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HTGR ACCIDENT INITIATION AND PROGRESSION ANALYSIS STATUS REPORT

VOLUME VI. EVENT CONSEQUENCES AND UNCERTAINTIES DEMONSTRATING SAFETY R&D IMPORTANCE OF FISSION PRODUCT TRANSPORT MECHANISMS

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1. SUMMARY

This volume of the report presents Phase I analysis of consequences for the events that were selected for detailed risk analysis during the preliminary scoping analysis reported in Volume III. Consequence assessments are presented in the form of radiological doses in rem to representative site boundaries of 500 m [exclusion area boundary (EAB)] and 2500 m [low population zone boundary (LPZB)] for a reference 3000-MW(t) HTGR. In addition, man-rem doses to the surrounding environment are derived based on WASH-1400 population densities. A health effects model based on somatic effects of radiation exposure is developed following guidelines of the Environmental Protection Agency (EPA). This model was used to establish an index to assess the relative safety R&D significance of fission product isotopes and the transport mechanisms that contribute to the radiological doses predicted in each event analyzed. The evaluation of the dose consequences and statistical uncertainties for the spectrum of events considered in the detailed Phase I risk assessment (Volume IV) supplies needed input to that assessment.

Five accident conditions resulting from the (Volume III) initiating events were considered. Two relatively high risk conditions were evaluated for input to the Phase I risk assessment: (1) core heatup resulting from the loss of offsite power (LOSP) and earthquake (EQ) events, and (2) reheater tube leak. Three relatively low risk design basis events were selected for evaluation: (1) slow depressurization, (2) rapid depressurization (DBDA), and (3) steam ingress from the steam generator main bundle tube rupture. These events involve virtually all existing fission product technologies that have been developed to date. In addition, the steam ingress consequences were important to provide more meaningful tradeoffs in the design option study for moisture protection safeguards presented in the preliminary study (Volume III).

This study traces the transport phenomena of fission product isotopes from the inventories associated with normal operation (Volume V) to radiological doses at the site boundaries. Compared with the results in Volume III, the present results are characterized by more realistic initial fission product inventories, more realistic modeling of release, and more realistic evaluation of radiological consequences. In particular, median values and statistical uncertainties of the input parameters and resulting doses were derived. A summary of event radiological consequences and uncertainties, and of fission product safety ranking, is given in Sections 7 and 6, respectively.

The relatively most severe consequences among the events analyzed occur during core heatup, where the 30-day whole body dose (0.2 rem for whole body gamma plus beta median dose with a 95% confidence level uncertainty factor of ~ 25) at the LPZB is controlling. Dose magnitudes at the site boundaries are below regulatory limits (25 rem whole body gamma plus beta per 10CFR100) and well below estimated limits for acute health effects.

For the core heatup event, release of initial gasborne activity is small compared to fuel body activity release, even for the 2-hr dose. For release of fuel body activity, the noble gases (chiefly Xe-133) have the greatest significance in the safety index employed. This stems from the importance of the whole body gamma dose. The median inhalation thyroid and bone LPZB doses are lower than the median whole body gamma LPZB dose at 30 days by factors of 63 and 11,000, respectively. Considering the respective uncertainties, the upper 95% confidence doses for inhalation thyroid and bone are still lower than the upper 95% confidence whole body gamma dose by factors of 4 and 140, respectively.

These lower doses are due mainly to plateout of iodine and metallic fission products primarily in the PCRV and to a lesser extent in the containment building, since in the case of HTGRs the PCRV acts as a conventional barrier against iodines and metallics. Also, holdup of metals in the graphite represents a contributing effect. In particular, Sr, Ce, and Ru are

predicted to plate out extensively, while I, Te, Sb, and Cs have relatively more moderate plateout characteristics. However, Cs experiences a significant delay in graphite. As a result, Te-132 and Sb-125 are the highest contributors to the median inhalation bone dose, while strontiums and cesiums are less contributing. The important fission product transport mechanisms are the rate of release (primarily Xe-133 and Xe-135) from the fuel particles, the fuel particle coating failure characteristics as a function of temperature, and plateout (primarily of iodines) in the PCRV and containment.

For the reheater tube leak event, the important dose is the whole body (primarily 30-day LPZB) dose resulting from direct and immediate release of gasborne activity to the atmosphere. Thus, the dominant isotopes in the gasborne inventory, chiefly Kr-88 and Kr-87, also are key contributors to the dose. This indicates that the primary coolant inventory parameters, e.g., release rate to birth rate ratio (R/B) and possibly gasborne dust (owing to the current low level of technical information), are of significance.

For the steam ingress event, release of noble gases due to hydrolysis of UC_2 kernels with prior failed coatings is the largest contributor to the doses, outweighing initial gasborne activity release. Thus, the important fuel body noble gas, Xe-133, is the key isotope contributor for the release of fission products into the containment during a steam ingress event. Secondary contributors are Kr-88 and Rb-88. The important parameters that affect fission product release are, in order, initial failed fuel fraction, hydrolysis reactions, and diffusion of steam in graphite, all of which affect the release due to hydrolysis.

Depressurization events are the only case analyzed where the thyroid dose and, to a lesser extent, the bone dose become important in the index of safety significance. The major source of these doses comes from liftoff of plated-out fission products and blowdown of activity on potentially circulating dust. By far the most significant fission product parameter

is plateout or fallout, principally of iodines, in the containment. Corresponding key isotopes are I-133, I-131, and Ce-144. Other important fission product mechanisms are the initial plateout inventory and the partition between plateout on dust and on metallic surfaces.

2. INTRODUCTION

2.1. BACKGROUND

The primary objective of the AIPA study is to provide guidance for HTGR safety research and development. Another objective is to consider alternative design options through use of risk assessment techniques. In the preliminary scoping study reported in Volume III, a coarse-cut analysis of seventeen initiating events was performed to select for more detailed evaluation the few events that are expected to contribute most to the attainment of the primary AIPA objective. In this volume are reported the detailed consequence evaluations performed as a part of the Phase I risk assessment described in Volume IV. Additional consequence evaluations are reported to provide a basis for more meaningful design option studies (Volume III) and in particular to provide direct guidance to HTGR safety R&D concerning fission product parameters. This study makes use of the data presented in Volume V for the initial inventories of fission products. A block diagram showing the relationship between Volumes III, IV, V, and VI is presented in Fig. 2-1.

2.2. OBJECTIVES

Within the framework of the AIPA objectives, three objectives were established for the consequence analyses reported in this volume:

1. In support of the detailed risk assessments documented in Volume IV, perform realistic consequence assessments of the appropriate event sequences, taking into account parameter and modeling uncertainties.

Fig. 2-1. Block diagram interrelationships of Volumes III through VI

2. Update (from the preliminary analysis) the consequence assessments of selected relatively low risk design basis events to provide a more meaningful basis for design option trade-off studies and for satisfaction of the third objective over the complete spectrum of events where current fission product research is being applied.
3. Identify key fission product isotope transport mechanisms that contribute to the statistically integrated offsite dose consequences, the purpose being to provide guidance for fission product R&D programs by quantitatively establishing the relative safety importance of supporting data.

The three basic outputs of this volume, as shown in Fig. 2-1, are consistent with these three objectives.

2.3. SCOPE

The scope of these analyses includes the detailed evaluation of the statistically median consequences based on realistic analysis assumptions and associated uncertainties for the events selected in the preliminary scoping study (Volume III). In addition, the quantitative safety R&D significance of fission product transport mechanisms, parameters, and data is identified in comparison with other parameters (design, meteorology, material, etc.) which constitute sources of uncertainty in the consequences. This effort is unique to HTGR research in that the methodology is based primarily on risk assessment and statistical error analysis techniques, as described in Volume II.

In accordance with the findings of the preliminary analysis, three initiating events were selected for the more detailed risk assessment presented in Volume IV: reheater leak (RHL), loss of offsite power (LOSP), and earthquake (EQ). In conjunction with the event trees developed for the LOSP and EQ events, it was found that all points of relatively high risk involve the core heatup accident condition. Hence, the satisfaction of

objective 1 above entailed performing a detailed consequence assessment for the two accident conditions of reheater tube leak and core heatup.

A short summary description of and justification for all events analyzed in this report is presented in Table 2-1. The RHL has a relatively simple consequence evaluation, while the core heatup involves complex fission product transport and retention mechanisms such as fuel failure, diffusion and evaporation in graphite, and transient plateout on metal surfaces. Thus, core heatup is also of particular importance for satisfaction of objective 3 above.

Detailed risk analysis of the steam ingress accident condition [the initiating event is a steam generator economizer-evaporator-superheater (EES) tube rupture] provides a basis for a more rigorous and meaningful design option study to satisfy objective 2 above. This event has received considerable attention in previous HTGR safety analyses since it is unique to the HTGR and since it involves the design of the moisture monitor, PCRV pressure relief, and steam generator isolation and dump systems. Considerable fission product and primary coolant chemistry technology is applied in the consequence evaluation of this event. Thus, this event is also important for the satisfaction of objective 3 above.

Another candidate for design option studies but not included in the FY-75 design option work was the class of PCRV depressurization events. In this volume, the depressurization events were analyzed to satisfy the consequence data needs anticipated for other design option studies that are projected for the future. In addition, the events incorporate significant fission product research data so that detailed consequence evaluation is pursuant to the more complete satisfaction of objective 3 above.

The sequence of steps was analysed from fission product birth through distribution under operating conditions (Volume V) to anticipated exposure and plant shielding requirements (Volume VII) and predicted dose at HTGR

TABLE 2-1
DESCRIPTION AND JUSTIFICATION OF EVENTS ANALYZED IN VOLUME VI

Initiating Event	Accident Condition	Event Choice Justification	Type of Fission Product Release
Relatively High Risk Events			
SG reheater tube leak (RHL)	Leakage of primary coolant activity through reheater to air ejector exhaust stack	Relatively high risk due to relatively high occurrence probability. Consequences were not extensively analyzed previously since they fall far short of regulatory limits.	Direct gasborne activity release to environment
LOSP or EQ	Core heatup	Relatively high risk due to relatively high consequence. Abundant and complex fission product transport phenomena are involved.	Gasborne and fuel body activity release to containment
Selected Relatively Low Risk Events			
Steam generator EES tube rupture	Steam ingress to primary coolant	Basis for design of moisture protection and SG isolation and dump systems. Subject of design option study. Analysis incorporates extensive fission product transport data.	Gasborne and limited fuel body activity release to containment
Large penetration closure failure (DBDA)	Rapid depressurization of primary coolant	Basis for depressurized CACS performance requirements and design of containment structure. Benchmark event for licensing. Involves significant fission product transport mechanisms.	Gasborne inventory and liftoff of plate-out activity released to containment
Failure of connecting piping to primary coolant boundary	Slow depressurization of primary coolant	Higher risk than DBDA (Vol. III) but quantification of low doses for generic class of depressurizations can serve as a basis for a containment design option study, if desired.	Gasborne activity release to containment

site boundaries for potential accidental releases (present volume). The basic radiological consequence model is discussed in Section 3, while more general aspects of the consequence methodology are presented in Volumes II and V.

3. CONSEQUENCE MODELING ASSUMPTIONS

3.1. PLANT CONDITIONS

Consequences of the events analyzed in this volume are based on realistic modeling of the effect of release of radioactive fission products from the reference 3000-MW(t) HTGR to the environment. The plant is assumed to be located at a typical site with EAB at a radius of 500 m and LPZB at a radius of 2500 m. In contrast to the preliminary scoping analysis, which focused on 30-day LPZB whole body gamma doses, the analysis of this report considers both the 2-hr EAB and 30-day LPZB doses of the following types:

External exposure whole body penetrating gamma dose

External whole body beta dose

Inhalation iodine contributing to thyroid dose

Inhalation bone-seekers contributing to bone dose.

These are the major doses based on general practices of nuclear reactor safety. Reasons for and details of considerations of these doses are presented in Sections 3.4 and 3.5.

The initial fission product inventories and decay heat generation rate are dependent, to some degree, on the plant operating history. Inventories of long-lived isotopes with half-lives on the order of months or years are affected by prior periods of shutdown and decay and, in general, by plant capacity factors. Inventories of isotopes with intermediate half-lives

on the order of hours or days are affected by the plant operating power levels in the days immediately prior to postulated occurrence of the initiating event.

The effect of reactor operating history (shutdowns and load-following transients) is illustrated qualitatively in Fig. 3-1. The most significant change in fission product inventory, in terms of fractional change and response time, is observed in the isotopes of intermediate half-lives. While it may be possible to derive from existing compilations of gas-cooled reactor operating history a statistically median plant capacity factor and average power factor, this was outside the scope of the present analysis. For the decay heat calculation and the Volume V derived fission product inventories, the design HTGR capacity factor* of 80% was assumed neglecting any shutdown time. Also, the operating power prior to the initiating event was assumed to be 100%. The former assumption is consistent with typical capacity factor for gas-cooled reactor experience (Ref. 3-1).** However, the assumed 100% operating power is conservative considering availability*** experience with other gas-cooled reactors (Ref. 3-1). It is estimated that an average operating power**** around 89% is appropriate for periods when the HTGR is operating. Since this would affect total fission product inventories and alter heat rates, this effect should be investigated further in future work.

*Capacity factor is the quotient of the actual net electrical energy output produced by the unit in a given period divided by the maximum net electrical energy output the unit would have produced if operated at the maximum dependable capacity for the entire period.

**References are listed at the end of the section.

***Availability is the quotient of the time the unit actually operated in a given period divided by the total time in the given period.

****Average operating power is the quotient of the capacity factor divided by the availability, or the quotient of the total energy produced divided by the total energy possibly produced for a given time period (P_{avg}/P_{max}).

Fig. 3-1. Effect of shutdown time (a) and load-following transient (b) on fission product inventory

3.2. FISSION PRODUCT INVENTORIES

3.2.1. Fuel Body Activity

The median fuel body activities for the pertinent isotopes and the activity at the 95% upper confidence limit are presented in Volume V and summarized here. In general, the median values and upper 95% confidence values fall below values used for the current HTGR design and PSAR accident analysis calculations.

Initial fuel body activity is defined and calculated in Volume V as the inventory of a nuclide contained in a core fuel element including graphite and fuel rods during steady-state operation at a core-averaged location. Table 3-1 summarizes the fuel body activities classified according to groups of isotopes that affect the whole body dose (noble gases), the inhalation thyroid dose (iodines), and the inhalation bone dose. This tabulation gives the activities for the most important nuclides, the ones that are the greatest potential contributors to the consequences of the events analyzed in this report. Total core inventories are obtained by multiplying the average element 4-year values by 3944, the number of fuel elements in the core, and adjusting for an equilibrium core at the middle of cycle (e.g., one-fourth of the elements are each 0.5, 1.5, 2.5, and 3.5 years of age). Total core inventories for major dose contributors are presented in Table 3-1. Note that only the long-lived isotopes, e.g., Sr-90, require adjustment for fuel age, since the remaining isotopes reach equilibrium rapidly.

The results show that the AIPA median values for fuel body activity average about 13% less than current HTGR design values. Also, the values are very well known, with the upper 95% confidence values being slightly greater than 1% of the median value for each isotope. Noble gases consist of about one-eighth of the total inventory but, because of their high mobility, comprise the most important class of isotopes for the dose

TABLE 3-1
FUEL BODY ACTIVITY INVENTORIES (FROM VOLUME V)

Isotopes	Average 4-Year Fuel Element (Ci)			AIPA Median Equilibrium Core Inventory (Ci)
	AIPA Median Values	95% Confidence Deviations(a)	HTGR Design Values	
Noble Gases				
Kr-88	3.07 x 10 ⁴	300	3.73 x 10 ⁴	1.21 x 10 ⁸
Xe-133	4.06 x 10 ⁴	500	4.59 x 10 ⁴	1.60 x 10 ⁸
All others	<u>24.3 x 10⁴</u>	<u>3100</u>	<u>28.8 x 10⁴</u>	
Total	31.3 x 10 ⁴	3900	37.1 x 10 ⁴	1.23 x 10 ⁹
Iodines				
I-131	2.03 x 10 ⁴	200	2.29 x 10 ⁴	8.01 x 10 ⁷
I-133	4.06 x 10 ⁴	500	4.59 x 10 ⁴	1.60 x 10 ⁸
All others	<u>12.2 x 10⁴</u>	<u>1300</u>	<u>14.8 x 10⁴</u>	
Total	18.3 x 10 ⁴	2000	21.7 x 10 ⁴	7.22 x 10 ⁸
Metallics				
Sr-89	3.48 x 10 ⁴	400	3.91 x 10 ⁴	1.36 x 10 ⁸
Sr-90	2.99 x 10 ³	34	3.25 x 10 ⁴	7.44 x 10 ⁶
Cs-134	5.11 x 10 ³	60	5.36 x 10 ³	1.43 x 10 ⁷
Cs-137	2.94 x 10 ³	35	3.15 x 10 ³	7.31 x 10 ⁶
Te-132	2.78 x 10 ⁴	300	3.00 x 10 ⁴	1.10 x 10 ⁸
All others	<u>201.0 x 10⁴</u>	<u>3.5 x 10⁴</u>	<u>228.2 x 10⁴</u>	
Total	<u>208.4 x 10⁴</u>	<u>3.6 x 10⁴</u>	<u>239.2 x 10⁴</u>	8.1 x 10 ⁹
Grand total	258 x 10 ⁴	4.2 x 10 ⁴	298 x 10 ⁴	10.1 x 10 ⁹

(a) Deviation is defined as 95% confidence level minus median value. See Fig. 3-2 for lognormal distribution.

assessments. The most abundant noble gas, by a slight amount, is Xe-133, and with its 5.27-day half-life it is generally the most important contributor to the whole body dose. Kr-88 is among the most abundant of the kryptons and, for some events, is a significant contributor.

Iodines, which contribute to the inhalation thyroid dose, make up less than 10% of the total fuel body activity. The remaining isotopes comprise about 80% of the total fuel body activity, but most of these nuclides are relatively nonvolatile, and this fact reduces their effect on the dose consequences. Because iodines plate out in the PCRV and containment, they are not major contributors to the whole body dose in AIPA sequences.

3.2.2. Steady-State Circulating (Gasborne) Activity

The detailed justification of initial gasborne activity levels in the primary coolant is presented in Volume V of this report. An overview of the results, showing contributions of the nuclides important in the event consequences analyzed here, is presented in Table 3-2. The realistic AIPA analysis resulted in a much lower contribution of iodines and metallic isotopes to the total gasborne activity than in the HTGR design calculations. Thus, the median AIPA primary coolant activity consists of 99% noble gases compared to about 75% in design calculations. Of these noble gases, Kr-88 is the largest single contributor; Xe-133 is about one-third as abundant. Xe-135 is the second most abundant circulating noble gas.

Compared to design values, the median steady-state circulating activity levels are about 20 times lower for noble gases, about 500 times lower for iodines, and about 1000 times lower for volatile metals. Since every individual isotope was not analyzed, the total median gasborne particulate activity is not known precisely. An estimate was therefore made for the purpose of comparing the total median gasborne activity with the design value as shown in Table 3-2. A total of 7.6×10^3 Ci was determined for the median total gasborne activity, which is about 27 times lower than design.

TABLE 3-2
PRIMARY CIRCUIT GASBORNE AND PLATEOUT ACTIVITY LEVELS (FROM VOLUME V)

Isotope	Gasborne Activity (Ci)			Plateout Activity (Ci)		
	Median Value	Uncertainty ^(a)	HTGR Design Value	Median Value	Uncertainty ^(a)	HTGR Design Value
Noble Gases						
Kr-88	1750	940	3.94×10^4	--	--	--
Xe-133	596	363	8.65×10^3	--	--	--
All others	<u>5239</u>	<u>2986</u>	<u>10.8×10^4</u>	--	--	--
Total	7585	4289	15.6×10^4			
Iodines						
I-131	7.29×10^{-2}	0.32	41.1	8220	4850	12.0×10^4
I-133	0.445	2	258	5580	3290	8.37×10^4
All others	<u>8.6</u>	<u>37.6</u>	<u>4005</u>	<u>1.2×10^4</u>	<u>6860</u>	<u>17.54×10^4</u>
Total	9.1	39.9	4304	2.6×10^4	1.5×10^4	37.9×10^4
Metallics						
Rb-88	2.14	4.3	8.36×10^3	1760	920	4.09×10^4
Sr-90	6.8×10^{-7}	1.51×10^{-5}	7.86×10^{-3}	346	7643	1.54×10^4
Tc-132	2.9×10^{-2}	0.47	85.1	6390	3770	8.82×10^4
Cs-137	1.12×10^{-4}	3.1×10^{-3}	8.77×10^{-2}	4.02×10^4	1.2×10^6	15.9×10^4
Ce-144	6.56×10^{-7}	1.9×10^{-5}	1.54×10^{-2}	12.4	348	1.39×10^3
Cs-134	2.87×10^{-4}	8.0×10^{-3}	0.243	1.44×10^4	4.04×10^5	5.78×10^4
All others	--	--	--	--	--	<u>49.7×10^4</u>
Total	<u>~50</u>	--	<u>5.0×10^4</u>	<u>$\sim 13 \times 10^4$</u>	--	<u>86×10^4</u>
Grand total	~7644	--	21.0×10^4	$\sim 16 \times 10^4$	--	124×10^4

(a) Deviation in median value to upper 95% confidence level, defined as 95% confidence level minus median. See Fig. 3-2 for lognormal distribution.

Uncertainties in the median gasborne activity are relatively large compared to fuel body activity uncertainties but are substantially less than the factor of 10 applied for "design" inventories over "expected" inventories. For noble gases, the upper 95% confidence value represents an increase of about 56% over the median values. For iodines and the volatiles, the deviations are much larger (e.g., a factor of 5-1/2 for iodines), but, since these isotopes make up only about 1% of the gasborne activity, these uncertainties are not controlling.

3.2.3. Primary Circuit Plateout Activity

Median values and uncertainties for the plateout activity in the primary circuit are derived in Volume V of this report. These values pertain to 40-year plateout inventories. While the use of 40-year plateout activities may seem more appropriate for end of plant life than a more probable intermediate time, plateout activities at, for example, 20 years are the same as for 40 years for all isotopes of interest except the long-lived isotopes, e.g., Sr-90 and Cs-137, which would have activities roughly 25% lower. The only event which this affects is the DBDA, discussed in Section 5.2; the quantitative perturbations on the results of that event are also discussed in that section.

Table 3-2 also summarizes plateout activities for the pertinent isotopes of interest in the consequence analyses presented in this volume. Since all isotopes were not included in the analysis, the total median plateout inventory was estimated by averaging the change from design values for the important contributors and applying this to the nonanalyzed isotopes. Major contributors to the plateout activity are the cesiums (mainly 134 and 137), iodines, telluriums, and barium (Ba-137m). Strontiums have a very small plateout contribution, in terms of curies. Iodines comprise about one-sixth of the total median activity, which is a decrease from their relative contribution to design values. The cesiums comprise about one-third of the median plateout activity and are the largest element contribution to plateout activity.

Compared to design values, the median plateout activities are lower by factors that vary from element to element. Iodines are a factor of 15 lower, cesiums are 5 times lower, Sr-90 is 50 times lower, telluriums are 14 times lower, and Ce-144 is 112 times lower. The total median plateout inventory is estimated at about 1.6×10^5 Ci, which is about 8 times lower than design.

The deviations of the 95% upper confidence limit from the median iodine plateout activity can be seen from Table 3-2 to be just over 50% of their respective median values. For the volatile metallics, the uncertainties vary widely with different elements. For telluriums and rubidiums, the uncertainties are similar to iodines, while for cesium, barium, cerium, and all other metallics whose release is determined by the minimum R/B, the deviations range over orders of magnitude.

3.3. MEDIAN SOURCE TERMS AND UNCERTAINTIES

Calculations of the "source terms," or transient fission product (curie) release to the containment during the postulated event, are characterized in this volume by more realistic modeling of the transient phenomena than was the case in the preliminary analysis (Volume III). Table 3-3 summarizes the Phase I modeling assumptions. Note that the scope of the present analysis did not include completely realistic modeling in the source term computer programs such as OXIDE-3, CORCON, and SORS, which are described in the Appendix to Volume II. An effort was made to eliminate major sources of conservatism. However, within the scope of the FY-75 work it was not practical in all cases to remove all inherent conservatisms and assumptions that tend to maximize the curie releases. Several examples of conservatisms retained and removed are presented in Table 3-3.

The overall analytical procedure followed in the consequence and uncertainty assessment is summarized in Table 3-4. The analysis follows the procedures and method of a standard uncertainty analysis. The final

TABLE 3-3
SOURCE TERM MODELING ASSUMPTIONS IN VOLUME VI

- Event occurs at 100% power, assuming an 80% load factor.
- Initial fission product inventories for an equilibrium core at middle of cycle (except plateout as noted).
- Nominal plant operating conditions and transient response.

Examples:

- a. Nominal core temperatures and flow
- b. Nominal setpoints and response times of protective systems
- c. Nominal release mechanisms such as, in core heatup, coupled diffusion and evaporation of fission products from graphite

Exceptions:

- a. Same core power distributions used as in PSAR
- b. Conservative modeling assumptions unchanged where no realistic assumption can be identified, such as, in steam ingress, taking no credit for prior release of noble gases from failed fuel due to hydrolysis during steady-state operation.

TABLE 3-4
CONSEQUENCE AND UNCERTAINTY EVALUATION PROCEDURE

1. Establish which events are to be analyzed based on risk evaluation and fission product technology significance.
2. Calculate median parameter values for those parameters which affect doses.
3. Determine probability distribution for the parameter values and evaluate each parameter 95% confidence bounds.
4. Use the median values in the analytical models to compute the median dose.
5. Determine the influence of each parameter on the dose by varying each parameter singly in the respective analytical model to 95% confidence limits.
6. Establish probability distribution of dose consequences based on the parameter influence results, and, from the law of propagation of errors, calculate the 95% confidence bound on dose.
7. Compare the safety R&D significances of each fission product parameter based on the median dose value and uncertainties, to establish goals and priorities.
8. Use median dose values and uncertainties as input to the more detailed Phase I risk assessments presented in Vol. IV and to design option study in Vol. III.

products of the study are median values and uncertainties of event safety consequences and an indication of the influence of the major parameters for representative accident sequences consistent with the objectives specified in Section 2.2.

The first step in the analysis of each event is the review and specification of parameters. A list of parameters that affect the dose is established. For each parameter, median values and 95% confidence bounds are evaluated. Several probability distributions are used in these evaluations.

The normal (or Gaussian) probability distribution is the most familiar and was used in parameter uncertainty analysis where the real distribution of data could be approximated by the Gaussian distribution. Another distribution that is useful in the statistical analysis of analytical parameters is lognormal. The lognormal distribution enters where the logarithm of parameter values has a Gaussian distribution, and uncertainty factors are used to describe the statistical dispersion in parameter values. The uniform probability distribution was used in the parameter evaluation when all parameter values in a specific range are equally probable. It was chosen to express a lack of data except for upper and lower bounds. Examples of each of the distributions and calculations are given in Fig. 3-2.

