraction vs R/B as manufactured

Fig. 4-5. Core average release fraction vs R/B activation energy

Fig. 4-6. Core average release fraction vs core thermal power

Fig. 4-7. Core average release fraction vs total core coolant flow

Fig. 4-8. Core average release fraction vs core inlet helium temperature

Fig. 4-9. Core average release fraction vs region mismatch temperature

Fig. 4-10. Core average release fraction vs fuel rod thermal conductivity

Fig. 4-11. Core average release fraction vs graphite thermal conductivity

Fig. 4-12. Core average release fraction vs TRISO kernel hydrolysis

Fig. 4-13. Circulating activity as a function of core average R/B

Fig. 4-14. Activity as a function of R/B_{min}

Fig. 4-15. Activity as a function of purification rate

Fig. 4-16. Activity as a function of plateout rate

Fig. 4-17. Activity as a function of plateout rate

Fig. 4-18. Plateout as a function of core average R/B

Fig. 4-19. Plateout as a function of minimum R/B Zr-95, Nb-95

TABLE 4-1
AIPA RESULTS FOR CIRCULATING ACTIVITY (Ci)

Isotope	Median Activity (Ci)	Upper 95% Activity (Ci)	GASSAR Activity (Ci)	
			Expected	Design
Bromine-84	1.30	6.90	3.02	1310.0
Krypton-85m	776.0	1193.0	1810.0	1.8×10^4
Krypton-87	878.0	1350.0	2050.0	1.9×10^4
Krypton-88	1750.0	2690.0	4090.0	3.9×10^4
Krypton-89	397.0	610.0	931.0	9180.0
Krypton-90	151.0	232.0	354.0	4380.0
Krypton-91	39.8	64.4	93.4	1930.0
Strontium-91	5.7×10^{-4}	9.8×10^{-3}	5.2×10^{-3}	34.0
Strontium-94	0.203	3.50	1.05	1250.0
Yttrium-96	0.120	2.07	0.644	1050.0
Zirconium-95	3.7×10^{-6}	6.4×10^{-5}	2.1×10^{-5}	0.109
Niobium-95	6.8×10^{-6}	1.2×10^{-4}	3.8×10^{-5}	0.16
Antimony-133	0.0445	0.77	0.24	544.0
Tellurium-129m	4.6×10^{-4}	1.48×10^{-3}	3.8×10^{-3}	1.7
Tellurium-131m	7.24×10^{-3}	2.33×10^{-2}	0.06	27.0
Tellurium-132	0.029	0.093	0.246	85.0
Tellurium-133m	0.226	0.726	1.9	803.0
Tellurium-133	0.367	1.18	3.1	1010.0
Iodine-131	0.0729	0.39	0.113	41.0
Iodine-132	0.961	5.2	1.51	549.0
Iodine-133	0.445	2.4	0.698	258.0
Iodine-134	2.25	12.1	3.54	1430.0
Iodine-135	0.675	3.6	1.06	426.0
Iodine-136	4.70	25.3	7.40	1600.0
Xenon-131m	1.59	2.56	2.50	23.0
Xenon-133m	28.9	46.5	45.6	419.0
Xenon-133	596.0	959.0	939.0	8650.0
Xenon-135m	409.0	658.0	649.0	6700.0
Xenon-135	1010.0	1625.0	1590.0	1.6×10^4
Xenon-137	247.0	398.0	392.0	4790.0
Xenon-138	445.0	716.0	706.0	7910.0
Xenon-140	24.2	50.0	38.3	1520.0
Cesium-138	0.31	0.87	2.57	1200.0

TABLE 4-2
CONTRIBUTIONS TO VARIANCE IN PLATEOUT ACTIVITY

Isotope	Variance Due to		$\sigma_{\ln(A/\hat{A})}^2$ Total Variance	Median Activity (Ci)	Upper 95% Activity (Ci)	GASSAR Activity (Ci)	
	Core Average R/B	Minimum R/B				Expected	Design
Sr-91	0.0	0.52	0.52	67.2	221.0	122.0	4,950.0
Zr-95	0.0	2.6	2.6	31.8	455.0	31.0	3,470.0
Nb-95	0.0	2.6	2.6	48.0	687.0	47.0	5,210.0
Te-129m	0.120	0.0	0.120	1,050.0	1,670.0	1,620.0	1.85×10^5
Te-131m	0.120	8×10^{-6}	0.120	604.0	960.0	954.0	1.08×10^5
Te-132	0.120	8×10^{-6}	0.120	6,390.0	10,160.0	10,100.0	88,200.0
I-131	0.120	8×10^{-6}	0.120	8,220.0	13,070.0	12,900.0	1.2×10^5
I-132	0.120	8×10^{-6}	0.120	7,530.0	11,970.0	11,900.0	1.05×10^5
I-133	0.120	8×10^{-6}	0.120	5,580.0	8,870.0	8,820.0	84,000.0
I-135	0.120	0.0	0.120	2,350.0	3,740.0	3,720.0	38,000.0

TABLE 4-3
CIRCULATING ACTIVITY OF SPECIAL NUCLIDES

Nuclide	$\sigma_{\ln(A/\hat{A})}^2$ Circulating	Circulating (Ci)		GASSAR Activity (Ci)	
		Median	Upper 95%	Expected	Design
H-3	0.48	3.45	10.8	4.15	5.87
Rb-88	0.44	2.14	6.4	26.4	8360
Rb-89	0.44	0.58	1.7	7.07	2460
Rb-90m	4.17	0.01	0.3	0.056	149.0
Rb-90	1.06	1.32	7.2	15.1	3910
Rb-91	1.85	1.16	10.9	11.4	2820
Sr-90	3.62	6.81×10^{-7}	1.57×10^{-5}	5.50×10^{-5}	7.86×10^{-3}
Kr-85	0.065	1.10	1.68	2.93	35.0
Ag-110m	0.90	1.81×10^{-5}	8.66×10^{-5}	1.01×10^{-4}	4.01×10^{-2}
Cs-134	4.17	2.87×10^{-4}	8.3×10^{-3}	6.45×10^{-4}	0.243
Cs-140	2.23	0.754	8.9	5.13	2490
Cs/Ba-137m	4.17	$1.12 \times 10^{-4}/$ 5.65×10^{-3}	$3.2 \times 10^{-3}/$ 0.16	$2.25 \times 10^{-4}/$ 3.04×10^{-2}	$8.77 \times 10^{-2}/$ 46.3
Ba/La-140	3.62	$1.94 \times 10^{-5}/$ 1.42×10^{-4}	$4.48 \times 10^{-4}/$ 3.3×10^{-3}	$1.35 \times 10^{-4}/$ 7.9×10^{-4}	1.03/3.18
Ce/Pr-144	4.17	6.56×10^{-7} 1.2×10^{-2}	$1.95 \times 10^{-5}/$ 0.35	$3.65 \times 10^{-6}/$ 6.65×10^{-2}	$1.54 \times 10^{-2}/$ 223.0

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TABLE 4-4
PLATEOUT ACTIVITY OF SPECIAL NUCLIDES

Isotope	$\sigma_{\ln(A/\hat{A})}^2$ Plateout	Plateout (Ci)		GASSAR Activity (Ci)	
		Median	Upper 95%	Expected	Design
Rb-88	0.087	1,760	2,680	4,100	40,900
Rb-89	0.087	410	624	945	10,700
Rb-90m	4.17	2.0	58.0	2.13	289.0
Rb-90	0.73	169.0	691.0	372	6,380
Rb-91	1.65	52.9	441.0	107	3,420
Sr-90	3.62	346.0	7,989.0	4,100	15,400
Ag-110m	0.90	306.0	1,464.0	321	3,210
Cs-134	4.17	14,400.0	418,000	6,100	57,800
Cs-140	2.23	37.9	447	52.7	3,080
Cs/Ba-137m	4.17	4.02 x 10 ⁴ /	1.2 x 10 ⁶ /	16,200/	159,000/
		3.76 x 10 ⁴	1.1 x 10 ⁶	15,200	149,000
Ba/La-140	3.62	53.7/69.5	1,240/1,605	69.3/86.0	4,760/6,460
Ce/Pr-144	4.17	12.4/22.1	360./642.	13.1/23.2	1,290/2,460

TABLE 4-5
FUEL BODY ACTIVITY (Ci)
AVERAGE POWER ELEMENT AFTER FOUR YEARS OPERATION

Nuclide	Median Activity (Ci) (50% Confidence)	Upper Activity (Ci) (95% Confidence)	GASSAR Design Activity (Ci)
H-3	12.8	12.9	12.8
Br-84	8860.	8960.	1.13×10^4
Kr-85m	1.20×10^4	1.21×10^4	1.50×10^4
Kr-85	487.	494.	580.
Kr-87	1.81×10^4	1.83×10^4	2.10×10^4
Kr-88	3.07×10^4	3.10×10^4	3.73×10^4
Kr-89	3.16×10^4	3.20×10^4	3.43×10^4
Kr-90	2.89×10^4	2.93×10^4	3.28×10^4
Kr-91	1.48×10^4	1.50×10^4	2.21×10^4
Rb-88	3.10×10^4	3.14×10^4	3.77×10^4
Rb-89	3.43×10^4	3.47×10^4	3.82×10^4
Rb-90m	5150.	5210.	7330.
Rb-90	4.01×10^4	4.06×10^4	4.34×10^4
Rb-91	3.33×10^4	3.37×10^4	3.77×10^4
Sr-89	3.48×10^4	3.52×10^4	3.91×10^4
Sr-90	2990.	3024.	3250.
Sr-91	3.61×10^4	3.65×10^4	3.88×10^4
Sr-94	3.12×10^4	3.16×10^4	4.07×10^4
Y-91	3.63×10^4	3.67×10^4	3.92×10^4
Y-96	3.28×10^4	3.32×10^4	4.03×10^4
Zr-95	4.08×10^4	4.13×10^4	4.41×10^4
Nb-95	4.08×10^4	4.13×10^4	4.41×10^4
Ag-110m	1.17	1.23	0.521
Sb-133	1.42×10^4	1.44×10^4	2.20×10^4
Te-129m	1410.	1426.	1990.
Te-131m	4240.	4290.	5960.
Te-132	2.78×10^4	2.81×10^4	3.00×10^4
Te-133m	2.27×10^4	2.30×10^4	2.84×10^4
Te-133	1.79×10^4	1.81×10^4	1.90×10^4
I-131	2.03×10^4	2.05×10^4	2.29×10^4

TABLE 4-5 (Continued)

Nuclide	Median Activity (Ci) (50% Confidence)	Upper Activity (Ci) (95% Confidence)	GASSAR Design Activity (Ci)
I-132	2.91×10^4	2.94×10^4	3.22×10^4
I-133	4.06×10^4	4.11×10^4	4.59×10^4
I-134	4.20×10^4	4.25×10^4	4.87×10^4
I-135	3.50×10^4	3.54×10^4	4.34×10^4
I-136	1.56×10^4	1.58×10^4	2.37×10^4
Xe-131m	122.	123.	137.
Xe-133m	1150.	1165.	1280.
Xe-133	4.06×10^4	4.11×10^4	4.59×10^4
Xe-135m	6420.	6495.	7130.
Xe-135	4740.	4795.	5300.
Xe-137	3.82×10^4	3.86×10^4	4.03×10^4
Xe-138	3.71×10^4	3.75×10^4	4.24×10^4
Xe-140	1.51×10^4	1.53×10^4	2.60×10^4
Cs-134	5110.	5170.	5360.
Cs-138	4.21×10^4	4.26×10^4	4.58×10^4
Cs-140	3.48×10^4	3.52×10^4	3.96×10^4
Cs/Ba-137m	2940./2720.	2975./2750.	3150./2920.
Ba/La-140	4.01×10^4 /	4.06×10^4 /	4.28×10^4 /
	4.02×10^4	4.07×10^4	4.29×10^4
Ce/Pr-144	2.97×10^4 /	3.00×10^4 /	3.32×10^4 /
	2.98×10^4	3.01×10^4	3.32×10^4

TABLE 4-6
VARIABLES USED TO CALCULATE CORE AVERAGE RELEASE FRACTIONS

1. Kernel migration coefficient
2. TRISO pressure vessel failure
3. BISO pressure vessel failure
4. R/B, failed fuel
5. R/B, as-manufactured fuel
6. R/B activation energy
7. Core thermal power
8. Total coolant flow rate
9. Core inlet helium temperature
10. Region mismatch temperature
11. Fuel rod thermal conductivity
12. Graphite thermal conductivity
13. TRISO kernel hydrolysis
14. TRISO amoeba failure distance
15. BISO amoeba failure distance

TABLE 4-7
NOMINAL CORE AVERAGE GASEOUS RELEASE FRACTION

	Krypton-85m	Xenon-138
Nominal R/B	2.78×10^{-5}	3.07×10^{-6}
Relative contribution		
Fissile failure	0.37	0.53
Fertile failure	0.09	0.05
Contamination	0.54	0.42

TABLE 4-8
SUMMARY OF VARIABLE UNCERTAINTIES, CORE
AVERAGE RELEASE FRACTION

i	Variable	$\sigma_i^{(a)}$	Ref.
1.	Kernel migration coefficient	1.04	3-9
2.	TRISO pressure vessel failure	0.66	3-9
3.	BISO pressure vessel failure	0.61	3-9
4.	R/B, failed fuel ^(b)		
	Krypton-85m	0.50	(c)
	Xenon-138	0.50	(c)
5.	R/B, as-manufactured fuel		
	Krypton-85m	0.22	(c)
	Xenon-138	0.22	(c)
6.	R/B activation energy		
	Krypton-85m	0.193	(c)
	Xenon-138	0.55	(c)
7.	Core thermal power	0.007	3-10
8.	Total coolant flow rate	0.034	3-10
9.	Inlet helium temperature	0.017	3-10
10.	Region mismatch temperature (°F) ^(b)	28.9	Footnote (d), Table 3-6
11.	Fuel rod thermal conductivity	0.14	3-13
12.	Graphite thermal conductivity	0.095	3-12
13.	TRISO kernel hydrolysis	0.045	(d)
14.	TRISO amoeba failure distance (μm) ^(b)	16	3-9
15.	BISO amoeba failure distance (μm) ^(b)	16	3-9

(a) σ of $\ln(y_i/\hat{y}_i)$.

(b) Uncertainty given over linear scale.

(c) See Section 7.

(d) See Section 4.2.2.

TABLE 4-9
SUMMARY OF TERMS IN PROPAGATION OF ERRORS
FORMULA, CORE AVERAGE RELEASE FRACTION

i	Variable	Variance	
		Krypton-85m ^(a)	Xenon-138
1.	Kernel migration coefficient	0.00	0.00
2.	TRISO pressure vessel failure	(8) 6.0×10^{-4}	(6) 3.8×10^{-3}
3.	BISO pressure vessel failure	(3) 9.5×10^{-3}	(4) 9.5×10^{-3}
4.	R/B, failed fuel	(5) 6.4×10^{-3}	(5) 6.4×10^{-3}
5.	R/B, as manufactured fuel	(1) 3.4×10^{-2}	(2) 3.4×10^{-2}
6.	R/B activation energy	(4) 7.55×10^{-3}	(3) 2.2×10^{-2}
7.	Core thermal power	(10) 2.24×10^{-4}	(10) 1.04×10^{-4}
8.	Total coolant flow rate	(7) 2.78×10^{-3}	(8) 1.30×10^{-3}
9.	Inlet helium temperature	(9) 1.5×10^{-4}	(9) 7.3×10^{-4}
10.	Region mismatch temperature	(6) 3.0×10^{-3}	(7) 1.4×10^{-3}
11.	Fuel rod thermal conductivity	(11) 1.25×10^{-4}	(11) 7.06×10^{-5}
12.	Graphite thermal conductivity	(12) 1.09×10^{-4}	(12) 4.42×10^{-5}
13.	TRISO kernel hydrolysis	(2) 2.25×10^{-2}	(1) 4.11×10^{-2}
14.	TRISO amoeba failure distance	0.00	0.00
15.	BISO amoeba failure distance	0.00	0.00
	Total = $\sigma_{\ln}^2 [(R/B)/(\hat{R/B})]$	0.087	0.120

(a) The numbers in parentheses represent the ranking of each variable in order of importance of its contribution to the total variance of R/B for each isotope.

TABLE 4-10
VARIABLES USED TO CALCULATE CIRCULATING AND PLATEOUT
ACTIVITIES, NOMINAL VALUES AND UNCERTAINTIES

i	Variable	Nominal Value	Ref.	$\sigma_i^{(a)}$	Ref.
1.	Core average R/B				
	Krypton-85m	2.10×10^{-5}	Calculated	0.254	Calculated
	Xenon-138	3.07×10^{-6}	Calculated	0.281	Calculated
2.	Minimum R/B	1.0×10^{-7}	4-3	1.61	(b)
3.	Plateout removal rate constant				
	Bromine and iodine	0.1 sec^{-1}	Calculated	See text	Calculated
	Others	0.535 sec^{-1}	Calculated	See text	Calculated
4.	Purification removal rate constant	$4.27 \times 10^{-5} \text{ sec}^{-1}$	Calculated	0.058	Calculated
5.	Total circulating helium (M)	13,470 lb	4-6	672 lb	4-6
6.	Purification flow rate (r_s)	2070 lb/hr		51 lb	4-8
7.	Plateout per pass (τ)				
	Bromine and iodine	0.35	(c)	0.316	(c)
	Others	0.90	(c)	1.098	(c)

(a) σ of $\ln(y_i/\hat{y}_i)$.

(b) See Section 7.

(c) See Section 5.

TABLE 4-11
CONTRIBUTIONS TO VARIANCE IN CIRCULATING ACTIVITY

Isotope	Core Average R/B	Minimum R/B	Purification Removal Rate	Plateout Removal Rate	Total Variance $\sigma^2_{\ln(A/\hat{A})}$
Br-84	0.087	0.0	0.0	0.96	1.047
Kr-85m	0.087	0.0	2×10^{-4}	0.0	0.087
Kr-87	0.087	0.0	6×10^{-5}	0.0	0.087
Kr-88	0.087	0.0	1.8×10^{-4}	0.0	0.087
Kr-89	0.087	0.0	0.0	0.0	0.087
Kr-90	0.087	0.0	0.0	0.0	0.087
Kr-91	0.087	0.02	0.0	0.0	0.107
Sr-91	0.0	2.6	0.0	0.38	2.98
Sr-94	0.0	2.6	0.0	0.38	2.98
Y-96	0.0	2.6	0.0	0.38	2.98
Zr-95	0.0	2.6	0.0	0.38	2.98
Nb-95	0.0	2.6	0.0	0.38	2.98
Sb-133	0.0	2.6	0.0	0.38	2.98
Te-129m	0.120	0.0	0.0	0.38	0.50
Te-131m	0.120	0.0	0.0	0.38	0.50
Te-132	0.120	0.0	0.0	0.38	0.50
Te-133m	0.120	0.0	0.0	0.38	0.50
Te-133	0.120	0.0	0.0	0.38	0.50
I-131	0.120	0.0	0.0	0.96	1.08
I-132	0.120	0.0	0.0	0.96	1.08
I-133	0.120	0.0	0.0	0.96	1.08
I-134	0.120	0.0	0.0	0.96	1.08
I-135	0.120	0.0	0.0	0.96	1.08
I-136	0.120	0.0	0.0	0.96	1.08
Xe-131m	0.120	0.0	0.001	0.0	0.121
Xe-133m	0.120	2.6×10^{-6}	0.001	0.0	0.121
Xe-133	0.120	0.0	0.001	0.0	0.121
Xe-135m	0.120	0.0	4.0×10^{-6}	0.0	0.120
Xe-135	0.120	0.0	3.9×10^{-4}	0.0	0.120
Xe-137	0.120	0.0	0.0	0.0	0.120
Xe-138	0.120	0.0	4×10^{-6}	0.0	0.120
Xe-140	0.120	0.1	0.0	0.0	0.220
Cs-138	0.0	0.01	4×10^{-6}	0.38	0.39

TABLE 4-12
CIRCULATING ACTIVITY SENSITIVITY TO MINIMUM RELEASE FRACTION

Nuclide	$\frac{\partial \ln(A/\hat{A})}{\partial x_2}$ (a)	σ_{x_2}	$a^2 \sigma_{x_2}^2$
Kr-91	0.08	1.61	0.02
Sr-91	1.0	1.61	2.6
Sr-94	1.0	1.61	2.6
Y-96	1.0	1.61	2.6
Zr-95	1.0	1.61	2.6
Nb-95	1.0	1.61	2.6
Sb-133	1.0	1.61	2.6
Xe-131m	0.001	1.61	~0
Xe-140	0.196	1.61	0.1
Cs-138	0.068	1.61	0.01
Cs-140	0.44	1.61	0.51

TABLE 4-13
CIRCULATING ACTIVITY SENSITIVITY TO PURIFICATION
SYSTEM REMOVAL RATE CONSTANT

Nuclide	$\frac{\partial \ln(A/\hat{A})}{\partial x_3}$ (a)	σ_{x_3}	$a^2 \sigma_{x_3}^2$
Kr-85m	-0.48	0.058	2×10^{-4}
Kr-87	-0.23	0.058	6×10^{-5}
Kr-88	-0.40	0.058	1.8×10^{-4}
Xe-131m	-1.0	0.058	0.001
Xe-133m	-1.0	0.058	0.001
Xe-133	-1.0	0.058	0.001
Xe-135m	-0.06	0.058	4×10^{-6}
Xe-135	-0.58	0.058	3.9×10^{-4}
Xe-138	-0.06	0.058	3×10^{-6}
Cs-138	-0.06	0.058	3×10^{-6}

TABLE 4-14
 PLATEOUT ACTIVITY SENSITIVITY TO MINIMUM RELEASE FRACTION

Nuclide	$\frac{\partial \ln(A/\hat{A})}{\partial x_2}$ (a)	σ_{x_2}	$a^2 \sigma_{x_2}^2$
Sr-91	0.45	1.61	0.52
Zr-95	1.0	1.61	2.6
Nb-95	1.0	1.61	2.6
Te-129m	0	1.61	0
Te-131m	0.002	1.61	8×10^{-6}
Te-132	0.002	1.61	8×10^{-6}
I-131	0.002	1.61	8×10^{-6}
I-132	0.002	1.61	8×10^{-6}
I-133	0.002	1.61	8×10^{-6}
I-135	0	1.61	0

5. FISSION PRODUCT PLATEOUT DISTRIBUTIONS

This section discusses calculations of uncertainty in fission product plateout distributions. The specific activity (Ci/cm^2) at a given location in the primary coolant circuit is uncertain owing to uncertainty about the amount of activity released from the fuel and where it deposits or plates out. Sections 3 and 4 describe calculations of total core fission product release and the associated uncertainties. This section translates the uncertainties in total core release and the deposition process into confidence statements about the specific activity levels at local points throughout the primary circuit.

