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VOLUME I
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HTGR ACCIDENT INITIATION AND PROGRESSION ANALYSIS STATUS REPORT

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VOLUME I INTRODUCTION AND SUMMARY

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MANAGEMENT OF AIPA STUDY

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FOREWORD

This report summarizes the progress on 189 number 189a 13822, "Accident Initiation and Progression Analysis - HTGR." The work was performed under Contract E(04-3)-167, Project Agreement No. 51. The major portion of the work was performed between July 1, 1974, and June 30, 1975. The work for Volume VII of the report was completed in December 1975.

As a consequence of the recent ERDA reorganization, this study now falls within the purview of the Division of Nuclear Research and Applications (NRA).

ABSTRACT

Probabilistic risk assessment techniques have been applied to obtain guidance in choosing nuclear safety research and development that is most worthwhile for high-temperature gas-cooled reactor (HTGR) nuclear power plants. The probabilistic techniques used are similar to those employed in the Reactor Safety Study for light water reactors (LWRs), WASH-1400, directed by Dr. N. C. Rasmussen. The recommendations for research include studies related to core heatup even though this event poses a very low risk to the public. In fact, it was found that under the many conditions covered by the study to date, even very infrequent accidents in HTGRs (say, once in ten million years) will not produce fatalities. Potential cost reduction areas have been found where alternate design options protect the public and meet regulatory safety criteria.

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1. INTRODUCTION

The Accident Initiation and Progression Analysis (AIPA) study was initiated in February 1974 under the sponsorship of the Reactor Research and Development Division of the Energy Research and Development Administration (ERDA) (formerly a part of the Atomic Energy Commission). The study has been conducted by General Atomic Company (GA) and reviewed by Aerojet Nuclear Company (ANC).

The primary objective of the AIPA study is to provide guidance for safety research and development programs for high-temperature gas-cooled reactors (HTGRs). Such R&D efforts can be directed at identifying the design changes required to enhance public safety if that is found to be necessary, or to reduce the uncertainties involved in the realistic assessments of risk where deemed appropriate. Another objective is to consider alternative design options related to HTGR safety. Such options can include economics as a variable when safety criteria are still met. Additional objectives are to develop the methodology of risk assessment and to encourage the utilization of the method by designers. A companion effort, funded as a separate task by ERDA, is development of a data bank for probabilistic information applicable to gas-cooled reactors. The greatest portion of the effort to date has been to provide guidance for safety R&D.

The AIPA study employs probabilistic methods in examining the safety aspects of HTGRs. The overall techniques used in the program were established by June 1974 (Ref. 1-1). That effort utilized the prior work of Mulvihill (Ref. 1-2), Farmer (Ref. 1-3), Cave and Holmes (Ref. 1-4), Joksimovic (Ref. 1-5), and Rasmussen (Ref. 1-6), among others, as well as the papers presented at the BNES Symposium of 1969 (Ref. 1-7). Since that time, the techniques have been improved and applied to the HTGR. The

status of the study is presented in this report. Most of the results contained herein were completed by June 30, 1975 (FY-75). Some preliminary results were presented at the American Nuclear Society Summer Meeting on June 8, 1975 (Ref. 1-8).

The AIPA study is similar to the Reactor Safety Study, WASH-1400 (Ref. 1-9). However, the starting points as well as the purposes of the two studies are different. This study so far has been directed at the generation of HTGRs which passed NRC and ACRS review via the Summit and Fulton applications, whereas WASH-1400 was directed at LWRs that were in operation. The WASH-1400 study was an assessment of the safety of LWRs, while the gas-cooled reactor study provides guidance to safety research and development. At this stage, the AIPA study has the advantage of being performed by a safety and reliability group closely associated with both the design and research organizations for the HTGR. Hence, it cannot be claimed to be a completely independent study, although every effort is being made by GA and ERDA to be as objective as possible. In accord with this desire, ERDA has contracted with ANC for continual review of the AIPA program. WASH-1400 was an independent assessment and could not directly utilize the capabilities of the LWR design organizations. WASH-1400 had to provide accident consequences in absolute terms, while the risk computed in the HTGR study may be either absolute or comparative and still be usable in providing research and development guidance. The funding of the HTGR effort has so far been moderate, compared to the WASH-1400 effort, to allow primary emphasis to be placed upon the main gas-cooled R&D safety programs.

A major difference between this status report and WASH-1400 is that WASH-1400 is a final report that covered a large number of accident sequences, whereas this report has dealt with a much smaller number of sequences and may not have achieved a complete risk assessment. The cases covered are representative and broad in scope, but analysis must be continued to achieve greater completeness. Further development of methodology and data base is also important, although significant advancements were made in the course of the present work.

The AIPA study reported herein is based on the 3000-MW(t) reference design of General Atomic Company as of the spring and summer of 1975. Any application of these results to other designs of gas-cooled reactors should be done with that fact in mind, but major portions of the results provide insights into the generic characteristics of gas-cooled, graphite-moderated reactors. It is interesting to note that work by L. Cave (Ref. 1-10), which was a relatively small-scale effort in comparison with the AIPA study, led to results in terms of risk assessments with regard to the overall relative safety standing of HTGRs which are similar to those found in this study.

This report covers many portions of the AIPA study; these portions and their basic relation to one another are shown in Fig. 1-1. The study originates in the AIPA program, shown at the left of the figure. The program has had two parts: basic AIPA and fission product technology. The basic AIPA work is oriented toward safety risk, broad R&D recommendations, and design options; fission product technology work is oriented toward the R&D significance of fission product transport mechanisms, parameters, and data. Figure 1-1 also shows which report volume contains each block of work. The introductory volume addresses both portions of AIPA.

The basic AIPA work developed the method that was applied in part to the preliminary analysis and in full to the detailed risk analysis. The choices of initiating events for the detailed risk analysis and the design option study were based on the results of the preliminary analysis. The detailed risk analysis obtained results from the detailed consequence analysis for relatively higher risk cases, and it supplied information that allowed R&D recommendations to be made.

The fission product technology work covered both the initiating events that were found in the preliminary analyses to be of interest from the risk point of view, and the initiating events that form the major design bases and areas of fission product technology for the HTGR, such as the design

Fig. 1-1. Relationship between volumes of the report

basis depressurization accident (DBDA). In support of this work, fission product inventories in the primary helium circuit were analyzed probabilistically. The results were used to evaluate the R&D significance of the fission product transport parameters. The work includes the effects on shielding and personnel exposure of the very small normal release of radioactivity from the core.

A brief description of each volume is given below:

Volume I, "Introduction and Summary," provides an introduction and a brief technical summary of all volumes except number VII, which is to be published subsequently.

Volume II, "AIPA Risk Assessment Methodology," expands on the previously reported (Ref. 1-1) analytical and display techniques utilized in the study.

Volume III, "Preliminary Results (Including Design Options)," presents the results of a preliminary screening of event sequences resulting from seventeen different initiating events and identifies those initiating events with sequences that form an envelope of the risks as assessed for all of the events considered. A preliminary design option study is also reported, which is related to an initiating event well within the risk envelope.

Volume IV, "Phase I Analyses and R&D Recommendations," contains the results of a more detailed analysis of the three initiating events that formed the risk envelope, as identified in the preliminary study. A derivation of safety R&D recommendations based on the probabilistic analyses is also reported in this volume.

Volume V, "AIPA Fission Product Source Terms," provides a study of the quantities of fission product inventories of the fuel, primary coolant, and plateout surfaces. This information has been used in Volumes VI and VII.

Volume VI, "Event Consequences and Uncertainties Demonstrating Safety R&D Importance of Fission Product Transport Mechanisms," contains derivations of radiation doses used in Volume IV and an assessment of the relative R&D significance of fission product transport mechanisms as related to accidental releases.

Volume VII, "Occupational Man-Rem Exposures from Gas-Borne and Plateout Activities," provides probabilistic information on the relative importance of fission products associated with the primary coolant circuit during normal operation of an HTGR and the radiological exposures predicted for plant operating and maintenance personnel during normal plant operation.

The results of this study are a part of an emerging technology rather than an established one. Continuous improvement is under way in the methodology, improvement that is aided by a flow of ideas to the study group. Therefore, feedback from the readers of this report would benefit future development.

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2. HISTORY AND MANAGEMENT OF AIPA STUDY

The planning guide for HTGR safety and safety-related research and development (ORNL-4968) was prepared by the Oak Ridge National Laboratory (ORNL) (Ref. 2-1) with major contributions from GA. This guide, which was developed for use by the Division of Reactor Research and Development (RRD) of ERDA, identified seven interrelated task areas as follows:

1. Systems and safety analysis
2. Fission product technology
3. Primary coolant technology
4. Seismic and vibration technology
5. Confinement components (concrete pressure vessel and containment structures)
6. Materials technology
7. Safety instrumentation

The major objective of task area 1 was to analyze accident sequences and their consequences. In performing this work, probabilistic safety analyses would be employed and used to guide other tasks. While reviewing initial drafts of the guide (the final version was issued in May 1974), RRD recognized that probabilistic safety analyses or risk assessments provide an excellent management tool to guide safety R&D programs. The guide also

established priority designations A, B, C, and D based on the then-current judgments and some supplementary definitions. A further effort to establish a rationale for distributing limited funds revealed that the risk assessment approach can be used effectively to rank R&D priorities.

Other major contributing factors were the initiation and government support of the Reactor Safety Study (WASH-1400) and the existence of probabilistically trained gas-reactor-safety staff at GA, ERDA, and ORNL.

Having accepted the risk assessment approach as a management tool for safety R&D programs, ERDA encouraged GA and ORNL to submit proposals, which were presented on February 11, 1974, at a meeting in Germantown. The meeting was attended also by representatives from the Nuclear Regulatory Commission (NRC), Los Alamos Scientific Laboratory (LASL), and Brookhaven National Laboratory (BNL). As a result of this meeting, the AIPA program was initiated. The GA work scope was defined comprising the examination of analytical techniques, including those used by ORNL, and identification of accident families.

The GA 3000-MW(t) plant reference design was selected for the study. Dr. Donald Emon, Chief, Gas-Cooled Reactor Safety Branch, RRD, accepted the responsibility for ERDA management of the effort. Dr. Vojin Joksimovic, Manager, Safety and Reliability Branch of the Plant Engineering Division, accepted management responsibility for GA's participation in the study. Mr. W. J. Houghton, Section Leader in the Safety and Reliability Branch, was appointed the task leader. Mr. P. Rubel of ORNL participated in the early evaluation of the study, and the laboratory has continued some selected probabilistic analyses.

The first progress meeting was held in San Diego on April 22, 1974. At this meeting, the RRD accepted GA's method of identifying accident families as well as sequence diagrams as the basic methods to be used in the study. General Atomic agreed to evaluate Kaman Sciences GO code for utilization in the study and, after careful evaluation of the code, concluded that the GA MONSIM models provide a more realistic simulation of HTGR

operations. GA also agreed to make a best effort to provide RRD with a rationale to demonstrate that the proposed iterative screening process for identifying initiating events has sufficient generality so that, in principle, every significant event can be taken into account. General Atomic also agreed to extend the scope of the AIPA study in FY-75 by inclusion of the so-called fission product add-on task, which would provide a rationale for establishing priorities in fission product transport R&D programs.

The review meeting for the end of FY-74 was held in Germantown on June 19-20, 1974. By this time, the fundamental methodology was developed and illustrated. The HTGR accident families were identified, and GA agreed to furnish RRD with a proposed method for recommending relatively high risk events for more detailed study. General Atomic accepted the responsibility for the portion of the fission product add-on task dealing with normal operation and maintenance situations, while ORNL accepted the responsibility for the portion of the task dealing with accident conditions.

In July 1974, the AIPA study continued, with RRD contracting with ANC to provide for independent review and assessment of GA's activities. Mr. R. R. Stiger assumed management responsibilities for the task, with Mr. M. E. Stewart as the principal investigator.