The objective of the statistical analysis is to make a confidence statement on radiological consequence. This can be done with the following information:

1. Distribution - variable-consequence characteristics
2. Median - median variable values and random uncertainties
3. Variance - propagation of errors

<p>GAUSSIAN (OR NORMAL) DISTRIBUTION</p>	<p>PROBABILITY</p>	$\mu = \frac{\sum_{i=1}^M x_i}{M}$ $\sigma^2 = \frac{\sum_{i=1}^M (x_i - \mu)^2}{M - 1}$ $x_{0.95} = \mu + 1.645\sigma$ <p>EXAMPLE HTGR DECAY HEAT</p>
<p>LOGNORMAL DISTRIBUTION</p>	<p>PROBABILITY</p>	$\mu_{\ln} = \frac{\sum_{i=1}^M \ln x_i}{M}$ $x_M = e^{\mu_{\ln}}$ $\sigma_{\ln}^2 = \frac{\sum_{i=1}^M (\ln x_i - \mu_{\ln})^2}{M - 1}$ $x_{0.95} = e^{\mu_{\ln} + 1.645 \sigma_{\ln}}$ <p>EXAMPLE: GASBORNE FP INV.</p>
<p>UNIFORM DISTRIBUTION</p>	<p>PROBABILITY</p>	$\mu = \frac{MAX + MIN}{2}$ $\sigma^2 = \frac{(MAX - MIN)^2}{12}$ $x_{0.95} = 0.05 MIN + 0.95 MAX$ <p>EXAMPLE: PLATEOUT INVENTORY PARTITION ON CARBON DUST (0 OR 100%)</p>

Fig. 3-2. Probability distributions of parameters used to calculate consequences

Using the median parameter values in the analytical models, median dose values were calculated. Then, the influence of each parameter on the doses was determined by varying each parameter singly to its 95% confidence bound. From the results of this sensitivity study, it was established that a lognormal probability distribution best fit the dose versus parameter, i.e., the log of the dose varied linearly with the log of the parameter. Based on the lognormal distribution, the law of propagation of errors was applied for the 95% confidence bound on the doses.

Consider a particular dose D and a generalized variance σ for a given event sequence:

$$\log D = \log [D(\log a_1, \log a_2, \dots \log a_n)] \quad (3-1a)$$

$$\begin{aligned} \sigma_{\log D}^2 = & \sum_{i=1}^n \left(\frac{\partial \log D}{\partial \log a_i} \right)^2 \sigma_{\log a_i}^2 \\ & + 2 \sum_{i=1}^n \sum_{j>i}^n \rho_{\log a_i \log a_j} \left(\frac{\partial \log D}{\partial \log a_i} \right) \\ & \times \left(\frac{\partial \log D}{\partial \log a_j} \right) \partial_{\log a_i} \partial_{\log a_j} \quad (3-1b) \end{aligned}$$

Here, the calculated dose D is a function of n input parameters, $a_1, a_2, \dots, a_n$, and the variance is a function of parameter variance $\sigma_{\log a_i}$ and correlation coefficient $\rho_{\log a_i \log a_j}$.

The uncertainty in dose is associated with the uncertainty in many contributing variables. Some variables are functions of space or time, and others are independent of both space and time. This variation or lack of variation allows the uncertainties to be characterized as random or systematic, respectively. Random uncertainties tend to average out, whereas

systematic uncertainties do not. Thus, this study emphasizes the evaluation and propagation of systematic uncertainties and does not consider random variations. A more detailed description of this comparison is contained in Volume V.

For the majority of the dose evaluation, variables are uncorrelated, $\rho_{\log a_i \log a_j} = 0$, and since the random uncertainties tend to average out over the system, the median dose is calculated for nominal input parameters and the variance becomes the more familiar form for law of propagation of errors:

$$\log D = \log [D(\log a_1, \log a_2, \dots \log a_n)] \quad (3-2a)$$

$$\begin{aligned} \sigma_{\log D}^2 &= \sum_{i=1}^n \left(\frac{\partial \log D}{\partial \log a_i} \right)^2 (\sigma_{\log a_i})^2 \\ &= \sum_{i=1}^n (\Delta_i \log D)^2 = \sum_{i=1}^n (\log D_i - \overline{\log D})^2 \quad (3-2b) \end{aligned}$$

Now, a deviation in any particular parameter from the median value will result in a corresponding deviation in the median dose. The corresponding dose value for each parameter can be determined by varying each parameter singly to its 95% confidence limit, $\log a_{i,95\%}$. Thus, the dose corresponding to a 95% confidence limit may be calculated

$$\begin{aligned} \log D_{i,95\%} &= \log [D(\log a_1, \log a_2, \dots, \\ &\quad \log a_{i,95\%}, \dots, \log a_n)] \quad (3-3) \end{aligned}$$

Proceeding in this manner for each parameter, the overall 95% confidence limit uncertainty factor f can be calculated by law of propagation of error,

$$\Delta \log D_{95\%} = \exp \left[\sum_{i=1}^n (\log D_{i,95\%} - \log D)^2 \right]^{1/2}, \quad (3-4a)$$

$$f = \exp (\Delta \log D_{95\%}) \quad (3-4b)$$

Thus, a statistical statement of the radiological consequence of an event may be characterized by a lognormal distribution with a median dose D_{median} and error factor f , e.g.,

$$D_{\text{median}} = e^{\log D}, \quad (3-5a)$$

$$95\% \text{ confidence limit} = f \times D_{\text{median}}. \quad (3-5b)$$

3.4. DOSE CRITERIA AND HEALTH EFFECTS

To identify the safety R&D significance of fission product release mechanisms in the analyzed events, parameters that have the strongest effect or cause the greatest deviations in the median consequence were determined. The change in dose D due to parameter variation from the median value to its 95% confidence limit is one of the significant quantities denoting the effect of each parameter on the total uncertainty. Thus, reduction in the uncertainty associated with the highest ranked parameters is the efficient means of reducing the total uncertainty.

It is understood that a parameter can have safety significance either by exerting a moderate effect on relatively high risk events or by exerting a pronounced effect on lesser risk events. To cover both possibilities it is necessary to compare event consequence predictions with some measure of safety significance such as regulatory dose guidelines or limits of biological damage.

A major drawback of these possibilities is that they do not portray any rigorous assessment of relative importance of doses to different organs. Perhaps the most far-reaching assessment of this kind to date is provided by the EPA in its environmental study of the uranium fuel cycle (Ref. 3-2). Guidelines were derived for health effects appearing in exposed individuals for different kinds of doses (e.g., skin, whole body, thyroid, and bone). Therefore, for the present analysis, the guidelines established by the EPA were selected as a basis of comparison. It is recognized that these guidelines are not claimed to have a comprehensive data base; neither are they final. However, they were deemed appropriate for satisfaction of the stated objective 3 of this work.

Depending on the severity of the accident, the health effect consequence could include immediate acute effects such as mortality and radiation sickness or else late-manifested effects in the exposed individual or his offspring.

Acute exposures to the whole body in the range of 100 to 1000 rems produce clinical effects that range from minor illness to possible death of an individual. Exposures of less than 100 rems delivered to the whole body produce nothing other than minor temporary blood chemistry changes; disabling sickness does not occur, and little chance of death exists (Ref. 3-3).

Studies of potential HTGR event sequences analyzed to date have demonstrated that associated exposures are considerably less than the 100-rem threshold for significant health effects. Therefore, acute health effects could not be utilized to effectively meet objective 3 for this work. On the other hand, HTGR event consequences are predicted near the threshold of latent health effects.

Latent health effects are manifested in various forms of somatic effects resulting from exposure to radiation, including genetic modification, nonspecific life shortening, and cataracts. The information

available does not permit quantitative estimates of all possible health effects at this time (Ref. 3-4). However, data have been compiled to quantify observable health defects in exposed individuals. Thus, latent health effects for this study will be represented by EPA-tabulated somatic risk values for the exposed individual. Estimates of these somatic risks are provided in Table 3-5, based on information developed by the EPA.

To apply the health effects model to the various events, a conversion factor from rem to man-rem is required. The method used was similar to described in WASH-1400 (Draft), Appendix VI. The population density, $\rho(x,y,\tau)$, was based on Fig. VI-6 and on the evacuation model of Section 6.5.2 of WASH-1400. The conversion factor from rem to man-rem for a specific site boundary (and release height) is calculated as

$$\text{Conversion factor } \left(\frac{\text{man-rem}}{\text{rem}} \right) = \int_{x,y,\tau} \rho(x,y,\tau) \times \left[\frac{\dot{D}(x,y,\tau)}{D(\text{boundary},t)} \right] dx dy d\tau \quad (3-6)$$

where $\dot{D}(x,y,\tau)$ is the dose rate as a function of location and time, x is the downward direction, and y is the direction normal to x , parallel to ground direction.

The upper limit of integration was assumed in the present study to be 50 miles for all event sequences. Use of a constant conversion factor is a simplification of the WASH-1400 model, which evaluates the man-rem effects to a cutoff distance dependent on the dose consequences of a particular event. Such variable integration distance models were not available for the present AIPA consequence analysis but are intended to be incorporated in future work. The 50-mile integration limit corresponds to the distance at which the dose to an individual for the highest consequence AIPA event branch (LOSP BF) is about 1% of typical 30-year background radiation exposure. References 3-5, 3-6, and 3-7 tend to support a 50-mile distance.

TABLE 3-5
EPA SOMATIC HEALTH EFFECT FACTORS

Critical Organ	Health Effect	Health Effect Conversion Factors Events Per Man-rem		
		Mortality	Nonfatal Cancers	Total Cases
Total soft tissue organs other than bone or thyroid ^(a)	Cancers	2×10^{-4}	2×10^{-4}	4×10^{-4}
Thyroid ^(b)	Thyroid cancers	0	5.6×10^{-5}	5.6×10^{-5}
Bone	Cancer of the skeleton	1.6×10^{-5}	1.6×10^{-5}	3.2×10^{-5}
Skin	Cancer of the skin	0	3×10^{-6}	3×10^{-6}

(a) Includes leukemia mortality. Exposure pathway whole body irradiation.

(b) Average risk value obtained using population age/susceptibility weighting.

TABLE 3-6
CONVERSION FACTORS FROM REM TO MAN-REM USING WASH-1400
POPULATION MODEL

Site Boundary	Release Height (m)	Conversion Factor	Events
2 hr 500 m EAB	Ground level	16	Depressurization events, core heatup, steam ingress
30 day 2500 m LPZB	Ground level	390	
2 hr 500 m EAB	30	22	Reheater tube leak
30 day 2500 m LPZB	30	530	

Thus, the man-rem conversion factor is conservatively high for all events since their consequences are less severe than those in LOSP Branch BF.

The ratio of dose rates is based on differences in radioisotope atmospheric dispersion and decay as a function of location. These factors convert radiological dose at a specific site boundary and cumulative time to effective population-dose consequences. Table 3-6 presents the calculated conversion factor for rem to man-rem at several site boundaries and cumulative times.

Multiplication of the median calculated dose converted into man-rem times the appropriate health effect factor listed in Table 3-5 yields a median consequence in terms of individual organ health effect. Comparisons of the safety importance of specific parameters affecting the consequence can be made using these organ health effects and including the corresponding uncertainty in the consequence caused by the uncertainty of each parameter. To incorporate these uncertainties mathematically, a multiplication factor was derived as represented by the ratio $(U_T - U_{T-j})/U_T$, where U_T is the total uncertainty in the organ dose for all parameters and U_{T-j} is U_T without the uncertainty U_j due to a specific parameter j . This ratio is the fractional change in the total uncertainty due to elimination of the uncertainty associated with the parameter in question. Mathematically, this latter quantity is

$$U_{T-j} \equiv \exp [(\ln U_T)^2 - (\ln U_j)^2]^{1/2} \quad . \quad (3-7)$$

The ratio $(U_T - U_{T-j})/U_T$ qualitatively expresses the potential relative contribution of a parameter to reduce the total uncertainty in the specific organ dose. If a parameter uncertainty U_j is only a small part of U_T , the above ratio is close to zero. As $U_j \rightarrow U_T$, the ratio tends toward unity.

From the above discussion, the health effect index factor for a specific organ dose is

$$F_{\text{organ}} = \text{median dose (rem)} \times \text{conversion factor (man-rem/rem)} \\ \times \text{health effect (H.E./organ man-rem)} \\ \times (U_T - U_{T-j})/U_T, \quad (3-8)$$

where F_{organ} is either the whole body gamma dose, the thyroid dose, the bone dose, or the skin dose.* An indication of the parameter significance for the total individual radiation exposure F_T can be obtained from summing this factor over the kinds of doses for a given parameter in a given event:

$$F_T = \sum F_{\text{organ}} \equiv \left(F_{\text{whole body } \gamma} + F_{\text{thyroid}} + F_{\text{bone}} + F_{\text{skin}} \right) \text{ 30-day LPZB} \quad (3-9)$$

The LPZB factors are representative of the overall health effect for a particular event. A comparison of the total individual exposure factors F_T for the various parameters in a given event is used to identify which fission product transport mechanisms are most important regarding consequence uncertainties of that event. Section 6 presents these comparisons for all events analyzed.

*In this study, the health effect of the lung dose was generally not considered. An estimate of the health effect factors for the lung dose is contained in the safety R&D discussion (Section 6). Also, the lung dose contribution to the health effect factors of the remaining events was estimated as 10%. The consideration of lung dose would not affect the ranking of fission product parameters although it would increase the numerical importance factors for PCRV and containment plateout.

For the overall safety significance of a parameter that affects consequences of several events, the individual total exposure factors F_T are combined by weighting by the probability of event occurrence P in the summation over all events. Thus,

$$\text{parameter overall safety index} = \sum_{\text{events}} (F_T \times P)_i \quad . \quad (3-10)$$

The interpretation of this safety index is that it represents the total health effect of a parameter and its associated uncertainty on the risk (probability x consequence) summed over all events considered. The magnitude of the health effect is considered on a relative scale only for purposes of comparing parameter importance; no attempt is made to relate to any biological assessment at this stage of analysis.

In summary, median parameter values and uncertainties are established for the selected accident analysis. From these results, median radiological consequences and parameter influence on consequences are established. Using the law of propagation of errors, the overall uncertainties in dose are calculated. From the median dose, uncertainty, and EPA health effect model, a ranking of parameter influence is established. Relative comparisons have been used to establish a basis for guidance and priorities for safety research and development regarding fission product technology.

3.5. RADIOLOGICAL CONSEQUENCE MODEL

The evaluation of event consequences reported in this volume represents a second iteration improvement over the preliminary study results in Volume III. The present analysis incorporates newly calculated doses using the General Atomic dose models. More realistic consequences are obtained by starting with more realistic fission product inventories and incorporating more realistic parameter estimates for the time-dependent retention of fission products in the containment. Most notable among these are plateout and fallout of airborne radioactive material in the containment, and leak rate from the containment building. Thus, the doses for the

analysis reported here differ from the preliminary results due to the indicated differences in initial fission product inventories, and also due to more realistic modeling of fission product release source terms in the event models and more realistic modeling of fission product transport from the containment to the environment.

3.5.1. Containment Model

The transport of fission products from the containment building atmosphere to the environment is treated in a mechanistic and realistic fashion. Initially, over 260 separate radionuclide species may be present in the containment building atmosphere. The activity level of each nuclide on its pathway to the environment was determined by use of the analytical computer program TDAC (Ref. 3-8). TDAC is a computer program that calculates the time-dependent environmental radiological impact of HTGR nuclear power plant operations and accidents. The use of TDAC permits the prediction of both fission product transport (i.e., the spatial location and temporal behavior of released fission products) and the resultant environmental dose levels.

The fission product release to the environment from the containment depends directly on the (1) radionuclide inventory present within the containment at any given time and (2) the structural leak-tightness or integrity of the containment structure.

The initial source of radioactivity released to the containment may be categorized as follows:

Noble gases

Aerosol particulates

Halogens (including elemental iodine and methyl iodide)

The depletion mechanisms within the containment depend on the chemical and physical form of each specific fission product. In general, released materials will be present as solid aerosol particulates. The noble gases and halogens are present in gaseous form. The release of fission products to the containment is based on analytical models for the specific events analyzed. These models are discussed in the following paragraphs.

The fission product material existing in the HTGR containment at any time may be expressed by forming a source/sink balance on the entire gas phase:

$$\text{Accumulation} = \sum \text{sources} - \sum \text{removal sinks} \quad . \quad (3-11)$$

Some possible source terms for fission product input to the containment atmosphere are:

S_1 - release(s) from the PCRV

S_2 - desorption from structural surfaces in the containment

S_3 - desorption of activity from suspended particles in the containment

S_4 - production from vapor phase reactions

S_5 - radioactive decay of precursor radionuclides

In similar fashion, some removal mechanisms from the containment atmosphere are:

R_1 - diffusion across concentration gradients to plateout surfaces

R_2 - radioactive decay

R_3 - gravitational settling

R_4 - sorption onto suspended particles

R_5 - removal in vapor phase reactions

R_6 - leakage to the environs or other compartments

R_7 - removal by engineered filter systems.

Examination of the sources and sinks present in the containment leads to the conclusion that only a few of the terms are significant. Notably, the removal mechanisms of plateout (diffusion across concentration gradients to surfaces and absorption onto surfaces), radioactive decay, gravitational settling, leakage, and filtration are the more significant sinks and they are considered in the analysis.

The containment parameters and selected analysis assumptions (see Table 3-7) were combined with the event source terms(s) as described later in the accident analysis sections to develop the quantity of each radio-nuclide species released from the containment to the environment.

3.5.2. Dose Model

Doses to individuals in the environment are determined through the application of the TDAC computer code, which computes external submersion dose as well as internal inhalation dose. For this study, only those exposure pathways that contribute significantly to personnel exposure levels were considered and reported, i.e., external submersion whole body gamma and beta doses as well as inhalation pathway doses for the thyroid gland and skeleton bone. Dose levels resulting from ground deposition of released nuclides and food chain ingestion dose (e.g., contaminated water or foodstuff ingestion) were not considered, as their incorporation into the overall doses has a negligible effect on the consequences.

TABLE 3-7
PARAMETERS FOR RADIOACTIVE RELEASE INTO CONTAINMENT AND SUBSEQUENTLY TO ATMOSPHERE

Parameter	Median Condition	95% Confidence Interval	Comment
1. Containment atmosphere fission product removal system characteristics			
a. Engineered fission product removal system			
Startup delay after accident	1 hr	0.5-2 hr	Uncertainty is result of delays in startup of cleanup loop to allow for cooldown of containment atmosphere to level where filtration is assured.
Recirculation loop filtration rate	100% vol/hr	50-200% vol/hr	Rate consistent with current HTGR engineered system designs
Noble gas removal filter efficiency	0%	0%	Noble gases will not be removed to any significant degree
Particulate removal filter efficiency	99%	98-99.3%	Represents decontamination factor variation of $\pm 50\%$
Iodine removal filter efficiency	95%	90-96.7%	
b. Natural fission product removal mechanisms (i.e., fallout and plateout) for particulates and halogens only			

TABLE 3-7 (Continued)

Parameter	Median Condition	95% Confidence Interval	Comment
Initial early removal	1.4 hr ⁻¹	0.1-14 hr ⁻¹	Expected natural removal processes of gravitational settling and sorption and desorption processes are modeled in a fashion similar to that presented in Ref. 3-4. Uncertainties were assigned relying heavily on engineering judgment.
2. Containment building leak rate			Upper bound estimate consistent with current HTGR design. Lower bound estimate represents typical secondary containment with functioning secondary leak collection systems. See Section 5.3 of Volume II for detailed calculation method and uncertainties on non-isolated containment.
a. Isolated containment	0.5% vol/day	0.01-0.25% vol/day	
b. Nonisolated containment	0-1 hr	580% vol/hr	
	1-2 hr	12% vol/hr	
	2-8 hr	5.2% vol/hr	
	8-16 hr	0.48% vol/hr	
	16 hr +	0.04% vol/hr	
3. Containment release height	≈1 ft	Factors of 1/70 and 70	From containment building penetrations reference design, average release height and uncertainties were evaluated based on penetration number and size. Resultant change from ground level release to release from 1 ft is negligible, so that for calculational ease, ground level release was assumed. Uncertainty factor was applied in manner recommended for elevated release heights in Ref. 3-9. For

TABLE 3-7 (Continued)

Parameter	Median Condition	95% Confidence Interval	Comment
4. Atmospheric transport dilution factors			reheater accident, elevated release height of 30 meters was used and no uncertainty need be applied.
a. EAB, 500 m (sec/m ³)			See Volume II for detailed discussion on development of median atmospheric dilution factors and corresponding uncertainties.
Release height = 0 m	2×10^{-4}	3×10^{-3}	
Release height = 30 m	2×10^{-5}	2×10^{-4}	
b. 2500 m (sec/m ³)			
Release height = 0 m	8×10^{-6}	1×10^{-4}	
Release height = 30 m	4×10^{-6}	9×10^{-5}	
5. Biological factor breathing rate	2.32×10^{-4} m ³ /sec	N.A.	No uncertainties were assigned to breathing rate in this study.

The external whole body gamma and beta doses in air are calculated using the semi-infinite cloud approximation as described in Ref. 3-9. Corrections are made for decay and buildup of radioactive material enroute to the dose receptor location. However, fallout scavenging of the resultant plume is not considered, which results in simplified but slightly conservative doses.

The resultant external gamma and beta dose levels from all nuclides and their progeny are calculated as

$$D_{\gamma X}^{t_1-t_2} = 900 \sum_i \int_{t_1}^{t_2} \bar{E}_{\gamma X_i}(t) dt \quad (3-12)$$

and

$$D_{\beta X}^{t_1-t_2} = 828 \sum_i \int_{t_1}^{t_2} \bar{E}_{\beta X_i}(t) dt \quad (3-13)$$

Dose integration intervals of 0 to 2 hr or 0 to 720 hr (30 days) following the event occurrence were selected for this analysis. The former time is representative of doses that might be delivered to an area immediately adjacent to the plant prior to evacuation, or representative of accidents with fairly short radionuclide release durations, such as the reheat-leak event. The latter dose integration interval of 0 to 720 hr (30 days) is more representative of the total dose in surrounding areas for longer release durations (like core heatup). Extending this latter dose interval to time intervals greater than 720 hr results in only a minor (~1%) increase in whole body dose. Therefore, extension of the time interval in excess of 720 hr can be deemed unnecessary.

The inhalation pathway dose to the thyroid and skeleton are evaluated by the relation

$$D_{m,i} = \int_{t_{in}} \chi_i(t) \beta' \epsilon_{m,i} dt \quad , \quad (3-14)$$

where $D_{m,i}$ = dose to organ m from inhalation of nuclide i, rem,

t_{in} = inhalation time interval, sec,

$\chi_i(t)$ = airborne concentration of nuclide specie i
at time t, Ci/m³,

β' = individual breathing rate, m³/sec, and

$\epsilon_{m,i}$ = effectivity of nuclide i, referenced to organ m (organ-rem per Ci inhaled).

The dose levels $D_{m,i}$ for the various event sequences were estimated using the radionuclide sources described later in the event description sections combined with the meteorological and dose parameters described in Table 3-7.

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4. RELATIVELY HIGH RISK EVENTS

4.1. REHEATER TUBE LEAK

4.1.1. Description

The reheat steam system operates at a lower pressure than the primary coolant system over the normal operating range of the reactor. A leak in the steam generator reheater section inside the PCRV results in leakage of primary coolant helium into the reheat steam system. Detection of leakage of primary coolant into the reheat steam system is provided by activity monitors located in the hot reheat line from each of the six reheaters. These activity monitors are a part of the engineered plant protection system (PPS). If the activity monitors detect radioactivity in a hot reheat line, the affected reheater is isolated and the loop is shut down.

In this volume of the report, the following event sequence is analyzed for Branch LA, which was identified in Volume III as the highest risk leak:

1. Offset shear rupture of a reheater tube.
2. Isolation and loop trip signal on high activity at about 0.5 sec after activity reaches the monitor (PPS).
3. Cold reheat line isolated by helium circulator turbine stop valve 1 sec after high activity signal (PPS).
4. Hot reheat isolation valve closure 5 sec after high activity signal (PPS action).

Because of the delayed closure of the hot reheat isolation valve, uncontained leakage to the secondary coolant system occurs for about 5.5 sec. This would result in attenuated release to the environment through the 30-m air ejector exhaust (AEE) and subsequent radiological consequences.

Lower risk sequences for the reheater leak have also been analyzed for branches other than LA. They vary from the Branch LA sequence by virtue of different leak rate or different system or operator response.

4.1.2. Consequence Evaluation

The consequence calculations were performed as follows:

1. The median release rate from the primary coolant system for the three classes of leaks was established.
2. The median time to isolate for each event sequence was estimated.
3. The fission product activity release into the reheat section was calculated by multiplying the release rate and the isolation time. Because the condenser filters any halogens and particulates, the releases from the condenser air ejector and from the condenser bypass were calculated separately.
4. The TDAC code was used to calculate the dose at the EAB according to median weather conditions.
5. The uncertainty in consequence was determined for the branches with the highest risk.

Reheater leak sizes were classed according to helium leakage rates as indicated in Table 4-1. The classification was chosen to best describe the responses of the plant and its operator personnel to leaks of varying size.

TABLE 4-1
LEAK SIZE CLASSIFICATION FOR REHEATER TUBE LEAK

Leak Classification	Range of Helium Flow Rate (lb/sec)	Representative Log Average Helium Flow Rate (Uncertainty)
Small leaks	$8.2 \times 10^{-5} - 2.7 \times 10^{-3}$	4.7×10^{-4} (factor 5.7)
Intermediate leaks	$2.7 \times 10^{-3} - 0.27$	2.7×10^{-2} (factor 10)
Large leaks	0.27 - 8.2	1.5 (factor 5.5)

Large leaks are in the range of automatic PPS response; their consequences are sufficient to require immediate automatic response. Intermediate leaks are in the range of annunciation by the AAE monitors; their consequences are such that they do not require automatic response but the leak must be terminated by the operator in a short time. Small leaks are of serious consequence only if they remain unterminated for extended periods of time. The upper bound leak rate (95% confidence) was chosen to be that for an offset shear rupture of an individual reheater tube. The median value compares well with the 1 Ci/sec estimated value of a leak rate through a leak of 1-sq-in. flow area, as was reported in the literature (see Volume IV).

The isolation times for the branches were chosen as follows (see Volume IV for a description of these branches):

Branch LA: based on response time of the PPS system and closure time of the valves for offset shear rupture.

Branches LB, LD, LG, LJ, LM, IB, and IG: based on operator intervention time as described in Volume IV.

Branches LC, LE, LH, LK, and LN: based on the time the reheat steam is bypassed to the environment after a reactor or turbine trip.

Branches IC and IH: based on average repair time.

Branches SG and IA: based on average operator response time.

Branches SH, SJ, and SK: based on average operator response time and average repair time.

Branches ID, IE, IJ, IK, IM, IN, SM, and SN: based on time to reach low PCRV pressure trip.

Branches LF, LL, LO, IF, IL, IO, SL, and SO: based on release of circulating activity.

The median dose and uncertainty analysis included the effects of two fission product parameters. First, the equilibrium gasborne fission product inventories and uncertainties for the reheater leak consequences were those established in Volume V, as discussed in Section 3.2 of this volume. Also, the effect of gasborne dust on the fission product inventories has been estimated, and the calculational methods are discussed in Section 5.2 of this volume.

Using the equations for molecular diffusion in gas phase and mass transport to condenser surfaces, estimates of fission product plateout through the main condenser during a large reheater leak event have been made. These analyses include the fact that the solubility or sorptive capacity of iodine is large compared to the amount of iodine ingress for water and condenser surfaces available. Also, the comparatively low temperatures on condenser surfaces as compared to fission product melting points lead to the assumption that any nuclide colliding with the surfaces must adhere to the surface. The results show that a negligible amount of iodine is anticipated to be released in gaseous form from the main condenser. Plateout on fixed surfaces was estimated as 99% of the available iodine. Absorption into the water vapor or particles was estimated as 100% of available iodine. The combination of these effects would significantly reduce the release of iodine during the reheater leak event. Since the boiling point of iodine is much lower than that of other nuclides, except for the noble gases and tritium, this may be considered a limiting case. That is, all other nuclides except the noble gases will release less than iodine. Therefore, non-noble-gas fission products should not be released as gasborne nuclides from the condenser. The effects and estimate of condenser filtration time are discussed in Volume IV of this report.