Nominal plateout distributions and uncertainties were calculated for strontium-90, silver-110m, tellurium-131m, iodine-131, cesium-137, lanthanum-140, and cerium-144. The method employed was essentially that described in Section 2, with exceptions discussed in the following sections.

The results of the analysis of fission product plateout distributions are summarized in Tables 5-1 through 5-7 and in Figs. 5-1 through 5-7. As illustrated in these tables and figures, a large uncertainty exists regarding the specific plateout activity at local points in the primary circuit.

The dominant variables responsible for the large uncertainties can be identified by inspection of Tables 5-1 through 5-7. The largest contributions to the total variance are summarized in Table 5-8, where the ranking reflects decreasing uncertainty from top to bottom and left to right.

The largest uncertainties are in cesium and iodine plateout distribution, with core release and metal sorptivity dominating. The latter

is uncertain because a sufficient number of sorption experiments have not been performed on pertinent metals under the appropriate conditions to provide the proper data base. The uncertainty in the plateout distribution of strontium-90 is caused by large uncertainties in strontium release characteristics, but the consequences are small because the nominal release of strontium-90 is small compared with the GASSAR expected release criterion (see Section 4). The uncertainty in the silver and cerium distributions is dominated by uncertainty in core release and mass transfer processes. The latter should be readily reduced by analysis of experimental plateout data. Uncertainty in the tellurium-131m distribution is also a mass transfer effect. The uncertainty in the lanthanum-140 distribution is quite small because the deposition in the cold leg (where uncertainties in mass transfer are most dominant) is controlled by xenon-140 decay.

Another point to consider is that the results are biased by the models used to simulate the transport processes. The mass transport model assumes transport of atomic species only. Significant quantities of circulating particulate matter have been observed in the Peach Bottom and AVR reactors. If fission products sorb on this dust, their transport characteristics will be dramatically altered. Since aerosol transport is still an emerging science, it is very difficult to accurately model the phenomena from first principles. All that can presently be said is that the limited experimental plateout profiles generally indicate molecular transport. Other areas of modeling difficulties involve the extent to which sorbed fission products penetrate the bulk of structural materials. The precise nature of the sorption of iodine on steels is also unclear. Further work in these areas is needed.

5.1. METHODS AND PROCEDURE

The propagation of errors formulation, essentially as described in Section 2, was used to calculate the uncertainty in specific plateout

activity at various points in the primary circuit. Slight differences in notation were employed, as summarized below. The general treatment is similar to that described in Section 3.2.

If a_k is the specific activity at the k^{th} location in the primary circuit, then

$$a_k = a_k(y_1, y_2, y_3, \dots y_n) \quad , \quad (5-1)$$

where $y_i = i^{\text{th}}$ variable affecting the activity. The nominal activity $\hat{a}_k$ (i.e., the activity calculated using nominal values for all parameters) is defined as

$$\hat{a}_k = \hat{a}_k(\hat{y}_1, \hat{y}_2, \hat{y}_3, \dots \hat{y}_n) \quad , \quad (5-2)$$

where $\hat{y}_i =$ nominal value of i^{th} variable. To make the linearizing transformation,

$$A_k = \log \left(\frac{a_k}{\hat{a}_k} \right) \quad , \quad (5-3)$$

and

$$x_i = \log \frac{y_i}{\hat{y}_i} \quad . \quad (5-4)$$

Then,

$$A_k = A_k(x_1, x_2, x_3, \dots x_n) \quad . \quad (5-5)$$

If A_k is a linear function of the x_i over the range of interest, the total variance of A_k is given by

$$\sigma_{A_k}^2 = \sum_{i=1}^n \left(\frac{\partial A_k}{\partial x_i} \right)^2 \sigma_{x_i}^2 + 2 \sum_{i=1}^{n-1} \sum_{j>1}^n \rho_{x_i x_j} \left(\frac{\partial A_k}{\partial x_i} \right) \left(\frac{\partial A_k}{\partial x_j} \right) \sigma_{x_i} \sigma_{x_j} \quad , \quad (5-6)$$

where $\sigma_{A_k}^2$ = variance of A_k ,
 $\sigma_{\log a_k}^2$ = variance of $\log a_k$,
 $\sigma_{x_i}^2$ = variance of x_i ,
 $\rho_{x_i x_j}$ = correlation coefficient between x_i and x_j .

It is assumed that A_k has a normal distribution about a mean value of zero, and hence a_k itself has a log normal distribution. It is also assumed that x_i values vary only systematically. Hence, from the discussion in Section 2, the nominal and median plateout distributions are the same.

Plateout distribution calculations at GA are currently performed with the PAD code, a transient one-dimensional mass transfer code. A finite difference solution is obtained for the coupled, nonlinear differential equations describing the conservation of mass with a convective boundary condition (Ref. 5-1). It is assumed that the metallic fission products are released from a point at the coolant exit from the core.

The coolant and surface concentrations are coupled by a concentration-dependent sorption process (either Langmuir or Freundlich isotherms can be employed). The model allows for production by precursor decay and treats recirculation in a closed loop. The PAD code has three options for treating deposition: (1) no sorption (e.g., a nonadsorbing noble gas); (2) no desorption (the surface is a perfect sink, or more precisely, the vapor pressure over the surface is zero for all surface concentrations); and (3) desorption (an adsorption isotherm is employed such that at a given surface temperature and partial pressure, an equilibrium surface concentration exists beyond which no further accumulation occurs).

The procedure followed for each isotope was as follows:

1. The nominal plateout distribution was calculated.
2. Sensitivity studies were performed with the PAD code to determine the partial derivatives in Eq. 5-6.

3. The median plateout distribution was calculated. Since all variables affecting the plateout distribution were considered to be only systematically uncertain, the median and nominal plateout distributions are identical.
4. The variance of the specific activity at local points in the circuit was calculated from the systematic uncertainties and the partial derivatives.
5. Confidence statements were made about the specific activity at local points in the primary circuit.

Plateout distributions were obtained for the following regions in the primary circuit:

1. Active core.
2. Lower replaceable reflector.
3. Transition block and core support floor.
4. Steam generator inlet plenum and cross ducts.
5. Reheater.
6. Hot duct.
7. Superheater.
8. Evaporator and economizer.
9. Circulator.
10. Core inlet plenum and cross ducts.
11. Upper reflector.

Sensitivity studies were performed at the inlet and outlet of each of these regions. Confidence statements were made regarding the specific activity of each isotope of interest at the inlet and outlet of each region. In this report, the regions are sometimes referred to by the above numbers rather than by name.

5.2. VARIANCE IN PLATEOUT DISTRIBUTIONS

Uncertainty in the plateout distribution arises from uncertainties in the core release, the convective mass transfer and deposition processes, and the data used to characterize these processes. Variables considered uncertain in this analysis are summarized in Table 5-9; the choice of variables is dictated by the transport model formulated in the PAD code. This uncertainty analysis does not include contributions arising from any variables other than those listed or those from modeling uncertainties. Justifiable assessments of these indeterminate effects cannot be made at this time. The variables in Table 5-9 are assumed to be uncorrelated.

The uncertainty in core release is treated in Sections 3 and 4. Gaseous and metallic release are important to plateout calculations because iodine release is modeled as xenon release, and some important nuclides have multiple sources (e.g., cesium-137 results from release of xenon-137 precursor and from direct metal release).

The convective mass transport process occurring in the primary coolant circuit is believed to be quite complex. However, in the PAD code the process is simplistically modeled by use of an empirically determined correlation which gives a mass transfer coefficient analogous to a film heat transfer coefficient (Ref. 5-1). The correlation (Ref. 5-2) in PAD, which was derived from experimental data with Reynolds numbers from 2000 to 35,000, Schmidt numbers from 0.6 to 2.5, and gas pressures from 0.1 to 3 atm is

$$Sh = 0.023 Re^{0.83} Sc^{0.44} , \quad (5-7)$$

where Sh = Sherwood number, kd/D ,
 Re = Reynolds number, $\rho Vd/\mu$,
 Sc = Schmidt number, $\mu/\rho D$,

or

$$\frac{kd}{D} = 0.023 \left(\frac{\rho V d}{\mu} \right)^{0.83} \left(\frac{\mu}{\rho D} \right)^{0.44} . \quad (5-8)$$

It follows that

$$k \propto (\rho)^{0.39} (V)^{0.83} (d)^{-0.17} (\mu)^{-0.39} (D)^{0.56} , \quad (5-9)$$

where k = mass transfer coefficient (cm/sec),

ρ = helium density (g/cm³),

V = velocity (cm/sec),

d = hydraulic diameter (cm),

μ = helium viscosity (g/cm-sec),

D = gaseous diffusivity (cm²/sec).

Because of very limited experimental data, the gaseous diffusivity of radionuclides in helium is quite uncertain. In addition, the appropriateness of the empirical coefficients is uncertain given HTGR operating conditions and complex geometry. For simplicity, an overall uncertainty in the mass transfer coefficient was treated in this study; the uncertainty was rather arbitrarily estimated to be a factor of 2 at 1σ . Although no attempt was made to quantify individual contributions, the uncertainty in the gaseous diffusivity is clearly a significant factor.

The sorptive capacity of primary circuit structural materials for important radionuclides (iodine, cesium, etc.) is at present poorly defined. Uncertainties are large, but quantification of these uncertainties is very complex. Estimates of the variances in the sorption data for cesium and iodine on graphite, high-alloy steels, and low-alloy steels are presented in Sections 6 and 8.

5.3. CESIUM-137 PLATEOUT DISTRIBUTION

The results obtained for cesium-137 are presented in detail to illustrate the application of the methods discussed in Sections 5.1 and 5.2. The plateout distribution of cesium-137 is uncertain because of systematic uncertainties in all the variables listed in Table 5-9. From Eq. 5-6, the variance of the specific activity at the k^{th} location is

$$\sigma_{A_k}^2 = \sum_{i=1}^5 \left(\frac{\partial A_k}{\partial x_i} \right)^2 \sigma_{x_i}^2 = \sum_{i=1}^5 \left(M_{ik} \sigma_{x_i} \right)^2, \quad (5-10)$$

where $M_{ik} = (\partial A_k / \partial x_i)$. The base case for all cesium sensitivity studies was taken to be the GASSAR expected total plateout inventory.

5.3.1. Xenon-137 Precursor Release

The effect of varying the total core release of xenon-137 by factors of 2 and 4 relative to the base case (GASSAR expected) is shown in Fig. 5-8. At all locations, the change is either small or negligible because direct release of cesium-137 metal completely dominates the xenon-137 component. The sensitivities are quantitatively estimated by calculating the slope of sensitivity curves; the results are presented in Fig. 5-9. From Table 4-11, for xenon-137, converting from $\ln$ to $\log$,

$$\sigma_{x_1}(\log) = 0.151.$$

Therefore, from Fig. 5-9, over the range $\pm\sigma = \pm 0.151$, the slopes and M_{1k} can be estimated. (To keep the calculations tractable, slopes were determined only for local points corresponding to the inlet and outlet of each major circuit component.) The sensitivity plots are relatively linear over the $\pm\sigma$ range. The $(M_{1k} \sigma_{x_1})^2$ terms are zero or small (see Table 5-1).

5.3.2. Cesium-137 Direct Release

The effect of varying the total core release of cesium-137 metal by factors of 2 and 10 is shown in Fig. 5-10. Changes in these sections up to and including the superheaters are small. In the cold leg, the resultant changes are large. Sensitivity plots are given in Fig. 5-11.

From Table 3-2, for cesium-137,

$$\sigma_{x_2}(\log) = 0.887 \quad .$$

Over the $\pm 1\sigma$ range, the sensitivity plots for selected local points vary from approximately linear to highly nonlinear. In the latter case, two slopes were determined: the slope at the origin and the average slope $\bar{M}$, where

$$\bar{M} = \frac{A_k(+\sigma) - A_k(-\sigma)}{+\sigma - (-\sigma)} \quad . \quad (5-11)$$

For conservatism, M_{2k} values were taken as the larger of the two calculated slopes.

5.3.3. Mass Transfer Coefficient

The mass transfer coefficient was calculated within PAD from the local fluid properties, the flow geometry, and a reference gaseous diffusivity which is an input quantity. The mass transfer coefficient was varied by varying the diffusivity of the fission product atoms in helium.

Since from Eq. 5-9, $k \propto D^{0.56}$, it follows that

$$\log \left(\frac{k}{\bar{k}} \right) = 0.56 \log \left(\frac{D}{\bar{D}} \right) \quad . \quad (5-12)$$

Therefore, sensitivity plots of A_k versus $\log (\hat{D}/D)$ from which the desired slopes can be readily computed can be prepared. From Eq. 5-12,

$$\frac{\partial A_k}{\partial \left(\log \frac{k}{\hat{k}} \right)} = \frac{1}{0.56} \frac{\partial A_k}{\partial \left(\log \frac{D}{\hat{D}} \right)} \quad (5-13)$$

As stated in Section 5.2, the 1σ uncertainty in the mass transfer coefficient is arbitrarily taken to be a factor of 2. Hence,

$$\sigma_{x_3} (\log) = 0.301 \quad ,$$

and in the sensitivity plots,

$$\sigma_{\hat{D}/D} (\log) = 0.301/0.56 = 0.54 \quad .$$

Thus, in determining M_{3k} values, the range of interest is ± 0.54 .

The effect of varying the gaseous diffusivity by factors of 2 and 4 (equivalent to changes of 1.5 and 2.2 in k , respectively) is given in Fig. 5-12; the base case used the cesium-in-helium diffusivity determined from GAIL IV diffusion probe data (Ref. 5-3). Sensitivity plots are given in Fig. 5-13; the plots range from linear in the hot leg to concave downward in the cold leg. There is no effect from the hot cross ducts through the superheaters because sorption effects are controlling and recirculation effects are negligible over the variation range. Since the experimental data are meager, estimation of

$$\sigma_{x_3} (\log) = 0.301$$

is rather arbitrary. Slopes were estimated over the $\pm 1\sigma$ range, again selecting the larger of the average and the tangent at the origin. The calculated $(M_{3k} \sigma_{x_3})^2$ terms ranged from zero in the hot leg to increasingly

larger values in the cold leg because errors in estimating the rate of depletion of activity from the coolant are cumulative.

5.3.4. Graphite Sorptivity

Although the vapor pressure of cesium over graphite powder has been exhaustively investigated, the extent to which cesium metal can penetrate the bulk of the hot reflector block graphite is not well known. Preliminary results from postirradiation examination of Peach Bottom fuel elements suggest substantial penetration. Therefore, it is currently assumed that the entire internal surface area (as determined by BET measurements) is available for cesium sorption.

The effect of changing the sorptivity of graphite by factors of 10 and 100 is shown in Fig. 5-14. By inspection, only the lower reflectors are substantially affected, because if the reflector graphite were a perfect sink (as is the case when the sorptivity is increased by factors of 10 or more), only approximately 16% of the total cesium released would deposit there. Thus, the effect on the overall mass balance is small. The sensitivity plots are shown in Fig. 5-15; the slopes are essentially zero in most sections and are highly nonlinear. The tangent at the origin is consistently the larger slope.

The vapor pressure of cesium over graphite affects fuel release and plateout distribution. In general, experimental data are fit with a Freundlich isotherm of the form (see Section 8)

$$\log p = \left(A + \frac{B}{T} \right) + \left(D + \frac{E}{T} \right) \log C \quad , \quad (5-14)$$

where p = vapor pressure (atm),
 T = absolute temperature (°K),
 C = concentration (μ mole Cs/g graphite),
 A, B, D, E = experimentally determined constants.

In Section 6.2.3, it is shown that in the Freundlich region the uncertainty in the vapor pressure for a given loading is given by

$$\sigma^2 (\log p) = 0.00134 + 0.00264 (\log C) + 0.0152 (\log C)^2 \quad . \quad (5-15)$$

From PAD code calculations, the maximum cesium loading is 1.4 $\mu\text{mole/g}$. Therefore,

$$\sigma^2 (\log p) = 1.94 \times 10^{-3} \quad ,$$

and

$$\sigma (\log p) = 0.044.$$

However, the more appropriate variable for deposition calculation is the uncertainty in the equilibrium surface concentration for a given partial pressure and temperature; in fact, this is the quantity which was varied in Fig. 5-15. Calculating $\sigma (\log C)$ from $\sigma (\log p)$ is straightforward using Eq. 5-14:

$$\log p = f (\log C) \quad , \quad (5-16)$$

$$\sigma^2 (\log p) = \left(\frac{\partial f}{\partial \log C} \right)^2 \sigma_{\log C}^2 = \left(D + \frac{E}{T} \right)^2 \sigma_{\log C}^2 \quad , \quad (5-17)$$

or

$$\sigma (\log p) = \left(D + \frac{E}{T} \right) \sigma (\log C) \quad . \quad (5-18)$$

5.3.5. Cesium-137 Metal Sorptivity

Conceptually, the effects of metal sorptivity uncertainties are determined in the same manner as those for graphite (see Section 5.3.4).

The base case presumes a metal sorptivity throughout the hot leg equal to that of oxidized 304 stainless steel (Ref. 5-1).

The effect of changing metal sorptivity by factors of 10 and 100 relative to the base case is shown in Fig. 5-16, and sensitivity plots are given in Fig. 5-17. The plots range from linear to highly nonlinear; in fact, some even exhibit maxima at zero (e.g., the outlet of the superheater). The reason for this extreme behavior is that changing the metal sorptivity greatly perturbs the overall mass balance.

The uncertainty in the vapor pressure of cesium over high-alloy steel is estimated in Section 8.3 to be

$$\sigma^2 (\log p) = 1.78 + 0.163 (\log C) + 0.132 (\log C)^2, \quad (5-19)$$

where in this case C is in units of $\mu\text{g}/\text{cm}^2$; again, the desired quantity is $\sigma (\log C)$. By applying Eq. 5-18 and assuming a maximum loading of $46 \mu\text{g}/\text{cm}^2$, the uncertainty is estimated to be

$$\sigma (\log C) = 1.32.$$

Clearly, over the range $\pm 1\sigma$, a number of the sensitivity plots in Fig. 5-17 are highly nonlinear. The log transform fails to condition the function, so Eq. 5-10 cannot be rigorously applied. Higher-order derivatives are required. However, as a matter of expediency, the calculation was made using the rationale that in general, equilibrium conditions are maintained in the hot leg over a wide range of sorptivities such that the slopes of the corresponding sensitivity plots tend to be approximately unity. Therefore, local points which were obvious singularities (e.g., superheater outlet) were assigned values of $(\partial A_k / \partial x_5) = 1.0$ as an apparent upper limit. Finally, the $(M_{5k} \sigma_{x_5})^2$ terms were determined and were generally large (see Table 5-1).

5.3.6. Confidence Statements on Cesium-137 Distribution

Once all the variables in Eq. 5-10 have been determined, the total variance in the specific cesium-137 activity at local points can be calculated. Variances at the inlet and outlet of each of the 11 regions of the primary circuit are summarized in Table 5-1.

Given the variance in A_k and the nominal (median) plateout distribution, confidence statements can be made. To obtain the upper 95% confidence specific activity level at local points, the following expression is applied:

$$a_{k95} = \hat{a}_k 10^{1.65 \sigma_{A_k}} \quad (5-20)$$

Taken collectively, these local point confidence statements provide a 95% confidence plateout distribution; the results are given in Fig. 5-1, along with the median distribution and the expected and design distributions calculated with GASSAR expected and design inventories.

5.4. STRONTIUM-90 PLATEOUT DISTRIBUTION

The methodology for calculating the uncertainty in the strontium-90 plateout distribution is analogous to that for cesium-137 with the exception that the metal surfaces are treated as perfect sinks for strontium and the $(M_{5k} \sigma_{x_5})^2$ term vanishes from Eq. 5-10. The perfect sink assumption for strontium is based on its chemical similarity to barium, for which perfect sink behavior has been observed. No experimental data are known for strontium-90, a beta emitter (Refs. 5-4, 5-5). Again, the base case for the sensitivity calculations was the GASSAR expected inventory.

5.4.1. Krypton-90 Precursor Release

The effect of varying the core release of krypton-90 by factors of 2 and 4 is shown in Fig. 5-18, and sensitivity plots are given in Fig. 5-19.

From Table 4-11, converting from $\ln$ to $\log$, the uncertainty in krypton-90 release is

$$\sigma_{x_1}(\log) = 0.128 \quad .$$

Over the $\pm 1\sigma$ range, the sensitivity plots are linear.

