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Technical Considerations Related to Interim Source Term Assumptions for Emergency Planning and Equipment Qualification

S. J. Niemczyk
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HEALTH AND SAFETY RESEARCH DIVISION
OAK RIDGE NATIONAL LABORATORY

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Health and Safety Research Division

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S. J. Niemczyk
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CONTENTS

Page

LIST OF TABLES v

ACKNOWLEDGEMENTS vii

SUMMARY. ix

1. INTRODUCTION 1
 1.1 PERSPECTIVE 1
  1.1.1 Purpose. 1
  1.1.2 Equipment Qualification Versus Emergency Planning 1
  1.1.3 Accident Spectrum Approach 2
  1.1.4 Technical Bases Report 10
  1.1.5 Scope. 10
  1.1.6 Scale. 11
  1.1.7 Overall Approach 11
 1.2 PRESENTATION. 12
 References 14

2. QUALITATIVE DESCRIPTIONS OF SOURCE TERMS 15
 2.1 INTRODUCTION. 15
 2.2 RELEASES FROM CORE MATERIALS. 16
  2.2.1 Releases Inside the Reactor Vessel 16
  2.2.2 Releases Outside the Reactor Vessel. 18
 2.3 TRANSPORT THROUGH THE PRIMARY COOLANT SYSTEM. 20
 2.4 TRANSPORT THROUGH THE CONTAINMENT 24
  2.4.1 Natural Processes. 25
  2.4.2 Effects of Engineered Safety Features. 25
 2.5 DIFFERENCES AMONG REACTORS. 28
 2.6 ACCIDENT SPECTRUM 29
 2.7 SUMMARY 30

3. METHODS. 33
 3.1 INTRODUCTION. 33
 3.2 OVERALL APPROACH. 33
 3.3 INDIVIDUAL ACCIDENTS. 36
  3.3.1 Releases from Core Materials 36
  3.3.2 Transport in the Primary Coolant System. 38
  3.3.3 Transport Through the Containment. 40
 3.4 SPECTRUM OF ACCIDENTS 41
  3.4.1 Equipment Qualification. 41
  3.4.2 Emergency Planning 42
 3.5 METHODS VERSUS REALITY. 43
 References 46

CONTENTS (Cont'd)

	Page
4. EQUIPMENT QUALIFICATION	49
4.1 INTRODUCTION	49
4.2 DESIGN BASIS ACCIDENT SPECTRUM	49
4.3 RELEASES FROM THE CORE MATERIALS	51
4.4 TRANSPORT THROUGH THE PRIMARY COOLANT SYSTEM	53
4.5 TRANSPORT WITHIN THE CONTAINMENT	53
4.6 UNCERTAINTIES	55
4.7 SOURCE TERMS FOR REGULATORY GUIDANCE FOR EQUIPMENT QUALIFICATION	56
References	60
5. EMERGENCY PLANNING	61
5.1 INTRODUCTION	61
5.2 ACCIDENT SPECTRUM	61
5.3 RELEASES FROM CORE MATERIALS	63
5.4 TRANSPORT THROUGH THE PRIMARY COOLANT SYSTEM	66
5.5 TRANSPORT WITHIN THE CONTAINMENT	69
5.6 UNCERTAINTIES	71
5.7 SOURCE TERMS FOR REGULATORY GUIDANCE FOR EMERGENCY PLANNING	72
References	77
6. LIMITATIONS AND RECOMMENDATIONS	79
6.1 LIMITATIONS	79
6.1.1 Assumptions and Procedures	79
6.1.2 Illustrative Estimates	82
6.2 RECOMMENDATIONS	83
References	85
APPENDIX A. ACCIDENT DESCRIPTIONS AND TERMINOLOGY	A-1
APPENDIX B. FISSION PRODUCT RELEASE FROM FUEL	B-1
APPENDIX C. TRANSPORT THROUGH THE PRIMARY COOLANT SYSTEM	C-1
APPENDIX D. TRANSPORT THROUGH THE CONTAINMENT	D-1
APPENDIX E. SOURCE TERMS	E-1

LIST OF TABLES

Table		Page
1.1	Differences between the DBA approach and the probabilistic risk assessment approach.	3
2.1	Potential effects of some natural processes in the PCS during accidents.	21
2.2	Potential effects of some natural processes and some ESF's in the containment during accidents.	26
3.1	Problems in estimating magnitudes of source terms	44
4.1	Release fractions from core materials	52
4.2	Total release fractions for design basis accident for equipment qualification.	54
4.3	Impacts of problems in estimating magnitudes of source terms on equipment qualification accident sequences	57
4.4	Source terms for equipment qualification.	59
5.1	Release fractions from core materials	64
5.2	Summary of primary coolant system escape fractions.	67
5.3	Primary coolant system escape fractions for transient-initiated meltdowns	68
5.4	Containment escape fractions for some representative accident sequences	70
5.5	Impacts of problems in estimating magnitudes of source terms on emergency planning accident sequences	73
5.6	Estimated values of updated total containment escape fractions.	74
5.7	Estimated values of updated total containment escape fractions for generic accident classes.	75

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SUMMARY

The source terms recommended in the current regulatory guidance for many considerations of light water reactor (LWR) accidents were developed a number of years ago when understandings of many of the phenomena pertinent to source term estimation were relatively primitive. The purpose of the work presented here was to develop more realistic source term assumptions which could be used for interim regulatory purposes for two specific considerations, namely, equipment qualification and emergency planning.

The overall approach taken was to adopt assumptions and models previously proposed for various aspects of source term estimation and to modify those assumptions and models to reflect recently gained insights into, and data describing, the release and transport of radionuclides during and after LWR accidents. To obtain illustrative estimates of the magnitudes of the source terms, the results of previous calculations employing the adopted assumptions and models were utilized and were modified to account for the effects of the recent insights and data. Basically, some assumptions and calculations presented in the Reactor Safety Study (RSS; USNRC, 1975) and the reports of the Reactor Safety Study Methodology Applications Program (RSSMAP; Carlson et al., 1981; Kolb et al., 1981; Hatch, Cybulskis, and Wooton, 1981; Cybulskis, 1981; Wooton, 1981), as well as some proposed models and related calculations described in the Technical Bases Report (USNRC, 1981), were adapted for use.

Two accident spectra, one for equipment qualification and another for emergency planning, were considered. Only limited-core-damage accident sequences constituted the equipment qualification spectrum while both limited-core-damage and meltdown accident sequences constituted the emergency planning spectrum.

Releases of radionuclides and other species from the core materials before reactor vessel failure were estimated using temperature-dependent release rates derived by curve fitting to experimental data (Wichner, Kress, and Lorenz, 1981). The needed thermal-hydraulic estimates, as functions of time, were taken from the RSSMAP work (Wooton, 1981).

Releases after vessel failure were treated differently. Specifically, the total releases of materials after vessel failure were estimated with an empirically-derived equation relating the gas velocity and the melt temperature to the aerosol concentration in the sparging gases (Murfin and Powers, 1980; Wichner, Lorenz and Kress, 1981). The estimates of individual radionuclide releases at that time were obtained by combining element-dependent vaporization fractions with gas release rates (USNRC, 1975).

Transport through the primary coolant system (PCS) was estimated using perspectives gained from recently completed calculations of retention in the PCS (Gieseke and Kuhlman, 1981). The escape of any gaseous species was assumed to depend on the temperatures encountered in the PCS. The escape of any aerosol species was taken to be a function of both the aerosol concentration in the PCS and the residence time of the species in the PCS. Approximate aerosol residence times in the PCS for various types of accident sequences were based on calculations in the Technical Bases Report (USNRC, 1981) and on considerations of the RSSMAP work (Wooton, 1981).

Transport within and escape from the containment were considered using the assumptions and models presented in the RSS and the results of associated calculations made for the RSSMAP reports. In those considerations, the effects of both natural removal processes and engineered safety features were taken into account. The RSSMAP estimates were modified primarily to account for changes in the timing of the releases from the core materials assumed in the present study.

For some accident sequences previously estimated to dominate emergency planning considerations, in particular, for certain transient-initiated accidents and some small-break loss-of-coolant accidents, some of the radionuclide releases were predicted to be noticeably smaller than previously anticipated because of much larger radionuclide retention in the PCS. In contrast, some of the radionuclide releases to the environment for certain other dominant sequences were estimated to be larger than formerly assumed for two reasons: larger estimates of the releases from the core materials and later releases of some radionuclides from the core materials and/or coolant system. Source terms were estimated to be smaller for many of the less important emergency planning sequences because natural processes were predicted to be much more effective in preventing releases than had been previously assumed.

For equipment qualification, for sequences involving partial melting of the core, the estimates of the releases of some of the more volatile species to the containment were much larger than those currently in use but the estimates of the releases of some of the less volatile species were much smaller than those currently recommended. In contrast, for sequences involving only minor core damage, the releases of some of the more volatile radionuclides were smaller than those currently used but the estimates of the releases of some of the less volatile species were larger than those previously assumed. As was the case for emergency planning, the larger releases to the containment were the result of larger, later releases from the core materials, whereas the smaller releases to the containment were the result of smaller initial releases. For all types of accidents, the possible range of initial distributions of the radionuclides between the containment atmosphere and water in the containment was predicted to be much larger than that previously assumed. The broader range was the result of more realistic consideration of the possibilities.

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1. INTRODUCTION

1.1 PERSPECTIVE

1.1.1 Purpose

The purpose of the work reported in this document was to review some assumptions currently used for regulatory guidance related to light water reactors (LWR's). In particular, it was to consider the appropriateness of certain assumptions concerning post-accident radionuclide releases and behavior in the containment which are used for two distinct purposes:

1. equipment qualification, and
2. emergency planning.

Based on the review, the project was to provide recommendations for updated source term assumptions for use in each of those two areas. More exactly, the project was to describe the accident spectrum designated to be appropriate for considering each of the two indicated areas and it was to develop assumptions for estimating the magnitudes of the radionuclide releases for accidents within each of those spectra.

This document presents the detailed technical considerations which factored into the work for the project. A related report (NUREG/CR-2629; Niemczyk and McDowell-Boyer, 1982) summarizes the results of the project.

1.1.2 Equipment Qualification Versus Emergency Planning

In equipment qualification, the immediate concern is the potential hazard to the equipment inside the nuclear plant during and after an accident. If a not-too-serious accident occurs, the necessary equipment to keep the accident from progressing should be able to operate. If a more serious accident occurs, the appropriate equipment to monitor the conditions and to ensure prolonged confinement of the radioactivity within the containment should continue to function. Equipment qualification includes estimation of the potential radiation doses to the equipment-of-concern.

In emergency planning, the primary concern is the potential radiation hazard to the human population located in the vicinity of the nuclear plant at the time of an accident. If a serious accident occurs, certain members of that population might need to take some type of protective action to mitigate the possible radiation doses. Emergency planning includes estimation of the potential radiation doses to the human population, along with development of procedures which might be employed to implement various types of protective actions.

Obviously, the physical domains of interest for equipment qualification and for emergency planning are essentially nonoverlapping. Whereas the ultimate region of interest for the former purpose is inside the containment (or reactor building), the region of interest for the latter purpose is outside the containment (or nuclear plant).

Current regulatory guidance for source terms for equipment qualification is based on the concept of the Design Basis Accident (DBA). In contrast, current considerations for emergency planning are often based on a risk assessment approach. A comparison of the design basis approach and the risk assessment approach is presented in Table 1.1. [For a brief review of the history of regulatory practice in these two areas, the reader might consult Section C.3 of Appendix C of *Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions* (Pasedag, Blond, and Jankowski, 1981). For descriptions of the source terms currently used in both these areas, the reader might refer to Appendix E of this report.]

1.1.3 Accident Spectrum Approach

In this report, the assumptions in the current regulatory approaches for equipment qualification and for emergency planning are updated by "mechanistic" consideration of a spectrum of accidents appropriate for each area. In this mechanistic treatment, a plausible scenario is developed and utilized for each accident investigated. Such consideration of individual accidents leads to an improved description of the overall spectrum for each area. For equipment qualification and its associated design basis approach, such "realistic" consideration of the indicated spectrum leads to a more meaningful description of the

Table 1.1 Differences between the DBA approach and the probabilistic risk assessment approach

	DBA approach	Risk assessment approach
Accidents	DBA's only	Degraded core and meltdown accidents
Descriptions	Nonmechanistic[a] "Conservative"	Mechanistic "Realistic"
Framework	Deterministic (more severe accidents than DBA's are not credible; make the world absolutely safe from DBA's)	Probabilistic (all accidents are included in a weighted fashion according to their probabilities of occurrence; consider overall acceptable risk/time)

[a]The current regulatory guidance for DBA's is based on a non-mechanistic, conservative approach. In contrast, the approach for DBA's in this report is a "mechanistic," "realistic" one.

design basis. For emergency planning and its corresponding risk assessment approach, such realistic consideration of a broad spectrum leads to an updated basis for estimating the risk from both individual accidents and all accidents.

1.1.3.1 Equipment Qualification

1.1.3.1.1 Current Approach

Many current regulatory procedures, such as those for equipment qualification, are based on the concept of defense-in-depth against what were perceived some years ago to be the worst credible nuclear reactor accidents. The defense-in-depth approach includes both diversity and redundancy of safety functions and systems, along with multiple physical barriers to prevent releases of radioactivity to the environment. At the time the current regulations were formulated, the worst credible accidents were taken to be enveloped by the fictitious Design Basis Accidents (DBA's). In particular, the worst credible accidents, or the DBA's, were taken to include the following possibilities:

1. loss-of-coolant accidents (LOCA's),
2. rod ejection accidents for pressurized water reactors (PWR's) and rod drop accidents for boiling water reactors (BWR's); and
3. steam generator-tube ruptures and steamline breaks outside the containment for PWR's.

All of these accidents were assumed to result in less than complete meltdown of the reactor core. (See Section A.2 in Appendix A for general descriptions of each of these types of accidents.)

The postulated releases of radioactive materials for the DBA's were not developed on a mechanistic basis. Instead, the amounts of the radioactive materials released as the result of any accident included in the DBA envelope were taken to be large enough that they were thought to bound the amounts of materials which could be released in any credible accident. The population-of-concern for such accidents was assumed to be the human population. Consequently, to compensate for those radionuclides which would be released in any DBA accident in presumably small but unknown quantities, extra amounts of very radiologically effective radionuclides such as the radioiodines* were added to the anticipated DBA releases. Thus the postulated values of those source terms were thought to include a conservative margin. In no sense were the postulated releases given in the regulatory guidance assumed to be realistic.

Unfortunately, the releases postulated on consideration of the hazard to the human population were adopted for considering the hazard to other populations-at-risk. For example, the DBA source terms initially proposed for siting guidance were adopted for equipment qualification guidance.

The DBA used for both those purposes is the DBA-LOCA and involves the largest assumed releases of radionuclides of any of the DBA's. The characteristics for the DBA-LOCA are determined from the following assumptions:

1. substantial melting of the reactor core occurs, implying less than optimal performance of those systems designed to prevent melting,
2. containment integrity is maintained; and
3. the engineered safety features (ESF's) designed to mitigate the consequences function properly.

1.1.3.1.2 TID-14844

In current regulatory guidance for equipment qualification, the magnitudes of the radioactive releases after any potential accident within the design basis are taken to be encompassed by the releases suggested in TID-14844 (USAEC, 1962). The releases constituting the TID source term were taken to result from the DBA-LOCA, that is, from complete rupturing of a major coolant system pipe followed by a partial melting of the core and a release to the containment of some part of the radionuclide inventory of the core. Possible scenarios for the

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*Some of the radioisotopes such as some of the radioiodines result in a much larger radiation dose per unit of activity (e.g., μCi) than other isotopes.

accident were not considered. At the time the TID source term was suggested, not enough data existed to enable the radionuclide releases resulting from various accident conditions to be estimated realistically. Although much more information has become available over the years since it was formulated, the TID source term has continued to be part of the regulatory guidance for many purposes such as equipment qualification.

Since the current regulatory bases were established, it has become apparent that the TID source term is not applicable for at least some purposes. In particular, for those areas in which the radiation hazard to the population-of-concern is dominated by radionuclides other than the noble gases and the halogens (the major components of the TID source term), there have been indications that the TID source term might not be conservative.

1.1.3.1.3 Approach in This Report

For equipment qualification, the approach in this report is in part analogous to the current approach in that the design basis concept is utilized.* However, the approach taken here is to generate source terms using "realistic" analyses of possible worst-case accidents rather than using bounding (enveloping) analyses of undefined accidents. This approach is adopted because the consequences of some of the accidents investigated here are potentially large and because what is conservative is not always apparent. The accident analyses undertaken are then used to determine a more reasonably based envelope for the design basis source terms.

1.1.3.2 Emergency Planning

1.1.3.2.1 Current Approach

There is currently no regulatory guidance for emergency planning source terms. Thus, a "modified design basis" source term is used by

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*The restriction of the accidents considered for equipment qualification to just those within the current design basis was part of the definition of the scope of this project by the funding agency.

some. (In particular, the TID source term is used in combination with an assumption of massive breaching of the containment.) In contrast, the current regulatory guidance for environmental impact analyses is followed by others.

Current regulatory guidance for environmental impact analyses requires consideration of all possible accidents in the context of their probabilities of occurrence; that is, probabilistic risk assessments are indicated. Not only are lesser accidents, such as those within the design basis envelope, considered, but also much more severe accidents, such as those involving melting of the complete reactor core, are included.

In a probabilistic risk assessment, the consequences of various types of accident sequences representative of the total accident spectrum are estimated. Then, the overall risk for those accidents is calculated by weighting the representative accidents according to both their likelihoods of occurrence and their estimated consequences. Thus the approach attempts to put into perspective the relative importance of low-probability high-consequence accidents and that of high-probability low-consequence accidents, as well as that of accidents with all other possible combinations of probabilities and consequences. The starting point for such an assessment is the description of the source terms for the indicated accident sequences.

When the original licensing basis was being developed, little information was available about the likelihoods of occurrence of various types of possible accidents. In addition, as was already indicated, little information was available about the magnitudes of the associated potential releases of radioactivity. Thus a risk assessment approach was not reasonable. However, since the development of the original licensing basis, much more information has become available on both the probabilities of the various types of accidents which might occur and the amounts of radioactive materials which might be released as a result of any of them. Although some of this information has subsequently been factored informally into certain considerations of emergency planning, none of it has become a part of official regulatory guidance for emergency planning. Thus, unlike consideration of equipment qualification, some treatments of emergency planning involve the use of source term assumptions based on some of the more recent information. Unfortunately, some other treatments are still based on the source terms used for equipment qualification.

1.1.3.2.2 Reactor Safety Study and Reactor Safety Study Methodology Applications Program

In some current considerations of emergency planning, the radionuclide releases after any accident are taken to be described by the assumptions used in the relatively recent Reactor Safety Study (RSS; USNRC, 1975). The work performed for that study was the most comprehensive attempt to date to assess the relative contributions to the risk to the human population of various types of meltdown and other degraded core accidents. Because that work had as its goal the estimation of the risk to the human population, its emphasis was on accidents involving complete meltdown of the reactor core, although it also considered accidents involving limited core damage.

In that work, two specific nuclear plant designs were reviewed in detail. Possible accident sequences covering a large portion of the entire accident spectrum for each plant were delineated. Event trees and fault trees were constructed and estimates of the probabilities of various types of accidents were developed. The postulated accident sequences were grouped into nine PWR categories and five BWR categories, according to those features upon which the potential atmospheric pathway consequences depend. Phenomenological models were used to consider the behavior of the radioactive materials and to describe the transport and movement of the radioactive material both inside and outside the reactor containment for a spectrum of accidents for each plant. In addition, the consequences to the human population were described for representative accidents in each spectrum. Furthermore, the aggregate risk to the population from all the various types of potential accidents was estimated. Thus, the accident sequences dominating the risk to the human population for those two plant designs were identified.

Since the completion of the RSS, the Reactor Safety Study Methodology Application Program (RSSMAP) has applied the methods developed in the RSS to the consideration of the accident spectra for four other plant designs (Carlson et al., 1981; Kolb et al., 1981; Hatch, Cybulskis and Wooton, 1981; Cybulskis, 1981; Wooton, 1981). As in the RSS, both the probabilities and the magnitudes of the radionuclide releases of accident sequences potentially important for estimating the risk to the human population were quantified. Unlike the RSS, the overall risk to the human population was not estimated for any of the RSSMAP plants. To ensure consistency with the RSS, the basic assumptions in the RSSMAP regarding source terms were taken to be the same as those used in the RSS.

At the time the RSS source terms were generated, they were thought to be conservatively realistic, that is, they were thought to be chosen to err on the conservative side when not enough information was available to make a realistic assessment. Recently, however, it has been suggested that the source term assumptions used in both those studies, and therefore in some current considerations of emergency planning, might be highly conservative for a variety of reasons. Namely, recent research has filled in some of the gaps existing in the knowledge at the time the RSS was performed.

1.1.3.2.3 Approach in This Report

The approach in this report is to utilize the risk assessment framework currently used by some for emergency planning. The emphasis here is on updating the assumptions to be used for estimating the magnitudes of the associated radionuclide releases on the basis of relatively recent findings. As is required by either consideration of emergency planning for individual accidents or estimation of the risk from all possible accidents, the analyses are directed to obtain realistic or best estimate values, rather than conservative values, for the emergency planning source terms. Unlike the situation for a DBA analysis, best estimates for the individual accidents are used both for accident planning and for a risk analysis; they are not used to generate an envelope of the source terms for all relevant accidents.

1.1.4 Technical Bases Report

A recent report, entitled *Technical Bases for Estimating Fission Product Behavior During LWR Accidents*, was written by experts at several laboratories (Battelle Columbus Laboratories, Oak Ridge National Laboratory, and Sandia National Laboratories) to summarize all the post-RSS work relevant to estimating the magnitudes of the radionuclide releases for LWR accidents (USNRC, 1981). Its views are thought by many to represent the state-of-the-art of a number of non-probabilistic aspects of source term estimation.

The work performed for the Technical Bases Report considered in detail many of the factors affecting the releases of radionuclides from the core materials, the transport of those radionuclides in the primary coolant system, the subsequent transport in the reactor containment, and the eventual escape of some of the radionuclides into the environment. Portions of several hypothetical accident sequences were considered for each of four basic types of nuclear plants. Estimates of the effects of the important factors affecting the magnitudes of the radionuclide source terms were made separately for the initial releases, the transport in the coolant system and the transport in the containment. The interactions between the effects in the various portions of the overall problem were not treated.

1.1.5 Scope

The scope of this study includes all major factors affecting the magnitudes of radionuclide source terms for light water reactor accidents involving core damage. The emphasis is on determination of assumptions and/or procedures which can be used to estimate those magnitudes for certain regulatory purposes.

Some of the assumptions and procedures adopted here for estimating the magnitudes of the source terms for accidents are based on the work in the Technical Bases Report. However, the scope of this project goes farther than that of the Technical Bases Report in that the methods recommended in that study are applied to complete accident sequences rather than to just portions of them. Furthermore, this study quantifies the effects of factors only alluded to in the Technical

Bases Report. In particular, it considers interdependencies of effects not previously treated. In addition, it attempts to place the new source term assumptions and illustrative estimates in an appropriate framework.

Some of the other assumptions and procedures adopted here are based on the work in the Reactor Safety Study. The scope of this project is much different than that of the Reactor Safety Study in that the goal here is consideration of only source term magnitudes and not the estimation of the overall risk. In addition, the emphasis here is on obtaining source terms suitable for regulatory purposes. Furthermore, this project is concerned not only with source terms appropriate for considering the hazards to humans but also with source terms appropriate for considering the hazards to equipment.

1.1.6 Scale

Although the scope of the project reported here was rather broad, the scale of the project was very limited. Unfortunately, many aspects of estimating source terms are extremely complicated and have not been adequately treated in the past. Because of the limited scale of this study, it was not possible to consider most topics at a greater level of detail than had been done previously. In addition, it usually was not possible to develop new approaches for problems which had been either inadequately or even inaccurately treated in the past.

1.1.7 Overall Approach

To develop assumptions appropriate for estimating the amounts of radionuclides which might be released during various reactor accidents, previous source term work, as summarized in both the Reactor Safety Study and the Technical Bases Report, was reviewed. Based on this review, a set of assumptions and procedures was adopted and/or developed for the estimation of post-accident source term magnitudes. To test the overall approach and to illustrate its use, it was applied to obtain estimates of radionuclide releases for a broad spectrum of accidents.

1.2 PRESENTATION

The body of this report contains a summary of the important issues addressed and the results of this study. The appendices of this report contain detailed discussions of the myriad of considerations which factored into obtaining those results.

Chapter 2 describes the processes concerning the release of radionuclides from the core materials and the subsequent transport of those materials within the nuclear plant which are relevant for estimating the consequences of various accident sequences. The emphasis is on presenting a qualitative description of the most important factors affecting the source terms.

Chapter 3 summarizes the approach which has been used in this study to estimate source terms for a spectrum of light water reactor accident sequences. It reviews both the procedures used to describe individual accidents and the considerations used to place those accidents in an accident spectrum framework.

Chapters 4 and 5 present the source term estimates which have been obtained in this study for illustrative purposes. In addition, they consider the effects of such source term estimates on equipment qualification and emergency planning.

Chapter 6 outlines some of the limitations of the work presented here. In addition, it lists some suggestions for future work.

Appendix A reviews the terminology used throughout the report for discussing reactor accidents. In addition, it includes descriptions of many the accident sequences specifically investigated in this study.

Appendices B, C, and D contain discussions of the detailed considerations given to the releases of radionuclides from the core materials, the behavior of the released materials in the primary coolant system, and the transport of the materials within the containment, respectively. Each of these appendices reviews the previous and on-going work relevant to source term estimation and outlines the development of the updated assumptions proposed in this report for interim consideration of source terms for regulatory purposes. In addition, each of these appendices includes detailed numerical illustrations of the adopted assumptions in the form of release or escape fractions. Furthermore, each appendix discusses the shortcomings and limitations of the knowledge used to develop the updated source term assumptions and the related illustrative estimates.

Appendix E summarizes the total source terms estimated from the work in the previous three appendices. In particular, it combines the estimated release and escape fractions from the previous three appendices into a total release fraction, or source term, for each radionuclide and each accident sequence considered. In addition, it discusses the overall impacts of the uncertainties associated with those source terms.

This report differs from the other report for this project, NUREG/CR-2629 (Niemczyk and McDowell-Boyer, 1982), in that whereas the other report emphasizes just the assumptions which, based on this project, are thought to be most appropriate for source term estimation, this report emphasizes the rationale used to choose the adopted assumptions from among the many possibilities. This report also considers in much more detail the factors affecting the use of the adopted assumptions to obtain numerical estimates for source terms. In addition, this report contains the details of the illustrative estimates.

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2. QUALITATIVE DESCRIPTIONS OF SOURCE TERMS

2.1 INTRODUCTION

Releases of radionuclides from the core materials could occur as a result of many types of accidents. In lesser accidents, such as limited-core-damage accidents, restricted overheating of at least some portion of the core usually would occur. In more severe accidents, such as meltdown accidents, extensive overheating of almost the entire core would occur.

A limited-core-damage accident typically would result from a basic system failure such as a coolant line break or a transient event (event in which either the reactor power increases too much or the heat removal capacity of the coolant system drops too low). Typically, the requisite engineered safety features (ESF's) would function as anticipated and would soon return the cooling of the core to normal. Although a meltdown accident would usually result from one of the same basic accident initiators (coolant line break or transient) as a limited-core-damage accident, unlike such an accident, it would be accompanied by a failure of one or more of the ESF's required to maintain or restore adequate cooling. Thus the ESF's could not return the reactor to normal conditions and the core would melt "completely." An accident intermediate between a "typical" limited-core-damage accident and a complete meltdown accident, for example, an accident such as the one at Three Mile Island, frequently would result from the one of the basic accident initiators, accompanied by partially impaired performance of one or more of the requisite ESF's. Consequently, the cooling might not be restored to an adequate level until after severe damage had been done to the core.

The amount of radioactivity released from the core materials during any accident would depend upon the length of time and the extent to which the core was overheated. For accidents in which the cooling was soon restored to an adequate level, radioactive releases could be expected to be relatively small. However, for accidents in which the system was either slowly or never restored to adequate cooling conditions, the radioactive releases could be substantial.

This report considers the entire spectrum of accidents from those involving relatively little core damage, and therefore small releases of radionuclides, to those involving potentially large fractions of the radioactive content of the core, and therefore large radionuclide releases. This chapter reviews those factors which affect the magnitudes and the rates of the radioactive releases resulting from such accidents. (More detailed descriptions of the qualitative factors affecting the source terms for such accidents are presented in Appendices B–D of this report.)

2.2 RELEASES FROM CORE MATERIALS

During any accident involving core damage, radioactivity would be released from the core materials into the reactor pressure vessel (RPV). In addition, in accidents in which the core melted through the RPV, radioactivity could be released directly into the containment.

2.2.1 Releases Inside the Reactor Vessel

Releases of radionuclides from the core materials could occur as the result of several different processes. The relative importance of each of these processes in determining the overall releases would depend on the nature of the processes, the course of the accident, and the radionuclides being considered.

Most accidents of concern in this report would result in a loss of coolant water from the reactor vessel so that at least part of the core would become uncovered. This, in turn, would permit overheating of the core. If, as would be typical, the region around the core were filled with water vapor as the core were heating up, then steam-metal reactions would occur, resulting in the evolution of substantial quantities of both hydrogen and heat. The heat released by such reactions would be on the order of the decay heat of the reactor and thus could substantially increase the rate of heating of the core. The reducing atmosphere provided by the hydrogen* and the steam would affect significantly the chemical forms of some of the released radionuclides.

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*The heated zirconium in the fuel rod cladding would be the initial source of reduction.

In contrast, if the region around the core were dry during heatup, then only the decay heat from the core would be available for heating and so the rate of heatup would be relatively slow. In addition, if the region were dry, the atmosphere in the core would not necessarily be a reducing one. Thus different chemical forms might be expected.

As the core uncovered and heated up, the fuel rod claddings would rupture, allowing the release of the gases which had accumulated in the fuel-cladding gaps. Such releases would consist primarily of noble gases and other volatile species. For a given rod, this release would be essentially instantaneous. However, inasmuch as different rods could be expected to burst at different times, the overall fuel-cladding gap release would occur over a period of time. That release would be expected to form a relatively small component of the total radioactive release for any meltdown accident but could form a significant part for other, lesser accidents.

The rupture of the cladding also would permit radionuclides which diffused or otherwise escaped from inside the fuel rods after rupture to be released. As the temperatures increased, the rate of this release, and hence the amount of material released in this manner, would also increase. This "diffusion" release would be relatively small for most meltdown accidents but could be significant in accidents involving substantial periods of elevated core temperatures somewhat less than those resulting in melting.

As the temperatures increased farther, the fuel would begin to melt. During the melting, some of the more volatile components could be evaporated from the various liquid surfaces. Both structural and core materials would be included in this release. The slower the overall process was, the more structural materials would be melted and incorporated into the corium (molten core plus structural materials).

As the accident progressed farther, some or all of the melted materials could fall ("slump") or otherwise move into the bottom of the RPV. If the molten material formed a single mass there, the rates of releases of all the radionuclides might be substantially decreased because of the much smaller relative surface area of the material in the bottom of the reactor vessel than in the fuel rods.

Because of the large temperature gradients within the core during most accidents, the core would melt on a region by region basis. Thus, like the gap release, the total release of any given radionuclide or other material would occur over a period of time. However, in general, the more volatile radionuclides would tend to be released early in the heatup and melting. In fact, some of them would be totally released long before the core completely melted.* Consequently, large quantities of such radionuclides could be released even for accidents involving less damage than complete meltdown. In contrast, the less volatile radionuclides would tend to be released toward the end of melting. Therefore, they would not be released in large quantities unless substantial melting of the core occurred. In a complete meltdown, their releases would continue after RPV melt-through. Indeed, the largest quantities of at least some of them could be released after that melt-through.

Any time after the fuel cladding failed, water might contact the core materials. For any accident terminated prior to complete melting of the core, this would always be the situation. Because of the elevated temperatures of the fuel at the time of contact, significant fragmentation of a sizeable fraction of the fuel might occur. Both the initially elevated temperatures and the large surface areas resulting from fragmentation could cause substantial leaching of some of the radionuclides to occur very rapidly. Such a leach release would be highly sequence and element dependent. It could be especially important for accidents involving both substantial cladding failure and less than complete melting of the core.

2.2.2 Releases Outside the Reactor Vessel

If the melting continued, the corium would melt its way through the bottom of the reactor vessel and fall onto the containment basemat. The interaction of the melt with the concrete basemat could result in

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*In the case of a complete meltdown, it is conceivable that if melting in the core region occurred rapidly enough, a certain fraction of the core might not even melt. Thus, that fraction would not be available for some types of radionuclide releases.

the release of large quantities of materials into the containment atmosphere. Both sparging of gases through the melt and evaporation of materials from the melt would be expected to be significant for such release of materials, with sparging generally being more important. Not only structural and core materials but also concrete decomposition products would be included in this release, with the latter component dominating the mass of the release.

If water were present in the reactor cavity at the time of RPV melt-through, the melt might be quenched and so the releases of radionuclides might be delayed until the water had boiled or drained off the melt. At such time, however, the releases could resume. Eventually, the molten mass would cool due to both loss of decay heat and incorporation of structural materials and concrete residua. Although complete cooling could take months or years, it would probably only take several hours at most for the mass to cool sufficiently that further radionuclide evolution would be negligible. Therefore, evolution of radionuclides would taper off relatively rapidly after RPV melt-through.

Another possibility if water were present in the reactor cavity at the time of RPV melt-through would be a steam explosion. Such an explosion could result in the rapid production and release to the containment of finely divided corium particles from a large fraction of the mass of the corium. The release of many radionuclides would be significantly enhanced by such particle formation due to the increased surface area. Alternatively, such a steam explosion could result in the scattering of substantial masses of the corium from the main corium mass. The scattered masses might be cooled more rapidly than the main mass and consequently the potential for release of some radionuclides could be decreased.

Relative to the reducing atmosphere typically found in the core region, the atmosphere in the containment might have more of an oxidizing character.* Thus the chemical forms of certain species released in the

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*This would not apply to the atmosphere found inside inerted drywells of Mark I BWR's (boiling water reactors) before containment failure.

containment might be much different than those released in the core region. In addition, an oxidizing atmosphere would greatly enhance the release of certain radionuclides if a steam explosion occurred.*

2.3 TRANSPORT THROUGH THE PRIMARY COOLANT SYSTEM

During an accident, radionuclides released from the core region would enter the primary coolant system (PCS). In many accidents, such radionuclides would be swept by steam and/or water flows from the PCS into the containment. However, in some accidents, significant amounts of those radionuclides could be retained in the PCS as the result of various types of interactions. For example, any released gases could adsorb or condense onto particulates and coolant system surfaces, react chemically with other species in the coolant system atmosphere or with coolant system surfaces, or dissolve in and/or otherwise react with any water present in the system. The particles released from the core could agglomerate onto other particles and eventually be removed from suspension, plateout onto coolant system surfaces by various processes, or react chemically. Any of the removed material could subsequently be resuspended, revaporized, and/or otherwise re-released so that it could be entrained in the coolant system fluids and subsequently transported out of the PCS. (See Table C.1 in Appendix C for a detailed review of the major natural processes in the PCS and their effects; see Table 2.1 for a summary of that review.)

A major means of retention in the PCS for certain radionuclides in some accidents would involve aerosol phenomena. In particular, if the accident conditions were such that high concentrations of aerosols were present, those radionuclides residing on or in particles could be retained in the PCS as the result of various aerosol processes. Inasmuch as removal by such processes could increase disproportionately with the total aerosol concentration, those accidents in which large quantities

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*A steam explosion also could occur during the slumping of the molten core into the bottom of the RPV. Inasmuch as the atmosphere in the RPV would typically not be an oxidizing one, the enhancement of the release of certain radionuclides would probably not be as large as in a comparable steam explosion in the containment.

Table 2.1 Potential effects of some natural processes in the PCS during accidents

Processes	Effects[a]
Aerosol agglomeration and deposition	Low aerosol concentrations would usually not be rapidly decreased by these processes; however, high aerosol concentrations could be rapidly decreased in the PCS if steam velocities were low
Condensation onto (and evaporation from) particles	Accidents with low temperatures in the PCS could involve substantial deposition of gases onto aerosols if enough aerosols were present; accidents with high temperatures throughout the PCS would not involve such deposition.
Condensation onto (and evaporation from) surfaces	Accidents with low temperatures in the PCS could involve substantial deposition onto surfaces if large enough surface areas were available; accidents with high temperatures throughout the PCS would not involve such deposition.
Chemical reactions and radioactive decay	The chemical form of a species could change and thus alter the likelihood of retention in the PCS; interactions with aerosols or PCS surfaces could affect retention.
Scrubbing	Passage through water could substantially reduce both the gases and aerosols escaping from the PCS.

[a]For details, see Table C.1 of Appendix C.

of aerosols would be generated could frequently be affected much more noticeably than those in which only small quantities were produced.

Whether a given radionuclide was present as an aerosol would depend on its volatility. Thus, in part, the retention in the PCS would depend on the volatility of the radionuclide. In general, the more volatile elements would tend to enter the PCS and transport through it as gases while the less volatile elements would tend to rapidly condense into aerosol particles and transport as such. The physical phases of the elements of intermediate volatility would be functions of the thermal conditions in the PCS.

The extent of retention of any radionuclide in the PCS would depend not only upon the volatility of the radionuclide but also upon several accident characteristics, including the length of the path through the system, the temperatures of the fluids and surfaces in the system, the velocities of gases and particulates through the system, the rate of release of aerosols into the system, and the presence (or absence) of water in the dominant pathway(s) through the system. Shorter physical paths, higher temperatures, higher velocities, and lower aerosol generation rates would usually tend to decrease the likelihood of retention in the coolant system for most species. In contrast, the presence of water would increase the amount of retention in the coolant system of most species.

The dominant paths which the radionuclides would follow through the coolant system generally would be determined by the location of the pipe break in the case of a loss-of-coolant accident (LOCA) or by the nearest relief or safety valve in the case of a transient-initiated event. Large hot-leg pipe breaks usually would result in the shortest paths to the containment (or drywell) and thus the largest releases of radioactivity escaping to the containment, if other conditions were equivalent.

The temperatures in the primary coolant system would depend upon the rate of heatup and/or melting, with sequences having more rapid heatup (melting) frequently being expected to have both higher temperatures and larger temperature gradients in the coolant system. (If heatup [melting] occurred more slowly, then some of the heat added to the coolant system would be dissipated to other locations during the course of the accident.)

High steam temperatures in the PCS would tend to prevent the condensation of compounds of high and intermediate volatility onto suspended aerosols; likewise, high surface temperatures in the PCS would tend to discourage the condensation of those compounds onto surfaces. Thus high temperatures in the PCS would tend to preclude condensation of large quantities of both highly and moderately volatile compounds there. For accidents involving less than complete meltdown of the core, the temperatures in the PCS, especially in those regions far from the core region, would be relatively low and so might permit deposition of large amounts of such compounds. For accidents involving meltdown, the temperatures in at least some of the PCS would generally be high and would tend to discourage condensation of such compounds in those regions. (Although condensation in the PCS might occur early in such an accident, the deposited radionuclides would often be re-evaporated and then flushed from the PCS as it heated up during the accident. Alternatively, however, such deposited radionuclides might react with the PCS surfaces as they heated up and thereby be retained.)

In regions of relatively low temperatures, whether deposition occurred onto aerosols or onto surfaces would always depend in part on the relative surface areas presented by those two media. If deposition were onto aerosols, the radionuclides could still be swept from the PCS with the aerosols. However, if deposition were onto surfaces, permanent retention in the PCS would often be more likely.

The velocities of radionuclides through the coolant system would depend upon the driving forces. If water were present around the core during overheating, the large quantities of hydrogen and steam evolved during heatup would tend to rapidly flush any radionuclides from the core region through the coolant system. In contrast, if water were not present, only expansion of the gases due to heating would drive the radionuclides from the core region; this process would be relatively slow, with substantial fractions of the generated radionuclides often not escaping into the containment until long after their initial releases. In many accidents involving meltdown, steam flows and velocities through the coolant system would be relatively high both early in the accident as the water was being boiled off from around the core and later in the accident as water was being evaporated out of the bottom of the RPV due to slumping of the melt into it. The steam flow rates and velocities at other times would be highly dependent on both the accident scenario and the reactor design.

LOCA's, especially ones involving large pipe breaks, often would have relatively large steam velocities and therefore short radionuclide residence times in the PCS. The retention of aerosols in the PCS would frequently be small for such accidents. In contrast, some transient-initiated accidents and certain small LOCA's typically would have low steam velocities, and consequently long radionuclide residence times in the PCS. For some such accidents, the delay could be such that the majority of the radioactivity would not even be released to the containment unless and until rupture of the coolant system occurred, for example, by RPV melt-through. The retention of aerosols and some gases in the PCS could be significant in such accidents.

The aerosol generation rates would depend upon the rate of release of materials into the PCS. Those materials potentially forming aerosols would be composed of moderately and less volatile radionuclides, control rod components and structural materials. Higher aerosol generation rates would tend to result in higher aerosol concentrations in the PCS and thus could promote sharply enhanced removal of aerosols within the PCS.

The presence of liquid water in the dominant flow path(s), for example, in the pressurizer quench tank in a transient-initiated accident in a pressurized water reactor, would permit scrubbing of some of both the more soluble gases and the aerosol particles from the fission product stream. In addition, it might provide an obstacle to the flow to the containment for all radionuclides.

2.4 TRANSPORT THROUGH THE CONTAINMENT

The materials entering the containment indirectly via the primary coolant system would consist of mostly core structural materials and fission products. In contrast, the materials entering the containment directly would be composed primarily of concrete residua, along with much smaller amounts of fission products and structural materials. Natural processes, as well as the operation of ESF's, would affect the removal of radionuclides from the containment atmosphere.

2.4.1 Natural Processes

Both the gases and the particulates released to the containment would undergo basically the same natural processes as the materials initially released to the coolant system. (See Table D.1 in Appendix D for a review of the major natural processes in the containment and their effects; see Table 2.2 for a summary of that review.) The principal differences would be due to the dissimilarities in the physical and chemical conditions present in those two places. The conditions in the containment would depend upon both the accident initiators and the performance of various ESF's. These, in turn, would be very accident specific.

If the core melted through the RPV, the region just above the melt would be very hot. However, in general, for any accident the temperatures in most of the containment would be much lower than those in either the core region or the primary coolant system. Thus more species would be present as aerosols in the containment than in the coolant system. However, due to both retention in the coolant system and dilution in the containment atmosphere, the total aerosol concentrations in the containment generally would be much lower than those in the coolant system. (They might not be lower if aerosol residence times, and therefore the aerosol concentrations, in the PCS were relatively small.) As a result, the concentration-dependent removal processes which frequently would dominate the behavior of the aerosols, and thus the behavior of the associated radionuclides, in the coolant system often would be much slower in the containment (see Table D.6 in Appendix D).

2.4.2 Effects of Engineered Safety Features

During any accident, various ESF's might operate (see Tables D.2 and D.3 in Appendix D for a review of the major ESF's and their effects;

Table 2.2 Potential effects of some natural processes and some ESF's in the containment during accidents

Processes or ESF	Effects[a]
Aerosol agglomeration and deposition	Low aerosol concentrations would usually not be rapidly decreased by these processes; however, high aerosol concentrations could be rapidly decreased by these processes if the containment were unruptured.
Condensation onto (and evaporation from) surfaces	Substantial deposition onto surfaces would be expected for many gases.
Containment leak-tightness	An unruptured containment would retard releases of radionuclides to the environment and thereby give natural processes and ESF's time to be effective.
Containment sprays	Operation of sprays would reduce rapidly high aerosol concentrations; sprays generally would be effective in lowering most radionuclide concentrations in the containment atmosphere.
Suppression pool	Passage through a suppression pool could reduce substantially the amounts of aerosols and some gases escaping.
Ice-bed condenser	Passage across unmelted ice beds would reduce substantially the concentrations of many radionuclides in the containment atmosphere.
Filter systems	Passage through unclogged filters could reduce substantially the concentrations of both aerosols and some gases in the containment atmosphere.
Chemical reactions	Interactions with surfaces could lower the concentrations of some radionuclides in the containment atmosphere; interactions with chemical additives (in sprays, ice beds,...) could hinder escape or return of some species to containment atmosphere.

[a]For details, see Tables D.1 and D.2 in Appendix D.

see Table 2.2 for a summary of that review.) For any accident, one of the most important ESF's would be containment leak-tightness. The containment would tend to confine the radioactivity and thus permit time for both natural processes and processes due to the operation of the other ESF's to remove radioactivity from the containment atmosphere. Consequently, an unbreached containment usually would result in substantially lower amounts of most radionuclides being released to the outside environment than a breached one. For example, if severe rupturing of the containment occurred early in the accident scenario, a large fraction of all the radionuclides released to the containment could escape to the environment. In contrast, if severe rupturing occurred much later in the scenario, then large fractions of only the more volatile and less reactive radionuclides, for example, the noble gases, might escape. Likewise, if only a relatively small leakage from the containment occurred throughout the accident, then significant fractions of only the less reactive, more volatile species would tend to escape.

Among the other ESF's which might affect the amounts of radioactivity escaping would be containment sprays, suppression pools, and ice-bed condensers. During accidents, operation of any of these features would tend both to lower the pressure in the containment and to remove radioactivity from the containment atmosphere. Consequently, any of these features would tend to lower the amounts of radioactivity released to the environment outside the nuclear plant. These various types of pressure suppression systems would, in general, be effective in substantially lowering those amounts escaping to the outside provided these systems were still operating during the releases from the core materials and/or from the PCS. The major exceptions would be those accidents involving early catastrophic breaching of the containment and those accidents in which the dominant radionuclide pathways would bypass the containment. For consideration of the environment inside the plant, the primary effect of such ESF's would be to redistribute the radionuclides within the containment. The various suppression systems would tend to transfer the radionuclides from the containment atmosphere to the water in the reactor building and to a lesser extent to the surfaces in the system.

Other ESF's, such as the containment recirculation filter systems and the standby gas treatment (filter) systems, would tend primarily to remove radioactivity from either the containment atmosphere or from the escaping gases. These ESF's would be most effective for accidents in which the containment remained unruptured. In contrast to the pressure suppression ESF's, the filter systems would usually be effective in substantially lowering the total amount of the radioactivity released only for less severe accidents. In general, the filter systems would tend to localize some of the activity in the filter systems themselves.

2.5 DIFFERENCES AMONG REACTORS

The magnitudes and the probabilities of the various releases which might occur at a given site are both dependent on certain features of the nuclear plant under scrutiny. In particular, they depend on factors including the size and the status of the core, as well as the engineered safety features (ESF's) of that plant (see Table D.3 in Appendix D). In addition, they depend upon the basic designs of the core, the coolant system and the containment(s). Furthermore, they are a function of the detailed construction characteristics of the plant. All of these factors vary widely among plants.

As a result of all these differences, the magnitudes and the probabilities of the potential accidental releases differ substantially from plant to plant. Indeed, the types of accident sequences which dominate the risk to any population-of-concern at certain plants are frequently negligible contributors to the risk to the analogous populations at other plants. Furthermore, the anticipated relative magnitudes of the radionuclide releases resulting from a given type of accident sequence vary greatly among plants. Likewise, the probability for a given type of accident sequence varies widely even among apparently similar plants.

2.6 ACCIDENT SPECTRUM

Of the many accidents which might occur at any plant, only a very limited number of types of these accidents are typically important for considering either the worst possible consequences or else the overall risk to any population-of-concern. For example, in a design basis accident approach such as that used for equipment qualification, the only important accidents in the accident spectrum are those which result in releases of radionuclides which bound the possible releases for all accidents within the design basis. In a risk assessment approach such as that used for certain emergency planning considerations, the only important accidents are those which dominate the risk, that is, those whose combined likelihoods of occurrence and potential consequences make them among the riskiest.* In contrast, a much larger number of types of accidents are typically relevant for considering the potential consequences of all possible individual accidents to the same population-of-concern. Thus, for detailed emergency planning considerations for individual accidents, a broad, relatively detailed spectrum of accidents must be investigated.

The ranges of each of the accident spectra are delimited by their worst-case and their best-case accidents. For equipment qualification, the worst case for most equipment involves partial meltdown of the core, rapid airborne transport through the coolant system, and limited leakage from the containment building.† The best case involves minor core damage, slow waterborne transport through the coolant system and extensive leakage from the containment. In contrast, for emergency planning, the worst case involves a complete meltdown of the core accompanied by substantial breaching or circumventing of the containment early in the

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*This assumes that the bounding cases and the risk-dominant sequences can be easily identified. If this is not the situation, then a larger portion of the spectrum of possible accidents must be investigated.

†For at least some monitoring equipment, the worst case would involve complete melting of the core, rapid transport through the coolant system and limited leakage from the containment. However such equipment is not considered in this report.

accident. The best case involves minor core damage accompanied by very limited leakage from the containment.

The relative importance of any accident in a design basis spectrum such as that considered for equipment qualification depends on only the magnitude of the resultant radionuclide releases. The relative importance of any accident in a risk assessment spectrum such as that considered for certain emergency planning purposes depends on both its probability of occurrence and the magnitude of the resultant radionuclide releases. In contrast, the relative importance of any accident in a spectrum such as that considered for emergency planning for individual accidents depends only on the magnitude of the resultant radionuclide releases.

2.7 SUMMARY

The fraction of any radionuclide released from the core materials during an accident could range from a negligible one to nearly the entire core inventory, depending on the radionuclide and the accident scenario. More volatile radionuclides, such as noble gases and halogens, would always tend to be released more completely than less volatile ones. In meltdown accidents, almost the entire inventory of some of the more volatile radionuclides would be released before RPV melt-through while only small fractions of the less volatile radionuclides would be released by that time. Larger amounts of some of the less volatile materials could be released after that time, although the total fraction of those materials released typically would be small. Accidents involving less heating of the core materials would tend to result in smaller releases of all radionuclides.

In general, radionuclides present as gases would tend to pass through the primary coolant system unaffected, except for radioactive decay, although some of those radionuclides could undergo substantial retention in the PCS as the result of condensation or chemical reactions. In contrast, most of the radionuclides present as aerosols could undergo substantial attenuation if there were large delays in the movement through the PCS, with the fraction of any aerosol retained in the PCS ranging from a very small one to a substantial fraction, depending on the accident. In particular, the amount of retention would depend on both the concentration of the aerosol and the length of time the material was contained in the PCS. In general, large LOCA's would tend to result in negligible aerosol retention due to delays while some transients and some small LOCA's would result in significant delays and hence potentially substantial retention in the PCS. Retention in the PCS of both gases and aerosols could also be large in those accidents in which the radionuclides had to pass through standing water to reach the containment.

The fractions of the radionuclides ultimately escaping from the containment would depend on the status of the containment when the radionuclides reached it, as well as on the effects of the ESF's such as containment sprays. Typically, accidents either with bypassing of the containment or with early catastrophic containment failure would result in the largest releases to the environment. Of those accidents with early containment failure, the ones with early failure of the radionuclide removal systems would result in the largest releases. In such accidents, natural removal processes would have little time to affect the amounts of radioactivity escaping.

The accident spectrum for equipment qualification for most equipment consists of accident sequences involving core damage less than complete melting. For a design basis approach, only a very limited number of those accident sequences must be considered. The sequences important to the equipment vary from plant to plant.

The accident spectrum for emergency planning consists of both meltdown and limited-core-damage accident sequences. For either a risk assessment or a planning evaluation for all possible accidents, many types of accident sequences must be considered. The specific sequences dominating the risk to the human population and those important to any planning evaluation vary substantially from plant to plant.

3. METHODS

3.1 INTRODUCTION

The discussions in the previous chapter focused on reviewing the basic factors involved in considering the release of radionuclides from the core materials and the subsequent transport of those radionuclides through the nuclear plant. This chapter outlines the approach that has been utilized for this study to quantify the aspects of those factors relevant for estimating source terms, by detailing the procedures followed in this project to obtain illustrative estimates. Chapter 3 of the other report for this project (Niemczyk and McDowell-Boyer, 1982) dwells on a description of the "assumptions" behind the adopted procedures. All of the adopted procedures and/or assumptions and the factors affecting their utilization are considered in much greater detail in Appendices B through D of this report.

3.2 OVERALL APPROACH

The emphasis of the work reported here was on estimating the magnitudes of the radionuclide releases for various groups of accidents in each of the accident spectra of concern, with the requisite information about the accident definitions and characteristics being taken directly from other previous work. In particular, the emphasis here was to determine and demonstrate the use of the best methods for estimating radionuclide releases and behavior for LWR accidents. Because the purpose of this project was to provide source term assumptions which could be used on an interim basis until a much more thorough investigation of source terms could be completed, the effort described here concentrated on consideration of procedures and methods which were already available and not on those which might become available in the future.

As was noted previously, many of the assumptions and/or procedures adopted here for consideration of radionuclide releases and behavior were based on work presented or summarized in the recently published Technical Bases Report (USNRC, 1981). In particular, the method proposed in that report for estimating the initial radionuclide releases from the core materials was adopted. In addition, the insights provided by that document concerning radionuclide retention in the primary coolant system were used to develop assumptions for estimating escape from the coolant system. Furthermore, procedures previously proposed and summarized in that report were used to develop assumptions for estimating escape fractions from the containment.

In general, the state-of-the-art of certain aspects of the estimation of source term magnitudes involves the use of complex computer codes and, in some cases, subsequent development of estimates from the outputs of those codes. Due to the limited scale of this project, it was not possible for this project either to describe in detail all the assumptions implicit in calculations by such codes or to perform complete sets of calculations for a variety of accidents with the indicated codes. As a result, the overall approach taken here to illustrate the assumptions and to obtain estimates was to utilize the outputs of previously performed suitable calculations and to modify those outputs to reflect acknowledged shortcomings in the codes used to generate them, as well as to reflect recently gained insights into the release and transport of radionuclides during and after nuclear reactor accidents. For those aspects of the problem for which previous calculations were not directly applicable, the approach was to utilize any available outputs and other information to aid in formulating assumptions and estimates.