Separate calculations were made for releases from the hot reheat bypass and from the condenser AEE stack. The release point for both was assumed to be at turbine building roof level (30 m).

4.1.3. Results And Uncertainties

A summary of the consequences for the various reheater leak event sequences is presented in Table 4-2. In all cases, the highest dose (rem) was found to be the 2-hr whole body dose at the EAB. For sequences with no release from the hot reheat bypass, inhalation doses were calculated to be negligible. In the other sequences, the inhalation thyroid dose was always at least an order of magnitude lower than the corresponding whole body dose. Thus, the whole body dose is concluded to be the important dose for this event.

Comparison of the calculated median doses in Table 4-2 with the probabilities of occurrence for each event sequence is made in Volume IV. It is shown that the highest risk sequence is Branch LA, in common with the preliminary scoping results in Volume III. Therefore, the detailed uncertainty analysis was performed for Branch LA.

A comparison of Branch LA dose consequences with regulatory limits and current PSAR calculations is shown in Fig. 4-1. Median doses are orders of magnitude lower, and upper 95% bounds on these doses, including the overall uncertainties derived as discussed below, are also well below the PSAR doses.

The uncertainties in the calculations were attributed to the following parameters: (1) activity on gasborne dust, (2) leak size, (3) gasborne fission product inventory, (4) decay and buildup enroute, and (5) meteorology. Table 4-3 presents median and upper 95% confidence bound values of these parameters. Corresponding uncertainty factors on the median doses are also presented in the table for Branch LA. It was found that meteorology uncertainty causes the highest uncertainty in the doses.

TABLE 4-2
CONSEQUENCES FOR VARIOUS EVENT SEQUENCES OF REHEATER TUBE LEAK

Branch	Median Release Rate	Time To Isolation (min)	Median Release (C1)		2-Hr EAB Doses	
			From Air Ejector Exhaust	From Bypass	Whole Body γ	Inhal. Thyroid
LA	1.5 lb/sec	0.092	2.75	0	1×10^{-5}	0
LB		5.5	125	40	1×10^{-3}	6×10^{-6}
LC		17.5	↓	400	2×10^{-3}	6×10^{-5}
LD		5.5		40	1×10^{-3}	6×10^{-6}
LE		17.5		400	2×10^{-3}	6×10^{-5}
LF		(a)		6300	3×10^{-2}	1×10^{-3}
LG		5.5		40	1×10^{-3}	6×10^{-6}
LH		17.5		400	2×10^{-3}	6×10^{-5}
LJ		5.5		40	1×10^{-3}	6×10^{-6}
LK		17.5		400	2×10^{-3}	6×10^{-5}
LL		(a)		6300	3×10^{-2}	1×10^{-3}
LM		5.5		40	1×10^{-3}	6×10^{-6}
LN		17.5		400	2×10^{-3}	6×10^{-5}
LO		(a)		6300	3×10^{-2}	1×10^{-3}
IA	2.7×10^{-2} lb/sec	2.6	1.6	0	5×10^{-6}	0
IB		5.5	3.3	↓	1×10^{-5}	↓
IC		180	108		4×10^{-4}	
ID		(b)	1350		4×10^{-3}	
IE		(b)	1350		4×10^{-3}	
IF		(a)	6760		2×10^{-2}	
IG		5.5	3.3		1×10^{-5}	
IH		180	108		4×10^{-4}	
IJ		(b)	1350		4×10^{-3}	
IK		(b)	1350		4×10^{-3}	
IL		(a)	6760		2×10^{-2}	
IM		(b)	1350		4×10^{-3}	
IN		(b)	1350		4×10^{-3}	
IO		(a)	1350		2×10^{-2}	
SG	4.7×10^{-4} lb/sec	30	0.18	↓	5×10^{-7}	↓
SH		210	1.3		4×10^{-6}	
SJ		210	1.3		4×10^{-6}	
SK		210	1.3		4×10^{-6}	
SL		(a)	6760		2×10^{-2}	
SM		(b)	1350		4×10^{-3}	
SN		(b)	1350		4×10^{-3}	
SO		(a)	6760		2×10^{-2}	

(a) Exceeds time necessary to release gasborne inventory.

(b) Isolation at reactor trip on low PCRV pressure (20% release of gasborne inventory).

Fig. 4-1. Summary of radiological dose consequences and uncertainties compared to regulatory guidelines of reheater tube leak event Branch LA

TABLE 4-3
SUMMARY OF MEDIAN DOSES AND PARAMETER UNCERTAINTY ANALYSIS FOR REHEATER
TUBE LEAK BRANCH LA

Parameter	Parameter Values		Uncertainty Factors Whole Body $\gamma + \beta$ Dose	
	Median	Upper 95% Bound	2-Hr EAB	30-Day LPZB
Activity on gasborne dust vs fixed surfaces	50%	95%	0.84	0.83
Leak rate	1.5 lb/sec	8.2 lb/sec	5.5	5.5
Gasborne inventory (noble gases)	7585 Ci	11,874 Ci	1.53	1.61
Decay and buildup enroute	Effect considered	Effect neglected	1.2	1.0
Meteorology (χ/Q) (a)	See Table 3-7		12	27
Total uncertainty factor			21	42
Median dose, rem			2.2×10^{-5}	1.4×10^{-6}

(a) Release height = 30 m.

Second highest dose uncertainty is attributed to variation in the leak size. Much lower is the uncertainty due to gasborne activity and, lastly, to the parameter activity on gasborne dust and decay and buildup enroute. Basically, the whole body dose was found to vary proportionally with each of the top three parameters so that the dose uncertainty factors reflect the uncertainty ranges in these parameter values. Therefore, narrowing of the key parameter value uncertainties causes a directly proportional narrowing of the dose uncertainties.

Figure 4-2 displays the radioisotope contributions of significance for Branch LA median doses. The largest contributor is Kr-88, since it is the highest constituent of the gasborne activity. Although this isotope, with its 2.8-hr half-life, decays significantly over a 30-day interval, its high effect on doses in the initial hours is sufficient to cause the greatest contribution among all noble gases to the 30-day LPZB dose.

4.2. CORE HEATUP

4.2.1. Description

In the preliminary AIPA analysis (Volume III), event sequence diagrams and risk logic trees were constructed for the loss of offsite power (LOSP) and earthquake (EQ) initiating events. For LOSP, a relatively high risk branch (BI, Volume III) was identified in which, after the initiating loss of off-site power, the probabilities of failure of the main turbine generator to remain on line and of added failure of CACS startup combine with relatively severe consequences to cause risk comparable to a reheater tube leak. The condition of interest resulting from this chain of events is a core heatup with the reactor tripped and the containment functioning as designed. A simplified diagram for this event sequence is presented in Fig. 4-3 (the reader should refer to Volume IV, LOSP Branch BD).

A condition of loss of offsite power can occur when current flow from the main transformer to the power grid is interrupted. In the standard HTGR plant there are at least two grid paths away from the plant site.

Fig. 4-2. Reheater tube leak Branch LA dose consequences

Fig. 4-3. Sequence of events for core heatup Branch BD of LOSP

Should the current flow be interrupted for any reason, the reactor is designed to be brought to 25% power, where it is held until offsite power is restored. During this period all power requirements are satisfied by the plant turbine generator.

If a turbine trip occurs in conjunction with LOSP, the plant protection system (PPS) will simultaneously initiate startup of the CACS. In this case, electrical power is supplied to the essential I-E buses by the diesel generators. If offsite power is restored, it is also possible to transfer to the 34.5-kV line from the offsite substation for power to the I-E buses.

It is only through multiple failures of the electrical supply systems and the CACS system that core heatup can occur. The fuel, moderator, reactor internals, and primary coolant begin to heat up gradually due to fission product decay heat in the absence of forced circulation. The core temperature rises at a maximum rate of approximately 10°F/min. It is still possible to return the plant to forced cooling providing that the system can be repaired soon enough to prevent irreparable damage within the PCR. It was postulated that the limiting primary system component for thermal transients (with a full helium inventory) is the thermal barrier cover plate in the upper plenum area, which could fail at temperatures exceeding 1500°F and potentially lead to partial blockage of primary coolant flow. In this analysis, restoration of core cooling following failure of the CACS was not considered after the temperature of the thermal barrier exceeds 1500°F. This time period, designated as the allowable duration of LOSP, is a function of the time at which CACS failure occurs, owing to changes in initial temperatures and decay heat generation rate with time.

As the core temperatures increase, intact fuel particle coatings begin to fail due to temperature and internal gas-pressure-induced stresses. Fission products within the fuel kernels then diffuse into the matrix bedding of the fuel rod. From the matrix bedding, the nuclides migrate or diffuse across the fuel/graphite interface through the graphite web to the

graphite coolant channel surface. Small convection currents carry the fission products, which are in the vapor phase, into cooler core and PCRV regions where condensation occurs. Gasborne isotopes that do not plate out are available for transport to the containment.

If the heatup condition cannot be corrected, then release of fission products to the containment building will occur through the relief valves. These valves are design features of the HTGR to prevent overpressurization of the PCRV. The time of relief after initiation of core heatup is established from the primary coolant temperature transient and varies with CACS operation time. For zero CACS operation time, the time to pressure relief was calculated to be 45 min.

If fission products reach the containment building as a result of relief valve activation, automatic isolation of the containment ventilation system and activation of the containment filtration system occur. These measures allow natural decay and containment cleanup to reduce activity levels of fission products that migrate to the containment from the primary coolant and the fuel.

Although studies performed under regulatory guidelines (Maximum Hypothetical Fission Product Release, MHFPR) have shown that release levels are low, there will still be some release of radioactive nuclides to the environment because of leakage from the containment building. Failure of the containment building to isolate will release more fission products and allow less chance for decay. In this event, the worst consequence considered in the analysis, the release of fission products to the environment, occurs through transport between the high-pressure helium atmosphere in the containment and the ambient air. This sequence is denoted as Branch BF of the LOSP event in Volume IV.

The consequence evaluation for the EQ initiating event is similar to that for core heatup, also with consideration for the probability of containment damage.

Owing in part to the risk of this event, the LOSP initiating event was chosen for study in the detailed Phase I AIPA analysis described in Volume IV. The described event branch, denoted in Volume IV as BD, core heatup branch, serves in this volume as the focal point or base event for the detailed evaluation of LOSP and EQ consequences, including median prediction and uncertainties on the prediction. Consequences and uncertainties for other LOSP and EQ branches are obtained by modifying the Branch BD results as described below and in Volume IV.

The core heatup analysis involves several high-temperature fission product release technologies; the event is therefore appropriate to use for determining the safety significance of these technologies. To this end, the core heatup is divided into three parts. The first part involves the evaluation of space-time temperature distribution of the core and PCRV internals during the event using the CORCON computer code (Ref. 4-1). The parameters of interest in this thermal analysis calculation include the afterheat, heat capacity, and effective core heat conductivity. The second part of the analysis consists of the SORS computer calculation (Ref. 4-2). Fission product transport rates from the fuel to the primary coolant system and subsequently to the containment building atmosphere are obtained as a function of the temperature distribution as calculated with CORCON. Some of the key parameters are release rate from the fuel, fuel failure characteristics, and fission product plateout on PCRV internals. The final part of the analysis consists of the dose evaluation as a function of fission product release rates to the containment. The key parameters include fission product fallout and plateout in the containment, containment leak rate, and meteorological conditions. A more complete description of the methods and assumptions used in the analysis can be found in Refs. 4-1 through 4-3.

4.2.2. Parameter Evaluation

There are numerous parameters that affect the consequences of a core heatup, such as core thermal parameters (CORCON code), core activity release parameters (SORS code), and containment activity release parameters (TDAC code). Table 4-4 presents a comprehensive list of parameters categorized according to the code in which they are used, the type of parameter, and the functional variation. Out of 17 total parameters, 10 are fission product parameters, 5 are design parameters, 2 are material parameters, and 1 is a meteorology parameter.

4.2.2.1. Core Thermal Parameters (CORCON). Experience with the CORCON computer code in analysis of core heatup has shown that three parameters are of importance to the temperature transient of the core materials. These parameters are discussed in the following paragraphs.

Afterheat. The heat generation in the core after the reactor is tripped is a fundamental fission product parameter. The core afterheat simulation in CORCON accounts for fission fragment decay, the effect of delayed neutrons, neutron capture, and the actinide nuclides. The basis for afterheat assessment is presented in Ref. 4-4. The afterheat calculation was done for 4-year operation at 80% load factor, neglecting any shutdown time. (This is equivalent to 3.2 years continuous operation at 100% load factor.) This level is higher than would exist in an equilibrium HTGR, because the average age of the fuel in the equilibrium reactor varies between approximately 1-1/2 and 2-1/2 years.

Using recommendations from Ref. 4-5 for decay heat calculations as a function of irradiation life, the median afterheat value was calculated as ~99% of that calculated from Ref. 4-4 for the core heatup times of interest. This formulation neglects the variation due to plant availability or shutdown times as discussed in Section 3.2. This parameter would only affect the short-lived isotopes, which are a small contribution to the decay heat and whose equilibrium inventories are built up quickly upon startup.

TABLE 4-4
SUMMARY OF CORE HEATUP PARAMETERS

Parameter	Type Of Parameter	Functional Dependence
Core Thermal Parameters (CORCON Code)		
Decay heat	Fission product	Time
Core effective thermal conductivity	Material	Temperature
Heat capacity	Material	Temperature
Core Activity Release Parameters (SORS Codes)		
Initial coolant activity inventory	Fission product	Steady-state operation
Initial fuel activity inventory	Fission product	Steady-state operation
Fuel particle coating failure	Fission product	Temperature
Fuel particle release rates	Fission product	Temperature, isotope
Diffusion through graphite	Fission product	Temperature, isotope
Vaporization from graphite	Fission product	Temperature, isotope
Plateout in PCRV	Fission product	Isotope
Containment Activity Release Parameters (TDAC Code)		
Release height	Design	--
Liner leak rate	Design	--
Plateout in containment	Fission product	Isotope
Decay and buildup of nuclides	Fission product	Time, isotope
Atmosphere dilution factor	Meteorology	Site, weather
Cleanup system recirculation rate	Design	--
Cleanup system startup time	Design	--
Cleanup system filter efficiency	Design	--

The variations for the afterheat were those recommended in Table 4 of Ref. 4-4. These variations represent 95% confidence bounds and were applied directly. Thus, the core heatup temperature transient was calculated using the CORCON code and applied to the fission product release SORS code calculations.

Core Thermal Conductivity. Experimental data on the thermal conductivity of HTGR core graphites have been reviewed and are presented in Ref. 4-6. Thermal analyses of HTGR core performance require precise values for the thermal conductivity of the moderator block graphite as a function of fast neutron fluence, irradiation temperature, and measurements. Measurement of the thermal conductivity of irradiated HTGR nuclear graphite has been performed and the results are shown in Fig. 4-4. The effects of irradiation are described in Ref. 4-6. The maximum uncertainty calculated ($\pm 35\%$ from Fig. 4-4) was applied uniformly about the base case value of graphite thermal conductivity.

Core Specific Heat. The large heat capacity of the core graphite results in a slowly rising core temperature transient, thus causing a significant delay in fuel failure and fission product release. The irradiated graphite and composite active core specific heat values shown in Fig. 4-5 are used as median conditions (Ref. 4-1). The sensitivity of dose is calculated by varying the specific heat over an uncertainty range of $\pm 10\%$.

4.2.2.2. Core Activity Release Parameters (SORS Code). Experience with the SORS computer code has shown that a number of fission product parameters are important to determine the activity release of radionuclides from the core during core heatup. These parameters and how they are specified or calculated in SORS are discussed in the following paragraphs.

Fuel Body Activity. The fuel body activities, circulating inventory, plateout levels, and the associated uncertainties used in the analyses have been calculated in Volume V of this report and are presented in Section 3.2 of this volume. In general, the 95% confidence limit and the median values

Fig. 4-4. Axial and radial graphite thermal conductivity as a function of temperature

Fig. 4-5. Irradiated graphite and composite active core heat capacity

for nuclide inventories are less than those for design calculations. Thus, a decrease in the dose consequences, from those predicted for the design calculation, is anticipated.

Fission Product Release from Fuel. The release behavior of each nuclide from HTGR fuel under transient conditions is expressed by a fractional release rate R based on experimentally derived curves, as a function of temperature, type of fuel (BISO or TRISO), and coating condition (intact or failed). The release rate is calculated assuming an exponential decay behavior, i.e.,

$$\frac{dN}{dt} = -RN \quad , \quad (4-1)$$

where N = fission product inventory,

R = release rate as a function of temperature, fuel type, fuel particle integrity, irradiation history, and nuclide, and

t = time.

R represents a fractional release rate and is best characterized from postirradiation annealing experiments over a range of time where R is constant. Because of the rapid temperature transient, the release rate characteristics are represented most accurately over short periods of time after the start of annealing experiments, where the release rate is larger than at longer annealing times.

A review of annealing experiments was conducted and a statistical analysis of General Atomic and HNL (previously ORNL) data was performed with the appraisal of factors of uncertainty, such as fuel sample characteristics and data treatment. Specifically, a set of release curves

(Figs. 4-6 and 4-7) has been derived from experimental data. Standard deviations and remarks for each curve are given in Table 4-5. These standard deviations were converted to a 95% confidence limit for fission product release rate in the SORS code.

Fuel Particle Coating Failure. One of the primary barriers to the release of fission products into the HTGR primary coolant system is the coating on each individual fuel particle. The release rate of a typical nuclide from a specific fuel particle is dependent on temperature, fuel type (BISO or TRISO coated particles), and particle coating integrity (intact or failed). The particle coating condition is characterized as a function of temperature, previous irradiation history, and fuel particle type.

TRISO and BISO fuel particle failure mechanisms include kernel migration, pressure vessel failure, and missing or defective coatings. In addition, TRISO particles may be subject to another failure phenomenon, that of SiC layer-fission product interactions. Failure models of these characteristics and failure model uncertainties have been evaluated for HTGR fuel from work described in Ref. 4-29.

For this analysis, the failure models were applied for average fuel burnup and fluence instead of the previously assumed maximum burnups and fluences for design calculations. The design fuel failure model used in MHFPR analysis has factors of 10 times the anticipated pressure vessel failure. The conservatism was also removed from the fuel failure characteristics.

The resultant input data for upper bound, median, and lower bound fuel failure models are given in Tables 4-6 and 4-7. A comparison of the median fuel failure model and confidence limits is shown graphically in Fig. 4-8 for 4-year-old BISO particles.

Fig. 4-6. Fission product release rate versus temperature for intact fuel particles during core heatup conditions

Fig. 4-7. Fission product release rate versus temperature for failed fuel particles during core heatup conditions

TABLE 4-5
SUMMARY OF FUEL PARTICLE RELEASE RATE DATA

Curve	Data Ref. No.	No. of Points	Standard Deviation Factor(a)	Comments
Iodine release from failed BISO particles	4-7 4-8 4-9	14 12	1.7	This data on mechanically cracked, high-burnup UO ₂ fuel particles was related to in-pile failed particle behavior by data of Ref. 4-9 for the difference in failure mechanism extrapolated beyond 1500°C.
Iodine release from failed TRISO particles	4-7	11	2.7	Mechanically cracked particles with low burnup treated as in iodine release from failed BISO. Uncertainties account for possible increase in release due to burnup extrapolated beyond 1400°C.
Iodine release from intact particles	4-10 4-11 4-12	7 3 2	4.7	The release rate was maximized by conservatively extrapolating the annealing time in heat experiments to 1 hr, from 1100°C to 1700°C.
Xenon release from failed particles	4-12	8	3.2	The effects of burnup and type of kernel were conservatively represented in this data evaluation. Data range from 1000°C to 2000°C annealing temperatures.
Xenon release from intact particles	4-10 4-13	4 6	4.7	Annealing temperature range from 1000°C to 1700°C with reasonable spread in data.
Krypton release from failed particles	4-12	34	3.2	Mean curve was also evaluated by comparison with xenon release with same results Refs. 4-2 and 4-4 through 4-9.

TABLE 4-5 (Continued)

Curve	Data Ref. No.	No. of Points	Standard Deviation Factor ^(a)	Comments
Krypton release from intact particles	4-14 through 4-20	14	4.7	Comparison of xenon and krypton release showed increase factor from xenon was 3.6. This was applied to krypton release from intact particles in range 1370°C to 2000°C.
Cesium release from failed particles	4-21	9	2.0	3 laser-drilled fuel particles yield maximum release and remaining 6 are from Peach Bottom element compacts and particles in range 1000°C to 1400°C.
Cesium release from intact BISO particles	4-22 4-23	3 8	3	Comparison with additional data (Ref. 4-24) shows good agreement in range 100°C to 2000°C.
Strontium release from intact BISO particles	4-10 4-22 4-23 4-25 4-26	4 3 3 1 4	~5.4	Range of observations for 1100°C to 2000°C annealing temperatures.
Strontium release from failed particles	4-27	1	5.4	Comparison of strontium release from failed and intact BISO particles shows release rate for failed is 8.2 times that for intact.
Strontium release from intact TRISO particles	4-28	6	10	Data from Ref. 4-28 shows no effect of burnup on release rate over temperature range 1400°C to 1800°C.
Cerium release from intact BISO particles	4-23		10	High burnup data for 1100°C to 2000°C annealing temperature range.

TABLE 4-5 (Continued)

Curve	Data Ref. No.	No. of Points	Standard Deviation Factor ^(a)	Comments
Cerium release from failed particles	4-27	1	10	The similarity between cerium and strontium release behavior was used to establish release curve for cerium as factor of 8.2 times that for intact BISO particle release.
Barium release from intact BISO particles	4-11 4-12 4-22 4-24 4-25	3 2 3 7 1	5.4	Annealing experiments in temperature range 1100°C to 2000°C and large range of burnup.
Barium release from failed particles	4-27	1	10	Barium failed particle release curve was established as 8.2 times that for BISO intact. Based on strontium and barium similarity in release behavior.
Barium release from intact TRISO particles	4-25	6	10	

(a) The standard deviation factor is the exponential function of the lognormal standard deviation (e^{σ}).

TABLE 4-6
TRISO FUEL PARTICLE FAILURE (%) AS A FUNCTION OF TEMPERATURE AND FUEL AGE

Age (Yr)	Temperature (°C)						
	1250	1400	1612	1628	1659	1714	2000
Upper Limit - 95% Confidence Level							
1	0.16	0.16	0.16	0.16	0.16	0.16	100
2	0.22	0.23	0.24	0.24	0.24	0.16	100
3	0.38	0.43	0.47	0.49	8.8	24	100
4	0.55	0.66	0.78	4.9	13	27	100
Mean - 50% Confidence Level							
1	0.06	0.06	0.06	0.06	0.06	0.06	100
2	0.090	0.092	0.093	0.094	0.10	16	100
3	0.16	0.19	0.21	0.22	8.5	23	100
4	0.26	0.32	0.39	4.5	12	27	100
Lower Limit - 5% Confidence Level							
1	0.025	0.025	0.025	0.025	0.025	0.025	100
2	0.034	0.034	0.035	0.035	0.035	16	100
3	0.05	0.055	0.059	0.06	8.4	23	100
4	0.068	0.077	0.07	4.2	12	26	100

TABLE 4-7
BISO FUEL PARTICLE FAILURE (%) AS A FUNCTION OF TEMPERATURE AND FUEL AGE

Age (Yr)	Temperature (°C)				
	1250	1800	2000	2164	2600
Upper Limit - 95% Confidence Level					
1	0.1	0.1	100		
2	0.2	0.2	100		
3	0.31	0.31	100		
4	0.42	0.42	100		
Mean - 50% Confidence Level					
1	0.1	0.1	55	100	
2	0.19	0.19	55	100	
3	0.28	0.28	55	100	
4	0.37	0.37	55	100	
Lower Limit - 5% Confidence Level					
1	0.097	0.097	10		100
2	0.18	0.18	10		100
3	0.26	0.26	10		100
4	0.34	0.34	10		100

Fig. 4-8. Comparison of median fuel failure model and confidence limits for 4-year-old BISO particles

Fission Product Transport Through Graphite and Release to Coolant. A diagram of the release mechanisms modeled in the core heatup analysis is shown in Fig. 4-9. Fission products introduced at the fuel hole surface must migrate across the graphite web to the coolant hole surface, where they vaporize or are released into the coolant channel. Transport of the volatile nuclides, noble gases, and halogens to the coolant channel is similar to diffusion through the helium coolant and is quite rapid as compared to the diffusion-vaporization transport of metallic fission products. Thus, the volatiles are assumed to be directly released upon escape from the fuel, and the metallic fission product transport is described by the classical Fickian model for diffusion and convective mass transfer for vaporization.

Transport Model. The diffusion rate equation for metallic fission product transport in the effective cylindrical geometry of the HTGR fuel element graphite is expressed as

$$j_D = \frac{2DL (C_a - C_b)}{\ln (b/a)} , \quad (4-2)$$

where j_D = diffusion flux, g/sec,

D = diffusion coefficient, cm^2/sec ,

L = cylindrical length, cm,

C = element concentration, g/cm^3 ,

b = outside radius, cm, and

a = inside radius, cm.

Fig. 4-9. Fission product release mechanisms from fuel particles to primary coolant

In the MHFPR for HTGR PSAR analysis, this diffusion rate of the element is compared with the calculated surface evaporation rate equation,

$$j_E = H W \frac{P_b - P_c}{P_S} \quad (4-3)$$

where j_E = evaporation or condensation rate, g/sec,

H = local mass transfer coefficient (dimensionless),

W = carrier gas flow rate, g/sec,

P_S = total system pressure, atm,

P_b = temperature- and concentration-dependent vapor pressure at the graphite surface, atm, and

P_c = element partial pressure in the coolant stream at the entrance to the core segment, atm.

For the MHFPR analysis, the smaller of these two values is conservatively used as the limiting release rate from the graphite coolant channel surfaces of the core. More realistically, the release rate is actually a combination of the effects of diffusion and evaporation. Thus, the above conservative simplifying assumption was replaced by the more realistic representation described as follows.

The transport may be treated by an electrical analogy as two resistances in series, as shown in Fig. 4-10. Following the notation in the figure, the solution for the fission product mass flux by analogy is

$$j = \frac{P_a - P_c}{R_D + R_E} \quad (4-4)$$

WHERE

J = FISSION PRODUCT
MASS FLUX
P = ELEMENT PARTIAL
PRESSURE
R = MASS TRANSFER
RESISTANCE

SUBSCRIPTS

a = FUEL GRAPHITE
SURFACE
b = GRAPHITE COOLANT
SURFACE
c = COOLANT CHANNEL
D = DIFFUSIONAL
E = EVAPORATIONAL

$$J = \frac{P_a - P_c}{R_D + R_E}$$

WHERE

i = ELECTRICAL CURRENT
E = ELECTRICAL POTENTIAL
R = ELECTRICAL RESISTANCE

$$i = \frac{E_a - E_c}{R_D + R_E}$$

Fig. 4-10. Fission product transport through graphite and release to coolant and electrical analogy

The terms R_D and R_E are calculated as

$$R_D = \frac{j_o}{P_a - P_b} \quad . \quad (4-5)$$

and

$$R_E = \frac{j_e}{P_b - P_c} \quad . \quad (4-6)$$

This model is used in this study to represent the fission product transport rate j from the core fuel graphite interface to the coolant channel.