5.4.2. Strontium-90 Direct Release

The effect of varying the core release of strontium-90 metal by factors of 3 and 10 is presented in Fig. 5-20, and corresponding sensitivity plots are shown in Fig. 5-21. From Table 3-2, the uncertainty in direct strontium-90 release is

$$\sigma_{x_2}(\log) = 0.826 \quad .$$

In general, the sensitivity curves are linear over the $\pm 1\sigma$ range.

5.4.3. Mass Transfer Coefficient

The effect of changing the gaseous diffusivity (mass transfer coefficient) of strontium in helium by factors of 2 (1.5) and 4 (2.2) is shown in Fig. 5-22. The distribution is shown to be strongly dependent upon this variable. The increasing slopes in the circulator and upper plenum reflect a net increase in the strontium-90 partial pressure at these locations, which results from more rapid production of strontium-90 from krypton-90 decay rather than depletion by plateout.

The sensitivity curves are shown in Fig. 5-23. As is the case for cesium-137, the 1σ range in the mass transfer coefficient was taken to be a factor of 2. The plots are approximately linear over the $\pm 1\sigma$ span.

5.4.4. Graphite Sorptivity

As is the case for cesium, the vapor pressure of strontium over graphite has received considerable attention, but the depth of penetration of hot reflector block graphite is unknown. The limited data available show very modest penetration. Therefore, it is assumed that only the geometric surface area is available for strontium sorption, in contrast to the total block availability for cesium.

The effect of changing the sorptivity of graphite for strontium by factors of 10 and 100 is shown in Fig. 5-24, and sensitivity curves are shown in Fig. 5-25. The effect is confined to the hot lower reflectors where the M_{4k} slopes are unity. Everywhere else the slope is zero.

In Section 6.3.3, the variance σ^2 in the strontium vapor pressure over graphite is given as

$$\sigma^2 (\log p) = 0.0239 - 0.0321 (\log C) + 0.0302 (\log C)^2 \quad . \quad (5-21)$$

Using a maximum concentration of 1 $\mu\text{mole/g}$,

$$\sigma (\log p) = 0.155 \quad .$$

From Eq. 5-18 and using the values for D and E given in Section 6, $\sigma (\log C)$ at 780°C can be determined to be

$$\sigma (\log p) = 3.79 \sigma (\log C) \quad ,$$

$$\sigma (\log C) = 0.155/3.79 = 0.041 \quad .$$

Thus, the $(M_{4k} \sigma_{x_4})^2$ terms can finally be determined.

5.4.5. Confidence Statements on Strontium-90 Distribution

Specific activity variances for strontium-90 at the inlet and outlet of each region of the primary circuit are summarized in Table 5-2. The upper 95% plateout distribution is given in Fig. 5-2, along with median, expected, and design distributions.

5.5. IODINE-131 PLATEOUT DISTRIBUTION

The plateout distribution of iodine-131 is determined by the distribution of the more plentiful stable isotopes iodine-127 and iodine-129, which totally dominate on an atom basis. Given the relatively high volatility of iodine, it is not surprising to find that hot graphite and metals have a very low sorptivity for iodine. Recent data have drastically decreased the estimated iodine sorptivity of graphite, high-alloy steel, and low-alloy steel; these results are presented in Section 8. The median iodine distribution reflects this change, as do most of the sensitivity calculations.

5.5.1. Iodine Direct Release

The release of iodine from the fuel is calculated by assuming that iodine atoms behave like xenon (Ref. 4-1). The effect of increasing the core release of all iodine isotopes by factors of approximately 2 and 4 relative to the median base case is shown in Fig. 5-26. Sensitivity curves are plotted in Fig. 5-27 and are approximately linear over the $\pm 1\sigma$ range given in Table 4-2 as $\sigma_{x_1}(\log) = 0.151$.

5.5.2. Mass Transfer Coefficient

The deposition of iodine is sorption controlled throughout the primary circuit and is negligible in the hot leg. Consequently, variations of the gaseous diffusivity (mass transfer coefficient) by factors of 2 (1.5) and 4 (2.2) had essentially no effect on the iodine plateout distribution. The results are not shown in this section.

5.5.3. Iodine-Graphite Sorptivity

The sorptivity of graphite for iodine is such that negligible quantities are sorbed on the hot lower reflectors. Therefore, changing the sorptivity of graphite by factors of approximately 10 and 100 is expected to have little effect except on the activity on the cold upper reflector blocks. This result is confirmed in Fig. 5-28, and the sensitivity curves from Fig. 5-29 give slopes of either zero or unity.

From Section 8.2.1, the uncertainty in the vapor pressure of iodine over graphite is given by

$$\sigma^2 (\log p) = 4.87 + 4.23 (\log C) + 3.44 (\log C)^2, \quad (5-22)$$

where C is in units of $\mu\text{g I/g graphite}$. From PAD calculations, the maximum loading is taken to be $0.09 \mu\text{g I/g graphite}$, giving

$$\sigma^2 (\log p) = 4.20.$$

Finally, upon application of Eq. 5-18, the uncertainty in $\log C$ at 340°C is estimated to be

$$\sigma (\log C) = 1.29.$$

The calculated $(M_{4k} \sigma_{x_4})^2$ terms are negligible for all sections except the upper reflectors.

5.5.4. Iodine-Metal Sorptivity

The sorption of iodine on metals is a complex phenomenon involving the formation of metal halides under certain conditions. The sorptivity also appears to vary greatly with alloy composition. Although the data are limited and badly scattered, high-alloy materials (Incoloy 800, Hastelloy-X, etc.) seem to sorb less iodine than low-alloy materials under comparable

conditions, with neither class sorbing significant quantities at temperatures above about 500°C. Therefore, for reactor design, the sorptivity of the low-alloy materials (chromaloy steels) used at low service temperatures is of greatest concern.

Using the low-alloy steel isotherms in Section 8.2.3, a parametric study was performed to determine the effect of varying the metal sorptivity factors by 10 and 100. The results are presented in Figs. 5-30 and 5-31. Large reductions in metal sorptivity result in substantial saturation of the primary circuit in less than 40 yr of plant operation. Consequently, late in life the plateout per pass diminishes, and the circulating iodine inventory increases. At the saturation limit, iodine behaves like a nonadsorbing noble gas, and the circulating iodine inventory is determined by a balance of the core release rate and the helium purification rate. Thus, for cases of metal sorptivity reduced by factors of 10 and 100 (Fig. 5-30), the plateout inventories are significantly less than the core release because substantial iodine is removed by the purification system. As with cesium-137, some of the sensitivity curves (Fig. 5-31) have maxima at the origin; the slope in these cases was again taken to be unity. Using previously defined procedures, the uncertainty was calculated from the estimated variance in Section 8.2.3 to be

$$\sigma (\log C) = 1.1 \quad .$$

5.5.5. Confidence Statements on Iodine-131 Distribution

Following standard procedures, the variances, multipliers, and upper 95% confidence activities were calculated. The results are tabulated in Table 5-3 and presented in Fig. 5-3.

5.6. LANTHANUM-140 PLATEOUT DISTRIBUTION

Calculation of the lanthanum-140 plateout distribution is complicated by multiple sources. It has been shown that these sources can be lumped

into a gaseous precursor component coming from xenon-140 and a metallic release component comprised of barium-140 and lanthanum-140. The limited available plateout data suggest that graphite and metal surfaces are perfect sinks for lanthanum, eliminating the need to quantify sorption effects and removing the $(M_{4k} \sigma_{x_4})^2$ and $(M_{5k} \sigma_{x_5})^2$ terms from Eq. 5-10. Under the perfect sink assumption, the choice of base case (in this instance, the GASSAR expected inventory criterion) is arbitrary because the relative distribution is independent of total inventory. The median lanthanum-140 distribution was calculated with median core releases from Section 4.

5.6.1. Xenon-140 Precursor Release

The effect of increasing the core release of the xenon-140 precursor by factors of 2 and 4 is shown in Figs. 5-32 and 5-33. By inspection, the consequences range from modest in the hot leg to moderate in the cold leg. Table 4-11, converted from $\ln$ to $\log$, gives

$$\sigma_{x_1}(\log) = 0.204 \quad ,$$

and the sensitivity curves are linear over that range.

5.6.2. Direct Metallic Release

The effect of changing the core release of barium-140 and lanthanum-140 by factors of 5 and 10 is shown in Figs. 5-34 and 5-35. In the hot leg the consequences are large; in the cold leg they are small. The release of barium and lanthanum isotopes is controlled by contamination levels (minimum R/B, see Section 4). In Section 7, it is determined that the uncertainty in the minimum R/B is

$$\sigma_{x_2}(\log) = 0.699 \quad .$$

Given this large σ_{x_2} , the $(M_{2k} \sigma_{x_2})^2$ terms are significant.

5.6.3. Mass Transfer Coefficient

Changes in the gaseous diffusivity (mass transfer coefficient) affect lanthanum-140 plateout distribution as shown in Figs. 5-36 and 5-37. The effect is substantial in sections where the direct release component dominates the gaseous precursor; in this instance, exceptions are the circulator and upper plenum. The sensitivity curves are reasonably linear over the $\pm 1\sigma$ range.

5.6.4. Confidence Statements on Lanthanum-140 Distribution

The estimated variances in the lanthanum-140 activity at local points are given in Table 5-4. The median and 95% confidence plateout distributions and the expected and design distributions are presented in Fig. 5-4. The median distribution is less than or equal to the expected distribution, and the upper 95% distribution is well below the design distribution.

5.7. CERIUM-144, SILVER-110m, AND TELLURIUM-131m PLATEOUT DISTRIBUTIONS

Under the design assumptions currently employed, the distributions of cerium-144, silver-110m, and tellurium-131m (as well as certain other nuclides) are quite similar in that all are directly released with no significant gaseous precursors and are believed to exhibit perfect sink deposition behavior. The uncertainties in their plateout distributions are, in fact, different primarily because of different core release uncertainties.

5.7.1. Direct Metallic Release

Under perfect sink deposition conditions with a single source, the relative plateout distribution is independent of the total amount released;

the specific activity everywhere varies in proportion to the total core release. Consequently, all $(\partial A_k / \partial x_1)$ and $(\partial A_k / \partial x_2)$ slopes are equal to unity. The uncertainty in the total release of individual nuclides, however, varies substantially with the chemical nature of the element. The release of cerium-144 is controlled by the minimum R/B, and thus $\sigma_{x_2}(\log) = 0.699$. The uncertainty of silver-110m release has been established as $\sigma_{x_2}(\log) = 0.412$ (Section 4.4). The release of tellurium isotopes is estimated by assuming xenon-like behavior; for tellurium-131m, $\sigma_{x_1}(\log) = 0.151$. Given the above information, $(M_{1k} \sigma_{x_1})^2$ and $(M_{2k} \sigma_{x_2})^2$ can be easily calculated.

5.7.2. Mass Transfer Coefficient

The deposition process is dominated throughout the primary circuit by mass transport effects. Consequently, as demonstrated for cerium-144 in Figs. 5-38 and 5-39, changes of a factor of 2 in the gaseous diffusivity can result in order of magnitude changes in the specific activity at local points since the change is cumulative. The sensitivity plots are approximately linear over the $\pm 1\sigma$ range. As before, the larger of the tangents at the origin and the average slope is used. Although the diffusivity of cerium, silver, and tellurium in helium would be expected to vary somewhat because of differences in their physical chemistry, the magnitude of this variation is believed to be small compared with the uncertainty in the data. Therefore, the same slope is used for all three nuclides.

5.7.3. Confidence Statements on Distributions

The variances in specific cerium-144, silver-110m, and tellurium-131m activities are tabulated in Tables 5-5, 5-6, and 5-7, respectively. The corresponding upper 95% confidence plateout distributions are shown in Figs. 5-5, 5-6, and 5-7, respectively. In general, the distributions are quite similar qualitatively, with the median case at or below the expected and the 95% profile at or slightly above the design level.

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- 5-5. Hanson, D. L., "Results of the General Atomic Deposition Loop Program," ERDA Report GA-A13140, General Atomic, to be published.

Fig. 5-1. Plateout distribution of Cs-137

Fig. 5-2. Plateout Distribution of Sr-90

Fig. 5-3. Plateout distribution of I-131

Fig. 5-4. Plateout distribution of La-140

Fig. 5-5. Plateout distribution of Ce-144

Fig. 5-6. Plateout distribution of Ag-110

Fig. 5-7. Plateout distribution of Te-131m

Fig. 5-8. Effect of Xe-137 precursor on Cs-137 plateout

Fig. 5-9. Determination of M_{1k} for Cs-137 (Xe-137 precursor release)

Fig. 5-10. Effect of metal release on Cs-137 plateout

Fig. 5-11. Determination of A_{2k} for Cs-137 (direct Cs-137 release)

Fig. 5-12. Effect of gaseous diffusivity on Cs-137 plateout

Fig. 5-13. Determination of $\partial A_k / \partial \log (D/\hat{D})$ for Cs-137 (mass transfer coefficient)

Fig. 5-14. Effect of graphite sorptivity on Cs-137 plateout

Fig. 5-15. Determination of M_{4k} for Cs-137 (graphite sorptivity)

Fig. 5-16. Effect of metal sorptivity on Cs-137 plateout

Fig. 5-17. Determination of M_{5k} for Cs-137 (metal sorptivity)

Fig. 5-18. Effect of Kr-90 release on Sr-90 plateout

Fig. 5-19. Determination of M_{1k} for Sr-90 (Kr-90 precursor release)

Fig. 5-20. Effect of Sr-90 direct release on plateout distribution

Fig. 5-21. Determination of M_{2k} for Sr-90

Fig. 5-22. Effect of gaseous diffusivity on Sr-90 plateout

Fig. 5-23. Determination of $\partial A_k / \partial \log (D/\hat{D})$ for Sr-90 (mass transfer coefficient)

Fig. 5-24. Effect of graphite sorptivity on Sr-90 plateout

Fig. 5-25. Determination of M_{4k} for Sr-90 (graphite sorptivity)

Fig. 5-26. Effect of core release on I-131 plateout distribution

Fig. 5-27. Determination of M_{1k} for I-131 (core release)

Fig. 5-28. Effect of graphite sorptivity on I-131 plateout distribution

Fig. 5-29. Determination of M_{4k} for I-131 (graphite sorptivity)

Fig. 5-30. Effect of metal sorptivity on I-131 plateout distribution

Fig. 5-31. Determination of M_{5k} for I-131

Fig. 5-32. Effect of Xe-140 release on La-140 plateout distribution

Fig. 5-33. Determination of M_{1k} for La-140 (Xe-140 precursor release)

Fig. 5-34. Effect of Ba-140/La-140 release on La-140 plateout distribution

Fig. 5-35. Determination of M_{2k} for La-140 (direct metal release)

Fig. 5-36. Effect of gaseous diffusivity on La-140

Fig. 5-37. Determination of $\partial A_k / \partial \log (D/\hat{D})$ for La-140 (mass transfer coefficient)

Fig. 5-38. Effect of gaseous diffusivity on Ce-144 plateout distributions

Fig. 5-39. Determination of $\partial A_k / \partial \log(D/\hat{D})$ for Ce-144 (mass transfer coefficient)

TABLE 5-1
VARIANCE IN CESIUM-137 PLATEOUT DISTRIBUTION

Section No.	$(M_{1k}\sigma_{x_1})^2$	$(M_{2k}\sigma_{x_2})^2$	$(M_{3k}\sigma_{x_3})^2$	$(M_{4k}\sigma_{x_4})^2$	$(M_{5k}\sigma_{x_5})^2$	σ_{A_k}	$10^{1.65\sqrt{\sigma_{A_k}^2}}$	Cesium Median (Ci/cm ²)	Upper 95% Confidence Level (Ci/cm ²)
1-Inlet	0	3.15	2.43	9.40-06	1.259	6.839	2.065+04	4.57-06	9.44-02
1-Outlet	0	3.15	6.66	9.40-06	1.259	11.065	3.08+05	2.89-07	8.90-02
2-Inlet	0	0.68	0.069	1.55-06	0	0.749	26.82	8.08-04	2.17-02
2-Outlet	0	0.68	0.059	1.55-06	0	0.739	26.18	7.62-04	1.99-02
3-Inlet	0	0.39	0.018	1.15-04	0	0.408	11.33	2.06-04	2.33-03
3-Outlet	0	0.41	0.018	1.15-04	0	0.428	12.02	2.05-04	2.46-03
4-Inlet	0	0.09	0	1.30-06	1.813	1.903	188.8	2.80-06	5.29-04
4-Outlet	0	0.09	0	1.30-06	1.813	1.903	188.8	2.80-06	5.29-04
5-Inlet	0	0.01	0	1.30-06	1.742	1.752	152.8	5.24-05	8.01-03
5-Outlet	0	0.005	0	1.30-06	1.673	1.678	137.3	8.37-05	1.15-02
6-Inlet	0	0.018	0	1.30-06	1.673	1.691	139.9	3.09-05	4.32-03
6-Outlet	0	0.018	0	1.30-06	1.673	1.691	139.9	3.09-05	4.32-03
7-Inlet	0	0.015	0	1.30-06	1.172	1.187	62.7	6.81-05	4.27-03
7-Outlet	0	0.39	0	1.30-06	1.742	2.132	256.7	1.15-04	2.95-02
8-Inlet	0	2.55	0.061	3.63-06	0.929	3.540	1.27+03	7.07-04	8.99-01
8-Outlet	0	2.55	1.953	3.63-06	0.929	5.432	7.01+03	1.38-05	9.67-02
9-Inlet	4.56-06	2.55	2.396	3.63-06	0.929	5.875	9.98+03	7.76-06	7.75-02
9-Outlet	2.30-05	2.55	2.396	3.63-06	0.929	5.875	9.98+03	7.34-06	7.33-02
10-Inlet	6.39-05	2.55	2.396	3.63-06	0.929	5.875	9.98+03	4.62-06	4.61-02
10-Outlet	1.62-03	2.55	2.600	3.63-06	1.259	6.411	1.506+04	3.79-06	5.71-02
11-Inlet	1.76-03	3.15	2.600	3.63-06	1.259	7.011	2.338+04	7.73-06	1.81-01
11-Outlet	1.82-03	3.15	2.430	3.63-06	1.259	6.840	2.067+04	4.76-06	9.84-02

TABLE 5-2
VARIANCE IN STRONTIUM-90 PLATEOUT DISTRIBUTION

Section No.	$(M_{1k} \sigma_{x_1})^2$	$(M_{2k} \sigma_{x_2})^2$	$(M_{3k} \sigma_{x_3})^2$	$(M_{4k} \sigma_{x_4})^2$	$\sigma_{A_k}^2$	$1.65\sqrt{\sigma_{A_k}^2}$ 10	Strontium Median (Ci/cm ²)	Upper 95% Confidence Level (Ci/cm ²)
1-Inlet	1.23-02	0	1.416-03	0	1.374-02	1.561	5.0-07	7.805-07
1-Outlet	1.23-02	0	1.45-01	0	1.580-01	4.527	5.52-08	2.499-07
2-Inlet	0	4.0-02	0	1.989-03	4.199-02	2.178	3.08-07	6.709-07
2-Outlet	0	4.0-02	0	1.989-03	4.199-02	2.178	3.08-07	6.709-07
3-Inlet	0	4.0-02	0	1.989-03	4.199-02	2.178	6.15-08	1.340-07
3-Outlet	0	4.0-02	0	1.989-03	4.199-02	2.178	6.17-08	1.340-07
4-Inlet	0	6.4-01	8.425-02	0	7.243-01	25.36	4.52-06	1.146-04
4-Outlet	4.44-05	5.9-01	7.514-02	0	6.652-01	22.17	4.23-06	9.377-05
5-Inlet	4.44-05	5.9-01	7.514-02	0	6.652-01	22.17	1.91-05	4.234-04
5-Outlet	4.44-05	5.9-01	1.04-03	0	5.911-01	18.56	7.91-06	1.468-04
6-Inlet	4.44-05	5.9-01	1.04-03	0	5.911-01	18.56	1.16-05	2.153-04
6-Outlet	2.77-04	5.2-01	4.172-02	0	5.620-01	17.26	5.79-06	9.991-05
7-Inlet	2.77-04	5.2-01	4.172-02	0	5.620-01	17.26	2.38-06	4.107-05
7-Outlet	3.56-04	5.2-01	3.624-01	0	8.828-01	35.50	6.61-07	2.347-05
8-Inlet	3.56-04	5.2-01	3.624-01	0	8.828-01	35.50	6.20-07	2.201-05
8-Outlet	8.49-03	3.7-01	1.416-01	0	5.201-01	15.48	5.18-08	8.021-07
9-Inlet	8.49-03	3.7-01	1.416-01	0	5.201-01	15.48	3.57-08	5.528-07
9-Outlet	1.16-02	8.0-03	9.361-03	0	2.895-02	1.909	7.04-08	1.344-07
10-Inlet	1.16-02	8.0-03	9.361-03	0	2.895-02	1.909	4.73-08	9.028-08
10-Outlet	1.23-02	0	5.096-02	0	6.328-02	2.601	3.87-07	1.006-06
11-Inlet	1.23-02	0	5.096-02	0	6.328-02	2.601	8.30-07	2.158-06
11-Outlet	1.23-02	0	3.496-03	0	1.582-02	1.613	5.20-07	8.385-07