In October 1974, the FY-74 status report was issued. At the same time, GA accepted responsibility for the portion of the fission product add-on task dealing with accident conditions. Dr. V. Joksimovic appointed Mr. A. W. Barsell, Section Leader in the Safety and Reliability Branch, to be the fission product add-on task leader. The Core Performance Branch of the Fuels Division assumed responsibility for the portion of the task dealing with normal operation, with Mr. M. J. Haire acting as coordinator. The Shielding Section of the Plant Engineering Division assumed responsibility for the portion of the task dealing with shielding and maintenance considerations, with Mr. R. E. Sund acting as principal investigator.

The first progress meeting in FY-75 was held in Germantown on October 31, 1974, when GA presented preliminary risk assessments associated with some of the initiating events representing various accident families identified in June. It was concluded that consequences, in addition to being expressed in whole body rem, would also be calculated in man-rem at a later stage.

A mid-year review meeting was held in Germantown on February 13-14, 1975. Preliminary risk assessments associated with seventeen initiating events were summarized. From the seventeen events, three relatively high risk events were selected for a more detailed study: reheater leak, loss of offsite power, and earthquake. Important relatively low-risk event sequences were recommended for design option study. General Atomic agreed to review a new ANC format for displaying the selection of initiating events. A GA proposal was accepted by the RRD for continuation of the inclusion of dependent failures (common mode failures) and for initiation of a scoping study of containment atmosphere response.

At a progress meeting held in San Diego on April 18, 1975, it was concluded that radionuclide inventories assessed under the fission product add-on task would be used as a basis for detailed risk assessments associated with the three relatively high risk events. In addition, GA reported acceptance of the ANC fault-tree-type format for selection of initiating events.

The first publication in the open literature that resulted from the AIPA study was a paper by K. N. Fleming on common mode failure methodology (Ref. 2-2). The paper was presented at a reliability symposium in Pittsburgh in April 1975. It was followed by an invited paper coauthored by W. J. Houghton, V. Joksimovic, and D. E. Emon, "Methods of Probabilistic Safety Analyses for Gas-Cooled Reactors" (Ref. 2-3), presented at the ANS summer meeting in New Orleans in June, which summarized the results of the study as of February 1975.

About 50 participants from ERDA, NRC, BNL, LASL, ORNL, ANC, and GA attended the FY-75 end-of-the-year review meeting on June 26, 1975, in Germantown. The GA presentation summarized the risk assessments arising from a detailed (Phase I) study of relatively high risk events. In addition, the presentation contained recommendations for safety R&D programs as well as design options associated with moisture ingress events. It was agreed by GA and ANC to review the new "RRD Decision Criteria on Gas-Cooled Reactor Safety R&D."

In view of the large volume of information that had been generated during the course of the study, the decision was made in July 1975 to dedicate substantial effort toward writing a multivolume status report. As a part of its independent review, ANC reviewed GA's reheater leak and core auxiliary cooling system assessments. Also in July, ANC summarized its independent review activities by concluding that "GA appears to be performing the analyses in an unbiased and professional manner."

In September 1975, the three parties (RRD, GA, and ANC) agreed upon a set of criteria to be used as decision criteria for gas-cooled reactor safety R&D.

On November 4, a progress review meeting was held in San Diego. RRD reviewed GA's progress on the multivolume report, completion of the fission product add-on task, and future work.

Issuance of the seven-volume status report early in 1976 is a major milestone of the study. The study is continuing as described in Section 8 of this volume.

Several more publications are planned. Two are scheduled for the special IEEE issue in August 1976: "Reliability and Safety Analysis Methodology in the Nuclear Program of ERDA" by W. H. Hannum, F. X. Gavigan, and D. E. Emon, and "Common Mode Failure Theory and its Application to HTGR

Cooling System Reliability" by K. N. Fleming and G. W. Hannaman. A paper entitled "Realistic Assessment of Consequences of HTGR Accidents" by A. W. Barsell and V. Joksimovic is being prepared for publication in Nuclear Safety.

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3. METHODOLOGY

3.1. PUBLIC HEALTH AND SAFETY EXPRESSED AS RISK

The AIPA study is fundamentally concerned with the public health and safety. The characterization of public risk can take many forms, as can the selection of criteria for comparison of risks. A characterization of risk was needed for the AIPA study that would allow appropriate quantitative analysis of the public health and safety.

Early work on the subject by Farmer (Ref. 3-1) defined risk as curies of I-131 released to the environment per year, as an average value predicted for accidents in nuclear power plants. Starr (Ref. 3-2), in work considering additional possible public hazards, treats risk as the probability of death per person per year. The Reactor Safety Study (WASH-1400 Draft) (Ref. 3-3), defined risk as consequence per unit time. It defined the equation as

$$\text{risk} \left\{ \frac{\text{consequence}}{\text{unit time}} \right\} = \text{frequency} \left\{ \frac{\text{events}}{\text{unit time}} \right\} \\ \times \text{magnitude} \left\{ \frac{\text{consequences}}{\text{events}} \right\}$$

The AIPA program (Ref. 3-4) chose the same definition in saying that risk equals the product of the annual occurrence probability of an event (or frequency) times the consequence expressed in rem or man-rem. The choice of using simply rem or man-rem for consequence was basically the result of an AIPA programmatic decision to emphasize work on accident sequences within the HTGR rather than to devote major resources to the details of atmospheric dispersion, potential population injuries, and public financial losses.

The characterization of risk in the AIPA study is then achieved by providing values for probability per year for each event and the rem and man-rem doses resulting from the event. Values are customarily plotted on a summary risk plot of probability versus consequence.

It has been recognized over the years (and in the references cited above) that the criteria for acceptable risk or for comparison of risks are not established and will not necessarily constitute constant risk values based on the equation given above. In addition, concern has been expressed (e.g., Ref. 3-5) that the public and the legislators may have difficulty in dealing with risk criteria that involve probabilities (especially very low probabilities). One suggestion (Ref. 3-6) for a new approach to this is to base the criteria on the consequence divided by the time delay in the receipt of the consequence. The equation can be defined as

$$\text{risk} = \text{consequence} \div \text{time delay} \quad .$$

The time delay is especially important for cases that may require evacuation of a local population. Further development of criteria in the future is certainly in order.

The interim criterion for judging comparative risks in the AIPA study up to this time has been that higher risks, as defined in the equation above for the AIPA program, are more important. This criterion had some bearing on the conclusions drawn from the preliminary analyses. However, for the more detailed analyses, the foremost conclusions are not sensitive to the exact formulation of the criterion equation as long as both probability and consequence are included.

3.2. PROBABILISTIC RISK ANALYSIS

The probabilistic methodology was adopted for use in AIPA since it provides a viable basis for attaining the AIPA objectives of recommending safety R&D and performing design option investigations. The conclusion

that probabilistic methodology is suitable follows from several considerations. First, the probabilistic approach can treat a complete spectrum of initiating events. With the use of event trees, probabilistic analyses can encompass all plant response sequences that can follow each particular initiating event, including the consideration of multiple failures where appropriate. Second, the probabilistic approach is oriented toward producing realistic results and assessing the uncertainty therein. This is achieved by treating both probability and consequence as statistically distributed parameters to reflect uncertainties in the knowledge of their true values. The results can be interpreted in terms of median values or in terms of lower and upper limits to reflect the degree of uncertainty entailed in the analysis. Thus, the results of quantitative probabilistic safety analyses provide a basis for ranking event sequences with respect to relative safety importance and identifying areas of major contribution to uncertainty. The highlights of the differences between the probabilistic and deterministic approaches to safety analyses are shown in Table 3-1.

The general method of probabilistic safety analysis as performed in AIPA is shown in Fig. 3-1. The first step in the method is to identify the initiating events that have the potential for the release of radionuclides to the environment. This is accomplished by identifying all sources of radionuclides in the HTGR and the physical barriers that normally prevent release. The next step is to identify the anomalous events that can trigger plant response sequences that could lead to breaches of these barriers and radionuclide transport to the environment. Each such event is termed an "initiating event" since it represents the starting point for event sequences that could lead to radionuclide release. The remaining steps in the risk analysis methodology are performed separately for each initiating event.

TABLE 3-1
COMPARISON OF PROBABILISTIC AND DETERMINISTIC SAFETY ASSESSMENT METHODOLOGIES

	Probabilistic	Deterministic
Initiating events studied	Complete spectrum	Limited to "credible" and preselected events
Plant response sequence	All possible plant response sequences, including multiple failures	Restricted to sequences satisfying "single failure" criterion
Probability evaluation	Median values and uncertainties assessed	Usually not assessed
Consequence evaluation	Median values and uncertainties assessed. Assumptions are realistic and always mechanistic.	Upper bound values evaluated without measure of uncertainties. Assumptions are conservative and nonmechanistic in some cases.
Relative safety significance of events (for guidance in recommending safety R&D)	Established quantitatively by means of summary risk plots	Established qualitatively since probabilities and margins on consequence are not assessed

Fig. 3-1. Probabilistic safety analysis (risk assessment)

Once an initiating event is defined, an event tree is constructed to display the response of the plant to that event in terms of the possible sequences of events that could follow the initiating event. With the use of the plant design, operator procedures, and various types of information from systems analysis, an expected sequence of events is defined for the initiating event. The next step is to identify any possible variations from the expected sequence that may have an impact on the sequence. These variations usually consist of different combinations of successes and failures of the plant systems designed to control the plant and prevent or mitigate the consequences of reactor accidents. The expected sequence and associated variations represent the information needed to construct the event tree.

The construction of the event tree is crucial to risk analysis since this is where the quantitative aspect of the analysis is formulated. Each branch of the tree represents one possible outcome of the initiating event, which needs to be studied in terms of likelihood (or probability of occurrence) of the associated radiological consequence.

One advantage of the event tree format is that it permits the formulation of event sequence probabilities in terms of probabilities of specific events, which are more easily handled from an analytical point of view. An example of an event tree is given in Fig. 3-2. The sequence probability corresponding to branch C, designated $P(C)$, can be expressed as

$$P(C) = P_1 Q_2 Q_3 \quad , \quad (3-1)$$

where $P(C)$ = sequence probability for branch C,

P_1 = probability that event 1 occurs,

Q_2 = probability that event 2 does not occur,

Q_3 = probability that event 3 does not occur.

Fig. 3-2. Example event tree

Each of the three terms on the right side of Eq. 3-1 is termed a branch point probability since it is associated with a branching point on the event tree. In estimating the values of Q_2 and Q_3 , the effects of preceding branch points in the sequence are taken into account.

The calculations of the branch point probabilities for systems have been done with reliability diagrams, fault trees, and physical logic methods as suitable. The current emphasis is to calculate failure probabilities with abbreviated fault trees. The details of the techniques used in each case account for common mode failures and their effect on the prediction of failure probabilities. A new common mode technique is used, called the beta factor method (Ref. 3-7). It allows estimates of the portion of failures that are common mode to be incorporated into the analyses without the necessity of identifying every interaction in detail.

Prediction of consequences, such as radiation doses to the public, has historically been done with great conservatism. The objective of this study is to perform realistic calculations of consequence and provide explicit estimates for the uncertainty in the prediction.

Computer codes were used in many cases to calculate the consequences. A brief listing of these codes is as follows:

1. RECA, for core thermal analysis.
2. CORCON, for fuel temperature transients in heatup sequences.
3. SORS, for fission product transport during temperature transients.
4. RATSAM, for PCRV blowdown transients.

5. CONTEMPT, for containment building response to depressurization events.
6. TDAC, for radionuclide transport from the containment building and the evaluation of radiological dose to the public.
7. OXIDE, for evaluating the effects of steam and air ingress to the primary coolant system (graphite oxidation and fuel hydrolysis).
8. RISQUE, for assessments of the effects of meteorological variations on dose.

Interpretation and display of the analytical results are important to provide clarity and meaning to the large quantities of data provided by probabilistic assessments. The event trees and consequence tables assist in this process. The most useful additional display is the summary risk plot. As used in this study, the coordinate axes of the risk plot are the event tree branch probabilities (in occurrences per year) on the ordinate and the branch consequence (in rem or man-rem) on the abscissa. Logarithmic scales are used on both coordinates; therefore, lines of constant risk, defined as probability multiplied by consequence, are straight and inclined at 45° on this plot. Important branches on event trees can then be visually distinguished by observing which plotted points have the highest risk.