Diffusion Coefficient. The dominant transport mechanism in the graphite web is diffusion in a concentration gradient as described by the classical Fickian model (Eq. 4-2). The diffusion coefficient in the HTGR graphite for core heatup conditions is assumed to be temperature-dependent in accordance with the equation

$$\log D = \log D_o - \frac{B_D * 10^3}{T(^{\circ}K)} \quad (4-7)$$

The coefficients D_o and B_D for this equation have been derived from a semiempirical analysis in Ref. 4-30.

Few experimental studies have been done at the fission product concentration levels and fast neutron fluence of interest for major core heatup transients. In Ref. 4-30, a conservative approach was formulated. The relation of diffusion coefficient to element heat of vaporization was established based on limited experimental data. This correlation was conservatively evaluated for high metal concentrations on all metallic fission product species. Also, any increase in graphite sorptivity under fast neutron irradiation is neglected. Thus, fission product transport will occur more slowly than is described by the diffusion coefficients recommended. Therefore, the recommended values of Ref. 4-30 have been assigned as upper limit values, and a median value of one-tenth of the recommended diffusion coefficients was established. These limits are based on data review and engineering judgment, and do not reflect statistical treatment. However, for this study the upper limit and mean values described above have been used as the upper 95% confidence limit and median value.

Vapor Pressure. The temperature- and concentration-dependent element vapor pressure at the graphite surface from Eq. 4-3 is derived from experimental measurements and discussed in Ref. 4-30. There are two basic uncertainties associated with the vapor pressure evaluation: (1) the vapor pressure data and (2) the monolayer concentration.

From review of the Ref. 4-30 vapor pressure data, the standard deviations have been developed as listed in Table 4-8 for the species of radiological monolayer. Evaluation of the monolayer concentration uncertainties is not as straightforward as evaluation of the vapor pressure data.

The effective surface area for sorption evaluation depends on uncertainties in the radius of adsorbing species, how well an adsorption parameter matches the substrate structure, and how well the surface area of the substrate itself is established. There is very little in the way of

TABLE 4-8
STANDARD DEVIATION^(a) FOR GASEOUS SPECIES ELEMENTAL VAPOR PRESSURE FROM
HTGR GRAPHITE SYSTEM

Species	$\sigma^{(a)}$ In Terms Of Log P	Comments
Rb	0.7	Not reference graphite (Ref. 4-31)
Cs	0.5	Curvature adds to uncertainty (Ref. 4-32)
Ba	0.5	Not reference graphite (Ref. 4-33)
Sr	0.3	(Ref. 4-34)
Sb ^(b)	0.3	(Ref. 4-35)
Ce	0.5	(Ref. 4-36)
Y	0.5	(Ref. 4-37)
Zr	0.5	(Ref. 4-37)
Pr	0.5	(Ref. 4-38)
La	0.5	(Ref. 4-39)
Pm	2.0	No experimental data (Ref. 4-39)

^(a) 95% uncertainty bound is defined as $10^{1.65\sigma}$.

^(b) Highest vapor pressure characteristics for species.

experimental evaluation of these quantities. In work on cesium, the isotherms achieved were characteristic for H-327 graphite. Thus, the questions of radius and substrate do not enter. There remains only the question of whether the surface area of a sample correctly models the surface area of the graphite utilized in the reactor. In fact, studies that have been made with other metals, e.g., Ba and Sr, indicate that the uncertainty in adsorption can be associated with uncertainty in surface area of the graphite. It is recommended that a standard deviation of ± 0.5 in the logarithm of the monolayer concentration or in the actual surface area be utilized to handle surface area uncertainties, surface growth or attenuation under irradiation, annealing behavior during the core heatup event, and actual monolayer concentration per surface area unit uncertainties. This factor will tend to cover the manufacturing uncertainties and the surface area attenuation of matrix graphite during irradiation.

PCRV Escape Fractions Simulating Fission Product Plateout. One of the predominant processes that governs fission product transport during core heatup conditions is fission product plateout or adsorption. Fission products remaining gasborne in the primary coolant gas as it exits from the core due to local convection currents will subsequently condense or plate out on the locations of lower temperature in the primary coolant circuit, such as the water-cooled PCRV liner and penetration opening. Fission product transport will be principally by free convection. The degree of adsorption that will occur has been determined experimentally. Twelve experiments were conducted to study the transport behavior of certain fission products in and from a small-scale HTGR mockup during simulated core heatup conditions (Ref. 4-40). Several types of core heatup experiments were performed. All data are summarized in Table 4-9.

The first type of experiment involved annealing of irradiated fuel particles in a graphite tube King furnace at temperatures up to 2500°C. Radioassay of the furnace internals and external traps was performed to

TABLE 4-9
COMPILATION OF TEST RESULTS OF PCRV ESCAPE FRACTIONS

Test Type	Test Designation	FSV FSAR Reference	Element	PCRV Escape Fraction
First type - furnace annealing tests of fuel particles	P1	Table D.3-3	Cs	0.14
			Sr	0.002
	P2		I	0.014
			Sr	4×10^{-5}
	P3		I	0.0012
			Cs	0.0017
	P4		I	0.024
			Te	0.01
			Cs	0.012
			La	$< 8 \times 10^{-6}$
			Eu	$< 2 \times 10^{-7}$
			Nd	$< 8 \times 10^{-7}$
			Ce	$< 9 \times 10^{-9}$
			Zr	$< 3 \times 10^{-7}$
	P5		I	0.02
			Te	0.035
			Cs	0.021
			Sr	4.9×10^{-5}
	P6		I	0.055
			Te	0.018
			Cs	0.057
			Ce	$< 1 \times 10^{-7}$
			Nd	$< 3 \times 10^{-7}$
			Eu	$< 1.2 \times 10^{-5}$
			Sr	$< 6.3 \times 10^{-5}$
Iodine tracer tests in modified furnace	I1	Table D.3-11	I	8.6×10^{-5}
	I2		I	3.6×10^{-4}
Iodine tracer tests in upper plenum mockup	I3	Table D.3-12	I	5.5×10^{-4}
	I4		I	7.7×10^{-4}
Sr tracer tests in King furnace	Sr 1	Section D.3.1.3	Sr	3.4×10^{-4}
	Sr 2		Sr	5.3×10^{-5}

measure the fission product distributions. The furnace dome and water-cooled electrodes simulated the reactor vessel and PCRV metal liner. Helium purge gas flow was used to conservatively represent the transport of fission products from the PCRV to the containment. The majority of the experiments were performed at these conditions; however, a more accurate representation of core heatup conditions was obtained in the second type of experiment, in which a modified King furnace apparatus was used. Accurate determination of fission product transport was obtained in these experiments by use of tracer elements. Two separate tests (I1 and I2) used a 1/12th-scale model of the reactor PCRV and steel liners on the radiation shields, which more closely approximated the HTGR conditions. A decreased flow rate and a similitude of the other diffusional characteristics (concentration) were also applied. These experiments were performed only for iodine. Additional tracer experiments on strontium were performed using the modified King furnace.

In the HTGR licensing analysis (MHFPR), the most significant element has been considered to be iodine. Thus, the best information has been accumulated for iodine. This is the case for the plateout observations. The iodine tracer experiment I2 most closely simulates the diffusional mass transport phenomenon anticipated for the event. The simulation of the diffusional transport mechanisms is considered the limiting factor and, therefore, the most important modeling parameter for plateout. Thus, the result of this experiment was used as a median value for iodine release fraction.

The strontium and cesium plateout measurements did not accurately model the fission product transport system as well as the I series experiments. A comparison of the P series and I series tests shows a factor of approximately 40 difference in the iodine plateout fractions. Since the sorptive capacity of carbon steel (PCRV liner material) is much greater for strontium and cesium than for iodine, and since the diffusion coefficient in helium (Ref. 4-41) is approximately the same for all fission

products considered, it is reasonable to assume that strontium and cesium would also have a plateout fraction 1/40th that for the P series tests as observed in the I series. This assumption was also drawn for the remaining sorptive elements, based on the fact that the diffusional process is controlling the plateout behavior and is about the same for all elements at these conditions.

The total uncertainty on iodine and strontium plateout was derived considering the data scatter and uncertainty in the applicability of the experiment to the core heatup conditions. For both elements, the maximum data scatter is approximately a factor of 10 about the median using statistical methods as described in Ref. 4-42 for small amounts of data. For iodine, an uncertainty factor of 50 was estimated to account for the experiment applicability. This factor is intended to account for lack of corroborating data and differences, if any, in geometry (scaling effects), flow rate, surface temperature, saturation effects, etc. For strontium, the above uncertainty factor for experiment applicability was increased by the multiplication factor of 40 used to normalize the raw strontium data to the iodine I series data. Thus, the strontium experiment applicability is assumed to be uncertain by a factor of 2000.

Calculated total uncertainties for iodine and strontium were obtained by combining the data scatter and experiment applicability uncertainty factors, U_O and U_α , respectively, in a manner consistent with the lognormal law of propagation of errors,

$$U = \exp [(\ln U_O)^2 + (\ln U_\alpha)^2]^{1/2} .$$

Table 4-10 presents the calculated plateout uncertainties using this equation. For iodine, the uncertainty is about a factor of 100, while the corresponding uncertainty for strontium is 3000. This reflects the assessed value of the iodine tracer tests (I series) as more indicative of core heatup conditions compared to either the P series or Sr series tests.

TABLE 4-10
MEDIAN PCRV ESCAPE FRACTIONS AND UNCERTAINTIES FOR CORE HEATUP

Species Number Used In SORS	Fission Product Elements (Characteristic Element Underlined)	Median PCRV Escape Fractions	Total Uncertainty Factor Including Data Deviations
1	<u>Sr</u>	1.9×10^{-6}	3000
2	<u>Cs</u> , Rb	3.1×10^{-4}	3000 ^(a)
3	Ba, <u>Eu</u> , Sm	3.0×10^{-7}	3000 ^(a)
4	<u>Ce</u>	2.5×10^{-9}	3000 ^(a)
5	Xe	1.0	1
6	Kr	1.0	1
7	<u>Zr</u> , Tc, Mo, Nb	2.5×10^{-9}	3000 ^(a)
8	<u>La</u> , Y, Nd, Pr, Pd, Pm, Sn	2.0×10^{-7}	3000 ^(a)
9	Ru, Rh	2.0×10^{-7}	3000 ^(a)
10	<u>I</u> , Te, Sb, Se	3.6×10^{-4}	100

(a) Uncertainty assumed to be the same as for Sr.

For cesium, the data scatter was observed to be about the same as for iodine or strontium. The uncertainty factor for experiment applicability was taken to be the same as for strontium, notwithstanding the fact that cesium was observed in P series tests to plate out similar to iodine rather than like strontium. The reason for using the strontium uncertainty factor is that, like strontium, no data were obtained for cesium at the I series test conditions.

Plateout data for iodine, cesium, and strontium for all tests are shown graphically in Fig. 4-11. All P series and Sr series data are normalized in this figure by the 1/40 factor to apply to I series test conditions. Uncertainty bands about the median values are shown for the elements. The extremely wide, and likely conservative, uncertainty for cesium is evident, causing the upper limit nonplateout fraction to approach unity.

Median values for elements other than the above three were calculated from the data in Table 4-9 where available. For elements having no data, median values were assumed from thermodynamically similar elements for which data existed. Table 4-10 lists the grouping of thermodynamically similar elements and the median values. Uncertainties for all groups except noble gases and the iodine groups were assumed to be the same as for strontium and cesium, as shown in Table 4-10.

4.2.2.3. Containment Activity Release Parameters. The major dose parameters and uncertainties are given in Section 3.5. Since there are generally few or no data on dose parameters not associated with containment design features, large uncertainties were used on these parameters, e.g., containment fallout and plateout, and meteorology factor. The uncertainties on these engineered safety features were generally based on judgment and available data.

Fig. 4-11. Summary of King furnace plateout data evaluation for important fission product elements

4.2.3. Consequence Analysis

The objective of support input for Volume IV AIPA probabilistic risk assessment requires computation of consequences for various event sequences. Eight different sequences were identified for the LOSP and EQ initiating events. The sequences are of interest because of their relatively high risk (probability-consequence product) as described in Volume III. The analysis for Branch BD, the relatively highest risk sequence, was used as the basis for generating the parameteric results. Other sequences considered variations in the CACS operation time. The assumptions of CACS operation time are based on the probabilistic discussions of Volume IV, Section A.2.6. Using these assumptions and the pressure response to core heatup from RECA results (Ref. 4-44), calculations were made of PCRV valve relief times. The effect was calculated of fission products decaying and plating out in the PCRV cavity until this pressure relief time. Several sequences consider variation in the containment cleanup system operation. The final variation was containment building isolation or immediate release to the environment. The probabilistic choices involved in these specifications for analysis are contained in Volume IV, Section A.2.9.

The analysis can be broken down into three parts: (1) core, PCRV, and primary coolant heatup thermal analysis, (2) fission product release calculations, and (3) dose evaluation. Each of these is described in the next paragraphs for LOSP Branch BD and other LOSP and EQ sequences.

4.2.3.1. Thermal Analysis. The thermal analysis results are exemplified by the active core average temperature transient for Branch BD, as shown in Fig. 4-12. These median temperatures are slightly lower than would be calculated using the MHFPR licensing methods (Ref. 4-1) due to a more realistic choice of parameter values. Fuel failure due to the heatup transient occurs in the temperature range 1600° to 2000°C (2900° to 3630°F). Thus, on a core average basis, the times for fuel coating failure range from

Fig. 4-12. Core temperature history from CORCON code analysis for 3000-MW(t) HTGR

about 7 to 14 hr. Some fuel failure occurs locally at both earlier and later times. The earliest time for the hottest region to experience fuel failure is about 2 hr.

When the core heatup event occurs, the pressure and temperature of the primary coolant (helium) increase until the PCRV relief valve opens. The nominal set point for the rupture of the disk and subsequent opening of the relief valve for the initiation of the PCRV blowdown is 762 psig. The relief valve, which has a net flow area of 20 sq in., is connected to the inlet plenum of the core.

The relief valve opens about 45 min after event initiation. The inlet plenum temperature at this time has been calculated as $\sim 1200^{\circ}\text{F}$. This temperature is above the usual maximum service temperature of 850°F for the valve-body pressure parts, bolts, and nuts (Ref. 4-45). Also, the mechanical properties of these ferritic materials are subject to a change at the 1200°F temperature. Therefore, the pressure relief valve cannot be relied upon to reseal at the predicted conditions. The relief time is calculated based on the PCRV temperature and convection transients, and the valve is postulated to remain open through the event sequence from that time.

Since the helium is released from the top of the PCRV above the top plenum, the probability of limited mixing in the containment is higher than might be expected for cases where the PCRV experiences a blowdown to the containment from the hot leg. Because of the partial mixing effect as compared to the assumed complete mixing of the DBDA, the containment liner in the dome area may be subjected to potentially higher gas temperatures than those anticipated for the DBDA. The containment liner in the dome (rather than the containment wall) is considered not only because it is the first surface to come in contact with the potential hot column of helium released from the PCRV, but also because it is the limiting surface for containment integrity since the concrete thickness at the dome is thinner than at the containment wall.

As input for determining the probability of containment non-isolation or failure for the event sequences, the peak values of containment pressure and liner temperature are given in Table 4-11 for the core heatup event. These values do not exceed typical containment design values; thus the integrity of the containment during a core heatup event is highly probable.

TABLE 4-11
CONTAINMENT PRESSURE AND LINER TEMPERATURE PEAK VALUES
DURING CORE HEATUP

Parameter	Low He Mixing	High He Mixing
Pressure (psia)	47	48
Liner temperature (°F)	321	190

For various sequences, the containment building was postulated to be nonisolated. The containment flow transient is assumed equal to the PCRV flow transient for calculations involving the "open" containment building where isolation valve failure is presumed to occur. Figure 5-3 of Volume II shows the flow modeling equations for the open containment flow rates. As mentioned previously, the 30-day time period required to obtain most of the dose contribution from core heatup also accounts for release of most of the containment inventory of radioactive gases. The leak rate is given in Table 3-1 for a nonisolated containment.

4.2.3.2. Fission Product Release. The fission product release and retention mechanisms used in the analysis of the core heatup event can be summarized as follows:

1. There is a small effect early in core heatup of release from intact fuel particles (Fig. 4-6) and from fission products sorbed in graphite.
2. Initial gasborne inventory and fission products added to primary coolant by mechanism 1 decay within the PCRV to lower levels before the PCRV relief valve opens.
3. Fuel particle coatings fail in a time-dependent manner (over many hours) due to internal gas stresses or chemical reactions (Fig. 4-8) induced by the time-dependent temperature transient.
4. There is a small instantaneous release of gaseous fission products from the buffer on coating failure; thereafter the fission product inventory of the kernel is released to the graphite in a time (temperature) dependent manner (Fig. 4-7).
5. Metallic fission products (Cs, Sr, etc.) diffuse through the graphite and are vaporized at the coolant channel (Fig. 4-10). No holdup in graphite is assumed for noble gases, iodines, telluriums, and seleniums as indicated by existing data.
6. Fission product vapors are transported relatively rapidly (assumed instantaneously) into upper and lower plenums due to small convection flow in the coolant channels.
7. Iodines and metallic fission products condense in cooler portions of the coolant channel and later become reentrained with increasing temperatures.

8. Fission products listed in Table 4-10 plate out on the thermal barrier, PCRV liner, and other metallic surfaces of the PCRV structure and relief valve train according to empirical plateout data (Table 4-10).
9. Plated-out fission products are assumed to remain fixed on the PCRV surfaces and are not available for reentrainment. No formation of volatile oxides, such as RuO_3 , is accounted for since during heatup the gas egresses from the PCRV (rather than air ingressing) and the primary coolant is devoid of oxidants.
10. The simplifying assumption was made that fission products not plated out are released instantaneously to the containment.
11. Plateout and fallout of fission products other than noble gases in the containment are accounted for.
12. Removal of fission products other than noble gases by the containment cleanup system is considered in certain branches.
13. Fission products decay or build up to different levels in the containment or en route to the site boundaries.

Mechanisms 9 and 10 are two phenomena about which relatively little is known. It is believed that errors introduced by this simplified treatment will tend to cancel for the two effects. However, to compensate for some amount of indeterminate uncertainties, the uncertainties on plateout fractions in Table 4-10 were maximized. Further analysis in this area is needed for more sophisticated treatment.

All of the above mechanisms are consistent with the SORS and TDAC computer programs used in the analysis. More detailed descriptions of the transport of fission products to the containment are presented in the SORS code documentation (Ref. 4-2).

For the above release mechanisms, the time-dependent partition of the fission product inventories in the various barriers (fuel kernels, graphite, plateout in PCRV, containment) and the fraction released to the environment were calculated as illustrated in Figs. 4-13 through 4-15. These figures trace the deposition history of several key isotopes for LOSP Branch BD. They dramatically show the slow time-dependent fission product release which is characteristic of the HTGR as opposed, for example, to that for LWRs. Figure 4-13 presents the results for the key noble gas, Xe-133, whose inventory in the fuel becomes depleted primarily in the time frame of about 10 to 40 hr. This release is immediately transferred to the containment in the calculational method. In fact, the Xe-133 inventory in the containment builds up somewhat at early times because of I-133 and Xe-133m precursors. The effect of the isolated containment on holdup of Xe-133 release to the environment can be roughly inferred by comparing the curves for cumulative release to the containment and cumulative release to the environment. This comparison shows a difference of over two orders of magnitude. More detailed calculations indicate that no containment holdup would increase the 30-day LPZB whole body dose by about a factor of 400 while a nonisolated containment causes an increase of about a factor of 100. These factors are used in the Volume IV risk assessment.

Corresponding results for I-131 are shown in Fig. 4-14. Here, the release from the fuel is slightly slower than for xenon because of release rate characteristics (Fig. 4-7). Nearly all iodine plates out in the PCRV in this median case calculation since the cumulative release to the containment corresponds to the median nonplateout fraction in Fig. 4-11. The small fraction (4×10^{-4}) of I-131 released to the containment decreases relatively quickly after 10 hr due mainly to the cleanup system (radioactive decay has a smaller effect). As a consequence, the containment

Fig. 4-13. Time-dependent partition of Xe-133 inventory in various HTGR fission product barriers for LOSP Branch BD

Fig. 4-14. Time-dependent partition of I-131 inventory in various HTGR fission product barriers for LOSP Branch BD

Fig. 4-15. Time-dependent partition of Sr-90 inventory in various HTGR fission product barriers for LOSP Branch BD

with operating cleanup system has a much larger mitigating effect on release to the environment for iodine compared to noble gases, as shown in Fig. 4-14.

Results are shown in Fig. 4-15 for a metallic fission product, represented by Sr-90. Strontium is released from the fuel primarily in the period of 5 to 20 hr. Holdup in the graphite, however, prevents its release to the coolant (thus to containment or plateout surfaces) until about 20 to 60 hr. Consistent with Fig. 4-11, the median PCRV nonplateout fraction is small ($\sim 2 \times 10^{-6}$). The effect of the containment barrier on Sr-90 release to the environment is similar to that for iodine. The net effect of the PCRV plateout and the containment barrier is to allow very little strontium release to the environment. Other metallic fission products are released in different proportions primarily according to their plateout fractions given in Table 4-10.

The fission product release in Branch BD may be characterized for groups of isotopes for purposes of comparing the time-dependent release from the PCRV with corresponding results obtained in the MHFPR licensing analysis for contributing isotopes to the whole body, thyroid, and bone doses. This comparison is depicted in Fig. 4-16 and pertains to release from the start of the event (loss of circulation) and not from the time of blowdown to the containment. For the core heatup, the blowdown occurs at 45 min compared to time zero for the MHFPR. However, the differences between the two cases are due mainly to differences in fuel release rate and plateout and are not due to holdup in the PCRV prior to 45 min for the core heatup case. The effect of this latter delay is estimated to be a second-order effect. For example, considering decay of noble gases, the inventory reduction for PCRV release at 1.25 hr instead of 0.75 hr (the nominal time of relief valve opening) is less than 10%. This reduction comes at the expense of the very short-lived isotopes, which generally do not contribute to the doses, especially the 30-day LPZB dose.

Fig. 4-16. Comparison of MHFPR to AIPA core heatup time-dependent fission product release for major dose contributors

4.2.3.3. Dose Consequences. The results of the AIPA consequence analysis are presented in Table 4-12 for the representative event sequences. The first entry in the table is the AIPA highest risk LOSP branch and is also the consequence branch for the fission product technology branch core heatup analysis. This branch assumes no period of core cooling after reactor shutdown. The proper operation of containment building isolation valves and containment building cleanup systems is assumed. Successive entries in the table assume failure of the containment cleanup (entry 2) and failure of both containment cleanup and isolation of the containment (entry 3). The latter is a branch in the 1.5 magnitude EQ analysis, where the probability of an open path for fission products to leave the containment is relatively high. Successive entries describe further variations of the LOSP and EQ consequence and account for longer periods of prior shutdown operation with the core auxiliary cooling system (CACS time).

4.2.4. Uncertainty Analysis (Branch BD)

The influence of each parameter on the core heatup radiological consequence was determined by varying each parameter singly to the upper 95% confidence value in Branch BD. The resulting uncertainty factors are presented in Table 4-13 along with the corresponding calculated median 2-hr EAB and 30-day LPAB doses. The uncertainties of the containment and meteorological factors are generally the overriding uncertainties for the whole body doses, while plateout dominates thyroid and bone dose uncertainties. Variations of the weather and containment parameters are performed mainly to evaluate the total uncertainty. Table 4-14 presents influence coefficients, the measure of dose sensitivity to parameter value without regard for the parameter uncertainty range, for the weather and containment parameters. For the most important of these parameters, the dose is found to vary proportionally with the parameter variation.

TABLE 4-12
SUMMARY OF CONSEQUENCES FOR VARIOUS LOSP AND EARTHQUAKE SEQUENCES

Event Sequence Identification (Vol. 4 Notation)	CACS Operation Time Prior to LOFC (hr)	PCRV Pressure Relief Time (hr)	Containment Building		Consequences (rem) 30-Day LPZ Doses			Comments
			Cleanup	Isolation	Whole Body γ	Inhalation Thyroid	Inhalation Bone	
LOSP BD	0	0.75	Yes	Yes	0.11	2×10^{-3}	9×10^{-6}	Base event analyzed. Consequences of other sequences were scaled from these results.
LOSP BE	0	0.75	No	Yes	0.11	0.01	5×10^{-5}	Varied from base event by taking no credit for containment cleanup.
LOSP AZ	5	8	Yes	Yes	0.09	1×10^{-3}	8×10^{-6}	Varied from base event by taking credit for 5 hr CACS operation before LOFC.
1.5 EQ S or LOSP BF	0	0.75	No	No	11	5	0.03	Varied from LOSP BE by taking no credit for containment isolation.
1.5 EQ P	5	8	No	No	5	3	0.02	Varied from LOSP AZ by taking no credit for containment isolation or cleanup.
1.1 EQ O	5	8	No	Yes	0.1	8×10^{-3}	4×10^{-5}	Varied from LOSP AZ by taking no credit for containment cleanup.
1.1 EQ K	50	12	Yes	Yes	0.07	1×10^{-3}	7×10^{-6}	Varied from base event by taking credit for 50 hr CACS operation before LOFC
1.5 EQ M	50	12	No	No	2	1	8×10^{-3}	Varied from 1.1 EQ K by taking no credit for containment isolation or cleanup

TABLE 4-13
SUMMARY OF PARAMETER UNCERTAINTY FACTORS FOR CORE HEATUP (BRANCH BD)

	2-Hr 500-m EAB Doses			30-Day 2500-m IPZ Doses		
	Whole Body $\gamma + \beta$	Inhalation Thyroid	Inhalation Bone	Whole Body $\gamma + \beta$	Inhalation Thyroid	Inhalation Bone
Median dose (rem)	6.5×10^{-3}	9.3×10^{-4}	6.5×10^{-6}	2.2×10^{-1}	1.6×10^{-3}	8.8×10^{-6}
<u>Fission Product Parameters</u>						
Fuel body activity	1.01	1.01	1.01	1.01	1.01	1.01
Fuel release rate	1.16	3.56	3.58	1.30	1.43	1.41
Decay heat	1.8	2.13	2.10	1.19	1.18	1.48
PCRV plateout fraction	1.39	96	294	1.4	96	632
Graphite transport diffusion coeff.	1.00	1.0	29.6	1.0	1.0	1.14
Fuel failure model	1.37	1.41	1.40	1.03	1.03	1.03
Containment fallout and plateout	1.00	10.0	10.0	1.0	10.0	10.0
Decay and buildup enroute	1.00	1.01	1.00	1.00	1.02	1.01
Gasborne activity	1.00	1.00	1.00	1.00	1.00	1.00
Vaporization from graphite	1.00	1.00	1.00	1.00	1.00	1.00
<u>Material Parameters</u>						
Core specific heat	1.08	1.08	1.08	1.08	1.08	1.08
Core graphite heat transfer coeff.	1.09	1.09	1.09	1.09	1.09	1.09
<u>Building and Meteorological Parameters</u>						
Containment leak rate	5.00	5.00	5.00	5.00	5.00	5.00
Cleanup system filter efficiency	1.00	1.01	1.00	1.00	1.00	1.00
Cleanup system recirculation rate	1.04	1.13	1.13	1.10	1.83	1.90
Meteorology atmospheric dispersion factor	14.0	14.0	14.0	16.0	16.0	16.0
Release height	0.195	0.195	0.195	0.96	0.96	0.96
$UF = \exp \left[\sum_P (\ln f)^2 \right]^{1/2}$						
Overall uncertainty factor	36	583	2924	25	436	2015

TABLE 4-14
INFLUENCE COEFFICIENTS FOR BUILDING AND METEOROLOGICAL PARAMETERS
IN GROUND RELEASE EVENTS

Rank	Parameter	Parameter Ratio $P_{\max}/\bar{P}$	Dose Ratio $D_{\max}/\bar{D}$	Influence ^(a) Coefficient
30-Day LPZB Inhalation Thyroid Dose				
1	Atmospheric dispersion factor χ/Q	16	16	1.0
2	Containment leak rate	5	5	1.0
3	Cleanup recirculation rate	2	1.83	-0.83
4	Cleanup filter efficiency	0.95	1.04	-0.83
5	Release height	70	0.96	-6×10^{-4}
30-Day Whole Body $\gamma + \beta$ LPZB Dose				
1	Atmospheric dispersion factor χ/Q	16	16	1.0
2	Containment leak rate	5	5	1.0
3	Cleanup recirculation rate	2	1.10	0.10
4	Release height	70	0.96	-6×10^{-4}
5	Cleanup filter efficiency	N/A	N/A	0

(a) Influence coefficient $\approx \left(\frac{D_{\max}}{\bar{D}} - 1 \right) / \left(\frac{P_{\max}}{\bar{P}} - 1 \right)$.