Many of the more complicated assumptions and/or procedures adopted in the Technical Bases Report are fundamentally the same as those used in, or developed as a result of, the Reactor Safety Study (RSS). Thus, the illustration of the assumptions and/or procedures adopted in the project described in this report was based on a set of calculations performed utilizing many of the basic procedures used in the RSS.

In particular, the results of a set of radionuclide escape and transport calculations performed by Battelle Columbus Laboratories (Cybulskis, 1981; Wooton, 1981; Kolb et al., 1981; Carlson et al., 1981; Hatch, Cybulskis, and Wooton, 1981) were used as the starting point of most of the estimates. The Battelle results were chosen for consideration here because those results, along with Battelle Columbus and Sandia National Laboratories' associated probabilistic assessment work, form the only set of finished calculations available which have considered in a consistent manner source terms for "complete" spectra of potential accidents at several different types of nuclear plants.* As was mentioned previously, the work by Battelle and Sandia (the Reactor Safety Study Methodology Applications Program, RSSMAP) was part of an extension of the methodology developed for the Reactor Safety Study (RSS; USNRC, 1975) to plant designs other than the two considered in detail in the RSS. The RSSMAP work included a rebaselining of the source terms of the two RSS plants [a large containment standard PWR (Surry) and a Mark I BWR (Peach Bottom)], along with an RSS-like treatment of the potential accident source terms for three other plant designs [an ice condenser PWR (Sequoyah #1), a Mark III BWR (Grand Gulf #1), and another type of large containment PWR (Oconee #3)].† (Some of the pertinent results of the Battelle-Sandia RSSMAP work are summarized in the addendum of Appendix D.)

For this project, the Battelle-Sandia RSSMAP work was used to identify accident sequences of potential interest for the various plants considered. The phenomenological descriptions of those sequences developed in the RSSMAP reports were also adopted. Those descriptions and the associated reactor thermal-hydraulic descriptions and containment escape estimates were utilized in the work performed for this report as the basis for re-estimating the magnitudes, and the rates of the radionuclide releases both to the containment and to the outside environment. The initial release rates from the core materials, the potential retention in the primary coolant system, and the rates of escape from the

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*A more extensive version of the work being performed for this study is currently being conducted at Battelle Columbus Laboratories. This work will include performing another complete set of source term estimates for a variety of plant types and will include some more recent considerations than those considered in the Battelle-Sandia work utilized here.

†Another large containment PWR (Calvert Cliffs #2) was also considered in the RSSMAP work. However, the results of that work have not been published yet.

containment were all reconsidered and consequently re-estimated in this study for each of the accident sequences considered.

3.3 INDIVIDUAL ACCIDENTS

To describe quantitatively the source terms for any specific accident, the primary quantities of interest are the amounts of the initial releases of the radionuclides and other materials, the fractions of all those materials escaping from the primary coolant system, and the fractions of those radionuclides ultimately escaping from the containment. Both sets of escape fractions depend not only on the amounts, but also on the rates, of the initial releases.

For all source term considerations, the radionuclides were grouped into the same seven basic element categories that were used in the RSS (see Table D.4 in Appendix D). Each radionuclide was assumed to be released according to the basic properties (mainly volatility) of its element category. For the most part, the detailed chemistries of the various radionuclides were not explicitly considered.

3.3.1 Releases from Core Materials

3.3.1.1 Method

Most of the initial releases of radionuclides from the core materials would depend directly on processes associated with overheating of the core materials. Therefore, the sum of all the release processes before slumping of the core into the bottom of the reactor vessel was assumed to result in a net release rate for any radionuclide which would be dependent only on the temperature of the material containing the radionuclide (Wichner, Kress and Lorenz, 1981). The indicated temperature-dependent release rates for various elements were obtained from fits by others of the limited empirical information available (see Figure. B.1 in Appendix B). These rates include the releases by all considered processes except leaching and "vaporization" (i.e., sparging). Thus, the procedures utilized here differ from those in the RSS (and in the RSSMAP) in that the contributions of the various phenomena (e.g., gap release, meltdown release, etc.) to the total releases are not treated as discrete events.

Estimates of the fractions of the various radionuclides released before slumping were obtained as a function of time by combining the temperature-dependent release rates just described with the appropriate time-dependent temperature profiles of the different regions of the core for the accident being considered. Release fractions for structural and other materials were determined in basically the same manner as the release fractions for the radionuclides. The radionuclide and nonradionuclide release fractions estimated in this way were coupled with calculated core inventories for such materials to predict the total aerosol mass released prior to slumping.

After slumping of the molten material into the bottom of the reactor vessel, two different procedures were used to estimate the releases from the core materials. In one procedure, the method used before slumping, and just outlined, was used without modification. In the other procedure, the method used before slumping was altered to take into account the much lower relative surface area of the molten materials in the bottom of the reactor vessel (Parker, 1982).

In contrast to the procedures followed before vessel failure, the aerosols formed after vessel failure, during concrete decomposition, were treated much differently. At such times, the release rates were estimated from empirically-derived equations relating aerosol production to gas flow rates through a molten mass interacting with concrete (Murfin and Powers, 1980; Wichner, Lorenz and Kress, 1981). The estimates of individual radionuclide releases after vessel failure were obtained somewhat analogously by combining element-dependent vaporization fractions with gas release rates (USNRC, 1975). The detailed time dependence of the radionuclide releases after vessel failure was ignored.

For all estimates, the core material temperatures as a function of time were taken from thermal-hydraulic code estimates of others (Wooton, 1981). In the computer code used to obtain those estimates, before slumping of the core into the bottom of the reactor pressure vessel (RPV), the core is divided into 120 regions, with the temperature calculated in each as a function of time. After slumping and before vessel melt-through, the core debris temperature is considered homogeneous. After vessel melt-through and during the attack of the concrete by the core debris, the melt is divided into two regions, a metallic phase and an oxidic phase, with the temperature evaluated for each.

Various accidents would differ in the temperature-time profiles of the core materials in varying degrees. Time-dependent estimates of the radionuclide and aerosol releases were obtained for several representative accident sequences for each of the two RSS plants. The results were generalized to important sequences for each of the other three plant designs considered.

In those accidents in which leaching of the core materials also might be a relatively important release mechanism, the fractions leached were estimated using single-temperature empirical leach rates. The resulting leach fractions were added to the fractions released directly by overheating.

3.3.1.2 Input

The temperature-dependent release rate estimates for each element considered were taken from the Technical Bases Report (Wichner, Kress, and Lorenz, 1981), with some modification. The vaporization fractions for each of the elements were taken from the RSS (USNRC, 1975). MARCH (Wooton and Avci, 1980) calculations, performed for the RSS baselining work, were used to obtain both the time-temperature profiles for the core materials and the gas velocities (after reactor vessel melt-through) for each of the considered accident sequences (Wooton, 1981). ORIGEN (Bell, 1973) calculations, performed for this study, were utilized to obtain the radionuclide core inventories at the time of the accident (Alexander, 1981). Recently published leach rates (Mitchell, Goode and Vaughn, 1981) were used to estimate leach releases.

3.3.2 Transport in the Primary Coolant System

3.3.2.1 Method

The most important effect of the primary coolant system during an accident would be retention of some of the materials passing through it. Because retention of many radionuclides in the primary coolant system would occur almost entirely as the result of aerosol deposition processes, removal of such radionuclides was taken to be mainly a function of the two major factors affecting aerosol deposition, namely, the total aerosol concentration and the coolant system aerosol residence time (Gieseke and Kuhlman, 1981).

Whether a material would be present as a gas or an aerosol was taken to depend on its volatility. More specifically, radionuclides belonging to the iodine, cesium, and tellurium groups were permitted to transport through the primary coolant system as both gases and particles. All other radionuclides, except the noble gases, were transported through the system as particles (aerosols).

Insights obtained from considering two somewhat complementary sets of previously performed calculations (Gieseke and Kuhlman, 1981) were used to aid in the development of particle escape fractions for the primary coolant system. One set of these calculations was utilized to estimate the aerosol concentrations and the residence times for various segments of several accident sequences. The other set was used to estimate roughly the amount of aerosol-related radionuclide removal, and therefore retention, which might occur in the PCS for various initial aerosol concentrations.

The results of these two sets of calculations were combined with aerosol residence time and generation rate estimates for various times during the course of an accident. From this, total retention in the primary coolant system was estimated separately for species of low volatility and for those of intermediate volatility for some accident sequences for each of the two RSS plants. PCS escape fractions for those species were not estimated for any of the sequences investigated for each of the other three plant designs considered.

Although retention resulting from aerosol processes would typically be most important for many radionuclides, retention of gases by condensation and sorption would be most important for certain other radionuclides. However, it generally was not estimated here because of a lack of both appropriate previous calculations and simple procedures.

In those accidents in which radionuclide retention due to passage through water in the PCS would be significant, noble gases were assumed to be unaffected by such passage. In contrast, the airborne concentrations of all other radionuclides were taken to be reduced by such passage through water, with the amount of reduction depending upon both the depth and the conditions of the water. These assumptions were used to modify the PCS escape fractions estimated for certain appropriate accident sequences.

3.3.2.2 Input

Recent MARCH-TRAP (Wooton and Avci, 1980; Baybutt and Jordan, 1977; and Jordan, Gieseke, and Baybutt, 1979) and QUICK (Gieseke, Jordan, and Lee, 1979) calculations (Gieseke and Kuhlman, 1981) were used as part of the basis for the primary coolant system retention estimates developed here. The results of other MARCH calculations (Cybulskis, 1981; Wooton, 1981) were also used.

3.3.3 Transport Through the Containment

3.3.3.1 Method

The primary effect of the containment would be to delay the turnover of the containment atmosphere to the outside environment until various chemical and physical processes had reduced the airborne concentrations of radionuclides in that atmosphere to lower levels. In this study, most radionuclides were assumed to remain in the same chemical forms as those in which they entered the containment. Thus, for most radionuclides, only physical removal due to natural processes and the operation of ESF's was considered.

For each accident sequence, the results of previously performed containment transport calculations (Cybulskis, 1981; Wooton, 1981; Kolb et al., 1981; Carlson et al., 1981; Hatch, Cybulskis and Wooton, 1981), which included consideration of removal of radionuclides from the containment atmosphere by both natural and engineered processes, were adjusted to account for potentially neglected or misestimated effects. In particular, the results of the adopted base calculations were modified to account for the differences in the initial fractional release rates employed in this report and those used in the base calculations (and in the RSS). Furthermore, they were altered to compensate for prior removal in the coolant system beyond that previously assumed.

For each group of radionuclides, the releases entering the containment were divided into two parts: the fraction entering the containment indirectly via the primary coolant system and the fraction entering the containment directly. The former fraction was assumed to behave basically like the same radionuclide group in the gap and the meltdown releases in the code used for the base calculations (and in the RSS), while the latter was assumed to behave basically like the same radionuclide group in the vaporization release in that code (and in the RSS). The primary difference was that potential effects due to dissimilarities in the prescriptions for the magnitudes and the timing of the releases into the containment were taken into account.

Containment radionuclide escape fractions were estimated for some representative accident sequences for each of the five RSS and RSSMAP plant designs considered. These values were used, in turn, to estimate containment escape fractions for each of the classes of accident sequences used for equipment qualification and emergency planning.

3.3.3.2 Input

MARCH-CORRAL (Wooton and Avci, 1980; Postma, Owzarski and Lessor, 1975; Burian and Cybulskis, 1977) calculations performed for the RSS rebaselining work and the associated RSSMAP work were utilized as the basis for the containment escape fraction estimates (Cybulskis, 1981; Wooton, 1981; Kolb et al., 1981; Carlson et al., 1981; Hatch, Cybulskis, and Wooton, 1981).

3.4 SPECTRUM OF ACCIDENTS

3.4.1 Equipment Qualification

3.4.1.1 Method

The accident spectrum for equipment qualification was taken to be inadequately covered by the accident sequences explicitly considered in the RSS and the RSSMAP reports. Both the indicated range of sequences and the relevant types of sequences within that range were taken to be different from those studies. Unlike those previous studies, sequences with only partial or delayed functioning of the emergency core cooling system were included in the range of accidents utilized to form the basis for the equipment-qualification spectrum. Only potentially worst-case limited-core-damage accidents within the design basis were included in the covered range. Two representative accident sequences for a single plant design were investigated in detail. The magnitudes of the radionuclide releases for each accident sequence were estimated using the methods described earlier, in Section 3.3 of this chapter.

3.4.1.2 Input

The descriptions of the two limited-core-damage accident sequences considered in detail were adapted from the Technical Bases Report (Denning, 1981).

3.4.2 Emergency Planning

3.4.2.1 Method

The accident spectrum for emergency planning was taken to be almost adequately covered by the accident sequences explictly considered in the RSS and the RSSMAP reports. Thus, primarily sequences with each of the ESF's either failed completely or else functioning as intended were included for consideration. For the most part, only the sequences identified in the RSS and the RSSMAP reports as being potentially dominant in each RSS accident category were investigated in detail for emergency planning. A broad distribution of both limited-core-damage accidents and meltdown accidents were included in the covered range. The magnitudes of the radionuclide releases for each accident sequence were estimated using the methods described earlier.

The accident sequences considered were subsequently regrouped into several distinct classes of sequences differing primarily in the magnitudes of the resulting releases to the environment. Thus, the emergency planning spectrum was reduced to a set of several classes of reactor accidents, with the magnitudes of the releases of the composite classes being representative of all five plant designs considered in this study.

3.4.2.2 Input

The descriptions and the relative importances of the various possible accident sequences were taken from the RSS and the RSSMAP reports (USNRC, 1975; Kolb et al., 1981; Carlson et al., 1981; Hatch, Cybulskis, and Wooton, 1981; Cybulskis, 1981; Wooton, 1981).

3.5 METHODS VERSUS REALITY

Some of the shortcomings of the methods used here are outlined in Table 3.1. They are discussed in much greater detail for all the topics covered in this report in Appendices B-D and in general in Appendix E. The shortcomings result primarily from inadequate data to describe some of the relevant processes. In many cases, this lack of data has severely hampered the development of appropriate models. The potential impacts of all these problems on the source term estimates are discussed in the next two chapters.

Table 3.1. Problems in estimating magnitudes of source terms

Processes	Method	Problems[a]
Releases of fission products and other products from core materials and structural materials before reactor vessel failure by processes other than leaching.	Empirically-derived curves describe temperature-dependent rates for each element	Adequate information not available for all important elements, for all temperatures, and for in-situ conditions; chemical forms largely ignored.[b]
	Phenomenological code describes thermal-hydraulic conditions.	Thermal-hydraulic input not adequately known; some important processes neglected or ignored.
	("First-principles" code describes fission product releases).[c]	(Most required data not available).
Leaching of fission products.	Empirical data describes leaching.	Data not available for most elements, for an appropriate range of temperatures, and for accident (in-situ) conditions.
	("First-principles" code describes leaching).[c]	(Most required data not available).
Releases of fission products and other materials after vessel failure.	Empirically-derived release rate expression describes total aerosol generation; element-dependent vaporization fractions determine radionuclide release rates.	Limited data available for total aerosol generation rates; inadequate data available for many elements.
Transport through primary coolant system atmosphere.	"First-principles" code describes PCS transport.	Important aerosol removal processes not included; thermal hydraulic conditions not adequately known.
	"First-principles" containment codes describe aerosol processes.	Ignore thermal hydraulic conditions; ignore structure of coolant system; some data not available.
	Phenomenological code describes thermal-hydraulic conditions.	Thermal-hydraulic input...

Table 3.1. (continued)

Processes	Method	Problems[a]
Scrubbing in primary coolant system.	Empirically-based decontamination factors describe scrubbing.	Adequate data not available for all important elements and conditions.
Transport through containment.	Empirically-based code describes containment transport.	Effects of potentially important removal processes may be underestimated; effects may not be correctly scaled from experimental conditions; containment rupture modes and timing not adequately predicted; thermal-hydraulic conditions not adequately known; many processes not explicitly modeled; inadequate element-dependent and condition-dependent data available.
	Phenomenological code describes thermal-hydraulic conditions.	Thermal-hydraulic input ..
	("First-principles" code describes containment transport).[c]	(Most required data not available; not all processes are included in any specific code.)
Leakage from containment.	Empirically-based equation describes leak rate with respect to size.[c]	Mode and timing of containment failure very uncertain.

[a]See the discussions in Appendices B-D of the other report for this project for details.

[b]To use this method, one needs either appropriate empirical data or knowledge of the chemical form(s) for each element.

[c]Not used in this project.

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4. EQUIPMENT QUALIFICATION

4.1 INTRODUCTION

Many of the safety-related components used in a nuclear plant are selected partially on their ability to withstand on a long-term basis the radiation fields which are encountered during normal operations and partially on their ability to withstand on a short-term basis the exceptional fields which might be encountered during accidents. This chapter illustrates the assumptions which are taken to be appropriate for the estimation of radionuclide source terms for such considerations, that is, for equipment qualification.

4.2 DESIGN BASIS ACCIDENT SPECTRUM

For the treatment of equipment qualification in this report, only accidents whose releases would bound the design basis envelope need to be considered. One such bounding accident* might involve melting of a large fraction of the core† as the result of delayed functioning of the emergency core cooling system following a large pipe break in the primary coolant system. Although the initial radioactive releases would encounter a relatively dry pathway to the containment, at least the releases toward the end of the accident would encounter water in that pathway.

Another possible bounding accident might involve melting of a large fraction of the core as the result of delayed functioning of the emergency core cooling system following a transient event.* In a PWR, any radionuclides released from the core in such an accident might pass through water in the pressurizer quench tank on their way to the containment. In a BWR, the released radionuclides would typically

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*This description is based on the Technical Bases Report (Denning, 1981a).

†In general, if more than approximately 50-80% of the core melted, melting probably could not be terminated and the entire core would melt (USNRC, 1975). Accidents with complete meltdown of the reactor core are outside the design basis.

bypass the drywell, be released through the relief valves and enter the suppression pool.

An important difference between the two accidents just described would be in the paths followed by the releases from the core materials. Whereas for the pipe-break accident the entire path to the containment often would be relatively dry until the emergency core cooling started to function, for the transient-initiated sequence the released materials might encounter water in that path throughout the accident. Thus a substantial fraction of the released radionuclides might be entrained by the water in the latter, transient sequence.

Another potentially important difference between the two accidents would be in the residence times of the released radionuclides in the primary coolant system. Although the pipe-break accident might have relatively short coolant system residence times during much of the accident, the transient-initiated accident might have relatively long coolant system residence times during a large portion of the accident. Thus if aerosol releases were large, aerosol deposition processes within the coolant system might be much more important for the latter, transient accident. This would, however, be very dependent on both the reactor design and the details of the accident scenarios.

To consider both accident sequences, the overheating of the core was assumed to proceed until temperatures sufficient to permit melting in 50% of the core were attained. (More exactly, temperatures sufficient to permit melting in 50% of the core regions, according to MARCH calculations, were assumed.) Then further overheating was taken to be prevented by activation of the emergency core cooling system. That system was assumed to fail completely until it started to function to cool the core adequately. For the loss-of-coolant accident (LOCA), a short unobstructed path through the primary coolant system was assumed. The overheating of the core was assumed to follow the same history for both these accident sequences.

4.3 RELEASES FROM THE CORE MATERIALS

The fractions of the core materials predicted to be released for both accident sequences investigated in the illustrative calculations are the same and are given in Table 4.1. Also given are the corresponding TID source terms (USAEC, 1962) currently used for many regulatory considerations. As can be seen in the calculations performed for this study, the releases for three important element groups (Cs-Rb, Te-Sb, and Ba-Sr) are estimated to be much larger than the TID source term values while the releases for one other group (La) are estimated to be much smaller than the TID values.

It is important to note, however, that these estimated amounts are highly dependent upon the exact accident scenario considered. For example, the sequence used in the illustrative calculations was a terminated AD* for a large containment PWR (the RSS PWR). This sequence is postulated to take place over a relatively short period of time. Other sequences which involve comparable total core damage but which involve slower overheating of the core might be expected to result in somewhat larger releases of certain materials. According to the methods used here, somewhat larger releases of all the groups from Te-Sb in Table 4.1 would occur in a relatively slow sequence. In calculations of certain slow sequences with core cooling initiated after 50% of the core regions was melted, the estimated releases of the last four groups might easily be double the amounts given in Table 4.1. (See the detailed time-dependent release results presented in Appendix B.)

In addition, the definition of the maximum extent of melting must be noted to be somewhat arbitrary. A definition other than that used here would result in different releases of at least some radionuclides. Also the exact point in the scenario of any accident after which complete melting could not be prevented is not known.

In general, all the differences due to consideration of various accident sequences and those due to different accident definitions would not noticeably affect the estimated releases of the more volatile

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*See Appendix A for an explanation of this notation.

Table 4.1. Release fractions from core materials

Report	Release	Fractions released from core materials						
		Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
This report[a,b]	Gap + "1/2" meltdown	1.00	1.00	1.00	0.50	0.05	0.01	1(-4)[c]
	Leach	0.00	0.00	0.00	0.01	0.001	0.001	1(-4)
TID-14844	Total	1.00	1.00	0.01	0.01	0.01	0.01	0.01

[a] "Best" estimate.

[b] These vaues have been rounded off. The exact estimates are given in Appendix B.

[c] 1(-4) denotes 1×10^{-4}.

species, such as the noble gases, the halogens and the alkali metals (Cs-Rb). The differences would be most evident for radionuclides in all the other, less volatile groups. Most importantly, the differences potentially caused by both considerations are well within the uncertainties of the methods used to obtain the source term estimates.

For bounding accidents such as those considered here, the additional fractions released by leaching usually would be relatively small compared to the fractions initially released by overheating (see Table 4.1). It should be noted that these fractions would be very accident dependent and that the rates of leaching are not well known.

4.4 TRANSPORT THROUGH THE PRIMARY COOLANT SYSTEM

The two general accident sequences considered admit an almost infinite variety of initial distributions of radionuclides between the containment atmosphere and the water in the containment. Radionuclides initially retained in the PCS by processes such as aerosol removal could be entrained in the coolant water when the core was reflooded. Thus most of the released radioactivity generally could enter the containment in one way or another. The only question often would be the distributions of the radionuclides between the air and the water.

The extreme initial distributions entering the containment are listed in Table 4.2. For comparison, the source terms currently used for regulation are also presented (USAEC, 1974). Those regulatory source terms are basically the TID ones, with the amounts of "solid" fission products being doubled for conservatism.

4.5 TRANSPORT WITHIN THE CONTAINMENT

For equipment qualification, the containment constitutes the environment of concern. Thus, the only important factor in the containment for source term considerations is the transfer across the interface between the containment atmosphere and any water in the containment. This is described by the consideration of partitioning given in Section 3.3 of Chapter 3 of the other report for this project (Niemczyk and McDowell-Boyer, 1982). All other considerations of movement within

Table 4.2. Total release fractions for design basis accident for equipment qualification

Report	Initial distribution	Fractions of core inventory released to containment but not to environment						
		Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
This report[a]	Airborne	1.00	1.00	1.00	0.50	0.05	0.01	1(-4)[b]
	Waterborne[c]	0.00	0.00	0.00	0.01	0.001	0.001	1(-4)
	Waterborne	1.00	1.00	1.00	0.50	0.05	0.01	1(-4)
	Waterborne[c]	0.00	0.00	0.00	0.01	0.001	0.001	1(-4)
Regulatory Guide 1.89	Airborne	1.00	0.25	0.00	0.00	0.00	0.00	0.00
	Waterborne	0.00	0.50	0.01	0.01	0.01	0.01	0.01
	Plateout	0.00	0.25	0.01	0.01	0.01	0.01	0.01

[a]"Best" estimates for the two extremes of initial distributions within the containment.

[b]1(-4) denotes 1×10^{-4}.

[c]The values on this line represent the additional releases due to leaching to the core materials.

the containment are beyond the scope of this project and so are not considered here.

4.6 UNCERTAINTIES

The uncertainties associated with estimates of the releases from the core materials are dependent on the species involved. For example, for accidents involving melting of a large fraction of the core, essentially the entire amounts of the more volatile species generally would be released. The main uncertainty in predicting releases for many such species is associated with estimating the amount of those species present in the core inventory both at the time of accident initiation and later in the accident. In contrast, for some species of low volatility, the fractions of the core inventory released are highly uncertain because the release rates for certain of those species are not well known. A lack of appropriate data and adequate models for describing releases contributes to that problem. Furthermore, there is again the problem of estimating the amounts present in the core inventory for such species. In the procedures used in this report, there is the additional uncertainty associated with using element groups instead of considering radionuclides on an element by element basis because there is often a wide range of release rates associated with the various elements within a given group. Furthermore, although the more volatile species would typically be entirely released in any accident involving melting of a large fraction of the core, the amounts of the less volatile elements releases would be highly scenario dependent. And even if the scenario is specified, its thermal-hydraulic description is somewhat uncertain and that affects the amounts of at least the less volatile species predicted to be released.

The uncertainties associated with the estimates of the releases from the primary coolant system are also dependent on the species involved. For example, noble gases would be entirely released from the PCS. In contrast, other species could be retained in varying amounts depending on their volatilities, their reactivities, and the accident scenario. However, because the design basis approach used for equipment qualification is a bounding approach, these uncertainties are not overly important if even just one accident can be envisioned which would probably result in very low retention of all species. Consequently, the most important uncertainties for estimating design basis source terms for equipment qualification are those associated with the factors which determine the initial releases of radionuclides from the core materials (see Table 4.3). (For more extensive discussions of the detailed sources of uncertainties, see Appendixes B-D. For a closely related discussion of the general source of uncertainties, see Appendix E.)

4.7 SOURCE TERMS FOR REGULATORY GUIDANCE FOR EQUIPMENT QUALIFICATION

A reasonable procedure for estimating source terms for equipment qualification would be to use the detailed procedures and/or assumptions described in Chapter 3 to consider various accidents potentially bounding the design basis. Thus, best estimate design basis source terms appropriate for a given nuclear plant could be estimated. However, because of the complexity of some of the indicated procedures and assumptions, there is a significant likelihood that errors would be made in their application by persons not thoroughly knowledgeable in source term estimation. Thus, it would probably be more reasonable to use "generic" (i.e., for a typical plant) source terms which had been generated by persons expert in the indicated procedures. Given the current level of sophistication of source term estimation, such a set of source terms would be relatively reactor independent and therefore could probably be used for a large number of plants.

In any case, because of the many problems associated with the definition and the description of the bounding accident(s) for the design basis for any plant, the worst-case accident(s) for any particular plant cannot be described exactly. Thus, the best-estimate source terms for either a specific plant or a generic plant should be modified to account for some of the uncertainties.

In particular, because of the large uncertainties associated with certain aspects of the estimation of source terms for equipment qualification, it is suggested that the best estimates of sequences such as

Table 4.3 Impacts of problems in estimating magnitudes of source terms on equipment qualification accident sequences

Phenomena	Description	Problems[a]	Potential impact	
			Individual accident	Design basis envelope
Releases				
Releases other than leaching	Empirical temperature-dependent rates	Insufficient data	Large	Large
	Thermal-hydraulic estimates	Insufficient data; inadequate models		
Leaching releases	Empirical rates	Insufficient data	Small	Small
Transport Through Primary Coolant System				
Transport through coolant system atmosphere	"First-principles" transport code	Insufficient data; inadequate models	Large	Small
	Thermal-hydraulic estimates	Insufficient data; Inadequate models		
Scrubbing in coolant system	Empirical rates	Insufficient data; Inadequate models	Large	Small

[a]See Table 3.1 in Chapter 3 for more details.

those considered here be multiplied by an adjustment factor. More exactly, the value of 1.3 is suggested for species of high volatility (noble gases, halogens, and alkali metals) and the value of 2.0 is suggested for all other species. Examples of such source terms are presented in Table 4.4.

For the more volatile species, which would be released entirely regardless of the details of the accident description, this adjustment factor encompasses the uncertainty associated with radionuclide core inventory estimates provided by such codes as ORIGEN (Bell, 1973). For the less volatile species, whose estimated releases would depend upon the conditions assumed, this would include some variations due to the arbitrary boundary conditions chosen. It would not account for the very large uncertainties associated with the release rates for some of those less volatile species. However, it might be noted that, in general, the release rates used in this study for those species with highly uncertain release rates have been taken to fall toward the high ends of their possible ranges.

Table 4.4. Source terms for equipment qualification

Initial distribution	Fractions of core inventory released to containment but not to environment						
	Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
Airborne	1.00	1.00	1.00	1.00	0.010	0.02	2(-4)[a]
Waterborne[b]	0.00	0.00	0.00	0.00	0.002	0.002	2(-4)
Waterborne	1.00	1.00	1.00	1.00	0.010	0.02	2(-4)
Waterborne[b]	0.00	0.00	0.00	0.00	0.002	0.002	2(-4)

[a] 2(-4) denotes 2×10^{-4}.

[b] The values on this line represent the additional releases due to leaching to the core materials.

References for Chapter 4

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Denning, R. S., 1981. "Accident Sequence Characteristics," Chapter 3 in *Technical Bases for Estimating Fission Product Behavior During LWR Accidents*, NUREG-0772, U. S. Nuclear Regulatory Commission, Washington, D. C.

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5. EMERGENCY PLANNING

5.1 INTRODUCTION

The procedures which would be followed to mitigate the radiation hazard to the human population during a reactor accident would depend on the anticipated releases of radioactivity to the environment. This chapter illustrates the assumptions which are taken to be appropriate for the estimation of radionuclide source terms for such considerations, that is, for emergency planning.

5.2 ACCIDENT SPECTRUM

For emergency planning, a broad spectrum of possible accidents needs to be investigated. A representative variety of postulated accidents and accident sequences for a few nuclear plants are described in some detail in Appendix A of this report and in greater detail in the reports from which those descriptions have been adopted. In this chapter, for illustrative purposes, the following accidents are considered: a normally-terminated, large pipe loss-of-coolant accident (LOCA) involving gap releases only; a belatedly-terminated, large pipe LOCA involving melting of 50% of the core; and a variety of meltdown accidents. They are described briefly in the next several paragraphs.

For the normally-terminated LOCA, only minor overheating of the core would occur. Because the core would heat up rather unevenly in such an accident, releases beyond just the classic gap releases would occur for some fuel rods while not even the classic gap releases would occur for many other rods. Overheating would soon be terminated by reflooding of the core by the emergency core cooling system. Escape of radioactivity to the environment would be relatively slow, by leakage from the containment. For this study, it was assumed that overheating would occur only until 50% of the core was above the minimum cladding rupture temperature.* (More exactly, "gap" releases were defined to

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*This artifice was adopted for this illustrative calculation because of a lack of readily available appropriate thermal-hydraulic results for a normally-terminated LOCA.

be the releases which would occur before 50% of the regions of the core reached, according to MARCH calculations, the temperature at which cladding would rupture.*)

For the accident involving delayed functioning of the emergency core cooling system, the overheating of the core would be much more extensive than for the first accident considered. Some parts of the core could experience temperatures sufficient to cause melting while others would experience only much lower temperatures. Eventually activation of the emergency core cooling system would reflood the core and prevent further melting. Escape of radioactivity from the containment would be by leakage only. For this study, it was assumed that overheating would occur until 50% of the core was above the melting temperature. (More exactly, temperatures sufficient to permit melting in 50% of the regions of the core, according to MARCH calculations, were assumed.) For both this accident and the normally-terminated LOCA, the pathway through the primary coolant system was taken to be short and unobstructed. In addition, leaching of the reflooded core materials was assumed to occur.

For any meltdown accident, more overheating of the core would occur than for either of the preceding two accidents. Any meltdown accident would involve melting of a large fraction of the core, as well as the potential for substantial interactions of at least parts of of the core with both the reactor vessel and the concrete basemat of the containment. A wide variety of accident conditions would be possible for meltdown accidents, ranging from those which would result in very small releases of radioactivity to the environment to some others which would result in relatively large releases. The specific sequences considered included some involving relatively rapid meltdown of the reactor core, as well as others involving relatively slow meltdown. In addition, the sequences investigated included some with the potential for substantial permanent retention of at least some radionuclides in the primary coolant system, in addition to other sequences with little likelihood of any permanent retention in that system. Furthermore, the

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*See the footnote on the previous page."

sequences considered varied widely in both the modes and the timing of containment failure assumed, and included both sequences with early massive rupturing of the containment and concomitant flows of large masses of radioactivity from the plant, as well as some with only slow leakage of small amounts of radioactivity from the plant. Descriptions of some of the specific individual meltdown sequences investigated are given in Appendix A.

5.3 RELEASES FROM CORE MATERIALS

Accidents in the overall spectrum for emergency planning range from those involving releases of essentially negligible fractions of all radionuclides from the core materials to those involving releases of substantial fractions of many radionuclides. The fractions predicted to be released from the core materials in all the illustrative calculations performed for this study are given in Table 5.1. For consideration of releases during limited-core-damage accidents, the accidents investigated were the two described in the previous section. For consideration of releases during meltdown accidents, the accidents investigated were seven different sequences in the two Reactor Safety Study (RSS) plants* (AD, V, S_2D, and TMLB´ in the RSS PWR and TC, TW, and TQUV in the RSS BWR).† The detailed results for all the sequences considered are presented in Appendix B. The corresponding RSS source terms sometimes used for certain emergency planning purposes are also given in Table 5.1 for comparison (USNRC, 1975).

The releases estimated for the accident involving only "gap" releases depend on the details of the accident description adopted. For example, the estimates of the "gap" releases can vary substantially, depending on the rupture temperature assumed for the fuel rod cladding. In all cases, however, those estimated releases are relatively small. For a cladding failure temperature of 750°C, the estimated releases are

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*The two "generic" reactors considered in the Reactor Safety Study were a large dry containment PWR (Surry) and a Mark I BWR (Peach Bottom).

†See Appendix A for descriptions of these sequences.

Table 5.1. Release fractions from core materials

Report	Release	Fractions of core inventory released from core materials[a]						
		Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
This report	"gap"	0.04(0.96)[b]	0.01	0.01	0.001	1(-4)[c]	1(-6)	1(-7)
RSS	"gap"	0.03	0 017	0.05	1(-4)	1(-6)	0	0
This report	gap + "1/2" meltdown	1.00	1 00	1.00	0.50	0.05	0.01	1(-4)
RSS	gap + "1/2" meltdown	-	-	-	-	-	-	-
This report	gap + meltdown[d,e]	1.00	1.00	1.00	1 00(0.78-1.00)	0.35(0.10-0.50)[f]	0.02(0.003-0 02)[g]	0.001[2(-4)-0.002]
						0.15(0.04-0.23)	0.15(0.06-0.25)	
This report	gap + meltdown[h]	1.00	1.00	1.00	0.80(0.57-1.00)	0.10(0.06-0.46)	0.004(0.002-0.02)	0.001
						0.04(0.03-0.21)	0.05(0.02-0.20)	
RSS	gap + meltdown	0 90	0.90	0.81	0.15	0.10	0.03	0.003
This report	vaporization[e]	0.00	0.00	0 00	0.00(0.00-0.22)	0.04(0.02-0.04)[f]	0.02(0.02-0.03)[g]	0.01
						0.00	0.12(0.08-0.16)	
RSS	vaporization	0 10	0.10	0.19	0.85	0 01	0.05	0.010
This report	total melt[d,i]	1.00	1 00	1.00	1.00	0.39(0.14-0.52)[f]	0.04(0.02-0.04)[g]	0.01
						0.15(0.04-0.23)	0.27(0.21-0.37)	
This report	total melt[h]	1.00	1 00	1.00	1 00	0.14(0.10-0.48)	0.02(0.02-0.04)	0.01
						0.04(0.03-0.21)	0.18(0.16-0.32)	
RSS	total melt[e]	1 00	1 00	1 00	1.00	0.11	0.08	0.013

[a]Value given denotes "representative" value for sequences considered; values in parentheses denote calculated ranges for those sequences.

[b]Denotes additional release by leaching.

[c]1(-4) denotes 1×10^{-4}

[d]These values assume no decrease in the fractional release rates after slumping of the core into the bottom of the reactor vessel.

[e]Based on RSS terminology.

[f]Upper line is based on barium data, lower line is based on strontium data

[g]Upper line is based on ruthenium data, lower line is based on molybdenum data.

[h]These values assume a decrease in the fractional release rates after slumping of the core into the bottom of the reactor vessel.

[i]Includes gap + meltdown + vaporization.

essentially just the classic gap releases for half of the core (see Table 5.1).*

The releases estimated for the accident involving partial melting of the core also depend upon the particulars of the postulated scenario. The releases for such an accident were discussed in the previous chapter and so they are not discussed further. They are given in Table 5.1.

The radionuclide releases for accidents involving complete melting of the core likewise depend strongly upon the accident descriptions. From the ranges of the radionuclide releases for the seven meltdown accident sequences given in Table 5.1, it can be seen that for the first four groups of elements in the table (Xe-Kr, I-Br, Cs-Rb, and Te-Sb), the total amounts of the radionuclides predicted to be released are the same as in the RSS. However, as can also be seen in the table, the timing of those releases from the core materials with respect to before and after reactor vessel failure is much different in this study and in the RSS for some of those species. For each of the next two groups in the table (Ba-Sr and Ru), both the magnitudes and the timing of the releases are somewhat different than in the RSS. For the last group in the table (La), both the magnitude and the timing are approximately the same as in the RSS. All of the differences in the estimates of the releases from the core materials in this study and in the RSS are due to both the much different approaches used for estimating

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*According to "best-estimate" calculations for a normally terminated large pipe LOCA (Johnson, Childs and Broughton, 1976), the conditions estimated in this study to be present when 50% of the core has reached 750°C are approximately equivalent to, or slightly worse than those expected for such a LOCA. Thus the choice of 750°C seems reasonable to describe a "gap" release, given the other large uncertainties incurred in describing such a release.

The conditions described here have been chosen on the basis of availability of MARCH output to use for these estimates. More appropriately, the output of computer codes better suited than MARCH to describe terminated LOCA's should be employed to consider "gap" releases. (See the brief discussion in Section C.3 in Appendix C which compares MARCH and more appropriate codes such as RELAP [Aerojet Nuclear Company, 1976].) The use of such codes would obviate the need for an artificial definition of gap releases such as that used in this study.

releases in the two studies and the use of some recent data in this study.

In general, because of the use of temperature-dependent release rates in this study, the total estimates releases of the less volatile species depend strongly on the length of time any overheating occurs, so that accidents involving slower heating often result in larger releases of many of those species. This accounts for most of the variation seen in Table 5.1 for different meltdown sequences. The other major source of variation for the meltdown sequences in that table is due to the two alternate assumptions employed to describe the releases after slumping of the core materials into the bottom of the reactor vessel. As can be seen in that table, only the estimates for the Ba-Sr group and the Ru group are very dependent on whether the release rates are assumed to remain the same or to decrease after slumping.

5.4 TRANSPORT THROUGH THE PRIMARY COOLANT SYSTEM

The continuum of accidents which should be considered for emergency planning includes an essentially infinite variety of possiblities with respect to transport through the primary coolant system (PCS). Accidents ranging from those involving almost complete retention of some released radionuclides in the PCS to those involving retention of none of the released radionuclides are possible. In addition, transport through the PCS could affect whether the radionuclides would escape to the containment as airborne species or as waterborne ones.

For the illustrative calculations, the escape fractions from the PCS were taken, based on the considerations reviewed in Chapter 3, to be those given in Tables 5.2 and 5.3. (These fractions represent the greatest retention which might currently be considered.) These escape fractions apply only to the amounts of materials entering the containment via the primary coolant system. Typically, in previous calculations such as those performed for the RSS, no retention in the PCS has been assumed. (The one exception in the RSS in which retention in the PCS was considered was a meltdown accident in a BWR. See the discussion in Appendix C.)

Table 5.2 Summary of primary coolant system escape fractions

Type of accident	Escape fractions: Xe	$I^{a,b}$	Aerosols[a]
A. Accident sequence without water in the pathway through the PCS			
Large LOCA's	1.0	1.0	1.0 (0.8-1.0)
Small LOCA's	1.0	1.0	1.0 (0.3-1.0)
Transient-initiated LOCA's	1.0	1.0	1.0 (0.3-1.0)
Other transients	1.0	1.0	-[c] (0.1-1.0)
Accident involving "gap" release only	1.0	1.0	1.0 (0.1-1.0)
Accidents involving partial melting	1.0	1.0	1.0 (0.1-1.0)
B. Accident sequences with water in the pathway through the PCS[d]			
Meltdowns	1.0	0.1 (0.01-1.0)	0.1 (0.01-1.0)
Accidents involving "gap" release only	1.0	0.1 (0.01-1.0)	0.1 (0.01-1.0)
Accidents involving partial melting	1.0	0.1 (0.01-1.0)	0.1 (0.01-1.0)

[a]The values given in parentheses denote ranges of reasonable values for escape fractions.

[b]Applies to iodine present as elemental iodine.

[c]The escape fractions for these sequences are highly dependent on the rate of flow of steam through the PCS during the accident. See Table 5.3 for the details.

[d]Obviously, the amount of retention in the PCS water would depend upon the path taken by the radionuclides, that is, upon the amount and temperature of the water encountered.

Table 5.3 Primary coolant system escape fractions for transient-initiated meltdowns

Accident sequence	Escape fractions		
	Xe	I[a]	Aerosols[b]
TMLB′ (PWR)[c]	1.0	1.0	1.0 (0.8-1.0)
TC (BWR)	1.0	1.0	0.33 (0.10-0.33)
TW (BWR)	1.0	1.0	0.33 (0.10-0.33)
TQUV (BWR)	1.0	1.0	0.67 (0.33-1.00)

[a]Applies to iodine present as elemental iodine.

[b]The values given in parentheses denote ranges of reasonable values for escape fractions.

[c]The values given here do not include the effects of possible scrubbing by water in the pressurizer quench tank (see Table 5.2).

As can be seen, the assumptions reviewed in Chapter 3 in general do not result in large permanent retention in the PCS for most accidents. They can, however, result in a redistribution of a sizeable fraction of some species between the PCS atmosphere and any water in the PCS. In addition, for a few sequences, they can result in significant permanent retention for some radionuclides groups which are both transported as aerosols and released mostly before vessel failure, e.g., Cs-Rb and Te-Sb.

5.5 TRANSPORT WITHIN THE CONTAINMENT

The total fractions of the various radionuclides estimated to be released from the containment in some representative accident sequences for three different plants considered in the Reactor Safety Study and in the Reactor Safety Study Methodology Applications Program (RSSMAP; Carlson et al., 1981; Kolb et al., 1981; Hatch, Cybulskis, and Wooton, 1981; Cybulksis, 1981; and Wooton, 1981) are listed in Table 5.4. The estimated releases for many more sequences are given in the addendum of Appendix E of this report.

As can be seen by the comparisons presented in Table 5.4 (and the tables in Appendix E), the values estimated in this study and in the RSS and in the RSSMAP follow-on work are comparable except for the following situations: (1) transient-initiated meltdowns with scrubbing of radionuclides assumed in the pressurizer quench tank (e.g., TMLB'); and (2) accidents with substantial aerosol removal assumed in the primary coolant system (e.g., TC). In both types of accidents, the source terms estimated in this report tend to be somewhat lower than those in the RSS and in the RSSMAP. In contrast, for all accidents, the Ba-Sr group releases and the Ru group releases are generally larger in this report if the potential decreases in the initial release rates after slumping into the bottom of the reactor vessel are ignored; however, if such decreases are assumed, then the releases of both the Ba-Sr group and the Ru group are typically lower in this report. (For example, see the values given in Table 5.4 for event V.) It should be noted that the differences indicated for the two element groups are well within the uncertainties associated with the descriptions of source terms.

Table 5.4. Containment escape fractions for some representative accident sequences

Report	Sequence	Reactor	Cumulative fracture of core inventory released to environment[a]						
			Xe-Kr	I-Br[b]	Cs-Rb	Te-Sb	Ba-Sr[c]	Ru[d]	La
This report	V	RSS PWR	1.00	0.70	0.94	0.94	0.20(0.47)	0.025(0.21)	0.004
This report	V[e]	RSS PWR	1.00	0.70	0.94	0.80	0.04(0.11)	0.01(0.09)	0.004
RSS[f]	V	RSS PWR	1.00	0.64	0.82	0.41	0.10	0.04	0.006
This report	TMLB′-ε	RSS PWR	0.86	0.003	0.001	0.001	0.0002(0.0004)	3(-5)(0.0003)	1(-5)[g]
This report	TMLB′-ε[h]	RSS PWR	0.86	8(-4)	3(-4)	3(-4)	6(-5)(0.0001)	2(-5)(0.0001)	1(-5)
RSS[f]	TMLB′-ε	RSS PWR	0.70	4(-4)	0.001	0.001	1(-4)	1(-5)	1(-5)
This report	TC-γ′	RSS BWR	1.00	0.47	0.67	0.67	0.04(0.14)	0.02(0.15)	0.006
This report	TC-γ′[i]	RSS BWR	1.00	0.47	0.22	0.26	0.013(0.064)	0.02(0.11)	0.006
RSS[f]	TC-γ′	RSS BWR	1.00	0.45	0.67	0.64	0.07	0.05	0.008
This report	TQUV-γ	RSS BWR	1.00	0.02	0.05	0.07	0.002(0.010)	0.003(0.02)	0.001
This report	TQUV-γ[i]	RSS BWR	1.00	0.02	0.03	0.05	0.002(0.009)	0.003(0.020)	0.001
RSS[f]	TQUV-γ	RSS BWR	1.00	0.02	0.06	0.11	0.006	0.007	0.001
This report	S_2HF-γ	IC PWR[j]	1.00	0.13	0.63	0.63	0.06(0.16)	0.02(0.13)	0.005
RSSMAP	S_2HF-γ	IC PWR	1.00	0.13	0.57	0.49	0.07	0.04	0.007
This report	AD-δ	IC PWR	1.00	7(-7)	3(-7)	3(-7)	6(-8)[2(-7)]	6(-9)[6(-8)]	6(-10)
RSSMAP	AD-δ	IC PWR	1.00	7(-7)	3(-7)	3(-7)	3(-8)	2(-8)	4(-9)
This report	A	RSS PWR	4(-6)	2(-7)	1(-7)	1(-8)	1(-10)[5(-10)]	1(-11)[5(-10)]	1(-12)
RSS	A	RSS PWR	3(-6)	1(-7)	6(-7)	1(-9)	1(-11)	0	0

[a] For other sequences, see Appendix E. Also see Appendix E for details.

[b] Represents fraction of iodine present at elemental iodine which escapes.

[c] Based on strontium data; value in parentheses based on barium.

[d] Based on ruthenium data; value in parentheses based on molybdenum.

[e] Release rates after slumping reduced by surface-to-volume considerations.

[f] Based on the RSS-rebaselining work in the Reactor Safety Study Methodology Applications Program (RSSMAP).

[g] 1(-5) denotes 1×10^{-5}.

[h] Removal due to scrubbing by passage through pressurizer quench tank included.

[i] Aerosol retention in primary coolant system included.

[j] Denotes ice condenser PWR considered in the RSSMAP.

5.6 UNCERTAINTIES*

For a given accident sequence, one of the largest sources of uncertainty is in the basic description of the associated accident scenario(s). In particular, even though the likelihood of conditions suitable for the occurrence of certain events can be predicted, the likelihood of the events themselves often cannot be. For example, the computer codes used to describe the environment in the containment can be utilized to predict the presence of conditions which might result in failure of the containment but the actual mode and timing of containment failure during any accident cannot be predicted and must be assumed. Likewise the accident descriptions abound with many other examples of phenomena which are assumed and not predicted. Consequently, it must be realized that the description of any given accident sequence involves many sometimes rather arbitrary assumptions and that if some of those assumptions were changed, the description of the accident might likewise change significantly. Because an accident spectrum is composed of many accident sequences whose basic descriptions are highly uncertain, various aspects of the spectrum are also uncertain.

The uncertainties associated with estimates of the releases from the core materials and retention in the primary coolant system are dependent on essentially the same factors already described in Section 4.5 for equipment qualification. The main difference is that because of the different approaches used for equipment qualification and emergency planning, the uncertainties in the retention in the primary coolant system are more important for emergency planning. Although for many sequences with little retention anticipated, the uncertainties associated with the fractions escaping from the PCS would be relatively low for most species, for some other sequences with significant retention possible, the uncertainties would be relatively high for many species. In addition, such retention of some species in certain sequences would be dependent on the details of the comparatively uncertain thermal-hydraulic conditions.

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*A more complete description of many types of uncertainties is given in Appendix E.

Like the uncertainties for both release from the core materials and escape from the coolant system, the uncertainties associated with the estimates of escape from the containment are also dependent on the species, with the uncertainties for noble gases being smallest. In addition, the uncertanties depend on the accident scenario. For example, for accidents involving early containment failure the uncertainties associated with transport within the containment are similar for most species whereas for accidents involving either delayed failure or no failure of the containment, the uncertainties are different and depend upon the possible fates of the various species. Except for iodine, these detailed fates are not generally addressed and so the uncertainties are large for some species. The largest uncertainties of importance are those associated with the mode and timing of containment failure. In a meltdown accident if the timing of the failure of the containment is changed somewhat, the releases of radioactive materials to the environment can change substantially (see Table 5.5).

5.7 SOURCE TERMS FOR REGULATORY GUIDANCE FOR EMERGENCY PLANNING

To estimate source terms for emergency planning, the assumptions and procedures outlined in Chapter 3 of this report might be implemented to generate source terms for an appropriate spectrum of accidents for the plant under consideration. Unfortunately, such an undertaking is a difficult process and includes the potential for many problems. Thus a more appropriate procedure might be to use "generic" source terms for various classes of accidents (Pasedag, Blond, and Jankowski, 1981).

For example, on the basis of the estimated magnitudes of the radionuclide releases to the environment for all the sequences considered in this project, all accident sequences can be divided into several classes. As one possibility, the six PWR classes and the five BWR classes listed in Table 5.6 might be considered. Alternatively, four composite classes, such as those summarized in Table 5.7, might be formed.

Table 5 5 Impacts of problems in estimating magnitudes of source terms on emergency planning accident sequences

Phenomena	Description[a]	Problems[a]	Potential impact: Lesser-core damage accidents	Potential impact: Meltdown accidents
Releases				
Releases before vessel failure other than leaching	Empirical temperature-dependent rates	Insufficient data	Large	Large[b]
	Thermal-hydraulic estimates	Insufficient data, inadequate models		
Leaching releases	Empirical rates	Insufficient data	Small	Small
Releases after vessel failure	Empirical temperature-dependent rates	Insufficient data, inadequate model	-	Large
Transport Through Primary Coolant System				
Transport through coolant system atmosphere	"First-principles" transport code	Insufficient data, inadequate models	Large	Large
	Thermal-hydraulic estimates	Insufficient data, Inadequate models		
Scrubbing in coolant system	Empirical rates	Insufficient data, Inadequate models	Large	Large
Transport through containment				
Transport in containment	Empirically-based code	Insufficient data, inadequate models	Small	Large
	Thermal-hydraulic estimates	Insufficient data, inadequate models		
Leakage or other release from containment	Empirically-based rates	Inadequate models for failure modes	-	Large

[a]See Table 3 1 in Chapter 3 for more details

[b]For some radionuclides

Table 5.6 Estimated values of updated total containment escape fractions

Class	Corresponding RSS categories	Cumulative fractions of core inventory released to atmosphere[a]						
		Xe-Kr[b]	I-Br[c]	Cs-Rb	Te-Sb	Ba-Sr[d]	Ru[e]	La
PWR I′	PWR 2 (V only)	1.0	0.8	0.9	0.9	0.5;0.2	0.03;0.2	0.005
PWR I[f]	PWR 1	-	-	-	-	-	-	-
PWR II	PWR 2 + PWR 3	1.0	0.3	0.6	0.6	0.5;0.2	0.04;0.3	0.01
PWR III	PWR 4 + PWR 5	1.0	0.01	0.01	0.01	0.01;0.004	0.003;0.03	0.002
PWR IV	PWR 6 + PWR 7	1.0	0.003	0.001	0.001	4(-4)[g];2(-4)	3(-5);3(-4)	1(-5)
PWR V	PWR 8 + PWR 9	1.0	5(-4)	0.001	3(-4)	2(-5);1(-5)	1(-6);1(-5)	3(-8)
BWR I′	BWR 2	1.0	0.6	0.8	0.7	0.1;0.05	0.01;0.1	0.003
BWR I	BWR 1	-	-	-	-	-	-	-
BWR II	BWR 2 + BWR 3	1.0	0.5	0.2	0.3	0.1;0.06	0.02;0.1	0.007
BWR III	BWR 4	1.0	0.001	0.005	0.006	0.003;5(-4)	4(-4);0.003	2(-4)
BWR IV	BWR 5	1.0	3(-10)	1(-8)	2(-9)	2(-10);1(-10)	1(-11);1(-10)	3(-13)

[a]The values given here for each class are not necessarily all-inclusive; instead they are representative of those sequences explicitly considered in this report.

[b]Ignores decay.

[c]Represents fraction of iodine present as elemental iodine which escapes.

[d]First value based on barium data; second value based on strontium data.

[e]First value based on ruthenium data; second value based on molybdenum data.

[f]Steam explosions were not considered in this report.

[g]4(-4) denote 4×10^{-4}.

Table 5.7 Estimated values of updated total containment escape fractions for generic accident classes

Class	Corresponding RSS categories	Cumulative fractions of core inventory released to atmosphere[a]						
		Xe-Kr[b]	I-Br[c]	Cs-Rb	Te-Sb	Ba-Sr[d]	Ru[e]	La
1	PWR 1 + PWR 2 + PWR 3; BWR 1 + BWR 2	1.0	0.8	0.9	0.9	0.5;0.2	0.03;0.2	0.005
2	PWR 4 + PWR 5; BWR 4	1.0	0.01	0.01	0.01	0.01;0.004	0.003;0.03	0.002
3	PWR 6 + PWR 7	1.0	0.003	0.001	0.001	4(-4)[f];2(-4)	3(-5);3(-4)	1(-5)
4	PWR 8 + PWR 9 BWR 5	1.0	5(-4)	0.001	3(-4)	2(-5);1(-5)	1(-6);1(-5)	3(-8)

[a]The values given here for each class are not necessarily all-inclusive; instead they are representative of those sequences explicitly considered in this report.

