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Identification of Severe Accident Uncertainties

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ABSTRACT

Understanding of severe accidents in light-water reactors is currently beset with uncertainty. Because the uncertainties that are present limit the capability to analyze the progression and possible consequences of such accidents, they restrict the technical basis for regulatory actions by the U. S. Nuclear Regulatory Commission (NRC). It is thus necessary to attempt to identify the sources and quantify the influence of these uncertainties.

As a part of ongoing NRC severe-accident programs at Sandia National Laboratories, a working group was formed to pool relevant knowledge and experience in assessing the uncertainties attending present (1983) knowledge of severe accidents. This initial report of the Severe Accident Uncertainty Analysis (SAUNA) working group has as its main goal the identification of a consolidated list of uncertainties that affect in-plant processes and systems. Many uncertainties have been identified. A set of "key" uncertainties summarizes many of the identified uncertainties. Quantification of the influence of these uncertainties, a necessary second step, is not attempted in the present report, although attempts are made qualitatively to demonstrate the relevance of the identified uncertainties.

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NOMENCLATURE

T e following abbreviations, when used in this report, have the meanings given here.

ADS	Automatic depressurization system
AEC	(U.S.) Atomic Energy Commission
AECB	Atomic Energy Control Board (Canada)
AFW	Auxiliary feedwater
AFWS	Auxiliary feedwater system
AGR	Advanced Gas-cooled Reactor
ANO-1	Arkansas Nuclear One - Unit 1
APRS	Automatic pressure-relief system
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated transient without scram
BCL	Battelle's Columbus Laboratory
BWR	Boiling water reactor
B&W	Babcock and Wilcox
CCDF	Complementary cumulative distribution function
CCW	Component cooling water
CE	Combustion Engineering
CR-3	Crystal River - Unit 3
CRD	Control-rod drive
CST	Condensate storage tank
CVCS	Chemical and Volume Control System
DC	Design contractor
DF	Decontamination factor
ECC	Emergency core cooling
ECCS	Emergency core-cooling system

EHC	Electro-mechanical hydraulic control
EPRI	Electric Power Research Institute
ESF	Engineered safety feature
FCI	Fuel/coolant interaction
FITS	Fully-instrumented test series
FP	Fission product (and other) radionuclides
FSAR	Final Safety Analysis Report
FWCI	Feedwater coolant injection
GE	General Electric
HEP	Human Error Probability
HPCI	High-pressure coolant injection
HPCS	High-pressure core spray
HPI	High-pressure injection
HPIS	High-pressure injection system
HPME	High-pressure melt ejection
HRA	Human reliability analysis
HVAC	Heating, ventilating, and air conditioning
IDCOR	Industry degraded-core rulemaking
INEL	Idaho National Engineering Laboratory
IPRDS	In-plant Reliability Data System
IP-2	Indian Point - Unit 2
IP-3	Indian Point - Unit 3
IREP	Interim* Reliability Evaluation Program (*NUREG-0544 says "Integrated")
KfK	Kernforschungszentrum Karlsruhe (Center for Nuclear Research, Karlsruhe)
LER	Licensee event report
LOCA	Loss-of-coolant accident

LOHSA	Loss-of-heat-sink accident
LPCI	Low-pressure coolant injection
LPCRS	Low-pressure coolant-recirculation system
LPCS	Low-pressure core spray (system)
LPIS	Low-pressure injection system
LPRS	Low-pressure recirculation system
LWR	Light-water reactor
MMI	Modified Mercalli intensity
MSIV	Main steam isolation value
NPRDS	Nuclear Plant Reliability Data System
NRC	(U.S.) Nuclear Regulatory Commission
PCS	Passive containment system
PORV	Power-operated relief valve
PNS	Projekt Nucleare Sicherheit (Project for Nuclear Safety)
PRA	Probabilistic risk assessment
PWR	Pressurized-water reactor
QUEST	Quantitative Uncertainty Estimation for the Source Term
RCIC	Reactor core isolation cooling
RCS	Reactor coolant system
RHR	Residual-heat removal
RMIEP	Risk Methods Integration and Evaluation Program
RPS	Reactor protective system
RPV	Reactor pressure vessel
RSS	Reactor Safety Study (see WASH-1400)
RSSMAP	Reactor Safety Study Methodology Applications Program
SARP	Severe Accident Research Program
SARRP	Severe Accident Risk Reduction Program