For the fission product and material property parameters, the dose is found generally to vary less than proportionally with the parameter variation. Figure 4-17 presents the sensitivity curves on a log-log plot showing the relative dose variation as a function of relative parameter variation. The dose variation is most sensitive to plateout variation for inhalation thyroid and bone doses, while the whole body dose is relatively insensitive to fission product parameters. The most significant dose is the 30-day whole body dose at the LPZB, which is 0.2 rem for beta + gamma median dose with an uncertainty factor of 25. This is well below regulatory limits, as shown graphically in Fig. 4-18. The median is also below the corresponding PSAR calculated value, while the upper 95% confidence bound slightly exceeds the PSAR value. On the other hand, inhalation thyroid and bone doses are below PSAR values even when adding uncertainty bounds as shown in Fig. 4-18.

The calculated doses are also well below estimated limits for acute health effects. Using the EPA recommended somatic health effects per man-rem, and converting site boundary doses to man-rem, the total estimated median exposure risk for all doses summed over all critical organs is one health effect per ten events with a 95% confidence uncertainty factor of 25. Details of these calculations are presented in Section 6.

The individual isotope contribution to the median dose indicates that Xe-133 and I-131 are the greatest contributors to the whole body and inhalation thyroid dose, respectively (Figs. 4-19 and 4-20). The major contributors to the inhalation bone dose are Te-132, Sb-125, and Te-129m. Strontium-90 is not a dominant contributor to the bone dose, because its PCRV escape fraction is orders of magnitude less than its competitor. The major contributor to all doses for this calculation is Xe-133, since the inhalation doses are low compared to the whole body penetrating gamma. Although the inhalation dose isotope contributions are small for the median dose, their importance cannot be neglected, because of the large uncertainties associated with these isotope parameters.

Fig. 4-17. Dose sensitivity to parameter variation for important fission product variables (Branch BD)

Fig. 4-18. Summary of radiological dose consequences and uncertainties compared to licensing guidelines for core heatup event

Fig. 4-19. Major isotope contributors to core heatup Branch BD median 2-hr EAB dose

Fig. 4-20. Major isotope contributors to core heatup Branch BD median 30-day LPZB dose

Because of the large differences in plateout uncertainties for different elements, the contribution of isotopes to the thyroid and bone inhalation doses would change when other than median nonplateout fractions are used. If upper 95% limits of nonplateout fractions were used for all elements simultaneously, the contributions to the 30-day LPZB inhalation bone dose would have been estimated as follows:

Cs-137	42%
Cs-134	27%
Sr-90	10%
Te-132	6%
Sb-125	4%
Other	11%

Cesiums are also estimated to be highest for the corresponding lung dose. It is concluded that cesiums should also be counted among the bone and lung dose isotope contributors most requisite of safety research and development.

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5. SELECTED RELATIVELY LOW RISK EVENTS

5.1. STEAM INGRESS EVENT

5.1.1. Description

The analysis in this section is concerned with an updating (from Volume III) of the consequences and risk assessment resulting from the initiating event of steam generator tube rupture in the economizer-evaporator-superheater (EES) section of the HTGR. Secondary coolant pressure at this location is between 2500 and 3000 psia at full power compared to the helium pressure of 725 psia. Thus, a tube rupture results in the accident condition of water/steam ingress into the PCRVR until the secondary coolant pressure equalizes with the primary system. The PPS safety systems for moisture detection and isolation and dumping of the leaking steam generator were accounted for (their mitigating effects and failure probabilities). The analysis of this section updates in detail the consequences of the most important event sequence branches, and scales these results to other branches as has been done in Volume III. In addition, several improvements are made in the event occurrence probabilities to provide a more meaningful risk assessment for the design option study in Volume III. One improvement is a more realistic probability for moisture monitor failure to identify and isolate the leaking steam generator. A second is consideration of the probability for flammability of CO and H₂ in the containment as these gases are generated in the PCRVR and egressed during pressure relief, if any.

Steam ingress sequences delineated in the preliminary AIPA analysis (Volume III) showed the relatively highest risk sequence to be Branch AC. The Branch AC event sequence consists of a single steam generator feedwater

tube rupture coupled with an incorrect identification of the defective loop by the moisture monitor system, causing a dumping of a nonleaking steam generator. Postulated lack of manual correction leads to failure to dump the leaking steam generator, increasing the duration of leak and quantity of steam inleakage. The resultant mass addition to the coolant causes the safety relief valve to lift briefly at 510 to 545 sec after event initiation due to excess coolant pressure. This venting leads to egress of PCRv gases to the containment and consequent small environmental doses. The Branch AC event sequence includes nearly all of the fission product transport and retention mechanisms experienced in steam ingress analyses, including fuel hydrolysis and steam-graphite reactions. Consequently, further investigation of the safety significance of parameter uncertainties was directed toward Branch AC of the preliminary study. Using this and the analysis for Branches AE and AG as described below, the entire steam ingress event tree was updated as described in Section 5.1.4.

Probabilities for burning of CO and H₂ egressed to the containment building as well as the resulting consequences thereof have been evaluated. Table 5-1 lists those event sequence branches from the preliminary analysis for which burning is of concern. For burning to occur, there must be sufficient production of flammable gases in the PCRv and a subsequent venting of these gases to the containment building. Consequently, only those branches are listed in Table 5-1 for which the leaking steam generator is not isolated and dumped, and either a penetration failure occurs or the relief line fails to close. Branches AK, P, AM, and L result in less chance of burning than either Branch AE or AG, because they result in less steam ingress to the PCRv. Consequently, only Branches AE and AG were considered with regard to burning. It is notable that Branches AC, AE, and AG encompass all of the fission product release technologies employed in steam ingress accident analyses.

TABLE 5-1
PRELIMINARY RESULTS OF STEAM INGRESS BRANCHES
WITH POTENTIAL BURNING (VOLUME III)

Branch	Sequence	Risk ^(a) (rem/yr)	Probability ^(a) (yr) ⁻¹
AE	Wrong loop dump Main loop run down Relief line remains open	4.4×10^{-12}	2.2×10^{-9}
AK	Wrong loop dump Immediate loss of main loops Relief line remains open	4×10^{-13}	$2 \times 10^{-12}^{(b)}$
AG	Wrong loop dump Main loop run down Penetration failure	5.2×10^{-16}	$2.6 \times 10^{-13}^{(b)}$
P	No isolation, no dump Immediate loss of main loops Relief line remains open	1.2×10^{-18}	$1.3 \times 10^{-15}^{(b)}$
AM	Wrong loop dump Immediate loss of main loops Penetration failure	4.2×10^{-17}	$2.1 \times 10^{-16}^{(b)}$

(a) Probabilities of flammability not included in preliminary results.
Risk = dose consequence x probability.

(b) For the purposes of this volume the notation of ϵ to indicate $<10^{-10}$ as described in Volume IV of this report was not used.

The method used to determine the accident conditions and the fission product release source terms is based on the OXIDE-3 computer code. The OXIDE-3 code was developed to analyze potential steam or air inleakage accidents for HTGR cores (Ref. 5-1). This code is capable of investigating the phenomena of: (1) possible release of fission products to the primary system due to chemical reactions of steam and air with core graphite and fuel, (2) possible PCRV pressure relief through a safety relief valve, or a postulated penetration failure, to the containment building and attendant activity leakage from the containment, and (3) resulting thermodynamic behavior in the containment due to hot gases that may be exhausted from the PCRV, including potentially flammable reaction product gases CO and H₂.

To execute OXIDE-3, transient coolant flow rates, coolant inlet temperatures, and steam leak rates into the PCRV are required input. The computer codes RECA (Ref. 5-2) and TAP (Ref. 5-3) were used in this analysis to determine primary coolant forcing functions. The computer code FLASH (Ref. 5-4) was used to determine steam inleakage rates into the PCRV.

5.1.2. Parameter Evaluation

Fission product transport during steam ingress events is governed by three reactions:

A. Steam reaction with graphite,

B. Steam hydrolysis reaction of failed uranium carbide fuel,

+ hydrogen + hydrocarbons

where X and Y are variable real and non-negative numbers.

C. Oxygen reaction with graphite,

The parameters associated with each reaction are shown in Table 5-2. Also listed are the parameters that describe the plant's fission product status prior to the accident, and the flammability limits for burning. The median value and 95% confidence bounds for each parameter are discussed below, using the designations shown in the table.

Steam-Graphite Reaction (Reaction A). The steam-graphite reaction consumes a portion of the inleaking steam and produces the flammable gases hydrogen and carbon monoxide. For a given steam leak, the more steam that reacts with graphite, the less steam there will be that is available to react with fuel. It is pointed out in Section 5.1.3 that the steam-fuel reaction is the greatest dose contributor for steam ingress events. Thus, the competing steam-graphite reactions may be expected to significantly alter the dose consequences of steam ingress events. As the graphite is oxidized, those fission products that had previously diffused into the graphite may also be released to the coolant. However, this release mechanism is negligible with regard to dose consequences.

The products of the steam-graphite reaction are the source of flammable gases during steam ingress accidents. Consequently, the steam-graphite reaction is of importance in assessing probabilities for burning in the containment.

1. Steam-Graphite Reaction Rate (R_{st})

The steam-graphite reaction rate is based on the Langmuir-Hinshelwood equation (Refs. 5-1, 5-5):

$$R_{st} = \frac{K_1 P_{st} F_b F_c}{1 + K_2 P_{hyd}^n + K_3 P_{st}} \quad , \quad (5-1)$$

TABLE 5-2
PARAMETER CLASSIFICATIONS SIGNIFICANT IN STEAM INGRESS EVENTS

A. Steam-Graphite Reaction

1. Steam-graphite reaction rate versus temperature and partial pressures of hydrogen and steam (R_{st})
2. Steam-graphite reaction rate versus burnoff (F_b)
3. Steam-graphite reaction rate versus catalyst concentration (F_c)
4. Catalyst distributions affecting steam-graphite reactions (ϕ)
5. Effective diffusion coefficient of steam through graphite (D_{eff})
6. Transverse flow effect on steam transport (D_{tot}/D_{eff})

B. Steam-Fuel Reaction

7. Fuel hydrolysis reaction rate versus temperature and steam partial pressure (R_F)
8. Moles of steam consumed per mole of fuel hydrolyzed (X)
9. Mass of UC_2 available for hydrolysis due to fuel particle coating failure (M_f)
10. Fractional release of noble gases from hydrolyzed fuel versus temperature (N_f)

C. Air-Graphite Reaction

11. Rate of air ingress to the PCRV (M_A)

D. Fission Product Initial Distributions

12. Gasborne activity (A_c)
13. Fuel body activity (A_F)

E. Burning in the Containment

14. Flammability limits

where R_{st} = rate of reaction, %/hr,

P_{st} , P_{hyd} = partial pressures of steam and
hydrogen, atm, and

F_b , F_c = linear modifiers for burnoff and
catalyst effects.

The temperature-dependent Arrhenius coefficients,

$$K_1 = 1069 \times 10^7 \exp(-40,900/RT), \text{ \% / hr - atm},$$

$$K_2 = 0.0166 \exp(28,600/RT), \text{ atm}^{-0.75}, \text{ and}$$

$$K_3 = 0.0531 \exp(27,500/RT), \text{ atm}^{-1},$$

are derived from experimental data, as is the exponential
constant $n = 0.75$.

Reference 5-1 presents test conditions and the graphite types for the important data sources involved in determining the empirical constants presented above. Some data for HTGR reference graphite, H-451, has been compiled (Ref. 5-6 and subsequent progress reports), but these data are too sparse to determine new rate constants. Confidence bounds for steam-H-451 graphite reaction rates were determined by assuming a lognormal distribution of data about a median given by Eq. 5-1. The standard deviation for the log of the steam-H-451 graphite reaction rates was determined and 95% confidence bounds were applied as multiplicative factors on the median reaction rates. For steam partial pressures below 0.1 atm these factors were determined to be 7.4 and 0.14.

For steam partial pressures above 0.1 atm, Eq. 5-1 is still believed to be the best representation of the reaction rate. However, there is evidence to suggest that the reaction rate may vary approximately as the square root of the steam partial pressure above 0.1 atm (Ref. 5-7). The data showing the greatest dependence on steam partial pressure, the 0.64 power, was chosen as an upper 95% confidence bound. For steam partial pressures above 0.1 atm, the upper bound multiplicative factor is given by:

$$F = \exp (1.65 \sigma_T)$$

where

$$\sigma_T = \left\{ \left[\frac{\ln(7.4)}{1.65} \right]^2 + \left[\frac{0.64}{1.65} \ln \left(\frac{P_{st}}{0.1} \right) \right]^2 \right\}^{1/2} \quad (5-2)$$

The first term in Eq. 5-2 is the standard deviation due to data scatter and uncertainties, and the second term results from the 0.64 power dependence on steam partial pressure suggested by Ref. 5-7. The two uncertainties are thus combined in a statistical manner. Equation 5-1 and the corresponding upper and lower confidence bounds for zero hydrogen partial pressures are shown in Fig. 5-1 for steam pressures above 0.1 atm.

2. Steam-Graphite Reaction Rate Versus Burnoff (F_b)

The effect of burnoff on the steam-graphite reaction rate is accounted for by a multiplying factor F_b on Eq. 5-1. An analysis was made of available data on burnoff factors for various types of graphite. Data trends appear to be more a function of experimental uncertainties than of graphite type. Figure 5-2 shows all available burnoff factor data normalized to the

Fig. 5-1. Median values and typical uncertainties (at zero H_2 pressure) for steam-graphite reaction rates

Fig. 5-2. Reaction rate burnoff factor as a function of burnoff (see Fig. 3-9 of Ref. 5-1 for reference to original data sources)

measured reaction rate at a burnoff of 1%. If burnoff b is expressed in percent, F_b may be represented by (Ref. 5-1)

$$F_b = 0.447 + 0.8094 b - 0.3221 b^2 + 0.0681 b^3 \\ - 0.0613 b^4 + 12.32(10^{-6}) b^5 + 2.89(10^{-5}) b^6 \\ - 1.15(10^{-6}) b^7 \quad . \quad (5-3)$$

Upper and lower 95% confidence bounds may be found from Eq. 5-3 by assuming that the data are normally distributed and applying additive constants of +0.2 and -0.2, respectively.

3. Steam-Graphite Reaction Rates Versus Catalyst Concentrations (F_c)

Experimental work (Ref. 5-8) has shown that, of the three fission product elements that sorb to any extent on graphite, barium, strontium, and cesium, barium is the critical fission product catalyst to the steam-graphite reactions, strontium is less catalytic, and cesium is not catalytic at all. Experimental measurements for barium and strontium as functions of concentration and temperature are shown graphically in Fig. 4 of Ref. 5-8. The rate data were normalized to the reaction rates for noncatalyzed graphite and expressed as a rate increase factor F_c . The rate increase factor F_c was found to be a function of both temperature and catalyst concentrations:

$$F_c = 1 + C_c \exp[12.153 - 4.264(10^{-3})T] \quad (5-4)$$

where C_c is the local catalyst concentration in mg/g graphite and T is in °F.

Upper and lower 95% confidence bounds were obtained by applying multiplicative factors of 1.62 and 0.62, respectively, which were found by assuming a lognormal distribution of the data. Figure 5-3 compares these bounds with the median value of Eq. 5-4 for an example barium concentration of 0.57 mg/g.

4. Catalyst Distributions Affecting Steam-Graphite Reactions (ϕ)

As discussed earlier, strontium and barium act as catalysts for the steam-graphite reaction. Their concentrations within the fuel rods and graphite of the core are calculated from experimental and analytical results as those existing at the time of the steam ingress event. Since parameters that describe fission product production (such as power and yields) are accurately known, the concentrations of these catalysts within fuel rods are also well known. Empirical factors for the decrease in catalyst concentrations across the fuel rod-graphite boundary are used to determine the maximum concentrations in the graphite. The equilibrium concentration gradients are calculated based on the diffusion of metallic species from the fuel rod-graphite interface to the coolant channel from current reference cesium transport data. Investigation of these techniques and data shows that the major uncertainty in metallic fission product concentrations is due to the sorption ratio for the concentration ratio across the fuel rod-graphite gap. From metallic fission product release calculations (Ref. 5-9), sorption ratios for the maximum, mean, and minimum local releases of strontium have been obtained. In addition, the sorption ratios for cesium and barium median local releases have also been established. Upper and lower bounds for cesium and barium were assumed to vary linearly with those for strontium. Table 5-3 gives the sorption ratios and associated upper and lower 95% confidence bounds.

Fig. 5-3. Barium catalyst factor effect on steam-graphite reaction rate

TABLE 5-3
METALLIC FISSION PRODUCT SPECIES SORPTION RATIOS

Isotope	Upper 95% Confidence Bound	Median	Lower 95% Confidence Bound
Sr	5.2	13.5	21.0
Cs	5.2 ^(a)	13.5	21.0 ^(a)
Ba	15.8 ^(a)	41.0	63.8 ^(a)

(a) Cesium and barium upper and lower limits are equal to the product of the median value times the ratio of the strontium sorption rates for upper and lower bounds to the median value.

5. Effective Diffusion Coefficient of Steam in Graphite (D_{eff})

The amount of steam transported into the graphite and, therefore, available to react with it is primarily determined by the diffusion coefficient of steam in graphite. The diffusion coefficient of steam in helium through graphite has been derived from experimental and analytical methods (Refs. 5-10 through 5-14) as

$$D_{eff} = 1.06 \times 10^{-6} T_P^{1.58} \text{ cm}^2/\text{sec} \quad , \quad (5-5)$$

where T is in $^{\circ}\text{K}$ and P is in atm.

From a review of the data in connection with the NRC licensing review of Ref. 5-1, 95% confidence bounds of -50% and +30% were obtained.

6. Transverse Flow Effect on Steam Transport (D_{tot}/D_{eff})

Transverse flow, the pressure-induced bulk flow of steam from the coolant holes to the fuel holes, may also affect the net steam transport to the fuel. Little data or analysis concerning this process is available. A factor of 2 on the diffusion coefficient of steam in graphite is believed to conservatively bound the transverse flow effects. A factor of 1 on the diffusion coefficient due to transverse flow effects implies that the uncertainty is negligible. A constant probability distribution is assumed between 1 and 2 for this transverse flow factor. Also, a median value of 1.5 is used.

Steam-Fuel Reaction (Reaction B). The present HTGR reference design incorporates mixed oxide and carbide fuel particles in the fuel rods. Fertile particle kernels consist of thorium oxide, while fissile particle

kernels consist of uranium carbide. It is not expected that steam will react chemically with the oxide particles, even those with nonintact coatings. However, it is known that steam can react with heavy metal carbides if their particle coatings are not intact. The reaction transforms the carbide compounds to heavy metal oxides and, if enough steam is present, water molecules can attach to these oxides in a process referred to as fuel hydrolysis. The deleterious consequence of hydrolysis is a swelling of the fuel particles with the attendant release of fission products, namely noble gases. These noble gases are major contributors to the dose consequences of steam ingress accidents.

7. Fuel Hydrolysis Reaction Rate Versus Temperature and Steam Partial Pressure (R_F)

Recent experimental data (Ref. 5-15) for UC_2 were utilized to evaluate a fuel hydrolysis rate function R_F and multiplicative factors for 95% confidence bounds:

$$R_F = 568 \exp(-5303/T) \left(\frac{P_{st}}{0.03} \right)^n \quad (\%/min) \quad , \quad (5-6)$$

$$\log_{10} (\text{uncertainty factors}) = \pm 2 [0.263 - 317.2/T + 1.18(10^5)/T^2] \quad ,$$

where T is in $^{\circ}K$, P_{st} is in atm, and $n = 0.5$.

The data used pertain to uncoated UC_2 particles exposed to a constant steam partial pressure of 0.03 atm and are conservative for particles with fractional coatings. No data for the dependence of the fuel hydrolysis reaction on steam partial pressures are available. In theory, the power dependence of

steam partial pressure lies between 0 and 1. Consequently, a uniform probability distribution is assumed for n , resulting in a median value of 0.5. Equation 5-6 is illustrated in Fig. 5-4 along with 95% confidence bounds for a steam partial pressure of 0.03 atm.

8. Moles of Steam Consumed Per Mole of Fuel Hydrolyzed (X)

The reaction of steam with UC_2 is given above (Reaction B). In this formula, the quantity X is the number of moles of steam that react with each mole of UC_2 . The quantity Y is the fraction of carbon released from UC_2 during the hydrolysis reaction but which remains in the hydrolysis product (experimental sample). The two hydrogen compound products in Reaction B represent gaseous molecules that are released from the sample during hydrolysis. The quantity X is important because it is used to determine the amount of steam still available for fuel hydrolysis. Unfortunately, the values of X and Y cannot be determined from simple weight gain data. However, upper and lower bounds on X may be obtained from weight gain data by assuming that all carbon from reacted fuel remains as fuel carbon, or all of it leaves the sample as hydrocarbons. Some data have been analyzed in this manner (Ref. 5-15) and are presented in Fig. 5-5. A constant value of $X = 2.6$ is used for temperatures less than 600°C , and $X = 2.0$ is used for temperatures greater than 600°C . Uncertainties of ± 0.6 are used on X to represent 95% confidence limits.

9. Fuel Failure Fraction Effect on Mass of UC_2 Available for Hydrolysis

The amount of UC_2 available for hydrolysis is dependent on the fraction of TRISO particles that have failed prior to the event. No further fuel failure is experienced during steam

Fig. 5-4. UC_2 fuel hydrolysis reaction rate versus inverse temperature

Fig. 5-5. Moles of water (X) reacting per mole UC_2 as a function of temperature

ingress events. A fuel particle coating failure table versus in-core irradiation life has been developed in Ref. 5-16. This treatment includes the kernel migration, defective and missing coating, and internal pressure-induced stress failure mechanisms. Table 5-4 gives the variation in fuel failure with in-core irradiation life as well as upper and lower 95% confidence bounds. The total inventory of UC_2 present in failed TRISO coated particles of a fuel element may be obtained by multiplying the values in Table 5-4 by the time-dependent UC_2 mass per fuel element. This latter mass varies also with irradiation life due to burnup; values are 0.47, 0.33, 0.26, and 0.22 kg per fuel element. Uncertainties in these latter mass values make a negligible contribution to the dose results.

10. Fractional Release of Noble Gases from Hydrolyzed Fuel Versus Temperature (N_f)

Knowing the extent of fuel hydrolysis, the release of fission products contained in fuel particle kernels may be computed based on available data. Evidence (Refs. 5-15, 5-17) shows that this fission product release consists essentially of noble gases. Iodine and other volatile fission products were not found to be released. Data for noble gas release fractions from completely hydrolyzed fuel versus temperature are shown in Fig. 5-6. Upper and lower 95% confidence bounds were calculated assuming a lognormal distribution of the data. The data presented are for krypton but are assumed to apply for xenon also. The amount of radioactive noble gases released due to fuel hydrolysis can be found from the product of three quantities: (1) the amount of noble gases contained in TRISO coated UC_2 particles, (2) the fraction of UC_2 that has hydrolyzed, and (3) the noble gas release fraction obtained from Fig. 5-6.

TABLE 5-4
STEADY-STATE FAILED TRISO FUEL PARTICLE FRACTIONS

In-Core Irradiation Life (yr)	Fraction of TRISO Particles Failed		
	95% Upper Limit	Median	95% Lower Limit
1	0.0016	0.0006	0.00025
2	0.0022	0.0009	0.00034
3	0.0038	0.0016	0.00050
4	0.0055	0.0026	0.00068

Fig. 5-6. Noble gas release from hydrolyzed fuel as a function of temperature

Air-Graphite Reaction (Reaction C). The air-graphite reaction need not be considered for Branch AC, the relatively highest risk steam ingress sequence. For this branch, no air gets into the PCRV during the brief time in which the safety relief valve lifts. However, the air-graphite reaction may be of importance when considering Branches AE and AG with regard to burning in the containment building. For these accidents, in which a passageway from the PCRV to the containment building is available, significant air ingress to the PCRV occurs.

11. Rate of Air Ingress to the PCRV (M_A)

For Branches AE and AG, analyses with the OXIDE-3 computer program (Ref. 5-18) have shown that a conservative air ingress rate of 0.34 lb/sec may be assumed. This value is considered to be an upper bound, since the model never considers two-way flow through the failed penetration even though such a possibility exists. A uniform probability distribution for air ingress rate was assumed with a median value of 0.17 lb/sec and deviations of ± 0.17 lb/sec to the 95% confidence bounds.

Fission Product Initial Distribution (D in Table 5-2). The distribution of fission products within the PCRV under normal operating conditions is important in assessing the dose consequences of steam ingress events. The dose contribution from the vented circulating activity is directly proportional to the total gasborne activity present, A_C . The dose contribution from the noble gases released during fuel hydrolysis is proportional to the amount of noble gases contained in the failed uranium carbide particles. This quantity is proportional to the fuel body activity, A_F .

12, 13. Gasborne (A_C) and Fuel Body Activities (A_F)

Median and upper 95% confidence limits for gasborne and fuel body activities are derived in Volume V. The median and upper

95% confidence bounds are presented in Section 3.2 of this report for the key noble gas nuclides.

Burning in the Containment (E in Table 5-2). Analyses of Branches AE and AG incorporate the probability of combustion of CO and H₂ exhausted from the PCRV to the containment. A key parameter for evaluation of such a probability is the flammable limits of the gas mixture of CO and H₂ in air with helium or steam as diluent.

14. Flammability Limits

The temperature-dependent flammability limits were obtained from Ref. 5-19. The curves used are shown in Figs. 5-2 through 5-6 in Ref. 5-19. For those lower limits away from the "noses" of the curves, 95% confidence bounds of $\pm 0.1\%$ were assigned considering data scatter and uncertainties in the temperature-dependence function.

Greater uncertainties are expected in the "nose" portions of the curves due to the relatively rapid change in fuel fractions. The 95% confidence bounds assigned to those limits were therefore $\pm 1.0\%$.

A summary of median values and upper and lower 95% confidence bounds is given in Table 5-5 for all parameters.