Decisions on safety research can be guided by the probabilistic results, including the uncertainties in the predictions, which can also be shown in the trees, tables, and plots.

Alternate design options can be an important subsequent consideration. They may be of interest to enhance safety in some cases, or in other cases they may be worthwhile to reduce electrical power generation cost when specific, potential safety problems do not, in fact, create a public hazard.

Probabilistic risk analysis is discussed in more detail in Volume II.

3.3. UNCERTAINTIES

It is recognized that a large majority of the event sequences considered in the AIPA study are not expected to occur and, indeed, have not occurred. As a result, it is often difficult to estimate the probabilities and consequences with sufficient accuracy to make point estimates meaningful. For this reason, most parameters used in risk analysis are treated in terms of statistical distributions that quantify the uncertainties in the knowledge of the true parameter values. Computations of probability, consequence, and risk involve the proper statistical combining of the uncertainty distributions for the parameters. An overview of the approach used to account for parameter uncertainties is given below, with a more detailed description given in Volume II.

It has been found that the lognormal distribution provides a good, practical approximation for the uncertainty distributions for most of the parameters considered in the Phase I AIPA analysis. This use of the lognormal distribution for parameter uncertainties is consistent with the approach used in the Reactor Safety Study (Ref. 3-3). A realistic (or central) estimate for any parameter X is given by the median value X_m of the lognormal uncertainty distribution for X . The uncertainty in the estimation of the true value of X is expressed numerically in terms of an uncertainty factor f . The value of f is computed from the lognormal uncertainty distribution and is used to compute the lower and upper limits of an uncertainty interval as follows:

$$\text{lower limit: } X_5 = X_m / f \quad , \quad (3-2)$$

$$\text{upper limit: } X_{95} = X_m f \quad . \quad (3-3)$$

The value of f is determined such that the estimated probability is 0.05 that the true value of X is below X_5 , 0.90 that it is between X_5 and X_{95} , and 0.05 that it is above X_{95} . In other words, the interval defined by X_5 and X_{95} represents an estimated 90% uncertainty interval associated with the evaluation of the true value of X .

In the above discussion, the parameter X represents any parameter of interest in the probabilistic risk assessment. It can stand for risk itself or for the two parameters (probability and consequence) whose product is used to define risk. It can also be interpreted to represent any of the parameters upon which probability depends (such as failure rate or mean repair time) or the parameters upon which consequence depends (such as fuel particle release rate, plateout, or decay heat generation rate). In order to perform the quantitative risk assessment, some means had to be employed to evaluate, for example, the median and uncertainty factor estimates for probability in terms of the estimated values of the medians and uncertainty factors for the parameters upon which probability depends. Two basic approaches were used: Monte Carlo simulation and analytical formulations based on conventional propagation of error techniques used in statistical analysis. The same basic approaches were used in the consequence and risk evaluations. Since these computational approaches are too complex to be adequately described in this summary volume, their description has been left to the more detailed discussion of methodology presented in Volume II.

3.4. COMMON MODE FAILURES

The term "common mode failure" is often applied to denote combinations of failures that are not independent of one another. A common mode failure can occur in redundant equipment by two different mechanisms: (1) the so-called "common cause" mechanism in which a common cause results in the failure of redundant components, or (2) the so-called "causal" mechanism in which the failure of one redundant component causes the failure of the other components. Methods have been developed to provide estimates for such failures even in cases where complete definition of the equipment or complete data are not yet available.

Failures resulting from a common cause are generally found to make a significant impact on the predictions of equipment failure, while the effects of considering causal mechanisms are generally negligible. A technique has been developed, called the beta factor method (Ref. 3-7), to provide a quantitative assessment of the impact of considering common cause mechanisms in the estimation of probabilities. Some of the fundamental concepts of this method are summarized below, together with a graph illustrating the results of its application. The details of the computational formulas and their derivations are presented in Volume II. A similar methodology has been developed for the causal mechanism and is likewise presented in Volume II.

The methodology for analyzing the common cause mechanism makes use of a factor β , which can be described simply as the proportion of all failures of a redundant component that would be shared by its redundant brother components by virtue of one or more common cause mechanisms. In other words,

$$\beta = \frac{C}{I + C} \quad , \quad (3-4)$$

where $(I + C)$ represents the total failure history of a component, C represents the failures that would occur in all redundant components due to a common cause (such as by loss of a common power supply), and I represents failures that would occur independently among redundant components. Thus, the value $\beta = 1$ signifies an extreme case of total common mode dependence among redundant components; that is, the failure of one redundant component would always be accompanied by failures of the other components. In this case, the system failure probability is relatively high, being equal to that of a single component. At the other extreme, the value $\beta = 0$ signifies total independence of the redundant components with respect to the common mode mechanism. In this case, the system failure probability is relatively low, being substantially less than that of one component.

The effects of variations in the common cause coupling parameter β are illustrated in Fig. 3-3 for the simple case of a system with two components in parallel, only one of which is required for successful functioning of the system. Notice that for any λt value for a single component, increases in β (which signify increases in common cause coupling of the two components) result in increases in the failure probability of the system.

3.5. HUMAN OPERATOR RELIABILITY

Some of the fault trees describing branch points in the AIPA event sequences identify sequences that require operator action to mitigate the event consequences. A method was developed to estimate the likelihood of an operator response for the specific conditions identified, and to relate this likelihood to the time required for the operator to perform a particular response. The time for an operator response becomes an important safety variable when the operator actions can affect a time-dependent consequence (i.e., stopping a radioactive leak). The operator response model utilized in this report provides a consistent method of treating operator actions for probability and response time. The method also enables the safety analyst to assess uncertainties in both response time and probability of action.

The basic modeling assumptions, which will provide a simple model approximating general operator behavior, are as follows:

1. The operator has a probability of zero for making any response instantaneously.
2. Given enough time, the operator will eventually take some action, which most likely will not increase the potential event consequences.

Fig. 3-3. One-out-of-two system unreliability versus λt for various values of β

3. If the operator finds that the first actions are not sufficient, he will take further corrective action until a mitigating condition is achieved.

For the more detailed AIPA analysis, an exponential distribution model was used for the time variable. This expression relates the probability of successful action within a time following an abnormal occurrence (initiating event sequence leading to the fault tree action under consideration). The formulation for operator success is

$$P_{os}(t) = 1 - e^{-(t/MTOR)} \quad (3-5)$$

In recognition of the fact that operator reliability is limited, the value of $P_{os}(t)$ was never allowed to exceed an upper limit value designated by P_s . In Eq. 3-5, t is the time allowed for operator action or the time when a major change in the consequences can occur due to operator inaction. The MTOR is the mean time to a correct operator response and may be viewed as the time within which 63% of the trained operators would perform the action successfully. For the purpose of estimating their values, MTORs have been thought of as the expected response time for an average, trained operator. A lognormal distribution on an MTOR can be used for the purpose of assessing uncertainty in the response time of different operators. A plot of Eq. 3-5 is shown in Fig. 3-4, together with the upper limit value of P_s on the probability of successful operator action.

To use Eq. 3-5, several experts who have experience with operator training, plant operations, and details of the fault tree conditions were asked to estimate time required for the average operator to respond to the various events under consideration. This was done in lieu of obtaining actual data. The estimates were increased by 10% to account for stress levels that would tend to slow down the operator response time. Based on these estimates, the upper 5%, 50%, and 95% confidence limits of MTOR were determined.

Fig. 3-4. Plot of Eq. 3-5

This simple model is useful in treating various operator conditions identified through fault tree analysis. It identifies the response times and probability objectives that can be incorporated into the operator training programs, and provides measurable parameters needed in the assessment of operator reliability. As more real data regarding the operator response become available, this model could be extended to include more complicated distributions; however, a more detailed examination of equipment failure rate uncertainties should also be made if this is done.

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4. PRELIMINARY ANALYSIS

4.1. BACKGROUND

The preliminary analysis was performed pursuant to two AIPA objectives—the primary objective of providing guidance for HTGR safety research, and another objective of considering alternate design options related to HTGR safety. The analysis began with a broad study of seventeen initiating events and served as an economical scoping study that provided an early sorting of the relative importance of several representative initiating events. It led to the identification and selection of three initiating events for more detailed risk evaluation during the subsequent Phase I analysis (discussed in Section 5) and then, as a result of the more detailed analysis, led to the research and development recommendation discussed in Section 6. The results of the preliminary scoping analysis also provided some of the insight needed to identify three initiating events as potential candidates for design option studies, one of which was selected as a pilot for such study efforts, as discussed in Section 4.4.

Very early in the AIPA study, the program management recognized that a detailed analysis of all possible anomalies in HTGR operation (whether caused internally or externally) would be overly and unnecessarily costly in terms of time, manpower, and funding requirements. An intuitive assessment indicated that only a few initiating events and plant response sequences would be important [a fact borne out by the Reactor Safety Study (WASH-1400) and later confirmed by the preliminary and Phase I AIPA analysis results]. Thus, it was recognized that the AIPA objectives could most effectively and efficiently be met by partitioning the study into two basic parts—screening and detailed. The screening was performed during the

preliminary analysis, which was performed with four objectives in mind: (1) the study must be comprehensive, covering a broad spectrum of initiating events and plant response sequences; (2) it must provide realistic (rather than conservative) numerical results; (3) it must provide numerical results of sufficient accuracy to serve as a basis for ranking the relative importance of events; and (4) it must be performed quickly and economically. To achieve these objectives, many analytical approximations and shortcuts were used to obtain realistic (median) estimates of occurrence probabilities and radiological consequences of the event sequences analyzed. Accordingly, the accuracy of the results varies depending on the degree of technical sophistication existing at the time the analyses were performed. However, it was later found that the preliminary results are generally within the uncertainty ranges of the subsequent, more detailed analyses, as reported in Volumes IV and VI. This tends to confirm the validity of using the quick, coarse-cut preliminary analysis approach as a basis for providing guidance in selecting the more important areas for more detailed investigation.

4.2. INITIATING EVENTS

An initial screening of possible initiating events was performed to identify a representative set of events for the HTGR. This led to the selection of the following set of seventeen initiating events for evaluation during the preliminary analysis:

1. Helium instrumentation line break
2. PCRV purge header break
3. Rapid PCRV depressurization
4. Slow PCRV depressurization
5. Drop of spent fuel shipping container
6. Gas waste surge tank rupture
7. Loss of offsite power
8. Moisture inleakage into primary coolant
9. Reheater tube leak

10. Rupture of neutron source
11. Main steam pipe rupture outside containment
12. Liquid waste tank rupture
13. Drop of solid waste container
14. Drop of irradiated hardware container
15. Drop of recycle fuel container
16. Safe shutdown earthquake
17. PCRV structural failure

The above initiating events were selected as being representative of the complete spectrum of classes of radioactive sources in the plant, each event having potentially the highest occurrence probabilities within each class. While this assured a considerable degree of completeness, it also resulted in evaluations of many events of relatively insignificant risk.

It is recognized that the above list of initiating events served as a starting point for the preliminary analysis. The list is representative for the HTGR but is not necessarily complete. Now that the Phase I detailed analysis has been completed, it is necessary to cycle back with the knowledge gained therefrom and identify additional initiating events for inclusion in a continuation of the preliminary analysis and, if appropriate, subsequent detailed analyses to assure the completeness of the overall AIPA study effort.

4.3. PRELIMINARY RISK ANALYSIS

Each of the seventeen initiating events listed above was analyzed to identify the potentially important alternative plant response sequences. These various potential event sequences were displayed in event trees in the same format as illustrated in Fig. 3-2. For each branch (or sequence) in the event tree, the objective was to estimate realistic values for the branch probability (P), consequence (C), and risk ($P \times C$)--an objective

that could not be attained in several instances (particularly for the consequence assessments). The probability values indicate the annual occurrence rates of the branches, and the consequences represent the corresponding estimates of whole body dose (in rem per occurrence) evaluated for a receptor at the outer boundary of the low population zone (LPZ), based on median weather conditions and "expected" (rather than "design") values for circulating activity.