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TABLE 5-3
VARIANCE IN IODINE-131 PLATEOUT DISTRIBUTION

Section No.	$(M_{1k}\sigma_{x_1})^2$	$(M_{4k}\sigma_{x_4})^2$	$(M_{5k}\sigma_{x_5})^2$	$\sigma_{A_k}^2$	$1.65\sqrt{\sigma_{A_k}^2}$ 10	Iodine Median (Ci/cm ²)	Upper 95% Confidence Level (Ci/cm ²)
1-Inlet	0	0	0	0	1.0	0	0
1-Outlet	0	0	0	0	1.0	0	0
2-Inlet	0.012	1.664	0.018	1.694	140.5	1.80-07	2.53-05
2-Outlet	0.012	1.664	0.018	1.694	140.5	1.79-07	2.51-05
3-Inlet	0.012	1.664	0.018	1.694	140.5	3.59-08	5.04-06
3-Outlet	0.012	1.664	0.018	1.694	140.5	3.68-08	5.17-06
4-Inlet	0	0	1.168	1.168	60.7	1.05-13	6.37-12
4-Outlet	0	0	1.168	1.168	60.7	1.00-13	6.07-12
5-Inlet	0.008	0	1.168	1.176	61.6	2.68-13	1.65-11
5-Outlet	0.005	0	1.168	1.173	61.2	5.18-13	3.17-11
6-Inlet	0.014	0	1.168	1.182	62.2	1.78-13	1.11-11
6-Outlet	0.014	0	1.168	1.182	62.2	1.73-13	1.07-11
7-Inlet	0.004	0	1.052	1.056	49.6	3.60-13	1.79-11
7-Outlet	0.002	0	1.052	1.054	49.4	1.59-12	7.85-11
8-Inlet	0.002	0	1.052	1.054	49.4	3.26-06	1.61-04
8-Outlet	0.068	0	1.210	1.278	73.3	6.29-05	4.61-03
9-Inlet	0.020	0	1.210	1.230	67.6	1.12-05	7.57-04
9-Outlet	0.009	0	1.210	1.219	66.3	8.85-06	5.87-04
10-Inlet	0.032	0	1.210	1.242	69.0	8.55-06	5.90-04
10-Outlet	0.063	0	1.210	1.273	72.7	8.03-06	5.84-04
11-Inlet	0.010	1.664	0.681	2.355	340.5	4.74-07	1.61-04
11-Outlet	0.010	1.664	0.736	2.410	364.4	4.73-07	1.72-04

TABLE 5-4
VARIANCE IN LANTHANUM-140 PLATEOUT DISTRIBUTION

Section No.	$(M_{1k}^{\sigma x_1})^2$	$(M_{2k}^{\sigma x_2})^2$	$(M_{3k}^{\sigma x_3})^2$	$\sigma_{A_k}^2$	$1.65\sqrt{\sigma_{A_k}^2}$ 10	Lanthanum Median (Ci/cm ²)	Upper 95% Confidence Level (Ci/cm ²)
1-Inlet	3.349-02	0	1.043-02	4.392-02	2.217	2.18-07	4.833-07
1-Outlet	3.349-02	0	1.669-01	2.004-01	5.477	2.45-08	1.342-07
2-Inlet	0	4.5-01	7.812-02	5.281-01	15.82	7.58-07	1.199-05
2-Outlet	0	4.5-01	6.113-02	5.111-01	15.12	6.96-07	1.053-05
3-Inlet	5.358-05	4.5-01	6.113-02	5.112-01	15.13	3.27-07	4.946-06
3-Outlet	5.358-05	4.2-01	7.514-02	4.952-01	14.49	2.96-07	4.289-06
4-Inlet	5.358-05	4.2-01	5.113-02	4.812-01	13.95	2.28-07	3.180-06
4-Outlet	1.772-03	2.4-01	5.850-02	3.003-01	8.020	2.64-07	2.117-06
5-Inlet	1.772-03	2.4-01	5.850-02	3.003-01	8.020	1.19-06	9.543-06
5-Outlet	1.772-03	2.3-01	1.156-02	2.433-01	6.515	5.04-07	3.284-06
6-Inlet	1.772-03	2.3-01	1.156-02	2.433-01	6.515	7.37-07	4.802-06
6-Outlet	9.407-03	1.2-01	9.361-03	1.388-01	4.118	5.35-07	2.203-06
7-Inlet	9.407-03	1.0-01	9.361-03	1.188-01	3.704	2.27-07	8.408-07
7-Outlet	1.127-02	8.0-02	1.849-01	2.762-01	7.364	7.05-08	5.192-07
8-Inlet	1.127-02	8.0-02	1.849-01	2.762-01	7.364	6.68-08	4.919-07
8-Outlet	3.086-02	8.0-04	2.430-02	5.596-02	2.457	1.85-08	4.545-08
9-Inlet	3.086-02	8.0-04	1.953-02	5.119-02	2.362	1.30-08	3.071-08
9-Outlet	3.349-02	2.0-04	1.398-02	4.767-02	2.292	2.84-08	6.509-08
10-Inlet	3.349-02	2.0-04	1.398-02	4.767-02	2.292	1.92-08	4.401-08
10-Outlet	3.349-02	0	6.113-02	9.462-02	3.218	1.68-07	5.406-07
11-Inlet	3.349-02	0	6.113-02	9.462-02	3.218	3.61-07	1.162-06
11-Outlet	3.349-02	0	1.043-02	4.392-02	2.217	2.27-07	5.033-07

TABLE 5-5
VARIANCE IN CERIUM-144 PLATEOUT DISTRIBUTION

Section No.	$(M_{2k} \sigma_{x_2})^2$	$(M_{3k} \sigma_{x_3})^2$	$\sigma_{A_k}^2$	$1.65 \sqrt{\sigma_{A_k}^2}$ 10	Cerium Median (Ci/cm ²)	Upper 95% Confidence Level (Ci/cm ²)
1-Inlet	4.89-01 ↓	5.824+00	6.313+00	1.399+04	7.85-11	1.098-06
1-Outlet		1.180+01	1.229+01	6.072+05	4.22-12	2.562-06
2-Inlet		8.739-02	5.764-01	1.789+01	3.03-07	5.422-06
2-Outlet		7.514-02	5.641-01	1.735+01	2.785-07	4.832-06
3-Inlet		6.113-02	5.501-01	1.674+01	1.31-07	2.193-06
3-Outlet		6.113-02	5.501-01	1.674+01	1.24-07	2.076-06
4-Inlet		6.656-02	5.556-01	1.698+01	8.99-08	1.526-06
4-Outlet		6.565-02	5.556-01	1.698+01	8.16-08	1.385-06
5-Inlet		5.096-02	5.400-01	1.631+01	3.68-07	6.002-06
5-Outlet		3.500-03	4.925-01	1.439+01	1.53-07	2.201-06
6-Inlet		3.500-03	4.925-01	1.439+01	2.24-07	3.222-06
6-Outlet		1.147-01	6.037-01	1.914+01	1.04-07	1.991-06
7-Inlet		1.147-01	6.037-01	1.914+01	4.24-08	8.116-07
7-Outlet		6.328-01	1.122+00	5.593+01	1.15-08	6.431-07
8-Inlet		6.328-01	1.122+00	5.593+01	1.07-08	5.094-07
8-Outlet		4.692+00	5.181+00	5.698+03	2.36-10	1.345-06
9-Inlet		4.715+00	5.204+00	5.808+03	1.49-10	8.654-07
9-Outlet		4.715+00	5.204+00	5.808+03	1.40-10	8.131-07
10-Inlet		4.715+00	5.204+00	5.808+03	8.83-11	5.128-07
10-Outlet		5.024+00	5.513+00	7.485+03	6.40-11	4.790-07
11-Inlet		5.024+00	5.513+00	7.485+03	1.33-10	9.954-07
11-Outlet		5.824+00	6.313+00	1.399+04	8.19-11	1.146-06

TABLE 5-6
VARIANCE IN SILVER-110m PLATEOUT DISTRIBUTION

Section No.	$(M_{2k} \sigma_{x_2})^2$	$(M_{3k} \sigma_{x_3})^2$	$\sigma_{A_k}^2$	$1.65 \sqrt{\sigma_{A_k}^2}$ 10	Silver Median (Ci/cm ²)	Upper 95% Confidence Level ₂ (Ci/cm ²)
1-Inlet	1.7-01 ↓	5.824+00	5.994+00	1.096+04	1.92-09	2.104-05
1-Outlet		1.180+01	1.197+01	5.103+05	1.03-10	5.256-05
2-Inlet		8.739-02	2.574-01	6.872+00	7.42-06	5.100-05
2-Outlet		7.514-02	2.451-01	6.561+00	6.82-06	4.474-05
3-Inlet		6.113-02	2.311-01	6.211+00	3.21-06	1.994-05
3-Outlet		6.113-02	2.311-01	6.212+00	3.04-06	1.889-05
4-Inlet		6.656-02	2.366-01	6.346+00	2.20-06	1.396-05
4-Outlet		6.656-02	2.366-01	6.346+00	2.00-06	1.269-05
5-Inlet		5.096-02	2.210-01	5.965+00	9.02-06	5.380-05
5-Outlet		3.500-03	1.735-01	4.867+00	3.75-06	1.825-05
6-Inlet		3.500-03	1.735-01	4.867+00	5.49-06	2.672-05
6-Outlet		1.147-01	2.847-01	7.592+00	2.55-06	1.936-05
7-Inlet		1.147-01	2.847-01	7.592+00	1.04-06	7.896-06
7-Outlet		6.328-01	8.028-01	3.009+01	2.82-07	8.485-06
8-Inlet		6.328-01	8.028-01	3.009+01	2.62-07	7.883-06
8-Outlet		4.692+00	4.862+00	4.348+03	5.78-09	2.513-05
9-Inlet		4.715+00	4.885+00	4.434+03	3.65-09	1.619-05
9-Outlet		4.715+00	4.885+00	4.434+03	3.43-09	1.521-05
10-Inlet		4.715+00	4.885+00	4.434+03	2.16-09	9.578-06
10-Outlet		5.024+00	5.194+00	5.760+03	1.57-09	9.043-06
11-Inlet		5.024+00	5.194+00	5.760+03	3.26-09	1.878-05
11-Outlet		5.824+00	5.994+00	1.096+04	2.01-09	2.202-05

TABLE 5-7
VARIANCE IN TELLURIUM-131m PLATEOUT DISTRIBUTION

Section No.	$(M_{1k} \sigma_{x_1})^2$	$(M_{3k} \sigma_{x_3})^2$	$\sigma_{A_k}^2$	$1.65 \sqrt{\sigma_{A_k}^2}$ 10	Tellurium Median (Ci/cm ²)	Upper 95% Confidence Level (Ci/cm ²)
1-Inlet	1.49-02 ↓	5.824+00	5.839+00	9.709+03	3.62-09	3.515-05
1-Outlet		1.180+01	1.181+01	4.685+05	1.95-10	9.135-05
2-Inlet		8.739-02	1.023-01	3.371+00	1.40-05	4.719-05
2-Outlet		7.514-02	9.004-02	3.127+00	1.28-05	4.002-05
3-Inlet		6.113-02	7.603-02	2.851+00	6.04-06	1.722-05
3-Outlet		6.113-02	7.603-02	2.851+00	5.72-06	1.631-05
4-Inlet		6.656-02	8.146-02	2.958+00	4.14-06	1.224-05
4-Outlet		6.656-02	8.146-02	2.958+00	3.76-06	1.112-05
5-Inlet		5.096-02	6.586-02	2.651+00	1.70-05	4.507-05
5-Outlet		3.500-03	1.840-02	1.674+00	7.05-06	1.180-05
6-Inlet		3.500-03	1.840-02	1.674+00	1.03-05	1.724-05
6-Outlet		1.147-01	1.296-01	3.926+00	4.79-06	1.880-05
7-Inlet		1.147-01	1.296-01	3.926+00	1.95-06	7.655-06
7-Outlet		6.328-01	6.477-01	2.128+01	5.30-07	1.128-05
8-Inlet		6.328-01	6.477-01	2.128+01	4.93-07	1.049-05
8-Outlet		4.692+00	4.707+00	3.800+03	1.09-08	4.142-05
9-Inlet		4.715+00	4.730+00	3.877+03	6.87-09	2.663-05
9-Outlet		4.715+00	4.730+00	3.877+03	6.45-09	2.501-05
10-Inlet		4.715+00	4.730+00	3.877+03	4.07-09	1.578-05
10-Outlet		5.024+00	5.039+00	5.056+03	2.95-09	1.492-05
11-Inlet		5.024+00	5.039+00	5.056+03	6.13-09	3.100-05
11-Outlet		5.824+00	5.839+00	9.706+03	3.77-09	3.659-05

TABLE 5-8
SUMMARY OF AIPA PLATEOUT UNCERTAINTIES

Element ^(a)	Major Sources of Uncertainty ^(b)
Cesium	Metal sorptivity, direct release
Iodine	Metal sorptivity, graphite sorptivity, stable iodine release
Strontium	Direct release, mass transfer
Silver	Mass transfer, direct release
Cerium	Min R/B, mass transfer
Tellurium	Mass transfer
Barium/ Lanthanum	Min R/B, mass transfer

(a) In order of decreasing uncertainty.

(b) In order of decreasing uncertainty (e.g., for cesium, direct release is a lesser source of uncertainty than metal sorptivity).

TABLE 5-9
VARIABLES AFFECTING PLATEOUT DISTRIBUTION

Symbol	Name
x_1	Core gaseous release
x_2	Core metallic release
x_3	Mass transfer coefficient
x_4	Graphite sorptivity
x_5	Metal sorptivity

6. UNCERTAINTIES IN CESIUM AND STRONTIUM TRANSPORT DATA

In this section, estimates are made of the uncertainties associated with cesium and strontium transport data. These estimates were used in the calculations of Section 3. The statistical procedures used to obtain the uncertainties are described in the following subsection. Subsequent subsections present details on the treatment of the transport data and a summary of the uncertainties used in Section 3.

6.1. PROCEDURES FOR ESTIMATING UNCERTAINTIES

Uncertainties were determined for two quantities: diffusion or permeation coefficients and vapor pressures over sorptive materials. The functional forms used to represent the two quantities are as follows. For diffusion coefficient,

$$D = D_0 \exp\{-Q/RT\} \quad , \quad (6-1)$$

where D = diffusion coefficient (cm^2/sec),

D_0 = a constant (cm^2/sec),

Q = temperature coefficient or activation energy (cal/mole),

R = gas constant,

$= 1.987 \text{ cal/mole-}^\circ\text{K}$,

T = temperature ($^\circ\text{K}$).

For vapor pressure,

$$\log p = \alpha(T) + \beta(T) \log C \quad , \quad (6-2)$$

where p = vapor pressure (atm),

$\alpha(T)$, $\beta(T)$ = coefficients which may be temperature dependent,

C = concentration ($\mu\text{mole sorbate/g graphite}$).

The diffusion coefficient D (and D_0) may be replaced in Eq. 6-1 by the permeation coefficient π (and π_0), the use of which is more appropriate for characterizing cesium transport through graphite.

A least squares straight line fit is made for the data pairs (D,T) and (p,c) in the form

$$y = a + bx \quad , \quad (6-3)$$

where y is either $\ln D$ or $\log p$, and x is either $10^3/T(^{\circ}K)$ or $\log C$. The use of the natural logarithm of D and the base ten logarithms of p and C is arbitrary. The coefficients a and b are $\ln D_0$ or $\alpha(T)$ and $-Q/(1000R)$ or $\beta(T)$, respectively.

The uncertainties derived below are the standard deviations from the straight line fit of Eq. 6-3. Two such standard deviations were calculated. In one case, the standard deviation of the average estimated value of the dependent variable was obtained; in the other case, the standard deviation of a single estimated value of the dependent variable was obtained.

For the standard deviation of the average value of the dependent variable, the variance is given by (Ref. 2-1, p. 43)

$$S_o^2 = S_y^2 \left[\frac{\sum x_n^2 - 2x \sum x_n + kx^2}{k \sum x_n^2 - (\sum x_n)^2} \right] \quad , \quad (6-4)$$

where $\{x_n\}$ = data set of independent variables,

k = number in the set,

S_o^2 = variance of the average estimated value of the dependent variable,

x = the value of the independent variable at which S_o^2 is to be evaluated.

Also,

$$S_y^2 = \frac{\sum (\delta y_n)^2}{k - 2} \quad , \quad (6-5)$$

$$\sum (\delta y_n)^2 = \sum y_n^2 - \frac{(\sum y_n)^2 \sum x_n^2 - 2 \sum x_n y_n \sum x_n \sum y_n + k (\sum x_n y_n)^2}{k \sum x_n^2 - (\sum x_n)^2} \quad , \quad (6-6)$$

where $\{y_n\}$ = the corresponding data set of dependent variables.

For the standard deviation of a single estimated value of the dependent variable the variance is given by

$$S^2 = S_o^2 + S_y^2 \quad . \quad (6-7)$$

In applying Eqs. 6-4 and 6-7 to the uncertainty analysis, a judgment was made. When the samples on which measurements were made were representative of the reactor materials and subject to reactor conditions to develop the distribution of fission products measured, reporting the uncertainty in terms of the deviation of the average estimated value of the dependent variable was judged to be reasonable. The data used in these uncertainty analyses did not generally conform to these restrictions. In cases in which the data were subject to known variations which were nevertheless not quantitatively characterized, the use of the deviation of a single estimated value of the dependent variable was judged to be reasonable. Thus, in the hypothetical case of a transport coefficient suspected to depend on the various materials used in the measurement and for which the dependence cannot yet be characterized, a particular selection of one of the materials could easily result in transport coefficient values which would lie outside the deviation of the average estimated value of the dependent variable, but would fall within the

deviation of a single estimated value. The standard deviation of the single and average estimated values of the dependent variable are reported in subsequent sections.

In Section 6.3.3, a case arises in which experimental data from two laboratories are combined to derive an estimate of the uncertainty. However, the results from one of the laboratories are only available in the form of Eq. 6-3; as a consequence, the results of the other laboratory are also used in the form of Eq. 6-3. In this situation, the variance corresponding to the standard deviation of the average estimated value of the dependent variable can be shown to be

$$S_o^2 = 4S_{y_c}^2 \left[\frac{\sum x_i^2 - 2x \sum x_i + kx^2}{k \sum x_i^2 - (\sum x_i)^2} \right], \quad (6-8)$$

where

$$S_{y_c}^2 = \frac{(a_1 - a_2)k + (a_1 - a_2)(b_1 - b_2) \sum x_i + (b_1 - b_2)^2 \sum x_i^2}{2(k - 2)}, \quad (6-9)$$

and a_i and b_i are the coefficients of Eq. 6-3 for the different laboratories ($i = 1$ or 2), and k , x_i , and x_i^2 represent the combined quantities for the experiments in both laboratories. In the absence of any information about the latter quantities for the experiments of one of the laboratories, the assumption was made that the known values of these quantities for the experiments in the other laboratory apply equally well to the experiments in both laboratories. In deriving Eq. 6-8, further assumptions were made; i.e.,

$$\frac{\partial a_1}{\partial y_i} \equiv \frac{\partial a_2}{\partial y_i},$$

$$\frac{\partial b_1}{\partial y_i} \equiv \frac{\partial b_2}{\partial y_i}.$$

Note that the variance of Eq. 6-8 applies to the least squares straight line given by

$$y = \gamma + \delta x \quad , \quad (6-10)$$

$$\begin{aligned} \text{where } \gamma &= (a_1 + a_2)/2, \\ \delta &= (b_1 + b_2)/2. \end{aligned}$$

A transformation of Eqs. 6-4 and 6-7 in terms of the deviations of coefficients a and b of Eq. 6-3 is convenient for applications. Thus, Eq. 6-4 can be written as

$$S_o^2 = S_a^2 - 2\rho_{ab} S_a S_b x + S_b^2 x^2 \quad , \quad (6-11)$$

where the definitions of S_a^2 and S_b^2 are immediately obtained upon comparison of Eqs. 6-4 and 6-11, and

$$\rho_{ab} = \bar{x}_n / \sqrt{\overline{x_n^2}} \quad , \quad (6-12)$$

$$\left. \begin{aligned} \bar{x}_n &= (\sum x_n)/k \quad , \\ \overline{x_n^2} &= (\sum x_n^2)/k \quad . \end{aligned} \right\} \quad (6-13)$$

A further transformation is necessary to allow the results of the present uncertainty analysis to be used in release calculations performed with the usual formats for input data. The necessity for this transformation is restricted to the diffusion (or permeation) coefficient. In place of Eq. 6-1, the following functional form is used in release calculations:

$$D = D_{1000} \exp \left\{ -\frac{Q}{R} (1/T - 1/1273) \right\} \quad ,$$

where D_{1000} (cm^2/sec) is the diffusion coefficient at 1000°C (1273°K), and other symbols are defined above. If the least squares straight line is used in the form

$$y = a' + b'x' \quad , \quad (6-15)$$

then y retains the interpretation given following Eq. 6-3, and

$$a' = \ln D_{1000} \quad ,$$

$$b' = Q/(1000R) \quad ,$$

$$x' = \{(1000/1273) - [1000/T(^{\circ}\text{K})]\} \equiv c - x \quad .$$

Using these definitions, it can be shown that

$$S_o^2 = S_a^2 - 2\rho_{a'b'} S_a S_b (x') + S_b^2 (x')^2 \quad , \quad (6-16)$$

where

$$S_{a'}^2 = S_y^2 \left[\frac{\Sigma x_n^2 - 2c\Sigma x_n + kc^2}{k\Sigma x_n - (\Sigma x_n)^2} \right] \quad , \quad (6-17)$$

$$S_{b'}^2 = S_b^2 \quad , \quad (6-18)$$

$$\rho_{a'b'} = \frac{c - \bar{x}_n}{\sqrt{c^2 - 2c\bar{x}_n + \bar{x}_n^2}} \quad . \quad (6-19)$$

To obtain the variance corresponding to the standard deviation of a single estimated value of the dependent variable, Eq. 6-7 is used in conjunction with Eq. 6-16. Note carefully that where this combination has

been used in Sections 6.2, 6.3, and 6.4, the first number appearing in the expressions for S^2 is the value of $S_a^2 + S_y^2$.