[b]Ignores decay.

[c]Represents fraction of iodine present as elemental iodine which escapes.

[d]First value based on barium data; second value based on strontium data.

[e]First value based on ruthenium data; second value based on molybdenum data.

[f]Steam explosions were not considered in this report.

[g]4(-4) denote 4×10^{-4}.

Although the accident sequences of interest vary greatly among different plant designs, sequences which fall within each of the accident sequence classes have been found at each type of plant considered to date. (See the addendum in Appendix E.) As a result, although the probabilities of occurrence of each class probably vary widely among plants (see Table E.4 of Appendix E), the magnitudes of the potential releases vary over the same basic ranges.

In general, most aspects of source term estimation which are currently thought to be potentially conservative and which might yield reductions of the estimated releases as the result of work in the near future will not result in reductions of the releases for all accident sequences. Therefore, in a sense, such improvements to a large extent may just change the relative probabilities of the various classes of accidents and not the anticipated releases for the classes.

Inasmuch as source terms for emergency planning would often be used in a probabilistic framework, best estimates such as those obtained in the illustrative calculations presented in the chapter are indicated. As a result, it is probably reasonable to adopt accident classes and associated source terms such as those presented in Table 5.7 for regulatory guidance for emergency planning.

References for Chapter 5

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6. LIMITATIONS AND RECOMMENDATIONS

6.1 LIMITATIONS

6.1.1 Assumptions and Procedures

The project reported in this document was to use the information provided in the Technical Bases Report (USNRC, 1981) as the starting point for formulating assumptions appropriate for use in regulatory guidance. Regretfully, the information presented in the Technical Bases Report was incomplete in some respects. Because of programmatic limitations, it was not possible to address most of the previously undertreated topics at a greater level of detail than had been done before. Thus this report necessarily suffers from many of the same problems as the Technical Bases Report.

Among the acknowledged shortcomings of the work performed for the Technical Bases Report (Sherry et al., 1981) are the following:

1. A systematic analysis of fission product transport from the fuel to the environment was not performed for that study. Thus the effects of certain important interdependencies in the overall problem of source term estimation were ignored.
2. No detailed analysis of the uncertainties was made. (Unfortunately, some of these uncertainties are large.)
3. In many cases only a cursory examination of the transport behavior of potentially important fission product species (e.g., Te, Ru, Sr) was made.
4. Only a very limited number of accident sequences were evaluated so that the full range of possible accident conditions may not have been adequately covered.
5. A number of physical processes which may have the potential to significantly affect fission behavior were not evaluated in depth. For example, the effects of hydrogen combustion in the containment and the effects of attenuation of aerosols along leak paths were not addressed.

6. For the most part, fission product behavior during actual previous reactor accidents was not analyzed.
7. Fission product release and transport during accident sequences in which steam explosions are postulated to occur were not addressed.

To a large extent, many of the shortcomings of the Technical Bases Report merely reflect the lack of appropriate data and/or understanding which currently exists in certain areas. For example, at least the following potentially important areas have been understudied experimentally in the past: (1) the chemistry of almost all elements in accident environments; (2) the effects of hydrogen burning; (3) the release rates from core materials and/or corium for many elements; (4) the retention of vapors and aerosols in the primary coolant system; (5) the effects of condensing steam environments on aerosol behavior; (6) the effects of core-concrete interactions; (7) aerosol formation and behavior in LWR accident conditions; and (8) the removal of radionuclides by suppression pools and ice-bed condensers. (The needs in many of these areas are discussed in more detail in the Technical Bases Report [Sherry et al., 1981] and so are not considered here.) Furthermore, most of these areas have not been adequately treated theoretically. Thus, many of the shortcomings of the Technical Bases Report are currently "necessary." As a result, these problems are of necessity not treated in detail in this report.

For the purpose of using the information provided in the Technical Bases Report in this project, there are some problems other than those already mentioned. Among those additional problems are the following:

1. The overall attitude of that document was in many respects a research oriented one, with much of the work not being directly applicable to regulation. For example, although some very interesting calculations were performed to investigate potential radionuclide retention in the primary coolant system, that work is not currently usable for regulatory considerations.

2. The rushed nature of the work performed for that study is apparent throughout the report. For example, there are certain inconsistencies and some potentially treatable areas not addressed in the report.
3. Variations among reactors were not adequately addressed in many portions of that document. This was especially the situation for the consideration of releases from the core and structural materials and transport through the primary coolant system.
4. Reasonable variations among possible accident conditions were not always addressed. For example, the effect of high pressures in the reactor vessel on releases from the core and structural materials was not considered.
5. The document sometimes implies that a consensus exists in certain technical areas where indeed no consensus of the experts does exist. Because definitive answers do not exist in many areas of source term estimation, the extent of disagreement among the experts must be regarded as one very important indicator of the uncertainty associated with various aspects of source term estimation.
6. No attempt was made to put the accident descriptions in any appropriate perspective. Namely, little consideration was made of the strong dependence of individual accident descriptions upon many somewhat arbitrary assumptions.

In addition to the limitations associated with the Technical Bases Report and the information behind it, there are other limitations of the assumptions and procedures adopted here. Many of them result basically from the inherent complexity of source term estimation. For example, because some of the state-of-the-art procedures for estimating source terms involve the use of complicated models and/or intricate computer codes, there is much room for error in the application of the indicated procedures. It is far beyond the scope of this project to describe all the potential pitfalls associated with the procedures. Furthermore, other limitations exist because the models and codes themselves include many sources of error, such as unmodeled processes, mismodeled processes, and so forth, and so they are not necessarily applicable for all or even any accidents. Other potential problems arise because the models and codes which represent the "best" methods were not developed with regulatory uses in mind. Consequently, because both the models and codes and the use of the models and codes are associated with large uncertainties, the general application of the adopted methods for source term estimation for regulatory guidance must be acknowledged to be fraught with many potential problems.

A further limitation associated with the work presented in this report is the consideration of only accidents within the current design basis for equipment qualification.* Such restricted consideration ignores the potential impact of more severe accidents such as meltdowns. In addition, it essentially precludes any useful consideration of the time-dependent character of the radionuclide releases for all accidents.

6.1.2 Illustrative Estimates

The approach taken in this document is built on the work of many others. While better methods and procedures than some of those used here probably can and will be developed, they are not presently available. In addition, although data better than much of that now available probably can and are being obtained, they are not currently available. Thus the source term estimates presented here in some sense represent the state-of-the-art. However, the quality of the estimates in this report necessarily has been restricted by the limited scale of this project. In particular, there has not been time to straighten out or resolve certain inconsistencies and differences of opinion found in previous work and in the technical community. Also, previously performed calculations not initially intended to be used for the purposes of this document have been utilized as the bases for the estimates. In addition to this, there are all the uncertainties associated with any assessment of source terms.

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*The restriction of the accidents considered for equipment qualification to just those within the current design basis was part of the definition of the scope of this project by the funding agency.

6.2 RECOMMENDATIONS

As the result of the insights gained during this study, the following recommendations are made for future efforts in source term work:

1. The experimental research areas mentioned in the discussion of understudied areas should be investigated. In addition, improved models and codes for estimating releases of radionuclides and aerosols, transport of materials through the primary coolant system, movement of materials within the containment and thermal-hydraulic conditions throughout the plant are indicated.
2. For some problems, a more scientific approach is indicated. For example, the fitting of curves to unrelated sets of data is often not a meaningful approach.
3. The sensitivities and uncertainties of source term estimation should receive far more consideration than they have received in the past. In particular, something beyond mere parametric analyses of computer codes is indicated. Much more effort should be expended to address the uncertainties associated with the problems themselves and not just those associated with the related models and computer codes.
4. Interaction among the people involved in various aspects of research and other investigations related to source term estimation should be fostered. In particular, more effort should be made to promote exchange of knowledge between chemists and nuclear engineers. It is clear that some chemists participating in research relevant to source term work are relatively ignorant of certain basic aspects of nuclear engineering which are important to source term estimation. Likewise, some nuclear engineers are comparatively unversed in the basic chemical factors which are important to source term estimation. In addition, more cooperation between workers at different laboratories involved in source term work would be beneficial.

5. An attempt should be made to incorporate much needed new perspectives and updated approaches into all aspects of source term work.

References for Chapter 6

Sherry, R. R., M. A. Cunningham, C. N. Kelber, and M. Silberberg, 1981. "Introduction, Summary and Conclusions," Chapter 1 in *Technical Bases for Estimating Fission Product Behavior During LWR Accidents*, NUREG-0772, U. S. Nuclear Regulatory Commission, Washington, D. C.

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APPENDIX A

ACCIDENT DESCRIPTIONS AND TERMINOLOGY

A.1 INTRODUCTION

Accidents involving core damage in light water reactors* can be divided into two basic groups: those accidents resulting in limited core damage (less than complete melting of the core) and those accidents resulting in complete meltdown. Examples of both of these groups are described in this appendix.

A.2 LIMITED-CORE-DAMAGE ACCIDENT DESCRIPTIONS

Limited-core-damage accidents are considered in this report for both equipment qualification and emergency planning purposes. On the basis of performance of the engineered safety features, two basic classes of limited-core-damage accident sequences can be defined: a class of accident sequences in which the engineered safety features function as anticipated after some accident initiating event, for example, a coolant system pipe break or a transient event, and another class in which one or more of the required engineered safety features function in a degraded mode after some accident initiator. On the basis of the characteristics of radionuclide transport in the primary coolant system, each of these classes can be further divided into two subclasses: one in which any released radionuclides must travel through water to reach the containment and another in which any released radionuclides do not pass through water to reach the containment during most of the accident. Examples of accidents in each of these classes and subclasses are described in the following paragraphs.

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*Descriptions of the two basic types of light water reactors, namely, pressurized water reactors (PWR's) and boiling water reactors (BWR's), are given in the Technical Bases Report (Denning, 1981a). Descriptions of the variations of these two basic types of reactors which are considered in this report are also given in the Technical Bases Report (Denning, 1981a), as well as in the Reactor Safety Study Methodology Applications Program (RSSMAP) reports (Carlson et al., 1981; Hatch, Cybulskis and Wooton, 1981; and Kolb et al., 1981).

A.2.1 Accidents With Expected Functioning of Engineered Safety Features

Rod ejection. The ejection of a control rod from the reactor core in a PWR (or the dropping of a control rod in a BWR) would result in overheating of only a very localized region of the core. If the reactor protection systems functioned as intended, the transient produced by rod ejection would be countered by functioning of the reactor shutdown system. Likewise, if the engineered safety features functioned appropriately, the primary coolant system rupture (in the head of the reactor vessel caused by the rod ejection) would be compensated for by activation of the emergency core cooling system. Typically, the only radionuclide releases would be "gap" releases from a small region of the core, although, in some cases, much more extensive releases resulting from damage up to and including catastrophic rupturing of fuel rods in the vicinity of the ejected control rod could occur. Because the core would be surrounded by water, any radionuclides released from the core would directly enter the water in the primary coolant system.

Main steam line break. In the event of a large pipe break in a main steam line in a PWR, steam would be lost rapidly, at least initially, from the steam line. The increased steam flow would cause increased heat removal from the primary coolant system and therefore decreased temperatures and pressures of the coolant in the primary system. If the engineered safety features operated as expected, any overheating of the core would be rather restricted. In general, the releases of radioactivity from the core would be very small; the only possible radionuclide releases would be "gap" and leaching releases from a few fuel rods located throughout the core. Any radionuclides released would enter the primary coolant system water directly. Unless a leakage path from the primary coolant system existed or developed, the released radioactivity would remain in that system.

Primary coolant system pipe break. In the event of a large pipe break in the primary coolant system, water would be lost rapidly from the core region. If the engineered safety features operated as expected, the overheating of the core before the emergency core cooling system reflooded the core region would be small. The overheating would generally be much more extensive than for either of the previously mentioned two accidents. "Gap" releases might occur for a large fraction of the core, although typically they would occur only for a small fraction of core. In addition, leaching of ruptured fuel rods might occur after the core was reflooded. In contrast to the situation for a large pipe break, a small pipe break would result in a slower rate of water loss from the core region. If the emergency core cooling functioned as expected, the core would not be uncovered and overheating would be minimal. Thus essentially no radioactivity would be released from the core.*

A.2.2 Accidents With Degraded Functioning of Engineered Safety Features†

Delayed functioning of the emergency core cooling system given a large pipe break. In the event of a large pipe break in the primary coolant system, coolant would be rapidly lost from the reactor vessel and the core would become uncovered. If the emergency core cooling system were to fail to deliver water to the reactor coolant system, the core would heat up and start to melt. The extent of melting, and therefore the amounts of various radionuclides released, would depend upon the length of the delay before the emergency core cooling system started to operate. The location of the break would have a substantial effect on the conditions encountered by radionuclides as they were transported through the reactor coolant system.§ Although the initial releases would often encounter a relatively dry pathway to the contain-

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*This description of a small pipe break depends upon the operator recognizing the existence of the break and activating the engineered safety features before the break is detected by safety instrumentation.

†The descriptions in this subsection were adopted from NUREG-0772 (Denning, 1981a).

§For a hot-leg pipe break close to the reactor pressure vessel, the path through the primary coolant system would be short and the attenuation due to transport through the coolant system often would be negligible; in contrast, for a cold leg pipe break, the path through the coolant system could be relatively long and retention in the coolant system could be substantial for some radionuclides.

ment, at least the releases toward the end of the accident would encounter water in that pathway.

Delayed functioning of the emergency core cooling system given a transient event. In the event of a transient with loss of the normal heat sinks, water would eventually be boiled off the core and lost through the relief valves. If the emergency core cooling system were to fail to deliver water to the reactor coolant system, the core would begin to melt. Melting would be terminated if the emergency core cooling system started to operate before too much of the core were melted and complete meltdown of the core became inevitable. In a PWR, any radionuclides released from the core might pass through water in the pressurizer quench tank on their way to the containment. In a BWR, the released radionuclides would typically bypass the drywell, be released through the relief valves and enter the suppression pool. In both cases, the amounts of many radionuclides ultimately reaching the containment (or reactor building) atmosphere and the environment might be much smaller than the total amounts released from the core.

A.3 MELTDOWN ACCIDENT DESCRIPTIONS

Meltdown accidents are considered in this report only for emergency planning purposes.* Some of the meltdown accident sequences thought to dominate the risk to the human population for various categories of accident sequences are described here for each of the five different types of reactors investigated in this study. The descriptions given here have primarily been excerpted and/or adapted from the indicated source documents.† [More detailed descriptions of all the accident

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*The work reported in this document considered only accidents within the design basis for equipment qualification.

†It should be noted that the descriptions presented here for many of the meltdown accident sequences are dependent upon certain assumptions utilized in the studies on which these descriptions are based. For example, the timing and the mode of containment failure are both highly dependent upon the assumed containment failure pressure. Different assumptions than those utilized in these studies would have resulted in some not so trivial differences in the descriptions of some of the accident sequences.

sequences presented here, in addition to descriptions of many other meltdown accident sequences, are given in the Reactor Safety Study (RSS; USNRC, 1975), the Reactor Safety Study Methodology Application Program (RSSMAP) reports (Carlson et al., 1981; Hatch, Cybulskis, and Wooton, 1981; and Kolb et al., 1981), and NUREG-0772 (Denning, 1981b).] The notation used in this appendix and throughout this report to characterize the accident sequences for each of the five reactors (the two RSS reactors and the three RSSMAP reactors) is summarized in Tables A1.1–A1.5.

A.3.1 RSS PWR (Large Containment PWR)*

TMLB'-δ, γ, ε – Loss of reactor coolant system heat removal and loss of all AC power given a transient event; containment failure due to hydrogen burning, overpressurization or basemat melt-through. Reactors are designed so that if offsite power is lost and the diesel generators which provide an emergency source of AC power fail to operate, decay heat can still be removed from the reactor coolant system through the steam generators fed by steam-turbine driven pumps.[†] With this mode of heat removal failed, however, emergency core cooling pumps, which are driven by AC power, would not operate and the inventory of reactor coolant water would eventually be boiled away through pressure relief valves. Similarly the AC powered containment safety features, such as the containment heat removal system, would not operate. The likelihood of early containment failure by overpressurization in this sequence would be very high and the consequences potentially severe. However, it water were present in either the pressurizer or the pressurizer quench tank, significant fractions of many of the radionuclides released prior to reactor vessel melt-through might be partially scrubbed by that water and prevented from entering the containment atmosphere and thus from escaping to the environment.

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*The descriptions in this subsection were adapted from NUREG-0772 (Denning, 1981a).

[†]It should be noted that not all reactors are equipped with such steam-driven pumps.

V – Interfacing systems loss of coolant accident. Check valves provide a barrier between the low pressure emergency core cooling system and the high pressure piping of the reactor coolant system. In the event that these valves should fail, pressures beyond the design capability of the low pressure system could be imposed on it. The subsequent failure of the system would result not only in loss of reactor coolant, but potentially also in failure of the emergency core cooling system. Since the low pressure piping is located in the auxiliary building, the failure of the reactor coolant system would be external to the containment building. Thus, radioactive material released into the primary coolant system would bypass the containment and thus the containment safety features. In the Reactor Safety Study, this sequence was assessed to be the highest contributor to risk to the human population for the specific PWR plant design analyzed.

S_2C-δ – Failure of containment spray injection given a small pipe break; containment failure due to overpressurization. In the RSS PWR, the early activation of the containment spray recirculation system after a small pipe break would lead to that spray system's failure if too little water were available in the containment sump. Failure of the containment spray recirculation in this plant design would also mean failure of containment heat removal. Therefore, in this accident sequence, the containment building would overpressurize and fail prior to core meltdown. Fuel melting would be delayed approximately one day after the start of the accident. Fission products would be released into a failed containment building. Consequently, the radioactivity released to the environment could be substantial.

S_2D-ε – Failure of the emergency core cooling system given a small pipe break; containment failure due to basemat melt-through. A small pipe break accident would result in a slower depressurization of the reactor coolant system than a large pipe break accident and, in the event of failure of the emergency core cooling system, a more delayed uncovering of the core. Containment safety features would be expected to be operational for this type of accident sequence. Inasmuch as the containment would probably not fail prior to basemat melt-through, the total releases of radioactivity from the containment would be relatively small.

A.3.2 RSS BWR*

TC-γ,γ´ - Failure of reactor shutdown systems given a transient event; containment failure due to overpressure with releases either direct to atmosphere or through reactor building. If the control rods failed to insert and the backup liquid neutron absorber system failed to operate in a transient event requiring reactor shutdown, the reactor power would level off at a heat generation rate well above decay heat. At the estimated power level, the high pressure coolant injection system would not have adequate capacity to match the boiloff of water from the coolant system. The core would eventually become uncovered, heat up, and possibly melt. The large quantity of heat transferred to the suppression pool would result in boiling in the pool, thus preventing further steam suppression and reducing the capability for scrubbing radioactive material. The containment would typically fail by overpressurization before meltdown. The resultant radioactive releases to the environment could be substantial.

TW-γ,γ´ - Failure of decay heat removal given a transient event; containment failure due to overpressure with releases either direct to atmosphere or through reactor building. If the decay heat removal system failed, the suppression pool would be predicted to heat up, boil, and, after an extended period of time, fail the containment by overpressurization. For the specific plant design analyzed, during depressurization of the containment, the emergency coolant pumps would be expected to cavitate with potential to stop delivering cooling water to the reactor vessel. Subsequently, the fuel could become uncovered, heat up, and melt. Fission products released to the reactor vessel would pass through the boiling suppression pool and might not be effectively scrubbed. Fission products released from the fuel to the drywell might pass through the boiling suppression pool or might bypass

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*The descriptions in this subsection were adapted from NUREG-0772 (Denning, 1981a).

the pool, depending on the location of containment failure. The consequences of this sequence could be large.

TQUV-γ,γ' – Failure of all makeup water given a transient event; containment failure due to overpressure with releases either direct to atmosphere or through reactor building. In this sequence, none of the potential sources of makeup water would be available following a transient-initiated shutdown of the reactor. In that situation, steam would be released from the reactor coolant system to the suppression pool through pressure relief lines. The fuel would become uncovered, heat up, and melt. The suppression pool would be subcooled and so could effectively scrub many of the radionuclides released prior to containment failure. The containment would typically fail some time after meltdown. As a result of both more effective scrubbing in the suppression pool and later containment failure, the radioactive releases to the environment would probably be smaller for this sequence than for the other two accident sequences considered for this plant.

A.3.3 Ice Condenser PWR*

S_2HF-γ† – Failure of the recirculation modes of the emergency core cooling system and the containment spray system given a small pipe break; containment failure due to hydrogen burning. For an ice condenser plant, a common cause failure of the two recirculation systems is relatively likely. In particular, the return lines from the upper deck to the containment sump were left closed or became blocked, water sprayed into the upper compartment would remain there. After the ice was all melted and the sump ran out of water, both the recirculation systems would fail. As a result, fuel would become uncovered, heat up and melt. The containment would be expected to fail about the time of reactor pressure vessel melt-through. Neither the ice nor the containment spray system would be available at this point to condense steam

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*Most of the descriptions in this subsection are based on Carlson et al. (1980).

†This description is based on Denning (1981a).

or remove radioactive material from the containment atmosphere. Therefore, the consequences to the human population could be substantial.

V – Interfacing systems loss of coolant accident. The discussion given in Subsection A.3.1 about the V sequence for the RSS PWR is appropriate.

S_1H-γ and S_2H-γ – Failure of the emergency core cooling recirculation system given a small pipe break; containment failure due to hydrogen burning. Although the emergency core cooling system would function initially, its failure in the recirculation mode would eventually result in uncovering of the core. Due to the small volume and low design pressure of the ice condenser containment, failure of that containment could very likely occur early in the accident as the result of substantial hydrogen burning. Because of the potential for early containment failure, the releases of radioactivity to the environment could be large, although the early functioning of both the containment sprays and the ice beds would partially mitigate the releases.

S_1HF-γ,δ – Failure of the recirculation modes of the emergency core cooling system and the containment spray system given a small pipe break; containment failure due to hydrogen burning or overpressurization. This sequence would be similar to the S_2HF sequence for this reactor.

TML-γ – Failure of reactor coolant system heat removal given a transient event; containment failure due to hydrogen burning. This sequence would involve loss of both the normal and the emergency means of adding water to the steam generators. Because of this, the steam generators would eventually boil dry and there would be no means of removing decay heat. The resultant increase in reactor coolant system pressure would lead to relief of steam through the safety and relief valves. Water would boil off from the core, leading to core melt. Containment failure would probably occur about the time of reactor vessel melt-through if hydrogen burning occurred. The associated radionuclide releases could be large. However, the operation of the sprays at least initially and the existence of unmelted ice throughout the accident would somewhat attenuate the releases to the environment. In addition, if water were present in the pressurizer quench tank, many of the radionuclides released prior to vessel melt-through might be scrubbed partially by that water.

S_1D-γ and S_2D-γ - Failure of the emergency core cooling injection system given a small pipe break; containment failure due to hydrogen burning. This sequence would be analogous to sequence S_2D for the RSS PWR. The major difference would be that due to the smaller volume and lower design pressure of the ice condenser containment, failure of the containment would probably occur much earlier than in the RSS PWR and would occur as the result of hydrogen burning. (Typically, in the RSS PWR, late failure would occur as the result of basemat melt-through.) Thus, the releases of radioactivity to the environment would be much larger than for the analogous sequences in the RSS PWR.

A.3.4 Mark III BWR*

TPQI-δ - Failure of decay heat removal given a transient event accompanied by a stuck open safety valve; containment failure due to overpressurization. This sequence would be characterized by containment failure preceding melting of the core. Following containment failure and subsequent core melting, the suppression pool would become saturated and the potential for radionuclide scrubbing could be diminished. Thus the radionuclide releases to the environment could be substantial.

TQW-δ - Failure of decay heat removal given a transient event; containment failure due to overpressurization. This sequence would be analogous to sequence TW for the RSS BWR. Containment failure would probably occur before melting of the core. As in the preceding case, the resultant radioactive releases could be large.

TC-δ - Failure of the reactor shutdown system given a transient event; containment failure due to overpressurization. This sequence would be similar to sequence TC for the RSS BWR. The containment would typically fail before meltdown started. Thus, the consequences could be substantial.

*These descriptions are based on Hatch, Cybulskis and Wooton (1981).

TQUV-δ,γ – Failure of emergency core cooling given a transient event; containment failure due to overpressurization or hydrogen burning. This sequence would be similar to sequence TQUV for the RSS BWR, except that containment failure by hydrogen burning is thought to be more likely in this plant. Containment failure by hydrogen burning would be expected to occur shortly after melting of the core whereas containment failure by overpressurization (due to generation of noncondensable gases) would not occur until some time after melting. Both modes of failure would be likely. The radionuclide releases would be smaller for this sequence than for the other three accident sequences discussed above for this plant because of both the attentuation of some of the radionuclides in the subcooled suppression pool and the delayed containment failure.

A.3.5 Alternate Large Containment PWR*

V – Interfacing systems loss of coolant accident. The discussion given in Subsection A.3.1 about the V sequence for the RSS PWR is applicable.

T_2MQ-FH-γ,ε† – Failure of the recirculation modes of both the containment spray and the emergency core cooling systems given a transient event accompanied by a stuck open safety valve; containment failure due to hydrogen burning or basemat melt-through. This sequence would be analogous to sequence S_2HF for an ice condenser PWR. As in the case for an ice condenser plant, failure of both recirculation modes could be caused by a common mode failure. However, in this reactor, the most likely common cause would be operator failure to realign both systems from the injection to the recirculation mode. Because of the larger size and the higher design pressure for this containment, early containment failure by hydrogen burning would not be as likely as in an ice condenser PWR and so delayed containment failure by basemat melt-through would also be probable in this plant

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*These descriptions are based on Kolb et al. (1981).

†Transient-induced LOCA's such as this were not analyzed in detail in the RSS.

design. The consequences for the former type of failure could be substantial whereas those for the latter type would be relatively insignificant.

S_3FH-γ,ε – Failure of the recirculation modes of both the containment spray and the emergency core cooling systems given a small pipe break; containment failure due to hydrogen burning or basemat melt-through. A stuck-open safety valve would be approximately equivalent to a small break LOCA. Therefore, the discussion just given for sequence T_2MQ-FH is applicable.

T_2MQ-H-γ,ε* – Failure of the recirculation mode of the emergency core cooling system given a transient event accompanied by a stuck-open safety valve; containment failure due to hydrogen burning or basemat melt-through. This sequence would be similar to sequence S_2H for an ice condenser PWR. Because of the larger size and higher design pressure for this containment, the conditional probability of early containment failure by hydrogen burning would not be as high and so there would also be a significant likelihood of delayed containment failure by basemat melt-through for this plant design. The radionuclide releases to the environment could be large if the containment failed relatively early.

S_3H-γ,ε – Failure of the recirculation mode of the emergency core cooling system given a small pipe break; containment failure due to hydrogen burning or basemat melt-through. A stuck-open safety valve would be equivalent to a small LOCA. Consequently, the discussion just given for sequence T_2MQ-H is applicable.

T_2MLUO-γ,ε – Failure of the feedwater system, the high pressure injection system and the reactor building cooling system given a transient event; containment failure due to hydrogen burning or basemat melt-through. Failure of the three indicated systems could result from a common cause, in particular, the failure of the low pressure service water (required for component cooling for all the systems). Basically, this sequence would be similar to sequence TML already

*Transient-induced LOCA's such as this were not analyzed in detail in the RSS.

described for the ice condenser PWR. Because of the larger size and the higher design pressure of this containment, the conditional probability of early containment breaching by hydrogen burning would not be as high and so there would also be a significant likelihood of later containment failure by basemat melt-through in this plant design. The releases of radioactivity in cases of early failure could be substantial. However, if water were present in the pressurizer quench tank, a significant fraction of the radioactivity released before reactor vessel melt-through could be scrubbed from the radionuclide stream to the containment.

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APPENDIX A1

NOTATION

A-16

Table A1.1. Symbols used to characterize RSS PWR accident sequences[a]

A -	Intermediate to large loss-of-coolant accident (LOCA).
B -	Failure of electric power to engineered safety features.
B′ -	Failure to recover either onsite or offsite electric power within about 1 to 3 hours following an initiating transient which is a loss of offsite AC power.
C -	Failure of the containment spray injection system.
D -	Failure of the emergency core cooling injection system.
F -	Failure of the containment spray recirculation system.
G -	Failure of the containment heat removal system.
H -	Failure of the emergency core cooling recirculation system.
K -	Failure of the reactor protection system.
L -	Failure of the secondary system steam relief valves and the auxiliary feedwater system.
M -	Failure of the secondary system steam relief valves and the power conversion system.
Q -	Failure of the primary system safety relief valves to reclose after opening.
R -	Massive rupture of the reactor vessel.
S_1 -	Small LOCA with an equivalent diameter of about 2 to 6 inches.
S_2 -	Small LOCA with an equivalent diameter of about 1/2 to 2 inches.
T -	Transient event.
V -	Low pressure injection system check valve failure.
α -	Containment rupture due to a reactor vessel steam explosion.
β -	Containment failure resulting from inadequate isolation of containment openings and penetrations.
γ -	Containment failure due to hydrogen burning.
δ -	Containment failure due to overpressure.
ε -	Containment failure due to basemat melt-through.

[a]Based on RSS (USNRC, 1975).

Table A1.2. Symbols used to characterize RSS BWR accident sequences[a]

A -	Rupture of reactor coolant boundary with an equivalent diameter of greater than 6 inches.
B -	Failure of electric power to engineered safety features.
C -	Failure of the reactor protection system.
D -	Failure of vapor suppression.
E -	Failure of emergency core cooling injection.
F -	Failure of emergency core cooling functionability.
G -	Failure of containment isolation to limit leakage to less than 100 volume percent per day.
H -	Failure of core spray recirculation system.
I -	Failure of low pressure recirculation system.
J -	Failure of high pressure service water system.
M -	Failure of safety/relief valves to open.
P -	Failure of safety/relief valves to reclose after opening.
Q -	Failure of normal feedwater system to provide core make-up water.
S_1 -	Small pipe break with an equivalent diameter of about 2 inches to 6 inches.
S_2 -	Small pipe break with an equivalent diameter of about 1/2 inches to 2 inches.
T -	Transient event.
U -	Failure of high pressure coolant injection system or reactor core isolation cooling system to provide core make-up water.
V -	Failure of low pressure emergency core cooling system to provide core make-up water.
W -	Failure to remove residual core heat.
α -	Containment failure due to steam explosion in vessel.
β -	Containment failure due to steam explosion in containment.
γ -	Containment failure due to overpressure - release through reactor building.
γ' -	Containment failure due to overpressure - release direct to atmosphere.
δ -	Containment isolation failure in drywell.
ε -	Containment isolation failure in wetwell.
ζ -	Containment leakage greater than 2400 volume percent per day.
η -	Reactor building isolation failure.
θ -	Standby gas treatment system failure.

[a]Based on RSS (USNRC, 1975).

Table A1.3. Symbols used to characterize ice condenser PWR accident sequences[a]

A	-	Large LOCA (>6 in.).
B	-	Failure of electric power system.
C	-	Failure of containment spray injection system.
D	-	Failure of emergency core cooling injection system.
E	-	Failure of emergency cooling functionability.
F	-	Failure of containment spray recirculation system.
G	-	Failure of containment heat removal system.
H	-	Failure of emergency core cooling recirculation system.
K	-	Failure of reactor protection system.
L	-	Failure of auxiliary feedwater system.
M	-	Failure of power conversion system.
P	-	Failure of safety/relief valves to open.
Q	-	Failure of primary system safety/relief valves to reclose after opening.
S_1	-	Small LOCA (2 in. $\leqq$ D $\leqq$ 6 in.).
S_2	-	Small LOCA (<2 in.).
T	-	Transient event.
U	-	Failure of chemical and volume control system.
V	-	Interfacing systems LOCA.
X	-	Failure of air return fan system.
Z	-	Failure of ice condenser system.
α	-	Containment rupture due to a reactor vessel steam explosion.
β	-	Containment leakage.
θ	-	Core debris fragmentation upon contact with water in the reactor cavity.
γ	-	Containment rupture due to hydrogen burning.
δ	-	Containment rupture by overpressurization.

[a]Based on Carlson et al. (1981).

Table A1.4. Symbols used to characterize Mark III BWR accident sequences[a]

A	-	Large LOCA.
C	-	Failure to render the reactor subcritical.
D	-	Failure of vapor suppression system.
E	-	Failure of the emergency core cooling system.
I	-	Failure of residual heat removal systems after a LOCA (including transient-induced LOCAs).
M	-	Failure of safety/relief valves to open.
P	-	Failure of a safety/relief valve to reseat.
Q	-	Failure of the power conversion system.
S	-	Small LOCA (the break area is less than one square foot).
T_1	-	A loss of offsite power transient.
T_{23}	-	Any other transient which requires an emergency reactor shutdown.
U	-	Failure of the high pressure core spray or reactor core isolation cooling system.
V	-	Failure of the low pressure emergency core cooling systems to provide core flow.
W	-	Failure of the residual heat removal systems after a transient.
α	-	Containment failure due to a steam explosion.
γ	-	Containment failure due to an overpressure caused by rapid hydrogen burning.
δ	-	Containment failure due to an overpressure caused by gas generation.

[a]Based on Hatch, Cybulskis, and Wooton (1981).

Table A1.5. Symbols used to characterize alternate large containment PWR accident sequences[a]

A	-	Large LOCA (D > 13.5").
B_3	-	Failure of emergency power system.
C	-	Failure of containment spray injection system.
D	-	Failure of emergency coolant injection system.
F	-	Failure of containment spray recirculation system.
G	-	Failure of low pressure recirculation system heat exchange.
H	-	Failure of emergency coolant recirculation system.
K	-	Failure of reactor protection system.
L	-	Failure of emergency feedwater system, recovery of power conversion system and high head auxiliary feedwater system.
M	-	Failure of power conversion system (normal operation).
O	-	Failure of reactor building cooling system.
O´	-	Failure of containment spray injection system.
P_1	-	Failure of demand for safety/relief valves.
P_2	-	Failure of safety/relief valves to open.
Q	-	Failure of reclosure of pressurizer safety/relief valves.
S_1	-	Intermediate LOCA (10" < D ≦ 13.5").
S_2	-	Small LOCA (4" < D ≦ 10").
S_3	-	Small-small LOCA (D ≦ 4").
T_1	-	Loss of offsite power transient.
T_2	-	Loss of power conversion system transient caused by other than a loss of offsite power.
T_3	-	Transients with the power conversion system initially available.
U	-	Failure of high pressure injection system.
V	-	Interfacing systems LOCA.
Y	-	Failure of reactor building cooling system during injection phase.
Z	-	Failure of reactor building coolant system during recirculation phase.
α	-	Containment rupture due to a reactor vessel steam explosion.
β	-	Containment leakage.
γ	-	Containment rupture due to hydrogen burning.
δ	-	Containment rupture by overpressurization.
ε	-	Containment failure by basemat melt-through.

[a]Based on Kolb et al. (1981).

APPENDIX B

FISSION PRODUCT RELEASE FROM FUEL

B.1 INTRODUCTION

During any accident involving damage to the core, radionuclides could be released from the core materials. Initially those releases would be into the reactor pressure vessel. If the accident progressed far enough, releases directly into the containment could also occur. The extent of release of any radionuclide from the core materials would be a function of the accident scenario, as well as of the characteristics of the radionuclide.

This appendix addresses the subject of fission product release from fuel in the following manner. First, qualitative descriptions of processes resulting in release from fuel are given, followed by a summary of the approach taken in the Reactor Safety Study (RSS; USNRC, 1975) to quantify these processes. Next, a discussion of information made available since the RSS work with which the release-from-fuel quantification might be updated is provided. An approach to estimation of the fission product release-from-fuel, expressed as the fraction of the core inventory released (at the time the accident begins) is then proposed, based on the review of currently available information. Following a discussion of the proposed approach are the results of calculations made using this approach to estimate fractional releases from fuel for the several representative accident sequences examined. Finally, limitations of the approach with regard to uncertainties and gaps in available knowledge on the subject are discussed, and a summary of this appendix is provided.

B.2 DESCRIPTION OF PROCESSES

Processes contributing to the release of fission products from fuel during a light water reactor (LWR) accident where core damage has occurred have been classified as follows (Baybutt, Nicolosi and Raghuram, 1981):

1) Cladding rupture,
2) Transport from solid fuel matrix prior to melting,
3) Evaporation from molten fuel in pressure vessel,
4) Leaching of fuel following cladding failure,

5) Oxidation of fragmented fuel, and
6) Sparging of fuel/concrete mixture by concrete degradation products after pressure vessel melt-through.

The degree to which any or all of these processes lead to fission product release is a function of many variables including the time/temperature profile of the core during the accident, the physical and chemical forms of fission products within the fuel prior to and during the accident, and the total inventory of fission products in the core at the time of the accident. In addition, recent experiments have indicated a potential pressure dependence of release from fuel (Malinauskas et al., 1980). Since we are dealing with release as a fraction of the core inventory here, total inventory will not come into the present discussion. The temperature profiles in the core as a function of time are extremely important in determining the fission product release. Not only are the six processes listed above very temperature dependent, but they are also affected by other time and temperature-dependent processes, such as cladding and structural material melt, which may alter the fission product release (Baybutt, Nicolosi and Raghuram, 1981; Wichner, Kress and Lorenz, 1981). Physical and chemical forms of fission products in the fuel throughout an accident will affect the amount released due to the volatility and migration behavior characteristic of each form (Wichner, Kress and Lorenz, 1981). These forms are, in turn, determined by temperature and other environmental factors, such as oxygen availability, which may change drastically throughout the course of an accident. Thus, release-from-fuel will depend to a great extent on the accident scenario described.

Cladding rupture may occur when the core heats up to temperatures between 750° and 1100°C. Such an event leads to a rapid release of volatile radionuclides which have accumulated in the plenum, or gap between the fuel and cladding, in pellet-pellet interfaces, and in pellet cracks during reactor operation. This burst release may be immediately followed by an additional release via diffusion of gas atoms which are associated with the interconnected void volume surfaces (Wichner, Kress and Lorenz, 1981). The combination of these two mechanisms of release is sometimes termed gap release (Baybutt, Nicholosi and Raghuram, 1981).

Diffusion from the solid fuel matrix, over and above the diffusion considered with a gap release, becomes appreciable above temperatures of 1350°C, during heat up of the core prior to melting (Baybutt, Nicolosi and Raghuram, 1981; Wichner, Kress and Lorenz, 1981). Another mechanism involved above 1350°C has been identified as the release of fission products accumulated at the grain boundaries within the fuel matrix as a result of the formation, swelling, and coalescence of bubbles of fission gas (Wichner, Kress and Lorenz, 1981). Depending on the accident scenario, the release from the core via these diffusion processes may be altered to varying degrees as a result of melting and oxidation of the zircaloy cladding (Baybutt, Nicolosi and Raghuram, 1981; Wichner, Kress and Lorenz, 1981).

Evaporation of fission products (melt release) is a process which occurs at temperatures above that at which the fuel becomes molten. Such evaporation can be expected to occur throughout the remainder of the accident, until the molten core material is quenched. The melting point of fuel pellets may vary depending somewhat on interactions of the fuel with molten cladding, structural materials, and control rods (Wichner, Kress and Lorenz, 1981). Diffusion of radionuclides to the surface of the molten mass which includes fuel, cladding, structural and control rod materials, and diffusion through the boundary layer is a function of their vapor pressure over the melt surface (Baybutt, Nicholosi and Raghuram, 1981; Wichner, Kress and Lorenz, 1981).

Leaching of fission products from fuel is a process which may follow cladding failure and core melting when water is introduced into the reactor vessel before vessel failure, or when the core drops into the reactor cavity which may be full of water. The degree of leaching will depend on the temperatures and chemistry involved for each fission product. Little discussion of the mechanisms behind this process has been found in the reactor accident literature.

Oxidation release is the term describing the phenomena of oxidation of finely divided UO_2 particles and subsequent vaporization of material existing in the particles following an event such as a steam explosion, in which fragmentation of fuel occurs (USNRC, 1975; Baybutt, Nicolosi and Raghuram, 1981). This process is distinguished from the evaporation process because of the extremely different geometries (i.e., fragmented particles versus a larger molten mass) which affect diffusion to the surface, and thus, release of fission products. Depending on the time during the accident in which this phenomenon occurs, releases may or may not be significantly enhanced.

Sparging is a process which occurs after the molten core and structural materials have melted through the reactor vessel to the reactor cavity, at which time the core material reacts with the concrete basemat. The mechanism referred to in this process is that by which fission products are released from the molten core to the containment as a result of gas sparging of the melt (Baybutt, Nicolosi and Raghuram, 1981; Wichner, Kress and Lorenz, 1981). Such sparging is a result of the production of gas from thermal decomposition of concrete (H_2O vapor and CO_2), and from reaction of H_2O and CO_2 with the corium to form H_2 and CO. Although most materials released by this process are nonradioactive, in the form of inorganic oxides, sulfides, chlorides and fluorides (Powers, 1982), volatile fission products may enter the bubbles formed within the melt, travel to the surface, and be released (Baybutt, Nicolosi and Raghuram, 1981). Two immiscible phases of the core debris, an oxidic phase and a metallic phase, are generally considered through which gas sparging occurs. The nonradioactive aerosols produced by this process may be important in their potential for removing radioactive aerosols in the containment (Murfin, 1980). The aerosol production process is complicated by the potential for formation of thick crusts of decomposition products over the melt, after core debris solidification (Murfin, 1980).

B.3 REACTOR SAFETY STUDY APPROACH

The RSS approach to quantifying release of fission products from fuel during various accident sequences considers all of the processes discussed in Section B.2 above except leaching and diffusion from the fuel matrix. Total release fractions for each process are provided, with crude temporal considerations associated with the melt release and sparging (termed "vaporization") release processes. The means by which the magnitude of each process was estimated in the RSS are described briefly below.

The gap release component in the RSS considers release only at the time of cladding rupture. The gap release fraction is divided into two fractions: (1) the "release fraction", and (2) the "escape fraction". The "release fraction" refers to the fraction of the core inventory of fission products accumulated in the plenum and other interconnected void spaces as a result of normal operation. This fraction is derived from three separate estimates, which are based on classical diffusion theory and empirically based diffusion coefficients. The amounts of short and long half-live (or stable) isotopes in the plenum and void space were averaged for each element to obtain these release fractions. The "escape fraction" refers to the fraction of the inventory in the gap and voids which escapes at cladding rupture, ignoring subsequent diffusion through voids (see Section B.1). All of the noble gases, one-third of the iodine (halogens), one-third of the cesium (alkali metals), 10^{-4} of the alkaline earths, 10^{-3} of the tellurium, and none of the other species in the gap are assumed to escape during the depressurization of the rod caused by the rupture. The rationale behind these fractions is based partially on volatility considerations, and thus, chemical form. The noble gases are, of course, considered inert. It is recognized in the RSS that iodine readily reacts with zirconium (present in the cladding), cesium, and hydrogen (from water vapor), forming compounds of varying volatilities. It is further recognized that thermodynamic analyses indicate CsI to be the most stable form of iodine at elevated temperatures, but that experimental data had not sufficiently established this compound as being the major chemical form of iodine, according to the RSS authors. However, experimental evidence existed suggesting partial retention of iodine by cladding, and thus, an escape fraction less than one was used. No chemical form is assumed for Cs (alkali metals), but the escape fraction was set equal to that for iodine in lieu of sufficient experimental data. Strontium (alkaline earths) is assumed to exist primarily in monoxide condensed phases, and the escape fraction was based on one set of experimental data. Tellurium is assumed to exist as either the element or oxide, and the escape fraction was again based on one set of experimental data.

The process of diffusion from a solid fuel matrix is not addressed in the RSS. The meltdown release component of the fission product source term in the RSS refers to the releases occurring during the period of time beginning with initiation of core melting and ending before pressure vessel melt-through. Release fractions are assumed to be proportional to the fraction of core melted, with all releases being assigned to the early period of first melting. This latter assumption is justified in the RSS by the observations that the highest steam flows and highest fuel surface areas are expected during the early period. Release fractions for each fission product were estimated from limited experimental data and thermodynamic considerations. Ninety percent of the noble gases and halogens (iodine) are assumed released. Alkali metals are considered not as volatile as the noble gases or halogens, and thus only eighty percent of the elements in this group are assumed released. Tellurium, although quite volatile as the element, is assumed to react extensively with unoxidized cladding, and thus release is estimated to range between only 5 and 25%. Similar reasoning and consideration of oxygen partial pressures, resulted in an assumed release fraction for the alkaline earths of 10%, and 3% for the noble metals. The rare earths and refractory oxides, although quite involatile, are assumed to vaporize at the same rate as UO_2, resulting in an estimated loss of 0.3%.

The oxidation release component of the RSS source term is assumed to occur if and when a steam explosion event takes place. Since such an event would occur as a result of contact of the hot core with water, it could theoretically occur prior to or following pressure vessel melt-through. The release fractions in the RSS as a result of a steam explosion were obtained from experimental data in which release during fuel oxidation by air at elevated temperatures were measured, since such an explosion would lead to exposure of fragmented fuel to the air-steam mixture in the containment or to the air outside the containment. The release is assumed to be instantaneous, with 90% of the noble gases, halogens, and noble metals, in the fraction of the fuel expected to be dispersed as a result of the explosion, being released. Sixty percent of the tellurium, selenium, and antimony are assumed released, and none of the remaining fission products are assumed released.

Finally, the sparging (vaporization) release component in the RSS is assumed to occur after pressure vessel melt-through. The constitution of the molten mass in the reactor cavity depends on many factors related to the phases existing within the mass, as a result of melting and oxidation of fuel, cladding, control rods, and structural materials. However, details of the properties of the molten mass were largely unknown, and thus, release estimates are stated to be highly uncertain. A simple approach was therefore taken, based on partitioning of fission products between the corium and gas which is sparging through the molten mass. Fission products remaining in the core and considered volatile (Xe, Kr, I, Br, Cs, Rb, Te, Se, and Sb) are assumed to be totally lost during the sparging phase, with a half-time of 30 minutes for the first 90 minutes of this phase, the remainder being lost in the next 30-minute period. This half-time was based on information indicating that sparging occurred to the greatest extent during the first half hour of core contact with the concrete. The remaining fission products, considered low in volatility, are assumed released to varying degrees depending on their volatility as oxides. Five percent of the noble metals are assumed to be released, and one percent of the alkaline earth and rare earth oxides are assumed released. The corresponding half-times for release of these low volatility compounds are assumed to be equal to those used for the volatile species.

The effects of fission product chemistry on source terms are considered in the RSS on the basis of thermodynamic calculations alone, with respect to the potential degree of oxidation of the fission product groups identified. Most of the discussion is qualitative with regards to the chemical forms existing in the core during an accident, especially with respect to the cladding interactions which are believed to greatly influence release of certain elements such as tellurium. It is recognized throughout the RSS that chemical form may greatly affect release fractions estimated, and that many complex interactions may occur during an accident sequence that are not quantifiable with information available.

The "best estimate" fractions provided in the RSS for each release process described above are summarized in Table B.1. It is cautioned that inventory balances for each fission product must be maintained in using the fractions, such that elements lost via one process are not available for loss by a subsequent process. The "best estimate" values were selected from a range of values given for each process, and considerable uncertainties were generally recognized for each value chosen.

B.4 WORK SINCE THE REACTOR SAFETY STUDY

A significant amount of both experimental and theoretical work has been carried out since the RSS for the purpose of evaluating fission product release from fuel during reactor accidents. Much of this work is ongoing, and thus, only preliminary results are, at best, available.

B.4.1 Experimental Work

A comprehensive review of the pertinent experimental work conducted up through 1980 has been provided in a report prepared for the Nuclear Regulatory Commission (USNRC, 1981). This report, hereafter referred to as the Technical Bases Report in this appendix, provides a description of the best technical information for estimating fission product release during postulated accidents that was available at the time the report was completed (March 1981). In the sections of the Technical Bases Report devoted to release from fuel, the best estimate fission product release rates derived are based on empirical data, and do not explicitly represent mechanisms of release as discrete events. Results of three separate sets of experiments were chosen to provide the data base (Lorenz et al., 1980a; Lorenz, Collins and Malinauskas, 1980; Lorenz et al., 1980b; Lorenz et al., 1981; Parker, Martin and Creek, 1963; Parker et al., 1967; Albrecht, Matschoss and Wild, 1979a; Albrecht, Matschoss and Wild, 1979b), pending more detailed analysis of other experiments described in the report. The experiments from which data were utilized were out-of-reactor tests, with one experimental design involving unclad fuel pellets (Parker, Martin and Creek, 1963; Parker et al., 1967). Tests were conducted in steam, inert, and air environments. Other

Table B.1. Fission product release-from-fuel fractions estimated in the Reactor Safety Study

Release component	Xe, Kr	I, Br	Cs, Rb	Te, Se, Sb	Sr, Ba	Ru, Rh, Pd, Mo, Te	Rare earths
Gas release (total)	0.03	0.017	0.05	10^{-4}	10^{-6}	---	---
release to gap	0.03	0.05	0.15	0.10	0.01	---	---
escape from gap	1	0.33	0.33	10^{-3}	10^{-4}	---	---
Meltdown release	0.90	0.90	0.80	0.15	0.10	0.03	0.003[a]
Vaporization release	1.0	1.0	1.0	1.0	0.01[b]	0.05	0.01
Oxidation release	0.90	0.90	---	0.60	---	0.90	---

[a] Includes Zr and Nb also.

[b] As refractory oxides.

experiments cited, but for which results had not been analyzed, include in-reactor tests and older out-of-reactor tests.

The empirical release data described above were used to estimate fission product release from fuel in the Technical Bases Report. Such data were utilized to generate graphical representations of release rate constants (fraction of the fission product inventory released per minute) as a function of temperature (Figure B.1). These graphical representations are in the form of smoothed curves through data points for each of the elements iodine, xenon, krypton, cesium, tellurium, silver, antimony, barium, strontium, zirconium, and ruthenium. Data for all experimental designs were used together with no distinction being made between clad or unclad fuel, the various environments into which the fission products were released, or the amount of material tested.

It is noted in the Technical Bases Report that the data show release rate constants for cesium and iodine to vary over two orders of magnitude at some temperatures, but that insufficient data on other fission products exist with which to estimate their associated uncertainties. It is also pointed out that the curves do not represent the rapidly occurring burst release of noble gases, which may result in loss of up to 4% of the initial inventory of these fission products in addition to release via other slower mechanisms (Wichner, Kress and Lorenz, 1981). Additional, but reportedly highly uncertain, release rate coefficients in a narrow temperature range are also provided for fuel, cladding, and structural materials in the Technical Bases Report.

In order to estimate releases of fission products from fuel during two accident sequences, chosen for illustrative purposes in the Technical Bases Report, the derived release rate constants were coupled with approximated temperature histories of the core, which were based on output from a thermal/hydraulic analysis code, MARCH (Meltdown Accident Response Characteristics) (Wooton and Avci, 1980). The MARCH code provides temperatures for a user-specified number of finite regions in a core as a function of time in prescribed accident scenarios. The temperature history approximation procedure in the Technical Bases Report necessitated the assumption that the temperature of a given

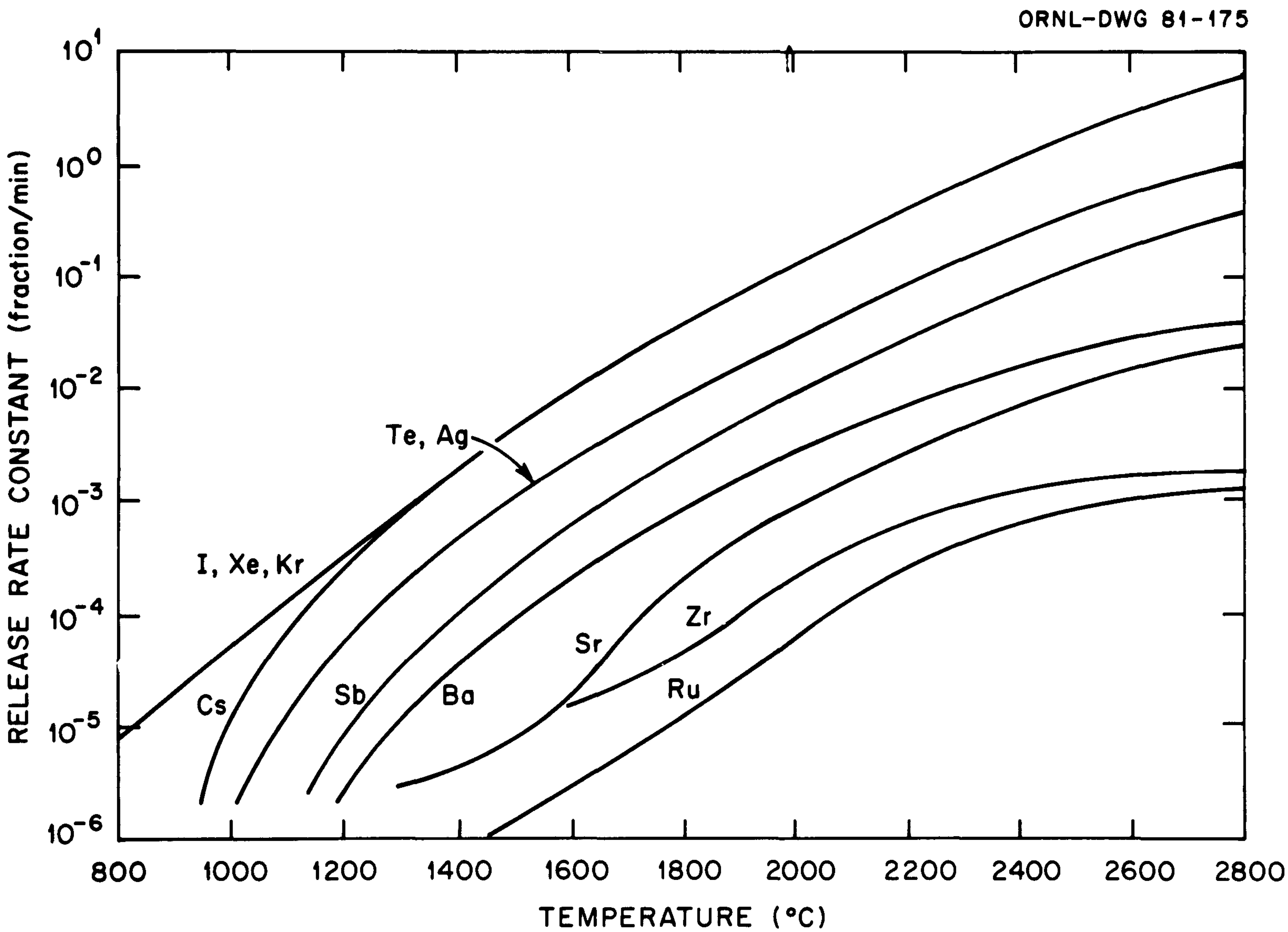

Fig. B.1. Fission product release rate constants from fuel-smoothed curves. (Courtesy of R. Wichner, Oak Ridge National Laboratory) finite core region increases linearly with time up to 2800°C, after which time the temperature remains at 2800°C. Results are presented for AB and S_2C sequences (terminology consistent with the RSS) in terms of the fission product release fraction as a function of time. The aerosol releases of cladding and structural materials are calculated based on the same temperature histories of the core.