SASA	Severe Accident Sequence Analysis
SAUNA	Severe Accident Uncertainty Analysis
SDCS	Shutdown cooling system
SGTR	Steam generator tube rupture
SLC	Standby liquid control
SNLA	Sandia National Laboratories, Albuquerque
SRV	Safety relief valve
SSE	Safe Shutdown Earthquake
TMI-2	Three Mile Island - Unit 2
TVA	Tennessee Valley Authority
T&M	Test and maintenance
USAEC	United States Atomic Energy Commission
USNRC	U.S. Nuclear Regulatory Commission
W	Westinghouse
WASH-1400	U.S. Nuclear Regulatory Commission. "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.
ZIP	Zion - Indian Point Study (W. B. Murfin et al, NUREG/CR-1409, NUREG/CR-1410, NUREG/CR-1411), 1980
ZPSS	Zion Probabilistic Safety Study, Commonwealth Edison Co., Chicago, 1981

1. INTRODUCTION

In this report, "uncertainty" refers to imprecision in the state of knowledge about physical processes, imprecision in models of physical processes or in the state of knowledge about the parameters of these models, or imprecision in the predictions of such models.

Because it is recognized that readers of this report are apt to have widely varying degrees of familiarity with nuclear power plant design criteria, probabilistic risk assessments (PRAs), severe accident progression, and severe accident research, Subsections 1.1 through 1.3 have been included to provide some general background and to indicate why severe accident uncertainties are important. Readers familiar with the aforementioned subjects may prefer to skip to Subsection 1.4, which discusses the purpose, scope, and limitations of this report. Subsection 1.5 provides a summary of the findings of this report and cross references to more detailed discussions in subsequent sections.

1.1 Background

The three basic questions that must be answered to determine the safety of any facility are:

1. What could go wrong?

2. How likely is it?

3. If it happens, what are the consequences?

Since the beginning of nuclear power plant design, scientists and engineers have been concerned with these questions. The safety design requirements for nuclear power plants in the United States evolved from the General Design Criteria set forth in the Code of Federal Regulations.[1] To a large extent this evolution preceded the Reactor Safety Study[2] and the accident at Three Mile Island.[3 4] At the risk of oversimplification, it may be said that the safety-design requirements which evolved in the US* used the General Design Criteria to answer question 2 above by implicitly emphasizing design against large-pipe-break loss-of-coolant accidents and against single failures of redundant safety systems. Thus, plants operating in the US today have been specifically designed to withstand a broad variety of accidents, including those initiated by large pipe breaks in the nuclear steam-supply system, and those that develop from the worst conceivable single failures of safety-related electrical, mechanical, or control systems. However, US plants have not generally been designed against

*No single document exists to define current safety design requirements for nuclear power plants in the US. In many cases, safety-design requirements are established in the plant-specific licensing process. However, the Regulatory Guides of the US Nuclear Regulatory Commission present the position of the NRC on many design matters.

accidents in which multiple failures capable of causing a complete loss of intended safety function(s) occur. Because such accidents can result in core damage, they are referred to as severe accidents, by contrast with conventional design-basis accidents.

The accident at Three Mile Island (TMI-2) demonstrates that multiple failures can lead to severe accidents. This is consistent with predictions of probabilistic risk assessments (PRAs), such as the Reactor Safety Study.[2] PRAs attempt to answer the three questions posed above by systematically identifying potential accidents and quantitatively estimating their frequencies (occurrences per year) and consequences (in terms of radiation doses, early fatalities, latent fatalities, dollars, or other units). The "risk" posed by a particular accident may be expressed as the product of its estimated frequency and its estimated consequence. The plant risk may be expressed as the sum of all such products. PRAs predict that severe accidents, despite their low frequencies, dominate the risk calculated in this way because of very high estimated consequences associated with core damage, possible catastrophic containment failure, and the uncontrolled release of radionuclides to the environment that might ensue. This is partly substantiated by the severe accident at TMI-2 which, although it did not result in high radiological consequences, did result in high monetary consequences.