In Section 6.2.4, a case arises in which sorption ratios must be used to estimate the uncertainty in sorption on fuel rod matrix material. The sorption ratio is given by

$$\phi = \frac{C_m}{C_g} \quad , \quad (6-20)$$

where ϕ = sorption ratio,

C_i = sorbate concentration in matrix material ($i = m$) or graphite
($i = g$) (μ moles sorbate/g sorbing material).

The sorption ratio data are combined with isotherm data for sorption on graphite to make the desired estimate. By using Eqs. 6-2 and 6-20, the variance in the dependent variable for sorption on fuel rod matrix material is obtained in terms of the variance for sorption on graphite, the variance in the sorption ratios, and the values of the sorptivity constants for the matrix material and graphite. The detailed expressions are given in Section 6.2.4.

6.2. UNCERTAINTIES IN CESIUM TRANSPORT DATA

Uncertainties in cesium transport data are obtained for transport-related processes which are modeled in metallic fission product release calculations (Ref. 3-2). These uncertainties were used to determine some of the values shown in Table 3-6.

6.2.1. Kernel Retention of Cesium

A clear demonstration has been given that a fraction of the cesium fission product in kernels is trapped; i.e., in postirradiation annealing experiments at relatively long anneal times, the fractional release of

cesium apparently approaches an asymptotic value of less than one (Ref. 6-1). This behavior has also been found at GA in postirradiation annealing experiments on particles irradiated in capsules for up to about 270 effective full-power days and in Peach Bottom gamma scans covering particles irradiated for 798 effective full-power days. Data from postirradiation anneals at Dragon Project also show cesium retention by kernels in particles irradiated for as long as 207 days (Refs. 4-9, 6-2). The particles used in these examinations include those with carbide and oxide kernels and those with FIMA values mainly in the range between 0% and 11%. In this section, the data are presented in graphic form, and a summary treatment of the data suitable for use in the AIPA study is given.

The kernel retention data for cesium are shown in Fig. 6-1 in the form of trapped fraction (TF) versus temperature without regard to kernel type, FIMA, or length of irradiation time. Data for kernels with FIMA of less than 11% are shown. For values of FIMA $\leq$ 10%, the recommended function for the AIPA analyses is mathematically represented by

$$TF = 0.65 \quad ; \quad T \leq 1400^{\circ}K \quad , \quad (6-21)$$

$$TF = 4.3 - 0.0026T \quad ; \quad 1400 \leq T(^{\circ}K) \leq 1650 \quad . \quad (6-22)$$

As can be seen in Fig. 6-1, this is clearly a conservative model of the data provided that the difference between the time of irradiation for the particles and that for the full fuel cycle is not significant for the trapped fraction. The deviation of the upper and lower bounds from the values of Eqs. 6-21 and 6-22 is taken to be 0.5 (TF). For values of FIMA $>$ 10%, the present recommendation is to set TF to zero. Further analysis and additional data will be needed to remove this latter conservatism.

For the AIPA analyses a mean value of the trapped fraction of 0.5 (corresponding to a fuel temperature of 1200°C) was used. The trapped fraction was applied to BISO fuel particles only, since FIMA for most TRISO

particles exceeds 10%. Thus, in this particular instance, a conservative, off-nominal value was chosen for an input variable and used in nominal release calculations.

6.2.2. Cesium Diffusion in Pyrocarbon

The data for the uncertainty analysis for diffusion of cesium in low-temperature isotropic (LTI) pyrocarbon are shown in Fig. 6-2. The data were obtained from GA and ORNL annealing tests (Ref. 6-1). In this figure, the least squares straight line for all the data is given by a continuous line. The upper and lower 95% confidence bounds are given by broken lines. A summary of the uncertainty analysis of the data is given in Section 6.5. Because the mean curve of Fig. 6-2 has to represent materials with a variety of properties and irradiation histories, the variance used is that for the deviation of a single point S^2 from the mean curve.

6.2.3. Cesium Sorption on Graphite

The uncertainty associated with the sorption of cesium on graphite is estimated from the combined data of seven experiments with H-327 graphite (Ref. 6-3). These experiments were conducted using the Knudsen cell method; the experiments differ in the use of different sample and cell sizes, cell composition, and in the case of one experiment, grain size of the sorbent. The data from these experiments used in the uncertainty analysis correspond to a temperature of 1300°K. The uncertainty at other temperatures is assumed to be the same. It should be noted that the available documented data for sorption of cesium on graphite are limited to the results of one laboratory and one method. This situation adds an uncertainty which was not quantified in this study.

The data are presented in Fig. 6-3 in terms of the logarithms of pressure (atm) and concentration ($\mu\text{mole/g}$) for cesium sorption on unirradiated H-327 graphite at 1300°K. A summary of the analysis of the

data of Fig. 6-3 is given in Section 6.5. The data are derived according to sorbate concentration regions in which either the Freundlich or Henrian isotherm is an adequate description. The 95% confidence bounds are also plotted in Fig. 6-3. In the Freundlich region, the bounds are based on the standard deviation of the mean from the least squares straight line, whereas in the Henrian region, the bounds are based on the standard deviation of a single point from the least squares straight line. The use of different measures of uncertainty in the Freundlich and Henrian regions is based on the greater number of data in the Freundlich region.

This analysis is based on sorption data for H-327 graphite, whereas the current reference graphite is H-451. However, for lack of extensive data on H-451 graphite, the assumption is made that the uncertainties associated with measurements on H-327 can be applied to H-451 graphite.

6.2.4. Cesium Sorption on Fuel Rod Matrix Material

There is only one satisfactory set of measurements of the sorption of cesium on a reference fuel rod matrix material (M-205). The data can be used to derive isotherms over an appropriate range of temperatures; however, good estimates of the uncertainties associated with these data cannot be readily made. Therefore, the estimates of the uncertainties have been made by using the more numerous data on isothermal sorption ratios combined with the uncertainties obtained for sorption of cesium on graphite. The sorption ratio $\emptyset$ is the ratio of the sorbate concentration in the matrix material to that in the graphite, as defined by Eq. 6-20. An approximation in the analysis has been introduced by assuming the sorption ratio for cesium to be independent of sorbate concentration and temperature. In all sorption ratios referenced, the graphite is H-327.

The sorption ratio data are given in Table 6-1. In the case of double $\emptyset$ entries, the average value was used. To obtain the estimated uncertainty in the vapor pressure over the matrix by combining the results of Table 6-1

with the uncertainties determined for sorption on graphite in Section 6.2.3, the following relations are used. For the Freundlich region,

$$S^2(\log p_m) = \frac{\beta_m}{\beta_g} \times S^2(\log p_g) + \beta^2 S^2(\log \emptyset) \quad , \quad (6-23)$$

For the Henrian region,

$$S^2(\log p_m) = S^2(\log p_g) + S^2(\log \emptyset) \quad . \quad (6-24)$$

For the transition concentration,

$$S^2(\log C_t^m) = S^2(\log C_t^g) + S^2(\log \emptyset) \quad . \quad (6-25)$$

In these equations, m refers to matrix and g refers to graphite. For cesium, $\beta_m = \beta_g$, with $\beta = D + E/T$, where D and E are constants. For simplicity in calculation, β^2 may be replaced by 13, corresponding to $T = 1000^\circ\text{K}$ for the case of cesium sorption. This is a conservative simplification and results in the uncertainty being independent of temperature, as previously assumed for cesium sorption on graphite. The uncertainties are given in Section 6.5.

6.2.5. Cesium Migration in Graphite

The uncertainty analysis of cesium migration in graphite was based on the compilation of data shown in Fig. 6-4. The data comprise a variety of types of experiments, including in-pile as well as out-of-pile tests. A reasonable approximation is to regard all these data as representing permeation coefficients π . In some cases (Refs. 6-4 and 6-5), the data are reported as permeation coefficients, which are coefficients derived from flux measurements, and in other cases (Refs. 6-6, 6-7, 6-8), the derived diffusion coefficients are from long-time experiments. The data of Fig. 6-4 are associated with concentrations of less than $0.2 \mu\text{moles/g}$ graphite. A concentration dependence of cesium migration exists, with the transport

of cesium being greater at larger cesium concentrations. The present uncertainty analysis applies to only concentrations below 0.2 $\mu\text{mole/g}$; the concentrations encountered in HTGRs under normal operating conditions are generally at or below this level.

The least squares straight line and the 95% confidence bounds corresponding to the deviation of a single point are shown in Fig. 6-4. The uncertainty expressions are given in Section 6.5.

6.3. UNCERTAINTIES IN STRONTIUM TRANSPORT DATA

Uncertainties in strontium transport data are obtained for transport-related processes which are modeled in metallic fission product release calculations (Ref. 3-2). These uncertainties were also used to determine some of the values in Table 3-6.

6.3.1. Strontium Diffusion in Oxide Fuel Kernels

There are very few data on the diffusion of strontium in oxide fuel kernels. These data are shown in Fig. 6-5. The uncertainty analysis is based on the average values of D' , which is the reduced diffusion coefficient, given by the ratio D/a^2 , where D is the diffusion coefficient and a is the radius of the equivalent sphere of the particle or grain through which diffusion occurs. Figure 6-5 shows the mean curve and the 95% confidence bounds for the deviation of the average estimated value of the independent variable. With such large uncertainties predicted for the temperature ranges of interest in normal HTGR operation, the use of the deviation of a single estimated value of the independent variable to obtain the confidence bounds would not sensibly alter the bounds shown in Fig. 6-5. The uncertainties are given in Section 6.5.

6.3.2. Strontium Sorption on Fuel Rod Matrix Material

The uncertainty analysis of strontium sorption on fuel rod matrix material was based on the compilation of data shown in Fig. 6-6. The data were obtained from tests at GA (Refs. 6-10 and 6-11). The reference curves and the 95% confidence bounds are shown for three temperatures. A summary of the uncertainty analysis of the data is given in Section 6.5. The uncertainty in the Henrian region is obtained by extending the 95% confidence bounds parallel to the reference Henrian curves.

At the time that the estimates of uncertainty of strontium sorption on fuel rod matrix material were made, there were insufficient data to obtain a reliable estimate. However, some qualitative assessment of the analysis is possible by considering recent measurements (Ref. 6-12) of strontium sorption on fuel rod matrix material. These measurements show that the isotherms of Fig. 6-6 yield vapor pressures that are equal to or greater than the vapor pressures derived from the recent measurements. In particular, since the recent measurements extend well into the Henrian region, i.e., to sorbate concentrations of less than 22.8 $\mu\text{moles/g}$ (Ref. 6-11), they confirm the estimates of the Henrian region isotherms shown in Fig. 6-6, which were based on the assumption that the transition from the Freundlich to the Henrian region occurred at the same vapor pressure as that occurring with strontium sorption on H-327 graphite.

6.3.3. Strontium Sorption on Graphite

The estimation of uncertainty in the data for sorption of strontium on graphite was determined from GA and Harwell data. The latter data constitute a determination of strontium sorptivity on H-327 graphite in an independent laboratory using a somewhat different detection technique (Ref. 6-11). Unfortunately, only a summary description of Harwell experiments in terms of the parameters of Eq. 6-3 is available. Therefore, the estimate of uncertainty is made according to Eqs. 6-8 and 6-9. The analysis applies

to only the Freundlich region of the isotherm. With the GA data, no definitive evidence exists by which to determine a transition to a Henrian region of the isotherm. The Harwell data exhibit a transition to Henrian behavior at a sorbate concentration of $0.12 \mu\text{mole/g}$ between 1500° and 1700°K . This transition has been incorporated in the reference curves, as discussed below.

The results of estimating the uncertainty in strontium sorption on H-327 graphite are presented in Table 6-2. These uncertainties have been applied to the reference isotherms, as shown in Fig. 6-7. The reference isotherms were established without attempting to incorporate the Harwell results in the Freundlich region. The uncertainty in the Henrian region is obtained by extending the 95% confidence bounds parallel to the reference Henrian curves. A summary of the uncertainty analysis is given in Section 6.5.

6.3.4. Strontium Diffusion in Graphite

The diffusion coefficients measured in a variety of graphite samples in the laboratory and under irradiation conditions are shown in Fig. 6-8. These data have previously been discussed (Ref. 6-11). The data of Fig. 6-8 were selected from a larger data set and were derived from experiments in which the strontium concentration was less than about $1 \mu\text{mole/g}$ graphite. Concentrations in the range below $1 \mu\text{mole/g}$ correspond to strontium concentrations expected under normal HTGR operation concentrations. At concentrations of sorbed strontium in excess of $1 \mu\text{mole/g}$, the diffusion coefficient for migration of strontium in graphite depends on the sorbate concentration.

The analysis of the data of Fig. 6-8 is given in Section 6.5. The least squares straight fit to the data, the 95% confidence bounds as applied to the least squares straight line fit, and the selected reference curve (Ref. 6-11) are also shown in Fig. 6-8. The reference curve differs

from the least squares straight line fit by intention; the selected reference curve is a conservative fit to the data. As can be seen from the figure, the reference curve lies just within the upper 95% confidence bound as applied to the least squares fit. Since there is a relatively large number of experimental data which represent a variety of graphite types in a variety of environments, the selection of the standard deviation of the average estimated value of the dependent variable from the least squares fit is justified as the measure of uncertainty.

6.4. LIMITATIONS OF THE ANALYSES

In treating the release of cesium from fuel particles, the assumption has frequently been made that the rate-controlling process is transport through the pyrocarbon coatings of the fuel particle, and therefore only this process need be considered in release calculations. There is evidence (Refs. 6-1, 6-2, 6-18) that a fraction of the cesium is retained by the kernel at specific temperatures and burnups, which will contribute to a smaller calculated release. Although this phenomenon is included in the uncertainty analysis in Section 6.2.1, the treatment is conservative. There are further uncertainties associated with cesium transport in fuel particle kernels. In recent and continuing experiments (Ref. 6-12), the diffusion coefficients for cesium in oxide fuel kernels have been measured and found to be quite small at relatively small burnups. The effects of these results on release calculations have yet to be determined. More importantly, the relation between these new diffusion data, which have been obtained by using bare kernels, and the deduced retention of cesium by kernels, based principally on in-pile experiments and tests with intact particles, remains to be established before the part played by the kernel in cesium release can be confidently taken into account.

The data analyzed for diffusion of cesium in pyrocarbon in Section 6.2.2 have not been classified according to structural differences in the pyrocarbon or effects of neutron fluence, although these factors do

influence the speed of transport for cesium in pyrocarbons. The usual characterization of pyrocarbon coatings in terms of density, anisotropy, and crystalline size is now believed to be insufficient for correlation of pyrocarbon properties with fission product release (Ref. 6-1). Thus, additional work is required before classification of pyrocarbons in regard to diffusion processes can be readily used for predicting fission product release. This means that the scatter in the available data cannot be adequately reduced by means of a functional relation between diffusion coefficients and pyrocarbon properties. Furthermore, changes in coating properties resulting from neutron irradiation have to be considered.

There are a number of limitations associated with the uncertainty analyses for sorption of cesium on graphite (Section 6.2.3) and on fuel rod matrix material (Section 6.2.4). For sorption of cesium on graphite, the analyzed data were obtained using a single experimental method, a Knudsen cell and mass spectrometer combination (Ref. 3-1). In addition, the data were from experiments on unirradiated graphite; data on irradiated graphite are available (Ref. 6-12), but were not included in the uncertainty analysis since no provision currently exists for using data on irradiated graphite in the cesium release calculations. The data for sorption of cesium on graphite from which the uncertainty was derived apply to H-327 graphite; if H-451 (reference) graphite sorption data are used in release calculations, then the assumption must be made that the uncertainties for the two types of graphites are the same. A further assumption was made in Section 6.2.3 that the uncertainties determined for cesium sorption in graphite at 1300°K can be applied at all other temperatures of interest in the release calculations for the normal operating conditions of the HTGR. The experimental sorption data are corrected for a grain size effect; no uncertainty associated with this correction has been included.

In the case of sorption of cesium on fuel rod matrix material, the uncertainty estimates were made on the basis of isothermal sorption ratio measurements. An insufficient number of isotherm measurements prevented using these as the basis for the uncertainty analysis.

In the analysis of the data for cesium migration in graphite, the included data were for cesium concentrations below 0.2 μ mole cesium/g graphite. Transport coefficients derived from concentration profiles in short-term experiments were excluded from the data set (Section 6.2.5).

The uncertainties determined for strontium data may generally be underestimated. For the diffusion of strontium in oxide fuel kernels, few data are available and these exist for only one burnup value. Thus, the uncertainty analysis in this case does not include the potentially important effect of burnup on the transport of strontium in the oxide fuel kernels. Certain deficiencies associated with the treatment of data for strontium sorption on fuel rod matrix material are discussed in Section 6.3.2. As in the case for cesium sorption, data on irradiated graphites for sorption of strontium have not been included in the uncertainty analysis.

In the general treatment of the data, the uncertainties were assumed to be in the dependent variable. Thus, in uncertainty analyses of diffusion, the uncertainty was taken to be in the diffusion coefficient rather than in the temperature. In analyses of sorption, the uncertainty was taken to be in the vapor pressure rather than in the sorbate concentration.

For some of the matters discussed above, such as the effect of neutron fluence on diffusion of cesium in pyrocarbon, and for other matters not mentioned above, such as the uncertainty about the qualitative and quantitative description of cesium migration in graphite, the uncertainties derived may to a large extent incorporate these effects, since the corresponding data are derived from situations in which, in one case, a range of burnups occur and in the other case, the several modes of cesium transport in graphite are operative. Thus, a thorough understanding of these effects would aid in reducing the uncertainty.

Several processes were omitted from the uncertainty analyses. Some of these omissions result from (1) the absence of a need in the current AIPA work because of assumptions in the models about the relative importance of processes and (2) the decision to accept a conservatism introduced as a result of the omission. For other processes, the data are totally inadequate or do not exist, and these cases are considered to be beyond the scope of the present effort. The omitted processes include the following:

1. Cesium diffusion in oxide and carbide kernels.
2. Strontium diffusion in carbide kernels.
3. Strontium diffusion in particle coatings.
4. Diffusion in matrix material.
5. Transport of cesium and strontium by transverse flow.
6. Effect of impurities on transport and sorption.
7. Interaction of various fission products in transport and sorption.
8. Sorption and transport in a neutron flux field.
9. Effect of graphite oxidation (burn-off) on transport and sorption.

6.5. SUMMARY: NOMINAL VALUES AND UNCERTAINTIES FOR CESIUM AND STRONTIUM TRANSPORT DATA

6.5.1. Kernel Retention of Cesium

For kernel retention of cesium,

$$S(TF) = 0.25 \quad , \quad (6-26)$$

where $S(TF)$ is the standard deviation in the trapped fraction TF , and the uncertainty is applied to fuel particles which have experienced $FIMA \leq 10\%$. For particles with $FIMA > 10\%$, the value of S and the trapped fraction are taken to be zero. Mean values are discussed in Section 6.2.1.

6.5.2. Cesium Diffusion in Pyrocarbon

For cesium diffusion in pyrocarbon (cf. Eq. 6-1),

$$D = 0.50 \exp[-75,950/RT] \quad , \quad (6-27)$$

$$S^2(\ln D) = 2.20 - 4.067 [0.7855 - (10^3/T)] + 9.269 [0.7855 - (10^3/T)]^2, \quad (6-28)$$

where $S^2(\ln D)$ is the variance in the natural logarithm of the diffusion coefficient D , and temperature T is in $^{\circ}K$; $S(\ln D)$ is the standard deviation of a single estimated value of the dependent variable.

6.5.3. Cesium Sorption on Graphite

For cesium sorption on graphite in the Freundlich region (cf. Eq. 6-2),

$$\ln p = (7.09 - \frac{27,600}{T}) + (-0.50 + \frac{4,100}{T}) \ln C \quad , \quad (6-29)$$

$$S_o^2(\log p) = 0.00134 + 0.00264 (\log C) + 0.01519 (\log C)^2 \quad , \quad (6-30)$$

where $S_o^2(\log p)$ is the variance in the base 10 logarithm of the vapor pressure p in atm, and C is the sorbate concentration in $\mu\text{mole cesium/g}$ graphite; $S_o(\log p)$ is the standard deviation of the average estimated value of the dependent variable. In the Henrian region,

$$\ln p = (7.09 - \frac{27,600}{T}) + (-1.50 + \frac{4,100}{T}) \ln (0.3) + \ln C, \quad (6-31)$$

$$S_o^2(\log p) = 0.101 + 0.050 (\log C) + 0.0212 (\log C)^2. \quad (6-32)$$

6.5.4. Cesium Sorption on Fuel Rod Matrix Material

In the Freundlich region, cesium sorption on fuel rod matrix material is given by (cf. Eq. 6-2)

$$\ln p = (8.39 - \frac{38,300}{T}) + (-0.50 + \frac{4,100}{T}) \ln C, \quad (6-33)$$

$$S_o^2(\log p) = 0.1643 + 0.00264 (\log C) + 0.01519 (\log C)^2. \quad (6-34)$$

In the Henrian region,

$$\ln p = (8.39 - \frac{38,300}{T}) + (-1.50 + \frac{4,100}{T}) \ln (4.0) + \ln C, \quad (6-35)$$

$$S_o^2(\log p) = 0.114 + 0.050 (\log C) + 0.0212 (\log C)^2. \quad (6-36)$$

6.5.5. Cesium Migration in Graphite

For cesium migration in graphite (cf. Eq. 6-1),

$$\pi = 3.62 \times 10^{-4} \exp[-25,300/RT], \quad (6-37)$$

$$S^2(\ln \pi) = 2.32 + 0.2607 [0.7855 - (10^3/T)] \\ + 1.353 [0.7855 - (10^3/T)]^2, \quad (6-38)$$

where $S^2(\ln \pi)$ is the variance in the natural logarithm of the permeation coefficient π .