All of the above release calculations are pertinent to releases prior to reactor vessel melt-through. Releases of radioactive materials after melt-through, when the molten core contacts the concrete basemat in the reactor cavity, were not determined. However, release of aerosols consisting of concrete decomposition products are estimated for a period of time after melt-through. These products are a result of the sparging of gas, produced by thermal decomposition, through the melt. A preliminary aerosol generation model, correlating the generation rate with gas velocity, melt surface area, and melt temperature, was implemented to estimate aerosol production via such processes. The gas velocity was approximated with the WECHSL code (Reimann and Murfin, 1978), which conducts a thermal analysis of the core/concrete interaction. Melt surface areas ranging from 29.2 m^2 to 45.4 m^2, along with temperatures of the melt generated by WECHSL, were used to calculate total aerosol production as a function of time for both accident sequences. Results for the AB sequence were presented as a function of time in the Technical Bases Report.

The major limitation in application of the above procedure for estimating fission product release from fuel and aerosol production, cited by the authors, involves the uncertainty in the derived release rate coefficients. The quantity of empirical data derived from consistent experimental designs is insufficient to evaluate the accuracy of these coefficients, particularly for compounds of relatively low volatility and for fuel, cladding, and structural materials. Because the smoothed release rate coefficient curves generated (Figure B.1) incorporate data from short-term experiments conducted in various environments, the inaccuracy involved in application to conditions in any given environment on a different time scale may be significant to varying degrees for different fission products. Also, because the zircaloy cladding is suspected to effect fission product release through interaction with the fuel and certain fission products (Wichner, Kress and Lorenz, 1981; USNRC, 1975), the inclusion of empirical data from experimental work with unclad fuel may introduce further error in the curves generated. Furthermore, some of the experiments involved the use of fuel pellets to which fission product simulants had been added before pellet formation (Albrecht, Matschoss and Wild, 1979a; Albrecht and Wild, 1981), leaving to conjecture whether actual fuel pellet conditions are adequately simulated. In addition to these limitations, extrapolation from small-scale experiments to in-reactor conditions necessitates that a certain degree of caution be exercised in interpretation of results, since unknown factors may alter the outcome on the larger scale.

Other pertinent experimental work conducted since the RSS involves measurements of leaching of radioactive materials from LWR fuel by water and a borate solution. Work of this nature is being conducted at Pacific Northwest Laboratories (PNL) in an investigation of leaching effects on storage of spent fuel (Katayama, 1979; Katayama, Bradley and Harvey, 1980), and at Oak Ridge National Laboratory (ORNL) in a study of leaching in a simulated LWR post-accident environment (Mitchell, Goode and Vaughen, 1981). To date, PNL results have been reported for leaching of unclad fuel fragments in deionized water at 25°C. Fractional leaching for ^{137}Cs, $^{239+240}Pu$, and ^{244}Cm, for three different degrees of burnup, as a function of time, are provided (Katayama, 1979). The data indicate that, for the fuel fragments tested, approximately 0.15% of the ^{137}Cs, 0.015% of the $^{239+240}Pu$, and $2.5 \times 10^{-4}\%$ of the ^{244}Cm have been leached after 10 days exposure to the water. Leaching of other elements are being tested (Katayama, Bradley and Harvey, 1980), but results have not been presented on a fractional release basis. Instead, leach rates in units of g of fuel fragments per cm^2 of surface area per day are given. These rates may be converted to fractional rates as a function of time using the approximate mass (15 g) and surface area (30 cm^2) of the fragments provided. Using this information, the estimated percentage leached after 10 days was obtained for several elements, and are presented in Table B.2. No results have been reported on higher temperature

Table B.2. Percent of Element Leached After 10 Days Immersion in Deionized Water at 25°C

Element	Approximate percent leached after 10 days (%)	
	PNL	ORNL[c]
I	-	1.5 - 3.0
Cs	0.15[a], 0.92[b]	0.09 - 0.25
Pu	0.15[a], 0.054[b]	-
Cm	2.5E-04[a], 0.19[b]	-
Ce	0.11[b]	0.006 - 0.04
Eu	0.11[b]	0.005 - 0.02
Ru	0.015[b]	0.009 - 0.05
Sr	0.10[b]	0.02 - 0.04
Sb	0.10[b]	0.15 - 0.06
U	-	0.004 - 0.01

[a]Taken from Katayama (1979).

[b]Calculated from leach rates given in Katayama, Bradley and Harvey (1980).

[c]Mitchell, Goode and Vaughen (1981).

experiments by PNL, although these are planned. The validity of extrapolation of these results to accident conditions is highly questionable due to the likely dependence of fractional leaching on geometry of the fuel fragments being leached, on the temperature and type of the leachant, and on cladding effects.

The ORNL study involved the use of a leachant chosen to approximate conditions in a post-accident reactor vessel. The experiments were conducted at 85°C and at 100°C, with three different fuel fragment sizes. The fuel fragments were characteristic of approximate mid-cycle burnup, but had not been exposed to the high temperature conditions generally encountered in an accident before fuel rod failure. Cumulative leach fractions as a function of time for the elements studied are given in Table B.2. The range of fractions given represent the range in particle sizes and temperatures used.

B.4.2 Theoretical Work

Theoretical work carried out since the RSS has involved the development of models to simulate fission product release from fuel on the basis of the mechanisms, or "first principles", involved. These mechanistic approaches incorporate varying degrees of detail in describing the processes involved, and thus, each relies on somewhat different information about each fission product for their application. Any mechanistic approach, however, will rely on temperature profiles in the core as a function of time since the mechanisms are so dependent on temperature. The two models briefly reviewed here necessitate specification of these temperature profiles as input, and thus, thermal/hydraulic analysis of an accident, such as is conducted in the MARCH code, is necessary.

A detailed model for the cladding-rupture releases of cesium and iodine as a function of temperature has been developed at ORNL, based on experiments conducted with fuel rods in steam over the temperature range 500° to 1200°C (Lorenz, Collins and Malinauskas, 1980). This model has been incorporated into a generalized source term model describing radionuclide releases from LWR fuel during degraded core accidents which is being developed at Battelle Columbus Laboratories (Baybutt, Nicolosi and Raghuram, 1981). The purpose of this modeling effort is to produce a model which is more descriptive of the physical processes to which the fuel is subjected in an accident than was the RSS release model, described in Section B.3 of this appendix, and thus, improve the predictive accuracies. To meet this objective, an attempt has been made to improve upon the models used to describe release mechanisms in the RSS, and to add other mechanisms not considered in the RSS. Thus, gap, melt, vaporization, and oxidation releases are considered, as was done in the RSS, as well as diffusion and leach releases. All of these mechanisms were qualitatively described in Section B.2. The model is termed "semi-mechanistic" by the Battelle authors due to its incorporation of parameters that may be determined from experimental data rather than further defining the parameters in mechanistic terms. This requires knowledge of such values as "effective diffusivities" of fission products for intact fuel and molten fuel under various accident conditions. The model is in a fairly early stage of development at present. Its usefulness will depend to a great extent on the availability of experimental data with which to evaluate the empirical parameters required, and the results of validation when model development is complete.

A mechanistic model is also under development at Argonne National Laboratory, the implementation of which is being carried out with the computer code, FASTGRASS (Rest, in press). The purpose of this model is also to characterize the mechanisms underlying fission product release from fuel to a greater extent than was carried out in the RSS, during premelt heat-up only. In contrast to the Battelle model, the FASTGRASS model provides a more detailed analysis of some of the mechanisms involved. For example, bubble nucleation, diffusion, and coalescence; channel formation; fuel microcracking; and grain boundary diffusion are handled explicitly in the code (Wichner, Kress and Lorenz, 1981), rather than utilizing empirically-derived effective diffusivities. At present, the FASTGRASS code has been applied to predict noble gas movement in the fuel, and some validation results have been obtained. Work is ongoing to allow application to other fission products, and the code is now operational for the cesium and iodine systems, although validation has not been performed for these latter elements. Thus, in terms of providing source term information for all fission products, this code is also in a fairly preliminary stage of development. Like the Battelle model, the usefulness of the Argonne model is dependent on the ability of the user to compile necessary input to the code on an element- and accident-dependent basis, and on validation results.

B.4.3 Chemistry

The importance of fission product chemistry, both prior to and during a degraded core accident, to the release from fuel was emphasized in the Technical Bases Report, as was the complexity of the subject. Oxygen availability and the presence of steam, determined by the accident scenario, and interactions between fission products and other materials present, such as fuel cladding, as well as with gases passing through the melt are phenomena which provide a controlling influence on the chemistry, and thus, degree of release. The empirical approach put forth in the Technical Bases Report does not necessitate explicit consideration of chemistry in calculation of release fractions if experimental designs can be assumed to simulate accident conditions. However, the theoretical approaches would require such consideration in order that the compound-dependent diffusion rates be obtained, as well as other parameters closely tied to chemical form. Thus, the ability to approximate accident conditions experimentally, and the ability to accurately simulate chemical phenomena will determine, to a great extent, which of the two approaches, empirical or theoretical, are desirable.

Iodine chemistry is an important consideration in estimating iodine releases from a reactor during an accident, due to the large differences in behavior of different chemical forms of this element (Campbell, Malinauskas and Stratton, 1981; Parker and Creek, 1981; Morewitz, 1981; Wichner, Kress and Lorenz, 1981). Although many believe that the aqueous chemistry of iodine in reactors specifies cesium-iodine as the form of iodine present after release from fuel in the presence of water, the actual form at the instant of release from fuel is still deemed uncertain by some (Wichner, Kress and Lorenz, 1981; Baybutt, 1981).

Before reactor vessel failure in an accident, when the core is heating to melting temperatures, H_2 produced by a steam/zirconium reaction creates a reducing environment will exist in most, if not all, accidents as a result of the interaction of the coolant with the cladding at high temperatures (Campbell, Malinauskas and Stratton, 1981; Parker and Creek, 1981). Therefore, many of the fission product species may exist in a reduced state rather than that predicted in air or at higher oxygen pressures. Under normal reactor conditions, the oxygen potential in the fuel may be in the range of -460 to -540 kJ/mol, at which value thermodynamics predict that tellurium, ruthenium, palladium, antimony, and molybdenum would exist in a metallic state for normal operating temperatures, whereas strontium, barium, yttrium, lanthanum, cerium, uranium, plutonium, zirconium, and niobium would exist as oxides (Cubicciotti et al., 1976). However, if the mechanism of competition for oxygen among elements present is altered by accident conditions such as the presence of steam, the chemistry of the elements may shift. After vessel failure, the presence of sparging gas during concrete decomposition may again create conditions in which chemistry of fission products are not easily predictable on the basis of thermodynamics at a set oxygen potential. For example, the gas passing through the melt may not be a passive carrier, but rather react with fission products in the melt to produce compounds of different volatilities than expected (Powers, 1982). This phenomena may affect releases prior to vessel failure also. Limited experimental evidence (Albrecht and Wild, 1981) indicate little difference in releases of most fission products released in steam and air environments from molten fuel pins. However, the sparging gases are not present in these experimental designs. After vessel failure, many of the more volatile fission products will likely be absent from the melt, thus eliminating the need to specify their chemistry. Unfortunately, time did not permit an in-depth review and analysis of the literature pertinent to fission product chemistry during reactor accidents in the present study. However, its importance with respect to potential releases from fuel and transport beyond that stage should be acknowledged.

B.5 PROPOSED APPROACH

The preceding discussions in this appendix serve to provide the technical bases upon which a state-of-the-art means of estimating fission product release from fuel during degraded core accidents may be proposed. The following approach was chosen on the basis of its practicality and simplicity relative to other possible approaches.

The proposed approach is believed to be an improvement over that provided by the RSS through incorporation of more recent experimental data, and is representative of state-of-the-art from an operational standpoint. It differs from the RSS approach in that fission product releases from fuel during a given reactor accident, except for leaching and sparging releases, are calculated as a function of core temperature only, throughout the pre-vessel failure phase of accidents, and release rate constants are strictly empirical in nature. The RSS, in contrast, considers gap releases separately, and does not consider other releases until the core starts to melt. Also, the total meltdown release in the RSS for each element group does not vary from accident to accident, while it may vary using the present methodology. The present approach is basically the approach put forth in the Technical Bases Report (Wichner, Kress and Lorenz, 1981), with a few minor alterations and additions in application. One of the major advantages of this empirically-based method for estimating fission product releases from fuel, is that fission product chemistry, which may be very complex during an accident, need not be specified in order to obtain release fractions as a function of time. This is not the case with more theoretical approaches, where one must either explicitly consider chemical thermodynamics at the temperatures involved, or must assume a chemical form for each fission product, in order to estimate diffusion out of the fuel. Also, failure to consider a mechanism of release is not a potential deficiency of the empirical approach, since all mechanisms were presumed operative during the reference tests. Furthermore, the empirical approach is simple in nature, thus allowing a clearer picture as to what the uncertainties in the estimations of fission product release might be attributed.

One of the major potential limitations of an empirical approach such as is proposed, is the error introduced by extrapolation from laboratory to actual in-reactor environments. Thus, the accuracy of this method will depend to a great extent on the similarity between the two environments. This and other limitations and uncertainties associated with the present application of this empirical approach will be discussed in Section B.7 of this appendix. The remainder of the present discussion will be devoted to a more detailed description of this approach, which was used to generate the estimated fission product release-from-fuel fractions presented in the next section.

The empirical approach proposed involves the use of fractional release rate constants for representative fission products of the chemical groups identified in the RSS, although such grouping is somewhat controversial. Separate values are, however, given for strontium and barium of the alkaline earth group, and ruthenium and molybdenum are split out from the noble metals groups as was done in the RSS. The actinides and rare earths are considered together, as in the RSS, with the release fraction for the uranium fuel used as representative of those groups of elements.

In general, release rate constants as a function of temperature (Figure B.1) are coupled with time-varying core temperatures generated by the MARCH thermal/hydraulic code for various accident sequences to estimate fractional releases as a function of time. These releases represent the gap, diffusion, and evaporation releases described in Section B.2 of this appendix. Leaching of degraded core materials and sparging releases are estimated separately, using empirical data when available. Oxidation releases which may occur during heat-up of fuel in an environment, in which oxygen may be more available, as a result of a steam explosion or after reactor vessel failure, are not considered explicitly in the approach described in the Technical Bases Report nor here. However, some of the empirical data making up the release rate curves in Figure B.1 were derived from heat-up experiments conducted in air and with unclad fuel pellets; conditions which might be considered representative of this oxidation release environment. Finally, aerosol generation due to heating of fuel, cladding, structural materials, and core/concrete interactions are estimated from empirically based models of generation rates as a function of temperature. Release fractions as a function of time are estimated for the fission products, or groups of fission products, for three distinct phases of an accident sequence: before core slumping, before reactor vessel failure, and after vessel failure. A description of the calculational procedures for each of these three phases follows.

B.5.1 Release before Core Slumping

As the core begins to heat up, following an event which reduces the heat removal efficiency of the coolant, the fission product release processes described in Section B.2 are initiated at different times for various locations within the core, due to differential heat-up of the core. Thus, it is necessary to consider distinct regions of the core in calculating releases, to account for the effect of the temperature differential throughout the core. The importance of this core regionalization may be inferred from examination of the curves in Figure B.1, where release rate constants are shown to increase exponentially with increasing temperature. Thus, if a situation exists where a significant portion of the core remains relatively cool while the remaining portions are at much higher temperatures, the average temperature of the core may indicate a much lower overall release than would separate consideration of release from the high and low temperature regions.

Core regionalization with respect to temperature is accomplished by the MARCH code, mentioned in Section B.4 above (Wooton and Avci, 1980). Output from the code, prior to core slumping, provides temperatures of up to 500 regions of the core (10 radial and 50 axial) at user-specified time intervals following accident initiation.

By coupling temperature values for each region with the release rate constants given in Figure B.1, empirically based fission product releases as a function of time may be calculated by summing over all regions at each time step. The Technical Bases Report states that use of this method may neglect the 3 to 4% gap release of the noble gases. Therefore, such a percentage must be added to the results for these fission products. It must be cautioned that this procedure may not adequately represent effects of molten materials other than fuel or fission products on release if experimental designs were not comparable to full-scale conditions.

In order to facilitate computations of this type, fractional release rate constants, k(T), in fraction/min, may be approximated by equations of the form

$$k(T) = A\,e^{BT} , \tag{B.1}$$

where T is temperature in °C, as was done in the Technical Bases Report (Kress and Wichner, 1981). Three sets of values for A and B were used in the present methodology, for each of the curves in Figure B.1 – for temperatures greater than 800°C and less than or equal to 1400°C; for temperatures greater than 1400°C and less than or equal to 2200°C; and for temperatures greater than 2200°C. Similar sets of values of A and B were approximated for the fission product molybdenum, such that the curve passed through two experimentally derived points for this element, shown in Figure 4.2 of the Technical Bases Report (Wichner, Kress and Lorenz, 1981). In the course of preparation of the present document, it was brought to the author's attention that some of the release rate coefficient curves and data provided in the Technical Bases Report were no longer considered representative of the best available information on some materials, following a reevaluation of published experimental data (Lorenz, 1982). Thus, some alterations to the derivation described above for release rate coefficients as a function of temperature were made for the fission product zirconium, fuel cladding, and structural materials. Other alterations are indicated for similar reasons for strontium, barium, and ruthenium, as the curves in Fig. B.1 may be too high, but it was suggested that such alterations may not be justified at this time until confirmatory data are made available (Lorenz, 1982). The alterations made were as follows. For the fission product zirconium, it is believed that the curve in Fig. B.1 is too high by about a factor of 100, and that a reduction in the release rate coefficients by a factor of 100 is justified. The same slope of the zirconium curve in Fig. B.1

was assumed, however. For fuel, the release coefficient values given in the Technical Bases Report are believed to be too low, by one to two orders of magnitude. Therefore, the two values given in the Technical Bases report for fuel were increased by a factor of 50, and a single value for A and B was used over the entire temperature range. For cladding, separate consideration of the zirconium and tin components was given. The cladding release coefficients for the zirconium component were assumed to be identical to those for the fission product zirconium, and coefficients for the tin component were assumed to be identical to those for the fission product antimony. This latter assumption was based on experimental results (Albrecht and Wild, 1981), indicating a similarity between release of tin and antimony. Finally, alterations were made in the structural materials' release coefficients, as the Technical Bases Report values are believed to be too low. The iron component of structural materials is believed to be more volatile than indicated by the values given. Some experimental data (Albrecht and Wild, 1981) indicate that the release coefficients of the iron component may be approximated by the zirconium curve from Fig. B.1 (not the altered curve for zirconium discussed above). Therefore, the A and B values derived for the zirconium curve in Fig. B.1 were used for iron. The values of A and B derived for equation B.1 are given in Table B.3.

In order to estimate total releases from the release rate constants, the effective temperature of each region throughout each time step can be estimated as follows. Temperature increases may be assumed to be linear with time during each time interval, such that the average temperature between two successive time steps could be used as the effective temperature during the entire interval. In this way, release rate constants associated with the effective temperatures are simply multiplied by the size of the time step to generate total releases for each core region. Fractional release may be calculated for the entire core at each time step in terms of the fraction of the original fission product inventory at accident initiation. Thus, the amount lost at each time step must be subtracted before the succeeding fraction is calculated. The total fractional release prior to slumping is computed by summing these fractional releases at each time step. As was mentioned earlier,

Table B.3. Values used for the constants A and B in the approximation of the release rate coefficients, $k(T)=Ae^{BT}$

Fission product group	800°C < T ≦ 1400°C		1400°C < T ≦ 2200°C		T > 2200°C	
	A	B	A	B	A	B
I, Xe, Kr	7.02E-09[a]	0.00886	2.02E-07	0.00667	1.74E-05	0.00460
Cs	7.53E-12	0.0142	2.02E-07	0.00667	1.74E-05	0.00460
Te, Ag	3.88E-12	0.0135	9.39E-08	0.00630	1.18E-05	0.00411
Sb	1.90E-12	0.0128	5.88E-09	0.00708	2.56E-06	0.00426
Ba	7.50E-14	0.0144	8.26E-09	0.00631	1.38E-05	0.00290
Mo	5.01E-12	0.0115	5.93E-08	0.00523	3.70E-05	0.00200
Sr	2.74E-08	0.00360	2.78E-11	0.00853	9.00E-07	0.00370
Zr[b]	6.64E-12	0.00631	6.64E-12	0.00631	1.48E-07	0.00177
Ru	1.36E-11	0.00768	1.36E-11	0.00768	1.40E-06	0.00248
Fuel[b]	5.00E-13	0.00768	5.00E-13	0.00768	5.00E-13	0.00768
Cladding[b] (Zr)	6.64E-12	0.00631	6.64E-12	0.00631	1.48E-07	0.00177
(Sn)	1.90E-12	0.0128	5.88E-09	0.00708	2.56E-06	0.00426
Structure[b]	6.64E-10	0.00631	6.64E-10	0.00631	1.48E-05	0.00177

[a] 7.02E-09 denoted 7.02×10^{-9}.

[b] The values for A and B for these elements were altered from the Technical Bases Report. See discussion in text.

a 4% loss of noble gases should be added to the fractional loss determined for these fission products during the first time step.

Total aerosol releases before core slumping may be estimated by coupling release fractions calculated for fission products, fuel, cladding, and structural materials with the amount of these materials present. Control rod materials should also be considered, as they are a potentially large source of aerosols. For reactors which use the Ag-In-Cd controls rods, the amount of Ag-In-Cd may be on the order of 2300 kg (Albrecht and Wild, 1981). Thus, if the release rate coefficient for the fission product silver were used for the control rod silver (the major constitutent of the control rod), this material would be the largest contribution to the aerosol mass released before pressure vessel failure (Albrecht and Wild, 1981). However, some recent experimental work done at ORNL suggests that a vapor suppression process, which occurs during melting of the materials in core, may severely reduce the volatility of silver control rods and other materials to varying degrees (Parker, 1982). Because of these large uncertainties surrounding the release rate coefficient for the control rod silver, it was not included in the aerosol mass calculations, although its potential importance should not be overlooked.

Initial inventories of materials other than fission products, appropriate for a typical pressurized water reactor (PWR), are given in the Technical Bases Report as:

UO_2	10^5 kg
Zr (clad)	2.0×10^4 kg
Sn (clad)	300 kg
Fe (structure).	2.5×10^3 kg (in core)
	2.5×10^4 kg (core + bottom structure)

Fission product inventories, present when the accident is assumed to begin, may be estimated using the ORIGEN computer code (Bell, 1973), for specified reactor power ratings and degrees of burnup. For an 800 MW PWR, at midcycle, estimated fission product inventories are:

Xe	260 kg
Kr	20 kg
I	12 kg
Cs	140 kg
Te	23 kg
Ag	3 kg
Sb	2 kg
Mo	160 kg
Ba	70 kg
Sr	54 kg
Zr	190 kg
Ru	110 kg

Leaching of fission products from fuel in which cladding has ruptured is a potential mechanism of release from fuel, as was discussed in Section B.2. Experimental data for in-reactor conditions are greatly lacking in this area. Katayama's data (Katayama, 1979: Katayama, Bradley and Harvey, 1980) may be extrapolated for the purpose of estimating such releases in lieu of more appropriate data.

Table B.2 indicates leaching may be an important contributor to fission product release for less volatile fission products. It seems that leaching is more likely to be important in an accident in which total meltdown does not occur, since other mechanisms of release in the more severe accidents will likely dominate the release.

B.5.2 Releases before Reactor Vessel Failure

After the core slumps into the lower plenum of the reactor vessel, any water present may quench or partially quench the molten core, which at this point is a mixture of molten core materials (UO_2, control rods, cladding, and structural materials). However, as this water boils off, the core materials again approach melting temperatures, resulting in the release of additional fission products prior to vessel failure. The MARCH code predicts one overall temperature of the molten core during this time interval, as well as the length of the time interval, in its subroutine HEAD (Wooton and Avci, 1980). This information may be used with the previously derived approximation equations for release rate constants (Section B.5.1) to calculate releases after slumping but before vessel melt-through. In doing so, however, an assumptions must be made that the surface to volume ratio difference between that specified by the experimental design and that of the core which has slumped into the vessel head will not greatly alter the releases. This latter assumption is likely to introduce significant errors in estimated releases, since the molten core in the vessel head will have a much smaller surface to volume ratio than that of a segment of fuel with mass less than one kilogram, such as has been used in experiments reported to date. Other factors related to geometry may also affect release from fuel, and thus a simple correction factor to scale up the experimental results is not intuitively justifiable. However, preliminary experimental evidence indicates that such a correction factor may greatly improve predicted releases during this phase of the accident (Parker, 1982). In light of the preliminary nature of this evidence, a correction factor was not used in the calculational results presented in this appendix, but rather an example calculation indicating the potential significance of such a correction to estimated releases for one accident sequence is provided (see Section B.6.3).

The total fraction of the initial fission product inventory lost before vessel failure may be calculated by summing these latter releases and the before-slumping releases discussed in Section B.5.1. Again, potential aerosol releases may be calculated from the release fractions determined and the amount of material present.

B.5.3 Releases after Vessel Failure

After the reactor vessel has failed, the molten core debris will fall into the reactor cavity, possibly boiling off any water present, and causing thermal decomposition of the concrete basemat, as discussed in Section B.2. Releases of fission products and non-radioactive aerosols may occur as the molten material drops onto the concrete, and after core/concrete interactions begin. Releases during the time of dropping are not considered here, because the time period of this process is quite short. However, it should be noted that the streaming melt may evolve lots of dust during transit through open air, and that it is possible that the melt would be scattered violently about the cavity under high pressure conditions (Powers, 1982). During the core/concrete interaction stage, releases as a result of gas sparging may occur to a significant extent. The sparging process before vessel failure is enhanced by the increased melt surface area exposed to gases bubbling through the molten mixture. The MARCH code provides temperatures of the oxide and metallic phases of the melt during this time period, as well as gas flows through the melt. This information is derived from the INTER code, which relies on qualitative and quantitative descriptions of convective stirring, substrate decomposition, admixture of substrate decomposition products to the melt, core and structural material oxidation and reduction reactions, interactions with water, radiative heat loss, and heat transfer coefficient variations to provide such information (Niemczyk et al., 1981).

The sparging component of the fission product release during core/concrete interactions may be best estimated at present using values generated in the RSS (Haaland, 1975). These theoretical values assume an equilibrium distribution of fission products between the melt and the gas bubbles passing through the melt, that 100% of the gas passes through the melt and temperatures of 3100°K. They are therefore considered overestimates, assuming that the correct chemical form has been assumed for the element for which values are given. The values are in terms of the fractional loss for fission product compounds, and no time-dependence is associated with them except a crude half-life approximation based on gas sparging rate decreases over time, which is highly uncertain (Haaland, 1975). It is suggested here, then, that fractional losses be considered to occur with a half-life that is consistent with the gas flow rates generated by the thermal/hydraulic code being used. Total fractional loss values for fission products via sparging are given in Table B.4 for the chemical forms indicated.

Estimation of aerosol production as a result of gas sparging is in a very preliminary stage of development. An equation is available,

Table B.4. Vaporization Release Fractions

Element	Fraction released via vaporization (range)[a]	Value used in present calculations
I	>0.999	1.0
Cs	>0.999	1.0
Te, Ag	>0.999	1.0
Sb	>0.999	1.0
Mo[b]	<0.001 - 0.15	0.15
Ba[c]	0.01 - 0.045	0.045
Sr[c]	<0.001 - 0.001	0.001
Zr	<0.001	0.001
Ru[d]	<0.01 - 0.02	0.02
U	0.001 - 0.006	0.006
Sn	0.66 - >0.999	1.0

[a] Considering from 20% to 100% of sparge gas passes through melt (RSS, 1975).

[b] Range is for the metal and dioxide; value used is for the more volatile dioxide.

[c] The monoxide.

[d] The metal.

which was derived from a minimum amount of experimental data, providing aerosol concentration in the sparging gas as a function of gas velocity and melt temperature (Kress and Wichner, 1981). Use of such an empirical fit to data implies that all mechanisms contributing to aerosol formation are represented during this phase of an accident, including sparging and bubble bursting (Powers, 1982). This equation, presented in the Technical Bases Report, is of the form

$$[A] = \exp(-19000/T)(24V + 3.3)\ 10000, \tag{B.2}$$

where [A] is the aerosol concentration in the gas (g/m^3), T is the absolute temperature of the melt (°K), and V is the superficial gas velocity (m/s). Superficial gas velocity may be calculated from the gas flow rates (g/s) for CO_2, CO, H_2O, and H_2 generated by the INTER subroutine called by MARCH, or by the WECHSL or CORCON codes, which are believed to be improvements over INTER (Murfin, 1980). The calculation involves converting gas flow rates in grams per second to volumetric flow rates for each gas, assuming ideal gases, and dividing these values by an estimated surface area of the melt. After calculating aerosol concentration in the sparge gas, total aerosol release (kg) to the containment during each time step may be found by multiplying this concentration by the volumetric flow rates and the length of the time step. Aerosol production due to evaporation of fuel, cladding, and structural materials is not explicitly calculated after vessel failure due to uncertainties in appropriate release rate constants for these materials, although it has been found that the aerosols generated during the process are composed mostly of concrete decomposition products (Baybutt, 1981).

It should be emphasized that the above procedure is preliminary in nature, and that neither the INTER, WECHSL, or CORCON codes, simulating temperatures and gas production rates during core/concrete interactions, nor Equation (B.2), have been verified as to their accuracy under various conditions. From Equation (B.2), and the discussion immediately following it, it is evident that an order of magnitude change in gas velocity will result in a change in aerosol generation rate of almost two orders of magnitude. The gas velocity is not only temperature dependent, but also depends on the model used to estimate its magnitude. As pointed out in Figure 5.3 of Murfin (1980), gas velocities estimated by INTER may generally exceed those estimated by WECHSL, sometimes by more than a factor of two. Thus, errors in temperature estimation and other modeling assumptions in INTER or WECHSL may greatly affect the aerosol generation calculation, and caution must be exercised in interpreting the values generated by this procedure.

B.6 ESTIMATED RELEASE FRACTIONS

Fractional releases of fission products from fuel were calculated using the procedures described in Section B.5 for nine accident sequences. Because these calculations were conducted under great time constraints, some of the values used for input, such as the MARCH code output, may not represent the most updated versions of the information available, but rather represent the most complete and readily available set of values that allows consistency in calculations throughout this report. The calculated values are specific to the reactors and conditions for which the MARCH code input was specified, and thus serve better to illustrate application of the proposed approach presented in Section B.5 than to be considered definitive state-of-the-art source term values. Uncertainties associated with these values are discussed in Section B.7.

The first seven accident sequences, in RSS terminology, are, for PWR's, the V, S_2D, AD, and TMLB´ sequences; and for boiling water reactors (BWR's), the TQUV, TW, and TC sequences. In addition, fractional releases were estimated for accidents in which 50% of the core exceeds the minimum and average temperatures at which cladding fails, or at 750°C and 900°C, respectively (Wichner, Kress and Lorenz, 1981), in an AD-type accident (termed an AD-"gap release" sequence). Fractional releases were also estimated for an AD-type accident when 50% of the core regions reach the melting temperature of fuel mixture, assumed to be 2275°C in MARCH, and this sequence was termed an AD-1/2 accident. This latter sequence may also be assumed to be representative of releases-from-fuel in a TMI-2-like accident sequence. Below is a brief description of the above accident sequences in terms of the temporal aspects affecting release from fuel. This is followed by a description of assumptions used to generate the calculational results presented at the end of this section.

B.6.1 Accident Sequence Descriptions

The V-type accident sequence, representing a check valve failure for the low-pressure injection system in a PWR, will result in a rapid uncovery of the core and subsequent melting, which may begin around 30 minutes after valve failure occurs according to MARCH code runs available. Slumping of the core into the pressure vessel head will result in partial quenching of the core, such that vessel head failure occurs approximately 40 minutes later. No water would be expected to be in the reactor cavity, such that core/concrete interactions begin immediately upon pressure vessel failure.

The S_2D sequence, involving a small-break loss of coolant (LOCA) and failure of the ECCS, results in a fairly rapid core uncovery time following accident initiation. Vessel failure may occur rapidly following slumping, since the system is at high pressure, but core/ concrete interactions may not begin for several hours due to the water in the reactor cavity.

The AD-type sequence, where a large LOCA occurs in conjunction with ECCS failure, will again result in rapid core uncovery. The time of vessel failure following slumping will occur in a time interval similar to that for a V-type accident due to relatively low pressures prevailing in this sequence. Core/concrete interaction may occur immediately following vessel failure if water is not present in the reactor cavity.

The TMLB´ sequence, representing a transient event with loss of electric power, auxiliary feedwater, and power conversion systems, may result in core uncovery approximately 3 hours post-initiation of the accident. Vessel failure may occur fairly rapidly after core slumping due to high pressures maintained in the vessel. Water in the reactor cavity from the accumulator discharge following vessel failure delays core/concrete interactions.

The BWR sequences listed above all represent transient events. In the TQUV and TC accidents, core slumping occurs fairly rapidly after uncovery (within approximately 40 minutes), whereas core slumping in the TW sequence is somewhat slower (approximately 2.5 hours). Vessel failure in all three sequences occurs 20 to 30 minutes after slumping according to MARCH. With no water in the drywell following vessel failure, core/concrete interactions immediately succeed vessel melt-through.

B.6.2 Assumptions Made for Calculations

Release fractions as a function of time were calculated for the three phases of each accident that were specified (Section B.5). In these cases, MARCH code output was made available by Battelle Memorial Institute in Columbus, Ohio. The output had been generated in 1978 for certain accident sequences with the Surry PWR and Peach Bottom BWR reactors, and thus, more recent modifications to the MARCH code were not reflected in the temperatures reported (See Section D.3). Output for the AD sequence was not available in this set of MARCH code runs; therefore, output for the V sequence, which is believed to result in a similar temperature history of the core before and after vessel failure, was used. Core slumping in these MARCH runs was assumed to occur when 75% of the 120 regions specified in these runs had reached the temperature at which fuel rods are assumed to melt, estimated to be 2275°C (4130°F) for these predictions.

Average temperatures of core debris after slumping but before vessel failure were assumed to be 2300°C for the PWR sequences considered, although these values were not obtained directly from MARCH output. The HEAD subroutine of MARCH predicts such temperatures, but the initial data obtained from Battelle did not include such output for these sequences. Due to time constraints associated with this project, it was not possible to track down these values. Therefore, the assumed temperature was chosen to approximate temperatures shortly after the core drops into the reactor cavity for all of these sequences, which were available from the subroutine INTER output. For the BWR sequences, HEAD output was available, but was based on the assumption of 100% clad reaction with steam for this particular set of MARCH runs, resulting in core debris temperatures after slumping around 3300°C. Because this assumption was believed to be unreasonable, a lower temperature of 2000°C for core debris was assumed for the after-slumping phase of the BWR accident sequences. This value was obtained from a user of the MARCH code at ORNL. The time of residence of core debris at these after-slumping temperatures, before vessel failure, was assumed to be consistent with those times being used in the RSS rebaselining study (Cybulskis, 1981) for PWR's. For the BWR sequences, times of residence were taken directly from the MARCH output made available by Battelle. It must be recognized and emphasized that, because of these necessary assumptions, the releases calculated after slumping and before vessel failure, which rely on such assumptions, reflect any errors in these assumptions. In addition, the releases after core slumping were calculated for all sequences without a surface area-to-volume ratio correction (see Section B.5.2), thus casting further doubt on their accuracy. To illustrate the potential effect of making such a correcton, a calculation for a V-type sequence was made using a correction factor of 40 (Table B.15), which was estimated to be appropriate for scaling SASCHA data to a full-scale core for illustrative purposes (Parker, 1982). Thus, by dividing the releases after slumping, which were used to generate results in Table B.5, by 40, the releases present in Table B.15 are estimated.

After vessel failure, temperature of the two phases of the melt were provided in the MARCH output every thirty minutes. Because the temperatures did not change rapidly for the sequences considered after an initial drop in temperature, aerosol release calculations based on the average temperature of the two phases were carried out every 2 hours after an initial calculation 30 minutes post-vessel failure. It should be noted that these MARCH code runs discontinued temperature calculations approximately 8 hours after vessel failure, at which time the melt had generally cooled to temperatures at which little additional fission product or aerosol release would be expected.

Aerosol releases (Table B.16) were estimated according to the procedure outlined in Section B.5.3, assuming surface areas of the debris

Table B.5. Release fractions calculated for a V-Type accident (PWR)

Element	Fraction lost before slumping	Fraction lost before vessel failure	Total
Xe, Kr	1.0	1.0	1.0
I	1.0	1.0	1.0
Cs	1.0	1.0	1.0
Te, Ag	0.74	1.0	1.0
Sb	0.31	1.0	1.0
Mo	0.04	0.18	0.30
Ba	0.10	0.49	0.51
Sr	0.04	0.21	0.21
Zr	1.0E-04	4.4E-04	0.001
Ru	0.004	0.02	0.04
U	1.8E-04	0.001	0.007

of 34 m^2 for PWR's, and 32 m^2 for BWR's. The radii of these areas (3.3 m and 3.2 m, respectively) correspond to values used in the subroutine INTER of the MARCH code runs that were available. The surface area of the debris may affect the predicted aerosol generation according to the calculational procedure described in Section B.5.3. In addition, the surface area of the debris, being an input parameter to INTER, will affect the temperature of the debris calculated in that subroutine, further affecting the estimated aerosol generation. More important than the surface area, however, is the effect of gas velocity on the calculated aerosol generation rate as was discussed in Section B.5.3. Therefore, caution must be exercised in interpreting the aerosol values generated here, realizing that the values are specific to the Surry and Peach Bottom reactors from the standpoint of the debris surface area, and that the values are very sensitive to the INTER modeling assumptions. The inventory of structural materials in the melt was assumed to include that in the core as well as in the bottom structure. Although this may overestimate the amount of structural material which is mixed into the molten core and thus in the aerosol during some accidents, the overall contribution of these materials to the aerosol concentration is insignificant using the release rates generated by this estimation procedure.

Sparging releases were assumed to occur instantaneously, due to the large uncertainty in the half-life approximation currently available. Time constraints associated with the present study did not allow further development with regard to the time-dependence of the sparging process.

Leaching releases were calculated for the AD-"gap release" and AD-1/2 sequences only. The leaching values given in Tables B.12 through B.14 represent the fractional losses 10 days after the contact of water with failed fuel rods initially begins.

B.6.3 Calculational Results

Following are results generated using the methods described in Section B.5 for estimating fractional releases of fission products from fuel for nine representative degraded core accident sequences. Tables B.5 through B.11 summarize results for complete meltdown accidents. Tables B.12 through B.14 provide values for degraded core accidents in

Table B.6. Release fractions calculated for a S_2D-type accident (PWR)

Element	Fraction lost before slumping	Fraction lost before vessel failure	Total
Xe, Kr	0.99	1.0	1.0
I	0.99	1.0	1.0
Cs	0.99	1.0	1.0
Te, Ag	0.65	1.0	1.0
Sb	0.28	0.65	1.0
Mo	0.02	0.06	0.22
Ba	0.07	0.18	0.22
Sr	0.03	0.08	0.08
Zr	6.0E-05	1.5E-04	0.001
Ru	0.003	0.008	0.03
U	1.7E-04	4.3E-04	0.006

Table B.7. Release fractions calculated for a TMLB′-type accident (PWR)

Element	Fraction lost before slumping	Fraction lost before vessel failure	Total
Xe, Kr	1.0	1.0	1.0
I	1.0	1.0	1.0
Cs	1.0	1.0	1.0
Te, Ag	1.0	1.0	1.0
Sb	0.80	0.89	1.0
Mo	0.09	0.12	0.25
Ba	0.26	0.34	0.37
Sr	0.13	0.17	0.17
Zr	2.0E-04	2.8E-04	0.001
Ru	0.01	0.01	0.03
U	0.002	0.002	0.008

Table B.8. Release fractions calculated for an AD-type accident (PWR)

Element	Fraction lost before slumping	Fraction lost before vessel failure	Total
Xe, Kr	1.0	1.0	1.0
I	1.0	1.0	1.0
Cs	1.0	1.0	1.0
Te, Ag	0.74	1.0	1.0
Sb	0.31	1.0	1.0
Mo	0.04	0.18	0.30
Ba	0.10	0.49	0.51
Sr	0.04	0.21	0.21
Zr	1.0E-04	4.4E-04	0.001
Ru	0.004	0.02	0.04
U	1.8E-04	0.001	0.007

Table B.9. Release fractions calculated for a TQUV-type accident (BWR)

Element	Fraction lost before slumping	Fraction lost before vessel failure	Total
Xe, Kr	0.99	1.0	1.0
I	0.98	1.0	1.0
Cs	0.98	1.0	1.0
Te, Ag	0.57	0.78	1.0
Sb	0.23	0.35	1.0
Mo	0.03	0.07	0.21
Ba	0.06	0.10	0.14
Sr	0.03	0.04	0.04
Zr	5.0E-05	9.0E-05	0.001
Ru	0.002	0.003	0.02
U	1.2E-04	1.6E-04	0.006

Table B.10. Release fractions calculated for a TC-type accident (BWR)

Element	Fraction lost before slumping	Fraction lost before vessel failure	Total
Xe, Kr	1.0	1.0	1.0
I	1.0	1.0	1.0
Cs	1.0	1.0	1.0
Te, Ag	0.80	0.90	1.0
Sb	0.38	0.47	1.0
Mo	0.04	0.08	0.22
Ba	0.11	0.15	0.19
Sr	0.04	0.05	0.05
Zr	9.0E-05	1.3E-04	0.001
Ru	0.004	0.005	0.03
U	2.6E-04	3.0E-04	0.006

Table B.11. Release fractions calculated for a TW-type accident (BWR)

Element	Fraction lost before slumping	Fraction lost before vessel failure	Total
Xe, Kr	1.0	1.0	1.0
I	1.0	1.0	1.0
Cs	1.0	1.0	1.0
Te, Ag	1.0	1.0	1.0
Sb	0.91	0.93	1.0
Mo	0.20	0.25	0.37
Ba	0.46	0.50	0.52
Sr	0.21	0.23	0.23
Zr	5.0E-04	5.6E-04	0.002
Ru	0.02	0.02	0.04
U	0.001	0.001	0.007

Table B.12. Release fractions calculated for an AD-"gap release"-type accident (PWR), 50% of core above 750°C[a]

Element	Fraction lost (before slumping)	
	Volatilization	Leaching
Xe, Kr	0.04	0.96[c]
I	3.E-03	0.03
Cs	3.E-03	3.E-03
Te, Ag	6.E-04	6.E-03[d]
Sb	2.E-04	6.E-03
Mo	5.E-05	5.E-04[e]
Ba	5.E-05	4.E-04[f]
Sr	1.E-05	4.E-04
Zr	5.E-08	1.E-04[g]
Ru	1.E-06	5.E-04
U	4.E-08	1.E-04

[a]50% of the core above minimum clad burst temperature.

[b]Using upper value of 10-day cumulative leach fractions from Mitchell, Goode and Vaughen (1981).

[c]Assumed remainder is released.

[d]Assumed same leach constant as Sb.

[e]Assumed same leach constant as Ru.

[f]Assumed same leach constant as Sr.

[g]Assumed same leach constant as U.

Table B.13. Release fractions calculated for an AD-"gap release"-type accident (PWR), 50% of core above 900°C[a]

Element	Fraction lost (before slumping)	
	Volatilization	Leaching[b]
Xe, Kr	0.14	0.86[c]
I	0.13	0.03
Cs	0.13	2.E-03
Te, Ag	0.03	6.E-03[d]
Sb	6.E-03	6.E-03
Mo	1.E-03	5.E-04[e]
Ba	2.E-03	4.E-04[f]
Sr	7.E-04	4.E-04
Zr	2.E-06	1.E-04[g]
Ru	7.E-05	5.E-04
U	3.E-06	1.E-04

[a] 50% of the core above mid-range clad burst temperature.

[b] Using upper value of 10-day cumulative leach fractions from Mitchell, Goode and Vaughen (1981).

[c] Assumed remainder is released.

[d] Assumed same leach constant as Sb.

[e] Assumed same leach constant as Ru.

[f] Assumed same leach constant as Sr.

[g] Assumed same leach constant as U.

Table B.14. Release fractions calculated for an AD-½ - or TMI-2 - like accidents (PWR)[a]

Element	Fraction lost (before slumping)	
	Volatilization	Leaching[b]
Xe, Kr	0.92	0.08[b]
I	0.91	4.E-03
Cs	0.91	4.E-04
Te, Ag	0.42	4.E-03[d]
Sb	0.15	5.E-03
Mo	0.02	5.E-04[e]
Ba	0.04	4.E-04[f]
Sr	0.02	4.E-04
Zr	4.E-05	1.E-04[g]
Ru	2.E-03	5.E-04
U	7.E-05	1.E-04

[a]50% of the core molten.

[b]Using upper value of 10-day cumulative leach fractions from Mitchell, Goode and Vaughen (1981).

[c]Assumed remainder is released.

[d]Assumed same leach constant as Sb.

[e]Assumed same leach constant as Ru.

[f]Assumed same leach constant as Sr.

[g]Assumed same leach constant as U.

Table B.15. Release fractions calculated for a V-type accident (PWR) with surface area-to-volume ratio correction[a]

Element	Fraction lost before slumping	Fraction lost before vessel failure	Total
Xe, Kr	1.0	1.0	1.0
I	1.0	1.0	1.0
Cs	1.0	1.0	1.0
Te, Ag	0.74	0.75	1.0
Sb	0.31	0.33	1.0
Mo	0.04	0.04	0.18
Ba	0.10	0.11	0.15
Sr	0.04	0.04	0.04
Zr	1.0E-04	1.1E-04	0.001
Ru	0.004	0.004	0.02
U	1.8E-04	2.0E-04	0.006

[a]Divided fractional releases which occurred after core slumping but before vessel failure by 40.

Table B.16. Total aerosols generated during nine accident sequences

Accident sequence	Aerosols before vessel failure (kg)[a]	Aerosols after vessel failure (kg)
V	750	6600[c]
S_2D	490	400[c]
TMLB′	670	400[c]
AD	750	6600[c]
TQUV	320	9100[d]
TC	390	9200[d]
TW	790	23000[d]
AD-"gap release"	<1[e], 20[f]	NA
AD-½	180	NA

[a]Includes Cs and Te as aerosols; excludes control rod materials.

[b]Numbers should be regarded as highly uncertain due to preliminary nature of the equation used and sensitivity of values to INTER results (Section B.6.2).

[c]Values for PWR sequences generated by the MARCH code assumption of 50% clad reaction with water in containment building.

[d]BWR values generated by the MARCH code assumption of 100% clad reaction with water in containment building, and thus temperatures are much higher.

[e]For 50% of the core above 750°C.

[f]For 50% of the core above 900°C.

which 50% of the core reaches a temperature of 750°C, representing the minimum temperature of cladding failure (Table B.12), of 900°C, representing a mid-range temperature for cladding failure (Table B.13), and of 2275°C, representing the temperature of incipient melting of the fuel rods (Table B.14), before the accident was terminated. All of the rods were assumed to experience clad rupture with respect to the noble gas release (i.e., 4%) in these partial meltdown sequences. Table B.15 shows results of calculations for a V-type accident, where a correction was made on release estimates for the dramatic reduction in surface area-to-volume ratio of the core debris after core slumping but before vessel failure. A comparison of Table B.5 with Table B.15 indicates a significant effect of such a correction on total estimated releases of Ba and Sr only. Finally, total aerosol generation for each accident is given in Table B.16 for various stages in an accident sequence. Release fractions as a function of time are given in the Addendum B1 to this appendix.

A comparison between the total fractional releases of fission products given in Tables B.5 through B.11 and those provided in the RSS may be made by combining the gap, meltdown, and vaporization release components in the RSS (Table B.2) into one value. The results of such a comparison are presented in Table B.17. It is apparent from this table that the total releases calculated via the present approach and the RSS methodology are in agreement for those fission products generally recognized as more volatile; i.e., the noble gases, iodine, cesium, and tellurium. However, the new approach indicated a potentially larger release of the alkaline earths, strontium and barium, and of molybdenum for the accident sequences assumed. Furthermore, if a comparison is made between the release fractions corresponding to before-vessel failure in the RSS (i.e., gap and meltdown releases), and the fractions for this time interval generated by the new approach, it is evident that the new approach suggests a much greater release fraction for the tellurium and antimony fission products (0.35 - 1.0) and for strontium and barium during this period than the RSS value (0.15). These differences are mainly a result of the consideration of recent pertinent experimental data since the RSS, which was summarized in the Technical Bases Report

Table B.17. Comparison of total fission product releases from RSS and proposed approach

Element	RSS value[a] (fractional release)	Range of values for nine accident sequences from new approach (fractional release)	
		Core melt	Degraded core
Xe, Kr	1.0	1.0	1.0
I, Br	1.0	1.0	0.03 - 0.91
Cs, Rb	1.0	1.0	6.E-03 - 0.91
Te, Se, Sb	1.0	1.0	6.E-03 - 0.42
Sr, Ba	0.11	0.04-0.52	5.E-04 - 0.04
Ru, Rh, Pd, Mo, Tc	0.08	0.02(Ru) - 0.37(Mo)	5.E-04(Ru) - 0.02(Mo)
Rare Earths[b]	0.01	1.E-03(Zr) - 8.E-03(U)	2.E-04

[a]Composite value for gap, meltdown, and vaporization (sparging) release.

[b]Includes Zr and Nb in the RSS (USNRC, 1975).

(Wichner, Kress and Lorenz, 1981). Other major differences between the RSS results and those calculated here involve the time dependence of the newer values that is lacking in the RSS values (See Addendum B1). This time dependence may be of great importance when the release-from-fuel fractions are coupled with those for the primary system and for the containment.

B.7 LIMITATIONS AND UNCERTAINTIES

Limitations of any approach used to estimate fission product release from fuel, as well as uncertainties involved in application of the approach, will serve to affect the degree of accuracy associated with results generated. These topics are briefly addressed here, as they pertain to the approach proposed in Section B.5. Gaps in available knowledge, which affect both the approach limitations and uncertainties in application, are also pointed out here.

B.7.1 Limitations of the Approach

The accuracy of results generated by the empirical approach described in this appendix depends to some extent on the accuracy of simulation of thermal/hydraulic conditions in the reactor, carried out by the MARCH code and its various subroutines, and on the amount of error introduced by extrapolation of release rate data from bench-scale, out-of-reactor test conditions to postulated full-scale accident conditions in a reactor. The number of simplifying assumptions necessitated by the paucity of available information on various release processes such as leaching, gas sparging, and fission product attachment to aerosols further affect the accuracy of the proposed approach.

Some of the uncertainties and limitations associated with the MARCH code approach to modeling the temperature histories of core materials before reactor vessel failure and to modeling core/concrete interactions after vessel failure are discussed in Murfin (1980). A more comprehensive discussion of MARCH limitations has been prepared, but was not available at the time the present document was written (Rivard, 1981). Before vessel failure, uncertainties are classified in Murfin as phenomenological or modeling uncertainties. The former term refers to uncertainties resulting from the lack of understanding, and thus of proper simulation, of physical interactions taking place during meltdown. The latter term refers to uncertainties resulting from improper or simplified simulation of processes that are believed to be understood. The identified uncertainties are not quantified in Murfin (1980), but it is noted that due to both a propogation of uncertainties throughout an accident sequence and greater phenomenological uncertainties as temperatures increase, the greatest uncertainties accumulate later in the accident.

Uncertainties in MARCH calculations related to loss of heat sink (i.e., loss of coolant), fission product release, and other heat transfer processes are cited as adding uncertainty to the time dependence of the temperature predictions. Phenomenological uncertainties associated with the Zr/H_2O reaction and fuel/clad melting – fuel/clad motion add further uncertainty to the time-dependent predictions, as do modeling uncertainties related to the core barrel failure, dropping of the core to the lower plenum of the vessel, and vessel breach.