Since the accident at TMI-2, the US Nuclear Regulatory Commission (NRC) and the nuclear industry have been actively reassessing the likelihood and potential consequences of severe accidents at US nuclear power plants. The objective is to identify whether changes should be made at existing plants, or in the design requirements for future plants, to reduce the risk posed by severe accidents. Such decisions, if based on today's knowledge, will be made in the presence of large uncertainties in the frequency and consequence estimates for severe accidents. One of the criticisms made by the Lewis Committee in their review of the RSS was that "the error bounds on those [risk] estimates are in general greatly understated."[5] The Lewis Committee recommended that PRA results be used "to guide the reactor safety research programs so as to reduce the uncertainties in the analyses, and to gain greater understanding of those points of risk uncovered." Although some actions have been taken to improve the treatment of uncertainties in PRAs, the uncertainties are still not well-quantified, they are believed to be large,[6 7] and they could easily be comparable to the risk reduction afforded by contemplated safety-improvement options. Thus, if the NRC is to make informed decisions regarding safety-improvement options and research priorities, careful consideration must be given to identifying and, to the extent possible, quantifying uncertainties in risk estimates.

1.2 Severe Accidents in LWRs

The only way that large quantities of radionuclides could be released from nuclear power plants to the environment is by the extreme overheating (approaching or including melting) of the fuel in the reactor core. The safety design of each nuclear reactor includes various systems to prevent such overheating. A containment building and associated safety systems are also included to keep radionuclides, which might be released from the fuel, from reaching the environment.

The redundant (and, in many cases, diverse) systems which are installed to prevent core overheating accomplish three basic safety functions: shutting down the fission chain reaction, keeping the core covered with coolant, and removing heat generated by the decay of radionuclides in the fuel. Extreme overheating of fuel in the reactor core is possible only in accidents that involve multiple failures of equipment designed to accomplish these functions. Accidents involving such failures are called severe accidents.

Certain key events would occur in any severe accident. These include the initiating event, equipment failures, and operator actions preceding fuel damage. The uncertainties in initiating events and resulting accident sequences and their likelihood are discussed in Section 2 of this report. Uncertainties in the in-vessel stages of a severe accident are the topic of Section 3. These stages can be marked by the onset of sustained core uncovering (which leads to core heatup), the onset of exothermic oxidation of fuel-rod cladding by steam to produce zirconium dioxide (ZrO_2) and hydrogen, cladding failure (which releases gaseous radionuclides from the fuel), the onset of fuel melting (which results in more substantial radionuclide releases from the fuel), slumping of molten material into the lower plenum of the reactor vessel (which may contain residual reactor coolant), and failure of the reactor vessel with consequent discharge of molten material into containment.

Molten material discharged from the reactor vessel can undergo interactions if it contacts water or concrete in containment. The result may be either cooled ex-vessel core debris, uncooled core debris (which attacks the concrete containment floor), or a combination of these two results. The melt ejection and melt-water and melt-concrete interactions provide sources of steam, hydrogen, and both radioactive and nonradioactive aerosols to the containment atmosphere. The melt-concrete interactions also yield other gases, predominantly carbon dioxide and carbon monoxide. Both hydrogen and carbon monoxide are combustible. Uncertainties in the ex-vessel processes that give rise to gases and aerosols are the topic of Section 4, while those arising in the fluid, heat transfer, and thermodynamics of the containment atmosphere are treated in Section 5.

If the containment does not fail during a severe accident, the consequences to public health and safety will be negligible. Various modes of containment failure and their uncertainties are the topic of Section 6. In general, the modes of containment failure that result in the expulsion of radionuclides directly to the outside atmosphere have the highest potential consequences. One such mode is failure due to a missile generated by a steam explosion occurring when molten core material slumps into water in the lower plenum of the reactor vessel. Other direct-expulsion modes are those attributable to high pressures and temperatures occurring within containment during the accident. High pressures and temperatures could result from the addition of steam and other gases mentioned above, from direct heating of the containment atmosphere by aerosols and gaseous radionuclides, and from possible combustion events (Section 5). Safety-design features, such as containment sprays, fan coolers, and ice condensers, would, if operating, act to reduce these pressure-temperature loads.