5.1.3. Results

5.1.3.1. Branch AC. Analysis of the consequences enabled new values to be established for median dose predictions of Branch AC including 95% confidence bounds. For this branch, burning of CO and H₂ in the containment is not considered since there is negligible probability of such

TABLE 5-5
STEAM INGRESS ACCIDENT PARAMETERS AND UNCERTAINTIES

Parameter	Median Value	Uncertainties Additive or Multiplicative Factors ^(a)	
		Upper 95% Limit	Lower 95% Limit
1. Steam-graphite reaction rate	Eq. 3-33 of Ref. 5-1	$\exp\left\{2\left[1 + (0.32 \ln P_{st}/0.1)^2\right]^{1/2}\right\}$	0.14
2. Burnoff effect	Eq. 3-40 of Ref. 5-1	+0.2	-0.2
3. Catalytic effect	Eq. 3-41 of Ref. 5-1	1.62	0.62
4. Phi factors	13.5 for Ba and Cs 41 for Sr	1.55	0.385
5. Diffusion coefficient of steam in graphite	Eq. 8-11 of Ref. 5-1	1.3	0.5
6. Transverse flow factor	1.5 x D	1.33	0.67
7a. Fuel hydrolysis rate vs temperature	568 exp(-5303/T) %/min	10^r ($r = 0.263 - 317.2/T + 1.18 \times 10^5/T^2$)	10^{-r}
7b. Fuel hydrolysis rate vs steam pressure	1/2 order reaction rate	First-order rate	Zero-order rate
8. Moles steam reacting per mole UC ₂	2.6 for T ≤ 600°C 2.0 for T > 600°C	+0.6	-0.6
9. Fuel failure fraction	0.000335λ + 0.000588	2.5	0.4
10. Noble gas release fraction in hydrolysis	0.00689 exp(0.00987 T) T ≤ 480°C 0.667 exp(0.000381 T) T > 480°C	1.79 for T ≤ 430°C 1.09 for T > 430°C	0.56 for T ≤ 534°C 0.92 for T > 534°C
11. Rate of air ingress	0.17 lb/sec	+0.17	-0.17
12. Circulating activity	3.86 x 10 ⁴ Ci	+2.39 x 10 ⁴ Ci	-1.48 x 10 ⁴ Ci
13. Fuel body activity	3.04 x 10 ⁵ Ci	+0.04 x 10 ⁵ Ci	-0.04 x 10 ⁵ Ci
14. Flammability limits	Figs. 5-2 through 5-6 of Ref. 5-21	+1% in nose	-1% in nose

(a) Factors are multiplicative except where noted by + or -.

burning due to the closing of the PCRV safety relief valve. Also, from the uncertainty analysis of Branch AC, conclusions may be made regarding the safety significance of parameters involved in analyzing steam ingress accidents, excluding flammability data, which are considered in the analysis of other branches.

Table 5-6 presents the median dose consequences and uncertainties found by varying the parameters in Section 5.1.2, one at a time, to their upper 95% confidence bound. The dose contributions due to the venting of gasborne activity and noble gases released due to fuel hydrolysis are presented in Table 5-7. Figure 5-7 presents the median doses and uncertainty bands compared with 10CFR100 regulatory limits and with Fulton Station PSAR calculated doses. The PSAR doses are above the 95% confidence upper bounds of the present study. The overall uncertainty factors listed in Table 5-7 were obtained by combining the individual parameter factors statistically. Those factors are shown in Fig. 5-8. The most safety-significant parameters in the source term model are those that directly affect the fuel hydrolysis reaction, fuel failure fractions M_f , and the fuel hydrolysis reaction rate R_f . This is due to the fact that noble gases released during fuel hydrolysis are the primary contributors to offsite doses.

The parameters that cause the strongest influence, e.g., have the greatest influence coefficients, are those that have the steepest slope in Fig. 5-8. A ranking of the influence coefficients for all parameters is presented in Table 5-8, which lists the sensitivity of the doses to these fission product transport or retention mechanisms, and indicates the primary escape paths and pertinent fission product mechanisms.

The dose contributions from key isotopes are shown in Figs. 5-9 and 5-10. The most significant isotope, with regard to whole body beta and gamma doses, is Xe-133. The isotope Kr-88 and its daughter Rb-88 are also important. Inhalation thyroid and inhalation bone doses are small compared

TABLE 5-6
SUMMARY OF MEDIAN DOSES AND PARAMETER UNCERTAINTY FACTORS FOR STEAM INGRESS EVENT (BRANCH AC)

	2-Hr 500-m EAB Doses			30-Day 2500-m LPZ Doses		
	Whole Body $\gamma + \beta$	Inhalation Thyroid	Inhalation Bone	Whole Body $\gamma + \beta$	Inhalation Thyroid	Inhalation Bone
Median dose (rem)	3.13×10^{-5}	1.50×10^{-8}	3.37×10^{-8}	8.73×10^{-6}	5.38×10^{-10}	1.54×10^{-9}
<u>Fission Product Parameters</u>						
Fuel body activity	1.012	1.012	1.012	1.012	1.012	1.012
Circulating activity	1.040	1.619	1.009	1.022	1.619	1.009
Steam-graphite reaction rate vs temperature	1.059	1.000	1.062	1.061	1.000	1.062
Burnoff effect	1.010	1.000	1.009	1.0009	1.000	1.009
Catalytic effect	1.020	1.000	1.021	1.020	1.000	1.021
Effective diffusion coeff. of steam	1.194	1.000	1.204	1.198	1.000	1.204
Transverse flow effect	1.213	1.000	1.223	1.218	1.000	1.222
Distribution coeff. of catalysts	1.001	1.000	1.001	1.001	1.000	1.001
Fuel hydrolysis reaction rate vs temperature	1.273	1.000	1.277	1.271	1.000	1.277
Fuel hydrolysis rate vs P_{ST}	1.257	1.000	1.269	1.263	1.000	1.269
Moles steam reacting per mole UC_2	1.203	1.000	1.213	1.208	1.000	1.213
Fuel failure	2.252	1.000	2.312	2.284	1.000	2.312
Noble gas release fraction vs temperature	1.069	1.000	1.075	1.074	1.000	1.075
Containment fallout and plateout	1.00	10.0	10.0	1.0	10.0	10.0
Decay and buildup enroute	1.0	1.01	1.00	1.0	1.02	1.01
<u>Building and Meteorological Parameters</u>						
Containment leak rate	5.0	5.0	5.0	5.0	5.0	5.0
Cleanup system filter efficiency	1.0	1.01	1.00	1.0	1.04	1.01
Cleanup system recirculation rate	1.038	1.13	1.13	1.098	1.83	1.90
Meteorology atmospheric dispersion factor	14.0	14.0	14.0	16.0	16.0	16.0
Release height	0.195	0.195	0.195	0.96	0.96	0.96
$UF = \exp \left[\sum_p (\ln f)^2 \right]^{1/2}$						
Overall uncertainty factors	37.2	67.7	73.9	28.5	55.8	61.4

TABLE 5-7
CONTRIBUTIONS OF GASBORNE ACTIVITY AND HYDROLYSIS RELEASE
TO OFFSITE DOSE CONSEQUENCES

Dose	Gasborne Activity (%)	Noble Gas Release During Hydrolysis (%)
2-hr EAB whole body γ or β	6	94
30-day LPZB whole body γ or β	4	96
Inhalation thyroid	100	0
2-hr EAB inhalation bone	2	98 ^(a)
30-day LPZB inhalation bone	2	98 ^(a)

^(a) Results from decay of kryptons to strontiums.

Fig. 5-7. Summary of radiological dose consequences and uncertainties compared to regulatory guidelines for steam ingress Branch AC

Fig. 5-8. Variation in 2-hr EAB whole body gamma plus beta dose within parameter uncertainties (Branch AC)

TABLE 5-8
INFLUENCE COEFFICIENTS FOR STEAM INGRESS
(BRANCH AC, 30-DAY WHOLE BODY $\gamma + \beta$ DOSE AT LPZB)

Rank	Parameter	Parameter Ratio $P_{\max}/\bar{P}$	Dose Ratio $D_{\max}/\bar{D}$	Influence ^(a) Coefficient
1	Fuel body Xe + Kr activity	1.01	1.01	1.0
2	Fuel failure fraction	2.5	2.28	0.85
3	Noble gas release in hydrolysis	~ 1.1	1.07	0.70
4	Moles steam per mole UC ₂ (X)	~ 1.3	1.21	-0.70
5	Steam diffusion coefficient	1.3	1.20	0.67
6	Transverse flow factor	1.33	1.22	0.67
7	Fuel hydrolysis rate versus P_{st}	~ 2	1.26	0.26
8	Fuel hydrolysis rate versus T	~ 3	1.27	0.14
9	Burnoff effect on parameter 12	1.2	1.01	-0.05
10	Catalytic effect on parameter 12	1.62	1.02	-0.03
11	Gasborne activity	1.625	1.02	0.03
12	Steam-graphite reaction rate	~ 8	1.06	-0.01
13	Distribution coefficient of catalysts ϕ	1.55	1.001	0.002

(a) Influence coefficient $\approx \left(\frac{D_{\max}}{\bar{D}} - 1 \right) \left(\frac{P_{\max}}{\bar{P}} - 1 \right)$

where D = dose and P = parameter value.

Fig. 5-9. Isotope contributions to 2-hr EAB doses for steam ingress Branch AC

Fig. 5-10. Isotope contributions to 30-day LPZB doses for steam ingress Branch AC

to the whole body doses. Consequently, the isotopes mentioned above are those with the greatest safety significance for steam ingress events. It is noted that primary source for the strontiums, which constitute the bulk of the inhalation bone dose, comes from the decay of kryptons (for example, $\text{Kr-89} \rightarrow \text{Rb-89} \rightarrow \text{Sr-89}$).

5.1.3.2. Branches AE and AG (Considering Burning). To evaluate the effect of flammability as a parameter influence on steam ingress consequences, median dose consequences were derived for the two main branches in which flammability can be important (Branches AE and AG, as discussed in Section 5.1.1). Parameters that affect flammability were then varied to their 95% confidence bounds and the effects on doses were assessed as described below. Dose uncertainties due to other parameters can be inferred from the analysis of Branch AC.

The probabilities and consequences of burning in the containment are difficult to determine quantitatively. The consequences of burning can conceivably range from absolutely no effect on doses to the other extreme of causing a breach in the containment building and failure of the CACS. This latter extreme, of course, results in the greatest dose consequences of any event sequence considered. It is important, therefore, to determine the probability of its occurrence due to burning. This analysis attempts to show whether or not significant occurrence probabilities of this extreme event or any other lesser event, as measured by dose consequences, may occur as a result of burning.

The problem will be broken into three stages: (1) the probability of obtaining a flammable mixture of gases in the containment building, (2) the probability of obtaining an ignition source within the flammable mixture, and (3) the possibility of damage occurring to components within the containment building as a result of burning. The problem is further complicated by the fact that different levels of burning may occur. For this analysis, burning is divided into three categories: (1) plume

burning, in which the flammable gases are burned as they emerge from the PCRV, (2) global burning, in which all of the flammable gases in the containment, although postulated to be perfectly mixed, are burned, and (3) pocket burning, in which only pockets of flammable gases within the containment are burned. The first two categories are modeled in OXIDE-3. Probabilities for the third, pocket burning, are inferred from the other two.

The median values of the parameters discussed in Section 5.1.2 were input to the OXIDE-3 code to determine whether plume or global flammable mixtures would occur in the containment building for either Branch AE or AG. For Branch AE, neither plume nor global flammable mixtures occurred in the median case. For Branch AG, a flammable plume was obtained but not a global flammable mixture. Even though the median gas mixtures may not exceed the flammable limits discussed in Section 5.1.2, a significant probability of burning may still exist provided some of the parameters deviate from their median in such a way as to increase the quantity of flammable gases flowing into the containment. In general, when the flammable limits are not exceeded it is due to insufficient quantities of $\text{CO} + \text{H}_2$ rather than a lack of air, since sufficient air is available in the containment building. Probabilities for attaining flammable mixtures can therefore be determined in the following manner. First, the median and 95% confidence bounds are determined for the maximum plume and global $\text{CO} + \text{H}_2$ molar fractions in the containment obtained in the analysis of both Branches AE and AG. The median and 95% confidence bounds of the corresponding $\text{CO} + \text{H}_2$ molar fraction flammable limits are also determined. Lognormal probability distributions are assumed for each of the fractions listed above. The probabilities of attaining a global or plume flammable mixture can then be approximated by the equation

$$P(\text{burning}) = \frac{1}{(2\pi)^{1/2}} \int_{\frac{Q}{\sigma}}^{\infty} \exp(-x^2/2) dx, \quad (5-7)$$

where $Q_1 = Q - Q_o$,

$$\sigma = [\sigma_1^2 + \sigma_o^2] \quad ,$$

Q_1 , Q_o are the logarithms of the median CO + H₂ flammability limit and the molar fraction obtained, and σ_1 and σ_o are the standard deviations in the logarithms of the flammability limit and the CO + H₂ fraction obtained.

The key objective is to determine the characteristics of the distributions. Optimally, the OXIDE-3 code could be used to determine the distribution of CO + H₂ molar fraction obtained by varying each parameter discussed in Section 5.1.2 in turn through its upper and lower confidence bounds. Unfortunately, the OXIDE-3 code requires large amounts of computer time for the long simulation times required to analyze burning in the containment. This approach was, therefore, deemed cost prohibitive. Instead, only the most important parameters affecting fuel fractions in the containment were varied. The OXIDE-3 runs made are listed below:

1. Branch AE, median parameters
2. Branch AE, upper bound on diffusion coefficient of steam (D_{eff})
3. Branch AE, upper bound on steam-graphite reaction rate (R_{st})
4. Branch AE, upper bound on air ingress rate.
5. Branch AG, median parameters
6. Branch AG, upper bound on steam-graphite reaction rate (R_{st}).

The short-time effects of all the parameters were obtained from the OXIDE-3 runs made for Branch AC. At short times, Branch AC is exactly the same as Branch AE. Linear correlations were assumed for the other

parameters between certain quantities obtained at short times and the CO + H₂ molar fractions obtained at long times. For global burning, the quantity of carbon monoxide produced at short times was used. For plume burning, the CO + H₂ molar fractions in the gases vented at short times were correlated. These correlations gave rise to 4 multiplication factors (global and plume for both branches AE and AG) for each parameter, which may be applied to the predicted CO + H₂ molar fractions to arrive at the upper confidence bounds. Table 5-9 lists these factors.

The overall multiplication factor is obtained by employing the equation

$$\text{overall multiplication factor} = \exp \left[\sum_p (\ln f_p)^2 \right]^{1/2}, \quad (5-8)$$

where f_p is the multiplication factor on the CO + H₂ molar fraction obtained by varying the pth parameter to its upper uncertainty limit.

The distributions for the flammable limits were more easily determined. The temperature uncertainties applicable to global and plume flammable mixtures for both Branches AE and AG were determined from the OXIDE-3 runs mentioned above. The corresponding temperature-induced uncertainties in flammable limits were combined with the experimental data scatter uncertainties for the limits discussed in Section 5.1.2. The probability distributions used for the CO + H₂ molar fractions obtained and their limits are shown in Figs. 5-11 and 5-12. Probabilities for obtaining flammable mixtures for the two steam ingress Branches AE and AG were found from Eq. 5-7, recognizing that the assumption of lognormal probability distributions is only approximate. The calculated probabilities for obtaining flammable mixtures in the containment are given in Table 5-10.

TABLE 5-9
MULTIPLICATION FACTORS ON CO + H₂ MOLAR FRACTIONS OBTAINED BY VARYING
PARAMETERS TO UPPER 95% CONFIDENCE BOUNDS

Parameter	Branch AE		Branch AG	
	Plume	Global	Plume	Global
Fuel hydrolysis reaction rate vs temperature	2.50	1.40	1.052	1.07
Fuel hydrolysis reaction rate vs steam pressure	1.76	1.25	1.03	1.04
Transverse flow factor	4.05	1.62	1.11	1.11
Diffusion coefficient of steam	3.70	1.55	1.09	1.09
Moles steam reacting per mole UC ₂	1.50	1.13	1.02	1.02
Steam-graphite reaction rate vs temperature	4.47	3.36	1.12	1.40
Catalytic effect	1.33	1.23	1.01	1.04
Burnoff effect	1.26	1.15	1.01	1.03
Catalyst concentrations	1.25	1.10	1.01	1.02
Air ingress	4.58	2.03	4.58	2.03
Fraction CO + H ₂ flammability limit	0.092	0.083	0.092	0.083
Maximum CO + H ₂ molar fraction	7.13×10^{-3}	5.18×10^{-4}	0.298	0.016

Fig. 5-11. Branch AE containment gas fuel fraction probability distribution

Fig. 5-12. Branch AG containment gas fuel fraction probability distribution

TABLE 5-10
PROBABILITIES FOR OBTAINING FLAMMABLE MIXTURES GIVEN
SEQUENCES AE AND AG

Event Branch	Type of Burning Considered	Probability of Obtaining Flammable Mixture
AE	Plume	0.05
	Global	2×10^{-10}
AG	Plume	0.73
	Global	2×10^{-5}

Few data are available to determine the occurrence probabilities of ignition sources required for burning. Ignition sources can conceivably result from electrical fires or sparks generated by the operating electrical equipment within the containment. Electrical equipment that will be operating in the containment during the event sequences investigated will vary from plant to plant due to the placement of this equipment. In all plants, the containment cleanup system will be operating. However, the system location and the motor enclosures are such that sparks produced by the equipment are not expected to reach the plume burning region.

Sparks may also be generated as a result of mechanical movements in the containment. Other hard-to-predict ignition sources are conceivable. Fires within the containment, electrically caused or otherwise, have a low probability of occurrence. Circumstances such as having combustible materials in the containment and failure to close electrical relay and switch boxes may serve to produce additional sources. Qualitatively, the occurrence of ignition sources for plume burning should be less frequent than for global or pocket burning. For this study, the probability of obtaining a global or pocket flammable ignition source is taken to be 0.5. For plume flammable mixtures, the probability used is 0.1. These factors are based on engineering judgment.

The third stage in the investigation of burning within the containment is to assess the extent of damage to components caused by burning. Of primary concern are the electrical cables providing power to the CACS, the containment liner, and the containment cleanup system.

The electrical cables providing power to the CACS are enclosed in steel conduits. These cables are known to be able to withstand temperatures of 750°F for 30 min. The containment liner and cleanup system can withstand pressures in excess of 75 psia and extremely high containment gas temperatures. Plume burning would not result in damage to any safety-related containment components. The quantity of CO + H₂ available for this type of burning is not sufficient to cause any damage. Plume burning for extended periods of time would not occur. Extended burning would lead to a pressure rise in the containment, which in turn would prevent further venting of flammable gases from the PCRV. Global burning could conceivably cause failure of the containment pressure boundary in some manner. The operability of electrical cables in the containment that are subjected to the high temperatures caused by global burning would be jeopardized. Consequently, global burning is assumed to always lead to a loss of forced circulation with containment failure, while plume burning is assumed to have an undetermined probability of causing such failures ranging from 0 to 1.

Depending on the size and location of the flammable pocket, the dose consequences of pocket burning range from those of global burning to none. The formation of flammable pocket mixtures will have a greater occurrence probability than that of global burning but less than that of plume burning. The probability is highly dependent on the amount of mixing that goes on in the containment building. The extent of this mixing is believed to be substantial. The gas circulation induced by the containment cleanup system and the venting of relatively high temperature gases from the PCRV should aid the mixing process. Separation of gases according to molecular weight is not expected to occur.

Results of the investigation of burning in the containment, for Branches AE and AG of the preliminary study, are summarized in Figs. 5-13 and 5-14. For these figures, global burning was treated as a subset of pocket burning. The global burning is the maximum or largest pocket burning which encompasses the whole containment building. The important branches are seen to be BK and CK, in which pocket burning occurs, and consequences range from no effect to a loss of forced circulation and containment failure. Occurrence probabilities of branches that include pocket burning have large uncertainties. To narrow these uncertainties, the extent of mixing that occurs in the containment building must be quantified. Once the locations and volumes of flammable pockets are identified, damages from pocket burning may be assessed.

The highest risk in the ranges assessed occurs for Branch BK, where the upper limit risk is 2×10^{-10} rem/yr, for the extreme of containment failure and loss of forced circulation caused by pocket burning. This risk slightly exceeds that for Branch AC so that it can be concluded that probabilities of flammability are sufficiently low that gas burning does not materially increase the statistical risk of steam ingress events.

From Figs. 5-11 and 5-12 one sees that the flammable limits are relatively well known. The experimental data base for flammability limits may then be regarded as sufficient. Sizeable uncertainties exist for the maximum $\text{CO} + \text{H}_2$ molar fractions determined by the computer analysis. The parameters primarily responsible for these uncertainties are (1) the rate of air ingress to the PCR, (2) the steam-graphite reaction rate versus temperature and partial pressures of steam and hydrogen, (3) the effect of transverse flow on steam transport, and (4) the diffusion coefficient of steam through graphite. These parameters primarily determine the quantity of flammable gases produced due to steam ingress. If an analysis of mixing in the containment indicates that pockets of flammable gases would form, these parameters would play a key role in determining the consequences of pocket burning.

Fig. 5-13. Event logic subtree for Branch AE. For the purposes of this volume, the notation of ϵ to indicate $<10^{-10}$ as described in Volume IV of this report was not used.

Fig. 5-14. Event logic subtree for Branch AG. For the purposes of this volume, the notation of ϵ to indicate $<10^{-10}$ as described in Volume IV of this report was not used.

5.1.4. Revision of Steam Ingress Event Sequences

The preliminary probabilistic analysis of the moisture inleakage event sequences was presented in Section 4.8 of Volume III. The present analysis represents an update of that preliminary work. The changes are reflected in both the occurrence probabilities and the dose consequences. Figure 5-15 represents the revised event sequence tree diagram of the preliminary work; the updated 30-day LPZ and 2-hr EAB whole body and thyroid doses are listed in Table 5-11.

5.1.4.1. Occurrence Probability. In recent analysis, the probability of failure of the moisture monitors to detect the defective steam generator was estimated to be 1.2×10^{-3} instead of 1.2×10^{-2} as reported in the preliminary analysis. This also includes the effect of common mode failure. The new analysis was based on a failure rate of $\lambda = 20 \times 10^{-6}/\text{hr}$. Also, in the preliminary analysis for the events leading to reactor depressurization with excessive steam inleakages, a burning probability of 1 was conservatively assumed in the containment. In the new analysis, the probability of obtaining a local (either plume or pocket) flammable mixture for each of the affected branches was reassessed based on the information generated in the previous section. Global burning was neglected since it has much lower importance. The probability of burning for each branch is a multiplication of the probability of attaining a local flammable mixture times the probability of initiation by an ignition source. In the present analysis, a constant probability of 0.1 for ignition generated by an ignition source was used for all the branches.

A simplified approach was taken in assessing consequences in that local burning was assumed to lead to either containment failure or no failure ($P = 1$ or 0 for failure) in each sequence, depending on the amount of water ingress, reaction products generated, etc. Thus, in sequences where the faulted steam generator is correctly isolated and dumped, the probability of containment failure is zero. In wrong-loop-dump sequences, the probability of containment failure due to local burning was

Fig. 5-15. Event logic tree for steam ingress (updated). For the purposes of this volume, the notation of ϵ to indicate $<10^{-10}$ as described in Volume IV of this report was not used.

TABLE 5-11
MEDIAN CONSEQUENCES OF STEAM INGRESS SEQUENCES

Branch Identification	Branch Probability (a) (per yr)	Whole Body $\gamma + \beta$			Inhalation Thyroid		
		EAB 2-hr Dose (rem)	LPZ 30-Day Dose (rem)	LPZ 30-Day Risk (rem/yr)	EAB 2-hr Dose (rem)	LPZ 30-Day Dose (rem)	LPZ 30-Day Risk (rem/yr)
A	2.5×10^{-3}	0	0	0	0	0	0
B	2.5×10^{-11}	4.6×10^{-5}	1.2×10^{-5}	3×10^{-16}	2.5×10^{-4}	9×10^{-6}	2.3×10^{-16}
C	8.9×10^{-17}	4.6×10^{-3}	1.2×10^{-3}	1.1×10^{-19}	2.5	9×10^{-4}	8×10^{-20}
D	9.4×10^{-8}	2.2×10^{-5}	6.2×10^{-6}	5.9×10^{-13}	1.1×10^{-8}	3.8×10^{-10}	3.6×10^{-17}
E	1×10^{-8}	8.6×10^{-5}	2.4×10^{-5}	2.4×10^{-13}	1.8×10^{-4}	7.3×10^{-6}	7.3×10^{-14}
F	1×10^{-11}	7.5×10^{-4}	2.2×10^{-4}	2.2×10^{-15}	1.8×10^{-4}	7.3×10^{-6}	7.3×10^{-17}
G	3.6×10^{-17}	7.5×10^{-2}	2.2×10^{-2}	7.9×10^{-19}	1.8	7.3×10^{-4}	2.6×10^{-20}
H	0	0	0	0	0	0	0
I	2.1×10^{-14}	7.5×10^{-4}	2.2×10^{-4}	4.6×10^{-18}	2.5×10^{-4}	9×10^{-6}	1.9×10^{-19}
J	7.5×10^{-20}	7.5×10^{-2}	2.2×10^{-2}	1.7×10^{-21}	2.5	9×10^{-4}	6.8×10^{-23}
K	1.8×10^{-8}	0	0	0	0	0	0
L	1.8×10^{-19}	1.3×10^{-3}	3.6×10^{-4}	6.5×10^{-23}	2.5×10^{-4}	9×10^{-6}	1.6×10^{-24}
M	6.4×10^{-22}	1.3×10^{-1}	3.6×10^{-2}	2.3×10^{-23}	2.5	9×10^{-4}	5.8×10^{-25}
N	1.4×10^{-11}	2.2×10^{-5}	6.2×10^{-6}	8.7×10^{-17}	2×10^{-8}	4.5×10^{-10}	6.3×10^{-21}
O	1.5×10^{-12}	1.9×10^{-4}	5.4×10^{-5}	8.1×10^{-17}	1.8×10^{-4}	7.3×10^{-6}	1.1×10^{-17}
P	4.4×10^{-18}	1.5×10^{-3}	4.3×10^{-4}	1.9×10^{-21}	1.8×10^{-4}	7.3×10^{-6}	3.2×10^{-23}
Q	4.7×10^{-21}	1.5×10^{-1}	4.3×10^{-2}	2×10^{-22}	1.8	7.3×10^{-4}	3.4×10^{-24}
R	1.2×10^{-7}	0	0	0	0	0	0
S	7.4×10^{-13}	2.2×10^{-5}	6.2×10^{-6}	4.6×10^{-18}	2×10^{-8}	4.5×10^{-10}	3.3×10^{-22}
T	7.9×10^{-14}	1.9×10^{-4}	5.4×10^{-5}	4.3×10^{-18}	1.8×10^{-4}	7.3×10^{-6}	5.8×10^{-19}
U	3.2×10^{-19}	2×10^{-3}	5×10^{-4}	1.6×10^{-22}	1.8×10^{-4}	7.3×10^{-6}	2.3×10^{-24}
V	2.8×10^{-22}	2×10^{-1}	5×10^{-2}	1.4×10^{-23}	1.8	7.3×10^{-4}	2×10^{-25}
W	6.2×10^{-16}	5×10^{-5}	1×10^{-5}	6.2×10^{-21}	3×10^{-8}	5×10^{-10}	3.1×10^{-25}
X	6.6×10^{-17}	3×10^{-4}	6×10^{-5}	4×10^{-21}	1.8×10^{-4}	7.3×10^{-6}	4.8×10^{-22}
Y	3.3×10^{-25}	3×10^{-3}	8×10^{-4}	2.7×10^{-25}	1.8×10^{-4}	7.3×10^{-6}	2.4×10^{-30}
Z	2.4×10^{-25}	3×10^{-1}	8×10^{-2}	1.9×10^{-26}	1.8	7.3×10^{-4}	1.8×10^{-28}
AA	2.8×10^{-25}	3×10^{-1}	8×10^{-2}	2.2×10^{-26}	2.5	9×10^{-4}	2.5×10^{-28}
AB	2.5×10^{-28}	3×10^{-1}	8×10^{-2}	2×10^{-29}	2.5	9×10^{-4}	2.3×10^{-31}
AC	2.7×10^{-6}	3.1×10^{-5}	8.8×10^{-6}	2.4×10^{-11}	1.5×10^{-8}	5.4×10^{-10}	1.5×10^{-15}
AD	2.9×10^{-7}	6.2×10^{-4}	1.8×10^{-4}	5.2×10^{-11}	1.8×10^{-4}	7.3×10^{-6}	2.1×10^{-12}
AE	1.5×10^{-12}	5.2×10^{-3}	1.5×10^{-3}	2.3×10^{-15}	1.8×10^{-4}	7.3×10^{-6}	1.1×10^{-17}
AF	1.0×10^{-15}	5.2×10^{-1}	1.5×10^{-1}	1.5×10^{-16}	1.8	7.3×10^{-4}	7.3×10^{-19}
AG	2.1×10^{-15}	5.2×10^{-3}	1.5×10^{-3}	3.2×10^{-18}	2.5×10^{-4}	9×10^{-6}	1.9×10^{-20}
AH	1.1×10^{-19}	5.2×10^{-1}	1.5×10^{-1}	1.7×10^{-20}	2.5	9×10^{-4}	1×10^{-22}
AI	2.3×10^{-9}	4×10^{-5}	9×10^{-6}	2.1×10^{-14}	3×10^{-8}	5×10^{-10}	1.2×10^{-18}
AJ	2.4×10^{-10}	3×10^{-4}	6×10^{-5}	1.5×10^{-14}	1.8×10^{-4}	7.3×10^{-6}	1.8×10^{-15}
AK	1.5×10^{-15}	5.2×10^{-1}	2×10^{-1}	3×10^{-16}	1.8	7.3×10^{-4}	1.1×10^{-18}
AL	8.7×10^{-19}	5.2×10^{-1}	2×10^{-1}	1.8×10^{-19}	1.8	7.3×10^{-4}	6.4×10^{-22}
AM	2×10^{-18}	5.2×10^{-1}	2×10^{-1}	4×10^{-19}	1.8	7.3×10^{-4}	1.5×10^{-21}
AN	9.1×10^{-23}	5.2×10^{-1}	2×10^{-1}	1.8×10^{-23}	1.8	7.3×10^{-4}	6.6×10^{-26}

(a) For the purposes of this volume, the notation of ϵ to indicate $<10^{-10}$ as described in Volume IV was not used.

taken to be zero or unity, depending on the specific branch sequence circumstances. These judgments were based on experience gained with OXIDE-3 computer analyses of the thermodynamic response in the containment resulting from burning under these circumstances.