The importance of these uncertainties in MARCH predictions before vessel failure lies in the relationship between temperature of core materials and fission product release rates. These rates are exponentially related to temperature in the approach used, as is evident in the curves of Figure B.1, and the equations fit to these curves (from Equation B.1). Unfortunately, the uncertainties associated with MARCH predictions have not been quantified, and thus, associated uncertainties in the estimated release fractions prior to vessel failure (Tables B.5 through B.15) cannot be provided. Such uncertainties are not likely to be important except for the less volatile elements in a core melt accident, since the volatile elements are lost rapidly once the core reaches near-melting temperatures.

After vessel failure, uncertainties will be related to those inherent in the HOTDROP and INTER subroutines of MARCH. The HOTDROP subroutine models the streaming of molten core materials onto the concrete from the reactor vessel following vessel failure, while INTER models molten core material/concrete interactions and solidified melt/concrete interactions (Murfin, 1980). A large degree of phenomenological uncertainty is attached to the former subroutine, but the degree to which such uncertainty affects subsequent calculations is not estimated.

Murfin (1980) states that the INTER code has been improved upon in more recently developed codes such as WECHSL and CORCON through added flexibility and more mechanistic and accurate descriptions of heat transfer to concrete, and of chemical reactions, although improvements in resultant accuracy and uncertainties are not discussed in a quantitative sense. Predictions by the INTER code of total gas production are approximately two times higher than comparable predictions by the WECHSL code at ten hours after core/concrete interactions begin (Murfin, 1980). It is believed by many that most available codes do not estimate gas generation well (Powers, 1982). The simplistic approach to chemistry in both codes, however, adds an undetermined amount of uncertainty to the gas production estimates.

After vessel failure, fission product release estimates are again quite sensitive to the gas production rates and temperatures estimated by INTER, as is evident from Eq. (B.2), used to calculate aerosol generation. Thus, inaccuracies in INTER output result in much larger inaccuracies in aerosol release rates after vessel failure. Unfortunately, potential inaccuracies are not known, and thus it is difficult to bound the calculated values (Section B.6.3) from this standpoint.

Limitations of the aspect of the proposed approach which entails extrapolation of fractional release rates from bench-scale tests to postulated full-scale accidents fall in the following two major categories. First of all, release rate fractions for portions of fuel rods, some unclad, and some spiked with fission product simulants, may not be representative of individual clad fuel rods and of the core as a whole in a reactor as temperatures increase during an accident. This is especially important as surface area-to-volume ratios change in the course of an accident, and become less similar to those in the experiments from which release rate fractions were estimated. Second, the environment in the reactor vessel or containment building may vary from experimental conditions, thus affecting chemistry and potentially the release rate. Errors introduced by extrapolation have not been quantified.

The method of estimating fission product leach rates incorporated into the proposed approach relies again on extrapolation from laboratory experiments to full-scale events, adding an undetermined degree of uncertainty to leaching estimates. These, however, would not be important in a meltdown accident. Sparging releases are estimated by assuming an instantaneous release, due to large uncertainties associated with the currently used 30 minute half-life (USNRC, 1975). Therefore, this estimation procedure likely overestimates initial sparging releases in this respect, although the final accuracy depends on the total sparging release fraction assumed, which is also uncertain, but most likely an overestimate (Section B.7.2). Thus, a number of limitations to the proposed approach exist which affect the accuracy of the estimated release values to an undetermined degree.

B.7.2 Uncertainties in Application of the Approach

In addition to limitations in the approach used to estimate fission product release from fuel, uncertainties in data used to apply the approach under specified conditions provide further potential for reducing accuracies in estimated values. These data uncertainties are due to the extremely small number of experimental values from which a parameter such as release fraction or leach constant must be estimated, and due to a lack of knowledge about accident conditions or processes to the extent that appropriate parameter values may not be selected for use in the thermal hydraulic codes used.

Uncertainties due to a lack of experimental data are present to a large degree in the estimated values of the fission product release fraction, the leaching constant, the sparging release fraction, and the aerosol production term. Uncertainties due to a lack of knowledge about such things as clad reactions with steam and other initial conditions which must be specified may affect the temperature and gas production values predicted by MARCH (Murfin, 1980) to an unknown degree, and thus affect the accuracy of source term estimations. Examples of these types of uncertainties have been pointed out throughout this appendix.

Together with limitations in the proposed approach, uncertainties which are introduced through application of the approach serve to necessitate a great deal of caution in interpreting release fraction values contained within this appendix, even though an effort has been made to insure conservatism of the estimated values. Quantification of the uncertainties involved would be quite complex, and thus, is not within the scope of this project. However, the importance of such quantification should not be overlooked.

B.7.3 Gaps in Knowledge

Gaps in available knowledge with which to estimate fission product release from fuel were discovered in the process of estimating releases from best-available information. All of these were either directly or indirectly mentioned in the discussion of limitations and uncertainties above. Specifically, knowledge about the accuracy of predictions of thermal/hydraulic codes available is largely unavailable in a quantitative sense. Furthermore, the effects of extrapolation from laboratory-scale to full-scale accident conditions on estimated fractional fission product releases are unknown. Chemistry and release of fission products after vessel failure are quite uncertain, and a more accurate means of characterizing these releases than that provided here is lacking. Releases due to the sparging process are particularly uncertain after failure, as is their time dependence. This is partially due to the uncertainty in chemistry of the fission products during this accident phase, since release estimates are based on a particular chemical form of the elements.

The proposed approach does not lend itself well to estimating fission product releases in less than meltdown accidents, since the MARCH code does not provide information on the number of rods failed. Such information is necessary for these latter accidents if only a portion of the core reaches clad-burst temperatures. (MARCH code predictions eventually result in all core regions above clad-burst temperatures). Therefore, a method of estimating releases in less than meltdown accidents is lacking.

B.8 SUMMARY

The purpose of this appendix was to review the subject of fission product release from fuel during LWR accidents from a qualitative and quantitative standpoint. Processes believed to be involved in release from fuel are first described in a qualitative sense (Section B.2), based on a review of recent literature on the topic. These processes, which potentially lead to fission product release, include, or result from, cladding rupture, transport from the fuel matrix, evaporation from molten fuel, leaching of failed fuel rods, oxidation of fragmented fuel, and sparging of the fuel/concrete mixture by gases produced in concrete decomposition.

Following the qualitative descriptions, the RSS approach to quantifying these processes is discussed (Section B.3). The RSS results are only time dependent with respect to melt releases and sparging releases. Release fractions for fission product categories specified in the RSS are presented in Table B.1 of this appendix. In summary, the entire inventory of noble gases, halides, alkali metals, and tellurium, selenium, and antimony are potentially released during a meltdown accident according to RSS results, while up to 12% of the alkaline earths, 5 to 90% of the noble metals, and 1% of the rare earths and refractory oxides are potentially released.

Section B.3 reviews work, both theoretical and experimental, which has been carried out since the RSS pertinent to predicting fission product release from fuel during LWR accidents. Most of this recent work was discussed in the Technical Bases Report (Wichner, Kress and Lorenz, 1981), and an updated empirical procedure for estimating fractional releases was presented in that report based on the results of the new work. A summary of that procedure is given in Section B.3, along with a discussion of the potential importance of fission product chemistry in release estimates.

A modification of the empirical approach presented in the Technical Bases Report for estimating fission product release during LWR accidents is proposed in Section B.5 as a state-of-the-art means of estimating release from fuel. Although other, more theoretical, means are being developed, they are not available for use for all fission products at present. The proposed approach is one which adds time dependence to the estimated release fractions through postulated accidents, and can be utilized without a firm knowledge of chemical forms present during an accident. This approach relies, as do the others, on knowledge of temperature profiles in the core throughout an accident, and thus necessitates the use of a thermal/hydraulic model to provide such information. Using this approach, fission product and total aerosol (radioactive and nonradioactive) releases are estimated for the following three stages in an accident: before core slumping into the lower plenum of the reactor vessel, after slumping and before vessel failure, and after vessel failure. Core/concrete interactions after vessel failure are considered in estimating aerosol production during this stage.

Using this approach, fission product releases during nine representative accident sequences (chosen from the RSS sequences) were estimated, the results of which are presented in Section B.6. The calculated values are more illustrative in purpose rather than definitive state-of-the-art values due to the associated uncertainties. Temperature profiles generated by the MARCH thermal/hydraulic code were used in these estimates. Assumptions necessary for carrying out the calculations are discussed in Section B.6.2.

The calculational results derived from the proposed approach are provided in Tables B.5 through B.16, and a comparison of cumulative releases estimated by the new procedure with RSS values given in Table B.17. Aside from the time dependence of the new values, differences between new estimated fractions and the RSS values lie in the somewhat larger potential releases predicted via the new approach for the alkaline earths, noble metals, and refractory oxides. Tables B1.1 through B1.10 of Addendum B1 present the new estimated release fractions as a function of time for the accident sequences considered.

Finally, a discussion of the limitations and uncertainties of the proposed approach, and gaps in knowledge pertinent to fission product release from fuel during LWR accidents is given in Section B.7. Quantification of uncertainties was not possible here, and therefore, the degree of accuracy to be attached to the estimated release fractions provided in Section B.6 has not been approximated. It is obvious, of course, that the total estimated release fraction of one for noble gases, halogens, alkali metals, and tellurium, selenium, and antimony represents an upper bound since the release fraction cannot exceed one.

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ADDENDUM B1

RELEASE FROM FUEL FRACTIONS AS A FRACTION OF TIME

Table B1.1. Release-from-fuel fractions as a function of time for a V-type accident sequence[a]

Element	Fractional release[b] [time after accident begins (min)]							
	30.5	36.0	41.5	47.0	52.5	91.0	121.0	Total
Xe, Kr	4.3(-2)[c]	0.25	0.47	0.20	3.1(-2)	0	0	1.0
I	2.6(-3)	0.26	0.49	0.21	3.2(-2)	0	0	1.0
Cs	2.6(-3)	0.26	0.49	0.21	3.2(-2)	0	0	1.0
Te, Ag	6.1(-4)	6.6(-2)	0.20	0.26	0.21	0.26	0	1.0
Sb	1.6(-4)	2.0(-2)	7.0(-2)	9.3(-2)	0.13	0.69	0	1.0
Mo	5.5(-5)	2.4(-3)	7.3(-3)	1.2(-2)	1.5(-2)	0.14	0.12	0.30
Ba	5.4(-5)	5.2(-3)	1.8(-2)	3.4(-2)	3.9(-2)	0.39	2.0(-2)	0.51
Sr	1.3(-5)	2.0(-3)	7.3(-3)	1.3(-2)	1.7(-2)	0.17	2.0(-4)	0.21
Zr	4.8(-8)	4.3(-6)	1.5(-5)	2.6(-5)	3.5(-5)	3.4(-4)	1.3(-3)	1.3(-3)
Ru	1.2(-6)	1.9(-4)	7.0(-4)	1.2(-3)	1.6(-3)	1.7(-2)	2.0(-2)	4.0(-2)
Fuel	4.3(-8)	9.0(-6)	3.3(-5)	6.0(-5)	8.0(-5)	9.5(-4)	6.0(-3)	7.0(-3)
Cladding (Zr)	4.8(-8)	4.3(-6)	1.5(-5)	2.6(-5)	3.5(-5)	3.4(-4)	1.0(-3)	1.3(-3)
(Sn)	1.6(-4)	2.0(-2)	7.0(-2)	9.3(-2)	0.13	0.69	0	1.0
Structure	4.8(-6)	4.3(-4)	1.5(-3)	2.6(-3)	3.5(-3)	3.4(-2)	1.0(-2)	5.1(-2)

[a]Began calculations when first core region reached 750° in MARCH code output.

[b]Refers to fractional release within the time step, not cumulative fraction.

[c]Notation 1.0(-6) equivalent to 1.0E-06.

Table B1.2. Release-from-fuel fractions as a function of time for S_2D-type accident sequence[a]

Element	Fractional release[b] [time after accident begins (min)]							
	5	11	17	23	27	38	146	Total
Xe, Kr	4.0(-2)[c]	8.7(-2)	0.41	0.38	7.0(-2)	1.0(-2)	0	1.0
I	1.0(-4)	9.1(-2)	0.43	0.40	7.3(-2)	1.0(-2)	0	1.0
Cs	8.5(-5)	9.1(-2)	0.43	0.40	7.3(-2)	1.0(-2)	0	1.0
Te, Ag	1.8(-5)	2.1(-2)	0.13	0.27	0.23	0.35	0	1.0
Sb	3.6(-6)	6.4(-3)	4.2(-2)	0.11	0.12	0.37	0.35	1.0
Mo	1.8(-6)	8.2(-4)	4.2(-3)	1.0(-3)	1.1(-2)	4.0(-2)	0.14	0.22
Ba	1.1(-6)	1.6(-3)	1.1(-2)	2.7(-2)	3.1(-2)	0.11	4.0(-2)	0.22
Sr	1.1(-6)	6.2(-4)	4.2(-3)	1.1(-2)	1.4(-2)	5.0(-2)	9.0(-4)	8.0(-2)
Zr	4.9(-9)	1.4(-6)	8.7(-6)	2.2(-5)	2.5(-5)	9.0(-5)	1.0(-3)	1.0(-3)
Ru	4.8(-8)	5.8(-5)	4.0(-4)	1.1(-3)	1.2(-3)	5.0(-3)	2.0(-2)	3.0(-2)
Fuel	1.8(-9)	2.7(-6)	2.0(-5)	6.0(-5)	9.0(-5)	2.6(-4)	6.0(-3)	6.0(-3)
Cladding (Zr)	4.9(-9)	1.4(-6)	8.7(-6)	2.2(-5)	2.5(-5)	9.0(-5)	1.0(-3)	1.1(-3)
(Sn)	3.6(-6)	6.4(-3)	4.2(-2)	0.11	0.12	0.37	0.35	1.0
Structure	4.9(-7)	1.4(-4)	8.7(-4)	2.2(-3)	2.5(-3)	9.0(-3)	1.0(-2)	2.5(-2)

[a]Began calculations when first core region reached 750° in MARCH code output.

[b]Refers to fractional release within the time step, not cumulative fraction.

[c]Notation 1.0(-6) equivalent to 1.0E-06.

Table B1.3. Release-from-fuel fractions as a function of time for a TMLB'-type accident sequence[a]

Element	Fractional release[b] [time after accident begins (min)]							
	5	15	25	35	46	56	357	Total
Xe, Kr	4.1(-2)[c]	0.35	0.54	7.0(-2)	0	0	0	1.0
I	1.1(-3)	0.37	0.56	7.3(-2)	0	0	0	1.0
Cs	1.1(-3)	0.37	0.56	7.3(-2)	0	0	0	1.0
Te, Ag	2.8(-4)	9.6(-2)	0.34	0.45	0.11	0	0	1.0
Sb	5.8(-5)	3.0(-2)	0.13	0.33	0.31	9.0(-2)	0.11	1.0
Mo	3.3(-5)	4.3(-3)	1.5(-2)	3.1(-2)	4.2(-2)	3.0(-2)	0.13	0.25
Ba	2.4(-5)	8.0(-3)	3.5(-2)	8.9(-2)	0.13	8.0(-2)	3.0(-2)	0.37
Sr	4.4(-6)	3.0(-3)	1.4(-2)	4.2(-2)	6.9(-2)	4.0(-2)	8.0(-4)	0.17
Zr	2.7(-8)	6.6(-6)	2.9(-5)	6.7(-5)	9.7(-5)	8.0(-5)	1.0(-3)	1.0(-3)
Ru	3.9(-7)	2.7(-4)	1.3(-3)	3.5(-3)	5.3(-3)	4.0(-3)	2.0(-2)	3.0(-2)
Fuel	1.5(-8)	1.2(-5)	6.0(-5)	4.7(-4)	1.2(-3)	2.3(-4)	6.0(-3)	8.0(-3)
Cladding (Zr)	2.7(-8)	6.6(-6)	2.9(-5)	6.7(-5)	9.7(-5)	8.0(-5)	1.0(-3)	1.3(-3)
(Sn)	5.8(-5)	3.0(-2)	0.13	0.33	0.31	9.0(-2)	0.11	1.0
Structure	2.7(-6)	6.6(-4)	2.9(-3)	6.7(-3)	9.7(-3)	8.0(-3)	1.0(-2)	3.8(-2)

[a]Began calculations when first core region reached 750° in MARCH code output.

[b]Refers to fractional release within the time step, not cumulative fraction.

[c]Notation 1.0(-6) equivalent to 1.0E-06.

Table B1.4. Release-from-fuel fractions as a function of time for an AD-type accident sequence[a]

Element	Fractional release[b] [time after accident begins (min)]							
	30.5	36.0	41.5	47.0	52.5	91.0	121.0	Total
Xe, Kr	4.3(-2)[c]	0.25	0.47	0.20	3.1(-2)	0	0	1.0
I	2.6(-3)	0.26	0.49	0.21	3.2(-2)	0	0	1.0
Cs	2.6(-3)	0.26	0.49	0.21	3.2(-2)	0	0	1.0
Te, Ag	6.1(-4)	6.6(-2)	0.20	0.26	0.21	0.26	0	1.0
Sb	1.6(-4)	2.0(-2)	7.0(-2)	9.3(-2)	0.13	0.69	0	1.0
Mo	5.5(-5)	2.4(-3)	7.3(-3)	1.2(-2)	1.5(-2)	0.14	0.12	0.30
Ba	5.4(-5)	5.2(-3)	1.8(-2)	3.4(-2)	3.9(-2)	0.39	2.0(-2)	0.51
Sr	1.3(-5)	2.0(-3)	7.3(-3)	1.3(-2)	1.7(-2)	0.17	2.0(-4)	0.21
Zr	4.8(-8)	4.3(-6)	1.5(-5)	2.6(-5)	3.5(-5)	3.4(-4)	1.3(-3)	1.3(-3)
Ru	1.2(-6)	1.9(-4)	7.0(-4)	1.2(-3)	1.6(-3)	1.7(-2)	2.0(-2)	4.0(-2)
Fuel	4.3(-8)	9.0(-6)	3.3(-5)	6.0(-5)	8.0(-5)	9.5(-4)	6.0(-3)	7.0(-3)
Cladding (Zr)	4.8(-8)	4.3(-6)	1.5(-5)	2.6(-5)	3.5(-5)	3.4(-4)	1.0(-3)	1.3(-3)
(Sn)	1.6(-4)	2.0(-2)	7.0(-2)	9.3(-2)	0.13	0.69	0	1.0
Structure	4.8(-6)	4.3(-4)	1.5(-3)	2.6(-3)	3.5(-3)	3.4(-2)	1.0(-2)	5.1(-2)

[a]Began calculations when first core region reached 750° in MARCH code output.

[b]Refers to fractional release within the time step, not cumulative fraction.

[c]Notation 1.0(-6) equivalent to 1.0E-06.

Table B1.5. Release-from-fuel fractions as a function of time for a TQUV-type accident sequence[a]

Element	Fractional release[b] [time after accident begins (min)]							
	5	15	25	35	40	58	88	Total
Xe, Kr	4.0(-2)[c]	9.2(-2)	0.46	0.34	5.8(-2)	1.0(-2)	0	1.0
I	1.5(-4)	9.5(-2)	0.48	0.35	5.6(-2)	2.0(-2)	0	1.0
Cs	9.7(-5)	9.5(-2)	0.48	0.35	5.6(-2)	2.0(-2)	0	1.0
Te, Ag	2.2(-5)	2.2(-2)	0.15	0.26	0.14	0.21	0.22	1.0
Sb	4.5(-6)	6.6(-3)	4.8(-2)	0.10	7.2(-2)	0.12	0.65	1.0
Mo	2.5(-6)	1.1(-3)	5.6(-3)	1.2(-2)	8.3(-3)	4.0(-2)	0.14	0.21
Ba	1.3(-6)	1.8(-3)	1.3(-2)	2.8(-2)	2.1(-2)	4.0(-2)	4.0(-2)	0.14
Sr	1.9(-6)	6.1(-4)	4.8(-3)	1.1(-2)	8.6(-3)	1.0(-2)	1.0(-3)	4.0(-2)
Zr	8.1(-9)	1.5(-6)	1.1(-5)	2.3(-5)	1.8(-5)	4.0(-5)	1.0(-3)	1.0(-3)
Ru	7.6(-8)	5.7(-5)	4.7(-4)	1.1(-3)	8.3(-4)	1.0(-3)	2.0(-2)	2.0(-2)
Fuel	2.8(-9)	2.5(-6)	2.1(-5)	5.0(-5)	4.0(-5)	4.2(-5)	6.0(-3)	6.0(-3)
Cladding (Zr)	8.1(-9)	1.5(-6)	1.1(-5)	2.3(-5)	1.8(-5)	4.0(-5)	1.0(-3)	1.1(-3)
(Sn)	4.5(-6)	6.6(-3)	4.8(-2)	0.10	7.2(-2)	0.12	0.65	1.0
Structure	8.1(-7)	1.5(-4)	1.1(-3)	2.3(-3)	1.8(-3)	4.0(-3)	1.0(-2)	1.9(-2)

[a] Began calcualtions when first core region reached 750° in MARCH code output.

[b] Refers to fractional release within the time step, not cumulative fraction.

[c] Notation 1.0(-6) equivalent to 1.0E-06.

Table B1.6. Release-from-fuel fractions as a function of time for a TC-type accident sequence[a]

Element	Fractional release[b] [time after accident begins (min)]							
	5	15	26	36	40	58	89	Total
Xe, Kr	4.0(-2)[c]	0.18	0.55	0.21	1.3(-2)	0	0	1.0
I	5.0(-4)	0.19	0.58	0.22	1.4(-2)	0	0	1.0
Cs	4.8(-4)	0.19	0.58	0.22	1.4(-2)	0	0	1.0
Te, Ag	1.2(-4)	4.5(-2)	0.23	0.33	0.19	0.10	0.10	1.0
Sb	2.3(-5)	1.4(-2)	7.8(-2)	0.15	0.14	9.0(-2)	0.53	1.0
Mo	1.4(-5)	2.1(-3)	8.7(-3)	1.7(-2)	1.4(-2)	4.0(-2)	0.14	0.22
Ba	1.0(-5)	3.7(-3)	2.1(-2)	4.2(-2)	4.1(-2)	4.0(-2)	4.0(-2)	0.19
Sr	2.8(-6)	1.3(-3)	8.0(-3)	1.7(-2)	1.8(-2)	1.0(-2)	1.0(-3)	5.0(-2)
Zr	1.6(-8)	3.1(-6)	1.7(-5)	3.6(-5)	3.3(-5)	4.0(-5)	1.0(-3)	1.0(-3)
Ru	1.8(-7)	1.2(-4)	7.6(-4)	1.6(-3)	1.6(-3)	1.0(-3)	2.0(-2)	3.0(-2)
Fuel	7.0(-9)	5.5(-6)	3.6(-5)	9.0(-5)	1.3(-4)	4.2(-5)	6.0(-3)	6.0(-3)
Cladding (Zn)	1.6(-8)	3.1(-6)	1.7(-5)	3.6(-5)	3.3(-5)	4.0(-5)	1.0(-3)	1.1(-3)
(Sn)	2.3(-5)	1.4(-2)	7.8(-2)	0.15	0.14	9.0(-2)	0.53	1.0
Structure	1.6(-6)	3.1(-4)	1.7(-3)	3.6(-3)	3.3(-3)	4.0(-3)	1.0(-2)	2.3(-2)

[a]Began calculations when first core region reached 750° in MARCH code output.

[b]Refers to fractional release within the time step, not cumulative fraction.

[c]Notation 1.0(-6) equivalent to 1.0E-06.

Table B1.7. Release-from-fuel fractions as a function of time for a TW-type accident sequence[a]

Element	Fractional release[b] [time after accident begins (min)]							
	30	60	90	120	150	180	210	Total
Xe, Kr	9.1(-2)[c]	0.86	5.2(-2)	1.6(-5)	0	0	0	1.0
I	5.4(-2)	0.89	5.4(-2)	1.6(-5)	0	0	0	1.0
Cs	5.3(-2)	0.89	5.4(-2)	1.6(-5)	0	0	0	1.0
Te, Ag	1.4(-2)	0.43	0.45	0.11	7.6(-3)	0	0	1.0
Sb	3.2(-3)	0.15	0.32	0.29	0.15	2.0(-2)	7.0(-2)	1.0
Mo	1.4(-3)	2.1(-2)	4.6(-2)	6.6(-2)	7.0(-2)	5.0(-2)	0.12	0.37
Ba	1.2(-3)	4.2(-2)	0.11	0.15	0.15	4.0(-2)	2.0(-2)	0.52
Sr	1.9(-4)	1.6(-2)	4.4(-2)	7.0(-2)	8.0(-2)	2.0(-2)	8.0(-4)	0.23
Zr	1.0(-6)	3.5(-5)	9.6(-5)	1.6(-4)	1.9(-4)	6.0(-5)	1.0(-3)	2.0(-3)
Ru	2.1(-5)	1.5(-3)	4.4(-3)	7.3(-3)	9.0(-3)	2.0(-3)	2.0(-2)	4.0(-2)
Fuel	7.5(-7)	7.0(-5)	2.1(-4)	3.6(-4)	4.5(-4)	7.0(-5)	6.0(-3)	7.0(-3)
Cladding (Zr)	1.0(-6)	3.5(-5)	9.6(-5)	1.6(-4)	1.9(-4)	6.0(-5)	1.0(-3)	1.5(-3)
(Sn)	3.2(-3)	0.15	0.32	0.29	0.15	2.0(-2)	7.0(-2)	1.0
Structure	1.0(-4)	3.5(-3)	9.6(-3)	1.6(-2)	1.9(-2)	6.0(-3)	1.0(-2)	6.4(-2)

[a]Began calculations when first core region reached 750° in MARCH code output.

[b]Refers to fractional release within the time step, not cumulative fraction.

[c]Notation 1.0(-6) equivalent to 1.0E-06.

Table B1.8. Release from fuel fractions as a function of time for a AD-"gap release"-type accident sequence, 50% of core above 750°C[a]

Element	Fractional release[b] [time after accident begins (min)]						
	1	2	3	4	5	14400[c]	Total
Xe, Kr	4.0(-2)[d]	3.6(-6)	1.6(-5)	1.5(-4)	2.4(-3)	0.96	1.0
I	7.6(-7)	3.7(-6)	1.7(-5)	1.6(-4)	2.5(-3)	0.03	0.03
Cs	7.8(-8)	7.1(-7)	7.9(-6)	1.5(-4)	2.4(-3)	3.0(-3)	6.6(-3)
Te, Ag	2.3(-8)	1.9(-7)	1.9(-6)	4.0(-5)	5.7(-4)	6.0(-3)	6.9(-3)
Sb	6.1(-9)	4.6(-8)	4.1(-7)	7.8(-6)	1.5(-4)	6.0(-3)	6.2(-3)
Mo	5.3(-9)	3.4(-8)	2.4(-7)	5.1(-6)	4.9(-5)	5.4(-4)	5.9(-4)
Ba	9.5(-10)	8.9(-9)	1.0(-7)	3.4(-6)	5.1(-5)	4.0(-4)	4.5(-4)
Sr	3.3(-8)	9.8(-8)	2.2(-7)	6.0(-7)	1.2(-5)	4.0(-4)	4.1(-4)
Zr	8.1(-11)	3.1(-10)	1.0(-9)	4.0(-9)	4.3(-8)	1.0(-4)	1.0(-4)
Ru	5.4(-10)	2.3(-9)	8.7(-9)	5.3(-8)	1.1(-6)	5.0(-4)	5.0(-4)
Fuel	2.0(-11)	8.5(-11)	3.2(-10)	1.9(-9)	4.1(-8)	1.0(-4)	1.0(-4)
Cladding (Zr)	8.1(-11)	3.1(-10)	1.0(-9)	4.0(-9)	4.3(-8)	e	4.8(-8)
(Sn)	6.1(-9)	4.6(-8)	4.1(-7)	7.8(-6)	1.5(-4)	e	1.6(-4)
Structure	8.1(-9)	3.1(-8)	1.0(-7)	4.0(-7)	4.3(-6)	e	4.8(-6)

[a]Began calculations when first core region reached 750° in MARCH code output.

[b]Refers to fractional release within the time step, not cumulative fraction.

[c]Leaching after 10 days exposure of damaged core to water.

[d]Notation 1.(-6) equivalent to 1.0E-06.

[e]Leaching of clad and structure neglected.

Table B1.9. Release from fuel fractions as a function of time for a AD-"gap release"-type accident sequence, 50% of core above 900°C[a]

Element	Fractional release[b] [time after accident begins (min)]						
	1	3	5	6	7	14400[c]	Total
Xe, Kr	4.0(-2)[d]	1.7(-4)	1.9(-2)	3.0(-2)	5.3(-2)	0.86	1.0
I	4.5(-6)	1.7(-4)	2.0(-2)	3.1(-2)	5.5(-2)	0.03	0.16
Cs	7.9(-7)	1.6(-4)	2.0(-2)	3.1(-2)	5.5(-2)	2.0(-3)	0.13
Te, Ag	2.1(-7)	4.2(-5)	4.5(-3)	7.1(-3)	1.3(-2)	6.0(-3)	3.1(-2)
Sb	5.2(-8)	8.3(-6)	1.3(-3)	2.2(-3)	4.0(-3)	6.0(-3)	1.2(-2)
Mo	4.0(-8)	5.4(-6)	2.2(-4)	2.9(-4)	4.4(-4)	5.0(-4)	1.5(-3)
Ba	9.8(-9)	3.5(-6)	3.6(-4)	5.7(-4)	1.0(-3)	4.0(-4)	2.3(-3)
Sr	1.3(-7)	8.2(-7)	1.3(-4)	2.1(-4)	3.9(-4)	4.0(-4)	1.1(-3)
Zr	3.9(-10)	5.0(-9)	3.0(-7)	4.6(-7)	8.6(-7)	1.0(-4)	1.0(-4)
Ru	2.9(-9)	6.1(-8)	1.2(-5)	1.9(-5)	3.8(-5)	5.0(-4)	5.7(-4)
Fuel	1.1(-10)	2.2(-9)	5.2(-7)	9.0(-7)	1.7(-6)	1.0(-4)	1.0(-4)
Cladding (Zr)	3.9(-10)	5.0(-9)	3.0(-7)	4.6(-7)	8.6(-7)	e	1.6(-6)
(Sn)	5.2(-8)	8.3(-6)	1.3(-3)	2.2(-3)	4.0(-3)	e	6.3(-3)
Structure	3.9(-8)	5.0(-7)	3.0(-5)	4.6(-5)	8.6(-5)	e	1.6(-4)

[a]Began calculations when first core region reached 750° in MARCH code output.

[b]Refers to fractional release within the time step, not cumulative fraction.

[c]Leaching after 10 days exposure of damaged core to water.

[d]Notation 1.(-6) equivalent to 1.0E-06.

[e]Leaching of clad and structure neglected.

Table B1.10. Release from fuel fractions as a function of time for a AD-½-type accident sequence[a]

Element	Fractional release[b] [time after accident begins (min)]						
	4	8	12	16	17	14400[c]	Total
Xe, Kr	4.5(-2)[d]	0.17	0.39	0.28	3.9(-2)	0.08	1.0
I	2.6(-3)	0.17	0.41	0.29	4.1(-2)	4.0(-3)	0.91
Cs	2.7(-3)	0.17	0.41	0.29	4.1(-2)	4.0(-3)	0.91
Te, Ag	6.1(-4)	4.2(-2)	0.13	0.19	5.3(-2)	4.0(-3)	0.42
Sb	1.6(-4)	1.3(-2)	4.4(-2)	7.4(-2)	2.3(-2)	5.0(-3)	0.15
Mo	5.5(-5)	1.5(-3)	4.7(-3)	7.9(-3)	2.4(-3)	5.0(-4)	1.7(-2)
Ba	5.4(-5)	3.3(-3)	1.1(-2)	2.0(-2)	6.4(-3)	4.0(-4)	4.1(-2)
Sr	1.2(-5)	1.3(-3)	4.5(-3)	8.0(-3)	2.6(-3)	4.0(-4)	1.6(-2)
Zr	4.8(-8)	2.7(-6)	9.4(-6)	1.7(-5)	5.4(-6)	1.0(-4)	1.4(-4)
Ru	1.2(-6)	1.2(-4)	4.3(-4)	7.6(-4)	2.5(-4)	5.0(-4)	2.1(-3)
Fuel	4.3(-8)	5.5(-6)	2.0(-5)	3.6(-5)	1.2(-5)	1.0(-4)	1.7(-4)
Cladding (Zr)	4.8(-8)	2.7(-6)	9.4(-6)	1.7(-5)	5.4(-6)	*e*	3.5(-5)
(Sn)	1.6(-4)	1.3(-2)	4.4(-2)	7.4(-2)	2.3(-2)	*e*	0.15
Structure	4.8(-6)	2.7(-4)	9.4(-4)	1.7(-3)	5.4(-4)	*e*	3.5(-3)

[a]Began calculations when first core region reached 750° in MARCH code output.

[b]Refers to fractional release within the time step, not cumulative fraction.

[c]Leaching after 10 days exposure of damaged core to water.

[d]Notation 1.(-6) equivalent to 1.0E-06.

[e]Leaching of clad and structure neglected.

APPENDIX C

TRANSPORT THROUGH THE PRIMARY COOLANT SYSTEM

C.1 INTRODUCTION

Radionuclides released from the core materials into the reactor pressure vessel (RPV) would need to pass through some portion of the primary coolant system (PCS) in order to reach the containment. Both the particular pathway followed and the conditions encountered would be functions of the accident scenario. The factors affecting those conditions and their impacts on radionuclide transport through the PCS are considered in this appendix.

The processes which govern transport of radionuclides in the PCS, along with the effects of certain PCS conditions on these processes, are discussed qualitatively in Section C.2. A review of previous considerations of transport through the PCS follows this discussion. In particular, in Section C.3, the approach used in the Reactor Safety Study (RSS) for such considerations is reviewed. In addition, a more recent treatment (Gieseke and Kuhlman, 1981), which represents the state-of-the-art of predicting PCS transport during LWR accidents, is summarized. Next, in Section C.4, a simple approach for estimating radionuclide escape fractions from the PCS is presented. That approach is based in part on the previously-considered state-of-the-art treatment. Because there are a number of significant limitations, uncertainties, and gaps in knowledge associated with the estimation of radionuclide escape fractions, a separate discussion (Section C.5) is devoted to these topics.

C.2 DESCRIPTION OF PROCESSES

Processes pertinent to radionuclide transport within the PCS during an LWR accident involve those responsible for flow out of the core region and through the PCS upon release from the core materials, those responsible for retention of radionuclides in the PCS, and those responsible for potential reentrainment of initially retained materials. The extent to which each of these sets of processes was operating in any given accident would affect the degree to which radionuclides were released to the containment and potentially to the environment.

Radionuclides volatilized from the core during LWR accidents would remain in vapor states for varying lengths of time, depending on the vapor pressures of the particular species and the temperatures encountered during transport through the PCS, as well as other factors. Those species with high vapor pressures would tend to be transported as gases. In contrast, those species with low vapor pressures might be transported initially as vapors but might condense into aerosols soon after their release from the core. In addition, aerosols might be released directly by entrainment of material in generated gases. Thus, radionuclides would be transported through the PCS as gases, vapors and aerosol particles.

The major means of transport of radionuclides through the PCS during any accident in which core temperatures were sufficient to cause at least cladding rupture would generally be the flowing steam produced as water surrounding the overheated core were vaporized. Combined with the steam, there would be hydrogen generated by steam-cladding interactions. Because the rate of movement of the steam and hydrogen could vary throughout an accident, the rate of transport of radionuclides through the PCS likewise could vary throughout an accident. Another means of transport of radionuclides would be expansion of the gases in the RPV due to heating. Typically, gas expansion would drive radionuclides out of the PCS at a much slower rate and to a lesser degree than steam flow. Still another means of transport through the PCS would be the flow of the coolant water if radionuclides were entrained in that water during the accident.

The PCS would serve as a barrier to release of radionuclides to the environment to the extent that retention mechanisms acted to reduce flow-through to the containment. For radionuclides in vapor form, processes potentially enhancing retention would include condensation onto surfaces of the PCS, onto aerosols, or in the steam; sorption onto surfaces; and "scrubbing" through interaction with water. The condensation and sorption processes would be highly dependent on the thermal-hydraulic conditions in the PCS during the accident. The scrubbing process would occur only if liquid water were encountered by the flowing gases at some point in the coolant system.

Processes likely to promote retention of aerosols would include condensation of vapors onto aerosols; condensation of steam onto aerosols; agglomeration of aerosols; deposition and impaction of aerosols onto PCS surfaces; and scrubbing, as with gases and vapors. Again, the condensation processes would be quite temperature dependent, and thus might alter the aerosol concentrations according to the temperature gradients present in the PCS. Condensation of vapors onto aerosols, agglomeration of aerosols, and deposition of aerosols from the atmosphere would all be processes which would be highly dependent on aerosol concentrations. Increased aerosol concentrations generally would enhance the rates of these processes. Both agglomeration and deposition processes would tend to increase in importance as steam velocities through the PCS decreased and aerosol residence times in the PCS increased.

Other processes which might signficantly affect the release of radionuclides from the PCS to the containment would include radioactive decay and chemical reactions. Both processes could alter the physical state of a species by making it more or less volatile and thus affect its likelihood of retention. In addition, a reaction of any species with either aerosols or PCS surfaces could directly affect its potential retention.

Some processes would operate to reverse the effect of, or oppose, retention processes in the PCS. One of these processes would be resuspension of particulates, whereby aerosols deposited on surfaces in the PCS might be reentrained in the gas flowing by. Gases which were condensed onto surfaces also might be reentrained in the gas flow as the result of revaporization. Further, gases which were initially absorbed onto surfaces might be desorbed and thus reenter the gas stream. Finally, electrostatic interactions between suspended aerosols and those deposited on surfaces might counter deposition as surface deposits increased.

All of these processes, their effects in general, and their potential impacts on escape from the PCS during both limited-core-damage and meltdown accidents are summarized in Table C.1. A more in-depth discussion of basic processes in the PCS can be found in other documents

Table C.1. Effects of natural processes in primary coolant system during accidents involving core damage

Process	Species	General effects	Effects on radionuclides escaping from primary coolant system: Limited-core-damage accidents	Effects on radionuclides escaping from primary coolant system: Meltdown accidents
Agglomeration (Brownian; gravitational; turbulent)	Aerosols	Increases particle weights	The low aerosol concentrations typical of some accidents would not be rapidly decreased by agglomeration and deposition; however, the high aerosol concentrations in some accidents could result in substantial agglomeration and deposition if steam velocities through the PCS were low	High aerosol concentrations could be rapidly decreased by agglomeration and deposition for those accidents in which low steam velocities through the PCS occurred
Steam condensation	Aerosols	Increase particle weights; compacts individual particles	Due to the relatively low temperatures in the PCS, the aerosols might be affected by condensing steam.	Due to the relatively high temperatures in the PCS, the aerosols generally would not be affected by condensing steam.
"Condensation" onto particles (evaporation from particles) (condensation; adsorption;...)	Gases; aerosols	Increases particle weights; decreases gas concentrations; changes aerosol composition	Due to the relatively low temperatures in the PCS, vapor deposition could be substantial for all compounds.	Due to the relatively high temperatures in at least parts of the PCS, vapor deposition would affect only compounds of intermediate and low volatility in much of the PCS.
Deposition (gravitational; diffusional; thermophoretic; diffusiophoretic)	Gases; aerosols	Decreases gas or aerosol concentration	The low aerosol concentrations typical of some accidents would not be rapidly decreased by agglomeration and deposition;[a] however, the high aerosol concentrations in some accidents could result in substantial agglomeration and deposition if steam velocities through the PCS were low	High aerosol concentrations could be rapidly decreased by agglomeration and deposition for those accidents in which low steam velocities through the PCS occurred.

Table C 1 (continued)

Process	Species	General effects	Effects on radionuclides escaping from primary coolant system	
			Limited-core-damage accidents	Meltdown accidents
Resuspension	Aerosols	Increases aerosol concentration	Probably not important	Might be large for accidents with substantially varying steam flow rates through the PCS
"Condensation" onto surfaces (evaporation from surfaces) (condensation, adsorption,)	Gases	Decreases gas concentration	Due to the relatively low temperatures of surfaces in the PCS, vapor deposition could affect compounds of all volatilities both inside and outside the RPV	Due to the relatively high temperatures of the surfaces of the PCS, vapor deposition generally would affect only compounds of intermediate volatility and would occur only outside the core region[b]
Electrostatic interactions	Aerosols	May hinder deposition	Probably not important	Probably not important
Radioactive decay	Gases, aerosols	Changes chemical composition of gases and aerosols, heats aerosols	Chemical changes potentially important	Chemical changes potentially important
Chemical reactions	Gases, aerosols	Changes chemical compositions of gases and aerosols, may decrease gas concentration	Potentially important (for example, formation of CsI and reaction of Te with PCS surfaces)	Potentially important (for example, formation of CsI and reaction of Te with PCS surfaces)
Scrubbing (diffusiophoresis,)	Gases, aerosols	Decreases concentrations	Passage through standing water in the PCS could substantially reduce both gas and aerosol concentrations exiting from the PCS	Passage through standing water in the PCS could substantially reduce both gas and aerosol concentrations exiting from the PCS

[a]The relatively large particles ejected during the gap release might be deposited in the PCS

[b]This assumes that compounds of low volatility would condense into aerosols before reaching any surfaces suitable for condensation

(e.g., Baybutt, 1981; Nuclear Energy Agency, 1979) and will not be provided here. What must be noted here, however, is that the effects of all of these processes would be dependent on the magnitude and the rates of the radionuclide and other releases from the core, the physical conditions in the primary coolant system, the coolant system's geometry, the composition of the surfaces in the system, and the chemical and physical processes undergone by the source term materials. Therefore, the impacts of the processes all would be subject to variation with accident conditions through time and would be dependent on reactor design. Both the accident scenario and the reactor design would help to determine such things as temperature along the steam flow path in the PCS, the steam flow path itself, and whether water were present along that path. Thus, all of these factors need to be considered in determining potential escape of radionuclides from the PCS during LWR accidents.

C.3 SUMMARY OF STATE-OF-THE-ART

Historically, the movement of radionuclides through the PCS has not received a level of consideration comparable to that given to either the releases of radionuclides from the core materials or the transport of those materials within the containment. This is in part because transport through the PCS is rather complicated and in part because early calculations appeared to indicate that the PCS generally would not be a region significantly affecting radionuclide source terms. In this section, the two previous technical documents which have considered retention of radionuclides in the PCS in comparatively comprehensive manners are reviewed.

C.3.1 Reactor Safety Study

The Reactor Safety Study (RSS; USNRC, 1975) included the first "complete" treatment of transport through, and retention in, the PCS. In that study, a simple bounding approach for estimating PCS retention factors, or escape fractions, was adopted, although the desirability of a much more detailed analysis was acknowledged at the time.

In discussing the escape of radionuclides from the PCS, the RSS recognized the potential for retention in the PCS via plateout and condensation processes. The RSS analysis of retention by such processes concentrated on consideration of just the upper region of the pressure vessel (Ritzman, 1975). Inasmuch as that region represented a very high temperature regime of the PCS for all the meltdown accidents considered, little retention in the PCS was predicted. Thus permanent retention resulting from plateout and condensation was concluded to be negligible for all meltdown sequences. (The possibility of significant permanent retention in the PCS by those two processes was discussed at some length for a cold-leg break meltdown in a PWR but was not treated numerically. Such retention was not even discussed in the RSS for limited-core-damage accidents.)

It was further concluded in the RSS that, with the exception of one special case, retention of any radionclides in the PCS by all other processes also would be only temporary,* and that escape fractions of unity should be used in estimating releases to the containment atmosphere for most accidents. The special case identified for which escape fractions less than one were suggested was an accident involving a break of a recirculation line in the coolant system of a boiling water reactor (BWR). In that accident, the emergency core cooling system was operational but did not effectively cool the core due to abnormal conditions in the RPV. Under these conditions, released radionuclides would have to pass through several feet of water to reach the containment. On the basis of retention only by the water, escape fractions of 0.1 were suggested for all species other than noble gases.

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*For example, a special case identified in which temporary retention by processes other than plateout and condensation would occur was that of a meltdown in a BWR in which the emergency core cooling system was not operational. Because gas flow through the PCS in such an accident would be a result of only gas expansion rather than steam flow, it was estimated that only about two-thirds of the radionuclides released from the core would expand out of the vessel before RPV melt-through. Based on this rationale, escape fractions of two-thirds were suggested for all radionuclides.

C.3.2 Work Since the Reactor Safety Study

In the consideration of radionuclide retention presented in the RSS, retention in the outer reaches of the PCS was neglected. In addition, aerosol processes were largely ignored as potential removal mechanisms. Recently, however, it has been indicated that aerosol processes in the previously neglected regions could be quite important for retention of many radionuclides for some accident conditions. To date, detailed considerations of the effects of such processes have been described in only one report, the Technical Bases Report (USNRC, 1981). In that report, the potential effects of aerosol processes, along with the anticipated effects of previously considered processes, were discussed (Gieseke and Kuhlman, 1981).

In the studies supporting the Technical Bases Report, two different computer codes, TRAP (Baybutt and Jordan, 1977; Jordan, Gieseke and Baybutt, 1979) and QUICK (Gieseke, Jordan and Lee, 1979), were used to consider retention of radionuclides in the PCS. Several severe core damage accident sequences, including some meltdown sequences, were investigated with each of the two codes. In addition, a minor core damage accident was investigated using only TRAP. In all the calculations, only three distinct groups of radioactive species, namely, elemental iodine, cesium iodide (CsI), and aerosols, were considered. For all considerations, TRAP was used to estimate the fates of the more volatile radioactive species, along with the residence times of the steam and hydrogen, and therefore of the aerosols, in the PCS during the course of an accident, while QUICK was utilized to estimate retention of aerosols in the PCS on the basis of the TRAP residence time estimates.

TRAP is the only existing computer code which has been developed specifically to describe radionuclide retention in the PCS. The processes considered in TRAP include condensation and evaporation of radionuclide vapors onto and from surfaces, condensation and evaporation of those vapors onto and from particles, sorption onto surfaces, inertial deposition from turbulent flow, plating onto surfaces by diffusion and thermophoresis, and turbulent and Brownian coagulation. TRAP does not take into account sorption onto particles, condensation of steam, gravitational agglomeration, scrubbing by water, gas phase-liquid phase partitioning, chemical reactions and radioactive decay. In addition, it may or may not, depending on the source consulted, include gravitational settling (Nuclear Energy Agency, 1979; Baybutt, 1981; Gieseke and Kuhlman, 1981).* TRAP was developed on the basis of theoretical considerations and results of small scale experiments of limited scope and has not been validated by results of more appropriate experiments. Typically, TRAP is best suited for consideration of the transport of relatively volatile species in environments with low net aerosol concentrations; it currently is not adequate for describing environments in which certain aerosol processes are important.†

In contrast to TRAP, QUICK was developed expressly to describe aerosol processes. Thus, in some respects, it is better suited than TRAP for considering transport of less volatile species in the PCS. More exactly, QUICK contains models for some of the processes not included in TRAP, for example, gravitational agglomeration. (For a listing of the processes modeled in QUICK, see Table D.5 in Appendix D.) However, QUICK does not include any vapor processes such as condensation, evaporation, and so forth. Furthermore, it is a single compartment code in many ways not appropriate for considering the details of transport through a system as complex as the PCS. Its originally intended purpose was for description of aerosol behavior in liquid metal fast breeder reactors (LMFBR's) after accidents involving severe core damage.

The accident considered in the work performed for the Technical Bases Report involving only minor fuel damage was a terminated cold-leg loss-of-coolant accident (LOCA) in which some core damage was postulated to occur before adequate cooling was established by operation of the emergency core cooling system. The severe core damage accident sequences

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*The authors of the Technical Bases Report did not distinguish clearly between those processes not included in TRAP and those processes not considered in the TRAP calculations presented in that report. Indeed, they seemed to suggest that all agglomeration processes were neglected in the presented TRAP calculations even though TRAP includes some such processes.

†Since the completion of the Technical Bases Report, TRAP has been modified to include models for gravitational agglomeration and settling. Thus it is now much better suited for describing aerosol behavior than it was at the time of that study.

considered were divided into two categories: degraded core sequences and core meltdown sequences. The sequences analyzed in the former category were a TMI-like transient-initiated stuck-open relief valve sequence and a large pipe break sequence, both with delayed functioning of the emergency core cooling system. The sequences analyzed in the latter, meltdown category for a PWR included a transient sequence with loss of heat removal (TMLB')* and a large pipe break with emergency core cooling system failure (AD). The core meltdown sequences analyzed for a BWR included a transient with failure to scram (TC) and a large pipe break with emergency core cooling system failure (AE).

For the accidents involving lesser core damage than complete melting, the TRAP analyses began with cladding rupture and ended with recovering of the core.† For those accidents in which complete core melting is postulated, the TRAP analyses began with melt initiation and ended just prior to reactor pressure vessel (RPV) failure.† In all cases, a constant input of aerosols into the PCS was assumed. All released materials were assumed to be transported instantly to the upper plenum after release from the core materials. Scrubbing by liquid water in the PCS was ignored for all accident sequences considered.

For the minor core damage accident (the normally terminated cold-leg LOCA), an appreciable fraction of elemental iodine (33-50%) was predicted to escape directly to the containment atmosphere. In contrast, only very small fractions of CsI and the aerosols were predicted to escape to that atmosphere. (Typically, larger fractions of all species would escape for a hot-leg LOCA.)

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*See Appendix A for descriptions of these accident sequences and an explanation of the notation used here.

†For the one accident which involved only minor core damage, RELAP-WREM code estimates of steam flow rates and temperatures in the PCS were used. For all other accidents, the conditions in the PCS were extrapolated from MARCH (Wooton and Avci, 1980) results. (Codes such as RELAP and TRAC incorporate sophisticated treatments of the thermal-hydraulics of the primary coolant system. The range of applicability of those codes extends only up to uncovering of the core. In contrast, MARCH includes a simplified treatment of the thermal-hydraulics of the primary coolant system. However, it was developed expressly for consideration of the thermal-hydraulics during all parts of meltdown accidents and thus can be used after uncovering of the core.)

The results presented in the Technical Bases Report for almost all the accidents analyzed, other than the minor core damage one, are summarized in Table C.2 and Figure C.1, both taken from that report (Gieseke and Kuhlman, 1981). Multiple results for a given sequence (e.g., TMLB´-1, TMLB´-2, ...) represent the effects of using different thermal-hydraulic and/or aerosol characteristics in the calculations. The various points indicated in Figure C.1 for a given treatment of a particular sequence represent the concentrations and "residence times" obtained for various unspecified steam velocities within each of those sequences. The "residence times" in that figure are based on an ignorance of when melt-through of the RPV occurs.

As can be seen by inspecting Table C.2, very little retention of elemental iodine was predicted for all the accident sequences investigated. (Potential reactions of iodine with other species and with PCS surfaces were not considered.) In contrast, as can be seen in Table C.2, as well as in Figure C.1, the estimated retention of both moderately volatile species, like CsI, and less volatile species, like strontium, varied considerably from essentially none to large fractions of the amounts released from the core materials for different accident sequences. Typically, of the sequences examined, large LOCA's, especially hot-leg ones (denoted in the table as AD*), resulted in the lowest predicted retention of both moderately volatile and not-too volatile species in the PCS while transient-initiated accidents resulted in the largest retention of all those species.

In the Technical Bases Report, useable estimates for PCS escape fractions were presented only for elemental iodine and those estimates were presented for only the few accident sequences considered. The escape fractions presented for CsI (Table C.2) did not include some aerosol effects. No overall escape fraction estimates were presented for aerosols, although ranges of retention for those species were seemingly indicated (Figure C.1). (The total aerosol escape fractions can be estimated from the presented results only by a subjective weighting of the values given for the various unspecified steam velocities.)

Table C.2. Summary of TRAP predictions of iodine distribution at the end of the accidents considered[a]

Case	t_f (s)	I_2 released to containment[b] (%)	CsI released to containment (%)		CsI retained in primary (%)	
			Vapor[c]	Suspended particles[d]	Deposited vapor[e]	Deposited particles[f]
TMLB´-1	1320	>99	0.4	92.6	6.2	0.7
TMLB´-2	1320	>99	0.4	92.8	6.3	0.5
TMLB´-3	1320	>99	4.3	86.1	9.3	0.3
TMLB´-4	1320	>99	22.5	40.2	37.1	0.1
AD-1/2	600	>99	1.3	80.2	18.0	0.3
AD-1	900	>99	10.8	70.7	16.6	1.6
AD-2	900	>99	11.5	53.7	33.9	0.6
AD-3	900	>99	12.4	26.2	61.1	0.1
AD-4	600	>99	11.3	22.8	51.9	13.8
AD*[g]	800	>99	86.1	13.6	0	0.1
TC	3025	95	8.6	44.6	45.9	0.7
AE	6050	>99	24.2	64.6	10.6	0.3

[a]Based on Table 6.1 in the Technical Bases Report (Gieseke and Kuhlman, 1981).

[b]Percent of I_2 mass released from fuel which escapes to containment.

[c]Percent of CsI mass released from fuel remaining in vapor state.

[d]Percent of same deposited on surfaces of suspended particles.

[e]Percent of same deposited on primary system surfaces from vapor state.

[f]Percent of same deposited on system surfaces via particle deposition mechanisms.

[g]The asterisk denotes a hot-leg break.