A summary risk plot for all seventeen initiating events is shown in Fig. 4-1. The coordinates are basically comparative scales. The ordinate is annual occurrence rate, and the abscissa is the 30-day LPZ dose in rem per branch occurrence. The diagonals are lines of constant risk in rem per year. Each point displayed in the plot is derived from one branch of one of the seventeen initiating events considered. The points are identified by a number (corresponding to the initiating event numbers listed on the figure) and one or two letters that identify the specific branch in the event tree for the specified initiating event. (Some of the more important branches are identified subsequently; identifications for the other branches are to be found in Volume III, Section 4.)

The lower portion of Fig. 4-1 (below a probability of 10^{-9} occurrences per year) is shown by dotted lines. This is done in recognition of the fact that very low probabilities are not meaningful since events with such low probabilities are expected to be overwhelmed by other events (like meteor impact at the plant) which have not been specifically included in the analysis but which could have substantially higher occurrence probabilities. About all that can realistically be said about such probability values is that they are small, and the degree of smallness is of no practical importance, particularly with respect to the objectives of the preliminary analysis. Thus, the format of Fig. 4-1 permits the display of a broad range of the numerical results for the initiating events analyzed while displaying the current criterion that all branches with annual occurrence rates below 10^{-9} should be regarded simply as low probability events.

Fig. 4-1. Risk summary -- seventeen initiating events

An enlarged plot of the most significant part of the overall risk summary shown in Fig. 4-1 is provided in Fig. 4-2. This plot covers nine initiating events that have branches of highest relative risk. This simplified summary risk plot provides a convenient basis for sorting the initiating events to satisfy the objectives of the preliminary analysis.

The overall risk level of each initiating event is essentially determined by only a few branches (usually one or two); the contributions of the other branches are negligible. This can easily be illustrated by the three points shown in Fig. 4-2 for the reheater tube leak, initiating event 9. The corresponding risk values were estimated to be as follows:

<u>Branch</u>	<u>Risk</u> <u>(rem per year)</u>
9A	1×10^{-7}
9N	8×10^{-8}
9J	2×10^{-10}

The total risk from these three branches is approximately 1.8×10^{-7} rem per year, where the contribution from branch 9J is negligible.

In like manner, the overall risk level of the plant is essentially determined by a few initiating events, namely, those initiating events with branches of highest relative risk. Using this rationale and the information displayed in Fig. 4-2, it is clear that the following initiating events are of greatest relative importance from the point of view of overall potential risk, listed in order of decreasing relative risk:

1. Reheater tube leak (Branches 9A and 9N)
2. Loss of offsite power (Branch 7BI)
3. Safe shutdown earthquake (Branch 16H)
4. Slow PCRV depressurization (Branch 4A)
5. Gas waste surge tank rupture (Branch 6C)

Fig. 4-2. Risk summary -- nine initiating events

As a result of an ERDA-RRD review in February 1975, the first three initiating events were selected for more detailed evaluation in the subsequent Phase I analysis (as discussed in Section 5), with the objective of providing recommendations for future safety R&D programs for the HTGR. Future updating should consider more detailed analyses of the remaining two initiating events, with a fairly high priority. Scenarios for the six branches included in the above list of initiating events can be summarized as follows:

1. Branch 9A: Large reheater leak occurs, followed by identification and isolation of the leaking loop.
2. Branch 9N: Intermediate reheater leak occurs, followed by a delay (of less than 2 hr) in identifying the initiating event and shutting down the reactor.
3. Branch 7BI: Loss of offsite power occurs, followed by trip of the main turbine generator, no restoration of offsite power within 30 sec, reactor trip, cooldown on main loops for 15 min,* no restoration of offsite power within 15 min, loss of CACS cooling prior to restoration of offsite power, intentional depressurization of PCRV by pumping helium to storage, inability to restore core cooling within allowable duration, successful isolation of containment, and successful operation of the cleanup filtration system.
4. Branch 16H: Safe shutdown earthquake occurs, followed by loss of offsite power, trip of main turbine generator, loss of main loop cooling, reactor trip, loss of CACS cooling within 10 hr, inability to restore CACS cooling, successful isolation of containment, and successful operation of containment filtration system.

* After completion of the preliminary analysis, a design change eliminated this main loop cooldown capability.

5. Branch 4A: Slow PCRV depressurization occurs, followed by successful isolation of containment, reactor trip, successful cooling by main loops, and successful operation of containment filtration system.
6. Branch 6C: Rupture occurs in gas waste surge tank more than 30 days after a steam dump necessitated by a steam generator leak, followed by successful operation of the filtration system in the reactor service building.

4.4. EXPLORATORY DESIGN OPTIONS STUDY

In Section 4.3, a priority ranking was established for five of the nine initiating events displayed in Fig. 4-2. That ranking was based on the identification of branches of highest relative risk, the intent being to identify candidates for more detailed analysis that offer the greatest potential for contributing to the development of HTGR safety R&D programs.

The priority ranking for the next group of initiating events identifiable from Fig. 4-2 has a different emphasis. These initiating events are of interest because they make an essentially negligible contribution to the overall risk level, thereby raising a question as to the availability of alternative design options that could reduce the cost of electricity generation while still maintaining a high level of public safety. This group of initiating events, listed in order of decreasing risk, is as follows:

1. Moisture inleakage into the primary coolant (Branches 8AC and 8AD)
2. Drop of solid waste container (Branch 13C)

3. Rapid PCRV depressurization (Branch 3A)

4. PCRV purge header break (Branch 2C)

The first event in this list was selected for an exploratory design option study, the objective of which was to investigate the potential for extending the use of the basic risk methodology to reduce overall plant costs while assuring that an acceptable level of safety is maintained. The associated branches for the moisture inleakage event (points 8AC and 8AD in Fig. 4-2) are about two orders of magnitude in risk below the risk envelope defined by points 9A, 9N, and 7BI in the summary risk plot. The next event in the list (the drop of a solid waste container, Branch 13C in Fig. 4-2) is about three orders of magnitude in risk below the risk envelope. Since no significant design options are apparent, a low priority is deemed appropriate for design option studies in this area. The item of third priority in the above list, the rapid PCRV depressurization, has more commonly been called a design basis depressurization accident (DBDA), since it contributes to the bases for the design of the containment and the auxiliary circulators. The occurrence of such an event is very unlikely because it would be equivalent to a breach of an ASME Class 1 nuclear pressure vessel. Since the design requirements derived from the DBDA have a significant impact on the cost of the plant and since the risk is very low (over three orders of magnitude of risk below the risk envelope in Fig. 4-2), there seems to be ample justification for the exploration of design options in this area on a high priority basis. The last entry in the above list, PCRV purge header break, is of even lower risk (over five orders of magnitude below the above-mentioned risk envelope). However, since there do not appear to be any major design options, the pursuit of design option studies in this area should be of a relatively low level of priority.

Before going on to describe the exploratory design option work performed with regard to moisture inleakage into the primary coolant, it is appropriate to briefly describe the scenarios for the five branches identified in the above list.

1. Branch 8AC: Rupture occurs in a steam generator tube (causing moisture to leak into the primary coolant), followed by failure of the moisture monitor system to identify the defective loop, erroneous isolation and dump of a nondefective loop, reactor trip, shutdown of all main loops, cooldown by CACS, and lifting and reseating of the safety relief valve.
2. Branch 8AD: Same as Branch 8AC except that the relief line block valve must be closed manually to compensate for failure of the safety relief valve to reseal.
3. Branch 13C: Drop of lower half of high-temperature filter adsorber during transit in the reactor service building, followed by loss of shipping container integrity and failure of filter system to operate in the reactor service building.
4. Branch 3A: Rapid PCRV depressurization (DBDA) occurs, followed by successful isolation of the containment, reactor trip, cooling by main loops, and operation of the containment filtration system.
5. Branch 2C: Break occurs in purge header, followed by closure of all purge line check valves.

The exploratory design options work was based on the moisture inleakage event. The study was centered on the moisture inleakage protection system and considered the six design options identified in Table 4-1; the abbreviations used therein are as follows:

MMS moisture monitor system
PPS plant protection system (safety-grade)
OPS operational protection system (non-safety-grade)
SGs steam generators

In the current design [the reference 3000-MW(t) plant], the moisture monitor system is a part of the safety-grade plant protection system. Three of the options considered investigated the effects of changing the MMS from the PPS to the non-safety-grade OPS. The various options also considered several alternatives with respect to dumping a steam generator, ranging from no dump at all to dumping to a dump tank or to the atmosphere. The no-dump option seemed appropriate for investigation since the two branches of highest relative risk for the moisture inleakage event (Branches 8AC and 8AD, as identified above) both involve the isolation and dump of the wrong steam generator.

The basic objective of the design option study is to identify the system option with the least cost and an acceptable level of risk. This requires a two-faceted analysis of each design option: one facet concerned with radiological risk assessments of the form performed in the preliminary risk analysis, and the other concerned with cost assessments.

The radiological risk analysis for each design option was formally performed by the use of the event tree methodology. The impact of design differences can be appropriately assessed in terms of probabilities, radiological consequences, and radiological risk.

TABLE 4-1
KEY CHARACTERISTICS OF DESIGN OPTIONS

Option A: Current System	Option B: Simplified MMS and Dump	Option C: No Dump	Option D: Simplified MMS, No Dump	Option E: Simplified Dump and Header SGs Together	Option F: Simplified MMS and Dump, Header SGs Together
MMS 3 modules per loop, PPS (2 out of 3)	MMS 1 module per loop, OPS 3 modules in plenum, PPS (2 out of 3)	MMS as in A, PPS	MMS as in B, OPS	MMS as in A, PPS	MMS as in B, OPS
Dump tank 1900 ft ³	No dump tank, dump to atmos- phere	No dump tank, no dump	No dump tank, no dump	No dump tank, dump to atmos- phere	No dump tank, dump to atmos- phere
Dump valves, PPS	Dump valves, PPS	No dump valves	No dump valves	Dump valves, PPS	Dump valves, PPS
Isolation valves, PPS	Isolation valves, PPS	Isolation valves, PPS	Isolation valves, PPS	Isolation valves, PPS	Isolation valves, PPS
No header, isolate and dump defec- tive loop	No header, isolate and dump defec- tive loop	No header, isolate defec- tive loop	No header, isolate defec- tive loop	Header each 3 SGs together, isolate and dump header	Header each 3 SGs together, isolate and dump header
Response: MMS, 22 sec; valve stroke, 5 sec; dump time, 40 sec	Response: MMS, 22 sec; valve stroke, 5 sec; dump time, 40 sec	Response: MMS, 22 sec; valve stroke, 5 sec; no dump	Response: MMS, 22 sec; valve stroke, 5 sec; no dump	Response: MMS, 22 sec; valve stroke, 5 sec; dump time, 50 sec	Response: MMS, 22 sec; valve stroke, 5 sec; dump time, 50 sec

The cost assessment for each design option, to be comprehensive and indicative of significant differences, must contain all economic parameters, including initial costs as well as operating costs incurred over the operating life of a plant. Accordingly, the model of system cost (C_s) for each design option was expressed as

$$C_s = C_I + C_F + C_M + C_D$$

The first cost term (C_I) covers all of the initial costs (materials, labor, indirect) and was evaluated by engineering cost estimates. The remaining three cost terms account for plant operating costs. The estimated cost of unscheduled down time (C_F) accounts for the cost of replacement power required for the unscheduled downtime of a plant throughout its entire 40-year life. The maintenance and replacement costs (C_M) over the 40-year life of a plant were assumed to be a fixed fraction of the initial costs: 50% for safety-grade equipment and 30% for non-safety-grade equipment. The last cost term (C_D) covers costs resulting from potential occurrences of moisture inleakage events during the life of the plant. This cost is estimated as follows:

$$C_D = L \sum_x P_x C_x$$

The plant life (L) is taken to be 40 years. Event trees developed for each design option for the moisture inleakage event provided values for the annual occurrence probability (P_x) for each branch (x) in each event tree. The corresponding cost of damage (C_x) is, to a first-order approximation, a function of the magnitude of the radiological consequences and consists of costs incurred for cleanup, repair, penalties (such as fines levied by regulatory authorities), and purchase of replacement power for the period of plant unavailability resulting from the moisture inleakage event. The actual computations revealed that the cost C_x was dominated by the cost of replacement power. Also, the summation over all branches (x) in each individual event tree was dominated by only one or two branches.