5.1.4.2. Consequences. The dose consequences of the preliminary analysis have been revised. As explained in Section 3 of this volume, the preliminary dose consequences were based on Fulton Station PSAR results, which were appropriately scaled down for a number of factors. The new dose consequences are based on detailed analysis for Branches AC, AE, and AG using similar extension factors for other branches as in the preliminary study.

5.1.4.3. Results and Conclusions. All the important sequences are presented in Fig. 5-16. Compared to similar plots of the preliminary work, the figure shows fewer branches are within the envelope of interest. This is mainly due to changes in the dose consequence. Table 5-12 summarizes the most important branches, which are similar to the preliminary work, although the order of the first two branches has been reversed and Branch AD has a higher risk than Branch AC. In conclusion, the revision of the preliminary analysis slightly decreased the risk level.

5.2. DEPRESSURIZATION EVENTS

5.2.1. Description

Depressurization events were investigated in the preliminary AIPA analysis (Volume III), in which event sequences and branch diagrams were constructed and preliminary risks were assessed. It was found that the slow depressurization event exhibits a higher risk than the DBDA, with the

Fig. 5-16. Risk plot for steam ingress branches

TABLE 5-12
HIGHEST RISK SEQUENCES FOR STEAM INGRESS

Branch Identifi- cation	Branch Description	Probability ^(a) (per yr)
AD	WLD, ^(b) safety valve failure, manual block valve closure	5.2×10^{-11}
AC	WLD	2.4×10^{-11}
D	Correct PPS action, loss of main loops	5.9×10^{-13}
E	Correct PPS action, main loops, manual block valve closure	2.4×10^{-13}
AI	WLD, loss of main loops	1.7×10^{-13}

(a) For the purposes of this volume, the notation of ϵ to indicate $<10^{-10}$ as described in Volume IV of this report was not used.

(b) Wrong loop dump.

risk of both events being below that for the reheater tube leak. The primary reasons for studying depressurization events were:

1. The DBDA was a design basis accident in the Summit and Fulton PSARs and, thus, is of interest to compare its consequences to those of other events.
2. The depressurization events involve important fission product technologies, which are also of interest in other PCRV blowdown events such as reheater leaks.

The probabilities and preliminary estimate of consequences for the depressurization event tree are given in Volume III. The event tree for a slow depressurization is similar, except for a different probability of the initiating event. The branch used as the base case for this detailed evaluation of the consequences and uncertainties, Branch A (in which all systems perform correctly after the initiating event), has the highest risk of all the depressurization event sequences. Dose consequences were evaluated for the median case (the case in which median values or best estimates of all parameters were used) and for upper and lower 95% confidence limits of each parameter individually.

5.2.2. Parameter and Consequence Evaluation

For a given initiating depressurization event of a specific hole size, the basic parameters that affect the fission product release from the PCRV to the containment are:

1. Initial primary circuit gasborne activity. Median values and 95% confidence limits of this parameter were derived in Volume V and are discussed in Section 3.2.

2. Initial plateout activity in the primary circuit. Median values and 95% confidence limits were derived in Volume V and are discussed in Section 3.2.
3. Release of initial plateout activity due to:
 - a. Liftoff from fixed surfaces
 - b. Blowdown of activity on potentially circulating dust
 - c. Decay of remaining plateout to gasborne products and subsequent release from PCRV.

For the latter quantity, the first two of the three identified mechanisms of release of initial plateout activity are directly dependent on the amount of dust circulating in the primary system. Carbonaceous dust may arise partially from mechanical and chemical processes, which are sufficiently complex so that accurate predictions of expected dust concentrations are not presently available. For the present study it was assumed that the two parameters, liftoff of fixed surface plateout and blowdown of activity on circulating dust, are perfectly correlatable with the controlling variable dust concentration, as explained below. The third parameter, decay of remaining plateout, is a straightforward calculation once the remaining plateout is determined by subtracting the immediate blowdown and liftoff release from the initial plateout inventory.

Given the total plateout inventories, it is necessary to determine what fraction will deposit on gasborne dust and what steady-state activity levels will exist in the primary coolant. To formulate a mechanistic model of the behavior of particulates in the primary circuit, the following information is necessary:

1. The amount of dust produced and the size distribution of the dust particles.

2. Parameters associated with the physical behavior of the dust, such as sticking factors for dust impacting on surfaces and shear force ratios required for liftoff of the growing dust filaments.
3. Behavior of the condensable nuclides released from the core, i.e., deposition on:
 - a. Gasborne dust
 - b. Fixed surfaces
 - c. Dust residing on immobile surfaces and the transfer of activity from this dust to the surfaces themselves.

Since these data are not available, a mechanistic model cannot be constructed. Rather, the full range of plateout on dust must be examined.

The probability distribution assumed for plateout on dust is a uniform distribution from 0% to 100%; this was chosen due to a lack of basis for a median value. The low limit of 0% plateout on dust is possible ideally for favorable primary circuit chemistry and dust formation dynamics. On the other hand, results of a preliminary study, using a simple boundary layer diffusion-limited model, have indicated that plateout of activity on dust is sensitive to dust size and that significant amounts of activity can be deposited on gasborne dust for moderate steady-state dust levels. (For example, in a parameter calculation, greater than 65% of the Ce-144 activity released from the core was plated out per pass on 1-micron-diameter particles at a maximum predicted steady-state dust level.

The following section describes in detail the way in which the dust-borne activities are calculated. The percent plateout on dust is actually the ratio of the plateout rate on dust to the total plateout rate. This

fraction was varied from 0.5 through 0.95 in going from lower to upper conditions. The conditions were:

1. Lower case - 5% dust plateout rate, median distribution, GAIL IV liftoff correlations (Ref. 5-20).
2. Median case - 50% dust plateout rate, design distribution, GAIL IV liftoff correlations
3. Upper case - 95% dust plateout rate, uniform activity distribution, Ce-144 liftoff correlation.

The method of calculating doses for these depressurization events is as follows. First, steady-state levels are obtained for gasborne activity and activity resident on gasborne dust. Then a calculation is made of that portion of activity which is lifted off surfaces in the primary circuit during the depressurization. This liftoff activity is added to the coolant-borne activities; the total obtained represents the activity effluent from the PCRV during depressurization. For the slow depressurization, the velocity of the exiting gas is so low that liftoff is negligible, so that only coolant-borne activity is figured in the dose. In both cases, decay to noble gases of the plateout remaining in the PCRV is also accounted for. A more detailed account of how the above quantities are calculated is given in subsequent paragraphs.

As described in the preceding section, the percent plateout on dust, the plateout distribution, and liftoff correlations are co-varied in going from the 5% to the 95% confidence limits in the dust-liftoff parameter. This is done because the amount of dust in the primary circuit affects (1) the fraction of the total plateout that is on the dust, (2) the distributions of these nuclides on fixed surfaces in the circuit, and (3) their propensity for liftoff in a depressurization. In terms of probability distributions, the parameters of percent plateout on dust and liftoff of

activity are correlated; therefore, to vary each one independently would be invalid.

To account for this dependence, two different activity distributions have been defined. The first is the distribution to be used with "normal" conditions in the loop, that is, when the fixed surface plateout occurs primarily by sorption of atomic species, and not by plateout on circulating dust.

The median, 95% confidence bound, and HTGR design and expected plateout distributions to be used for Sr-90, I-131, Cs-137, La-140, and Ce-144 throughout the primary coolant circuit are presented in Volume V. The total plateout surface area in the primary circuit of the 3000-MW(t) HTGR is $4.7 \times 10^8 \text{ cm}^2$. The relative distribution of I-131 is valid for all the iodine isotopes and the relative distribution of Ce-144 is valid for all the nuclides that do not have a noble gas precursor. Hence, these figures are applicable for all the important plateout nuclides in the primary coolant circuit. For all other nuclides that have a noble gas precursor, a uniform plateout distribution throughout the primary coolant circuit is applied.

When plateout profiles from these figures are normalized to a total plateout of unity, they can be applied to different values of total plateout.

The alternative distribution, to be used when most of the activity is presumed to be residing on particulates, is a uniform distribution of each nuclide over all surfaces in the loop. This would apply if all sites were equally susceptible to particle deposition and entrainment such that activity plated out on dust residing on surfaces could be reentrained and deposited elsewhere.

The percentage of activity lifted off a surface is a function of the shear force exerted at the surface and the adherence of the nuclides to the site. Thus, to calculate liftoff, a correlation of fractional liftoff

versus shear force ratio is necessary. It is evident that the relation depends on whether the activity is situated on dust on the surfaces or is sorbed into the metal surface itself. Therefore, two liftoff correlations must be defined to reflect (1) the liftoff correlation without dust and (2) the effects of primary coolant dust on liftoff.

The first liftoff correlation to be used with the median distribution is a series of curves (Fig. 5-17) derived from GAIL IV loop data (Ref. 5-20). These represent the best data available at present. As recommended in Ref. 5-21, a constant 1.1% liftoff for the iodines was used because the iodine data show considerable scatter, even at low shear ratios (see Fig. 5-17). When allowance is made for desorption, the value of 1.1% liftoff in all regions is obtained. For nuclides (and their isotopes) other than those for which correlations are given, the Ce-144 correlation is used.

For the second correlation, a layer of dust is postulated to coat much of the PCRV internal surface area. Fission products deposited on this dust layer would be more easily reentrained in a depressurization, causing greater amounts of activity to be released from the PCRV. Cerium, which is the least adherent of the tested nuclides, is believed to exhibit this particular behavior because of its association with larger particles. Reference 5-22 states that a portion of the GAIL deposited activity was associated with loosely adherent particulate matter and/or oxide scale. This is not to say that GAIL was a "dirty" loop; a filter in the loop precluded gasborne dust to a large extent. However, diffusion probe studies done at GAIL showed that some activity was associated with particles larger than atoms or diatomic particles (Ref. 5-23). Cerium tends to form a very stable carbide, which has a low volatility. It seems likely that cerium carbide particles were formed in the fuel and then entrained in the coolant by some means of erosion rather than by evaporation. These particles, when deposited, would be less adherent than a chemisorbed atom or molecule, which would explain their propensity for liftoff.

Fig. 5-17. Liftoff versus shear force ratio

The cerium liftoff correlation was used to represent dust liftoff for all nuclides based on reasons given above. Different interpretations of the data could lead to different dust liftoff correlations.

Having obtained the distribution of nuclides in the primary circuit, it is necessary to calculate the maximum shear force experienced so that the total amount of liftoff may be calculated. This is done in the RATSAM computer code, which is used for depressurization analysis (Ref. 5-22). This code yields as output the flow rates and shear force ratios experienced by regions in the primary circuit as functions of time for a given size hole and location. The maximum shear force ratio experienced in a plateout region is used to compute the liftoff in that region.

For this study, a 100-sq-in. rupture of the PCRV in the hot leg is postulated in the DBDA. For the 1-sq-in. hole considered in the slow depressurization case, the exit flow rate is not large enough to cause significant liftoff of plateout activity.

Mass balance equations for each fission product isotope can be written separately for gasborne inventories, dustborne inventories, and plateout inventories. These equations are:

$$\begin{aligned} \frac{dN_{g,i}}{dt} = & (R/B)_i B_i + \lambda_{i-1} X_{1,i} N_{g,i-1} + \lambda_{i-2} X_{2,i} N_{g,i-2} \\ & - (\lambda_i + r_s + r_p + r_d) N_{g,i} \quad , \end{aligned} \quad (5-9)$$

$$\begin{aligned} \frac{dN_{d,i}}{dt} = & r_d N_{g,i} + \lambda_{i-1} X_{1,i} N_{d,i-1} + \lambda_{i-2} X_{2,i} N_{d,i-2} \\ & - (\lambda_i + r_s) N_{d,i} \quad , \end{aligned} \quad (5-10)$$

$$\begin{aligned} \frac{dN_{p,i}}{dt} = & r_p N_{g,i} + \lambda_{i-1} X_{1,i} N_{p,i-1} + \lambda_{i-2} X_{2,i} N_{p,i-2} \\ & - \lambda_i N_{p,i} \quad , \end{aligned} \quad (5-11)$$

where $N_{g,i}$ = number of gasborne atoms of the i th isotope,

$N_{d,i}$ = number of dustborne atoms of the i th isotope,

$N_{p,i}$ = number of plated out atoms of the i th isotope,

λ_i = decay constant for i th isotope,

$(R/B)_i$ = release to birth ratio for i th isotope,

B_i = birth rate of i th isotope,

r_s = purification train removal rate,

r_p = removal rate for plateout on surfaces,

r_d = removal rate for plateout on dust, and

X_1, X_2 = precursor branching ratios, parent and grandparent.

These equations are solved to yield

$$\begin{aligned} N_{g,i} = & \frac{(R/B)_i B_i + \lambda_{i-1} X_{1,i} N_{g,i-1} + \lambda_{i-2} X_{2,i} N_{g,i-2}}{\lambda_{TOT}} \\ & \times \left(1 - e^{-\lambda_{TOT} t} \right) \quad , \end{aligned} \quad (5-12)$$

$$N_{d,i} = \frac{r_d N_{g,i} + \lambda_{i-1} X_{1,i} N_{d,i-1} + \lambda_{i-2} X_{2,i} N_{d,i-2}}{(\lambda_i + r_s)} \times \left(1 - e^{-(\lambda_i + r_s)t} \right), \quad (5-13)$$

$$N_{p,i} = \frac{r_p N_{g,i} + \lambda_{i-1} X_{1,i} N_{p,i-1} + \lambda_{i-2} X_{2,i} N_{p,i-2}}{(\lambda_i + r_p)} \times \left(1 - e^{-(\lambda_i + r_p)t} \right), \quad (5-14)$$

where $\lambda_{TOT} = \lambda_i + r_s + r_d + r_p$.

These equations give the number of atoms of each nuclide for the indicated locations; since the activity inventories are required, the above equations are multiplied on both sides by the decay constant of the i th nuclide to obtain the activities.

The equations are solved by treating the nuclides one by one in increasing order. Depending on whether or not the nuclide is first in a chain, X_1 and X_2 are either zero or represent the branching ratios from parent and grandparent nuclides. This calculation, considering two precursors, is accurate to within 1% for all chains and precursor activity.

5.2.3. Results

Containment and site specific parameters used to calculate doses for the depressurization events are given in Section 3.5, the radiological models. Tables 5-13 through 5-15 give the fraction of the total plateout

TABLE 5-13
PERCENTAGE OF PLATEOUT ACTIVITY EFFLUENT FROM PCRV FOR DBDA

Nuclide	Upper 95% Estimate	Median Estimate	Lower 95% Estimate
Rb-88	84	47	0.2
Rb-89	84.6	47.3	0.22
Rb-90m	88.2	49.4	0.22
Rb-90	88.6	49.6	0.23
Rb-91	89.2	49.9	0.22
Sr-90	0.02	0.05	0.13
Sr-91	28.4	15.9	0.22
Zr-95	0.26	0.23	0.23
Nb-95	0.17	0.17	0.21
Ag-110m	0.08	0.122	0.23
Te-129m	0.5	0.36	0.23
Te-131m	11.8	6.6	0.23
Te-132	4.9	2.8	0.23
I-131	1.9	1.3	0.5
I-132	12.4	7.1	0.5
I-133	15.6	8.9	0.5
I-135	36	20.3	0.5
Cs-134	0.04	0.01	0.23
Cs-137	0.02	0.005	0.01
Ba-137m	0.06	0.11	0.23
Ba/La-140	2	1.1	0.22
Ce/Pr-144	23.5	13.2	0.23
Sr-89	0.34	0.26	0.23
Y-91	0.16	0.18	0.23
Sb-125	0.031	0.093	0.23

TABLE 5-14
PERCENTAGE OF PLATEOUT ACTIVITY EFFLUENT FROM PCRV FOR SLOW
DEPRESSURIZATION

Nuclide	Upper 95% Estimate	Median Estimate	Lower 95% Estimate
Rb-88	84	47	0
Rb-89	84.6	47.2	
Rb-90m	88.2	49.3	
Rb-90	88.6	49.4	
Rb-91	89.2	49.9	
Sr-90	2.9×10^{-3}	1.66×10^{-3}	
Sr-91	28.4	15.9	
Zr-95	0.25	0.14	
Nb-95	0.16	0.1	
Ag-110m	0.065	0.04	
Te-129m	0.5	0.3	
Te-131m	11.8	6.5	
Te-132	4.8	2.7	
I-131	1.9	1.0	
I-132	12.4	6.9	
I-133	15.6	8.7	
I-135	36	20.1	
Cs-134	0.022	0.012	
Cs-137	2.9×10^{-3}	1.6×10^{-3}	
Ba-137m	0.045	0.025	
Ba/La-140	2	1.1	
Ce/Pr-144	23.5	13.2	
Sr-89	0.3	0.17	
Y-91	0.14	0.08	
Sb-125	0.016	9.1×10^{-3}	0

TABLE 5-15
FRACTION OF DBDA EFFLUENT ACTIVITY DUE TO LIFTOFF FOR MEDIAN AND UPPER
BOUND CASES (PERCENT)

Nuclide	Upper 95% Estimate	Median Estimate
Rb-88	0	0
Rb-89	0	0.2
Rb-90m	0	0.2
Rb-90	0	0.4
Rb-91	0	0
Sr-90	83.2	96.7
Sr-91	0	0
Zr-95	3.8	39
Nb-95	5.9	41.1
Ag-110m	19.7	67.2
Te-129m	0	16.6
Te-131m	0	1.5
Te-132	2	3.6
I-131	0	23
I-132	0	2.8
I-133	0	2.2
I-135	0	1
Cs-134	45	87.9
Cs-137	83.9	69.8
Ba-137m	25	77.3
Ba/La-140	0	0
Ce/Pr-144	0	0
Sr-89	11.8	34.6
Y-91	12.5	55.5
Sb-125	48.4	90.2

that is effluent for different values of fractional plateout on dust. It can be seen by comparing Tables 5-13 and 5-14 (DBDA and slow depressurization, respectively) that, for most significant dose contributors, the liftoff contribution to total effluent activity is small for the median and upper bound cases. This is illustrated in Table 5-15, where the percentage of effluent activity due to liftoff is given. The notable exception is Sr-90, the source of which is primarily from liftoff. This is because its long half-life causes the gasborne activity to be very small, (λN is the activity, and λ is very small) and thus the liftoff becomes relatively more significant. This behavior is also exhibited in the DBDA bone dose uncertainty factor, which is less than 1. This indicates that dose decreases as plateout percentage on dust increases, because there is less fixed surface plateout to be removed.

Figure 5-18 and Table 5-16 show the sensitivity of the doses to variation of the total plateout, plateout on dust, and gasborne inventory as these parameters were varied through their 95% confidence bands. The top three parameters affecting the thyroid dose were found to have a proportional influence on this dose. The sensitivity of the whole body dose to the input parameters was less than proportional.

Tables 5-17 and 5-18 give the composite and total uncertainties for the slow depressurization and the DBDA for each of the parameters (uncertainties are combined in a lognormal fashion) as well as the median doses for these events.

Figures 5-19 through 5-22 show the important isotopes for the median whole body gamma and beta, thyroid, and bone doses for both depressurizations at the EAB and LPZB. The contributors for the two events are nearly the same, except that the bone dose is due primarily to Sr-90 in the DBDA, whereas Ce-144 is the main source of bone dose in the slow depressurization. This is because Sr-90 is obtained mostly from liftoff in DBDA, and

Fig. 5-18. Dose variation due to gasborne and total plateout variation for DBDA and slow depressurization

TABLE 5-16
INFLUENCE COEFFICIENTS FOR DBDA 30-DAY LBZB WHOLE BODY $\gamma + \beta$ AND THYROID DOSES

Rank	Parameter	Parameter Ratio $P_{\max}/\bar{P}$	Dose Ratio $D_{\max}/\bar{D}$	Influence Coefficient (a)
A. Inhalation Thyroid Dose				
1	% plateout on dust	1.8	1.82	1.0
2	Total plateout inventory	1.58	1.59	1.0
3	Containment plateout and fallout	10	10	1.0
4	Decay and buildup enroute	0	1.02	-0.02
5	Gasborne activity	1.56	$1 + \epsilon$	ϵ
B. Whole Body $\gamma + \beta$ Dose				
1	Gasborne activity	1.56	1.36	0.64
2	Total plateout inventory	1.58	1.24	0.41
3	% plateout on dust	1.8	1.05	0.06
4	Decay and buildup enroute	0	1.004	-0.004
5	Containment plateout and fallout	10	1.0	0

(a) Influence coefficient $\approx \left(\frac{D_{\max}}{\bar{D}} - 1 \right) / \left(\frac{P_{\max}}{\bar{P}} - 1 \right)$ where D = dose and P = parameter value.

TABLE 5-17
SUMMARY OF PARAMETER UNCERTAINTY FACTORS FOR SLOW DEPRESSURIZATION EVENT

	2-Hr 500-m EAB Doses			30-Day 2500-m LPZ Doses		
	Whole Body $\gamma + \beta$	Inhalation Thyroid	Inhalation Bone	Whole Body $\gamma + \beta$	Inhalation Thyroid	Inhalation Bone
Median dose (rem)	1.2×10^{-5}	1.8×10^{-4}	1.5×10^{-6}	3.1×10^{-6}	7.3×10^{-6}	6.0×10^{-8}
<u>Fission Product Parameters</u>						
% plateout on dust	1.10	1.98	1.97	0.81	1.99	1.97
Total plateout inventory	1.16	1.59	21.1	1.19	1.6	21.3
Gasborne inventory	1.46	1.0	1.0	1.39	1.0	1.0
Containment fallout and plateout	1.00	10.0	10.0	1.0	10.0	10.0
Decay and buildup enroute	1.004	1.01	1.00	1.004	1.02	1.01
<u>Building and Meteorological Parameters</u>						
Containment leak rate	5.00	5.00	5.00	5.00	5.00	5.00
Cleanup system filter efficiency	1.00	1.01	1.00	1.00	1.04	1.01
Cleanup system recirculation rate	1.04	1.13	1.13	1.10	1.83	1.90
Meteorology atmospheric dispersion factor	14	14	14	16	16	16
release height	0.195	0.195	0.195	0.96	0.96	0.96
$UF = \exp \left[\sum_p (\ln f)^2 \right]^{1/2}$						
Overall uncertainty factors	34	72	186	26	60	161

TABLE 5-18
SUMMARY OF PARAMETER UNCERTAINTY FACTORS FOR RAPID DEPRESSURIZATION (DBDA)

	2-Hr 500-m EAB Doses			30-Day 2500-m LPZ Doses		
	Whole Body $\gamma + \beta$	Inhalation Thyroid	Inhalation Bone	Whole Body $\gamma + \beta$	Inhalation Thyroid	Inhalation Bone
Median dose (rem)	1.6×10^{-5}	2.4×10^{-4}	5.7×10^{-6}	3.1×10^{-6}	9×10^{-6}	2.2×10^{-7}
<u>Fission Product Parameters</u>						
% plateout on dust	1.31	1.77	0.77	1.05	1.82	0.77
Total plateout inventory	1.26	1.59	22.4	1.24	1.59	22.3
Gasborne inventory	1.45	1.00	1.00	1.36	1.00	1.00
Containment fallout and plateout	1.00	10.0	10.0	1.0	10.0	10.0
Decay and buildup enroute	1.004	1.01	1.00	1.004	1.02	1.01
<u>Building and Meteorological Parameters</u>						
Containment leak rate	5.00	5.00	5.00	5.00	5.00	5.00
Cleanup system filter efficiency	1.00	1.01	1.00	1.00	1.04	1.01
Cleanup system recirculation rate	1.04	1.13	1.13	1.10	1.83	1.9
Meteorology atmospheric dispersion factor	14	14	14	16	16	16
Release height	0.195	0.195	0.195	0.96	0.96	0.96
$UF = \exp \left[\sum_p (\ln f)^2 \right]^{1/2}$						
Overall uncertainty factors	34	70	186	25	59	159

WHOLE BODY GAMMA (60 % OF TOTAL)

WHOLE BODY BETA (40% OF TOTAL)

INHALATION
THYROID

INHALATION
BONE

Fig. 5-19. DBDA 2-hr EAB median dose contributors

Fig. 5-20. DBDA 30-day LPZB median dose contributors

Fig. 5-21. Slow depressurization 2-hr EAB median dose contributors

Fig. 5-22. Slow depressurization 30-day LPZB median dose contributors

there is no liftoff in the slow depressurization. It is noted that 40-year plateout values were needed to calculate doses; if 20-year values are desired, then the only effect will be to decrease the importance of Sr-90 in a DBDA. In this case, its contribution to total dose will be approximately 50% as opposed to 64% for 40-year plateout.