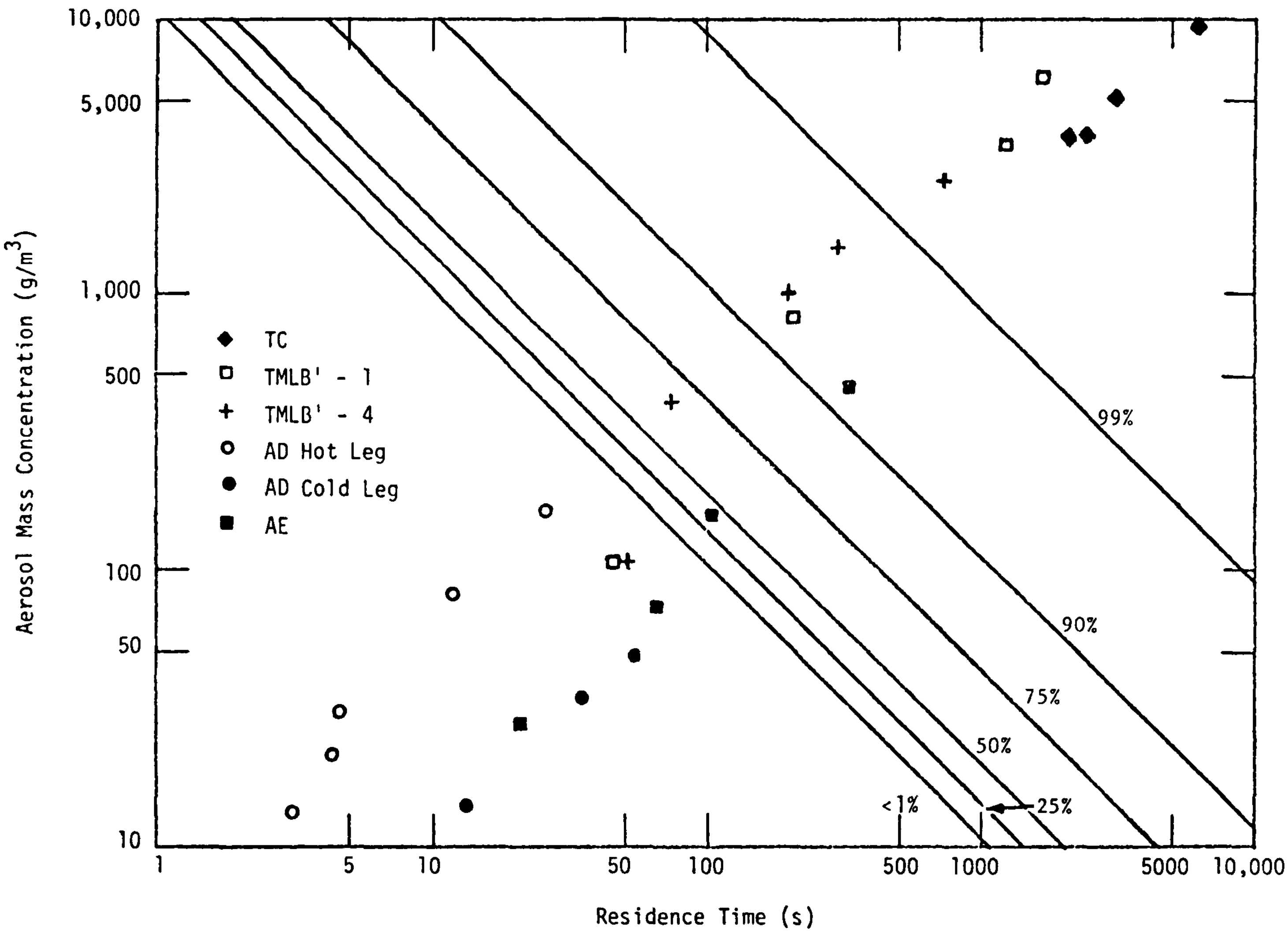

Fig. C.1. Aerosol mass concentration reduction predicted by QUICK code for initial mass concentrations and residence times (lines). Points represent TRAP calculations for various sequences. (This is Figure 6.3 of the Technical Bases Report [Gieseke and Kuhlman, 1981].)

Although one might use the information presented in the Technical Bases Report to estimate such escape fractions, the direct use of the results of the calculations presented in that report for describing retention in the PCS is questionable at least for the following reasons:

1. the volume of a large part of the RPV was ignored and thus the initial concentrations in the PCS were possibly overestimated in the calculations;
2. the aerosol generation rates were probably too high* for relatively long sequences like TC and thus the initial aerosol concentrations in the PCS were overestimated;
3. the material released during slumping of the core into the bottom of the RPV was ignored and thus the fractions escaping from the PCS were underestimated for some sequences;
4. the material released at RPV melt-through was ignored and thus the fractions escaping were underestimated for some sequences;
5. the time-dependent nature of the composition of the aerosol mass in the PCS was neglected and thus the escape fractions for some species were underestimated for some sequences; and
6. the deposited aerosols were not permitted to be reevolved after their initial deposition and therefore the fractions escaping were underestimated for some sequences.

In addition, the TRAP calculations in the Technical Bases Report utilized thermal-hydraulic input derived from MARCH (Wooton and Avci, 1980) calculations. Unfortunately, there are many problems associated with the use of MARCH for considering retention in the PCS because MARCH was developed to aid in considering transport of radionuclides within the containment and was not developed for considering transport through the

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*It should be noted that the aerosol release rates are highly uncertain. Some work indicates that the aerosol release rates used in the Technical Bases Report might be too low for many, if not all, sequences. Other work indicates that the release rates used might be too high.

PCS. Furthermore, some other problems always associated with the use of MARCH (Rivard, 1981) are associated with any MARCH-based TRAP calculations.

Another problem associated with the direct use of the results in the Technical Bases Report arises because some of the estimates obtained in that report were very sensitive to some poorly determined parameters. For example, the retention estimates for the two specific sequences investigated as a function of thermal-hydraulic conditions and/or aerosol characteristics were extremely sensitive to some small changes in the assumed descriptions. Unfortunately, large uncertainties exist in both the thermal-hydraulic and the aerosol descriptions for any prescribed accident scenario. In spite of all these shortcomings, the calculations in the Technical Bases Report still provide valuable insights into the possibilities of the effects of various accident conditions on vapor and aerosol retention in the PCS.

C.4 APPROACH FOR ESTIMATING RADIONUCLIDE RELEASES FROM THE PRIMARY COOLANT SYSTEM

The work presented in the Technical Bases Report, and described in the last subsection, represents the state-of-the-art for considering radionuclide retention in the PCS during LWR accidents.* Unfortunately, it is obvious from reviewing that work that there is no approach which currently can be used to provide definitive answers regarding retention during transport through the PCS.

The basic approach taken here was to utilize the results of previous calculations (namely, those presented in the Technical Bases Report) to develop an understanding of the possible extent of retention in the PCS for various plausible accident conditions. This understanding, along with consideration of phenomena neglected or not treated in the previous calculations, was used as the basis of the simplified assumptions presented here.

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*Much work extending the theoretical state-of-the-art is currently underway at Battelle Columbus Laboratories. Related experimental work is being performed at both Battelle Columbus Laboratories and Sandia National Laboratories.

C.4.1 Background

C.4.1.1 Volatility

The physical state of any species in the PCS during an accident would depend on the temperatures encountered there. For any accident in which boiloff of the coolant from the core was initiated, the temperatures would be relatively high, especially close to the core. The temperatures in the rest of the PCS, most importantly, along the path(s) followed by the radionuclides to reach the containment, would depend upon both the accident and the path. Locations farther from the core region would tend to be cooler than the core region. In general, accidents involving more extensive core damage would result in greater heating of at least some parts of the PCS.

Very volatile species, such as noble gases, would be transported through the PCS as gases regardless of the conditions encountered. Moderately volatile species would move as vapors in very hot regions but could condense onto particles and surfaces in cooler regions. If the steam temperatures were higher than the PCS surface temperatures, condensation of such species would tend to be onto surfaces; conversely, if the steam temperatures were lower than the surface temperatures, the condensation would tend to be onto aerosol particles. However, the relative amounts of such moderately volatile species condensing onto PCS surfaces and onto aerosols would also depend upon the relative surface areas of those two substrates, with greater condensation tending to occur onto the substrate with the larger area if the temperatures were the same. If PCS surfaces heated up as the accident progressed, condensed material could be revolatilized and moved to cooler regions and even transported out to the containment. Alternatively, if the surfaces heated up, the condensed material might react with the surfaces and be retained. Inasmuch as less volatile species generally would be rapidly condensed into aerosols and would move through most of the PCS as such, their fates would be determined almost entirely by that of the aerosol mass.

Besides condensing onto surfaces, many species present as either vapors or gases could chemisorb onto, or otherwise react with, surfaces in the PCS. The effects of chemisorption would be analogous in many ways to the effects of condensation except that the effects of chemisorption would not always be reversible and so could result in permanent retention on the internal surfaces of the PCS.* Likewise, chemisorption might result in incorporation into aerosol particles and thereby affect behavior in the PCS.

The environment encountered by the radionuclides would depend partly on the pathway through the PCS as determined by the accident scenario. For example, for a large hot-leg pipe break in the primary coolant system in a PWR, the path of the radioactivity through the coolant system would be relatively short. Materials would pass through the upper plenum of the reactor vessel and from thence exit through the pipe break into the containment. In contrast, for a transient-initiated event in a PWR, the path of the radioactivity might be through the upper plenum, then through the pressurizer and finally through the pressurizer quench tank. Whereas the relatively short, hot path of the radionuclides for the described pipe break accident would present scant resistance to flow out of the PCS for many radionuclides, the relatively long path for the considered transient might permit the condensation of large quantities of even more volatile radionuclides in the relatively cool pressurizer.

C.4.1.2 Aerosol Concentration

The aerosol concentration in the PCS would depend upon the size of the reactor core and its status at the time of accident initiation, as well as on the extent of core damage. Furthermore, it would depend upon the rate of movement of the released materials through the PCS. Thus a wide range of concentrations would be possible for various accidents.

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*For example, preliminary experimental work at Sandia National Laboratories on interactions of tellurium with metal (Inconel and 304 stainless steel) surfaces indicates that tellurium may be rapidly and strongly chemisorbed to such surfaces at least at temperatures in the range of 500-800°C. The species formed appear to be relatively stable metal tellurides (Sallach, 1982). Thus tellurium may be "permanently" sorbed to certain PCS surfaces during some accidents.

Because the rates of aerosol agglomeration and deposition increase disproportionately with increasing aerosol concentration, and because the possible aerosol concentrations in the PCS range over orders of magnitude, concentration-dependent effects could result in quantitatively much different aerosol behavior in the PCS for various accidents. In particular, for high aerosol concentrations, the possibility of sizeable deposition of aerosols in the PCS would exist, whereas for low aerosol concentrations, the deposition of aerosols in the PCS would generally be very small. (A notable exception would be the deposition of a possibly sizable fraction of the relatively large aerosol particles released at the time of cladding rupture.)

One of the critical issues determining whether aerosol processes could result in substantial retention in the PCS would be the time required for aerosol deposition processes to be important relative to the time needed for aerosol transport through the PCS. If the anticipated lifetime of a typical particle in an aerosol were long relative to its residence time in the PCS, little effect would be expected.

"Maximum" possible aerosol concentrations in the PCS for various postulated total aerosol releases are given in Table C.3. For comparison, the initial aerosol "half-lives" for a wide range of initial aerosol concentrations are given in Table C.4. As can be seen by comparing the various values in the two tables for some possible releases of aerosols into the PCS, the impacts of aerosol deposition could potentially be substantial. However, in general, the effects would be much less than those indicated by the maximum concentrations given in Table C.3 because both the gradual release of aerosols into the PCS and the flow of steam plus hydrogen through the PCS during the accident would result in much lower concentrations in the PCS than the maximum possible ones.

C.4.1.3 Steam Velocities

As was noted previously, steam flow usually would be the primary vehicle for transport of radionuclides from the core region to the containment. Periods of substantial steam generation, and therefore of potentially high steam velocities, could occur during at least the following activities: the blowdown of the coolant from the PCS after a

Table C.3 Maximum aerosol concentrations in the primary coolant system of typical reactors

Aerosols released (kg)	Maximum concentration $(g/m^3)^{a,b,c}$	
	PWR	BWR
20	100	30
100	500	140
500	2500	700

[a]The volumes assumed are 200 m^3 and 700 m^3 for the coolant system of a PWR and a BWR, respectively.

[b]By comparison, a very dense aqueous fog contains approximately 1 g/m^3.

[c]It must be noted that these values assume instantaneous dispersal of the aerosols throughout the PCS. In reality, the aerosols would not all be released simultaneously. In addition, the aerosols generally would not be dispersed rapidly throughout the PCS but instead often would be transported through the PCS in a fashion more appropriately described by laminar flow (of the carrier steam and hydrogen).

Table C.4 Approximate initial aerosol "half-lifes" for a typical aerosol[a]

Initial concentration (g/m^3)	"Half-life" (sec)[b]
3500	6
350	60
35	600
3.5	6000[c]

[a]These values are based on a specific set of QUICK calculations performed for NUREG-0772 (Kuhlman, 1981) but are thought to be adequately representative. The initial conditions for those calculations were as follows: r_g (median radius) = 0.1 μm, σ_g = 2.0, pressure = 171 atmospheres, temperature = 550 °C, and particle density = 10 g/cc. (The actual aerosol density for any severe core-damage or meltdown accident would probably be closer to 5 g/cc.) The initial total particle numbers were 1×10^{10} cm^3, 1×10^9 cm^3, and 1×10^8 cm^3 for the initial mass concentrations of 3500 g/m^3, 350 g/m^3, and 35 g/m^3, respectively.

[b]The "half-life" is defined to be the time it takes for the initial aerosol mass to decrease by one-half; after the initial rapid decrease, the effective half-life slows down, with the half-life at any time being essentially a function of the remaining suspended mass concentration.

[c]Estimate by extrapolation.

loss-of-coolant accident (LOCA); the operation of the coolant-system depressurization system (in a BWR); the boiloff of the water during heatup of the core; the cooling or quenching of the core by partial or delayed functioning of the emergency core cooling system; and the slumping of the core into the bottom of the RPV during a meltdown. In addition, high velocities could occur just after RPV melt-through, especially if the meltdown were at high pressure. The effectiveness of the steam in carrying radionuclides from the core region to the containment would depend upon the relative timing of the period(s) of high steam velocities and the periods of significant radionuclide releases from the core materials.

During most meltdown accidents, after the initial blowdown of the coolant system, the generation of steam would be driven primarily by the decay heat of the core. Thus the rate of steam production at the onset of melting would be the same for most accidents, typically to within a factor of two. In contrast, the velocity at which steam moved through the PCS just after blowdown would be highly dependent upon the system pressure and could vary by at least an order of magnitude for accidents with comparable steam mass generation rates (Denning, 1982).

The rate of steam production later in the accident would depend in part upon the fraction of the core still covered by water and this would, in turn, be highly accident dependent. Although the steam generation rates frequently would be relatively high early in any accident, as the fraction of the core covered by water decreased, the rate of steam generation would often diminish substantially. However, later as the core melted, steam generation rates would be increased by dropping of portions of the core into the bottom of the reactor vessel. In addition, the coolant system pressure could vary widely over the course of the accident. Consequently, the associated steam velocities in the PCS could change substantially during an accident due to changes in both the steam generation rate and the system pressure (Denning, 1982).

In addition to the large potential distribution of steam velocities within any given accident, there would be a wide range of "average" steam velocities associated with different basic "types" of accidents. For meltdown accidents involving intermediate and large LOCA's, the average steam velocities often would be sufficiently high that there would not be time for substantial aerosol retention in the PCS.* For meltdown accidents involving small LOCA's, the steam velocities would depend upon the size of the break, as well as on the performance of the emergency core cooling system. Frequently, average steam velocities would be an order of magnitude smaller than those for intermediate and large LOCA's. For transient-initiated accidents, the steam velocities likewise would depend upon the accident conditions. For transients involving stuck-open relief valves, the average steam velocities often would be comparable to those for small LOCA's. For other transients, the average steam velocity in the reactor coolant system typically would be somewhat smaller (Denning, 1982). Consequently, in some meltdown sequences initiated by either small LOCA's or transients, there could be sizeable retention of some species as the result of long residence times in the PCS.†

Because radionuclides of high volatility would tend to be released from the core materials relatively early in any accident,§ in accidents involving long residence times in the PCS, such radionuclides would often have more time to be deposited before their carrier steam exited from the PCS. However, such radionuclides would also have more time to

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*It should be recognized that there are many exceptions to most of the generalizations presented here. For example, in a BWR large LOCA with no emergency core coolant system injection, there might be little steam generation during the early stages of core melting.

†A related potential problem which might affect retention in at least certain accidents would be the plugging by aerosols of portions of certain pathways through the PCS. Such plugging could cause material to follow pathways to the containment which typically would not be thought of as being important. Substantial amounts of aerosols might be retained in the PCS even by partial plugging of dominant pathways. Severe plugging of the PCS generally would be precluded by the threat of overpressurization of that system.

§Because the core would heat up unevenly, radionuclides of differing volatilities would be released simultaneously from various portions of the core. Thus, this description of the releases somewhat simplified the actual timing of those releases. (See the discussions in Appendix B.) However, the overall description approximately would follow that presented here.

be revolatilized and potentially released to the containment. In contrast, radionuclides of low volatility would tend to be released from the core relatively late in any accident. In meltdown accidents involving long residence times in the PCS, those radionuclides frequently would have little time to be deposited before slumping and/or RPV melt-through could flush those materials from the system. Thus, disproportionately large fractions of the less volatile radionuclides might escape from the RPV in any accident with relatively long aerosol residence times in the PCS.

C.4.1.4 Standing Water

The extent of scrubbing of any radionuclides by water encountered in the pathway through the PCS would depend on factors including the depth and the temperature of the water. For deeper water and possibly for unheated water, scrubbing would be greater. (It is currently a matter of some debate whether heated water would be as effective in scrubbing aerosols as cooled water would be. See the footnote in Subsection D.2.2 of Appendix D.)

For core-melt accidents, the primary possibilities for water in the pathway would be water in the RPV of a BWR due to improper coolant circulation in that vessel (see the discussion in Subsection C.3.1) and water in the pressurizer quench tank of a PWR for certain transient-initiated accidents. In both cases, water would not be expected to be present throughout the melting. And even if it were present throughout an accident, it would probably be boiling by the end of the melting (Denning, 1982) and therefore possibly less effective in mitigating the amounts of most radionuclides escaping to the containment.

For limited-core-damage accidents, water could always be present in the pathway through the PCS during some portion of the accident. That is, by definition, accidents involving only limited damage would require that adequate cooling were established before complete melting occurred. The extent of scrubbing would depend upon the details of the accident scenario.

In many cases in which radioactivity entered the water, the activity could eventually enter the containment via that water. More volatile species, such as noble gases and elemental iodine, could then escape to the containment atmosphere by volatilization. (Often, however, substantial conversion of elemental iodine to ionic species would occur rapidly, thus preventing volatilization of large amounts of iodine. In addition, it is questionable whether much iodine would be present even initially in elemental form.)

In general, negligible quantities of ionic species, such as CsI, would be volatilized from the water. However, limited amounts of such species might be released to the containment atmosphere by other processes. For example, ionic species could be released to the atmosphere by "bubble busting" if the water containing them boiled or if gases bubbled through that water (Powers, 1982).*

C.4.1.5 Total Aerosol Mass

The fraction of aerosols escaping from the PCS in any accident would be very dependent on both the accident scenario and the reactor design. In addition, it would depend on the detailed composition of the core and the associated control-rod and structural materials. If other conditions were equivalent, retention as the result of aerosol effects generally would be larger for accidents involving greater generation of aerosols.

Reactors vary drastically in the size and status of their cores† and thus in the masses of fission products which could be released and form aerosols during any accident. These variations could result in the total masses of fission product aerosols generated during equivalent meltdown accidents at different reactors ranging from 10's of kg's to 100's of kg's.

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*More exactly, when any bubble burst, it would throw off small amounts of liquid which could form a fog. As the liquid evaporated, formerly-dissolved solid material could form aerosol particles. In addition, particles suspended in the water could be released to the atmosphere that way.

†See the related discussion in Subsection D.4.2.5 in the next appendix.

Reactors also vary widely in the amounts of non-fission product materials which could form aerosols. These variations could result in even larger differences in the masses of aerosols generated in equivalent accidents at different reactors.* The differences in the potential non-fission-product aerosol releases for meltdown accidents range from 10's of kg's to 1000's of kgs. As a result, the aerosol concentrations in the PCS could vary greatly among different reactors for even similar accident scenarios.

C.4.2 Assumptions

The assumptions presented here are based on both the foregoing qualitative discussions and the results of the calculations presented in the Technical Bases Report. It should be recognized that these assumptions are drastic simplifications of reality. They have been made so that at least some estimates of retention in the PCS can be made prior to the completion and application of more sophisticated procedures such as that involving the use of an improved version of TRAP.

C.4.2.1 Volatility

Following previous work, it has been assumed that all radioactive species can be partitioned into three basic groups for consideration of transport through the PCS. Specifically, those three groups are the following: (1) very volatile species (such as noble gases and elemental halogens) which would transport as gases; (2) moderately volatile species (such as CsI and tellurium) which could transport as either vapors or aerosol particles, depending upon the temperatures; and (3) less volatile species (such as strontium, ruthenium, and lanthanum)

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*For example, some reactors have control rods composed mostly of silver whereas other reactors have control rods composed primarily of boron carbide. Inasmuch as the control rods form a large mass and inasmuch as a large fraction of the silver might be volatilized while not much of the B_4C would be released, there could be substantial differences in the amounts of aerosols generated during meltdown accidents in different reactors. (It should be noted that there is a large uncertainty associated with the release estimates for silver.)

which would transport as aerosols. (In a more refined treatment than that considered here, the group of moderately volatile species might be further divided into two subgroups: (2A) CsI and other species which would tend to be released relatively early from the core in any meltdown accident; and (2B) tellurium and other slightly less volatile species which would tend to be released throughout the period the core was still in the RPV.)

It has been assumed that significant retention in the PCS of species present as gases or vapors would not occur for any meltdown accident. This assumption has been employed because it is difficult to make valid assumptions about gas and vapor retention which can be simply and reasonably quantified on the basis of the current state-of-the-art of PCS transport. Such retention could be expected to be highly species dependent for many accident sequences, with estimates of retention sometimes depending critically on the details of the predicted thermal-hydraulic conditions in the PCS. The only exception was assumed to be retention of elemental halogens in the PCS water in some accidents (see Subsection C.4.2.4).

In contrast, it has been assumed that significant retention in the PCS of aerosols species could occur for some meltdown accidents. Unlike the situation for gas and vapor retention, it is somewhat easier to make assumptions about aerosol retention which can be readily and meaningfully quantified. Such retention generally is much less species dependent, with estimates of aerosol retention usually not being overly dependent on the details of the predicted thermal-hydraulic conditions.

C.4.2.2 Aerosol Concentration

It has been assumed that aerosol concentrations in the vicinity of the melt would not be effectively lowered by aerosol processes because any deposited material might fall back into the very hot region around the melt and be revolatilized. Consequently, it has been assumed that substantial net removal would not occur until the aerosols had reached at least the upper portion of the RPV. After that point, the maximum concentration of aerosols in the RPV has been taken to be limited by deposition processes.

C.4.2.3 Steam Velocities

It has been assumed that the steam velocities would be largely a function of the basic type of accident, e.g., large LOCA, small LOCA, and so forth. Furthermore, it has been assumed that for meltdown accidents, at a minimum all the aerosols in the RPV at the times of both slumping and RPV melt-through would be released into the containment. The amounts of aerosols in the RPV at those times have been taken to be equal to the product of the fraction of the PCS occupied by the RPV and the total amounts of aerosols in the PCS at those times. In addition, it has been assumed that aerosols in the PCS at the time of any substantial steam velocities could also escape to the containment, with the steam velocities depending on the specific accident scenario considered.

C.4.2.4 Standing Water

For meltdown accidents, it has been assumed that standing water would typically not occur in the pathway through the PCS for any BWR accident sequence.* However, it has been assumed that water might be present in the pressurizer quench tank for PWR transient sequences. Furthermore, it has been assumed that if water were present in the tank, its maximum effect would be equivalent to a decontamination by a factor of 10 of approximately 80% of the material released into the RPV for a meltdown accident.

For those accidents involving partial melting of the core, it has been assumed that an infinite variety of conditions would be possible, ranging from those with all releases except leach releases moving along an entirely "dry" path through the PCS to those with extensive scrubbing of most of the non-leach releases into water in the PCS. For accidents involving only "gap" releases, it has been assumed that all the radionuclide releases would be directly into PCS water.

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*The suppression pool is not part of the PCS. Its potential effects are considered in the next appendix.

C.4.2.5 Total Aerosol Mass

It has been assumed for meltdown accident sequences that approximately 200 to 400 kg's of aerosols, or greater amounts, would be generated prior to RPV melt-through. For accidents involving lesser core damage, correspondingly smaller amounts of aerosols have been assumed.

C.4.2.6 Net Escape Fractions

The assumed net PCS escape fractions, presented in Tables C.5 and C.6, have been based on the general considerations just outlined. In those tables, both plausible ranges of the PCS escape fractions and "best estimates" are indicated.

The "best estimates" of the PCS escape fractions have been chosen, perhaps somewhat conservatively, as being equal to the upper ends of the indicated ranges. This has been done primarily for two reasons. First, there are large uncertainties associated with the estimated escape fractions for any given accident at any specific reactor. Second, there are even larger uncertainties associated with the estimated escape fractions for different reactors. Because of all these uncertainties, what is "realistic" is not always obvious.

To put these estimates in perspective, it should be noted that although the "best estimate" values given here often might be conservative, they would not necessarily always be so on the basis of today's understanding of retention in the PCS. In addition, they are not as conservative as the values currently used for regulatory guidance. (See Chapters 4 and 5.)

C.5 UNCERTAINTIES AND LIMITATIONS

Obviously, the uncertainties in the estimates of the PCS escape fractions are large. Currently, no set of models, and therefore no computer code, exists for accurately treating vapor and aerosol retention in the PCS for all possible LWR accident conditions. At best, the present models and/or codes can be used only to gain insight into the

Table C.5 Summary of primary coolant system escape fractions

Type of Accident	Escape fractions[a]			
	Xe	I	CsI; Cs; Te	Sr; Ru; La
A. Accident sequence without water in the pathway through the PCS				
Large LOCA's	1.0	1.0	1.0[b](0.8-1.0)	1.0(0.9-1.0)
Small LOCA's	1.0	1.0	1.0(0.3-1.0)	1.0(0.5-1.0)
Transient-initiated LOCA's	1.0	1.0	1.0(0.3-1.0)	1.0(0.5-1.0)
Other transients	1.0	1.0	-[c](0.1-1.0)	-[c](0.1-1.0)
Accident involving "gap" release only	1.0	1.0	1.0[c](0.1-1.0)	1 0(0.1-1.0)
Accidents involving partial melting	1.0	1.0	1.0[c](0.1-1.0)	1.0(0.1-1.0)
B. Accident sequences with water in the pathway through the PCS[e]				
Meltdowns	1.0	0.1(0.01-1.0)	0.1(0.01-1.0)	0.1(0.01-1.0)
Accidents involving "gap" release only	1.0	0.1(0.01-1.0)	0.1(0.01-1.0)	0.1(0.01-1.0)
Accidents involving partial melting	1.0	0.1(0.01-1.0)	0.1(0.01-1.0)	0.1(0.01-1.0)

[a]Within each indicated range, for aerosols the smaller estimated escape fractions generally correspons to relatively high net aerosol generation rates, small-size reactor vessels, low steam velocities through the PCS and short flow paths to the containment. Likewise, in each range for species of intermediate volatility, the smaller estimated escape fractions correspond to relatively low temperatures along the path(s) through the PCS and/or high chemical reactivities in the PCS, low steam velocities through the PCS, and short flow paths to the containment.

[b]1.0 is assumed due to the sensitivity of both the calculated results to the postulated thermal-hydraulic conditions and the amount escaping to possible pathway conditions for even a given accident sequence; deposition of intermediate volatility compounds such as these results primarily from vapor deposition.

[c]The escape fractions for these sequences are highly dependent on the rate of flow of steam through the PCS during the accident. See Table C.6 for the details.

[d]There is a possibility of substantial vapor deposition of intermediate volatility elements due to relatively low PCS temperatures; such deposition would be highly dependent on the pathway conditions.

[e]Obviously, the amount of retention in the PCS water would depend upon the path taken by the radionuclides that is, upon the amount and temperature of the water encountered

Table C.6 Primary coolant system escape fractions for transient-initiated meltdowns

Type of accident sequence	Escape fractions[a]			
	Xe	I	CsI; Cs; Te	Sr; Ru; La
TMLB´, TML (PWR)[b]	1.0	1.0	1.0 (0.8-1.0)	1.0 (0.9-1.0)
TC (BWR)	1.0	1.0	0.33[c] (0.10-0.33)	0.33[c] (0.10-0.33)
TW (BWR)	1.0	1.0	0.33[c] (0.10-0.33)	0.33[c] (0.10-0.33)
TQUV (BWR)	1.0	1.0	0.67[c] (0.33-1.00)	0.67[c] (0.33-1.00)

[a]The values given in parentheses denote ranges of reasonable values for escape fractions.

[b]The values given here do not include the effects of possible scrubbing by water in the pressurizer quench tank (see Table C.5).

[c]This value is probably high; it was assumed due to the large uncertainties in both the PCS retention estimates and the aerosol generation estimates.

possibilities.* In no sense can such treatments appropriately be utilized in a predictive mode.

For many accidents, it does not seem likely that retention of radionuclides in the PCS would be substantial. However, for some other accidents, it seems reasonable that such retention could be large, at least for certain radionuclides. Unfortunately, it is not possible to quantify that retention with any degree of certainty. Indeed, it is not even always possible to quantify the uncertainty to any useful degree of resolution. Thus, the accidents involving potentially sizable retention of radionuclides in the PCS currently pose a significant problem for source term estimation. Some of the major sources of uncertainty in considering radionuclide transport through the PCS, along with their potential impacts on source term estimates, are summarized in Table C.7.

One of the most important contributors to the uncertainty for the consideration of retention of aerosols in the PCS is the uncertainty associated with the estimation of the total amount of aerosols initially released into the PCS. Any estimate of aerosol retention in the PCS is crucially dependent upon the estimates of the total amount of aerosols generated before RPV melt-through. Unfortunately, this latter quantity is not known to within a factor of 5 even for any given accident at a particular reactor. The variation among reactors and different possible accidents is much larger. The result is that a large uncertainty must be associated with many values proposed in this or any other report for aerosol retention in the PCS. The effect of the total mass generated on the fractions retained would be expected to be greatest for those accidents in which aerosol residence times in the PCS would be relatively high.

An important source of uncertainty for consideration of retention of some vapors and gases in the PCS is the lack of detailed knowledge about the chemistries of those species. Inasmuch as retention of any

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*Although the recently improved version of TRAP can be used to obtain much more refined estimates of vapor and aerosol retention in the PCS than are currently available, any such set of calculations will still be handicapped by the uncertainties mentioned in this section.

Table C.7 Some major sources of uncertainty in primary coolant system transport consideration

Problem area	Effect	Problems in data	Problems in models or codes	Potential magnitude of effect on individual sequences	
				Limited-core damage accident	Meltdown
High aerosol concentrations	High aerosol concentrations are agglomerated and deposited much more rapidly than low concentrations	Aerosol generation rates are highly uncertain	TRAP does not include gravitational agglomeration or settling,[a] QUICK does not include vapor processes, QUICK is a single-compartment containment model	Large for accidents with low steam velocities	Large for accidents with low steam velocities
High steam concentrations	Condensing steam environments accelerate aerosol deposition	Data not available for condensing steam models	TRAP does not include condensation of steam	Large if temperatures in the PCS are low enough	Small
Scrubbing	Radionuclides (other than noble gases) passing through water, e.g., through water in the pressurizer quench tank, can be scrubbed	Availability of water not known; scrubbing data not generally available	TRAP does not include scrubbing	Large if water present in path through PCS	Large if water present in path through PCS
Timing of radionuclide releases to the PCS	Radionuclide releases just prior to or during large releases of gases from the PCS are more likely to escape to the containment	Radionuclide release rates are from the core materials are uncertain	All aerosols are lumped together in TRAP and most aerosol codes	Large for accidents with low steam velocities	Large for accidents with low steam velocities
Thermal-hydraulic conditions	Very low gas velocities permit aerosol deposition in the PCS; low temperatures permit condensation of vapors onto surfaces and particles	Much needed data not available	MARCH is based on simple models; RELAP and TRAC are not appropriate after uncovering of the core	Large	Large
Chemical form	Less volatile species transport as aerosols; more volatile species transport as gases	Much needed chemical data not available for many elements	Only TRAP considers phase changes	Large	Large
Chemical reactions	Reactions change chemical forms of some species; reactions with surfaces can affect retention	Much needed chemical data not available for many elements	No codes consider chemical reactions	Large	Large

[a]TRAP now includes gravitational agglomeration and settling.

relatively volatile species, other than a noble gas, would depend strongly on its interactions both with PCS surfaces and with other species, a large uncertainty is necessarily associated with estimates of retention for any species with potentially substantial interactions. Further uncertainty exists in the estimates of retention of some vapors and gases because those estimates depend in turn on the estimates of the thermal-hydraulic conditions in the PCS. Those estimates are also highly uncertain, especially for accidents involving complete meltdown of the core.

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APPENDIX D

TRANSPORT THROUGH THE CONTAINMENT

D.1 INTRODUCTION

After radioactivity was released into the containment, it would be transported within the containment and could subsequently be released to the outside environment. Those radionuclides which remained inside the containment would constitute a potential hazard to the equipment within the nuclear plant while those radionuclides which were released to the outside would constitute a potential hazard to the human population residing near the plant. Both the factors affecting the radionuclide transport within the containment and those factors affecting the amounts of various radionuclides which eventually would be released from the containment are considered in this appendix. In particular, considerations of all those factors are used to estimate the magnitudes of the containment-related portions of the source terms for various types of accidents for both equipment qualification and emergency planning purposes.

The various processes and phenomena which could affect radionuclide transport in the containment are outlined in Section D.2 of this appendix. The state-of-the-art for considering radionuclide transport within the containment and for estimating releases to the outside environment is reviewed in Section D.3. The approach used in this report both for considering that transport and for estimating those releases is discussed in Section D.4 and the results of using that approach for various reactor accident sequences are presented in Section D.5. The uncertainties associated with the containment-related portions of both the source term estimates given here and source term estimates in general are considered in Section D.6. The detailed results of previous studies which have been used to obtain the estimates presented here are contained in the addendum of this appendix.

An abbreviated description of the approach used in this appendix to estimate containment escape fractions is given in Section 3.3 of Chapter 3. The results of this appendix and their implications are summarized in Chapters 4 and 5. The results of this appendix, in combination with those of the previous two appendices, are also reviewed in Chapters 4 and 5 and are presented in detail in Appendix E.

D.2 DESCRIPTION OF PROCESSES

Materials entering the containment atmosphere would be subject to removal by both natural processes and processes resulting from engineered factors. During any accident, such processes could reduce substantially the amount of material in the containment atmosphere and, hence, the amount of radioactivity which could be released to the environment.

In this section, the basic processes which affect the movement and the reduction of materials in the containment atmosphere are reviewed briefly. For a more complete review of the natural processes and their effects, the reader might consult either Baybutt (1981), Gieseke et al. (1981), or Nuclear Energy Agency (1979). For a more comprehensive treatment of the effects of engineered features, the reader might review Pasedag, Postma, and Adams (1981).

D.2.1 Natural Removal Processes

During an accident, radionuclides could enter the containment either indirectly after transport through the primary coolant system (PCS) or else directly after escape from the core materials released to the containment by reactor pressure vessel (RPV) melt-through. Gases, vapors, and particulates all would be included among the released materials. Those materials could be either airborne or waterborne or both when they entered the containment. The amount of radioactive and other materials entering the containment, the rates at which they entered it, and the fluids containing those materials would depend upon the accident conditions. Although for minor accidents the amounts entering the containment would be very small, for some accidents the amounts would be substantial.

The natural removal processes undergone by the gases and vapors would be basically the same as those experienced in the primary coolant system. They would include sorption and condensation onto surfaces and particles, condensation into aerosol particles, chemical reactions with surfaces and other species in the atmosphere, and dissolution in any water present. The natural removal processes undergone by the particulate matter would include agglomeration into larger particles (by processes such as Brownian, gravitational, and turbulent coagulation) and subsequent removal of particulates from the containment atmosphere (by processes such as gravitational, diffusional, thermophoretic, and diffusiophoretic deposition). In all cases, processes would occur which would partially counteract some of these removal processes. For example, condensed vapors could be reevaporated and deposited particles could be resuspended. Thus, the concentration of materials in the containment atmosphere typically would be a complex function of the many processes which would take place. The relative importance of these processes in any accident would depend upon the details of the accident scenario. The more important of the natural processes and their effects on the post-accident concentrations of radionuclides in the containment atmosphere are summarized in Table D.1.

The more volatile species* typically would be present in gas or vapor form and could interact with other species in the containment. In contrast, the less volatile species generally would reside in or on aerosol particles and would not be available for interactions, except within the particles. Consequently, the behavior of the more volatile species usually would be governed by their reactivities in the containment while the fate of the less volatile (intermediate and low volatility) species would be determined by the behavior of the total aerosol mass in the containment. Inasmuch as most of the radionuclides other than the noble gases, and perhaps some of the halogens, could form aerosol particles in the containment, the post-accident behavior of the majority of radionuclides could be determined by the overall aerosol behavior.

Typically, in any aerosol, newly formed particles agglomerate into larger particles. Because agglomeration into heavier particles tends to hasten removal of the material from the atmosphere, the factors which affect agglomeration affect the history of the aerosol

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*The volatility of any radionuclide would be a function of its chemical form(s). For example, iodine present as elemental iodine would be much more volatile than iodine present as cesium iodine.

Table D.1. Effects of natural processes in containment during accidents involving core damage

Process	Species	General effects	Effects on radionuclide concentrations in containment atmosphere	
			Limited-core-damage accidents	Meltdown accidents
Agglomeration (Brownian; gravitational; turbulent)	Aerosols	Increases particle weights	The low aerosol concentrations would not be rapidly decreased by agglomeration and deposition.	The high aerosol concentrations generated during some accidents could be rapidly decreased by agglomeration and subsequent deposition.
Steam condensation	Aerosols	Increases particle weights; compacts individual particles.	The low aerosol concentrations would be more rapidly decreased by condensing steam.	The high aerosol concentration generated during some accidents would be more rapidly decreased in high steam concentrations, especially if accompanied by cooling.
"Condensation" onto particles (evaporation from particles) (condensation; adsorption...)	Gases; aerosols	Increases particle weights; decreases gas concentrations; changes aerosol composition.	Compounds not previously deposited in the PCS, especially ones of high volatility, could be deposited.	Compounds not previously deposited in the PCS, especially ones of high or intermediate volatility, could be deposited.[a]
Deposition (gravitational; diffusional; thermophoretic; diffusiophoretic)	Gases; aerosols	Decreases gas or aerosol concentration.	The low aerosol concentrations would not be rapidly decreased by agglomeration and deposition.	The high aerosol concentrations generated during some accidents could be rapidly decreased by agglomeration and subsequent deposition.
Resuspension	Aerosols	Increases aerosol concentration	Probably not important.	Probably not important.
"Condensation" onto surfaces (condensation; adsorption...)	Gases	Decreases gas concentration	Compounds not previously deposited in the PCS, especially ones of high volatility, could be deposited.	Compounds not previously deposited in the PCS, especially ones of high or intermediate volatility, could be deposited.[a]

Table D.1. (continued)

Process	Species	General effects	Effects on radionuclide concentrations in containment atmosphere	
			Limited-core-damage accidents	Meltdown accidents
Electrostatic interactions	Aerosols	May slow down rate of deposition or increase rate of resuspension	Probably not important.	Probably not important.
Radioactive decay	Gases; aerosols	Changes chemical composition of gases and aerosols; heats aerosols	Chemical changes probably not important; aerosol heating may affect steam condensation[b]	Chemical changes probably not important; aerosol heating may affect steam condensation and aerosol deposition.[b]
Chemical reactions	Gases; aerosols	Changes chemical composition of gases and aerosols; may decrease concentrations	Potentially important[c]	Potentially important.[c]
Evaporation	Dissolved species	Increases gas concentration	Evaporation from water (and surfaces) after the accident could form a long-term source of some radionuclides	Evaporation from water (and surfaces) after the accident could form a long-term source of some radionuclides.

[a] Vapors of low volatility fission products released directly into the containment after RPV melt-through would typically condense into aerosols shortly after being vaporized from the melt.

[b] Powers (1982).

[c] A notable example is the set of reactions which result in the formation of organic iodides. Another example is the set of reactions which might occur due to hydrogen burning.

and, thus, could significantly affect radionuclide behavior in the containment during and after an accident. Agglomeration is enhanced by at least one basic condition which might be typical of many light water reactor (LWR) meltdown accidents, namely, high aerosol concentrations. The effects of agglomeration, and therefore those of aerosol-related removal processes, generally increase substantially in relative importance with increasing aerosol concentration.

Aerosol removal is also augmented by another condition which would be typical of many meltdown accidents, that is, a condensing steam environment. The effects of steam usually increase with increasing steam concentration. Consequently, accidents with both high aerosol concentrations and condensing steam might involve significantly enhanced rates of removal of the particles from the containment atmosphere.

Depending on their forms, radionuclides present in the containment atmosphere could experience any or all of the processes just mentioned. Similarly, radionuclides present in any water in the containment could undergo various processes. For example, those radionuclides could undergo chemical reactions with other materials in the water, as well as with various constituents of the reactor building itself. In addition, any radionuclides present in the water as gases could escape to the containment atmosphere via volatilization. Unless the water were to escape from the containment building, most of the radioactivity entrained in the water typically would not escape to the environment.

D.2.2. Effects of Engineered Safety Features

As the natural processes were occurring during any accident, the functioning of some engineered safety features (ESF's) could cause additional removal of both gases and particulates to occur from the containment atmosphere. The ESF's which would be most effective in accelerating radionuclide removal would be those involving containment sprays, filter systems, suppression pools, and ice-bed condensers. The more important effects of the ESF's on the radionuclide concentrations in the containment atmosphere are summarized in Table D.2. The basic types of nuclear power plants in which these various ESF's are found are reviewed in Table D.3.

D 2 Effects of selected engineered safety features during accidents involving core damage[a]

Engineered safety feature	Plant	Electricity required	Effects	Effect on radionuclides in containment during limited-core-damage accidents	Effect on radionuclides in containment during complete meltdown accidents
Containment leak-tightness	all	no	Prevents radioactivity in containment from reaching environment, leakage often permits some release	Retards release to environment, permitting removal and decay mechanisms to lessen the releases	Retards release to environment, permitting various removal and decay mechanisms to lower the radionuclide concentrations in the containment, severe early containment rupture substantially lessens benefits of containment
Containment spray system	PWR's (restricted to upper compartment in ice condenser PWR's)	yes	Reduces pressure in containment, removes radioactivity and heat from containment atmosphere, promotes radionuclide removal by both washout and acceleration of aerosol removal processes	Removes some aerosols[b] and some iodine[c,d]	Large aerosol concentrations would be acted on most effectively,[b] very high aerosol concentrations could clog spray system
Containment recirculation filter system	some PWR's	yes	Removes radioactivity from containment atmosphere and cools containment atmosphere	Traps aerosols and some iodine[e]	Large aerosol concentrations in the containment could clog the filters and rapidly render them ineffective
Auxiliary building filter system	PWR's	yes	Removes radioactivity from auxiliary building atmosphere	Traps aerosols and some iodine[e]	Large aerosol concentrations in the auxiliary building could clog the filters and rapidly render them ineffective
Main stream isolation valve leakage control system	BWR's	yes	Filters gases leaked past inboard main steam isolation valve	Traps aerosols and some iodine[e]	For severe accidents, this system would generally be bypassed, when not bypassed, large aerosol concentrations could clog the system

Table D 2 (continued)

Engineered safety feature	Plant	Electricity required	Effects	Effect on radionuclides in containment during limited-core-damage accidents	Effect on radionuclides in containment during complete meltdown accidents
Pressure suppression pool	BWR's	no	Reduces pressure in containment (condenses steam released from drywell), scrubs some radioactivity from the drywell releases	Removes some aerosols[b] and some iodine[c]	Large aerosol concentrations would be acted on most effectively,[b] probably most effective if pool temperature is below boiling
Standby gas treatment system	BWR's	yes	Filters contaminants leaked from primary containment into secondary containment	Traps aerosols and some iodine[e]	For severe accidents, this system would generally be by-passed, when not by-passed, large aerosol concentrations could clog the system
Ice bed condenser	Ice-condenser PWR's	no (yes for long-term)	Reduces pressure (condenses steam) in containment, scrubs some radioactivity from the atmosphere, removes molecular iodine	Removes some aerosols[b] and some iodines[c,f]	Large aerosol concentrations would be acted on most effectively,[b] effective only if ice beds had not melted

[a]For more extensive discussions of the effectiveness of various engineered safety features, see both Chapter 8 of NUREG-0772 (USNRC, 1981) and Chapter 4 of NUREG-0771 (Pasedag, Blond, and Jankowski, 1981)

[b]The removal rate is greater for large particles than for small particles Inasmuch particles in high density aerosols tend to agglomerate into larger particles, removal rates by sprays usually can be expected to be relatively greater for situations with higher aerosol concentrations

[c]Organic iodine is not removed very effectively by this ESF

[d]At some reactors, chemical additives help to convert elemental iodine to less volatile iodides

[e]Organic iodine is not trapped as effectively as either elemental or particulate iodine

[f]Chemical additives help to convert elemental iodine to less volatile iodides

Table D.3. Presence of selected engineered safety features

Engineered safety feature	PWR	Ice condenser PWR	BWR-Mark I	BWR-Mark III
Containment leak-tightness	Yes (large volume, high pressure)	Yes (medium-large volume, medium pressure)	Yes (small volume, high pressure)[a]	Yes (medium-large volume, medium pressure)[b]
Containment spray system	Yes	Upper compartment only	No	Yes, but not necessarily activated
Containment recirculation filter system	Some	Yes	No	No
Auxiliary building filter system	Yes	Yes	No	No
Main steam isolation valve leakage control system	No	No	Yes	Yes
Pressure suppression pool	No	No	Yes	Yes
Standby gas treatment system	No	No	Yes	Yes
Ice-bed condenser	No	Yes	No	No

[a]Mark I's (and Mark II's) have small-volume drywells and large-volume secondary containments; whereas the drywells are designed to withstand high pressures (50-60 psig), the secondary containments are not designed to endure elevated pressures (design pressures are approximately 0.25 psig). (In a Mark I or Mark II BWR, the primary containment is composed of the drywell and the wetwell; the secondary containment is a shield building.)

[b]Mark III's have small-volume drywells and large-volume primary and secondary containments; both the drywells and the primary containments are designed to withstand moderate pressures (15-25 psig). The secondary containment (or shield building) is not designed to endure elevated pressures.

Containment sprays would result in radionuclide removal from the containment atmosphere directly by any of several processes including impaction, interception, and diffusion. In addition, the sprays would precipitate further removal by promoting the condensation of steam onto aerosol particles and the condensation of vapors onto both particles and surfaces. For accidents in which they were activated, they could be very effective in reducing the concentrations of most radionuclides in the containment atmosphere as long as they continued to function.

Filters would result in radionuclide removal from the containment atmosphere by both adsorption of gases and filtration of aerosol particles. For accidents in which there were small amounts of radionuclides released to the containment, the filters could substantially reduce both the concentrations of most radionuclides in the containment atmosphere and the amounts of those radionuclides leaked to the environment. For accidents in which there were large amounts of aerosols released to the containment, the filters might clog rapidly and lose their effectiveness.

A suppression pool would result in scrubbing of some of the materials escaping from the drywell. The effectiveness of the removal would be a function of both the temperature and the depth of the water in the suppression pool, as well as on the path taken by the radionuclides.* Passage through the suppression pool generally would be a very effective method of reducing the amounts of many radionuclides released to the environment as long as the containment remained unruptured and the pool remained cooled. Unfortunately, the suppression pool would be bypassed early in some important types of possible accidents and it would be boiling in some others.†

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*In addition, the effectiveness of the removal would be influenced by many other factors, including the noncondensible fraction in the flow stream, the flow rate, the bubble size, the aerosol size distribution, the aerosol size concentration, etc.

†It is currently a matter of some debate whether heated water would be as effective in scrubbing aerosols as cooled water would be. Unfortunately, adequate experimental and theoretical investigations do not appear to have been performed. Experimental investigations to address this issue are currently underway at Battelle Columbus Laboratories.

Ice-bed condensers would result in radionuclide removal from the containment atmosphere directly by any of several processes such as impaction or diffusion. They also could enhance further removal indirectly by lowering the temperature of the containment atmosphere, thus promoting the condensation of steam and accelerated aerosol removal. As long as the ice beds had not melted completely, they could be effective in reducing both the concentrations of most radionuclides in the containment atmosphere and the amounts of those radionuclides released to the environment.

The effectiveness of all the ESF's, as well as the effectiveness of the natural processes, in reducing the amounts of the radionuclides escaping to the environment in any accident would depend on the condition of the containment itself. For any accident in which the containment was integral and did not leak substantially, there would be time for all the various processes promoting radionuclide removal from the containment atmosphere to function and thus to prevent large releases to the environment. However, for any accident in which the containment either was severely breached early in the scenario or else was bypassed, all those processes would not have enough time to be as efficacious and so the releases to the environment of at least some radionuclides could be large.

D.3 SUMMARY OF STATE-OF-THE-ART

The processes which affect the transport of materials through the containment are exceedingly complex. As a result, most numerical considerations of those processes are performed using one or more of the various computer codes which have been developed expressly for that purpose. In this section, the features of those codes which are important for understanding their results, along with the contexts in which those results should be placed, are outlined. More complete discussions and comparisons of the codes and the models behind them can be found in Baybutt (1981), Gieseke et al. (1981), and Nuclear Energy Agency (1979), as well as in some of the other documents referenced in the following text.

D.3.1 Reactor Safety Study

The Reactor Safety Study (RSS) was the first comprehensive assessment of the risk from all types of accidents at light water reactors (LWR's) (see Section 1.1.3 of Chapter 1). In that study, models were formulated for considering the post-accident transport of radionuclides in the containment (Postma, Owzarski, and Lessor, 1975). An integrated, computerized version of the basic approach, CORRAL (Containment of Radionuclides Released After LOCA), was developed at the time of the RSS. A generalized version of that computer code, CORRAL-II, is still extensively used today (Burain and Cybulskis, 1977; Owzarski, 1978). In fact, it is probably the most widely used code for considering post-accident radionuclide transport in the containment. That code utilizes basically the same transport and removal models that were used in the RSS. The current results obtained using CORRAL-II differ from early results obtained using CORRAL (for example, those in the RSS) primarily because a somewhat more advanced procedure for estimating the necessary thermal-hydraulic input is now used [MARCH (Meltdown Accident Response Characteristics); Wooton and Avci, 1980]. In addition, current and early CORRAL results sometimes differ because certain accident description assumptions used to develop input for CORRAL have changed with time as a better understanding of the many processes involved in accidents has been developed. From here on, both CORRAL-II and its predecessor will both be referred to as CORRAL.

In CORRAL, all radioactive species are grouped, on the basis of fundamental transport properties, into four categories: noble gases, elemental halogens, organic iodides, and species present as aerosols. The releases of each of the four categories are divided, on the basis of the "processes" resulting in the releases from the core materials, into four components: gap, meltdown, oxidation, and vaporization. To permit the distributions of these releases over somewhat realistic time scales, each of these releases is further subdivided into discrete portions (see Subsection D.4.2.6). The transport within the containment of each portion of each release is followed independently for each of the four release categories. After the transport has been considered, the fractions of each of the seven RSS element groups (see Table D.4) escaping from the containment are estimated. The effects of both natural processes and ESF's on the radionuclides escaping are included in CORRAL.

The processes considered in CORRAL are listed in Table D.5. As can be seen by inspection of that table, many of the more important processes in the containment are not explicitly modeled in CORRAL. Instead some of them are included only implicitly. This is because CORRAL is a semiempirical code. It is based largely on the data obtained from the Containment Systems Experiments (CSE's) (Hilliard et al., 1971; Postma and Johnson, 1971). Those experiments were performed in a steam-filled atmosphere generated to simulate the conditions present after a loss-of-coolant accident in a pressurized water reactor (PWR). However, the conditions of those experiments are not representative of the conditions to be expected for some types of LWR accidents. Thus, CORRAL may not be appropriate for considering all types of accidents.