The results of the radiological risk and cost assessments are summarized in Tables 4-2 and 4-3, respectively, for all six of the alternative design options considered. These results indicate that Option F offers the most attractive cost saving considering initial costs plus operating costs over the 40-year life of the plant. The exploratory study shows that the radiological risks of all the design options investigated remain within the previously defined HTGR risk envelope. Potential cost savings over the plant lifetime among the design options are as follows:

1. Headering of steam generators has a potential of \$16 million cost saving over the 40-year life of the plant, most of it from the reduced cost of piping.
2. Changing the MMS from the PPS to the OPS classification has a potential saving of several million dollars (in 40 years) and appears worthwhile.
3. Eliminating steam generator dumping altogether may provide a considerable saving in the initial plant cost. However, the risk of costs associated with plant inoperability may outweigh initial saving if the probability of small leaks, which do not pose safety problems but affect plant availability, is high. These aspects need further analysis and elucidation.
4. Eliminating the dump tank and substituting atmospheric dump also offers cost savings. However, a greater saving can be achieved if slower valve stroking times are adopted. This additional cost saving could outweigh the cost of removal of large quantities of moisture.

The above conclusions are sensitive to the cost data of systems and components such as the MMS system, steam generator headering, and valves, and to the expected frequency of the steam generator tube rupture. Any

TABLE 4-2
RADIOLOGICAL RISK SUMMARY FOR ALL DESIGN OPTIONS

System Option	A	B	C	D	E	F
Total event risk ^(a) for each option [probability x consequence (rem/yr)]	2×10^{-10}	5×10^{-9}	8×10^{-9}	1×10^{-8}	3×10^{-9}	6×10^{-9}
Dominant branch						
Probability (yr ⁻¹)	3×10^{-7}	3×10^{-8}	1×10^{-5}	1×10^{-5}	3×10^{-8}	3×10^{-6}
Consequence (rem)	7×10^{-4}	9×10^{-2}	6×10^{-4}	5×10^{-4}	1×10^{-1}	1×10^{-3}
Risk (rem/yr)	2×10^{-10}	2×10^{-9}	7×10^{-9}	7×10^{-9}	3×10^{-9}	3×10^{-9}

(a) Total event risk is the sum of the risks of all branches in the event tree for the moisture inleakage event.

TABLE 4-3
SUMMARY OF ECONOMIC FACTORS FOR ALL DESIGN OPTIONS (\$)

System	A	B	C	D	E	F
Initial system cost	45.4×10^6	44×10^6	44.3×10^6	43.3×10^6	33×10^6	32×10^6
Maintenance cost (40 years)	2.5×10^6	1.8×10^6	2.1×10^6	1.5×10^6	1.8×10^6	1.1×10^6
Unscheduled downtime cost	3.3×10^6	2.3×10^6	2.3×10^6	1.7×10^6	3×10^6	2.3×10^6
Economic risk of moisture inleakage event	0.27×10^6	0.23×10^6	1×10^6	1×10^6	0.27×10^6	0.27×10^6
Total 40-year cost	51.5×10^6	48.3×10^6	49.7×10^6	47.5×10^6	38×10^6	35.7×10^6

change in the data could alter the conclusions of this study. The conclusions are preliminary and the analysis is continuing as a part of the AIPA effort.

5. PHASE I ANALYSIS

5.1. BACKGROUND

The preliminary scoping analysis led to the selection of the following three initiating events for more detailed risk evaluation:

1. Reheater leaks
2. Loss of offsite power
3. Large earthquakes (of the order of the SSE, and larger)

The detailed analysis of these events appeared most promising for achieving the primary objective of developing recommendations for HTGR safety R&D.

The Phase I analysis was initiated to perform a more detailed assessment of the probabilities and consequences associated with the three events. The analysis involved more detail where appropriate, as well as various expansions and improvements in methodology, such as:

1. Improved event and fault trees.
2. Accounting for common mode interdependencies, both intrasystem and intersystem.
3. Operator-equipment interactions.
4. Evaluation of uncertainties in probability estimates.
5. Inclusion of other organs (besides whole body) in consequence assessments.

6. More realistic bases for median consequence estimates.
7. Evaluation of uncertainties in consequence estimates.
8. Estimation of consequences in man-rem (in addition to rem).
9. Evaluation of uncertainties in risk estimates.

The details of the Phase I analysis are presented in Volume IV; the results are summarized in Section 5.2 of this volume. The specific R&D recommendations developed from the Phase I results are presented in Section 6.

As mentioned in Section 4.2, it is planned to continue the preliminary scoping study in an effort to ensure that the study of initiating events is complete (i.e., that no initiating event important to the study objective is inadvertently omitted from analysis). This may lead to a need to perform additional Phase I detailed analyses, thereby providing further guidance for safety R&D. Yet another aspect of the ongoing AIPA study is the introduction of certain portions of the Phase I R&D recommendations in accordance with plans formulated jointly by GA and ERDA and subject to ERDA authorization and approval.

5.2. RISK ANALYSIS

A detailed Phase I risk assessment was performed for three initiating events: reheater leak, loss of offsite power, and large earthquakes (ranging in magnitude from the SSE to twice the SSE). The results indicate that only a few plant response sequences are of interest in providing R&D guidance since they form an envelope on the summary risk plot shown in Fig. 5-1. These sequences, in order of decreasing median risk, are as follows:

1. Loss of offsite power with immediate turbine trip,
followed by no startup of the core auxiliary cooling

Fig. 5-1. Summary risk plot, Phase I analysis

system (CACS) and no subsequent restoration of CACS operation--LOSP, Branch BD.

2. Small reheater leak, followed by an inability to identify the leaking reheater, tripping of the turbine and reactor, and operator intervention to contain the activity within the reheat steam system--reheater leak, Branch SM.
3. Large reheater leak, followed by identification and isolation of the leaking reheater--reheater leak, Branch LA.
4. Large earthquake (approximately 1.5 times SSE), followed by reactor trip, no startup of the CACS, and an inability to isolate the containment--earthquake, $\alpha = 1.5$, Branch S.
5. Lesser earthquake (at or slightly greater than SSE magnitude), followed by reactor trip, early loss of the CACS without subsequent restoration to operation, successful isolation of the containment, and malfunction of the containment filter system--earthquake, $\alpha = 1.1$, Branch O.

Table 5-1 is a summary risk table for whole body doses; the more detailed listing in Table 5-2 includes risk assessments associated with bone and thyroid doses.

In the case of the reheater leak events, the important consequence is the whole body dose that results from the direct, immediate release of gasborne activity to the atmosphere. The dominant isotopes in the gasborne activity, chiefly Kr-88 and Kr-87, are also the primary dose contributors. With regard to the probability analyses, the probability of successful identification and isolation of the leaking reheater loop varies depending

TABLE 5-1
SUMMARY RISK TABLE

Initiating Event (a)	Branch Identification	Median Risk, R (rem/yr)	Uncertainty Factor, F(b)	Upper 95% Uncertainty Limit on Risk, R ₉₅ (rem/yr)	Rank (c)	
					R	R ₉₅
LOSP	BD	1×10^{-7}	40	4×10^{-6}	1	1
RHL-S	SM	2×10^{-8}	180	4×10^{-6}	2	2
RHL-L	LA	2×10^{-8}	10	2×10^{-7}	3	3
EQ. $\alpha = 1.5$	S	$\epsilon(3 \times 10^{-11})$ (d)	571	$\epsilon(2 \times 10^{-8})$	4	4
EQ. $\alpha = 1.1$	O	$\epsilon(4 \times 10^{-13})$	332	$\epsilon(1 \times 10^{-10})$	5	5

(a) LOSP = loss of offsite power, RHL-S = reheater leak - small, RHL-L = reheater leak - large, EQ $\alpha = 1.5$ = earthquake approximately 1.5 times the SSE, and EQ $\alpha = 1.1$ = earthquake approximately 1.1 times the SSE.

(b) Uncertainty factor (dimensionless) for 0.90 uncertainty band on risk.

(c) $R_{95} = R \times F$, the estimated upper 95% uncertainty limit on risk, rem/yr.

(d) ϵ means key probability numbers in the analysis are sufficiently low ($\leq 10^{-10}$) to have questionable meaning.

Note: Risk values are based on whole body doses.

TABLE 5-2
SUMMARY RISK TABLE DETAILED ANALYSIS

Basis of Risk	Initiating Event and Response Sequence ^(a)	Median Event Sequence Frequency (Occurrences/yr)	Uncertainty Factor For Probability	Median Individual Dose (rem) ^(b)	Median Population Dose (man-rem) ^(c)	Uncertainty Factor For Dose	Median Individual Risk (rem/yr)	Median Population Risk (man-rem/yr)	Uncertainty Factor For Risk
Whole Body - γ	LOSP BD	1×10^{-6}	23	1×10^{-1}	4×10^1	7	1×10^{-7}	4×10^{-5}	40
	RHL SM	4×10^{-6}	123	4×10^{-3}	9×10^{-2}	7	2×10^{-8}	4×10^{-7}	180
	RHL LA	2×10^{-3}	4	1×10^{-5}	2×10^{-4}	6	2×10^{-8}	4×10^{-7}	10
	EQ: $\alpha = 1.5$ S	$\epsilon(3 \times 10^{-12})$	519	1×10^1	4×10^3	3	$\epsilon(3 \times 10^{-11})$	$\epsilon(1 \times 10^{-8})$	571
	EQ: $\alpha = 1.1$ O	$\epsilon(4 \times 10^{-12})$	198	1×10^{-1}	4×10^1	11	$\epsilon(4 \times 10^{-13})$	$\epsilon(2 \times 10^{-10})$	332
Thyroid (Inhalation)	LOSP BD	1×10^{-6}	23	2×10^{-3}	8×10^{-1}	248	2×10^{-9}	8×10^{-7}	568
Bone (Inhalation)	LOSP BD	1×10^{-6}	23	9×10^{-6}	4×10^{-3}	1215	9×10^{-12}	4×10^{-9}	2354

(a) Sequence defined in Table 3-2 of Volume IV.

(b) 30-day exposure at LPZ boundary.

(c) Integrated man-rem within 50 miles of site.

on leak size. This accounts for a major intersystem dependency--the interactive effect of leak size and the sensitivity of loop activity monitors. Also, operator-equipment interactions are included by considerations of manual reactor trip and of operator isolation of the reheat steam system in response to instrumentation displays. From Table 5-2 it is seen that the upper 95% uncertainty limit values for whole body doses resulting from reheater leak events are no greater than 28 mrem (1.4 man-rem) per leak occurrence, and the corresponding upper 95% uncertainty limits for the risks are no greater than 4×10^{-6} rem per year (10^{-4} man-rem per year). It is clear from these very low dose and risk values that the reheater leak events are relatively (as well as absolutely) unimportant with respect to HTGR safety.

For the initiating event designated as loss of offsite power, the scenario for the branch of relatively highest risk (Branch BD, as identified previously) includes no startup of the CACS, which results in a core heatup transient. This is one of the LOSP branches of relatively highest consequence. Other branches with comparable consequence levels have lower occurrence probabilities than this branch.

For the core heatup event that could occur following an LOSP, the release of initial gasborne activity is small compared with the release of fuel body activity. With respect to the fuel body activity, Xe-133 is the greatest of the noble gas contributors to the whole body dose. The dose contributions to the other organs from the iodines and metallic fission products are relatively low due to their plateout in the PCRV and containment. Also, the holdup of metals in the graphite within the PCRV is another contributing effect. In particular, Sr, Ce, and Ru are predicted to plate out extensively, while I, Te, Sb, and Cs have more moderate plateout characteristics. However, Cs has a significant delay in graphite. As a result, Te-135 and Sb-125 are the highest contributors to the inhalation bone dose.