Finally, even if containment fails in a manner that permits the direct expulsion of radionuclides to the outside atmosphere, the consequences will depend on the quantity and characteristics of the radionuclides actually expelled. Retention of radionuclides in the reactor coolant system or in containment would act to reduce the consequences of severe accidents. Uncertainties in the in-plant release, transport, and retention of radionuclides are the topic of Section 7.

As noted in Section 1.1, to estimate the risk associated with a nuclear power plant one must identify the possible accidents and estimate their frequencies and consequences. However, severe accidents are rare. At the time of this writing the only civilian LWR accident in the US which has led to significant core damage in over 500 reactor-years of operation is that at TMI-2. Consequently, both the frequencies and the consequences of accidents identified in PRAs must be estimated by synthesizing the data which do exist regarding equipment failures, human actions, and individual phenomena. To accomplish this synthesis, one must develop and use models of both safety-system reliability and accident progression.

Because we can only model, not measure, severe-accident frequencies and consequences, the results of PRAs are subject to a variety of uncertainties. Uncertainties in model inputs (e.g., component failure frequencies, initial and boundary conditions, and material properties) will cause model outputs to be uncertain. Uncertainties or approximations in the modeling of individual processes will also contribute to uncertainty in the model outputs. Interrelationships between pieces of equipment or among various processes may not be well understood or properly modeled. Finally, there is always the question of completeness: Have we identified all of the important accident-initiators, equipment failure modes, and physical phenomena?

For the most part, consequence estimates in existing PRAs have been based on rather simple, parametric models such as those of the MARCH and CORRAL computer codes.[8 9] Such codes are known to have significant limitations[10], and efforts are underway to develop better tools.[11-14] These efforts include the industry's MAAP code and the NRC's MELCOR code, as well as more detailed phenomenological codes that treat particular accident stages or processes.[11-14] In parallel with their development, efforts are underway to apply these codes to define better (and, it is hoped, to reduce the uncertainty in) severe-accident-consequence estimates. The potential political and economic ramifications of such uncertainty reductions could be very significant. For example, it has been estimated that a tenfold reduction in the radioactivity released--the so-called "source term"--with respect to that estimated in the RSS would "eliminate the risk of early fatalities."[15] However, in light of our dependence on predictive models for both frequencies and consequences of severe accidents, it is important that efforts to justify such reductions be carried out in a forum that recognizes, insofar as possible, all of the sources of uncertainty that could affect such conclusions. This report is an initial attempt to identify such sources of uncertainty.

1.3 Severe Accident Research and Uncertainties

The principal, ongoing, instrument in elucidating the nature of severe accidents in LWRs is the Severe Accident Research Program (SARP) of the NRC. The plan for this research has been evolving over the past several years. The current approach is discussed at length in NRC's Nuclear-Power-Plant Severe-Accident Research Plan, NUREG-0900 (1983), which states that the goal of the plan is to obtain a better understanding of the likelihood and consequences of severe accidents, and to determine what changes in design or operation could reduce the risk from such events.[7]

It is significant that NUREG-0900 places a high priority on the issue of uncertainty in the understanding of severe accidents. To quote from the introduction: "There are substantial uncertainties in severe accident analysis because information upon which the calculations are based is either incomplete or largely judgmental at this time. Therefore, the identification, quantification, and reduction of important uncertainties is substantially significant to the severe accident research program."[7] This statement largely embodies the current NRC approach to uncertainties in severe-accident analysis.

In the "Severe Accident Decisions" draft of August 5, 1983, the SARP Senior Review Group states that "Agreement [for decision-making] will also be sought on descriptions of the magnitude of the uncertainties associated with the [severe accident] issues. If the uncertainties are too large, it may not be possible to base a decision on an accurate understanding of the issue. If further research can narrow the uncertainty, it may be better to defer the decision. Otherwise, it may be necessary to treat the issue conservatively and to proceed with the decision."[16] This clearly requires increased analysis of the uncertainties, including identification of important uncertainties, delineation of their potential effects, and (where possible) quantitative estimation of ranges, even for the least prescriptive of the above choices. This increase in uncertainty analysis is consistent with the high priority accorded to this issue by the Severe-Accident Research Plan.[7] The present study forms a part of this increased activity.