In particular, inasmuch as the CSE's were performed in essentially isothermal conditions, CORRAL, because of its dependence on those experiments, tends to underestimate the deposition by convective flow, thermophoresis, and diffusiophoresis for some accident situations. In addition, because the aerosol concentrations in those experiments were much less than the concentrations anticipated for some meltdown accidents, CORRAL tends to underestimate the importance of agglomeration in certain cases. Furthermore, because the CSE's were performed in low to moderate steam concentration environments, CORRAL is most appropriate for considering accidents in which such steam concentrations would occur (Hilliard and Postma, 1981; Hilliard et al., 1971; Baybutt, 1981). Still another factor which may affect the transferability of the results of the CSE's to certain accident situations, and hence may restrict the usefulness of CORRAL for those situations, is the relatively low temperature regime utilized in the experiments. Other factors which make extrapolation of the results of the CSE's less than completely straightforward, and therefore the use of CORRAL questionable, include the size of the overall experimental apparatus (much

Table D.4. RSS "element" groups

Noble Gases - Xe, Kr	
Halogens - I, Br	
Organic Iodide	
Cesium - Cs, Rb	
Tellurium - Te, Se, Sb	
Strontium - Sr, Ba	
Ruthenium - Ru, Mo, Pd, Rh, Tc, Co	
Lanthanum - La, Nd, Eu, Y, Ce, Pr, Pm, Sm, U, Np, Pu, Am, Cm, Zr, Nb	

Table D 5 Comparison of containment computer codes[a]

Characteristic	CORRAL-2	HAARM; QUICK; and ZONE	NAUA	PARIDESEKO-III and AEROSIM
Reactor Type	LWR	LMFBR	LWR	LMFBR
Containment Compartments	Multiple	Single (H,Q); Multiple (Z)	Single	Multiple (P); Single (A)
Fission Product Form	Vapor, aerosol	Aerosol	Aerosol	Aerosol
Natural Processes				
Brownian Coagulation	yes[b]	yes	yes	yes
Gravitational Coagulation	yes[b]	yes	yes	yes
Turbulent Coagulation	yes[b]	yes[c]	no	no
Steam Condensation on Particles	yes[b]	no[d]	yes	no[d]
Vapor Sorption on Particles	yes[b]	no	no	no
Gravitational Deposition	yes[e]	yes	yes	yes
Diffusional Deposition	yes[e]	yes	yes	yes
Thermophoretic Deposition	yes[e]	yes[c]	yes	yes
Diffusiophoretic Deposition	yes[b]	no	no	no
Resuspension	no	no	no	no
Leakage	yes	yes	yes	yes
Electrostatic Interactions	yes[b]	no	no	no
Radioactive Decay	no	no	no	no
Phase Changes	no	no	no	no
Chemical Reactions	no	no	no	no
Aerosol Concentrations	b,f	g	h	i

Table D.5. (continued)

Characteristic	CORRAL-2	HAARM; QUICK; and ZONE	NAUA	PARIDESEKO-III and AEROSIM
Engineered Processes				
Removal by Sprays	yes[j]	no	no	no
Removal by Ice Condensers	no[k]	no	no	no
Removal by Suppression Pools	no[k]	no	no	no
Removal by Filters	no[k]	yes	no	no
Source Term	fixed time dependence	arbitrary time dependence	arbitrary time dependence	arbitrary time dependence
Thermal-Hydraulic Conditions	input	input	input	input

[a]This is an extension of Table 7.1 of NUREG-0772. It is based on discussions in several documents (Bunz, Schikarski, and Schock, 1981; Gieseke et al., 1981; Baybutt, 1981; Nuclear Energy Agency, 1979).

[b]This process is not explicitly modeled in CORRAL-2, though because the code is empirically based, it may be thought to be included to some extent. Thus, the CORRAL-2 results cannot necessarily be expected to be appropriate for accidents in which the conditions are much different than those in the CSE's.

[c]This process is modeled but is not utilized due to uncertainties in the formulation and/or data.

[d]The effect of steam is sometimes partially taken into account by assuming that all particles are spherical

[e]Although this process is explicitly modeled, the model depends on the CSE results.

[f]The particle sizes are fixed within the code such that the sedimentation loss model predicts the attenuation observed in the CSE results. (In general, the CSE's had low aerosol concentrations).

[g]HAARM-3 assumes spatially homogeneous distribution of particles in the containment with the sizes of the particles being distributed log normally. QUICK and ZONE make no simplifying assumptions about particle sizes.

[h]NAUA assumes a spatially homogeneous distribution of particles but makes no simplifying assumptions about particle sizes. NAUA takes into account a size-dependent composition of the particles.

[i]Arbitrary particle size distributions can be considered.

[j]Organic iodides are not considered for this process.

[k]This process is not explicitly modeled but its net effect is accounted for.

smaller than any reactor containment) and the lack of forced convection devices.

D.3.2 Work After the Reactor Safety Study

D.3.2.1 Comparison of Codes

Since the completion of the RSS, several codes other than CORRAL have been developed to consider post-accident radionuclide transport in reactor containments. In general, these codes have been based on much more mechanistic models than CORRAL was. Typically the codes fall into two categories: codes developed expressly to consider LWR accidents; and codes initially intended to consider liquid metal fast breeder reactor (LMFBR) accidents but adapted to describe LWR ones. The phenomena and processes treated by some of the more widely used of these codes and the basic assumptions utilized in implementing the models in them are summarized in Table D.5.

One of the principal codes developed specifically for LWR accidents is NAUA (Bunz, Schikarski, and Schöck, 1980). It differs from the other codes mentioned here in that it is the only code which can legitimately be used to consider accidents with high steam concentrations. Unlike CORRAL, it also justifiably can be used to consider high aerosol concentrations.

Two of the LMFBR codes which have been adapted for LWR accident use are HAARM-3 and QUICK (Gieseke, Lee, and Reed, 1978; Gieseke, Jordan, and Lee, 1979; and Jordan et al., 1980). A related code, ZONE, is an extension of QUICK to a multicompartment system (Gieseke et al., 1980). These codes all differ from CORRAL in their ability to describe the high aerosol concentrations expected in some accidents. They are also unlike CORRAL in that they do not account for the effects of steam and the effects of the ESF's.

The results of using the different computer codes have been compared to various extents with the results of aerosol experiments. Consequently, to a certain degree, some of the codes can be thought to be validated.* However, a large number of the experiments, and thus any codes validated by considering them, are not necessarily appropriate for describing many potential LWR accidents. For example, inasmuch as most of the aerosol experiments have been performed to investigate hypothetical LMFBR accidents, their results, and thus the calculations by the associated LMFBR computer codes, are not necessarily directly applicable to LWR accidents. In LMFBR accidents, the primary component of the aerosols in the containment would be expected to be sodium oxide(s). In contrast, in LWR accidents, the primary component of those aerosols would be expected to be steam. Models which adequately describe sodium oxide aerosols in dry conditions cannot necessarily be expected to adequately describe radionuclide aerosols in the presence of large quantities of steam.†

Another reason the experiments are not necessarily directly applicable for considering LWR accidents is that they have been performed on relatively small scales. Consequently, the relative importance of certain surface-related removal mechanisms tends to be overestimated in such experiments. Unfortunately, the appropriate way to scale the experimental results is not always obvious.

Still another reason that the experiments are not always appropriate for considering some LWR accident conditions is that many of them have been performed at relatively low aerosol concentrations. Because the aerosol effects are dependent upon the concentration in a

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*Even the "first-principles" codes involve somewhat arbitrary adjustments of certain parameters or factors which are not treated on a first-principles basis.

†After vaporization during an accident, sodium typically would react with oxygen to form low vapor pressure oxides. If large amounts of such compounds were generated, then significant condensation of the oxides into aerosols would occur. Revolatilization of such compounds from the aerosols would generally not be significant. In contrast, for compounds with relatively high vapor pressures such as water, after the initial aerosol formation, revolatilization from the aerosol particles would be substantial in many accident environments. Indeed, in some situations, a repeating cycle of condensation and volatilization would be important for such relatively volatile species. LMFBR aerosol codes cannot describe such behavior.

rather complicated way, the extrapolation of low concentration results to high concentration cases can be misleading.

Other codes to consider transport in the containment are currently being developed. A partially mechanistic successor to CORRAL, MATADOR, is being developed at Battelle Columbus Laboratories. The CONTAIN code, a first-principles code initially intended to describe post-accident transport in LMFBR's, is being extended so that it will be able to consider LWR accidents (Senglaub et al., 1981). The TRAP code, a mechanistic code initially intended to consider just transport through the primary coolant system, is also being extended to describe behavior in the containment of an LWR (Baybutt and Jordan, 1977; Jordan, Gieseke, and Baybutt, 1979).

D.3.2.2 Comparison of Results of Codes

Some of the models behind the available containment computer codes differ substantially. This, coupled with the fact that none of those codes is completely applicable to the conditions expected for all LWR accidents, makes it adviseable to compare the basic characteristics of the predictions of the various codes.

As might be expected on the basis of the large differences between the models used, the descriptions of a specific accident sequence by various codes can differ significantly. For example, as is illustrated in Figure D.1 (Gieseke et al., 1981), the estimates of the fraction of the total aerosol mass airborne in the containment for a given accident sequence* differ by orders of magnitude at long times after the accident for four of the more frequently used codes. Indeed, it is common for the codes to differ substantially in their descriptions of many of the details, such as concentrations, aerosol particle size distributions, and so forth, as functions of time for any given accident sequence.

Fortunately, it can be said that the various codes often differ much more in the details of the descriptions they produce than in the

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*The accident sequence considered in Figure D.1 is TMLB'-ε in a large containment PWR. See Appendix A for an explanation of this notation and a description of this sequence.

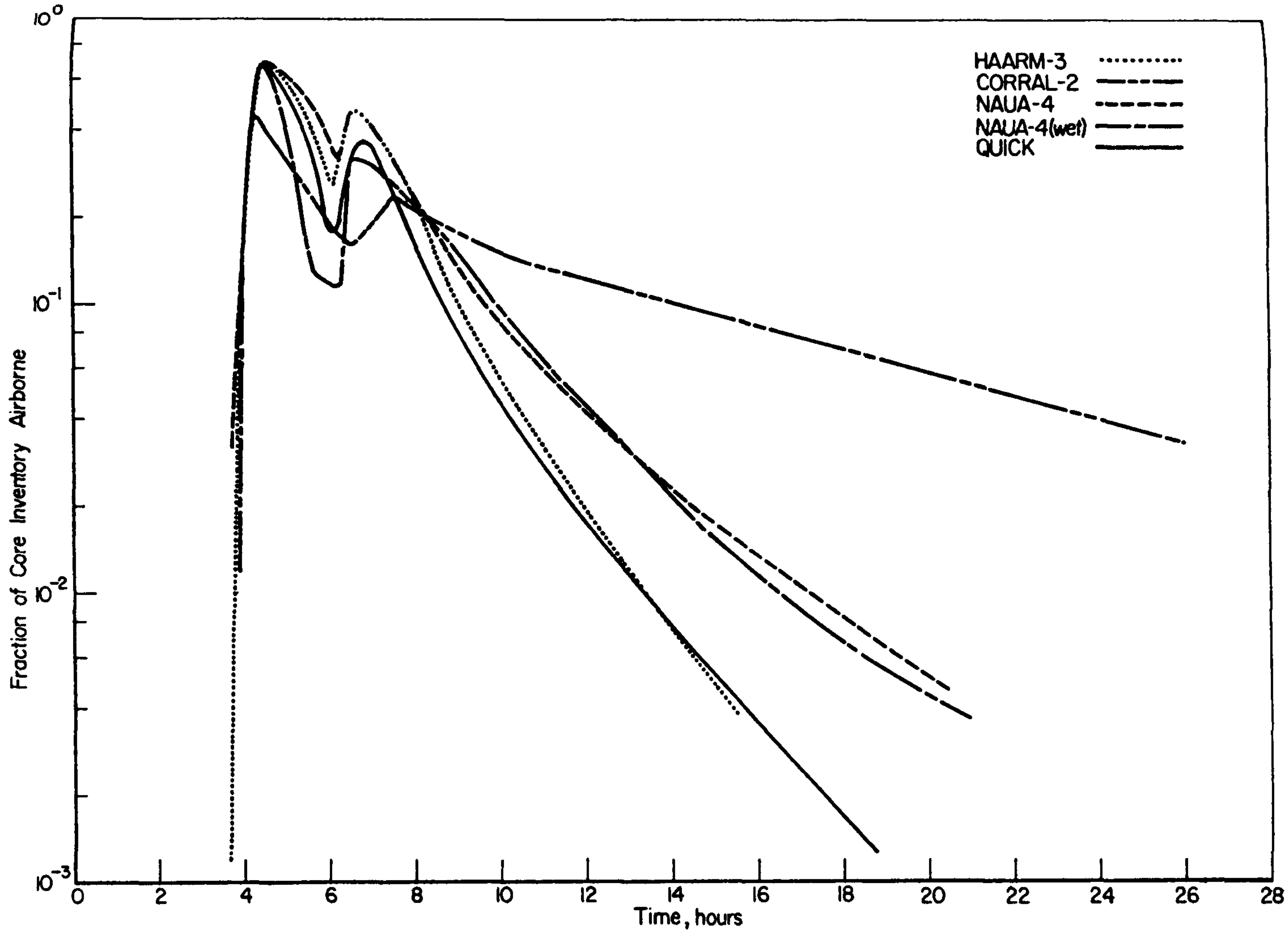

Fig. D.1. Comparison among HAARM-3, CORRAL-2, NAUA-4 and QUICK for total airborne particulate (Gieseke et al., 1981).

estimates of the total amounts of either the radionuclides released from the containment or the radionuclides remaining in the containment for any given accident. This is because the amounts estimated to be released frequently depend primarily on the early airborne concentrations in the containment and in the early stages of any accident, when those concentrations are relatively large, the descriptions of those concentrations by the various codes are rather close. This is typically the situation if the containment is assumed either to fail to the atmosphere early in the accident, not ever to fail to the atmosphere, or not to fail at all. However, if the containment is assumed to fail to the atmosphere at some intermediate time during the accident while the concentrations in the containment are still moderately high, the descriptions of the releases by the various computer codes can differ substantially.

D.4 APPROACH FOR ESTIMATING RADIONUCLIDE RELEASES TO THE ENVIRONMENT

D.4.1 General Approach

Because of the limited scale of this project, the overall approach taken to estimate the radionuclide releases from the containment, and thus also the amounts not released, was to take previously obtained CORRAL estimates for various important accident sequences* and to modify them to include some insights gained since the initial estimates were made (USNRC, 1981). The previous meltdown accident estimates were taken from the Reactor Safety Study Methodology Applications Program (RSSMAP) reports of Sandia and Battelle (Kolb et al., 1981; Carlson et al., 1981; and Hatch, Cybulskis, and Wooton, 1981)† and from the associated RSS

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*Despite its frequently criticized nonmechanistic basis, CORRAL appears to provide descriptions comparable to those of more rigorous codes of at least the amounts of radionuclides released and the amounts retained in the containment for many accident sequences. (See the discussion in the last subsection.) In addition, it takes into account the effects of the ESF's.

†The four plants analyzed in the RSSMAP were Sequoyah (a Westinghouse 4-loop PWR with a steel shell, ice condenser containment), Oconee (a Babcock and Wilcox 2-loop PWR with concrete dry containment), Calvert Cliffs (a Combustion Engineering 2-loop PWR with concrete dry

rebaselining work of Battelle (Cybulskis, 1981; Wooton, 1981) (see Section 1.1 for a brief outline of the scope of all that work). The CORRAL-based estimates of the radionuclide releases presented in that work, and used here, are summarized in Tables D.1.1 through D.1.7 of the addendum of this appendix.

Unfortunately, the RSSMAP reports did not include estimates for the radionuclide releases resulting from accidents involving less damage than complete meltdown of the reactor core. Consequently, to ensure the greatest consistency with the RSSMAP estimates for the meltdown accidents, the estimates for limited-core-damage accident sequences were taken from the RSS. Unlike the situation for meltdown accidents, sequences for only a single plant were considered.

D.4.2 Basic Issues

The insights developed concerning post-accident radionuclide transport in the containment since the initial estimates were made were taken to be mostly those mentioned in the recently published *Technical Bases for Estimating Fission Product Behavior During LWR Accidents* (USNRC, 1981). According to that report, for considering radionuclide transport through the containment, there are four major areas which need to be addressed in greater depth than has been done previously for a complete spectrum of accidents. All except one of these areas are related to the behavior of aerosols; they involve the effects of high aerosol concentrations, the effects of condensing steam, the effects of leak size, and the effects of iodine interactions (Gieseke et al., 1981). These areas are considered in the following four subsections. Two other areas of concern, the effects of both core size and status and the effects related to the timing of the releases into the containment, are dealt with in succeeding subsections.

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containment), and Grand Gulf (a General Electric BWR with Mark III containment system). The RSSMAP work on Calvert Cliffs had not been published when the containment escape estimates were performed for this study; therefore the work on Calvert Cliffs was not used in this study.

D.4.2.1 Effect of Aerosol Concentration

Accidents involving less damage than complete melting of the core typically would result in relatively low aerosol concentrations in the containment. In contrast, some accidents involving complete melting could result in comparatively large initial aerosol concentrations due to releases of aerosols from both the core materials and the concrete. Thus, inasmuch as aerosols of higher initial concentrations tend to experience proportionally much larger early losses due to deposition than aerosols of lower initial concentrations, concentration-dependent effects might be expected to result in somewhat different behavior of the aerosols after limited-core-damage accidents than after meltdown accidents.

The dependence of the total amount of material escaping from the containment on the total amount of aerosols generated is illustrated in Figure D.2 (Gieseke et al., 1981) for a given accident sequence*. In the sequence investigated, neither the containment sprays nor the recirculation fans operate; thus, removal of aerosols results from only natural processes and not from the operation of ESF's. As can be seen, for the conditions considered, the effects of aerosol concentration on the amount escaping would not be apparent for total aerosol masses of less than 700 kg. However, above that mass, concentration-enhanced aerosol removal processes could significantly affect the total amount of material released from the containment. (That is, there is an approximately linear relationship between the aerosol mass initially released and the predicted mass leaked below an initial release of mass of 700 kg. In contrast, above 700 kg, the relationship between the aerosol mass initially released and the predicted mass leaked departs from linearity as increased aerosol concentrations promote more rapid aerosol deposition.)

The rate at which material would be released from the containment to the environment is illustrated for the same accident sequence in Figure D.3 (Gieseke et al., 1981). In that figure, two possibilities,

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*The accident sequence considered is TMLB'-ε for a large containment PWR.

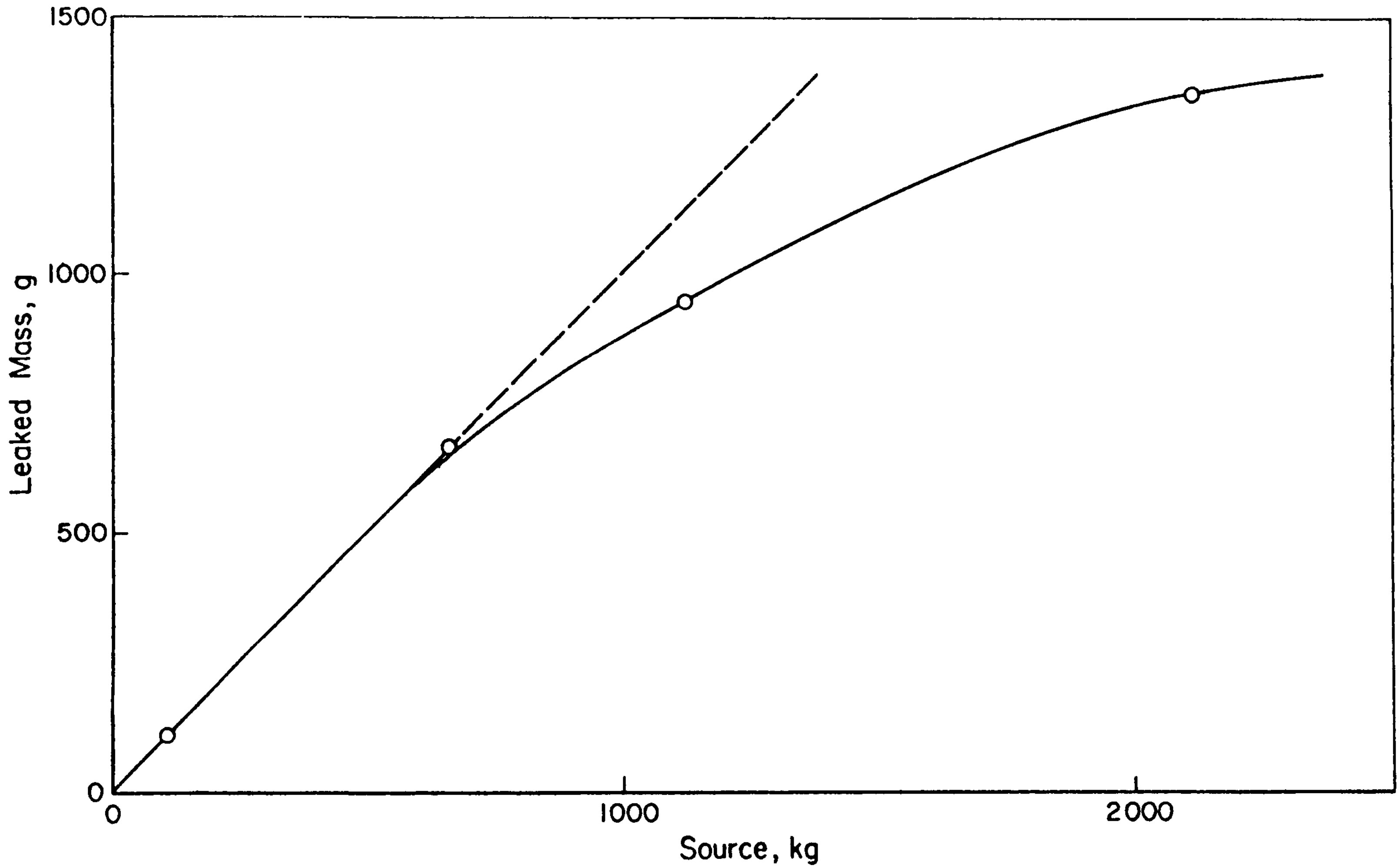

Fig. D.2. Effect of particle source on mass leaked from containment (Gieseke et al., 1981).

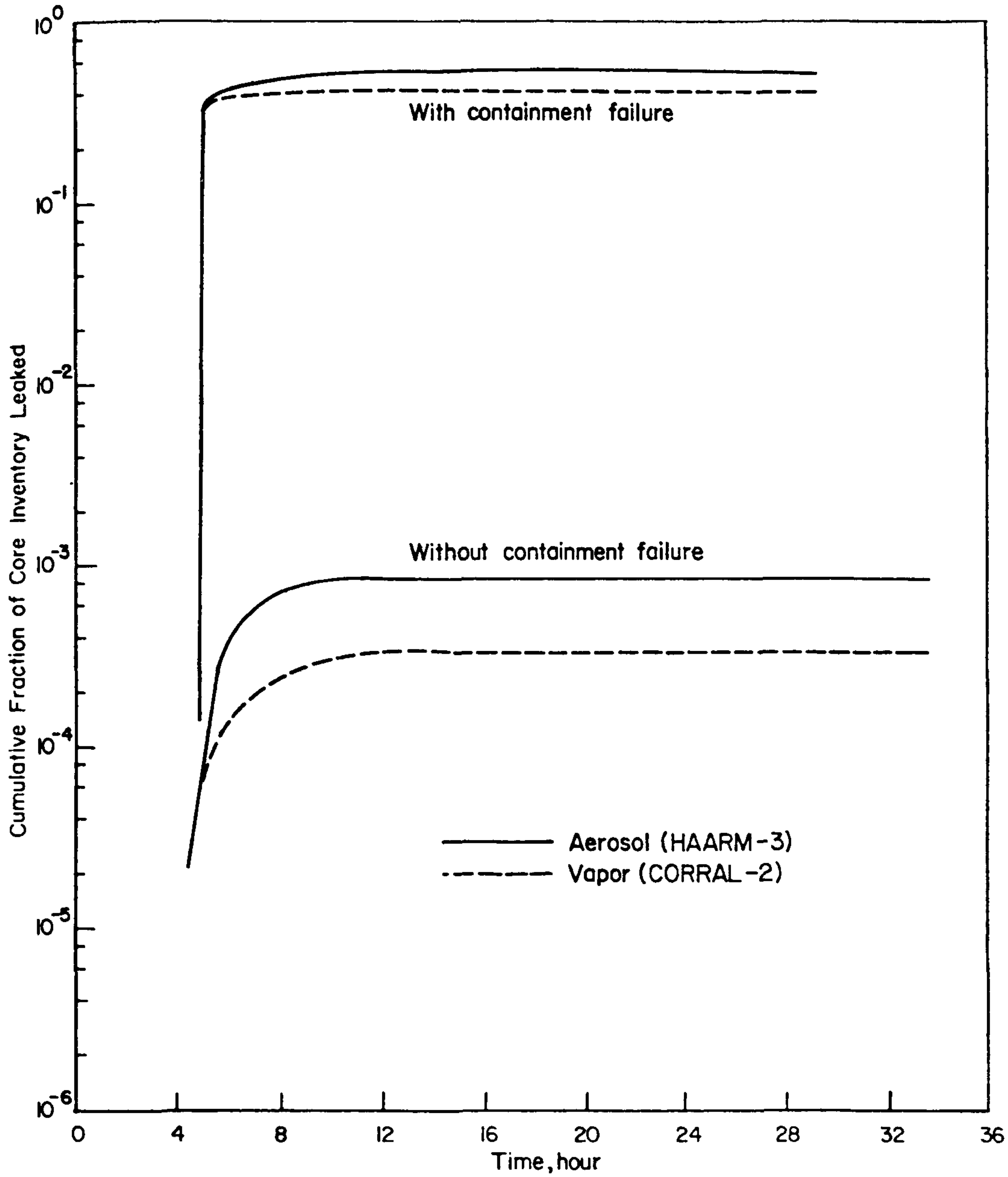

Fig. D.3. Effect of containment failure on mass leaked from containment (Gieseke et al., 1981).

one in which the containment fails early in the accident and another in which the containment does not fail except perhaps by basemat melt-through, are considered. Obviously, in either case, most released material would escape within the first several hours after melting was initiated. Indeed, if the failure occurred very early, most released material could escape within much less than an hour.

The rate at which natural processes would affect the aerosol concentration in the containment is illustrated for the same sequence in Figure D.4 (Nuclear Energy Agency, 1979). In that figure, both the aerosol concentrations and the amounts leaked as functions of time are depicted for two different total aerosol inputs, namely, 1000 and 2000 kg.* Because the time scale for natural removal for the considered sequence is comparable to the time scale for the majority of escape from the containment, the effects of such high aerosol concentrations on the amounts escaping would be most noticeable for situations without containment failure, for example, for the situation shown in Figure D.2. In contrast, for many situations involving early breaching of the containment, the effects of high aerosol concentrations in the containment would not even be apparent for the concentrations considered.

The disproportionate decrease of the time scales for natural removal with increasing aerosol concentration is illustrated in Table C.4 of the previous appendix. As can be seen by inspection of that table, for higher concentrations than those considered in the examples in Figures D.2-D.4, the effect of accelerated removal might be observable even for accidents involving early containment failure.

The aerosol concentration would depend upon the volume of the containment as well as on the total amount of material released. Figures D.2 and D.4 depict the situation for a relatively large containment PWR. Ice condenser PWR's and many boiling water reactors (BWR's) have containments which are typically a factor of several to an order

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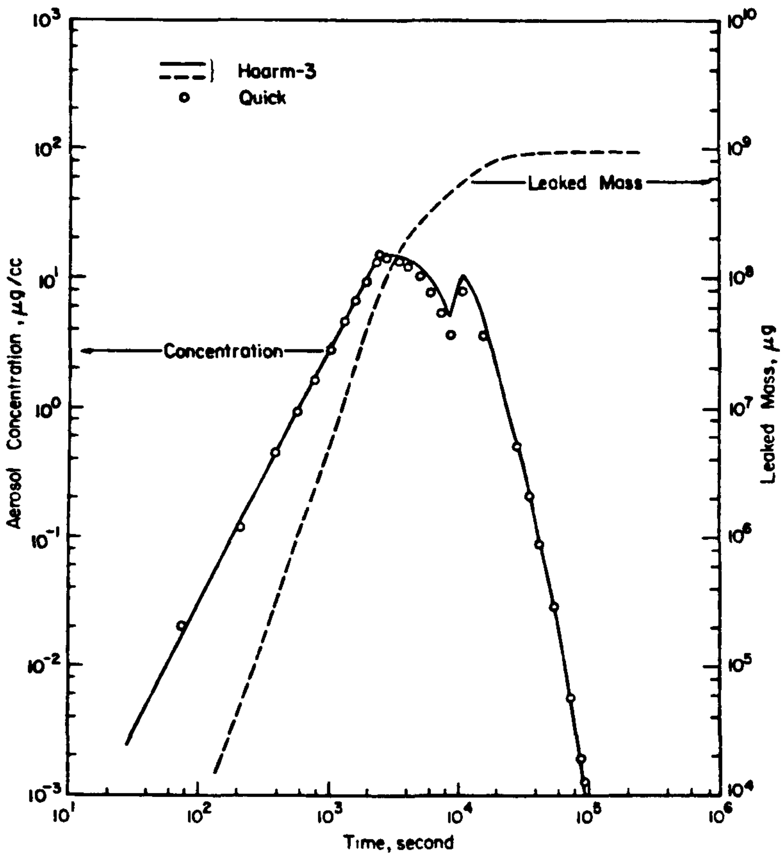
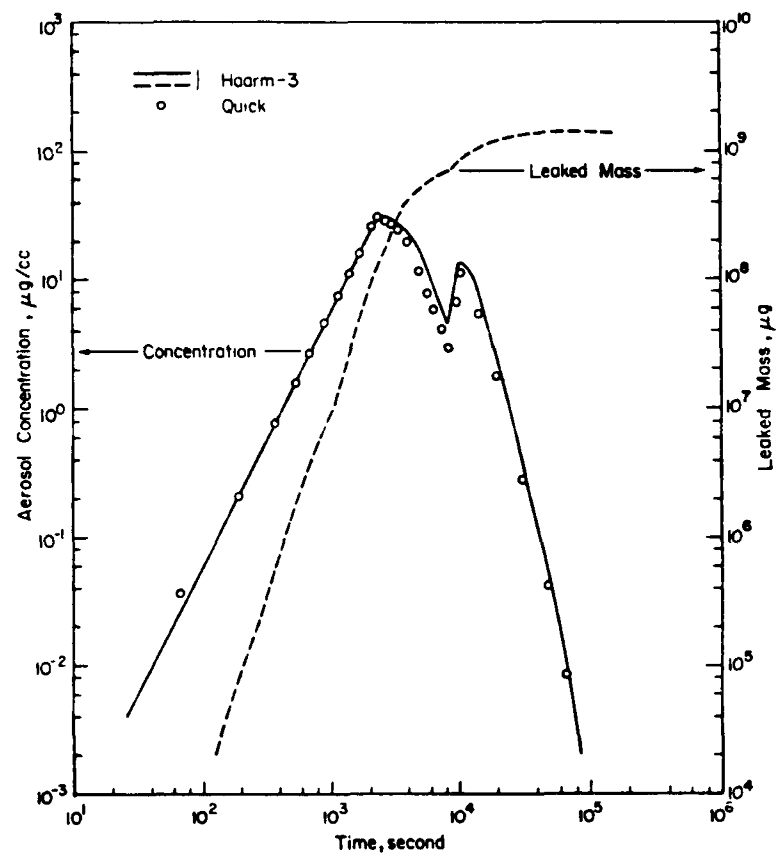
*For each release, the meltdown release aerosols constituted 73% of the total aerosol mass. The meltdown release aerosols were assumed to be generated at a linearly increasing rate for approximately a half hour. After a time period of no aerosol production, the vaporization release aerosols were taken to be generated at a constant rate over a 2-h period (Gieseke et al., 1981).



Airborne aerosol concentrations predicted for TMLB´ with 1000 kg aerosol source.



Airborne aerosol concentrations predicted for TMLB´ with 2000 kg aerosol source.

Fig. D.4.

of magnitude smaller than the volume for that PWR. Consequently, an even larger concentration effect generally might be expected for such plants.

"Maximum possible" aerosol concentrations for various types of nuclear plants are presented in Table D.6 for some postulated aerosol releases. The first two releases (20 and 100 kg) correspond to the total amounts of aerosols predicted to be released to the RPV and potentially to the containment during certain accidents involving only partial melting of the core. The next release (500 kg) corresponds to the total quantity of aerosols which might be released at least into the RPV during certain more severe accidents involving complete meltdown while the last amount (2500 kg) represents the total aerosol mass which might be released to the containment during such accidents. (It should be noted that the total amounts of aerosols which would be generated are highly uncertain and could be much smaller or larger than the values given here.)

To put the concentrations in Table D.6 in perspective, it is useful to consider the effect of concentration on the amount leaked in the sequence described in Figures D.2-D.4. For that case, for a maximum concentration of less than 20 g/m^3 (a total aerosol release of 1000 kg for the conditions assumed), the effects of concentration-enhanced natural removal would not be too noticeable in the total amount leaked from the containment. In contrast, for a maximum concentration of 45 g/m^3 (a total aerosol release of 2000 kg), the effects of natural removal would result in a reduction of approximately 30% in the total amount leaked beyond the situation with no concentration-enhanced removal.

Although the predicted "maximum" concentrations for many possible accidents are above the concentrations at which such enhanced removal could be significant, three factors would tend to lessen the relative importance of that removal. First, for relatively slow accidents, the actual concentrations always could be well below the maximum possible concentrations. Second, removal by ESF's often might exceed that by natural processes. And third, core-melt accidents in at least plants with small containments typically would result in containment failure

Table D.6 Maximum aerosol concentrations in the containments of typical nuclear plants

Aerosols released (kg)	Maximum concentration (g/m^3)[a]			
	Large containment PWR	Ice condenser PWR	Mark I BWR[b]	Mark III BWR[b]
20	0.4	2	4 (4)	2 (0.4)
100	2	10	20 (20)	10 (2)
500	10	50[c]	100 (100)[c]	50 (10)[c]
2500	50[c]	250[c]	500 (500)[c]	250 (50)[c]

[a]The estimation of these maximum concentrations unrealistically assumed instantaneous release of all aerosols into the containment. The volumes assumed are 5×10^4 m^3 for the large containment PWR; 1×10^4 m^3 for the ice condenser PWR; 5×10^3 m^3 and 5×10^3 m^3 for the drywell and the wetwell, respectively, for a Mark I BWR; and 1×10^4 m^3 and 5×10^4 m^3 for the drywell and the wetwell, respectively, for a Mark III BWR.

[b]The numbers in parentheses are for the wetwell.

[c]Containment failure often would occur prior to this concentration being reached.

early in the accident, long before the maximum concentrations might have been attained. (See Table D1.7 in the addendum of this appendix.)

Most of the accident sequences which dominate the risk to the human population for any nuclear plant involve either early substantial breaching of the containment or else a complete by-passing of the containment. Typically that rupturing would occur prior to or at the time of the melt-through of the reactor vessel. Thus, the net effects of agglomeration and subsequent precipitation within the containment generally might be small for most of the dominant emergency planning accidents due to the short residence time of the aerosols in the containment. However, the net effects of those processes would be large for many other accidents with either delayed containment failure or no containment failure.

In contrast, the accidents of importance to the equipment inside the plant involve no massive breaching of the containment. For such accidents involving complete meltdown of the core,* the aerosol concentration effect could decrease substantially the amount of material leaked from the containment; however, it would not affect significantly the amount retained and thus the amount contributing to the radiation field inside the reactor. It would noticeably affect the early time-dependent distribution (fractions airborne, waterborne, ...) of the material within the containment. For accidents involving less than a complete meltdown, the total amount of aerosols released into the containment often would be relatively small and so concentration-enhanced removal would not be expected to be important.

In view of all these considerations and because the prediction of the total amounts of aerosols generated is highly uncertain (see Appendix B), no modifications of the previous results have been made to account for the effects of high aerosol concentrations. However, if the uncertainties in the prediction of such amounts could be substantially reduced, reasonable modifications would be straightforward.

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*Meltdown accidents typically would be considered for equipment qualification purposes. However, they generally have not been considered in the work for this project. They have been included in this discussion for completeness.

D.4.2.2 Effect of Steam Environment

Most accidents involving partial or complete core melting would result in the release of very large amounts of steam from the evaporation of water in the reactor coolant system. The amount of steam released would generally be much greater than the total amount of all the other releases (fission products, structural materials, and hydrogen) combined. In PWR's, this release of steam generally would be directly into the containment. In BWR's, the endpoint of the steam release would depend upon the accident; typically, for loss-of-coolant accidents (LOCA's), the steam would be released into the drywell whereas for transient-initiated accidents, the steam would bypass the drywell and be released directly into the suppression pool in the wetwell.

For accidents involving complete melting of the core and eventual melt-through of the RPV, more steam would be released as the result of concrete decomposition. In addition, in some meltdown accidents, steam could be generated by evaporation of water from the reactor cavity. In all cases, the steam produced after RPV melt-through would be released into the containment.

The steam would be expected to have two primary effects. First, it would increase the size of aerosol particles by condensing onto them. Second, it would make all particles more spherical. Although the first of these effects would tend to result in accelerated removal of the aerosols from the containment atmosphere, the second of these effects would have just the opposite effect. The net result of the two effects generally would be enhanced removal. Such removal by condensation would be expected to be significantly hastened by any temperature decreases. The effectiveness of any steam in accelerating removal of radionuclide-containing aerosols from the containment atmosphere would depend partly upon the timing of its release into the containment, with much of the steam being released early in any accident potentially being condensed prior to the production of significant quantities of radioactive aerosols.

If containment sprays operated in any accident, they would tend to lower the steam concentration. Thus they would partially counteract the enhanced removal potentially caused by steam condensation. However, containment sprays would also decrease the containment temperature so that some types of aerosol and vapor removal processes would be promoted. In a steam-filled environment in which the sprays operated, the primary removal typically would be due to the operation of the sprays. Understanding of such environments is probably adequately covered by insights obtained from previous experiments.* However, supersaturated steam environments in which the sprays failed to operate do not appear to have been sufficiently investigated. (Experiments are currently in progress in Germany and in the U.S. to investigate such environments.)

Although the combined removal effects of high steam concentrations and sprays may be underestimated in current calculations, there is no unequivocal indication that this is the situation. Not only are the experimental data inadequate to make such a determination, but also the theoretical models are likewise not definitive. Only one containment computer code, NAUA, includes a model to account for the effects of condensing steam. Unfortunately, sufficient data are not available to use that model in a predictive mode. (Some of the experiments currently in progress in Germany and in the U.S. are attempting to obtain such data.)

As a result of all these considerations, no modifications of the previous results have been made to include the effects of very high steam concentrations. Such corrections must await the completion of more appropriate experiments.

D.4.2.3 Effect of Leak Size

Leakage of radioactivity from an essentially intact containment to the atmosphere outside potentially could occur through penetrations, welds, leaky valves, flaws, and so forth, in the containment. The

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*Perhaps the most complete set of experiments involving steam environments was the set known as the Containment Systems Experiments. They involved saturated steam-air environments at both elevated temperatures (250°F) and ambient temperatures (77°F). Unfortunately, from the results of those experiments, it is not possible to determine the effects of varying temperature in a steam environment. In addition, it is not possible to adequately deduce either the effects of very high steam concentrations or the combined effects of such high concentrations and the use of sprays.

rate of escape from a given leak would depend not only on the geometry of the leak path but also on the prevailing containment conditions such as pressure and temperature. (For example, for pressures in the containment less than the design pressure, a leak rate less than the design leak rate would occur; conversely, for pressures greater than the design pressure, a leak rate greater than the design leak rate might occur.)

For a relatively small leak, there is a possibility that substantial plugging of the leak path could occur by condensation and agglomeration of escaping steam and aerosols (Vaughan, 1978; Morewitz, 1982). As a result, many small leaks might be self-sealing to some extent.* Even for those leaks which were not self-sealing, removal of both vapors and particles could occur along the leak paths and thus lower the total amount of radioactivity released to the environment. This might be especially important for rather long, complicated leakage paths. [The annulus in a RSS-type BWR (Mark I) represents an example of such a path.] In addition, shortly after some accidents, the efflux from small leaks frequently could be countered by establishing and maintaining a slight negative pressure in the containment.

In contrast to the small leak situation, for a relatively large leak (rupture) there would be no possibility of self-sealing and often little potential for decontamination along the leak path. Thus the amount of radionuclides released to the environment in accidents with large leaks or ruptures would not be noticeably reduced by leak path attenuation.

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*However, if plugging of relatively small leaks by aerosols were as effective as has been suggested (Morewitz, 1982), the net effect of such plugging might not be a decrease in the amount of aerosols leaked. To the contrary, the net effect might be an increase in the amount "leaked" because of the failure of the containment by overpressurization. In particular, for some accidents in some plants, substantial leakage of gases from the containment during the accident would tend to prevent massive failure of the containment system. Thus, in such accidents, plugging of the leak paths from the containment could increase the releases of radioactivity to the environment rather than decrease them.

Although specific potential leak paths for any plant can be identified and characterized, it is not currently possible to estimate to what extent any certain leak path would actually be important in a given accident. Indeed, the mode and the timing of containment failure are among the least certain but most important factors for estimating the consequences of any degraded-core or meltdown accident. The important leak path(s) during any accident would depend upon both the accident scenario and the detailed design and construction characteristics of the containment. Obviously, these would be both accident and plant specific.

As was the case in the subsection about aerosol concentration, the modifications to the radionuclide release estimates indicated from considering leak path attentuation are often largest for those accidents involving small releases (leaks) of material from the containment and are smallest for those accidents involving large releases (leaks) from the containment. Inasmuch as the small leak accidents are negligible contributors to the overall risk to the human population, it is not important for estimating source terms appropriate for risk assessment purposes to consider the potentially large indicated corrections for those accidents. Likewise, inasmuch as the total anticipated consequences for small leak accidents are usually so small, it is generally not important to consider the potentially large indicated corrections for individual accidents for emergency planning. Similarly, because the correction is small and poorly described for large leak accidents, it is not necessary to consider any modification of the emergency planning source terms for those accidents either.

The discussion for the risk to the equipment likewise follows that given in the subsection on high aerosol concentrations. Consequently, the effects of leak path have not been used to modify the source terms for equipment qualification either.

D.4.2.4 Effect of Chemistry

D.4.2.4.1 Elements Other Than Iodine

Relative to the core region and parts of the primary coolant system (PCS), most of the containment might be regarded as being a region of comparatively low chemical activity. However, the containment would often be a region with a considerable number of important chemical reactions occurring. To a certain extent, the compounds which entered the containment would be the same as the compounds found in the containment atmosphere at later times. This would be primarily the result of the relatively low temperatures encountered in most of the containment for many accidents. Unlike the core region and other hot parts of the PCS in which the thermodynamically preferred forms of many elements would often be anticipated, such forms would not necessarily be expected in a large part of the containment. In contrast to the chemistry of the PCS, which usually would be governed by a reducing environment, the chemistry in most of the containment might be dominated by a steam-filled, oxidizing atmosphere. (This would not be the situation in an inerted Mark I BWR if the containment had not failed.)

For a meltdown accident, at least the vicinity of the melt after RPV melt-through would typically be a region of substantial chemical activity. Because of the extremely high temperatures and the sparging of gases through the melt, the thermodynamically expected forms of many species might be produced in that region. The chemistry in the vicinity of the melt might be dominated by a much different environment than that in the rest of the containment.

If burning of large amounts of hydrogen occurred in the containment during an accident, then the temperatures throughout the containment could be increased substantially and thus many otherwise kinetically unfavored reactions might be promoted. In addition, burning of hydrogen could result in a reducing atmosphere, at least while the burning continued. Thus the burning of hydrogen could significantly affect the fates of many radionuclides.

Typically, the chemistries of most of the radionuclides of concern have not been investigated in detail for considerations of transport of radionuclides through the containment. Therefore, with the exception of iodine, they have not been considered here.

D.4.2.4.2 Iodine

As for the other elements, it could frequently be assumed that the form(s) of iodine which entered the containment would be the form(s) which were present in the containment. A potentially important exception in certain accidents in some plants would be due to the conversion of elemental iodine in solution to ionic forms as the result of chemical additives in containment sprays and/or ice beds (see Table D.2). (Post-accident responses would also often add such substances to coolant system water.) Such conversion to ionic forms would hinder escape of the iodine in solution to the containment atmosphere. The extent of conversion to ionic forms, and hence the extent of retention of iodine, in the water would depend on the acidity of the water, the total concentration of iodine in solution, the other species in solution, the temperature of the water, as well as other factors.

In general, iodine would be converted to ionic form most completely in basic and/or dilute solutions. For any accident in which the water were relatively basic, the conversion usually would result in the total iodine concentration in solution initially being at least 100 times the total iodine concentration in the atmosphere. At later times, the total iodine concentration in solution often would be 10^2 to 10^6 times the total iodine concentration in the containment atmosphere (Elrick et al., 1981).

Another exception of potential importance would be due to the formation of organic iodides in the containment. Typically, organic iodides would not constitute a large fraction of all the iodine species present in the containment. However, because organic iodides are not effectively removed from the containment atmosphere by sprays, filters, or most natural processes, they could constitute a disproportionately large fraction of the iodine escaping to the environment for some accidents.

Both the genesis and the fate of the organic iodides in the containment are poorly understood. It is thought that the organic iodides, composed mostly of methyl iodide, are formed by reaction of elemental iodine with either any of the various lubricants, paints, and other carbon-based components of the reactors or else any of the products formed by decomposition of such components. However, the reactions responsible for the formation and decomposition of the organic iodides have not been unambiguously identified.

A "realistic" assessment of organic iodide production, based on the literature, results in an estimate that approximately 0.03% of any elemental iodine present in the containment atmosphere after an accident would be converted to organic iodide (Campbell, 1981; Elrick et al., 1981). In particular, 0.02% would be converted by nonradiolytic processes and 0.01% would be converted by radiolytic ones. (The effects of the different radiation fields resulting from various possible accident conditions were not considered in the referenced report.) In general, the iodine in the containment would be expected to be mostly in the form of iodides and not in elemental form. Thus, much less than 0.03% conversion of the total iodine in the containment would be anticipated. In addition, for any accident involving only slow leakage of iodine to the environment, the relatively rapid radioactive decay of most of the iodine would substantially decrease the fraction of radioiodine which could escape as organic iodide. (Over one-half the mass of iodine released from the core materials in any accident would typically be I-129. However, this radioiodine would include less than 1×10^{-6} of the total iodine radioactivity initially released from the core.)

In this report, to estimate the total fraction of radioactive organic iodide produced, the production fraction for organic iodide is combined with the fraction of the iodine assumed to be present as elemental iodine. The amount escaping is then obtained by adjusting that fraction to account for radioactive decay.

D.4.2.5 Effect of Core Size and Status

Most of the more recently built, under-construction and planned LWR's have approximately the same size cores. Thus, within the current limitations of source term estimation, they can be assumed to have approximately the same amounts of radioactivity released into the reactor containments for similar accident conditions. In contrast, some of the first reactors built had cores approximately an order of magnitude smaller than the more recent reactors. As a result, size-dependent effects might occur at least for the older reactors.

For example, because the amounts of the radionuclides in the core inventories differ substantially among reactors, the amounts of radioactivity which potentially could be released in a given accident vary widely. Furthermore, because the volumes of nuclear plants do not scale directly with core sizes and because the mass of aerosols generated in any given accident would not necessarily be proportional to the core size, the resulting aerosol concentrations in the containment (and in the PCS) might differ substantially from plant to plant. In addition, the likelihood of containment failure for any specific accident sequence probably varies with core size. Thus the amounts of radioactivity released to the environment during a given accident sequence could differ substantially not only directly because of differences in the core inventory but also indirectly because of various other size-dependent effects.

Not only would the size of the core be important in determining the amount of radioactivity released in any accident, but also the status of the core at the time of the accident could be relevant. Although the total radioactivity of the fission products in the core of any reactor attains approximately an equilibrium value and therefore is essentially independent of the operating history of the reactor, the total mass of those fission products increases almost linearly with time during a cycle and thus depends strongly on the core's status. For example, a given accident occurring near beginning of cycle could release approximately only half as large a mass of most fission products as one occurring toward end of cycle. Thus, concentration-dependent aerosol removal processes in the containment could be affected by the core's status. For limited-core-damage accidents in which all of the aerosols were generated from the core and structural materials, concentration-dependent processes in the containment might be noticeably affected by the core's status. For core-melt accidents in which the greater part of the aerosols were not generated from core materials, the status of the core usually would have much less net effect on such processes in the containment. (In addition, for any accident, if the total mass of the aerosols released initially to the PCS were dominated by fission products, then the core's status could significantly affect the total mass of materials released to the containment if concentration-dependent aerosol removal processes in the PCS played an important role in determining the fractions of materials ultimately escaping from the PCS.)

In this study, the effect of core size is not explicitly included. However, at least for reactors with relatively small cores, it should be considered. The effect of the status of the core is included here only in the consideration of uncertainties.

D.4.2.6 Effect of Timing of Releases

The escape of radioactivity to the environment generally would depend upon the flow of gases from the containment. The driving forces determining such flow would be a strong function of both the factors affecting the release of steam to the containment from the PCS and those affecting the generation of steam within the containment, with radionuclides in the containment being most likely to escape to the environment during high steam release and production periods. For example, if the containment failed early during a meltdown accident, escape would be most likely to occur during the following periods: boiloff of the water after slumping of the core into the bottom of the RPV; evaporation of water from the reactor cavity; and "vaporization" of water and aerosols from the concrete. Between periods of large steam generation rates, the flow of radionuclides from the containment would generally be comparatively low and deposition processes could dominate the behavior of the radioactive materials. Obviously, if the containment failed during an accident, its timing relative to the timing of both the radioactive releases and the steam generation periods would be important in determining the overall radionuclide releases to the environment.

The CORRAL calculations used as the basis for the estimates in this work were performed with a prescribed set of assumptions about the timing of both the radionuclide releases from the core materials and the transport through the primary coolant system. In particular, the prescription used in the RSS was utilized. That is, for the accidents considered in this document, the radionuclide releases were divided into three basic parts: gap, meltdown (before reactor vessel failure), and vaporization (after vessel failure). In the RSS prescription, the gap release occurs instantly at the start of melting. The meltdown release occurs in 10 equal amounts at evenly spaced intervals during the melting. The vaporization release takes place in 20 intervals, with the first 10 equally spaced over the first half hour of that release and the next 10 equally spaced over the next hour and a half. One-half of the vaporization release occurs in the first 10 intervals and the other half occurs in the next 10 intervals. Within each of the two groups of 10, the releases occur in exponentially decreasing amounts.

As was noted in the last two appendices, the predicted timing of the releases in this report is somewhat different than that assumed in those previous calculations. In particular, the meltdown releases of the more volatile radionuclides would occur relatively early in the melting period while those of less volatile radionuclides would tend to occur mostly toward the end of the melting period. Likewise, the vaporization releases of the less volatile radionuclides would tend to occur mostly within the first half hour after vaporization started. (However, it should be noted that the timing of the vaporization releases is extremely uncertain.) Because the release from the containment depends critically on the timing of the initial releases, the effects of these differences must be taken into account.

In general, it can be said that the indicated problems with the RSS prescription for the timing of the releases would be insignificant for all cases in which containment failure either did not occur or else occurred very late in the accident. For all other cases, the RSS prescription generally would tend to underestimate the releases to the environment for the less volatile radionuclides. Those predicted to be released too early in the RSS meltdown release would have time to settle out before the steam generated by slumping, and possibly by contact with water in the reactor cavity, could drive them from the containment. Likewise, those predicted to be generated too late in the RSS vaporization release would have much less steam to drive them out of the containment than those releases generated earlier. The exact amount of underestimation would depend on the accident, especially on the timing of the containment failure.

In this report, for any given accident sequence, the radionuclides entering the containment indirectly via the coolant system have been assumed to behave the same as the radionuclides in the meltdown releases in CORRAL. Likewise, the radionuclides entering the containment directly have been taken to behave the same as the radionuclides in the vaporization releases in CORRAL. These assumptions have been used to estimate the basic escape fractions for all of the radionuclide groups for each accident sequence considered.

To partially compensate for the effects of the "mistiming" of the releases in the CORRAL calculations used here, the meltdown containment escape fractions for some sequences involving early containment rupture have been modified for the low volatility elements. In particular, they have been increased by up to 20%, depending upon the timing of the sequence; sequences in which containment failure would occur long after meltdown have not been modified. In addition, those sequences which would involve sufficient leakage that gross containment failure would not occur were not modified. Although it is realized that the correction applied here is somewhat arbitrary, it is not thought to be unrealistic.*

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*Simple rerunning of the CORRAL code with a more correctly timed source term, if that were possible, would not entirely account for the problems being addressed here. In particular, inasmuch as all aerosols are lumped together in the CORRAL code, multiple runs would need to be made to account for the different timing of the various aerosol components in a given accident sequence. And even such multiple runs would not completely address the basic problems.

D.4.3 Summary of Approach

The effects of the three areas of aerosol transport which were considered in this appendix were all shown to be either negligible or not addressable due to a lack of data and/or models. Thus, the basic approach of escape fraction estimation used here involved primarily a reworking of the results of previous calculations to account for the proposed modifications due to differences in three factors: the rates of the initial radionuclide releases from the core materials; the fractions of those releases retained in the primary coolant system; and the chemistry of iodine. (See Appendices B and C for more details of the first two.)

For emergency planning, the containment escape fractions were estimated for representative core-melt and limited-core-damage accidents at each of the five plants considered in the RSSMAP work. For equipment qualification, the containment escape fractions were estimated for representative limited-core-damage accidents at a single plant.

D.5 RESULTS

Inasmuch as the approach in this appendix involved primarily a manipulation of the results of previous calculations, the resulting containment escape fractions are not presented in detail in this appendix. Instead, some illustrative examples for both emergency planning and equipment qualification sequences are given in this section. The detailed results obtained in this appendix, combined with those of the previous two appendices, are presented in the following appendix, Appendix E.

In general, the primary differences between the meltdown and the vaporization containment escape fractions in this report and those in the RSSMAP reports (on which these fractions are based) are due to modifications included here to account for the effect of different assumptions about the timing of the releases into the containment than those previously used. The primary differences between the total containment escape fractions in this report and those in the RSSMAP reports depend not only on those differences in the timing but also on differences in assumptions about the fractions of the core inventory initially released in both the meltdown and the vaporization releases and in assumptions about the fractions of the meltdown releases escaping from the coolant system into the containment. The major differences between the fractions used here and those in the RSS are all the differences just mentioned for the RSSMAP work plus differences due to both modifications in the methods for performing the thermal-hydraulic calculations used to develop input for CORRAL and some different basic assumptions used to describe certain accident sequences.

D.5.1 Emergency Planning

As an illustration of the approach used in this report, the fractions of the materials released to the containment which would escape to the environment for each of the dominant emergency planning sequences for each of the two RSS reactors are listed in Table D.7. As can be seen, for accident sequences dominating the emergency planning spectrum, substantial fractions of the materials reaching the containment would escape. (It should be noted that the fractions escaping for many other accident sequences would be much less than those for the dominant sequences.)