With regard to LOSP probabilities, the analysis included intersystem common mode considerations (such as multisystem dependence on preferred power) as well as intrasystem dependencies (such as the common mode characteristics of the diesel generators in the CACS). The analysis also included the consideration of the effects of operator responses as, for example, in the case of the evaluation of the probability of successful reactor trip. In addition, the potential benefits from performing maintenance were considered with regard to the restoration of malfunctioned equipment to service, as in the case of the CACS. It was found that the greatest contribution to the uncertainty in the estimated probability for Branch BD arises from the consideration of CACS start.

From Table 5-2 it can be observed that the median whole body, thyroid, and bone doses for LOSP Branch BD are:

<u>Median Doses</u>		<u>Organ</u>
<u>(mrem)</u>	<u>(man-rem)</u>	
110	40	Whole body
2	0.8	Thyroid
0.009	0.004	Bone

The corresponding values for the upper 95% uncertainty limits are:

<u>Upper Limit Doses</u>		<u>Organ</u>
<u>(mrem)</u>	<u>(man-rem)</u>	
770	230	Whole body
500	200	Thyroid
11	5	Bone

The occurrence probability for Branch BD is relatively low--a median value of 10^{-6} yr^{-1} , with an upper 95% uncertainty limit of $23 \times 10^{-6} \text{ yr}^{-1}$. The

median risk values are:

<u>Median Risk</u>		<u>Organ</u>
<u>(mrem/yr)</u>	<u>(man-rem/yr)</u>	
1×10^{-4}	4×10^{-5}	Whole body
2×10^{-6}	8×10^{-7}	Thyroid
9×10^{-9}	4×10^{-9}	Bone

The associated upper 95% limit values of risk are:

<u>Upper Limit Risks</u>		<u>Organ</u>
<u>(mrem/yr)</u>	<u>(man-rem/yr)</u>	
4×10^{-3}	2×10^{-3}	Whole body
1×10^{-3}	5×10^{-4}	Thyroid
2×10^{-5}	9×10^{-6}	Bone

The earthquake analysis considered earthquake magnitudes* ranging from the SSE to twice the SSE. It was estimated that earthquakes of such magnitudes would cause a loss of offsite power and a turbine trip with a probability of essentially one. For this reason, the LOSEP analysis was relied upon as a basis for the probability and consequence values developed in the earthquake analyses, with some modifications incorporated in an effort to account for earthquake effects.

* In this discussion, the term "magnitude" is used as a general measure of the size or intensity of an earthquake at a nuclear plant site. Numerical measures of earthquake magnitude are expressed in two forms: the ground acceleration at the site, and the ratio of the ground acceleration at the site to the ground acceleration prescribed for the safe shutdown earthquake for that site.

The consequences of earthquakes were taken to be the same as the consequences estimated for the corresponding LOSP event sequences, with a few modifications. Since the containment is an engineered safety feature designed to Seismic Category I, it is not expected to experience major structural damage as a direct result of a large earthquake. However, minor damage might be inflicted resulting in an increased rate of leakage of the containment atmosphere to the environment. The associated consequences were estimated to be approximately equal to those associated with a failure to isolate the containment. It was additionally recognized that damage resulting from an earthquake could alter the mechanistic details of containment cleanup and particulate fallout. Such effects were taken into consideration in the earthquake analysis by increasing the appropriate uncertainty factors involved in the consequence assessment.

The probabilities developed in the earthquake analysis are regarded primarily as indicators--not as rigorous, well-defined values. There are two basic reasons for this condition.* First, there is great difficulty in assessing the occurrence frequency of large-magnitude earthquakes. This is clear from the lack of concurrence among experts in the field, as evidenced in a recent survey by Okrent (Ref. 5-1). Further substantiation is provided in a paper by Newmark (Ref. 5-2), which shows that the assumed log-normal distribution of earthquake magnitudes is a good fit to the observed earthquake data over a probability range of about 0.001 to 0.999. However, the absence of data outside that interval leaves open to serious question any extrapolation into regions of lower probability (as required in the analyses presented in Volume IV). The second basic reason for regarding the probability results only as indicators concerns the evaluation of equipment failure probabilities. While equipment is designed in accordance with the appropriate seismic codes and standards, current design practice does not include the identification of potential modes and mechanisms of

* Additional considerations are presented in greater detail in Volumes II and IV.

earthquake-induced failure. This, coupled with a general lack of experience data for Seismic Category I equipment and structures subjected to large magnitude earthquakes (SSE or greater), makes the assessment of equipment failure probabilities very difficult. It was necessary to rely solely on engineering judgment, which could be considerably in error.

In referring to Table 5-2, it is evident that the median whole body dose is no greater than 10 rem (2×10^4 man-rem) per occurrence of an earthquake of very large magnitude, with an upper 95% uncertainty limit value of 30 rem (6×10^4 man-rem) per occurrence. Because of the low occurrence probabilities, the corresponding risk values are low. The median risk to the whole body is no greater than some such low value as 3×10^{-8} mrem per year (6×10^{-8} man-rem per year), and the corresponding upper 95% uncertainty limit values of risk are 2×10^{-5} mrem per year (3×10^{-5} man-rem per year).

In retrospect, every effort has been made in the Phase I analysis to develop event and fault trees that realistically display the events and event sequences that are significant for risk analysis. Likewise, all efforts have been directed toward realistic assessments of median probabilities and consequences and of their respective uncertainty factors. However, it is recognized that the constraints of time and technology have not always allowed the attainment of this goal of realism. Hence, it is expected that the uncertainties may, in some instances, be somewhat greater than the values developed thus far in this study. This is particularly true of the probability assessments performed in the earthquake analysis--a consideration alluded to in the foregoing discussion. It is clear that refinements in this area should be a part of subsequent analysis phases of the ongoing AIPA study.

The numerical results developed in the Phase I portion of the study, as displayed in Fig. 5-1 and Tables 5-1 and 5-2, clearly indicate a low safety hazard. This conclusion, which follows from considerations of consequences and of risk values, is evident by comparison with the 133 rem

value used in WASH-1400 as the threshold for acute fatalities (Ref. 5-3) and with the threshold for latent fatalities in the Biological Effects of Ionizing Radiation report (Ref. 5-4) (as further elaborated upon in Section 9 below).

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6. RESEARCH AND DEVELOPMENT RECOMMENDATIONS

6.1. BACKGROUND

The AIPA program involved two major analytical efforts: (1) the so-called "basic AIPA program" and (2) the so-called "fission product add-on task." The former was directed toward an overview of a standard (or "reference") 3000-MW(t) HTGR, the objective being to provide overall guidance in HTGR safety research and development; the resultant R&D recommendations are documented in Section 6.3. The latter was a more specialized study, which concentrated on fission product inventories and on the mechanisms for their transport and retention. The analytical results in this latter case provided valuable input to the basic AIPA program and served as a basis for ranking the isotopes and fission product parameters in terms of an index of relative safety R&D significance. The results obtained from the fission product add-on task are summarized in Section 7.

6.2. BASIS FOR RECOMMENDATIONS

The approach that was selected for R&D justification is probabilistic and considers specific initiating events and plant responses that could release radioactivity to the environment. The response alternatives of the plant are traceable through branches of event trees.

The summary risk table (Table 5-1) lists five event tree branches that serve as the basis for the R&D recommendations presented subsequently. These branches were included in the table based on two considerations:

1. The branch of highest relative risk (considering both the median and the upper 95% uncertainty limit) as determined in the Phase I study was selected for inclusion in the table, and

the R&D ideas associated therewith were identified.

2. Other branches analyzed in the Phase I study were considered in order of decreasing relative risk (again considering both the median and the upper 95% uncertainty limit). Branches that yielded repetitions of previously identified R&D ideas were not included in the table, while those that contributed new R&D ideas were included.

As was pointed out at the conclusion of Section 5.2, the numerical results of the Phase I study clearly indicate a low safety hazard. It is for this reason that the safety R&D recommendations developed during the Phase I study included no studies for the improvement of safety equipment. Instead, the R&D suggestions are directed toward the reduction of uncertainties associated with the prediction of probabilities and consequences in the branches of interest. The basic intent, then, is that the information derived from the proposed R&D would contribute to the validation of the low risk to public safety predicted for the HTGR.

To implement the approach outlined above, the following steps were taken. The important event tree branches were reviewed one at a time, in order of decreasing risk. For each such branch considered in turn, the individual branching points defined therein were compared to identify the specific event (or events) with the highest uncertainty in the estimated branch point probability. The specific event or events were analyzed to identify the specific R&D that would be required to reduce the uncertainty involved in the probability evaluation. In a like manner, the table of transport sequence factors (developed in support of the consequence calculations performed for the specific event tree branch under consideration) was reviewed to identify the factor or factors of highest uncertainty. This information then led to the identification of the R&D required to reduce the uncertainty in the consequence assessment. If the R&D ideas derived from a particular event tree branch were different from those

derived from previously analyzed event tree branches, then the appropriate data were included among the suggestions listed in Tables 6-1 and 6-2; otherwise, the information was excluded from these tabulations. This process was continued until a reasonable number of suggestions were accumulated covering a reasonably broad spectrum of events and technologies.

6.3. DISCUSSION OF R&D RECOMMENDATIONS

The information presented in Tables 6-1 and 6-2 is arranged in order of decreasing branch risk, where risk was evaluated in terms of whole body dose. As a result, the four R&D suggestions derived from LOSP Branch BD are listed first, followed by three suggestions from the two reheater leak branches and then two from the two earthquake branches.

The first R&D suggestion listed in Table 6-1 would be of benefit by improving the probabilistic prediction of CACS startup and operation. The second suggestion, the study of PCRV and containment integrity, benefits the calculation of the consequences of a core heatup transient that might occur following an LOSP. In addition, this R&D task could be used to corroborate or, if appropriate, modify the current judgment that the PCRV remains virtually intact in the LOSP branch in which core heatup could occur. The third and fourth suggestions involve tests to develop data concerning fission product release and plateout characteristics for application to core heatup transients. They also involve the use of the test data to verify and improve computer codes used in the analysis of fission product retention and transport characteristics.

The fifth R&D suggestion entails further work to better define the informational displays available to an operator during a small reheater leak, the ability of the operator to correctly interpret the displays and deduce the cause of anomalous plant operation, the effects of regulations and plant operating policies and procedures on operator response, and the operator's ultimate decision and response alternatives and their effects on

TABLE 6-1
RECOMMENDATIONS FOR HTGR SAFETY RESEARCH AND DEVELOPMENT

Initiating Event and Branch	Justification From AIPA				Area of R&D Interest	R&D Suggestion No.	Summary of Approach(a)
	Median Risk (rem/yr)	Uncertainty Factor	90% Uncertainty Band on Risk				
			R ₅ (rem/yr)	R ₉₅ (rem/yr)			
LOSP-BD	1 x 10 ⁻⁷	40	3 x 10 ⁻⁹	4 x 10 ⁻⁶	<ul style="list-style-type: none">• Probability of cooling system failure• Uncertainty in PCRV integrity in core heatup transient• Fission product release uncertainty	1 2 3 4	<ul style="list-style-type: none">• Probability model of core cooling.• PCRV integrity study and effect on containment.• HTGR fission product release and plateout tests.• Dev. and verification of fission product release and plateout code.
RHL-SM	2 x 10 ⁻⁸	180	1 x 10 ⁻¹⁰	4 x 10 ⁻⁶	<ul style="list-style-type: none">• Operator action uncertainty	5	<ul style="list-style-type: none">• Dev. better probabilistic operator response model.
RHL-LA	2 x 10 ⁻⁸	10	2 x 10 ⁻⁹	2 x 10 ⁻⁷	<ul style="list-style-type: none">• RH tube failures	6 7	<ul style="list-style-type: none">• Perform metallurgy studies leading to physical understanding of tube failure.• Tube failure model, test, and verification.
EQ-α = 1.5 S	ε(3 x 10 ⁻¹¹)	571	ε	ε(2 x 10 ⁻⁸)	<ul style="list-style-type: none">• Uncertainty in EQ. occurrence probabilities	8	<ul style="list-style-type: none">• Earthquake frequency studies.
EQ-α = 1.1 O	ε(4 x 10 ⁻¹³)	332	ε	ε(1 x 10 ⁻¹⁰)	<ul style="list-style-type: none">• Equipmt. survival probabilities	9	<ul style="list-style-type: none">• Confirm equipment and struct. survival.