6.5.6. Strontium Diffusion in Oxide Kernels

For strontium diffusion in oxide kernels (cf. Eq. 6-1),

$$D' = 1.96 \times 10^9 \exp[-142,600/RT] \quad , \quad (6-39)$$

$$S_O^2(\ln D') = 11.06 - 97.31 [0.7855 - (10^3/T)] \\ + 218.1 [0.7855 - (10^3/T)]^2 \quad . \quad (6-40)$$

6.5.7. Strontium Sorption on Fuel Rod Matrix Material

For strontium sorption on fuel rod matrix material in the Freundlich region (cf. Eq. 6-2),

$$\ln p = (42.8 - \frac{149,000}{T}) + (-8.52 + \frac{28,500}{T}) \ln C \quad , \quad (6-41)$$

$$S_O^2(\log p) = 5.16 - 5.812 (\log C) + 1.646 (\log C)^2 \quad , \quad (6-42)$$

where the variance for $T = 1273^\circ\text{K}$ is chosen for simplicity to be used in calculations at all temperatures.

In the Henrian region,

$$\ln p = (42.8 - \frac{149,000}{T}) + (-9.52 + \frac{28,500}{T}) \ln (22.8) + \ln C \quad , \quad (6-43)$$

$$S^2(\log p) \approx 0.3 \quad . \quad (6-44)$$

6.5.8. Strontium Sorption on Graphite

In the Freundlich region, strontium sorption on graphite is given by (cf. Eq. 6-2)

$$\ln p = (9.44 - \frac{42,400}{T}) + (-0.57 + \frac{4,590}{T}) \ln C \quad , \quad (6-45)$$

$$S_O^2(\log p) = 0.0239 - 0.0321 (\log C) + 0.0302 (\log C)^2 \quad , \quad (6-46)$$

where the variance for $T = 1300^\circ\text{K}$ is chosen.

In the Henrian region,

$$\ln p = (9.44 - \frac{42,400}{T}) + (-1.57 + \frac{4,590}{T}) \ln (0.12) + \ln C \quad , \quad (6-47)$$

$$S^2(\log p) \approx 0.079 \quad . \quad (6-48)$$

6.5.9. Strontium Diffusion in Graphite

For strontium diffusion in graphite (cf. Eq. 6-1),

$$D = 166 \exp[-64,000/RT] \quad , \quad (6-49)$$

$$S_O^2(\ln D) = 0.07982 - 0.6397 [0.7855 - (10^3/T)] \\ + 3.161 [0.7855 - 10^3/T]^2 \quad . \quad (6-50)$$

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Fig. 6-1. Kernel retention of cesium: trapped fraction (TF) vs temperature

Fig. 6-2. Diffusion coefficients of cesium in LTI pyrocarbon

Fig. 6-3. Cesium sorption on unirradiated H-327 graphite at 1300°K

Fig. 6-4. Data on cesium migration in graphite

Fig. 6-5. Reduced diffusion coefficient for strontium diffusion in oxide fuel kernels

Fig. 6-6. Isotherms for strontium sorption on fuel rod matrix

Fig. 6-7. Isotherms for strontium sorption on H-327 graphite

Fig. 6-8. Data on strontium diffusion in graphite

TABLE 6-1
ANALYSIS OF SORPTION RATIO DATA FOR CESIUM: SORPTION RATIOS
RELATIVE TO H-327 GRAPHITE

Material	Binder ^(a)	T(°C)	Matrix Concentration (μ mole/g)	ϕ	Measurement Means ^(b)
M-205	P. pitch	1200	Isotherm	18	MS
M-205	P. pitch	1200	26	10	Anneal
FSV ^(c)	C. pitch	1200	21	15	Anneal
FSV ^(c)	C. pitch	1200	33	12	Anneal
FSV ^(c)	C. pitch	1200	38	14	Anneal
M-155	C. pitch	1200	24	17	Anneal
M-156	C. pitch	1200	23	16	Anneal
M-266	P. pitch	1200	27	10	Anneal
M-259	P. pitch	1200	39	11	Anneal
A-240	P. pitch	1200	46	12	Anneal
B-018	P. pitch	1200	10	21	Anneal
M-224	C. pitch	1200	30	13	Anneal
Calcined petro- leum coke	C. pitch	1202	0.98	20,10	FTE
Calcined petro- leum coke	C. pitch	963	3.8	10,20	FTE
Calcined petro- leum coke	C. pitch	1057	3.8	5,10	FTE
Calcined petro- leum coke	C. pitch	954	0.02	14,11	FTE-P

(a) P. pitch = petroleum pitch; C. pitch = coal tar pitch.

(b) MS = mass spectrometer; FTE = Peach Bottom fuel test element diffusion sample; FTE-P = Peach Bottom fuel test element particle sample.

(c) FSV = Fort St. Vrain.

TABLE 6-2
ANALYSIS FOR COMBINED GA AND AERE DATA FOR STRONTIUM SORPTION ON H-327 GRAPHITE

T(°K)	GA (a)		AERE (b)		Variance (c)
	a ₁	b ₁	a ₂	b ₂	
1300	-10.0649	2.9608	(-10.6645) ^(d)	(2.59)	$S_o^2(\log p) = 0.0239 - 0.0321(\log C) + 0.0302(\log C)^2$
1500	-8.1763	2.49	-8.1763	2.59	$S_o^2(\log p) = 0.00367 - 0.00523(\log C) + 0.00728(\log C)^2$
1800	-6.1303	1.98	(-6.397)	(2.59)	$S_o^2(\log p) = 0.0191 + 0.0436(\log C) + 0.0450(\log C)^2$

(a) For GA,

$$\ln p = \left(9.44 - \frac{4.24 \times 10^4}{T} \right) + \left(-0.57 + \frac{4590}{T} \right) \ln C ,$$

$$a_1 = \left(9.44 - \frac{4.24 \times 10^4}{T} \right) (2.3026) ,$$

$$b_1 = \left(-0.57 + \frac{4590}{T} \right) .$$

(b) For AERE,

$$\log p = \left(4.697 - \frac{1.997 \times 10^4}{T} \right) + 2.59 \log C ,$$

$$a_2 = 4.697 - \frac{1.997 \times 10^4}{T} ,$$

$$b_2 = 2.59 .$$

(c) According to Eqs. 6-8 and 6-9.

(d) The range of AERE experiments was between 1500° and 1700°K; quantities in parentheses are for temperatures outside this range.

7. UNCERTAINTIES IN FISSION GAS RELEASE DATA

In this section, the results of fission gas release data uncertainty studies are presented. To a large extent, the data were obtained in experiments conducted by GA. The data are organized in a manner useful for the methods used in calculating the gaseous fission product inventories in the HTGR (Ref. 4-1). Thus, one begins with the fission product release to birth rate ratio R/B for krypton-85m and xenon-138 at 1100°C. The corresponding value for other isotopes of krypton and xenon is obtained by using an assumed dependence of R/B on nuclide half-life. To obtain values of R/B at other temperatures, an activation energy is employed for each element. Additionally, the dependencies of R/B on hydrolysis are considered in a general sense for all elements and their isotopes.

7.1. PROCEDURES FOR ESTIMATING UNCERTAINTIES

The uncertainties in fission gas release data are generally determined in the manner outlined in Section 6.1. Additional procedures used in this section are described below.

The expression usually used for the release fraction R/B as a function of temperature is (Ref. 4-1)

$$R/B = (R/B)_m + [(R/B)_o - (R/B)_m] \exp \left[\frac{Q}{R} \left(\frac{1}{T_o} - \frac{1}{T} \right) \right] , \quad (7-1)$$

where R/B = release fraction at temperature T (°K),
 $(R/B)_o$ = release fraction at temperature T_o (°K),
 $(R/B)_m$ = a fixed value of the release fraction at very low temperatures,

Q = activation energy (cal/mole),

R = gas constant.

With normal sets of data, i.e., R/B versus T at a variety of temperatures, Eq. 7-1 could be treated in the form of Section 6.1:

$$y = a' + b'x' \quad , \quad (7-2)$$

where $y = \ln[(R/B) - (R/B)_m]$,

$a' = \ln[(R/B)_o - (R/B)_m]$,

$b' = Q/1000R$,

$x' = (1000/T_o - 1000/T) \equiv c' - x$.

As a result, the variance in $\ln[(R/B) - (R/B)_m]$ would become

$$S_o^2 = (S_{a'})^2 - 2\rho_{a'b'}S_{a'}S_{b'} + (S_{b'})^2 (x')^2 \quad , \quad (7-3)$$

where S_i is the standard deviation in i , which is defined in Section 6.1, and

$$\rho_{a'b'} = \frac{c' - \bar{x}_n}{\sqrt{(c')^2 - 2c'\bar{x}_n + \bar{x}_n^2}} \quad , \quad (7-4)$$

where Eq. 7-4 differs from Eq. 6-19 in that $c' = 1000/1371$. Thus, T_o in Eq. 7-1 is 1100°C .

However, data are not uniformly available in sets of R/B versus T over a range of temperatures. Rather, many data have been obtained at the reference temperature of 1100°C , and fewer data have been obtained at other temperatures. Thus, the analyses have been separately performed on R/B data at 1100°C to obtain an uncertainty in $(R/B)_o$ and on limited data with temperature as the independent variable to obtain an uncertainty in Q . Advantage is taken of the disparity in numbers of data to make the simplifying assumption that $c' \approx \bar{x}_n$ in Eq. 7-4, and thus that $\rho_{a'b'} \approx 0$. Equation 7-3 then becomes

$$S_o^2 \approx (S_{a'})^2 + (S_{b'})^2 (x')^2 \quad (7-5)$$

Note that if the error in $(R/B)_m$ is regarded as zero, then S_o^2 is the variance in $\ln(R/B)$ and $(S_{a'})^2$ is the variance in $\ln(R/B)_o$.

At the time the uncertainty analyses of fission gas data were made, the form of the uncertainties required for use in Section 4 was not defined. Thus, the uncertainties in R/B and Q were obtained for the quantities themselves rather than their natural logarithms, as shown in Eq. 7-5. In this case, the variance in R/B is given by

$$S_o^2 = [(S_{\alpha'})^2 + \alpha^2 (x')^2 (S_{b'})^2] e^{2bx'} \quad (7-6)$$

where $(S_{\alpha'})^2$ = variance in $(R/B)_o$,
 $\alpha = (R/B)_o$,

and the corresponding approximation $\rho_{\alpha,b} \approx 0$ is made and the uncertainty in $(R/B)_m$ is taken to be zero. In general, there is no simple relation between Eqs. 7-5 and 7-6. However, if deviations from mean values are small, e.g., <20%, then the standard deviation of the natural logarithm of a quantity is approximately the same as the fractional standard deviation of the quantity.

The half-life dependence of R/B was treated by using the relation

$$(R/B) \propto t_{1/2}^n \quad (7-7)$$

where $t_{1/2}$ is the half-life for a specific isotope, and n is the exponent whose uncertainty is determined in this analysis.

7.2. R/B FOR PARTICLES WITH FAILED COATINGS

The uncertainties associated with the reference values of R/B at 1100°C for krypton-85m and xenon-138 in failed particles $(R/B)_f$ have been determined by analyzing several sets of data and combining these with

previously obtained results (Ref. 7-1). Analyses were performed on four sets of data: data from the HRB-4, HRB-5, and HRB-6 capsules (Ref. 7-2), F-30 capsule (Ref. 7-3) data, data from laser drilled particles, and Peach Bottom Core 1 data.

7.2.1. HRB-4, HRB-5, and HRB-6 Data

The data for krypton-85m from the HRB-4, HRB-5, and HRB-6 capsules (Ref. 7-2) are for UC_2 TRISO and ThO_2 BISO particles. The measured R/B values are normalized to the failure fraction for fertile particles when the fraction is nonzero or to the failure fraction for fissile particles when the failure fraction for fertile particles is zero. At end of life (EOL), about 80% of the fissions occur in the fertile particles, so the approximation in calculating the values of $(R/B)_f$, the nominal fractional release for failed particles, is not severe. The mean $(R/B)_f$ value is 0.7%.

7.2.2. F-30 Capsule Data

The data for krypton-85m from the F-30 capsule (Ref. 7-3) are for TRISO coated ThC_2 and $(\text{Th,U})\text{C}_2$ particles. For fertile particles, the mean value of $(R/B)_f$ is calculated to be 0.9%.

7.2.3. Laser-Failed Particle Data

Data on $(R/B)_f$ for krypton-85m from laser-drilled particles are given in Table 7-1. The particles were obtained from fuel test elements FTE-1 and FTE-2. Data on particles with UC_2 kernels cover the FIMA value range 25% to 54%, and those with $(\text{Th,U})\text{O}_2$ kernels cover the range 0% to 30%. The data on particles with UC_2 kernels are weighted according to the number of particles (n) involved in the tests. The average values of $(R/B)_f$ are given in Table 7-1 and are roughly 0.5%.

Differences exist between the values of $(R/B)_f$ calculated in this section and those found in Ref. 7-1. For particles with carbide kernels, a value of 0.4%, which is based on seven samples, is given in Ref. 7-1, and in this section, the value 0.5% is found. The latter value is based on ten samples, and the average is determined by using weighting factors. These specific differences contribute to the difference in the resultant values of $(R/B)_f$. For particles with oxide kernels, the same number of samples was used in this section and in Ref. 7-1, but in Ref. 7-1 the experimental results were incorrectly multiplied by a factor of 1.9. If the average value of $(R/B)_f$ obtained in Ref. 7-1 is corrected, agreement is reached with the present result of 0.46% shown in Table 7-1.

7.2.4. Peach Bottom Core 1 Data

From Peach Bottom Core 1 full-power conditions, R/B data obtained from fuel element purge gas samples have been examined for two isotopes, krypton-85m and xenon-138. The R/B data for krypton-85m are shown in Table 7-2 as a function of the accumulated thermal energy (ATE). The reported data show a significant rise in total core R/B at low values of ATE and thereafter only a slow decrease in values. For krypton-85m, the decline in total core R/B values is only slight over the range of ATE values reported (Ref. 7-4).

The values listed in Table 7-2 are restricted to the range of ATE values for which the change in R/B values is relatively small. From the least squares curve given in Table 7-2, the average value of R/B can be calculated to be 3.52(-3). This value must be corrected for temperature and failed fraction of particles. The temperature corresponding to the data of Table 7-2 is estimated from temperatures obtained for element D13-05, as shown in Table 7-3; the temperatures are weighted according to the weighting factor $\exp -Q/RT$, where Q is roughly the temperature coefficient for release of fission gases (see Section 7.3) and is chosen to be 13.0 kcal/mole. The weighted average temperature is found to be 918°C.

Correcting the average R/B value $3.52(-3)$ at $T = 918^{\circ}\text{C}$ to 1100°C yields an average R/B at 1100°C of $7.32(-3)$. The estimate of the average failure fraction is made with the data given in Table 7-4, with the result that the fraction is 0.88; correcting the R/B for failure fraction, a value of $8.3(-3)$ is calculated for the average $(R/B)_f$ for krypton-85m.

For xenon-138, the base data are given in Table 7-5, and the treatment of these data is the same as that used for krypton-85m. The uncorrected average $(R/B)_f$ for xenon-138 based on the least squares fit of Table 7-5 is $3.14(-4)$. Corrections for temperature and failure fraction yield an $(R/B)_f$ of $7.5(-4)$.

7.2.5. Total Data Set

The data previously reported (Ref. 7-1) are combined in Table 7-6 with the data examined in the preceding sections. The average value of $(R/B)_f$ for krypton-85m is 0.5% with a fractional standard deviation at the 95% confidence level of 0.86. This fractional standard deviation is based only on the data of Table 7-6. Uncertainty estimates for the individual $(R/B)_f$ values of Table 7-6 are available for only a few of the data; therefore, combining uncertainties for individual determinations of $(R/B)_f$ is not possible at present. The average value for xenon-138 has been determined only from the Peach Bottom Core 1 data in the present analysis, and the fractional standard deviation at the 95% confidence level was found to be 0.68. However, if the individual uncertainties for the laser-drilled particles and the Peach Bottom data (for krypton-85m and xenon-138) are combined with the standard deviation for the remaining data, the fractional standard deviation at the 95% confidence level becomes 0.86. Rather than use different values of this deviation for krypton-85m and xenon-138, a single value of 0.86 is reasonable to use for both isotopes until additional data for xenon-138 are analyzed.

Measurements of $(R/B)_f$ for krypton-85m by the laser drilling technique have been reported by French workers using GA mixed oxide, TRISO coated

particles. The measurements yielded (by interpolation of data obtained at 1230° and 1050°C) a value of 0.024 for the $(R/B)_f$ of krypton-85m at 1100°C. This is almost five times larger than that found in presumably comparable experiments at GA and is displaced from the mean obtained in Table 7-6 by about nine standard deviations. At present, no understanding of this difference exists; the assumption has been made at GA that laser drilling is a suitable simulation of failed particles since the R/B measurements from particles failed in pile or in other experiments were comparable to those obtained with laser-drilled particles. Differences in the manner of laser drilling have to be explored, and exchange of test particles between French and GA investigators has to occur before more definite statements about the relevance of the French results can be made. If the French result is included by adding it to the list in Table 7-6, i.e., by giving it a weight equal to any item of the list, then the average value of $(R/B)_f$ is increased by 32% and the standard deviation is increased by 159%.

7.3. TEMPERATURE DEPENDENCE OF R/B

The uncertainties associated with the temperature dependence of the fission gas release parameter R/B were initially determined from two sets of data obtained at GA. One set was obtained using an intact Fort St. Vrain fuel rod and the other using bare kernels of UO_2 in a matrix. The mean temperature coefficients determined from these two sets of data for the temperature range 900° to 1300°C were not significantly different at confidence levels of 50% or less. The uncertainties derived from the two sets of data were combined on the basis of the similarity of the mean temperature coefficients. After completion of the analysis of the two data sets, a third set was analyzed and found to yield results consistent with the previous analysis.

Data on fission gas release from the intact Fort St. Vrain fuel rod and the bare kernel plus matrix configurations are presented in Tables 7-7

and 7-8, respectively. Data for a number of isotopes of krypton and xenon are included; the number of determinations on which each entry is based is also listed. A representative plot of the data is given in Fig. 7-1 for the krypton-85m isotope. These and similar data were treated in the following way. The data below about 300°C were visually fit with a straight line such as that shown in Fig. 7-1. Using values of R/B from this line, the high-temperature data were corrected to obtain a set of points such as those shown by dots in Fig. 7-1. A straight line was visually drawn through this set of points, and the value of the temperature coefficient Q was determined using this line. The range of values for the data of Tables 7-7 and 7-8, determined by using the n contributions to R/B for each entry in these tables, was employed to construct error bars, as illustrated in Fig. 7-1. From lines through the extremes of these error bars, upper and lower bounds on the values of Q were estimated. For the purposes of the present uncertainty analysis, simple averages of Q over the isotopes of each element were calculated; also the estimated standard deviations for the isotopes of each element were averaged. The average values of Q and the associated standard deviations were combined to obtain the following results: $\hat{Q}_{Kr} = 12.6$, $f(S_o, 95) = 0.36$; and $\hat{Q}_{Xe} = 13.3$, $f(S_o, 95) = 0.75$, where $f(S_o, 95)$ is the fractional standard deviation at the 95% confidence level.

A third set of data on fission gas release is given in Table 7-9 for fuel rod XXK40. The average of the temperature coefficients obtained from these data is not significantly different at the 50% confidence level from the value determined above for krypton-85m.

7.4. HALF-LIFE DEPENDENCE OF R/B

Data on the half-life dependence of R/B have been assembled in Table 7-10. These data are from capsule (Ref. 7-5), loop (Refs. 7-6 through 7-9), and reactor measurements (Refs. 7-1, 7-4), and they represent a variety of materials and configurations, a range of the extent of particle failure

and coating contamination, and different experimental arrangements. No attempt has been made to sort the effect of these differences on the observed half-life dependence. If such an attempt were possible and successful, the uncertainty derived below could be reduced. However, the present treatment of the data yields a widely applicable average dependence of R/B on half-life with a correspondingly large uncertainty.

The temperatures given in Table 7-10 for the loop and reactor measurements were estimated by using data on temperature distributions and weighting temperatures according to the factor $\exp(-Q/RT)$, where Q (kcal/mole) has the value 13.0 kcal/mole. The estimated temperatures are between 850° and 1230°C. Although the half-life dependence of R/B may depend on temperature (Ref. 7-10), any temperature dependence is neglected in this analysis.

The analysis of the data in Table 7-10 is summarized (including further calculations) in Table 7-11. From Table 7-11, it is seen that the mean value of n (Eq. 7-7) is not significantly different from 0.5, which is the expected exponent if the release of fission gases is by diffusion in a spherical, homogeneous medium. However, at the 95% confidence level, the values of the exponents range from 0.3 to 0.8, which is a sufficiently large range and is consistent with the existence of several different mechanisms and combinations of mechanisms for the release of fission gas. These results are shown in Figs. 7-2 and 7-3 in terms of R/B values of various isotopes of krypton-85m or xenon-138, respectively. As these figures illustrate, the uncertainties in the relative values of R/B can vary by as much as a factor of 50.