1.4 Severe Accident Uncertainty Analysis

In response to the need, discussed above, to provide increased treatment of severe accident uncertainties, the NRC has directed Sandia National Laboratories, Albuquerque (SNLA) to account for the impact of uncertainties in several NRC-sponsored severe-accident research programs currently active at SNLA. In response to this requirement, participants from the research programs listed in Table 1-1 formed an informal Severe Accident Uncertainty Analysis (SAUNA) working group to coordinate work on

1. A consolidated list of severe accident uncertainties,

2. Valid methods for estimating the magnitude of severe accident uncertainties, and

Table 1-1 Studies requiring uncertainty analysis

Program	Reference
LWR Severe Core Damage Phenomenology	[17]
Melt Progression Phenomenology	Camp et al[18]
Severe Accident Sequence Analysis (SASA)	Haskin et al[19 20]
Severe Accident Risk Reduction Program (SARRP)	Benjamin[21]
MELCOR	Weigand et al[13] Sprung et al[14]
Accident Sequence Evaluation Program (ASEP)	Harper et al[22] Kolaczkowski et al[23]

3. Knowledge of what research should be most effective in reducing the severe accident uncertainties.

Perspective on the effort represented by this report by the SAUNA working group may be gained by referring to the three constituent parts of the uncertainty problem given in the quotation from the Severe Accident Research Plan[7] in Subsection 1.3. They are

1. Identification of uncertainties,

2. Quantification of uncertainties, and

3. Reduction of uncertainties.

(Quantification includes estimation of the quantitative effect of an uncertainty on risk measures. This is necessary, for example, to permit importance assessment.)

The present report attempts to provide the identification given as item 1 in this list. Little effort was expended in quantifying uncertainties (item 2), and the reduction of uncertainties (item 3) was clearly beyond the purview of the working group. Therefore, the effort represented herein should be regarded as a partial and incomplete response to NRC needs regarding severe accident uncertainties. It is, however, a necessary first step.

Sections 2 through 7 of this report list and discuss the specific severe accident uncertainties, restricted to the in-plant setting, that the working group has identified. The sections are arranged by categories that reflect the particular specializations of the members of the group. The reader should assume that certain biases related to the specializations of the authorship, including the tendency to discuss more thoroughly that which one understands best, remain, despite the extensive reviews which the various drafts have received. In Section 8, the relationships between these component uncertainties are discussed in the context of the principal risk-influencing factors. Also provided in Section 8 are some recommendations on future approaches to gaining a fuller understanding of uncertainties and their impact on reactor safety.

During the initial stage of any investigation, much of the knowledge accumulated serves to determine which facts are known about the problem, and to define the major areas of uncertainty. Subsequently, with the problem thus better defined, effort is increasingly directed at synthesizing the known facts and obtaining additional specific knowledge, much of which serves to reduce remaining uncertainties.

The process of investigating severe accidents in LWRs has the above characteristics. The identification of uncertainties is a part of the initial stage of the investigation, and the analysis of severe accidents corresponds to the synthesis of what is known about how they occur and progress. In many of the major areas of uncertainty, the process of obtaining additional knowledge to reduce the uncertainties is currently in progress. Examples include studies of the oxidation kinetics of zirconium, the mechanisms of steam explosions, and the combustion of hydrogen/air/steam mixtures. In other areas, studies aimed at resolving uncertainties are now being planned. Based upon the considerable progress achieved during the approximately five years since TMI-2, it is expected that many of the identified uncertainties will be substantially reduced in the future if appropriate effort is committed.

The goal of this report, a consolidated list of severe accident uncertainties, could be attained only by emphasizing those aspects of accident sequences that are highly uncertain in preference to those aspects which are relatively well-understood and thus less uncertain. The text is consequently one-sided in this respect and should not be taken as a balanced appraisal of the state of knowledge regarding all aspects of severe reactor accidents.