(a) Details are given in Table 6-2.

Note: ε means that key probability numbers in the analysis are sufficiently low (<10⁻¹⁰) to have questionable meaning.

TABLE 6-2
DETAILS OF R&D APPROACH

1. Probability Model of Core Cooling

Perform detailed probabilistic analysis of and identify events leading to a loss of core cooling. Study should result in reduced uncertainty of core heatup transient probability and should produce a more detailed event tree.

2. PCRV Integrity Study and Effect on Containment

- Conduct a survey of available data for large concrete structures under elevated temperature and pressure, such as rocket launch sites. Conduct supplemental testing as required. Develop computer programs to use the data to predict PCRV and PCRV internals integrity during core heatup transient. Computer programs would be used to assess effect of released heat and gases on containment building integrity.

3. HTGR Fission Product Release and Plateout Tests

Plan and conduct tests to determine transient temperature-related release and plateout characteristics of fission products during core heatup transient. Tests should quantify plateout characteristics of iodine.

4. Development and Verification of Fission Product Release and Plateout Codes

Using data from HTGR fuel fission product release and plateout tests, improve and verify codes for prediction of fission product release and plateout during core heatup transient. Data should be evaluated on a per isotope basis.

5. Develop Better Probabilistic Operator Response Model

Develop a more realistic model to account for such considerations as the information available to the operator during a small reheater leak, the operator's ability to correctly deduce the cause of the anomalous plant condition, and the operator's decision and response, which would be affected by such factors as pertinent regulations and overall plant operating procedures and policies.

6. Metallurgy Studies Leading to Physical Understanding of Tube Failures

Conduct studies leading to physical understanding of reheater tube failure mechanisms in HTGR environment. Studies should be of sufficient detail to identify cause and effect relations and should provide mathematical relations for these. Since the use of this information will involve probabilistic analysis, the effects must be identified probabilistically.

TABLE 6-2 (Continued)

7. Tube Failure Model, Test, and Verification

Using understanding of physical failures gained from metallurgy testing, conduct testing of tubes in simulated HTGR environment. Since failure probabilities are low, the failure frequency must be increased by amplification of the failure cause magnitude and/or frequency. The effort includes verification of failure mechanisms in the tubes and confirmation of ability to predict leak rate frequency distribution for HTGR.

8. Earthquake Frequency Studies

Study and gather data from earthquakes. Use statistical methods to develop probability distributions applicable to probabilistic studies. Earthquake data should also be related to particular sections of the country where nuclear power stations are planned for the future.

9. Confirm Equipment and Structural Survival

Establish priorities for probabilistic safety analysis of equipment and structures, based on information in current event trees. Make use of seismic design codes and practices, together with available seismic test data, to identify the modes and mechanisms of hardware failure, develop appropriate probability models, and estimate the pertinent hardware failure probabilities.

plant operation. Suggestions six and seven are concerned with the development of a better understanding of the mechanisms of reheater tube failure as a basis for a better assessment of the probability of tube leakage as a function of leak size.

The last two R&D recommendations were derived from the earthquake initiating event. The need for a more detailed investigation of occurrence frequencies as a function of earthquake magnitude is evident from the extremely large uncertainty in the occurrence probability that was found for the initiating event -- one of the largest of all uncertainties considered in the Phase I study. Also, the ability of structures and equipment to survive earthquakes of very large magnitudes constitutes a grey area requiring much more detailed evaluation to provide a realistic basis for assessing the required probabilities of successful operation.

Of all the R&D recommendations identified during Phase I, those derived from the LOSP initiating event are deemed to be the most important since they were derived from a branch of the highest relative risk of any of the branches considered in the study thus far. Furthermore, the occurrence probability and consequence are moderate, although each is quite low on the absolute scale of magnitude. On the other hand, even though the reheater leak branches are next in line on the risk scale, the two associated R&D suggestions are judged to be of much lesser importance since the consequences are so small as to pose no safety hazard to the public. Although the computed risk values associated with the earthquake initiating events are low compared with the LOSP and reheater leak risk values, the R&D recommendations derived from the earthquake analysis appear to be second in importance to those derived from the LOSP. This follows from the large uncertainties involved in the probability assessments and the indication that the consequences could be among the largest of those assessed in the Phase I analysis. Even though the earthquake-caused consequences might be the relatively largest of the events analyzed thus far, the absolute consequence magnitudes are still well within generally accepted threshold criteria and regulations that have been established to ensure public safety.

7. R&D SIGNIFICANCE OF FISSION PRODUCT TRANSPORT MECHANISMS

7.1. BACKGROUND

As mentioned in Section 6.1, the fission product add-on task provided a ranking of isotopes and fission product parameters in terms of an index of relative safety R&D significance. The analytical approach and results are summarized in the present section.

Five events (identified in Section 7.2) were selected as the basis for the analysis, based on the results of the preliminary analysis. The study traced the transport phenomena of fission product isotopes from the inventories associated with normal operation to the radiological consequences evaluated for the selected events. Compared to the preliminary results, the present results are characterized by more realistic fission product inventories, more realistic modeling of retention and release mechanisms, and more realistic evaluation of radiological consequences.

The scope of the fission product add-on task includes a detailed evaluation of the median consequences (based on realistic analysis assumptions) and associated uncertainties for the five selected events. In addition, a model was developed and applied to assess the relative R&D significance of fission transport mechanisms, parameters, and data compared to other fission product parameters that constitute sources of uncertainty in the consequence assessments. This effort is unique to HTGR research in that the methodology is based primarily on risk assessment and statistical error analysis techniques.

7.2. EVENTS SELECTED FOR ANALYSIS

Five events were selected for study in that portion of the fission product add-on task that deals with accident conditions. These events were found to involve virtually all of the fission product technologies that have been developed to date.

Two of the five events were derived directly from the results of the preliminary analysis, which recommended that the more detailed Phase I analyses be concentrated on three initiating events found to have event tree branches of the highest relative risk: reheater leak, loss of offsite power, and earthquake. Reheater leak was one of the events selected for study in the fission product add-on task. The second event selected for study was the core heatup transient. This condition was chosen because it appears in some of the relatively high risk branches of LOSP and earthquake, branches which entail loss of main loop and core auxiliary cooling.

The remaining three of the five events selected for study under the fission product add-on task are relatively low risk design basis events: (1) slow depressurization, (2) rapid depressurization (DBDA), and (3) steam ingress from a main steam generator. These events provide bases for the CACS performance requirements, the design of the containment structure, and the design of the moisture monitoring and steam generator isolation and dump systems and may serve as bases for future design option studies. Further credence for the potential for design option studies is provided by the updated results displayed in Fig. 7-1. All points display the results of the most current consequence assessments as presented in Volume VI. Consequence uncertainties include the effects of weather variations. Points with uncertainty bands on the probabilities were derived from the most current Phase I analysis, while those without uncertainty bands on the probabilities were obtained from the earlier preliminary analysis. It should be noted that the design basis slow and rapid depressurization Branch A consequences fall in an area designated as a region of low dose,

Fig. 7-1. Summary probability-consequence plot

as does the design basis steam ingress event Branch AC. Events in this region are deemed to be of no essential safety significance because of the low dose levels and, hence, to be prime candidates for design option studies. Of course, it is recognized that the concept of low dose is somewhat subjective. However, the 0.1-mrem cutoff for the region of low dose represents about 1% of the whole body gamma dose that would be accumulated during 30 days as a result of normal background radiation.

7.3. CONSEQUENCE MODELING ASSUMPTIONS

Consequences of the events analyzed for the fission product program are based on realistic modeling of the effect of release to the environment of radioactive fission products from a standard 3000-MW(t) HTGR. The plant is assumed to be located at a typical site, which is defined by an exclusion area boundary (EAB) at a radius of 500 m, the outer boundary of the low population zone (LPZ) at a 2500-m radius, and an averaged distribution of site meteorology--the same as given in WASH-1400 (Ref. 7-1). In contrast to the preliminary scoping analysis, which focused on the 30-day whole body gamma dose at the LPZ, the analyses performed for the fission product program considered both the 2-hr EAB and 30-day LPZ doses of the following types:

1. External exposure, whole body penetrating gamma dose.
2. External exposure, whole body beta (skin) dose.
3. Inhalation iodine, contributing to thyroid dose.
4. Inhalation bone-seekers, contributing to bone dose.

These are the major dose considerations in accordance with current general safety practices throughout the nuclear industry.

7.4. MODEL FOR EVALUATING RELATIVE R&D SIGNIFICANCE

To identify the relative safety R&D significance of fission product release mechanisms for the five events analyzed, parameters having the strongest effect or causing the greatest deviations in the median consequence were determined. The change in dose, D, due to parameter variation from the median value to its upper 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 an efficient means of reducing the total uncertainty.

It is understood that a parameter can have safety R&D significance either by exerting a moderate effect on relatively high risk events or by exerting a pronounced effect on events of lesser risk. To cover both possibilities, it is necessary to compare event consequence predictions with some measure of safety significance. For the analyses performed for the fission product add-on task, the guidelines established by the Environmental Protection Agency (Ref. 7-2) were selected as a basis for 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 basic study objectives.

Depending on the severity of the accident, the health effect consequences could include immediate acute effects such as mortality and radiation sickness, or there could be later manifestations of radiation effects in the exposed individual or his offspring. Both of these aspects of the health effects evaluation (acute and latent effects) must be considered in formulating a realistic model for assessing the relative safety R&D significance of the fission product parameters.

Exposures to the whole body in the range of 100 to 1000 rem produce immediate clinical effects, which range from minor illness to possible death of an individual. Exposures of less than 100 rem 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. 7-3).

Studies performed thus far on potential HTGR event sequences 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 the objectives of this study. On the other hand, some HTGR event consequences are predicted to be 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 estimation of all possible health effects at this time (Ref. 7-4). However, data have been compiled to quantify observable health defects in exposed individuals. Thus, latent health effects for this study were established from the EPA data in terms of health effect conversion factors provided in Table 7-1.

Multiplication of the median dose, expressed in man-rem,* times the appropriate health effect factor listed in Table 7-1 yields a median consequence in terms of the health effect to an individual organ. Comparisons of the safety R&D 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 in each parameter. To incorporate these uncertainties mathematically, the multiplicative factor

$$(U_T - U_{T-j})/U_T$$

was used, where U_T is the total uncertainty in the organ dose for all parameters, and U_{T-j} is U_T without the dose uncertainty U_j due to a

* The details of the man-rem dose calculation are presented in Volume VI.

TABLE 7-1
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.

specific parameter x_j . This ratio is the fractional change that would occur in the total dose uncertainty as a result of the elimination of the uncertainty associated with the parameter in question. The computational formula used to evaluate U_{T-j} is described in Volume VI.

The ratio $(U_T - U_{T-j})/U_T$ provides a measure of the relative contribution of the uncertainty in parameter x_j 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 approaches U_T , the ratio tends toward unity.

The information presented above can conveniently be combined and expressed in terms of the computational formula

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

where F_{organ} is a factor that provides a measure of the R&D significance of a specific fission product parameter (that is, parameter x_j) based on considerations of one of the five selected events and of a specific organ (evaluated in terms of the whole body gamma dose, thyroid dose, bone dose, or skin dose*). An indication of total parameter significance, F_T , for a specific event can be obtained by summing the F_{organ} factors over the four

* In this study, the health effect of the lung dose was generally not considered. An estimate of the health effect factor for the lung dose is contained in the core heatup section of Volume V. Also, the lung dose contribution to the health effect factors of the remaining events was estimated as $\approx 10\%$. 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.

organs for the specific parameter and event in question:

$$F_T = \sum_{\text{organ}} F_{\text{organ}} = \left(F_{\text{whole body } \gamma} + F_{\text{thyroid}} + F_{\text{bone}} + F_{\text{skin}} \right) 30\text{-day LPZ} .$$

The 30-day LPZ factors are representative of the overall health effect for a particular event. A comparison of the F_T factors for the various parameters in a given event is used to identify which fission product transport mechanisms are most important with respect to the consequence uncertainties for that event.