7.5. MINIMUM R/B

The minimum R/B concept is used primarily as an expedient way to calculate inventories of a large number of nuclides. The minimum R/B value for HTGR fuel results from contamination by fissile material on reactor

core surfaces, including coolant hole surfaces, fuel block surfaces, and fuel rod surfaces. At present, there are no data concerning contamination on graphite block surfaces, including coolant hole surfaces. When these surfaces are wiped, the measured values of contamination are within the background level. The measurement technique is not sufficiently sensitive to detect surface contamination. For fuel rod surfaces, however, the contamination level is much higher than that expected for fuel block surfaces.

The current specification for the large HTGR is as follows:

Exposed uranium-235 = less than 0.031 mg per element.

Exposed thorium-232 = less than 0.360 mg per element.

The average heavy metal loading is

U-235 = 0.408 kg per element.

Th-232 = 9.51 kg per element.

Hence, the average exposed fractions of heavy metal per element are 7.6×10^{-8} and 3.8×10^{-8} for uranium-235 and thorium-232, respectively.

The nominal value for the minimum R/B used for expected release calculations is 1×10^{-7} . Based on that value and the above results, for the AIPA analyses, 1×10^{-7} is taken as the nominal minimum R/B, with uncertainties ranging from 4×10^{-8} to 5×10^{-7} . The uncertainty range is arbitrary and represents a best guess.

7.6. HYDROLYSIS

Data on the effect of hydrolysis on measured R/B values are available from two studies (Refs. 7-11, 7-12). In one case (Ref. 7-11), R/B measurements were made as a function of time using laboratory prepared fuel rods

containing either bare kernels of UC_2 or laser-failed, TRISO coated particles with UC_2 kernels. These measurements were obtained in the temperature ranges of 300° to 500°C and 700° to 900°C. The data corresponding to the former range of temperatures are normally of interest in connection with conditions at reactor shutdown, and the data from the latter temperature range are of concern under the normal operating conditions of the HTGR. These measurements were obtained with particles experiencing little burnup. In the second case (Ref. 7-12), R/B measurements were made on loose, irradiated, BISO coated particles with UC_2 kernels. The range of burnups corresponded to FIMA values between 25% and 54%. In both cases, the fuel rods or particles were exposed to a carrier gas at 1 atm containing 100 ppm H_2O . In the case of the loose particles, the exposure times were shorter than in the case of the fuel rods.

An estimate of the uncertainty was made in the following manner. For the data of Ref. 7-11, a time-averaged value of the measured R/B values was computed for times greater than a minimum time t_{min} . These values are listed in Table 7-12. No distinction has been made for the different temperatures of these data. As shown in Fig. 7-4, the smoothed curves representing the data are scattered; thus, at 800 hr, for example, a least square fit to the (R/B)-T data yields only a 1% change in R/B between 300° and 900°C. The time-averaged data were combined with the data from Ref. 7-12 to obtain an average R/B value as shown in Table 7-12. The corresponding fractional standard deviation is 0.53 of the average.

7.7. LIMITATIONS OF THE ANALYSES

The reference value and associated uncertainty of R/B in failed particles for krypton-85m as well as xenon-138 were determined using data obtained at GA. However, French workers have found a significantly larger value of R/B for krypton-85m. Although the latter data were discussed in Section 7.2.5, there is difficulty in taking into account this discordant result. The difference in results could be due to the experimental

technique (laser drilling) or to more fundamental matters. Furthermore, incorporating the French results in spite of ignorance as to the cause of the difference raises the problem of the proper weight to give the French data. Thus, the reference value and the associated uncertainty obtained in Section 7.2 have these limitations.

In spite of the existence of data on the temperature dependence for R/B, these data are drawn from only a few samples of fuel rod or loose fuel particles. Additional limited data are available from the French and Dragon Project efforts. These limited data are not significantly different from the GA data. If details on the French and Dragon project efforts were available, these data could be readily incorporated into the present analyses. At present, use of the uncertainties derived must be made with awareness of the possible limited data base.

Data obtained at GA on the dependence of R/B on FIMA have consistently indicated no overall strong dependence. Recently, studies by the French on GA particles have shown a very strong dependence of R/B on FIMA in the FIMA range 0% to 6%. No resolution of this difference has yet been made and no recommendation is given. In the AIPA study on fission gas release, the possible dependence of R/B on FIMA was not taken into account. The minimum R/B in the reactor core which results from contamination by fissile material on the reactor core surface is arbitrarily chosen. No data are available to support the results of Section 7.5.

7.8. SUMMARY: FISSION GAS RELEASE DATA

Table 7-13 summarizes the mean values and associated uncertainties for fission gas release parameters. Uncertainties are expressed by the fractional standard deviation at the 95% confidence level. Thus, if the nominal value of a quantity Q is $\hat{Q}$, the limiting values of the quantity are $\hat{Q} \times [1 \pm f(95)]$. It should be noted that the values of $\hat{Q}$ obtained in this section are not in all cases the values used in the calculations in Section 4. Nominal values used in Section 4 were obtained from Ref. 4-1.

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Fig. 7-1. R/B vs reciprocal temperature for Kr-85m release from bare kernel (UO_2) plus matrix

Fig. 7-2. Relative values of R/B vs half-life for krypton isotopes

Fig. 7-3. Relative values of R/B vs half-life for xenon isotopes

Fig. 7-4. Time profiles of R/B for hydrolyzed UC_2 kernels in fuel rods (Ref. 7-12)

TABLE 7-1
DATA AND ANALYSIS FOR $(R/B)_f$ VALUES FROM LASER DRILLED UC_2 AND
 $(Th,U)O_2$ PARTICLES FOR KRYPTON-85m

Kernel	$n^{(a)}$	$w^{(b)}$	$(R/B)_f$ at 1100°C	% FIMA
$UC_2^{(c)}$	30	0.1875	3.15(-3)	25
	25	0.15625	2.66(-3)	25
	30	0.1875	8.32(-3)	30
	30	0.1875	7.26(-3)	54
	10	0.0625	2.61(-3)	25
	10	0.0625	3.54(-3)	30
	10	0.0625	3.70(-4)	54
	5	0.03125	8.08(-3)	25
	5	0.03125	9.86(-3)	30
	5	0.03125	6.26(-3)	54
$(Th,U)O_2^{(d)}$			2.49(-3)	0
			5.37(-3)	10
			6.16(-3)	10
			5.89(-3)	15
			2.87(-3)	30

Kernel	$(\hat{R}/B)_f$ at 1100 °C
UC_2	5.1(-3)
$(Th,U)O_2$	4.6(-3)

(a) n = number of particles used in tests.

(b) w = weighting factor; $\sum w = 1$.

(c) Particles obtained from crucibles 71P and 77P of FTE-1 and 72P of FTE-2.

(d) Particles obtained from crucibles 52P and 60P of FTE-1 and 53P of FTE-2.

TABLE 7-2
DATA ON TOTAL CORE (R/B) FROM PEACH BOTTOM CORE 1 FOR KRYPTON-85m^(a)

R/B at 1100°C	ATE(10 ⁵ MW/hr)	R/B at 1100°C	ATE(10 ⁵ MW/hr)
3.176(-3)	3.25	4.535(-3)	7.975
2.989(-3)	3.85	3.464(-3)	8.30
3.176(-3)	4.65	4.268(-3)	8.90
3.812(-3)	5.05	3.556(-3)	9.45
5.989(-3)	5.75	3.912(-3)	9.60
3.346(-3)	5.95	3.346(-3)	9.75
4.457(-3)	6.325	3.556(-3)	10.3
4.615(-3)	6.60	4.158(-3)	10.75
3.845(-3)	6.775	3.041(-3)	11.5
3.204(-3)	6.975	2.740(-3)	11.75
2.512(-3)	7.15	3.618(-3)	12.0
2.512(-3)	7.35	2.740(-3)	12.05
3.015(-3)	7.60		

^(a) $\ln (R/B) = -5.5689 - 0.0105 (ATE)$, where ATE = accumulated thermal energy.

TABLE 7-3
ESTIMATE OF EFFECTIVE TEMPERATURE FOR PEACH BOTTOM ELEMENT D13-05 (Ref. 6-8).

Compact No.	T(°C) (a)	exp(-Q/RT) (b)	Compact No.	T(°C) (a)	exp(-Q/RT) (b)
1	484	1.76(-4)	16	947	4.69(-3)
2	527	2.81(-4)	17	960	4.96(-3)
3	575	4.46(-4)	18	966	5.09(-3)
4	618	6.47(-4)	19	968	5.13(-3)
5	666	9.42(-4)	20	968	5.13(-3)
6	700	1.20(-3)	21	968	5.13(-3)
7	731	1.48(-3)	22	964	5.05(-3)
8	761	1.79(-3)	23	962	5.00(-3)
9	787	2.09(-3)	24	955	4.85(-3)
10	821	2.53(-3)	25	949	4.73(-3)
11	847	2.90(-3)	26	942	4.59(-3)
12	878	3.40(-3)	27	934	4.42(-3)
13	904	3.85(-3)	28	929	4.33(-3)
14	917	4.10(-3)	29	925	4.25(-3)
15	934	4.42(-3)	30	921	4.17(-3)

(a) Effective temperature = 918°C.

(b) Q = 13.0 kcal/mole.

TABLE 7-4
ESTIMATE OF AVERAGE FAILURE FRACTION AT 1100°C (Ref. 7-13)^(a)

Compact No.	% Failure	Weighting Factor ^(b)
1	6	1.76(-4)
8	97.5 ^(c)	1.79(-3)
10	88	2.53(-3)
15	92	4.42(-3)
20	94	5.13(-3)
25	80	4.73(-3)
30	85	4.17(-3)

(a) Weighted average % failure = 88; % standard deviation = 7.2.

(b) The weighting factors are values of $\exp(-Q/RT)$ taken from Table 7-3.

(c) Average of longitudinal and transverse sections.

TABLE 7-5
DATA ON (R/B) FROM PEACH BOTTOM CORE 1 FOR XENON-138^(a)

(R/B)	ATE(10 ⁵ MW/hr)	(R/B)	ATE(10 ⁵ MW/hr)
3.95(-4)	3.87	3.32(-4)	8.85
3.72(-4)	4.65	3.02(-4)	9.18
4.27(-4)	5.76	3.20(-4)	9.55
3.39(-4)	6.01	2.32(-4)	9.71
4.21(-4)	6.34	2.69(-4)	10.25
4.80(-4)	6.58	3.17(-4)	10.70
3.92(-4)	6.95	2.89(-4)	11.48
4.00(-4)	7.16	2.69(-4)	11.73
3.20(-4)	7.33	2.39(-4)	11.91
4.21(-4)	7.59	2.57(-4)	12.02
3.75(-4)	7.96	2.19(-4)	12.36
3.11(-4)	8.27		

^(a) $\ln R/B = -7.4116 - 0.07183 (ATE)$, where ATE = accumulated thermal energy.

TABLE 7-6
COMBINED DATA AND AVERAGE $(R/B)_f$ VALUE FOR KRYPTON-85m

Source	$(R/B)_f$	Ref.
Pluto 3A loop	0.005	7-1
Capsule 309	0.005	7-1
Capsule 6F2	0.004	7-1
GAIL IIIA loop	0.002	7-1
GAIL IIIB loop	0.005	7-1
GAIL IV loop	0.006	7-1
Peach Bottom Core 1	0.008	Section 7.2.4
Fuel rods, P13N	0.002	7-1
Laser drilled oxides	0.0046	Section 7.2.3
Laser drilled carbides	0.005	Section 7.2.3
Capsule F-30	0.009	Section 7.2.2
Capsules HRB-4, HRB-5 and HRB-6	0.007	Section 7.2.1
$(\hat{R}/B)_f$	0.005	
S_o	0.0022	
$f(S_o, 95)$	0.86	
Including CEA data:	0.024	
$(\hat{R}/B)_f$	0.0066	
S_o	0.0057	
$f(S_o, 95)$	1.69	

TABLE 7-7
DATA ON FISSION GAS RELEASE FROM INTACT FSV FUEL ROD (CR12-30049012)

T(°C)	$10^4/T(^{\circ}\text{K})$	$n^{(a)}$	R/B					
			Kr-85m	Kr-87	Kr-88	Kr-89	Kr-90	Kr-91
40.9	31.85	6	2.32(-6)	1.99(-6)	1.81(-6)	1.23(-6)		
98	26.95	2	2.23(-6)	2.09(-6)	1.82(-6)	1.46(-6)		
215	20.49	2	3.08(-6)	2.81(-6)	2.56(-6)	2.10(-6)		
280	18.08	2	3.88(-6)	3.39(-6)	2.96(-6)	2.39(-6)		
416	14.51	2	6.80(-6)	5.61(-6)	5.19(-6)	3.88(-6)		
525	12.53	2	6.52(-6)	6.63(-6)	5.39(-6)	4.61(-6)		4.62(-6)
650	10.83	2	1.04(-5)	1.04(-5)	8.60(-6)	7.03(-6)	6.85(-6)	1.00(-6)
713	10.14	2	1.02(-5)	1.00(-5)	8.80(-6)	6.81(-6)		4.36(-6)
814	9.20	3	1.68(-5)	1.70(-5)	1.38(-5)	9.83(-6)	8.19(-6)	1.06(-5)
1015	7.76	2	3.95(-5)	2.93(-5)	3.17(-5)	1.58(-5)	3.94(-6)	9.59(-6)
1080	7.39	2	6.66(-5)	5.19(-5)	5.38(-5)	2.10(-5)	1.05(-5)	1.17(-5)
1140	7.08	2	8.41(-5)	6.38(-5)	6.77(-5)	2.71(-5)	1.04(-5)	1.70(-5)
1278	6.45	2	1.37(-4)	1.03(-4)	1.15(-4)	4.75(-5)	2.5(-5)	2.03(-5)
			<u>Xe-133</u>	<u>Xe-135</u>	<u>Xe-135m</u>	<u>Xe-137</u>	<u>Xe-138</u>	<u>Xe-139</u>
40.9	31.85	6	1.31(-5)	2.38(-6)	1.69(-6)	1.36(-6)	1.19(-6)	7.41(-7)
98	26.95	2	2.36(-6)	1.69(-6)	1.11(-6)	1.47(-6)	1.09(-6)	9.55(-7)
215	20.49	2		2.30(-6)	1.73(-6)	1.69(-6)	1.80(-6)	1.22(-6)
280	18.08	2	5.35(-6)	2.54(-6)	2.02(-6)	1.89(-6)	1.56(-6)	1.32(-6)
416	14.51	2	2.65(-5)	5.75(-6)	3.04(-6)	2.75(-6)	2.14(-6)	2.04(-6)
525	12.53	2	1.20(-5)	4.66(-6)	4.03(-6)	4.02(-6)	2.91(-6)	2.85(-6)
650	10.83	2	1.34(-5)	7.49(-6)	6.38(-6)	5.31(-6)	4.21(-6)	4.12(-6)
713	10.14	2	1.19(-5)	7.13(-6)	5.36(-6)	5.90(-6)	3.93(-6)	3.21(-6)
814	9.2	3	2.16(-5)	9.04(-6)	7.47(-6)	6.13(-6)	4.39(-6)	4.74(-6)
1015	7.76	2	5.21(-5)	1.47(-6)	1.03(-5)	5.24(-6)	5.60(-6)	4.79(-6)
1080	7.39	2	1.04(-4)	2.77(-5)	1.31(-5)	7.17(-6)	7.27(-6)	5.28(-6)
1278	6.45	2	1.44(-4)	8.02(-5)	4.51(-5)	1.81(-5)	1.91(-5)	1.03(-5)

(a) n = number of determinations of R/B for which the listed R/B values represent an average.

TABLE 7-8
DATA ON FISSION GAS RELEASE FROM BARE KERNELS OF UC₂ PLUS MATRIX

T(°C)	10 ⁴ /T(°K)	n ^(a)	R/B					
			Kr-85m	Kr-87	Kr-88	Kr-89	Kr-90	Kr-91
41	31.8	3	8.70(-4)	8.49(-4)	7.18(-4)	4.35(-4)		
100	26.8	4	1.46(-3)	1.14(-3)	1.07(-3)	6.56(-4)		
310	17.2	4	1.76(-3)	1.72(-3)	1.40(-3)	8.57(-4)		1.29(-3)
435	14.12	2	3.19(-3)	3.00(-3)	2.52(-3)	1.77(-3)	1.43(-3)	1.92(-3)
603	11.42	3	4.16(-3)	4.05(-3)	3.46(-3)	2.62(-3)	1.67(-3)	2.98(-3)
715	10.12	2	6.87(-3)	6.28(-3)	5.81(-3)	4.20(-3)	2.92(-3)	4.31(-3)
810	9.23	2	1.04(-2)	9.75(-3)	8.50(-3)	5.04(-3)	3.71(-3)	5.53(-3)
950	8.18	3	1.52(-2)	1.30(-2)	1.23(-2)	6.73(-3)	4.83(-3)	6.68(-3)
1060	7.50	2	2.71(-2)	2.18(-2)	2.35(-2)	1.07(-2)	1.08(-2)	9.89(-3)
1080	7.39	1	2.71(-2)	2.16(-2)	2.34(-2)	1.06(-2)	1.36(-2)	5.17(-3)
1165	6.95	2	3.83(-2)	3.18(-2)	3.33(-2)	1.59(-2)	1.14(-2)	1.28(-2)
1265	6.50	2	5.53(-2)	4.20(-2)	4.70(-2)	1.93(-2)	1.59(-2)	1.70(-2)
			<u>Xe-133</u>	<u>Xe-135</u>	<u>Xe-135m</u>	<u>Xe-137</u>	<u>Xe-138</u>	<u>Xe-139</u>
41	31.8	2		5.81(-4)	5.70(-4)	5.86(-4)	4.36(-4)	2.76(-4)
100	26.8	3		7.91(-4)	6.67(-4)	4.78(-4)	5.06(-4)	2.06(-4)
310	17.2	3	1.48(-3)	1.16(-3)	1.21(-3)	1.33(-3)	8.64(-4)	7.14(-4)
435	14.12	2	4.13(-3)	2.33(-3)	1.40(-3)	1.18(-3)	1.01(-3)	8.52(-4)
603	11.42	3	3.23(-3)	2.06(-3)	2.12(-3)	1.76(-3)	1.40(-3)	1.36(-3)
715	10.12	2	5.15(-3)	2.88(-3)	2.50(-3)	2.15(-3)	1.74(-3)	1.42(-3)
810	9.23	2	1.05(-2)	3.54(-3)	3.03(-3)	2.33(-3)	2.10(-3)	2.22(-3)
950	8.18	3	6.37(-3)	3.83(-3)	3.04(-3)	2.49(-3)	2.13(-3)	2.14(-3)
1060	7.50	2	2.17(-2)	8.12(-3)	4.45(-3)	3.90(-3)	3.86(-3)	2.68(-3)
1080	7.39	1	4.61(-2)	1.36(-2)	1.00(-2)	6.35(-3)	1.00(-2)	2.08(-3)
1165	6.95	2	4.22(-2)	1.28(-2)	6.95(-3)	5.74(-3)	5.30(-3)	3.10(-3)
1265	6.50	2	4.12(-1)	6.33(-2)	3.04(-2)	6.13(-3)	8.18(-3)	4.36(-3)

(a) n = number of determinations of R/B for which the listed R/B values represent an average.

TABLE 7-9
DATA ON FISSION GAS RELEASE FROM FUEL ROD XXX40 (MARK-III-3 MEASUREMENTS)

T(°K)	$10^4/T(°K)$	n ^(a)	10^5 R/B				
			Kr-85m	Kr-87	Kr-88	Kr-89	Xe-138
1343	7.45	13	9.21	6.12	5.54	3.83	1.62
1291	7.75	1	11.4	7.08	6.71	3.86	1.06
1333	7.50	1	10.3	7.06	6.22	4.39	1.60
1364	7.33	1	8.82	5.93	5.32	5.3	2.1
1375	7.27	1	14.9	9.47	8.98	5.93	2.04
1447	6.91	5	18.7	11.8	11.9	7.1	3.15
1437	6.96	1	15.3	10.5	9.61	5.68	2.11
1427	7.00	3	17.2	11.3	10.5	6.19	2.32
1468	6.81	1	22.2	13.2	15.0	6.6	1.8
1547	6.46	1	30.9	18.0	19.8	10.1	5.1
1572	6.36	1	26.2	18.2	18.1	11.6	6.8
1562	6.40	3 ^(b)	23.9	16.0	16.7	8.37	3.6
1577	6.34	1	13.7	9.5	8.6	6.6	2.3
1463	6.84	1	18.2	12.8	12.7	6.6	3.3
1551	6.45	3	21.8	14.6	15.6	7.2	3.2
1541	6.49	1	30.9	19.9	21.0	10.4	4.1
1531	6.53	1	25.1	17.6	17.8	8.4	3.2
1557	6.42	1	19.6	14.3	14.3	7.2	2.5
1650	6.06	1	46.1	25.5	30.2	13.1	6.2
1656	6.04	3 ^(c)	34.4	22.6	22.1	11.8	5.0
1645	6.08	1	36.1	23.4	24.4	4.6	5.7
1676	5.97	1	35.8	24.4	24.5	12.8	--
1624	6.16	1	30.0	20.2	22.0	10.4	3.8
1635	6.12	1 ^(c)	24.8	9.3	9.6	5.8	3.6
1640	6.10	1	61.1	30.3	37.4	11.2	4.3
1750	5.71	5	44.2	26.0	29.8	14.5	8.0
1760	5.68	(d)	41.6	26.8	28.0	12.8	5.6
1739	5.75	2 ^(e)	41.4	27.4	27.4	14.3	7.2
1771	5.65	2	24.4	16.2	16.4	9.7	3.8
1781	5.61	1	41.7	28.4	28.9	14.0	6.2
1343	7.45	5	17.9	9.7	10.9	4.0	1.3
1353	7.39	3	25.6	12.3	15.9	4.0	2.3
1348	7.42	2 ^(f)	28.9	13.7	16.7	5.0	1.4
1323	7.56	1	11.7	8.1	7.3	5.6	2.7
1336	7.49	2	13.3	9.2	8.4	5.4	2.1
1362	7.34	2	14.7	10.0	9.0	6.2	2.2
1388	7.20	1	17.2	10.8	11.1	5.7	1.9
1358	7.36	5	14.9	9.3	8.1	5.0	2.2

(a) n = number of determinations of R/B for which the listed R/B values represent an average.