Figure 5-23 graphically illustrates the comparison of median dose levels and uncertainties with PSAR doses and licensing guidelines. It can be seen that PSAR doses are conservative relative to the 95% confidence limits, except in one case.

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Fig. 5-23. Summary of radiological dose consequence and uncertainties compared to regulatory guidelines for depressurization events

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6. SAFETY R&D SIGNIFICANCE OF FISSION PRODUCT DATA

The results of the preceding event consequence analyses can be consolidated for the purpose of identifying key fission products and transport mechanisms which have the highest contribution to consequence uncertainties in the events analyzed. The approach uses the formulations of health effect consequence factors presented in Section 3.4. In this approach, one first calculates separately for the whole body thyroid, bone, and skin doses (30-day LPZB) the health effect factor for each parameter (Eq. 3-6) in a given event. One then sums the health effects of the spectrum of doses to calculate F_T , as shown in Eq. 3-7. The parameter's indicative factor of safety significance for a given event is F_T . The overall significance of the parameter is obtained by summing F_T over all events that are affected by the parameter.

In this summation, the probability of occurrence for each event is used as a weighting factor, as shown in Eq. 3-8. These overall safety significance values are then ranked, parameter by parameter, to determine the relatively most significant fission product release and transport mechanisms affecting dose consequences. A summary of the calculations of the indicative factors of safety significance for the different events is presented in Tables 6-1 through 6-5.

The indicative factors F_T in Tables 6-1 through 6-5 range in magnitude up to 9×10^{-4} (for fuel particle release rate during core heatup). In fact, the first six ranked parameters in the core heatup event have higher F_T values than any parameter in other events. Just considering consequences without regarding the probability of occurrence, it is concluded that relative uncertainties in core heatup parameters outweigh parameter uncertainties in other events with regard to importance to health effects in the EPA model.

TABLE 6-1
INDICATIVE FACTORS OF FISSION PRODUCT PARAMETER SAFETY R&D
RANKING FOR REHEATER TUBE LEAK EVENT

			30-Day LPZB Doses			
			Whole Body γ	Thyroid	Bone	Skin
Median dose (rem) $\times 10^6$			0.75	0.75	0.75	1.3
Man-rem dose $\times 10^4$			4.0	4.0	4.0	6.9
Health effect $\times 10^8$			16.0	2.2	1.3	0.21
Total uncertainty factor			42	42	42	42
Rank	Parameter	$F_T \times 10^{10}$	% Contribution to F_T			
1	Gasborne activity	23.0	80.8	11.5	6.4	1.3
2	Dustborne activity	3.8	80.8	11.5	6.4	1.3
3	Decay and buildup enroute	0.19	80.6	11.4	6.6	1.4

TABLE 6-2
INDICATIVE FACTORS OF FISSION PRODUCT PARAMETER SAFETY R&D RANKING FOR
CORE HEATUP EVENT

			30-Day LPZB Doses			
			Whole Body γ	Thyroid	Bone	Skin
Median dose (rem) $\times 10^6$			100,000	102,000	100,000	217,000
Man-rem dose $\times 10^4$			394,000	402,000	394,000	855,000
Health effect $\times 10^8$			1,580,000	225,000	126,000	25,600
Total uncertainty factor			25	27	25	25

Rank	Parameter	$F_T \times 10^{10}$	% Contribution to F_T			
1	Fuel release rate	1,780,000	92.7	5.1	0.7	1.5
2	PCRV plateout	895,000	4.0	94.2	1.7	0.1
3	Decay heat	860,000	82.6	10.3	5.8	1.3
4	Containment plateout and fallout	51,100	neg.	100	neg.	neg.
5	Fuel failure vs T	24,900	82.5	10.4	5.8	1.3
6	Fuel body activity	2,780	82.4	10.5	5.8	1.3
7	Decay and buildup enroute	440	82.4	10.4	5.9	1.3

TABLE 6-3
INDICATIVE FACTORS OF FISSION PRODUCT AND CORE CHEMISTRY PARAMETER SAFETY
R&D RANKING FOR STEAM INGRESS EVENT

			30-Day LPZB Doses			
			Whole Body γ	Thyroid	Bone	Skin
Median dose (rem) $\times 10^6$			4.2	4.2	4.2	8.7
Man-rem dose $\times 10^4$			16.5	16.5	16.5	34.3
Health effect $\times 10^8$			66	9.2	5.3	1.03
Total uncertainty factor			29	29	29	29
Rank	Parameter	$F_T \times 10^{10}$	% Contribution to F_T			
1	Fuel failure fraction	780	79.8	11.1	6.3	2.8
2	Hydrolysis rate vs T	70	80.7	11.2	6.4	1.7
3	Hydrolysis rate vs P_{st}	64	80.1	10.9	6.2	2.8
4	Transverse flow factor	44	79.5	11.2	6.4	2.8
5	Moles steam per mole UC_2	41	80.7	10.4	6.0	2.9
6	Steam diffusion coeff.	38	80.7	11.2	6.4	1.7
7	Noble gas release in hydrolysis	6	81.3	10.0	5.8	2.9
8	Steam-graphite reaction rate	4	80.2	10.8	6.2	2.8
9	Gasborne activity	0.6	82.4	9.4	5.4	2.8
10	Catalytic effect on parameter 8	0.5	79.6	11.2	6.4	2.8
11	Fuel body Xe and Kr activity	0.2	77.1	12.8	7.3	2.8
12	Burnoff effect on parameter 8	0.1	76.4	13.2	7.6	2.8
13	Containment plateout and fallout	0.002	neg.	63.6	36.4	neg.
14	Distribution coeff. of catalysts	0.001	79.8	11.0	6.4	2.8

TABLE 6-4
INDICATIVE FACTORS OF FISSION PRODUCT PARAMETER SAFETY R&D RANKING FOR
SLOW DEPRESSURIZATION EVENT

			30-Day LPZB Doses			
			Whole Body γ	Thyroid	Bone	Skin
Median dose (rem) $\times 10^6$			1.6	8.9	1.7	3.1
Man-rem dose $\times 10^4$			6.3	35.1	6.7	12.2
Health effect $\times 10^8$			25.2	19.6	2.1	0.37
Total uncertainty factor			25	52	28	25

Rank	Parameter	$F_T \times 10^{10}$	% Contribution to F_T			
1	Containment plateout and fallout	360	neg.	99.8	0.2	neg.
2	% plateout on dust	135	29.4	62.4	7.7	0.5
3	Initial plateout inventory	82	17.2	63.2	19.3	0.3
4	Gasborne activity	46	89.0	2.4	6.9	1.7
5	Decay and buildup enroute	0.1	neg.	100	neg.	neg.

TABLE 6-5
INDICATIVE FACTORS OF FISSION PRODUCT PARAMETER SAFETY R&D RANKING FOR
DBDA EVENT

			30-Day LPZB Doses			
			Whole Body γ	Thyroid	Bone	Skin
Median dose (rem) $\times 10^6$			2	11	2	3
Man-rem dose $\times 10^4$			7.9	43.3	7.9	11.8
Health effect $\times 10^8$			31.5	24	2.5	0.35
Total uncertainty factor			26	52	33	26

Rank	Parameter	$F_T \times 10^{10}$	% Contribution to F_T			
1	Containment plateout and fallout	1010	neg.	99.3	0.7	neg.
2	Total plateout inventory	78	0.2	60.9	38.9	neg.
3	% plateout on dust	76	1.3	98.5	0.2	neg.
4	Gasborne inventory	43	89.1	2.3	7.0	1.6
5	Decay and buildup enroute	0.12	neg.	96.9	3.1	neg.

The final safety R&D significance index for fission product parameters in comparison with other parameters results from summing the F_T values times the probability of occurrence for each event. Table 6-1 presents these results. Parameters are ordered from highest values of $\sum F_T P$ to lowest, which is the final ranking of the safety significance of these fission product parameters and satisfies the third objective of Section 2.2. Contributions of the events to the total parameter safety significance are shown in the final columns.

These results show that the three top parameters indicated by the criteria are: (1) fuel particle release rate during core heatup, (2) transient plateout in the PCRV of fission products released from the core, and (3) decay heat generation rate. The importance of these parameters derives solely from their effect on the core heatup event. Further, the indicative safety index factor for fuel release rate is almost double that for PCRV plateout and decay heat generation rate, which have nearly equal safety index factors.

After the top three parameters, the safety index factors drop off significantly. The fourth ranked parameter, containment plateout and fallout, is important both for core heatup and slow depressurization. Next in the ranking is the initial gasborne inventory, which is important mainly in the reheater leak and also in slow depressurization. The sixth ranked parameter is the fuel particle coating failure model used to predict steady-state distributions of UC_2 exposed to hydrolysis during steam ingress and also used to predict transient coating failure during core heatup. Ranked seventh is percent plateout on dust, which affects both the initial gasborne activity and the initial activity plated out on metallic surfaces. The former effect is more important because of the high safety significance of gasborne activity. Pertinent events where this is important are reheater leak, slow depressurization, and DBDA.

The eighth and ninth ranked parameters are initial fuel body and plateout inventories, respectively. The latter is important primarily due to the thyroid and bone dose contributions of a slow depressurization event. For the former, the importance stems from the whole body contribution during the core heatup event. Similarly the importance of decay and buildup enroute stems from its contribution to the RHL whole body dose for the tenth ranked parameter.

Hydrolysis of failed UC_2 TRISO particles during steam ingress is ranked eleventh. A breakdown of the various hydrolysis parameters and their contribution to the overall hydrolysis safety index is presented in Table 6-6. It is shown that the rate of fuel-steam reaction as a function of the controlling variables is the largest contributor.

The health effect to the lung has not been analyzed in detail in this study but can be estimated from the total lung dose (whole body penetrating gamma plus inhalation lung doses calculated by the TDAC code), the man-rem conversion factor, and the EPA health effect factor for lung dose (H.E. - 5×10^{-5} from Ref. 6-1). The inhalation lung dose is calculated in a manner similar to that for the inhalation thyroid and bone doses as described in Section 3.5. From subsequent dose calculations, the lung dose health effect was estimated as approximately 90% of that of the thyroid doses for the core heatup, the reheater tube leak, and the steam ingress events, and about 10 to 20% of that of the thyroid doses for the DBDA and slow depressurization events.

To accurately establish the effect of lung dose on fission product parameter ranking, a sensitivity study on the lung dose as a function of parameter variation is required. An alternative analysis was used, determining the isotope contributions to the lung dose and applying the dose sensitivity for the established doses which best simulates the lung dose contributions. For example, in the core heatup analysis, the inhalation lung dose for median conditions is predominantly due to tellurium, iodine, and antimony isotopes. Therefore, in general, the sensitivity of the lung

TABLE 6-6
OVERALL RANKING OF FISSION PRODUCT AND CORE CHEMISTRY PARAMETERS THAT CONTRIBUTE MOST TO RISK
CONSEQUENCE UNCERTAINTIES OF EVENTS ANALYZED

Rank	Parameter	Index ^(a)	Contribution of Event (%) ^(b)				
			RHL	Core Heatup	Steam Ingress	DBDA	Slow Depress.
1	Fuel particle release rate	17,800		100			
2	Plateout in PCRV	8,950		100			
3	Decay heat generation rate	8,600		100			
4	Containment plateout and fallout	600		87.7		0.2	12.1
5	Gasborne inventory	360	94.3				5.7
6	Fuel failure model	270					
	a. Steady-state value				7.8		
	b. Temperature transient			92.2			
7	Percent plateout on dust	98	48.5			0.2	51.3
8	Initial fuel body inventory	29		99.9	0.1		
9	Initial plateout inventory	20				0.4	99.6
10	Decay and buildup enroute	11	98.4				1.6
11	Hydrolysis of UC ₂	5					
	a. Reaction rate vs temp				39		
	b. Reaction rate vs steam pressure				36		
	c. Moles H ₂ O reacting per mole UC ₂				23		
	d. Noble gas release				3		
12	Steam graphite reactions	2					
	a. Diffusion of steam				52		
	b. Transverse flow				43		
	c. Reaction rate equation				4.0		
	d. Catalytic effects				0.6		
	e. Burnoff effects				0.1		

(a) Index is defined as $10^{14} \sum_{\text{events}} F_{\text{T}} P.$

(b) Contribution is negligible if no percentage is listed.

dose was taken to be the same as for the thyroid dose, since 80% of the lung dose is due to isotopes that are treated similarly to those contributing to the thyroid dose.

Using this method for estimating the lung dose uncertainty and the lung dose health effect established in the previous paragraph, the effect on parameter ranking was evaluated. For the PCRV plateout and the containment fallout and plateout parameters, the numerical importance factors of Table 6-6 would increase less than a factor of 2. The remaining parameter index of importance would increase by 10 to 20%, but in no case would the ranking of the parameters change. The lung dose contribution, as well as the dose contributions for thyroid, bone, and skin doses, to the fission product parameter ranking and to overall consequences is generally small compared to the whole body penetrating gamma contributions. Therefore, the whole body gamma dose plays a major role in the HTGR safety evaluation.

Table 6-7 summarizes the ten top-ranked parameters and the important contributing kinds of doses as inferred from Tables 6-1 through 6-5. Also shown in Table 6-7 are the isotopes that are key contributors to these important doses as inferred from the results in Sections 4 and 5. Recommendations for future testing and data correlations, the parameters, and the isotope releases to be measured follow the priority ranking of Table 6-6.

REFERENCES

- 6-1. "Environmental Analysis of the Uranium Fuel Cycle, Part I - Fuel Supply, Part II - Nuclear Power Reactors, Part III - Nuclear Fuel Reprocessing," U.S. Environmental Protection Agency Report EPA-520/9-73-003-B, C, D, October 1973.

TABLE 6-7
IDENTIFICATION OF RELATIVELY HIGHEST DOSES AND CONTRIBUTING ISOTOPES FOR
HIGHEST RANKED PARAMETERS

Rank	Parameter (Events)	Relatively Highest Dose(s)	Key Isotopes
1	Fuel release rate (core heatup)	30-day LPZB whole body γ	Xe-133 Xe-135
2	PCRV plateout (core heatup)	30-day LPZB thyroid	I-131 I-133
3	Decay heat (core heatup)	30-day LPZB whole body γ	--
4	Containment plateout and fallout (core heatup, slow depressurization)	30-day LPZB thyroid	I-131 I-133
5	Fuel failure model (core heatup, steam ingress)	30-day LPZB whole body γ	--
6	Gasborne activity (RHL, slow depressurization)	2-hr EAB whole body γ	Kr-88 Kr-87
7	Percent plateout on dust (RHL, depressurizations)	2-hr EAB whole body γ	Kr-88 Kr-87
8	Initial fuel body inventory (core heatup, steam ingress)	30-day LPZB whole body γ	Xe-133 Xe-135
9	Initial plateout inventory (depressurizations)	2-hr EAB and 30-day LPZB thyroid and bone	I-133 I-131 Ce-144
10	Hydrolysis of UC ₂ (steam ingress)	30-day LPZB whole body γ	Xe-133 Kr-88

7. CONCLUSIONS AND EVALUATION OF RESULTS

7.1. RESULTS FOR RISK ASSESSMENT AND DESIGN OPTION STUDIES

Two primary objectives of the present study, as described in Section 2.2, were to provide updated* consequences for the selected relatively high risk events (Volume IV) and for the selected relatively low risk events. This subsection summarizes the results and conclusions attained in satisfaction of these objectives.

In comparison with the preliminary AIPA risk assessments of Volume III, the following results have been obtained by the present analyses:

1. The reheater tube leak consequences have been updated for the relatively highest risk sequence (Branch LA) in Table 4-3, and the consequences of the other branches of the reheater tube leak event have been updated in Table 4-5.
2. The core heatup consequences have been updated for the relatively highest risk LOSP sequence (Branch BD) in Tables 4-11 and 4-12, and the updated consequences of the other LOSP and EQ event branches are shown in Table 4-15.
3. The steam ingress occurrence probability and consequences have been updated for the relatively highest risk sequence (Branch AC) in Table 5-6; the consequences have been updated and flammability has been considered for Branches AE and AG; and the logic tree has been updated in Fig. 5-16.

*"Updating" is defined as the calculation of statistically median values and the associated uncertainties using the methods of analysis described in Section 3 of this report.

4. Depressurization consequences for the relatively highest risk sequences for the DBDA event (Branch A) and slow depressurization event (Branch A) are updated in Tables 5-17 and 5-18.

A graphical comparison of dose consequences and uncertainties obtained in this analysis with the preliminary consequences is presented in Fig. 7-1. For the reheater leak, the updated whole body dose is lower by about one order magnitude for the 2-hr EAB dose and 2 orders of magnitude for the 30-day LPZB dose. The updated core heatup 30-day LPZB whole body dose for Branch BD of LOSP, for example, decreased by less than an order of magnitude.

For selected relatively low risk events, the updated risk of steam ingress events decreased due to about an order of magnitude decrease each in occurrence probability and consequence. The reason for the lower probability is a more realistic assessment of the probability of moisture monitor failure. Updated consequences of depressurization events showed little change in whole body doses and generally slightly lower thyroid doses. Generally, the preliminary consequence results were within the uncertainty band of the updated results.

7.2. RESULTS FOR KEY FISSION PRODUCTS AND TRANSPORT MECHANISMS

The third objective of this study was to identify key fission product isotopes and transport mechanisms that contribute most to the event consequences and uncertainties. In satisfaction of this objective, Tables 7-1 and 7-2 summarize event consequences in terms of the median doses, total uncertainty factors for these doses, and the three major isotopes and fission product transport mechanisms in each case. Ranking of the isotopes follows their relative contribution to the median dose in each case, while ranking of the transport mechanisms follows the respective uncertainty factors. It is recognized that the uncertainty factor for each release mechanism parameter is the product of the sensitivity of the resultant

Fig. 7-1. Comparison of updated whole body and thyroid doses and uncertainty bands with preliminary (Volume III) results

TABLE 7-1
SUMMARY OF CONSEQUENCES AND UNCERTAINTIES FOR RELATIVELY HIGH RISK EVENT SHOWING KEY ISOTOPES AND TRANSPORT MECHANISMS

Parameter	2-Hr EAB Dose			30-Day LPZ Dose		
	Whole Body $\gamma + \beta$	Inhalation		Whole Body $\gamma + \beta$	Inhalation	
		Thyroid	Bone		Thyroid	Bone
<u>Core Heatup</u>						
Median dose (rem)	6.5×10^{-3}	9.3×10^{-4}	6.5×10^{-6}	2.2×10^{-1}	1.6×10^{-3}	8.8×10^{-6}
Uncertainty factor	36	583	2920	25	436	2015
Major isotope contributors	{ 1. Xe-133 2. Xe-135 3. Kr-85	I-131 I-133 I-132	Te-132 Sb-125 Te-129m	Xe-133 Xe-135 Kr-85	I-131 I-133 I-132	Te-132 Sb-125 Te-129m
Major fission product transport parameters	{ 1. Decay heat 2. PCRV plateout ^(a) 3. Fuel failure	PCRV plateout Cont. plateout Fuel release		Fuel release Decay heat PCRV plateout ^(a)	PCRV plateout Cont. plateout Fuel release rate	
<u>Reheater</u>						
Median dose (rem)	2.2×10^{-5}	neg.	neg.	1.4×10^{-6}	neg.	neg.
Uncertainty factor	21			42		
Major isotope contributors	{ 1. Kr-88 2. Kr-87 3. Rb-88			Kr-88 Rb-88 Kr-87		
Major fission product transport parameters	{ 1. Gasborne inv. 2. Dust activity 3. Decay enroute			Gasborne inv. Dust activity Decay enroute		

(a) Primarily plateout of Rb-88, which is produced from decay of Kr-88.

TABLE 7-2
SUMMARY OF CONSEQUENCES AND UNCERTAINTIES FOR SELECTED DESIGN BASIS EVENTS SHOWING KEY ISOTOPES
AND TRANSPORT MECHANISMS

Parameter	2-Hr EAB Dose			30-Day LPZB Dose		
	Whole Body γ + β	Inhalation		Whole Body γ + β	Inhalation	
		Thyroid	Bone		Thyroid	Bone
<u>Steam Ingress</u>						
Median dose (rem)	3.1 x 10 ⁻⁵	1.5 x 10 ⁻⁸	3.4 x 10 ⁻⁸	8.7 x 10 ⁻⁶	5.4 x 10 ⁻¹⁰	1.5 x 10 ⁻⁹
Uncertainty factor	37	68	74	29	56	61
Major isotope contributors	{ 1. Xe-133 2. Kr-88 3. Rb-88	I-133 I-131 I-135	Sr-89 Sr-90 Sr-91	Xe-133 Rb-88 Kr-88	I-133 I-131 I-135	Sr-89 Sr-90 Sr-91
Major fission product transport parameters	{ 1. Failed fuel 2. Hydrolysis 3. Diffusion	Cont. plateout Gasborne inv. Decay enroute	Failed fuel Hydrolysis	Failed fuel Hydrolysis Diffusion	Cont. plateout Gasborne inv. Decay enroute	Cont. plateout Hydrolysis Failed fuel
<u>DBDA</u>						
Median dose (rem)	1.6 x 10 ⁻⁵	2.4 x 10 ⁻⁴	5.7 x 10 ⁻⁶	3.2 x 10 ⁻⁶	9.0 x 10 ⁻⁶	2.2 x 10 ⁻⁷
Uncertainty factor	34	70	186	25	59	159
Major isotope contributors	{ 1. Xe-133 2. Rb-88 3. Kr-88	I-133 I-131 I-135	Sr-90 Ce-144 Te-132	Xe-133 Rb-88 Kr-88	I-133 I-131 I-135	Sr-90 Ce-144 Te-132
Major fission product transport parameters	{ 1. Gasborne inv. 2. Dust 3. Plateout inv.	Cont. plateout Dust Plateout inv.	Plateout inv. Cont. plateout Dust	Gas activity Plateout inv. Dust	Cont. plateout Dust Plateout inv.	Plateout inv. Cont. plateout Dust
<u>Slow Depressurization</u>						
Median dose (rem)	1.2 x 10 ⁻⁵	1.8 x 10 ⁻⁴	1.5 x 10 ⁻⁶	3.1 x 10 ⁻⁶	7.3 x 10 ⁻⁶	6.0 x 10 ⁻⁸
Uncertainty factor	34	72	186	26	60	161
Major isotope contributors	{ 1. Xe-133 2. Kr-88 3. Rb-88	I-133 I-131 I-134	Ce-144 Te-132 Sr-89	Xe-133 Rb-88 Kr-88	I-131 I-133 I-135	Ce-144 Te-132 Sr-89
Major fission product transport parameters	{ 1. Gas activity 2. Dust 3. Plateout inv.	Cont. plateout Dust Plateout inv.	Plateout inv. Cont. plateout Dust	Gas activity Dust Plateout inv.	Cont. plateout Dust Plateout inv.	Plateout inv. Cont. plateout Dust

dose to the parameter times the range of uncertainty in the parameter value. Thus, the uncertainty factors do not measure relative contribution of the release mechanisms to the median dose but rather how the dose can deviate from the median as the release mechanism parameter varies through its $\pm 95\%$ confidence range. Relative contribution of each release mechanism to the median dose can be inferred from the influence coefficient tables given in Sections 4 and 5.

Results in Tables 7-1 and 7-2 can be assessed in connection with the summary in Table 7-3 of major contributing activity sources (initial fuel body or plateout or gasborne activities, etc.) to the radiological dose of each event. For the core heatup event, release of initial gasborne activity is small compared to fuel body activity release, even for the 2-hr dose. This is due partially to the fact that the gasborne activity is not released until the PCRV relief valve lifts at 45 min. By definition, the 2-hr dose is counted from the time of release to the containment. Thus, at 2-3/4 hr past event initiation, a small but significant fraction of fuel failure is predicted with attendant fission product release. For release of fuel body activity, the volatile noble gases and chiefly Xe-133 (to a much lesser extent Xe-135 and Kr-85), which has a 5.3-day half-life combined with a high fission yield, are most significant. Noble gases do not plate out or decay significantly in graphite; therefore, the important transport mechanisms are rate of release from the fuel particles and fuel particle coating failure model. Contributions to the thyroid and bone doses are controlled mainly by PCRV and containment plateout and fuel release rate, in that order. Holdup of volatile metallic fission products in graphite is found to be relatively unimportant within the context of the health effects model employed.

For the reheater tube leak, the important dose is the whole body dose resulting from direct release of gasborne activity to the atmosphere. Thus, the isotopes dominant in the gasborne inventory, chiefly Kr-88 and Kr-87, also are key contributors to the dose; Rb-88, the daughter of Kr-88, is important mostly for the beta dose.

TABLE 7-3
SUMMARY OF MAJOR CONTRIBUTIONS TO EVENT CURIE RELEASE TO CONTAINMENT

Initial Gasborne Activity	Initial Fuel Body Activity	Initial Plateout Activity	Decay From Noble Gases
<u>Core Heatup</u> Released from PCRV; small effect on total dose at 2 hr and 30 days	Time-dependent release of volatiles (about 1/2 total activity) major source for all doses; fuel release rate; PCRV plateout and fuel failure dominant parameters	Negligible release	Small effect
<u>Steam Ingress</u> Released from PCRV; major source for thyroid doses; important for whole body doses; small effect on bone doses	Release of noble gases due to hydrolysis of failed fuel; important for whole body doses; amount failed fuel, hydrolysis rate, and steam diffusion dominant parameters.	Negligible release	Decay of Kr-89, Kr-90, and Kr-91 to corresponding strontiums major source of bone dose, but total bone dose small.
<u>DBDA</u> Released from PCRV; major source for whole body doses; small effect on thyroid and bone doses; initial gas inventory and large uncertainty in Rb-88 contribution dominant parameters	Negligible release	Liftoff from fixed surfaces major source for thyroid and bone doses; blowdown of dust activity minor; containment plateout, initial plateout inventory, and partition of plateout on dust and surfaces dominant parameters	Small effect
<u>Slow Depressurization</u> Released from PCRV; major source for whole body doses; small effect for thyroid and bone doses; initial gas inventory and large uncertainty in Rb-88 contribution dominant parameters	Negligible release	Blowdown of activity on dust major source for thyroid and bone doses; initial plateout inventory, containment plateout, and partition of plateout on dust and surfaces dominant parameters	Small effect
<u>Reheater</u> < 0.1% released directly to atmosphere; major source for whole body doses; initial gas inventory and percent plateout on dust dominant parameters	Negligible release	Negligible release	Major source for bone doses, but thyroid and bone doses negligible; small compared to whole body due to plateout of iodines and particulates

For the steam ingress event, there is essentially no contribution from plateout activity, and the sole fuel body activity release is of noble gases due to hydrolysis of UC_2 kernels with prior failed coatings. It is found that release due to hydrolysis is the largest contributor to the doses, outweighing initial gasborne activity release. Thus, the important fuel body noble gas, Xe-133, is the key isotope contributor. Secondary contributors are Kr-88 and Rb-88. Important release mechanisms are, in order, initial failed fuel fraction, hydrolysis reactions, and diffusion of steam in graphite, all of which affect the release due to hydrolysis. The most important dose is the 30-day whole body gamma dose at the LPZB.

Depressurization events are the only events analyzed where the thyroid dose, and to a lesser extent the bone dose, becomes important in the index of safety significance (Section 6). The major source of these doses is liftoff of plated-out fission products and blowdown of activity on dust. Inventories of gasborne iodines and bone seekers initially in the primary coolant are small in comparison. By far the most significant fission product parameter is plateout and fallout, principally of iodines, in the containment. Corresponding key isotopes are I-133, I-131, and Ce-144. Other important fission product mechanisms are the initial plateout inventory and the partition between plateout on dust and on metallic surfaces.