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

$$\text{parameter overall safety R\&D index} = \sum_{\text{events}} (F_T \times P)_i .$$

The interpretation of this safety R&D 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 was made to relate to any biological damage assessment at this stage of analysis.

7.5. FISSION PRODUCT DATA R&D SIGNIFICANCE

Based on the computational approach presented in Section 7.4, values of the R&D significance factor F_T were evaluated for each of the transport

and retention mechanism parameters for each of the five events selected for inclusion in the fission product add-on task. For each event, the values of the F_T factors were ranked, parameter by parameter, in order of decreasing F_T values to identify the relatively most significant fission product transport and retention mechanisms. The detailed results for all five events are presented in Volume VI. Because of the relatively greater importance of the LOSP initiating event (as discussed in Section 6), only the results for the associated core heatup event are presented herein.

The values of the significance factors for the seven highest ranked parameters are displayed in Table 7-2 for the core heatup event. The highest of all F_T values for all parameters and all events is 2×10^{-4} --the value for the fuel particle release rate associated with the core heatup event. As a matter of fact, the first six ranked parameters in the core heatup event have higher F_T values than any parameter in the other four events. Considering consequences without regard to occurrence probabilities, it is concluded that the consequences in the core heatup event outweigh the consequences in other events.

The overall index of safety R&D significance for each of the fission product parameters was found by summing (over all five events) the products of the F_T and occurrence probability values evaluated for the events for the parameter of interest. These results were then ranked, parameter by parameter, in order of decreasing index values. Table 7-3 presents these results, together with an indication of the relative contributions of the five selected events to the tabulated values of the parameter significance indices. It is clear from the table that the five highest ranked parameters are so ranked because of the core heatup event. This leads to the conclusion that the core heatup event is of relatively high importance, a direct confirmation of a conclusion of the preliminary and Phase I analyses of the basic AIPA program.

TABLE 7-2
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 7-3
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 $10^{14} \sum_{\text{events}} F_T P.$

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

The results presented in Table 7-3 show that the three highest ranked 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 index of safety R&D significance for fuel particle release rate is almost double the index values for PCRV plateout and decay heat generation rate.

After the top three parameters, the safety R&D index factors drop off significantly. The fourth ranked parameter, containment plateout and fall-out, 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 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 higher safety R&D 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 follows from the whole body contribution during the core heatup event. Similarly, the importance of decay and buildup enroute follows from its contribution to the reheater leak 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 R&D index is also

displayed in Table 7-3. It is shown that the rate of fuel-steam reaction as a function of the controlling variables is the largest contributor. The last parameter listed in Table 7-3 is steam-graphite reaction.

Table 7-4 summarizes the top ten ranked parameters and the organ doses making the most significant contributions to the R&D index values shown in Table 7-3. Also shown in Table 7-4 are the key contributing isotopes to these important doses. Recommendations for future R&D efforts in the areas of testing and data correlations, release and retention mechanisms, and isotope releases should follow the priority rankings as displayed in Tables 7-3 and 7-4.

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TABLE 7-4
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

8. FUTURE WORK

The AIPA program on HTGRs is continuing. The present status report represents a major milestone, but a greater degree of completeness is desirable.

Further development on the methods of identification of initiating events is required. The choice of the original seventeen representative initiating events served the principal purpose of encompassing all sources of radioactivity in the plant and providing a broad spectrum of events. The subsequent results of the preliminary analysis provided a quantitative basis for emphasizing some areas for further study and not devoting effort to other areas.

As new initiating events are identified, they should, in general, be the subject of preliminary analysis. This will allow a more economical sorting of events to assist in determining which events and sequences are worthy of detailed study.

Detailed risk assessments will be done on a few initiating events where it is found important to study problems in more depth. There may be phases in the detailed analysis where a Phase II extension beyond Phase I is done only in cases where even greater refinement or more detailed elaboration of event sequences is worthwhile. Core heatup sequences provide a good example. Studies of a more detailed nature may also be done on technical problems that are found to have a potentially large effect on the significant probability or consequence calculations. An example of this, which has arisen in the current results, is a study on PCRV structural integrity in a core heatup event and its effect on the containment integrity.

Limited effort will be devoted to risk analysis on alternative design options affecting safety for which safety criteria are met and economics is a significant variable.

The development of applicable methodology will proceed, and ways to apply the risk methods to engineering efforts important to safety will be considered.

The primary objective of these studies will continue to be the generation of information and insights that provide guidance to safety R&D for HTGRs. The criterion used in this report is keyed to the risk (product of annual occurrence probability times consequence) for a given sequence in an event tree. New criteria have been developed by NRA, GA, and ANC to provide a framework for making the R&D recommendations in the future. These criteria are:

1. The risk is small enough that the accident can clearly be identified as not being of potential key public safety* or environmental concern.** NRA effort may be oriented towards evaluation of risk sensitivity in various safety-related plant options.
2. The risk appears to be large enough that such accidents would be of potential key public safety concern. Additional NRA effort

* GCR accidents of potential key public safety concern should be taken in this context as including those accidents which could potentially constitute a public risk on the order of or greater than the current upper limits of the envelope risk values as given in WASH-1400 or other such reputable overall reactor safety risk assessments.

** GCR accidents of potential key environmental concern should be taken in this context as including those accidents which could potentially constitute a public risk on the order of or greater than interpretation of the current upper limits of the envelope risk values as developed in WASH-1400 or other such reputable overall reactor safety risk assessments to the low consequence region (exposures less than 10^4 man-rem) where environmental rather than human effects are of primary concern.

should be expended on design options which would reduce either the likelihood and/or the potential consequences, and which can be practically implemented.

3. The uncertainty in the risk is large enough that one cannot determine conclusively if the accident would be of potential key public safety or key environmental concern. Further NRA effort should be expended to investigate the primary contributors to uncertainty with an objective of reducing the uncertainty to an acceptable level. Design areas which might be changed to significantly lower the risk should be identified. Once the objective is accomplished, either criterion 1 or 2 becomes applicable.
4. Certain additional safety R&D efforts which do not fit into the first three categories but nonetheless have general application to risk analysis will be indicated and may be performed. The further development of common mode failure methodology is an important example.
5. It is recognized that, at this stage of development, risk assessment methodology is not all encompassing. Hence, some additional safety R&D may be identified on other bases such as judgment of experts.

9. SUMMARY AND CONCLUSIONS

The primary objective of this study, to provide guidance for HTGR safety R&D by employing risk assessment techniques, has been accomplished for a representative set of initiating events. The guidance was provided in two parts: (1) the broad-scale R&D ideas developed in Volume IV and (2) the safety R&D significance of specific fission product parameters developed in Volume VI. A preliminary analysis of seventeen initiating events was performed first to select, for more detailed evaluation, the few events that would be the most significant from the point of view of safety risk. The work done for the detailed risk assessments of these events provided the basis for deriving the broad R&D recommendations and the identification of specific fission product parameters of interest.

The most important R&D recommendations are (1) refinement of the probability model of core cooling, (2) study of PCRV structural integrity during core heatup, and the effects on the containment, and (3) study of fission product release and plateout during core heatup. The specific fission product isotopes and transport mechanisms of greatest significance are (1) fuel release rate (during core heatup) of Xe-133 and Xe-135, and (2) PCRV plateout (during core heatup) of I-131 and I-133.

The risk associated with core heatup sequences leads to the recommendations above, even though this risk was found to be low in absolute terms. The AIPA core heatup analysis may be compared with the analysis of the maximum hypothetical fission product release (MHFPR) used in the Nuclear Regulatory Commission (NRC) licensing process (Refs. 9-1, 9-2), and it will be found that the NRC safety criteria were met for the Summit and Fulton sites.

Based on results obtained during the preliminary analysis reported in Volume III, it can be safely concluded that certain selected design basis events are not important from a risk standpoint. Two cases in point are the design basis depressurization accident (DBDA) and the moisture ingress into the primary coolant system. These two events expose the public to very small risks that are considerably less than those predicted for the design basis reheater leak, which meets regulatory safety criteria (as shown by acceptance in NRC licensing actions on the Summit and Fulton applications).

Another study objective is the consideration of alternative design options related to safety. Several alternative designs for moisture ingress protection were analyzed and are presented in Volume III. The results indicate that the use of common headering of steam pipes and the change from PPS to OPS instrumentation can reduce costs by millions of dollars while maintaining a high level of public safety. Hence, a potential cost reduction area has been identified.

An additional objective of this study is to develop and improve the risk assessment methodology. The results are presented primarily in Volumes II, IV, and VI. Some highlights of these developments are:

1. Improved techniques for accounting for common mode effects in the fault tree analyses.
2. Improved techniques for including intersystem dependencies in event tree analyses.
3. New methods for predicting operator error.
4. Refinements in computational capabilities to provide more realistic estimates of radiation doses to the public.
5. Improvements in the estimation of parameter uncertainties as well as the methods for taking the uncertainties into account in the assessments of probabilities and radiation doses.

6. Improvements in the display of results, thereby facilitating their proper interpretation and communication.

Concerning the objective of encouraging the use of the method by designers, progress has been made by disseminating information and giving presentations about AIPA methodology and results to designers who can make effective use of the technology. In addition, numerous presentations have been delivered to personnel of outside organizations including ERDA, NRC, National Laboratories, and utilities. A paper was presented at a Reliability Symposium in April 1975 (Ref. 9-3) and another at the American Nuclear Society 1975 summer meeting (Ref. 9-4).

Final conclusions about the absolute and relative nuclear safety of the HTGR will be reached with care. When using the results of this study to consider such a conclusion, it should be noted that uncertainties in present results are, in many cases, substantial; in addition, the study so far has been based on a representative, but not all-inclusive, list of initiating events. Improvements in these areas can be effected in future work. However, the results provide a reasonable and strong indication that the HTGR has excellent absolute and relative safety. This is brought out in Figs. 9-1 and 9-2, which define a risk envelope for the HTGR. Points that are identified in these figures relate to Phase I predictions for reheater leak, loss of offsite power, and earthquake.

Figure 9-1 summarizes the results of the study in a plot of annual occurrence probability (or frequency) versus consequence, in whole-body rem. Each point represents the median estimate for a specific possible sequence of events following the indicated initiating event, based on median weather. Also shown at each point are the corresponding uncertainty bands for the probability and consequence predictions, again based on median weather.

Fig. 9-1. Summary risk envelope for HTGR [3000 MW(t)] showing median and uncertainty estimates

Fig. 9-2. Summary man-rem risk envelope for HTGR [3000 MW(t)] showing median and uncertainty estimates

It is difficult to form judgments about the results shown in Fig. 9-1 because the whole body doses are less than the approximately 320 rem expected to produce no early fatalities (Ref. 9-5). However, there is a possibility of deaths in the long term resulting from the exposure of the population to relatively low dose levels. The effects themselves, primarily cancers, are manifested only after an interval of years or decades following the irradiation. A plot needed to provide a means for estimating the possibility of these latent cancers is provided in Fig. 9-2, where the consequence coordinate is whole body man-rem. (For all the exposure "pathways" examined, whole body exposure is found to be the most significant exposure mechanism for HTGRs.)

The long-term effects of nuclear radiation have been correlated by the BEIR Committee (Ref. 9-6). The correlation they propose is that, on the average, about 10,000 man-rem of irradiation may produce one cancer death during a period of about 25 years following irradiation. The Environmental Protection Agency (Ref. 9-7) have also interpreted the BEIR Committee information, but they have selected the conservative upper bound consequence value of two cancer deaths per 10,000 man-rem of irradiation. The former consequence value is plotted in Fig. 9-2 since it is more representative of median conditions.

From the results presented in Fig. 9-2, it is concluded that, over a meaningful range of frequencies (say, once per year to once in ten million years), no accident sequence considered in this study would cause death by latent cancer. Events that were even less frequent (less than once in a billion years) might be predicted to cause one or two latent cancer deaths during a period of approximately 25 years following the occurrence of such an event. This result would be overwhelmed by the normal occurrence of cancer-induced deaths which, during that same 25-year period, would typically be about 5700 in a population of comparable size (Ref. 9-8).

The study therefore finds that HTGRs are exceptionally safe under the many conditions covered by the study to date.

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