(b) n = 2 for Kr-88.

(c) n = 2 for Kr-89 and Xe-138.

(d) n = 6 for Kr-85m; 7 for Kr-87; 5 for Kr-88; 7 for Kr-89; and 6 for Xe-138.

(e) n = 1 for Kr-88.

(f) n = 1 for Xe-138.

TABLE 7-10
R/B DATA FOR VARIOUS LOOPS, CAPSULES, AND REACTOR

Isotope	$t_{1/2}$ (hr)	λ (sec ⁻¹)	R/B										
			Ref. 7-5	Ref. 7-5	Ref. 7-5	Ref. 7-6	Ref. 7-7	Ref. 7-4	Ref. 7-4	Ref. 7-4	Ref. 7-8	Ref. 7-8	Ref. 7-1
Kr-85m	4.4	4.38(-5)	1.7(-3)	7.0(-3)	1.1(-2)	4.9(-3)	3.3(-5)	2.7(-4)	3.2(-4)	3.3(-3)	1.0(-5)	7.6(-4)	2.1(-4)
Kr-85	9.41(4)	2.04(-9)	--	--	--	--	--	--	--	--	--	1.2(-3)	--
Kr-87	1.27	1.52(-4)	5.4(-4)	2.8(-3)	3.5(-3)	2.1(-3)	1.7(-5)	1.1(-4)	1.1(-4)	1.5(-3)		5.4(-4)	7.0(-5)
Kr-88	2.79	6.90(-5)	7.7(-4)	3.4(-3)	4.5(-3)	2.3(-3)	2.2(-5)	1.2(-4)	1.4(-4)	1.8(-3)	6.6(-6)	6.2(-4)	1.1(-4)
Kr-89	5.3(-2)	3.63(-3)	1.8(-4)	1.5(-3)	1.65(-3)	8.6(-4)	5.3(-6)	3.6(-5)	2.6(-5)	2.5(-4)	9.6(-7)	1.9(-4)	9.0(-6)
Kr-90	8.97(-3)	2.15(-2)	1.3(-4)	1.4(-4)	4.0(-3)	2.2(-4)						5.4(-5)	
Kr-91	2.39(-3)	8.05(-2)	--	--	--	--						2.1(-5)	
Xe-133	1.265(2)	1.52(-6)	4.8(-3)	3.0(-2)	5.0(-2)	9.1(-3)	2.2(-5)	7.5(-4)	1.2(-3)	5.0(-3)	1.5(-5)		4.9(-4)
Xe-135	9.16	2.10(-5)	4.5(-4)	2.2(-3)	4.5(-3)	1.4(-3)	6.7(-6)	1.09(-4)	1.9(-4)	9.1(-4)	3.3(-6)		1.4(-4)
Xe-135m	2.62(-1)	7.35(-4)	--	--	--	--	--						
Xe-137	6.37(-2)	3.02(-3)	3.9(-5)	2.2(-4)	4.8(-4)	2.9(-4)					5.1(-7)		
Xe-138	2.37(-1)	8.12(-4)	7.2(-5)	6.5(-4)	6.5(-4)	2.5(-4)	1.8(-6)	3.7(-5)	2.5(-5)	2.6(-4)			1.4(-5)
Xe-139	1.11(-2)	1.73(-2)	--	--	--	--							
Xe-140	3.78(-3)	5.09(-2)	5.8(-6)	7.9(-6)	1.6(-5)	1.9(-5)	3.1(-7)				5.0(-7)		
Estimated temperature (°C)			1120	1230	1150	1134	990	<850	850	850	1129	--	--

TABLE 7-11
SUMMARY OF VALUES OF THE EXPONENT n

Ref.	n_{Kr}	n_{Xe}
7-5	0.375	0.590
7-5	0.513	0.688
7-5	0.439	0.681
7-6	0.424	0.520
7-7	0.391	0.388
7-4	0.394	0.468
7-4	0.508	0.613
7-4	0.552	0.463
7-8	0.513	0.334
7-1	0.679	0.570
$\hat{n}_i$	0.479	0.532
$S(n_i)$	0.092	0.116
$f_s(n_i)$	0.192	0.218
$f_s(95)$	0.376	0.427

TABLE 7-12
ANALYSIS OF DATA ON THE EFFECT OF HYDROLYSIS ON R/B VALUES (KRYPTON-85m
AT 1100°C) AT HYDROLYSIS TEMPERATURES IN THE RANGE 300° TO 900°C

T(°C)	Configuration ^(a)	R/B	Exposure Time (hr)	Ref.
300	bk	0.21	100	7-11
400	bk	0.18	15	
500	bk	0.23	15	
300	1f	0.21	300	
400	1f	0.12	300	
500	1f	0.10	300	
700	bk	0.28	10	
800	bk	0.35	10	
900	bk	0.26	7	
700	1f	0.28	295	
800	1f	0.23	107	7-12
900	1f	0.09	58	
900	b(54,30)	0.33	(b)	
900	b(54,10)	0.08	(b)	
900	b(30,30)	0.20	(b)	
900	b(30,10)	0.52	(b)	
900	b(25,30)	0.27	(b)	
900	b(54,25)	0.20	(b)	
900	b(25,10)	0.50	(b)	
Average		0.24		
Standard deviation		0.127		
f(95) ^(c)		1.0		

^(a) bk = bare kernels; 1f = laser failed; b(x,y) = BISO coated particles with x% FIMA and y in number.

^(b) The particles were exposed for relatively short times (18 to 67 hr) and were corrected for gross particle failure.

^(c) f(95) = fractional standard deviation at the 95% confidence level.

TABLE 7-13
SUMMARY OF UNCERTAINTIES IN FISSION GAS RELEASE DATA

Classification of Data	Quantity ^(a)	$\hat{Q}$	f(95)
Failed fuel particles at 1100°C	$(R/B)_f$		
Krypton-85m		5(-3)	0.86
Xenon-138		7.5(-4)	0.86
Temperature dependence of R/B (activation energy, kcal/mole)	Q_{Kr}	12.6	0.36
	Q_{Xe}	13.3	0.75
Half-life dependence of R/B	n		
Krypton isotopes		0.48	0.38
Xenon isotopes		0.53	0.43
Dependence of R/B on hydrolysis	$(R/B)_{WH}$	0.24	1.0
Minimum R/B	$(R/B)_{min}$	1×10^{-7}	9.8

(a) $\hat{Q}$ = the mean value of Q found in the present analysis; $(R/B)_f$ = the release rate to birth rate ratio for fission gas in failed fuel particles; Q_i = the temperature coefficient (activation energy) for element i; n = the exponent from Eq. 7-7; $(R/B)_{WH} = (R/B)_f$ for hydrolyzed fuel; $(R/B)_{min}$ = the minimum R/B due to contamination effects.

8. MEAN VALUES AND UNCERTAINTIES FOR SORPTION OF CESIUM AND IODINE ON PRIMARY CIRCUIT COMPONENTS

This section presents estimates for the mean values and uncertainties associated with cesium sorption on metals and iodine sorption on metals and graphite. The results of these analyses were used in the calculations in Section 5, where uncertainties in fission product plateout distributions were determined.

8.1. PROCEDURES FOR ESTIMATING MEANS AND UNCERTAINTIES

The uncertainties were generally determined in the manner outlined in Section 6.1. Several additional procedures used in the present analyses are described in this section. In several data sets, most of the data are narrowly spread about some nominal temperature, with very few data at significantly different temperatures. These few data had to be used to obtain the temperature dependence of the vapor pressure - sorbate concentration curves. To estimate the error associated with the use of these few data, the variance in the vapor pressure was treated as follows:

$$S^2 = S_A^2 + \frac{S_B^2}{T^2} + \left(S_D^2 + \frac{S_E^2}{T^2} \right) (\log C)^2 + \left(\frac{\partial \log p}{\partial \log C} \right)^2 S_{\log C}^2 \quad , \quad (8-1)$$

with

$$\log p = (A + B/T) + (D + E/T) \log C \quad , \quad (8-2)$$

where S^2 is the variance in $\log p$, S_i^2 is the variance in i , T is the temperature, and C is the sorbate concentration. Since the variance is known at the nominal temperature, the variance at temperature T becomes

$$s_T^2 = s_{T_n}^2 + \left\{ s_B^2 + s_E^2 (\log C)^2 \right\} \left\{ |T^{-2} - T_o^{-2}| \right\} , \quad (8-3)$$

where T_o is the nominal temperature, and the absolute difference in temperatures reasonably results in an increase in the variance at any temperature other than the nominal temperature. Equation 8-3 was used only over a restricted temperature range.

In using the few data available for determining the temperature dependence of the pressure/concentration curves, the values of B and E were determined by using the following relation:

$$\partial p / \partial (1/T) = B + E \log C \quad . \quad (8-4)$$

Values of $A + B/T$ and $D + E/T$ were known from the isotherm at the nominal temperature.

As in Section 6, the error in a sorption isotherm at a fixed temperature was assigned to the vapor pressure rather than the sorbate concentration. In one case, a mean isotherm curve was selected by considering the error to be assigned to the vapor pressure and the sorbate concentration. Thus, if $y \equiv \log p$ and $x \equiv \log C$ and the analyses yield

$$y = a + bx \quad , \quad (8-5)$$

$$x = \alpha + \beta y \quad , \quad (8-6)$$

then

$$\bar{y} = 1/2 (a - \alpha/\beta) + 1/2 (b + 1/\beta)x \quad , \quad (8-7)$$

where the error is formally assigned to y. As an estimate of the mean uncertainty associated with Eq. 8-7, the following equation was used:

$$\bar{s}_y = 1/2 \left[s_y + \frac{(b + \beta^{-1})}{2} s_x \right] \quad , \quad (8-8)$$

where S_y and S_x are the standard deviations associated with the dependent variables y and x of Eqs. 8-5 and 8-6, respectively.

8.2. MEAN VALUES AND UNCERTAINTIES FOR IODINE SORPTION

8.2.1. Iodine Sorption on Graphite

Previously used values for iodine sorption on graphite (Ref. 5-1) were used in the present analysis. The uncertainties in iodine sorption were estimated from the data which appear in Fig. 8-1. Mean values for iodine sorption derived from the data of Fig. 8-1 were not significantly different from the values derived using the older isotherms. The data in Fig. 8-1 are not for the exact same temperature, but in the analysis the data were regarded as corresponding to a nominal temperature of 500°C. The data were treated by first sketching boundary curves, as shown in Fig. 8-1. Based on these boundary curves, a mean curve was selected. This mean curve is shown in Fig. 8-1. The boundary curves were then quantitatively defined by using selected points on the estimated boundary curves to obtain the coefficients of the equation:

$$S_i^2 = a + b (\log C) + d (\log C)^2, \quad (8-9)$$

where S_i is the standard deviation for the upper ($i = u$) or lower ($i = \ell$) boundary curve; C ($\mu\text{g iodine/g}$) is the sorbate concentration; and a , b , and d are coefficients. The values of S_i used in the analysis were simply the difference between the estimated boundary and mean curves. Equation 8-9 is of the same form as Eq. 6-4. The mean and variance equations for iodine sorption on graphite are given in Section 8.5.

8.2.2. Iodine Sorption on Low-Alloy Metals

The data for sorption of iodine on 1% Cr - 1/4% Mo steel, iron oxide, and mild steel comprise those for low-alloy metals (Refs. 8-3 to 8-7). The

largest set of data points are for a nominal temperature of 400°C. These data were used in the following way. An isotherm at 400°C was derived using the data for 1% Cr - 1/4% Mo steel (Ref. 8-4) on the basis that this alloy is more like the low-alloy materials encountered in the HTGR. All the data were used in the calculation of the corresponding uncertainties since only one experiment was conducted on the 1% Cr - 1/4% Mo steel material. The data used to obtain the 400°C isotherm are shown in Fig. 8-2. These data are based on the geometric surface area. As shown in Table 8-1, the least squares fit was calculated for two cases: one case in which the experimental (random) error was regarded as being only in values of the pressure p and a second case in which only values of the sorbate concentration c contained errors. The mean of these cases was selected according to Eq. 8-7. The mean of the least squares fit is shown in Fig. 8-2.

The data used in the uncertainty analysis for the 400°C isotherm are given in Table 8-1. A nominal temperature of 400°C was taken for all data. Again, the cases in which all the error was regarded as being in either p or C were treated. To simplify the procedure for obtaining a mean of the uncertainties for the two cases, each uncertainty expression was tested for constancy of the variances over a relevant range of the values of p and C . The variances were found to change only slightly. Consequently, to obtain a mean variance, the constant values of the variances, as given in Table 8-1, were combined according to Eq. 8-8.

The temperature dependence of the vapor pressure was obtained by using two sets of data (Refs. 8-3, 8-5). The isotherm expression obtained with the desired temperature dependence is given in Section 8.5, and isotherms for various temperatures calculated with this expression are given in Fig. 8-3. As a result of the limited data used in obtaining the temperature dependence in the isotherm expression, errors were probably introduced into the expression. In order to account for this uncertainty, the errors were estimated by using Eq. 8-3. The result of this evaluation yields an

abnormally large variance. The calculated large variance follows from the 70% fractional standard deviation in the parameter B and is thus a reflection of the limitations of the data used to estimate the temperature dependence. An alternate method of estimating the uncertainty was finally employed. This alternate method consists of setting the ratio of the variance at temperature T to that at 400°C for the low-alloy data equal to the ratio for the high-alloy data. The variances for the high-alloy data are given in Section 8.2.3. Then, for example, the ratio of the variance at 200°C to that at 400°C for low-alloy data becomes 4.84 rather than 26 at $C = 5 \text{ } \mu\text{g}/\text{cm}^2$.

8.2.3. Iodine Sorption on High-Alloy Metals

The high-alloy data were taken from sorption measurements of iodine on Hastelloy-X, Incoloy 800, and 347 stainless steel. The analysis of the data was based on the assumption that all error is in pressure p. The corresponding line based on the assumption that all error is in concentration C differed negligibly; this difference was neglected.

The data employed in the analysis have a nominal temperature of 470°C. The temperature dependence was established based on deposition loop observations with the additional assumption that the slope of the isotherm on the $\log C - \log p$ plot remains unchanged over the temperature range 350° to 660°C. A plot of the curves obtained is shown in Fig. 8-4; the 95% confidence bounds for the 470°C isotherm obtained from the analysis are also shown. The additional error resulting from the estimation of the temperature dependence for the temperature range 350° to 660°C was determined as indicated in Section 8.1. In this instance, the term $S_T^2 (\log C)^2$ was neglected, and the additional error at 350° or 660°C was taken to be $(\log 50)/2$. The equations for the mean and variance of $\log p$ are given in Section 8.5.

8.3. CESIUM SORPTION ON METALS

This section treats cesium sorption on metals. Cesium sorption on graphite is discussed in Section 6.2.3.

The mean values for the sorption of cesium on metals were obtained from isotherms based on data for oxidized 304 stainless steel (Ref. 8-8). These data were treated by averaging to obtain the isotherm expression given in Section 8.5. The estimation of the associated uncertainties was based on additional data obtained on Incoloy 800, Hastelloy-X, and 304 stainless steel. The isotherm and variance expressions given in Section 8.5 were used in the analyses of Section 5 for sorption on high-alloy metals only. For low-alloy metals, perfect sink conditions were applied.

8.4. LIMITATIONS OF THE ANALYSIS

The mean curves and uncertainties derived are clearly limited by the lack of data for conditions of interest in HTGRs. Particularly severe are (1) the necessity to extrapolate the available data to the low vapor pressure of interest, i.e., to pressures in the range 10^{-9} to 10^{-13} atm; (2) the lack of data for some materials of interest and for the surface conditions under normal operating conditions; and (3) the question about the significance of differences in time available for equilibration under laboratory conditions as compared with reactor conditions, i.e., whether penetration of the reactor surfaces by sorbates is only significant under reactor conditions. The results of the present analysis can only be considered to be very preliminary estimates.

8.5. SUMMARY: CESIUM AND IODINE SORPTION ON PRIMARY CIRCUIT COMPONENTS

A summary of the mean curves and associated uncertainties is presented.

8.5.1. Iodine Sorption on Graphite

Iodine sorption on graphite can be expressed as

$$\log p = (0.954 + 3698/T) + (0.73 + 1250/T) \log C \quad , \quad (8-10)$$

$$S^2(\log p) = 4.87 + 4.23 \log C + 3.44 (\log C)^2 \quad , \quad (8-11)$$

where p (atm) is the vapor pressure, T ($^{\circ}\text{K}$) is the temperature, C (μg iodine/g graphite) is the sorbate concentration, $S^2(\log p)$ is the variance in the logarithm to the base 10 of the pressure, and S is the standard deviation of a single estimated value of the dependent variable $\log p$.

8.5.2. Iodine Sorption on Low-Alloy Metals

Iodine sorption on low-alloy metal is given as

$$\log p = (5.96 - 11,050/T) + (4.24 + 238/T) \log C \quad , \quad (8-12)$$

$$S^2 = 5.62 + \left\{ 6.08 \times 10^7 + 1.56 \times 10^6 (\log C)^2 \right\} \left| T^{-2} - (673)^{-2} \right| \quad , \quad (8-13)$$

where C (μg iodine/ cm^2 metal) is the sorbate concentration, and S^2 is an average of the variance in $\log p$ and $\log C$.

8.5.3. Iodine Sorption on High-Alloy Metals

Iodine sorption on high-alloy metals is given by

$$\log p = -(1.60 + 2846/T) + 0.947 \log C \quad , \quad (8-14)$$

$$S^2(\log p) = 0.227 + 1.20 \times 10^6 \left| T^{-2} - (743)^{-2} \right| \quad , \quad (8-15)$$

where the symbols have the same meaning as those for iodine sorption on low-alloy metals.

8.5.4. Cesium Sorption on Low-Alloy Metals

Perfect sink conditions are assumed for cesium sorption on low-alloy metals. Thus, the vapor pressure of the sorbed species (cesium) is taken to be negligible over the low-alloy metals for concentrations and temperatures encountered in the reactor under normal operating conditions.

8.5.5. Cesium Sorption on High-Alloy Metals

Cesium sorption on high-alloy metals is

$$\log p = (32.24 - 43,429/T) + (-37.82 + 43,120/T) \log C \quad , \quad (8-16)$$

$$S^2(\log p) = 1.78 + 0.163 \log C + 0.132 (\log C)^2 \quad , \quad (8-17)$$

where C ($\mu\text{g cesium}/\text{cm}^2$ metal) is the sorbate concentration, and other symbols have the same definitions as those for iodine sorption on graphite.

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Fig. 8-1. Iodine sorption on graphite

Fig. 8-2. Data on sorption of iodine on low-alloy metal (1% Cr - 1/4% Mo steel) at 400°C

Fig. 8-3. Sorption isotherms for iodine sorption on low-alloy metal

Fig. 8-4. Sorption isotherms for iodine sorption on high-alloy metals

TABLE 8-1
DATA AND UNCERTAINTY ANALYSIS FOR SORPTION OF IODINE ON LOW-ALLOY METALS^(a)

Ref.	T (°C)	log p (atm)	log C (μg/cm ²)	Ref.	T (°C)	log p (atm)	log C (μg/cm ²)
8-4	400	-4.959	1.318	8-7	400	-9.310	-1.260
		-5.081	1.365			-9.310	-0.7959
		-5.398	0.9638			-9.310	-1.658
		-5.398	0.9685			-9.310	-1.959
		-5.553	1.152			-9.310	-1.143
		-5.553	1.086			-9.310	-1.796
		-6.00	0.9031			-9.310	1.170
		-6.00	0.6021		427	-9.310	-1.921
		-7.046	0.8129			-9.310	-1.721
		-8.886	0.5798			-9.310	-1.678
8-6	400	-4.398	0.4771	8-5	400	-9.310	-2.357
		-9.00	-0.3010			-9.310	-1.018
8-3	425	-7.097	0.6989			-9.544	-0.880
	425	-7.097	0.7782			-8.538	-0.490
	400	-6.388	1.000			-7.529	-0.216
						-6.544	-0.170

(a) The error in log p is

$$s_o^2 = 0.038 + 0.00945 (\log C) + 0.0267 (\log C)^2 ,$$

$$s^2 = 1.20 + 0.00945 (\log C) + 0.0267 (\log C)^2 ,$$

$$s^2 \approx 1.25 ; \quad -2 \leq (\log C) \leq 2 .$$

The error in log C is

$$s_o^2 = 0.357 + 0.0885 (\log p) + 0.00577 (\log p)^2 ,$$

$$s^2 = 0.896 + 0.0885 (\log p) + 0.00577 (\log p)^2 ,$$

$$s^2 \approx 0.624 ; \quad -13 \leq \log p \leq -7 .$$

The chosen mean for errors in log p and log C is

$$\bar{s} = \frac{1.12 + 4.59 (0.79)}{2} = 2.37 .$$