D.5.2 Equipment Qualification

For most accidents of concern for equipment qualification, the containment would not rupture massively and the amounts of all radionuclides escaping would be much less than 1% of the amounts released to the containment (see Table D.8). Inasmuch as the amounts escaping from the containment would be much lower than the amounts of the radionuclides reaching the containment, the fractions escaping could typically be neglected for equipment qualification purposes. Of much more importance for equipment qualification would be the distributions of radioactive materials within the containment and these have not been dealt with in detail in this report.

Table D.7. Containment escape fractions for some potentially important emergency planning sequences at the RSS reactors[a]

Sequence	Containment escape fraction[b]				Potential effect on aerosols released	
	Meltdown			Vaporization		
	Xe	I[c]	Aerosols	Aerosols	Concentration	Timing[d]
RSS PWR						
V	1.00	0.70	0.94	0.32	No	No
TMLB′-δ	1.00	0.33	0.43	0.10	No	Yes
TMLB′-γ	1.00	0.33	0.43	0.10	No	Yes
S_2C-δ	1.00	0.056	0.35	0.16	No	Yes
TMLB′-ε	0.86	0.003	0.0047	0.0045	Yes	No
RSS BWR						
TW-γ′	1.00	0.10	0.21	0.40	No	Yes
TC-γ′	1.00	0.47	0.67	0.60	No	Yes
TQUV-γ′	1.00	0.10	0.30	0.40	No	Yes
TW-γ	1.00	0.003	0.10	0.10	No	Yes
TC-γ	1.00	0.072	0.14	0.10	No	Yes
TQUV-γ	1.00	0.02	0.05	0.12	No	Yes

[a]See Table D1.6 in the addendum of this appendix for a summary of the emergency planning accident sequences at each reactor considered. See Appendix A for a description of each of these sequences.

[b]For sequences other than involving the V event, these fractions apply to the amounts released into the containment and do not include the effects of the modifications indicated at the right; for the V event, these fractions apply to the amounts released from the core materials and represent the fractions escaping to the environment either from the containment for the meltdown release or from the auxiliary building for the vaporization release.

[c]Represents fraction of iodine present as elemental iodine which escapes.

[d]This effect is taken to apply only to the timing of releases prior to RPV melt-through.

Table D.8. Containment escape fractions for equipment qualification "sequences"

Sequence	Containment escape fraction[a,b]				Potential effect on aerosols[b]	
	Meltdown			Vaporization		
	Xe	I[c]	Aerosols	Aerosols	Concentration	Timing
Large LOCA(A)	1(-4)[d]	2(-5)	1(-5)	-	No	No
Large LOCA with delayed emergency core cooling	1(-4)	2(-5)	1(-5)	-	No	Yes
Transient with delayed emergency core cooling	1(-4)	2(-5)	1(-5)	-	No	Yes
Meltdown	6(-3)	2(-5)	1(-5)	1(-5)	Yes	Yes

[a]These fractions apply to the amounts released into the containment. The containment isolation system is assumed to function properly.

[b]Multiply correction factors for these effects times the previously estimated aerosol containment escape fractions.

[c]Represents fraction of iodine presented elemental iodine which escapes.

[d]1(-4) denotes 1×10^{-4}.

D.6 UNCERTAINTIES AND LIMITATIONS

As has been indicated in the previous sections, there are many sources of uncertainties associated with both the containment escape fractions presented in this appendix and containment escape fractions in general. Unfortunately, the potential impacts of some of those uncertainties on the estimated escape fractions are large. The magnitudes of the potential impacts of some of those sources of uncertainties are outlined in Table D.9. A related discussion of the general sources of uncertainties is presented in the following appendix.

As is the situation for many aspects of estimating source terms for reactor accidents, there are rather large uncertainties associated with the consideration of radionuclide transport within and escape from the containment. Some of the uncertainties are associated with the impossibility of exactly predicting the scenario a given accident would follow. For example, for any meltdown accident, the mode and timing of containment failure cannot be predicted definitely but instead must be estimated on the basis of engineering judgment. Thus the descriptions used in this report (and in other reports) to consider various accident sequences must be acknowledged to be merely hypothetical constructs which have been developed to gain understanding of the important factors affecting the source terms and the associated consequences.

Because of the somewhat arbitrary nature of the definition and choice of the accident sequences for consideration, the problems with many of the detailed models used to consider parts of those accidents are not necessarily overwhelmingly important for the overall description of any accident. For example, although the effect of temperature on the scrubbing of aerosols by suppression pools is not yet definitely known, that lack of knowledge is not as important in many cases as the inability to predict whether the radionuclides would even flow through the water being considered.

Among the more important factors affecting the potential consequences of many accidents would be the amounts of aerosols generated, the timing of the radionuclide releases during the accident, and the timing and mode of containment failure if it occurred. All of these

Table D 9 Some major sources of uncertainty in containment transport considerations

Problem area	Effect	Problems in data	Problems in models and/or codes	Potential magnitude of effect on individual sequences: Equipment qualification: Limited-core damage	Equipment qualification: Meltdown[a]	Emergency planning: Limited-core damage	Emergency planning: Meltdown
High aerosol concentrations	High aerosol concentrations are agglomerated and deposited relatively much more rapidly than low concentrations	Aerosol generation rates are highly uncertain	Aerosol aging in CORRAL is independent of total aerosol concentration	Small	Large only for distribution within containment for radionuclides other than noble gases	Small	Large for radionuclides other than noble gases for some sequences
High steam concentrations	High steam environments accelerate aerosol deposition	Data not available for condensing steam models	CORRAL does not include the effects of very high steam concentrations	Small	Large only for distribution within containment for radionuclides other than noble gases	Small	Large for radionuclides other than noble gases for some sequences
Timing of radionuclide releases	Radionuclide releases just prior to large releases of gases from the containment are more likely to be released from the containment	Radionuclide release rates from the core materials are uncertain	Timing of melt and vaporization releases in CORRAL is incorrect for most radionuclides	Large	Large for some radionuclides in some sequences	Large	Large for some radionuclides in some sequences
Differences among reactors (core, sizes, construction,)	Construction operation and design variations can result in different types of dominant accident sequences and different magnitudes for the associated releases	Few site-specific data are available	Only differences due to differences in ESF's present are addressed	Large	Large	Large	Large
Timing of containment failure	Radionuclides released to the containment just prior to containment failure are most likely to be released to environment	---	Description of containment failure is highly uncertain, it is not modeled but rather is based on engineering judgment	-	Large	-	Large
Chemistry	Chemical reactions change chemical forms and thus affect the fates of radionuclides	Adequate physiochemical data not available for all species	No codes explicitly consider chemical reactions	Large for distribution within containment	Large for distribution within containment	Large	Large

[a]Meltdown accidents typically would be considered for equipment qualification They generally have not been considered in the work for this project They have been included in this table for completeness

factors are highly uncertain. Certain variations of each of these factors could affect the estimated releases substantially.

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ADDENDUM D1

SUMMARY OF BACKGROUND ESTIMATES

D-54

Table D1.1. Rebaselined RSS PWR CORRAL results

Sequence[a]	RSS category	Probability[b]	Cumulative fractions of core inventory released to the atmosphere[c]						
			Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
V	PWR 2	$2 \times 10^{-6} (4 \times 10^{-6})$	1.0	0.642	0.820	0.410	0.097	0.044	0.006
S_2C-δ	PWR 3	2×10^{-6}	1.0	0.052	0.332	0.188	0.036	0.018	0.003
TMLB'-α	PWR 1	3×10^{-8}	1.0	0.484	0.527	0.361	0.063	0.402	0.002
TMLB'-δ	PWR 2	$3 \times 10^{-6} (2 \times 10^{-6})$	1.0	0.308	0.388	0.145	0.044	0.018	0.002
TMLB'-ε	PWR 6	6×10^{-7}	0.7	4(-4)[d]	0.001	0.001	1(-4)	7(-5)	1(-5)
S_2D-β	PWR 5	2×10^{-8}	1.0	0.002	0.009	0.007	0.001	6(-4)	9(-5)
S_2D-α	PWR 3	9×10^{-8}	1.0	0.263	0.321	0.272	0.040	0.295	0.002
AHF-α	PWR 1	$<1 \times 10^{-11}$	1.0	0.369	0.439	0.313	0.057	0.323	0.002
AHF-γ	PWR 2	2×10^{-11}	1.0	0.148	0.236	0.126	0.029	0.013	0.002
AHF-δ	PWR 2	$<1 \times 10^{-11}$	1.0	0.140	0.224	0.129	0.028	0.013	0.002
AHF-ε	PWR 6	1×10^{-10}	0.7	2(-4)	7(-4)	8(-4)	8(-5)	6(-5)	1(-5)

[a]The sequence notation for this reactor is explained in Table A1.1 in Appendix A.

[b]Taken from RSS unless value changed during RSS rebaselining; in those cases, value in parentheses is the RSS value.

[c]Based on a private communication (Cybulskis, 1981; Wooton, 1981).

[d]The notation 5(-5) is an abbreviation of 5×10^{-5}.

Table D1.2. Rebaselined RSS BWR CORRAL results

Sequence[a]	RSS category	Probability[b]	Cumulative fractions of core inventory released to the atmosphere[c]						
			Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
AE-α	BWR 1	5×10^{-9} (2×10^{-9})	1.0	0.29	0.57	0.42	0.071	0.29	0.004
AF-α_2	BWR 1	1×10^{-9} (2×10^{-9})	1.0	0.024	0.04	0.3	0.0025	0.28	0.002
TC-γ'	BWR 2	2×10^{-6} ($<1 \times 10^{-7}$)	1.0	0.45	0.67	0.64	0.073	0.052	0.0083
TW-γ'	BWR 2	3×10^{-6}	1.0	0.098	0.27	0.41	0.025	0.028	0.005
TQUV-γ'	BWR 2	3×10^{-7} (8×10^{-8})	1.0	0.095	0.30	0.36	0.034	0.027	0.005
TC-γ	BWR 3	8×10^{-6} (1×10^{-5})	1.0	0.07	0.14	0.12	0.015	0.01	0.002
TQUV-γ	BWR 3	1×10^{-6} (4×10^{-7})	1.0	0.02	0.055	0.11	0.006	0.007	0.0013
TW-γ	BWR 3	1×10^{-5}	1.0	0.003	0.011	0.083	0.011	0.007	0.001

[a]The sequence notation for this reactor is explained in Table A1.2 in Appendix A.

[b]Taken from RSS unless value changed during RSS rebaselining; in those cases, value in parentheses is the RSS value.

[c]Based on a private communication (Cybulskis, 1981; Wooton, 1981).

Table D1.3. Ice condenser PWR CORRAL results[a]

Sequence[b]	RSS category	Probability	Cumulative fractions of core inventory released to the atmosphere						
			Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
AD-α[c,d]	PWR 1	2×10^{-9}	1.0	0.23	0.37	0.42	0.045	0.41	0.003
AD-γ[c]	PWR 5	2×10^{-7}	1.0	5(-4)[e]	6(-4)	0.003	4(-5)	2(-4)	3(-5)
AD-γ[f]	PWR 5	2×10^{-7}	1.0	0.002	0.006	0.026	3(-4)	0.002	3(-4)
AD-δ[c]	PWR 7	$<2 \times 10^{-8}$	1.0	7(-7)	3(-7)	3(-7)	3(-8)	2(-8)	4(-9)
S_2H-$\gamma\theta$[c]	PWR 4	2×10^{-5}	1.0	0.005	0.040	0.15	0.003	0.009	0.002
S_2HF-α	PWR 1	5×10^{-10}	1.0	0.27	0.68	0.41	0.081	0.43	0.003
S_2HF-γ	PWR 2	5×10^{-6}	1.0	0.13	0.57	0.49	0.068	0.042	0.007
S_2HF-δ	PWR 3	$<5 \times 10^{-7}$	1.0	0.010	0.23	0.34	0.024	0.024	0.004
S_2HF-$\delta\theta$	PWR 3	$<5 \times 10^{-7}$	1.0	0.045	0.26	0.13	0.033	0.015	0.002
TMLB'-δ[c]	PWR 4	$<3 \times 10^{-8}$	1.0	0.007	0.014	0.050	0.001	0.003	6(-4)
TMLB'-$\delta\theta$[c]	PWR 3	3×10^{-7}	1.0	0.063	0.11	0.089	0.013	0.008	0.001
TMLB'-$\delta\theta$[f]	PWR 3	3×10^{-7}	1.0	0.063	0.14	0.095	0.017	0.009	0.001
TML-γ[c]	PWR 3	3×10^{-6}	1.0	0.016	0.080	0.31	0.005	0.019	0.004
V[g]	PWR 2	5×10^{-6}	1.0	0.77	0.80	0.51	0.093	0.049	0.007
V[h]	PWR 2	5×10^{-6}	1.0	0.53	0.50	0.32	0.058	0.030	0.004
V[h,i]	PWR 2	5×10^{-6}	1.0	0.48	0.42	0.086	0.052	0.016	0.002

[a] Adapted from Carlson et al. (1981).

[b] The sequence notation for this reactor is explained in Table A1.3 in Appendix A.

[c] Ice bed decontamination factor of 100 for iodine and particulates under 90% of the ice has melted.

[d] Ice bed bypassed after the steam explosion.

[e] The notation 5(-5) is an abbreviation for 5×10^{-5}.

[f] Ice bed decontamination factor of 10 for iodine and particulates until 90% of the ice has melted.

[g] Assuming release through the UHI equipment room.

[h] Assuming release through the auxiliary building.

[i] With the air return fans and ice condenser operating after reactor vessel melt-through.

Table D1.4. Mark III BWR CORRAL results[a]

Sequence[b]	RSS category	Probability	Cumulative fractions of core inventory released to the atmosphere						
			Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
TPQI-δ[c]	BWR 2	5×10^{-6}	1.0	0.57	0.52	0.31	0.058	0.030	0.0044
TQW-δ[c]	BWR 2	2×10^{-5}	1.0	0.21	0.58	0.55	0.063	0.044	0.0072
AC-δ[c]	BWR 2	---	1.0	0.50	0.67	0.25	0.080	0.032	0.0039
TQUV-γ[d]	BWR 3	8×10^{-7}	1.0	0.033	0.17	0.49	0.014	0.030	0.0058
TQUV-δ[e]	BWR 4	8×10^{-7}	1.0	4.4-4	6.2-3	0.016	5.1-4	9.8-4	1.9-4

[a]Adapted from Hatch, Cybulskis, and Wooton (1981).

[b]The sequence notation for this reactor is explained in Table A1.4 of Appendix A.

[c]No scrubbing of fission products assumed due to high suppression pool temperature.

[d]Containment failure due to hydrogen burn at head failure; gap and melt release scrubbed by pool; no fission product scrubbing by pool after containment failure.

[e]Cold suppresssion pool; containment failure delayed 4 hrs. after head melting; fission products scrubbed by pool.

Table D1.5. Alternate large containment PWR CORRAL results[a]

Sequence[b]	RSS category	Probability	Cumulative fractions of core inventory released to the atmosphere						
			Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
$T_1(B_3)MLUOO'$-δ[c]	PWR 2	2×10^{-8}	1.0	0.54	0.74	0.64	0.082	0.054	0.0085
TMLU-γ[d]	PWR 3	2×10^{-6}	1.0	0.035	0.17	0.58	0.0091	0.035	0.0069
TMLU-γ[e]	PWR 3	2×10^{-6}	1.0	0.45	0.74	0.70	0.081	0.056	0.0090
V	PWR 2	$<4 \times 10^{-6}$	1.0	0.48	0.79	0.44	0.092	0.045	0.0063
AYF-δ	PWR 2	$<1 \times 10^{-8}$	1.0	0.23	0.67	0.46	0.076	0.042	0.0063
AG-δ	PWR 2	$<1 \times 10^{-8}$	1.0	0.16	0.76	0.71	0.083	0.058	0.0094

[a] Adapted from Kolb et al. (1981).

[b] The sequence notation for this reactor is explained in Table A1.5 of Appendix A.

[c] Containment fails from rapid boiloff of water in reactor cavity.

[d] Containment fails from hydrogen burn at head failure. Case shows effect of containment spray scrubbing.

[e] Containment fails from hydrogen burn at head failure. Case shows effect of no containment spray scrubbing.

Table D1.6. Summary of release categories, probabilities and dominant sequences[a]

Reactor	RSS category	Total probability[b]	Dominant sequence and probabilities[c]		Corresponding RSS sequence
RSS PWR	PWR 1	2×10^{-7}	--	--	--
	PWR 2	8×10^{-6}	V	4×10^{-6}	V
			TMLB'-δ	2×10^{-6}	TMLB'
			TMLB'-γ	7×10^{-7}	TMLB'
	PWR 3	4×10^{-6}	S_2C-δ	2×10^{-6}	S_2C
	PWR 4	$<2 \times 10^{-7}$	--	--	--
	PWR 5	$<2 \times 10^{-7}$	--	--	--
	PWR 6	2×10^{-6}	TMLB'-ε	6×10^{-7}	TMLB'
	PWR 7	4×10^{-5}	S_2D-ε	9×10^{-6}	S_2D
			S_2H-ε	6×10^{-6}	S_2H
			TML-ε	6×10^{6}	TML
Ice Condenser PWR	PWR 1	$<5 \times 10^{-7}$	--	--	--
	PWR 2	1×10^{-5}	S_2HF-γ	5×10^{-6}	S_2HF
			V	5×10^{-6}	V
	PWR 3	3×10^{-5}	S_2H-γ	2×10^{-5}	S_2H
			S_1HF-γ,δ	3×10^{-6}	S_1HF
			TML-γ	3×10^{-6}	TML
	PWR 4	2×10^{-5}	S_1H-γ	1×10^{-5}	S_1H
			S_2D-γ	6×10^{-6}	S_2D

Table D1.6. (continued)

Reactor	RSS category	Total probability[b]	Dominant sequence and probabilities[c]		Corresponding RSS sequence
Ice condenser PWR (continued)					
	PWR 5	5×10^{-6}	S_1D-γ	4×10^{-6}	S_1D
	PWR 7	$<4 \times 10^{-6}$	S_2H-δ	$<2 \times 10^{-6}$	S_2H
Alternate PWR	PWR 1	1×10^{-7}	--	--	--
	PWR 2	1×10^{-5}	V	4×10^{-6}	V
	PWR 3	3×10^{-5}	T_2MQ-FH-γ	3×10^{-6}	S_2FH
			S_3FH-γ	2×10^{-6}	S_1FH, S_2FH[d]
			T_2MQ-H-γ	6×10^{-6}	S_2H
			S_3H-γ	5×10^{-6}	S_1H, S_2H[d]
			T_2MLUO-γ	4×10^{-6}	TML
			T_2KMU-γ	4×10^{-6}	TKMU
	PWR 4	1×10^{-7}	--	--	--
	PWR 5	5×10^{-7}	--	--	--
	PWR 6	7×10^{-6}	T_2MQ-FH-ε	3×10^{-6}	S_2FH
			S_3FH-ε	2×10^{-6}	S_1FH, S_2FH[d]
	PWR 7	4×10^{-5}	T_2MQ-H-ε	6×10^{-6}	S_2H
			S_3H-ε	5×10^{-6}	S_1H, S_2H[d]
			S_1D-ε	5×10^{-6}	AD

Table D1.6. (continued)

Reactor	RSS category	Total probability [b]	Dominant sequence and probabilities [c]		Corresponding RSS sequence
RSS BWR	BWR 1	$<5 \times 10^{-7}$	--	--	--
	BWR 2	7×10^{-6}	TW-γ'	3×10^{-6}	TW
			TC-γ'	3×10^{-6}	TC
	BWR 3	2×10^{-5}	TW-γ	1×10^{-5}	TW
			TC-γ	1×10^{-5}	TC
	BWR 4	$<1 \times 10^{-7}$	--	--	--
Mark III BWR	BWR 1	1×10^{-7}	--	--	--
	BWR 2	3×10^{-5}	TQW-δ	2×10^{-5}	TW
			TPQI-δ	5×10^{-6}	TPW
			TC-δ	5×10^{-6}	TC
			SI-δ	5×10^{-6}	SI, SJ
	BWR 3	1×10^{-6}	TQUV-γ	8×10^{-7}	TQUV
	BWR 4	1×10^{-6}	TQUV-δ	8×10^{-7}	TQUV

[a] Based on RSS and RSSMAP results.

[b] Probability per reactor year; no smoothing has been applied to these probabilities.

[c] The accident sequence notation used here is explained in Tables A1.1-A1.5 in Appendix A.

[d] See Table 4.1 of Kolb et al. (1981).

Table D1.7. Summary of estimated probabilities of containment failure modes[a]

Reactor	Meltdown probability	Containment failure mode probability (per reactor year)					
		Steam explosion	Hydrogen burning	Over-pressurization	Basemat melt-through	Leakage	By-pass
RSS PWR[b]	5×10^{-5}	$<2 \times 10^{-7}$	1×10^{-6}	4×10^{-6}	4×10^{-5}	$<4 \times 10^{-7}$	4×10^{-6}
IC PWR	7×10^{-5}	$<5 \times 10^{-7}$	6×10^{-5}	$<8 \times 10^{-6}$[c]	0[d]	---[e]	5×10^{-6}
Alternate PWR	8×10^{-5}	1×10^{-7}	4×10^{-5}	$<8 \times 10^{-7}$	4×10^{-5}	6×10^{-7}	$<4 \times 10^{-6}$
RSS BWR[b]	3×10^{-5}	$<5 \times 10^{-7}$	0	3×10^{-5}	0[d]	$<5 \times 10^{-7}$	0
Mark III BWR	4×10^{-5}	1×10^{-7}	1×10^{-6}	4×10^{-5}	0[d]	$<3 \times 10^{-7}$	0

[a]Based on the RSS and the RSSMAP reports.

[b]The total probabilities given here for the RSS reactors are not those given in the RSS but rather are those obtained by straight summation. (The summations of the probabilities in the RSS were calculated using a Monte Carlo technique whereas those in the RSSMAP work used straight addition. For a given reactor, the total probabilities estimated by the Monte Carlo technique tend to be higher than those obtained by addition). In addition, the smoothing technique used in the RSS was not utilized here.

[c]The probability for overpressurization failure late in any accident sequence was not estimated in the RSSMAP report for an ice condenser PWR. Instead, that probability was merely bounded for each sequence considered. Thus this number represents an upper bound of the total probability in the sense that it is based on an overestimate of the probability of overpressurization failure for each of the accident sequences included. (However, it may not be an overestimate of the overall probability because not all possible sequences were considered in obtaining the total).

[d]Although basemat melt-through might eventually occur at this type of reactor, some other type of containment failure is always predicted to precede melt-through.

[e]Leakage was not considered in the RSSMAP report for an ice condenser PWR.

APPENDIX E

SOURCE TERMS

E.1 INTRODUCTION

This appendix combines and summarizes the results of the last three appendices. In particular, it presents the "best estimates" for the source terms obtained in this work. In addition, it discusses the general factors affecting the uncertainties of those results. Detailed considerations of both the results and some specific factors affecting the associated uncertainties have been presented in the last three appendices.

E.2 EMERGENCY PLANNING

The accidents dominating the risk to the human population all would involve complete meltdown of the reactor core. The fractions of the core inventory predicted to be lost from the containment for many types of meltdown sequences are presented in Tables E1.1 through E1.5 in the addendum of this appendix. The five classes of accident sequences which were formed for emergency planning by consideration of the values in those tables, along with another class formed by consideration of limited-core-damage accidents, are described in Table E.1. In particular, the magnitudes of the radionuclide releases for each of the classes as estimated in this report are summarized in that table. For comparison, the magnitudes of the radionuclide releases for each of those classes as determined by consideration of the RSSMAP and the associated RSS-rebaselining work and by consideration of the RSS are presented in Tables E.2 and E.3, respectively. The probabilities associated with each of the classes for each of the five reactors investigated in this report are listed in Table E.4.

According to all the tables, the amounts of radionuclides estimated to be released to the environment in all meltdown accident sequences vary from substantial fractions (40-100%) of the entire core inventory of all the more volatile radionuclides and significant fractions (1-10%) of all the less volatile radionuclides to very small fractions (less

Table E.1 Calculated ranges of updated total containment escape fractions for emergency planning accident classes

Class	Corresponding RSS categories	Cumulative fractions of core inventory released to atmosphere[a]						
		Xe-Kr	I-Br[b]	Cs-Rb	Te-Sb	Ba-Sr[c]	Ru[d]	La
PWR I′	PWR 2 (V only)[e]	1.0	0.3-0.8	0.7-0.9	0.7-0.9	0.5 0.2	0.03 0.2	0.004-0.005
PWR I[f]	PWR 1	-	-	-	-	-	-	-
PWR II	PWR 2 + PWR 3	1.0	0.01-0.3	0.01-0.6	0.01-0.6	0.02-0.5 0.003-0.2	0.003-0.04 0.03-0.3	0.001-0.01
PWR III	PWR 4 + PWR 5	1.0	0.002-0.006	0.0005-0.01	0.0005-0.01	0.001-0.01 1(-4)-0.004	0.0002-0.003 0.008-0.03	6(-5)[g]-0.002
PWR IV	PWR 6 + PWR 7	1.0	7(-7)-0.003	3(-7)-0.001	3(-7)-0.001	2(-7)-4(-4) 6(-8)-2(-4)	6(-9)-3(-5) 6(-8)-3(-4)	6(-10)-1(-5)
PWR V[h]	PWR 8 + PWR 9	1.0	5(-7)-5(-4)	1(-6)-1(-3)	3(-7)-3(-4)	2(-8)-2(-5) 1(-8)-1(-5)	1(-9)-1(-6) 1(-8)-1(5)	3(-11)-3(-8)
BWR I′	BWR 2 (TPQI-δ,..)	1.0	0.5-0.6	0.4-0.8	0.4-0.7	0.06-0.10 0.02-0.05	0.007-0.010 0.05-0.09	0.002-0.003
BWR I[f]	BWR 1	-	-	-	-	-	-	-
BWR II	BWR 2 + BWR 3	1.0	0.2-0.50	0.03-0.20	0 03-0.30	0.01-0.10 0.002-0.06	0.002-0.02 0.02-0.10	0.001-0.007
BWR III	BWR 4	1.0	0.0004-0.0008	0.002-0.005	0.005-0.006	0.001-0.003 0.0001-0.0005	0.0003-0.0004 0.002-0.003	0.0001-0.0002
BWR IV[h]	BWR 5	1.0	3(-10)	1(-8)	2(-9)	2(-10) 1(-10)	1(-11) 1(-10)	3(-13)

[a]The ranges given here for each class are not all-inclusive; instead they are representative of those sequences explicitly considered in this report.

[b]Represents fraction of iodine present as elemental iodine which escapes.

[c]Upper set of values is based on barium data; lower set of values is based on strontium data.

[d]Upper set of values on ruthenium data; lower set of values is based on molybdenum data.

[e]The only potentially dominant sequences in this class are for V-type accidents. In general, the estimated escape fractions for such accidents are at the upper ends of the indicated ranges.

[f]Steam explosions were not considered in this report.

[g]6(-5) denotes 6×10^{-5}.

[h]Includes "gap" releases only.

Table E.2 Calculated ranges of RSS-rebaseline total containment escape fractions for emergency planning accident classes

Class	RSS categories	Cumulative fractions of core inventory released to atmosphere[a]						
		Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
PWR I′	PWR 2	1.0	0.6	0.8	0.4	0.1	0.04	0.006
PWR I	PWR 1	1.0	0.3-0.5	0.3-0.5	0.3-0.4	0.04-0.06	0.3-0.4	0.002
PWR II	PWR 2 + PWR 3	1.0	0.05-0.3	0.2-0.4	0.1-0.2	0.03-0.04	0.01-0.02	0.002-0.003
PWR III	PWR 4 + PWR 5	1.0	0.002-0.01	0.01-0.04	0.01-0.03	0.001-0.005	1(-3)-0.003[b]	1(-4)-4(-4)
PWR IV	PWR 6 + PWR 7	0.006-0.7	2(-5)-4(-4)	1(-5)-0.001	2(-6)-0.001	1(-6)-1(-4)	1(-6)-1(-4)	2(-7)-1(-5)
PWR V	PWR 8 + PWR9[c]	3(-6)-0.002	1(-7)-1(-4)	6(-7)-5(-4)	1(-9)-1(-6)	1(-11)-1(-8)	0	0
BWR I	BWR 1	1.0	0.2-0.3	0.4-0.6	0.4	0.05-0.07	0.3	0.003-0.004
BWR II	BWR 2 + BWR 3	1.0	0.003-0.5	0.01-0.7	0.1-0.6	0.01-0.07	0.01-0.05	0.001-0.008
BWR III	BWR 4	1.0	0.001	0.005	0.004	0.001	0.001	0.0001
BWR IV	BWR 5[c]	5(-4)	6(-11)	4(-9)	8(-12)	8(-14)	0	0

[a]The ranges indicated are not meant to be all-inclusive; instead they are representative of the calculated values for the sequences considered in the RSS-rebaselining work.

[b]1(-3) denoted 1×10^{-3}.

[c]Not considered in the RSS-rebaselining work.

Table E.3 Calculated ranges of RSS total containment escape fractions for emergency planning accident classes

Class	RSS categories	Cumulative fractions of core inventory released to atmosphere[a]						
		Xe-Kr	I-Br[a]	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
PWR I´	PWR 2	-	-	-	-	-	-	-
PWR I	PWR 1	0.9	0.7	0.4	0.4	0.05	0.4	0.003
PWR II	PWR 2 + PWR 3	0.8-0.9	0.2-0.7	0.2-0.5	0.3	0.02-0.06	0.02-0.03	0.003-0.004
PWR III	PWR 4 + PWR 5	0.3-0.6	0.03-0.09	0.01-0.04	0.005-0.03	0.001-0.005	1(-3)-0.003[b]	1(-4)-4(-4)
PWR IV	PWR 6 + PWR 7	0.006-0.3	2(-5)-0.001	1(-5)-0.001	2(-5)-0.001	1(-6)-1(-4)	1(-6)-1(-4)	2(-7)-1(-5)
PWR V	PWR 8 + PWR 9	3(-6)-0.002	1(-7)-1(-4)	6(-7)-5(-4)	1(-9)-1(-6)	1(-11)-1(-8)	0	0
BWR I	BWR 1	1.0	0.4	0.4	0.7	0.05	0.5	0.005
BWR II	BWR 2 + BWR 3	1.0	0.1-0.9	0.1-0.5	0.3	0.01-0.10	0.02-0.03	0.004
BWR III	BWR 4	0.6	8(-4)	5(-3)	4(-3)	6(-4)	6(-4)	1(-4)
BWR IV	BWR 5	5(-4)	6(-11)	4(-9)	8(-12)	8(-14)	0	0

[a]The ranges shown tend to be truncated on the lower sides because of the use of the composite RSS category results to obtain these ranges.

[b]1(-3) denotes 1×10^{-3}.

Table E.4 Probabilities of emergency planning accident classes

Class	RSS categories	Probabilities (per reactor year)[a,b]				
		RSS PWR	RSS BWR	Ice condenser PWR	Mark III BWR	Alternate PWR
I′	PWR 2	4×10^{-6}	-	5×10^{-6}	-	4×10^{-6}
I	PWR 1; BWR 1	2×10^{-7}	$<5 \times 10^{-7}$	$<5 \times 10^{-7}$	1×10^{-7}	1×10^{-7}
II	PWR 2 + PWR 3; BWR 2 + BWR 3	8×10^{-6}	3×10^{-5}	4×10^{-5}	3×10^{-5}	4×10^{-5}
III	PWR 4 + PWR 5; BWR 4	$<4 \times 10^{-7}$	$<1 \times 10^{-7}$	3×10^{-5}	1×10^{-6}	6×10^{-7}
IV	PWR 6 + PWR 7	4×10^{-5}	-	$<4 \times 10^{-6}$	-	5×10^{-5}
V	PWR 8 + PWR 9; BWR 5	4×10^{-4}	1×10^{-4}			

[a]Based on the RSS and the RSSMAP reports.

[b]It should be recognized that these probabilities are highly uncertain and have limited, if any, applicability to other reactor designs.

than 0.1%) of all the radionuclides.* As a result, the associated consequences can be expected to vary widely.

As can be seen by comparing the results given in Tables E.1 through E.3, there are two major differences between the previously estimated source terms for emergency planning and those obtained in this report. First, the estimated releases to the environment for the Ba-Sr element group are generally somewhat larger in this work for all the accident classes. Second, the releases for the class including V-type accidents are noticeably larger for most element groups in the work reported here.

From the probabilities in Table E.4, it can be seen that the probabilities of occurrence of the more severe emergency planning classes vary more noticeably among reactors than the total probability of meltdown at each of the five reactors. Thus, the overall risk to the human population may differ significantly among the considered reactors.

E.3 EQUIPMENT QUALIFICATION

The accidents currently of interest for equipment qualification range from accidents involving extremely small releases of radioactivity from the core to those involving potentially very large releases from the core. The fractions of the core inventory predicted in this study to be released to the containment for representative accident sequences from each of two basic classes of equipment qualification sequences are presented in Table E.5. For comparison, the fractions of the core inventory which are assumed in current regulations to be released to the containment are listed in Tables E.6 and E.7.

According to the tables, the amounts of the radionuclides estimated to be released to the containment in limited-core-damage accidents vary from large fractions of the more volatile radionuclides and significant fractions of the less volatile radionuclides to relatively small fractions of all the radionuclides.* In addition, the anticipated

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*In general, the smallest anticipated releases could be much smaller than those indicated in the tables.

Table E.5. Updated release fractions for equipment qualification classes

Class	Initial distribution	Fractions of core inventory released to containment but not to environment						
		Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
A	Waterborne	0.05	0.01	0.01	0.001	1(-4)[a]	1(-6)[a]	1(-7)
	Waterborne[b]	0.95	0.05	0.01	0.01	0.001	0.001	1(-4)
B[c]	Waterborne	1.00	1.00	1.00	0.50	0.05	0.01	1(-4)
	Waterborne[b]	0.00	0.00	0.00	0.01	0.001	0.001	1(-4)
	Airborne	1.00	1.00	1.00	0.50	0.05	0.01	1(-4)
	Waterborne[b]	0.00	0.00	0.00	0.01	0.001	0.001	1(-4)

[a]1(-4) denotes 1×10^{-4}.

[b]The values on this line represent the additional releases due to leaching of the core materials.

[c]This class also includes all various intermediate possibilities of initial distribution between the two indicated ones.

Table E.6. Current Regulatory Guide 1.89 release fractions for equipment qualification classes

Class	Distribution	Fractions of core inventory released to containment but not to environment						
		Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
A	Airborne	0.10[a]	0.05	0.00	0.00	0.00	0.00	0.00
	Waterborne	0.00	0.10	0.00	0.00	0.00	0.00	0.00
	Plateout	0.00	0.05	0.00	0.00	0.00	0.00	0.00
B	Airborne	1.00	0.25	0.00	0.00	0.00	0.00	0.00
	Waterborne	0.00	0.50	0.01	0.01	0.01	0.01	0.01
	Plateout	0.00	0.25	0.01	0.01	0.01	0.01	0.01

[a]30% of the Kr-85 is also included.

Table E.7. Draft Regulatory Guide 1.89 release fractions for equipment qualification classes

Class	Distribution	Fractions of core inventory released to containment but not to environment						
		Xe-Kr	I-Br	Cs-Rb	Te-Sb	Ba-Sr	Ru	La
A	Waterborne	0.10[a]	0.10	0.00	0.00	0.00	0.00	0.00
	Waterborne	1.00	0.50	0.50	0.01	0.01	0.01	0.01
B	Waterborne	1.00	0.50	0.00	0.00	0.00	0.00	0.00
	Waterborne	0.00	0.00	0.50	0.01	0.01	0.01	0.01
C	Airborne	1.00	1.00	1.00	1.00	0.11	0.08	0.013

[a]30% of the Kr-85 is also included.

initial distribution of the radioactivity in the containment for at least the largest radionuclide releases varies from having most of the radioactivity in the atmosphere to having most of it in the water. Thus the resulting radiation fields in the containment can be expected to vary widely because of both the sizes of the potential releases and their distributions within the containment.

As can be seen by comparing Tables E.5 and E.6, there are two primary differences between the previously assumed source terms for equipment qualification and those obtained in this report. First, the estimated releases to the containment for most element groups are typically somewhat larger in this report. Second, the source terms for the class of accidents involving partial melting of the core have a much wider range of possible initial spatial distributions within the containment in this work.

E.4 UNCERTAINTIES AND LIMITATIONS

Within the constraints imposed by the limited scale of this project, the updated source term estimates presented here represent best estimates for emergency planning and equipment qualification. However, it must be acknowledged that the source term estimates are associated with large uncertainties for a variety of reasons. Many of the contributors to these uncertainties have been discussed in detail in the previous three appendices. Because an appreciation of the uncertainties is required to place the updated source terms in an appropriate perspective, it is suggested that the reader refer to the detailed discussions in those appendices, as well as to the following discussion of general sources of uncertainties.

E.4.1 Introduction

Uncertainties reflect either a lack of understanding or else a lack of knowledge of the area being considered. Quantification of uncertainties essentially amounts to a detailed description of the extent of this ignorance. Such quantification to the desired resolution is not always possible. Unfortunately, estimation of radioactive source terms for LWR accidents involving core damage falls into the class of problems for which the uncertainties cannot currently be quantified to within narrow bounds. In this section are discussed five general sources of uncertainty: (1) lack of understanding of processes and phenomena; (2) difficulties in accident descriptions; (3) modeling limitations; (4) data-related problems; and (5) variations due to differences among reactors.

E.4.2 Description of Processes

Describing the multitude of processes which would occur during any degraded-core or meltdown accident is a complicated undertaking. Fortunately, to estimate the potential consequences, it is not necessary to accurately describe all the processes which would occur. Instead, it is required only that those processes which would be important in the overall description be included in an adequate manner. However, inasmuch as many of the postulated accidents which appear to be of concern involve combinations of events without precedent, it is not always straightforward to delineate all those processes which would ultimately be significant. Thus an uncertainty arises from the inherent complexity of the needed description. Although engineering judgments can be used extensively to reduce the overall problem of estimating radionuclide source terms to a tractable one, there is, of necessity, a large uncertainty in the resulting estimates.

E.4.3 Description of Reality

An essentially infinite variety of accidents is possible at light water reactors (LWR's). To be able to consider the overall risk posed to any population-of-concern by such accidents, that continuum of potential accidents is typically reduced to a finite set of accident sequences. Because only certain sequences are identified and investigated, there is a danger of neglecting other, potentially important sequences. Furthermore, although any given sequence corresponds to an infinite variety of possible accident scenarios and hence descriptions, only a few possible scenarios for certain accident sequences are usually investigated. As a result, only a very restricted portion of the total accident spectrum is considered in detail.

In general, it is not possible to predict many of the events which might occur during any particular accident sequence. Instead, it can only be estimated whether or not such events might be possibilities. For example, the existence of flammable conditions in the containment can be estimated but the occurrence of ignition and burning cannot be predicted. Thus, to investigate any accident sequence, a large number of assumptions must be made. Consequently, there is a large uncertainty inherently associated with assuming any correspondence between reality and the scenarios postulated in any study.

In addition to the foregoing, there is the added problem that the sequences which would be best to consider the risk to one specific population, for example, humans, might not be the best ones to use to estimate the risk to another population, for example, equipment. Furthermore, even if the best sequences to estimate the risks to two different populations-of-concern were the same, the most appropriate assumptions to be used in the descriptions of those sequences might not be the same for the two different populations. Unfortunately, most previous source term work has focused the risk to just one population, namely, the human population.

E.4.4 Modeling Limitations

Modeling problems can be divided into three basic areas: (1) omission of potentially relevant processes; (2) limited description of some processes; and (3) numerical and coding problems. All the computer codes used to describe various aspects of radionuclide releases and subsequent transport in both the primary coolant system and the containment have problems in at least the first two areas. For example, all those codes currently omit consideration of chemical reactions and radioactive decay. Of necessity, the descriptions in all the codes are always limited by the inability to reasonably carry out sufficiently detailed computations and/or insufficient understanding of the processes being modeled.

In principle, the effects of all three types of modeling problems can be at least bounded by rigorous and detailed comparison of code results to experimental results, that is, by validation. Unfortunately, there is a dearth of appropriate experimental data in at least several important areas: release rates of radionuclides from the fuel under accident conditions; release rates of aerosols resulting from concrete decomposition; and behavior of high density aerosols in a condensing steam environment. Thus validation of some of the important models and codes is not possible for many possible LWR accident conditions. As a result, the uncertainty associated with the use of such models and codes for many accidents must be said to be large.

Lack of validation over all ranges of possible conditions is perhaps not as important for mechanistically based codes as it is for non-mechanistically based codes. For codes such as CORRAL which are based largely on nonmechanistic considerations, great care must be exercised in using the codes beyond their initially intended ranges.

E.4.5 Data-Related Problems

The results of using any computer code or model are no better than the data used in the calculations. Unfortunately, large gaps exist in the data required to estimate radionuclide releases and transport after an accident. Some of the missing and/or inadequate data are in non-sensitive areas and so even terrible approximations for the values for these data do not significantly affect the final source term estimates. On the other hand, some of the missing data are in sensitive areas and so the values assumed can substantially affect the resulting estimates.

For meltdown accidents, one of the most important sets of sensitive data relates to the timing and the mode of containment failure. Another set of sensitive data involves the description of the thermal-hydraulic conditions present during the accident. Small changes in certain portions of either of these sets can result in large changes in the estimated radionuclide source terms.

E.4.6 Variations Among Reactors

Due to the very limited scale of this study, only five plants have been considered. Other plants differ from those five in many respects. The differences range from basic design features to construction-related factors. While the effects of many of these differences can be expected to be trivial, the effects of some may be important. For example, both design features and construction-related factors could result in the probabilities of some types of accidents being much different than the probabilities of the analogous accidents at the plants considered here. Those factors and/or features might even permit accidents not previously considered to occur. Alternatively, they might not change the probabilities of the accidents but they might drastically change the amounts of radionuclides likely to be released for some accident sequences.

For a given plant design, the probabilities of similar types of accident sequences would generally be essentially the same. Furthermore, the amounts of radionuclides which would be released for comparable accidents would be expected to be similar. The major differences among plants of a given design probably would be due to construction-related factors. Some basic design variations are covered by the plants discussed in the RSSMAP studies.

As can be seen by comparing the tables in the addendum of the previous appendix which summarize the probabilities and the radionuclide releases for the accident sequences considered in both the RSS rebaselining work and the associated RSSMAP work, there is a large variation among the five plants considered here as to both the types of accidents which would be most likely and the amounts of radionuclides which would be released to the environment by the accidents. Although the variation among the remaining plants would be thought to be less than that for the RSS and the RSSMAP plants, it is not assured that this is the situation. In addition, the variation resulting from heretofore unconsidered construction features could be substantial. Consequently, the uncertainties resulting from all plant-to-plant differences can be expected to be large. Unfortunately, they are not currently quantifiable.

E.4.7 Uncertainty and Sensitivity Studies

Few studies have been done to address the sensitivities of the source term estimates to various uncertain factors. And none of those sensitivity studies which have been done are generally available (for example, Baybutt and Kurth, 1981; Baybutt, Cox and Kurth, 1981), although some of the results of those studies have been alluded to in the literature (USNRC, 1981; Baybutt, 1981).

The few studies which have been performed involve only multi-parametric variations and do not include any probabilistically-based considerations. Consequently, the insights which can potentially be obtained from them are limited. In addition, those studies have been of extremely restricted scope. They have concentrated on the one particular set of models and associated codes. Thus they do not include the uncertainties associated with the models themselves. Furthermore, they do not include the uncertainties related to the variation of important factors among nuclear plants.

E.4.8 Overall Uncertainties

As has been indicated in the previous subsections, there are many potential sources of uncertainties in the estimation of post-accident radionuclide releases. Unfortunately, some of these result in large uncertainties in the various portions of the estimation. However, many of these factors involving large uncertainties are not ultimately important for estimating the radionuclide source terms.

Furthermore, the uncertainties of those factors which are important are not necessarily "additive;" in other words, the uncertainty of the radionuclide source term estimates are not equal to the "sum" of the uncertainties of the dominant factors affecting the estimates. Investigating certain potentially important processes very closely can result in an overestimation of the uncertainty contributed by those individual processes to the overall phenomenon. Often one can have a much better understanding of the overall set of processes than one does for each of the individual processes. Consequently, emphasizing the effects of the individual processes can distort the overall picture.

While this consideration probably results in a sizeable reduction of the uncertainty associated with the overall phenomenology of any accident, it probably does not hold for reducing the uncertainties caused by plant-to-plant variations. These have not been adequately addressed.

To the extent that the implementation of the overall methodology used here is unbiased, another factor may work to reduce the final error in the risk estimated from the source terms. That factor is the fortuitous cancellation of error. It can occur in the consideration of both the probabilities and the amounts of the radionuclides released. Although the error in any one estimate may be large, the overall error in the resulting risk may be small.

In the consideration of uncertainties, it should be noted that whereas it has often been alleged that overly conservative assumptions have been employed in obtaining source term estimates in the past, the veracity of that allegation is not at all evident. Along with the so-called conservative assumptions there have been many nonconservative assumptions. For example, in the RSS all reactors were taken to be adequately represented by two "generic" reactors. That this was an adequate treatment is not obvious; in no sense was such an approximation conservative. Other examples of nonconservative aspects of source term estimation abound in both the RSS and in subsequent source term work.

For all the reasons previously discussed, it must be acknowledged that the uncertainties involved in source term estimation are large. Hopefully, the current understanding of the processes involved and the data which are available permit one to make estimates which are in some sense reasonable.

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ADDENDUM E1

DETAILED RESULTS

This addendum contains the detailed estimates of the containment escape fractions for all the meltdown sequences considered in this study for emergency planning. For comparison with the results of the RSSMAP reports, see the addendum of Appendix D.

Table E1.1. Total containment escape fractions for meltdown sequences at RSS PWR

Sequence	RSS category	Cumulative fractions of core inventory released to atmosphere						
		Xe-Kr	I-Br[a]	Cs-Rb	Te-Sb	Ba-Sr[b]	Ru[c]	La
V	2	1.00	0.70	0.94	0.94	0.47 0.20	0.025 0.21	0 004
TMLB′-δ[d]	2	1.00	0.33	0.43	0.43	0.18 0.09	0.007 0.075	0.002
TMLB′-γ[d]	2	1.00	0 33	0.43	0.43	0.18 0.09	0.007 0.075	0.002
S_2C-δ	3	1.00	0.056	0.34	0.34	0.41 0.24	0.029 0.22	0.003
S_2D-β	5	1.00	0.002	0.01	0.01	0 002 0.0008	0.0002 0.002	6(-5)[e]
TMLB′-ε[d]	6	0.86	0.003	0.001	0.001	0.0004 0.0002	3(-5) 0.0003	1(-5)
AHF-ε	6	1 00	0.0002	0.0007	0.0007	0.0004 0.0001	3(-5) 0.0002	9(-6)
PWR 4[f]	4	1.00	0.09	0.047	0.047	0.010 0.004	0.001 0.008	0.0003
PWR 5[f]	5	1.00	0.03	0.01	0.01	0.002 8(-4)	0.0002 0.002	6(-5)
PWR 6[f]	6	1.00	0.0008	0.001	0.001	0.0002 6(-5)	3(-5) 0.0002	1(-5)
PWR 7[f]	7	1.00	2(-5)	1(-5)	1(-5)	2(-6) 8(-7)	3(-7) 2(-6)	1(-7)
V[g]	2	1.00	0.70	0.94	0.80	0.11 0.04	0.01 0.09	0.004
TMLB′-δ,γ[h]	2	1.00	0.10	0.12	0.12	0.045 0.020	0.003 0.027	0.001
TMLB′-ε[h]	6	1.00	0.0008	0.0003	0.0003	0.0001 6(-5)	2(-5) 0.0001	1(-5)

[a]Represents fraction of iodine present as elemental iodine which escapes.

[b]Upper value is based on barium data; lower value is based on strontium data.

[c]Upper value is based on ruthenium data; lower value is based on molybdenum data.

[d]Removal due to scrubbing by passage through the pressurizer quench tank has not been included.

[e]6(-05) represents 6×10^{-5}.

[f]Based on the composites presented in the Reactor Safety Study.

[g]All release rates after slumping reduced by surface-to-volume considerations.

[h]Removal due to scrubbing by passage through the pressurizer quench tank has been included; a decontamination factor of 10 was assumed.

Table E1.2. Total containment escape fractions for meltdown sequences at RSS BWR

Sequence	RSS category	Cumulative fractions of core inventory released to atmosphere						
		Xe-Kr	I-Br[a]	Cs-Rb	Te-Sb	Ba-Sr[b]	Ru[c]	La
TC-γ′	2	1.00	0.47	0.22	0.26	0.064 0.013	0.021 0.110	0.006
TW-γ′	2	1.00	0.10	0.07	0.07	0.053 0.020	0.010 0.069	0.004
TQUV-γ′	2	1.00	0.10	0.20	0.22	0.044 0.008	0.009 0.070	0.004
TC-γ	3	1.00	0.07	0.05	0.05	0.016 0.003	0.006 0.033	0.002
TW-γ	3	1.00	0.003	0.03	0.03	0.022 0.010	0.002 0.022	0.001
TQUV-γ	3	1.00	0.02	0.03	0.05	0.009 0.002	0.003 0.020	0.001
BWR 4[d]	4	1.00	0.0008	0.005	0.005	0.003 0.0005	0.0003 0.002	0.0001
TC-γ′[e]	2	1.00	0.45	0.07	0.07	0.034 0.003	0.019 0.089	0.006
TW-γ′[e]	2	1.00	0.10	0.02	0.02	0.019 0.005	0.008 0.053	0.004

[a] Represents fraction of iodine present as elemental iodine which escapes.

[b] Upper value is based on barium data; lower value is based on strontium data.

[c] Upper value is based on ruthenium data; lower value is based on molybdenum data.

[d] Based on composite presented in the Reactor Safety Study.

[e] Based on primary coolant system escape fractions of 0.1 for all aerosols (instead of 0.33).

Table E1.3. Total containment escape fractions for meltdown sequences at ice condenser PWR

Sequence	RSS category	Cumulative fractions of core inventory released to atmosphere						
		Xe-Kr	I-Br[a]	Cs-Rb	Te-Sb	Ba-Sr[b]	Ru[c]	La
S_2HF-γ	2	1.00	0.13	0.63	0.63	0.16 0.06	0.016 0.13	0.005
V	2	1.00	0.77	0.89	0.89	0.45 0.19	0.026 0.21	0.005
TML-γ[d]	3	1.00	0.016	0.013	0.013	0.017 0.003	0.008 0.050	0.004
TMLB′-δθ[d]	3	1.00	0.063	0.16	0.16	0.068 0.033	0.04 0.036	0.001
S_2HF-δ	3	1.00	0.009	0.21	0.21	0.060 0.017	0.009 0.075	0.004
S_2HF-δθ	3	1.00	0.048	0.32	0.32	0.074 0.032	0.005 0.040	0.001
TMLB′-δ[d]	4	1.00	0.006	0.005	0.005	0.004 0.0009	0.001 0.008	0.0006
S_2H-γθ	4	1.00	0.005	0.013	0.013	0.009 0.001	0.003 0.028	0.002
AD-γ	5	1.00	0.002	0.0005	0.0005	0.0009 0.0001	0.0006 0.004	0.0003
AD-δ	7	1.00	7(-7)	3(-7)	3(-7)	2(-7) 6(-8)	6(-9) 6(-8)	6(-10)

[a]Represents fraction of iodine present as elemental iodine which escapes.

[b]Upper value is based on barium data; lower value is based on strontium data.

[c]Upper value is based on ruthenium data; lower value is based on molybdenum data.

[d]Removal due to scrubbing by passage through the pressurizer quench tank has not been included.

Table E1.4. Total containment escape fraction for meltdown sequences at Mark III BWR

Sequence	RSS category	Cumulative fractions of core inventory released to atmosphere						
		Xe-Kr	I-Br[a]	Cs-Rb	Te-Sb	Ba-Sr[b]	Ru[c]	La
TPQI-δ	2	1.00	0.60	0.37	0.37	0.12 0.05	0.010 0.09	0.003
TQW-δ	2	1.00	0.21	0.19	0.19	0.13 0.06	0.016 0.13	0.007
AC-δ	2	1.00	0.52	0.78	0.73	0.06 0.02	0.007 0.05	0.002
TQUV-γ	3	1.00	0.03	0.06	0.19	0.03 0.002	0.011 0.08	0.006
TQUV-δ	4	1.00	0.004	0.002	0.006	0.001 0.0001	0.0004 0.004	0.0002

[a]Represents fraction of iodine present as elemental iodine which escapes.

[b]Upper value is based on barium data; lower value is based on strontium data.

[c]Upper value is based on ruthenium data; lower value is based on molybdenum data.

Table E1.5. *Total containment escape fraction for meltdown sequences at alternate large containment PWR*

Sequence	RSS category	Cumulative fractions of core inventory released to atmosphere						
		Xe-Kr	I-Br[a]	Cs-Rb	Te-Sb	Ba-Sr[b]	Ru[c]	La
$T_1(B_3)MLUOO'-\delta$[d]	2	1.00	0.55	0.76	0.76	0.33 0.16	0.022 0.19	0.008
V	2	1.00	0.53	0.90	0.90	0.46 0.18	0.026 0.20	0.004
AYF-δ	2	1.00	0.25	0.72	0.72	0.44 0.07	0.025 0.20	0.005
AG-δ	2	1.00	0.16	0.76	0.76	0.46 0.07	0.030 0.25	0.008
TMLU-γ[d]	3	1.00	0.03	0.03	0.03	0.03 0.006	0.014 0.09	0.007
TMLU-γ[d]	3	1.00	0.45	0.75	0.75	0.33 0.13	0.023 0.20	0.009

[a]Represents fraction of iodine present as elemental iodine which escapes.

[b]Upper value is based on barium data; lower value is based on strontium data.

[c]Upper value is based on ruthenium data; lower value is based on molybdenum data.

[d]Removal due to scrubbing by passage through the pressurizer quench tank has not been included.

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