Achieving the goal given has resulted in the identification of a significant number of uncertainties. Without consideration of the many advances and achievements in understanding severe accidents which have occurred in the five years of focused study since TMI-2, the list might appear formidable. A gloomy disposition concerning the list could result from comparison with optimistic understatements of the uncertainties that exist, coupled with an overestimation of the speed and economy of past research.

This report includes many explanations that have been designed to show how a given uncertainty can affect one or more larger results, these larger results being important because they relate to a major consequence of a reactor accident. For example, uncertainty in the pressurization resulting from a hydrogen combustion event is linked to the possible resulting uncertainty in containment failure, because the character of fission-product release to the environment is usually coupled to the failure of containment. These plausible links are included to give the reader an explicit reason why the uncertainty may be important. Without such linkages, the nonspecialist reader might have difficulty judging what the effect of the uncertainty on the outcome of accidents might be.

However, the plausible--sometimes even speculative--nature of these linkages should be emphasized. A rigorous defense of a linkage is often impossible because the linkage is itself uncertain or because the linkage often depends upon the outcome of the uncertain process lying within a particular range.

Many of the individual uncertainties listed in this report span ranges which permit outcomes from benign to hazardous. It is possible, by selectively combining exclusively pessimistic assumptions, or exclusively optimistic assumptions, to fabricate scenarios supporting extreme viewpoints regarding the outcome of severe accidents. However, careful reading of the relevant sections of the report will reveal that no evidence exists to support combinations yielding extreme outcomes in preference to combinations yielding intermediate outcomes. Thus, the statements about uncertainties herein should not be construed as supporting either extreme viewpoint regarding the safety of nuclear power plants.

Describing uncertainties in abstract terms such as "data-based uncertainty", "modeling uncertainty", or "random uncertainty" has not been emphasized in this report. Where such abstract terms as these are used in the report, their meanings are sufficiently clear from the context. A discussion of the different kinds of uncertainty would include some interesting and difficult concepts, and there may not be complete agreement as to their application in nuclear power plant accident analysis. We have therefore elected not to enlarge the sizeable literature on the abstract discussion of uncertainty in this report. A very recent study should be noted, however, in which Vesely and Rasmuson provide a general description of the different kinds of uncertainty that affect PRA, and some available approaches to their quantification and combination.[24] Although the classification system offered by these authors may not find general acceptance, it encompasses a wide range of qualitatively different kinds of uncertainties, from "data uncertainties" to "interpretation uncertainties" ("doubtfulness or vagueness in the interpretation of the results"). Certainly their two general categories of "experimental uncertainty" (variation in results in repeated "experiments") and "knowledge uncertainty" (lack of knowledge yielding "vagueness, indefiniteness, or imprecision in an analysis, a stated conclusion, or a stated value") represent an important distinction. The uncertainties identified in the present report support such a wide range of kinds of uncertainty, however categorized.

The main conclusions of Vesely and Rasmuson, that more complete uncertainty and sensitivity analyses are needed for PRA, and that these will add credibility and usefulness to PRA, closely reflect the recommendations given in Section 8. Thus these conclusions have been arrived at by identification of general classes of uncertainty as well as by consideration of a consolidated list of individually identified uncertainties in severe accident analysis.

1.5 Summary of Findings

As stated above, this report attempts to identify the in-plant uncertainties that influence the analysis of severe accidents. Sections 2 through 7 of this report indicate to us that a small group of key uncertainties--each relatively broadly encompassing--dominates the analyses of in-plant aspects of severe accident risks and consequences. The key uncertainties are summarized below, in the order in which they are discussed in Sections 2 through 7. Statements summarizing why we consider each of these uncertainties to be a key uncertainty are also provided, and the subsection numbers in which each key uncertainty is discussed are listed for ready reference.

1. Definition of accident sequences. Identification of sequence initiators, equipment failure modes (including partial failures), dependent failures, and modes of recovery from faults may be incomplete. Also, the definition and the effects of partial attainment of some success criteria are uncertain. [Subsection 2.1]

2. Frequencies of events and probabilities of faults. The limitations of data pertaining to initiating events, component failures, and human errors mean that estimates of the frequency of the combinations of these events in an accident sequence can be very uncertain. Time-dependent effects, plant-to-plant variability, and the use of incomplete or inaccurate data add to this uncertainty. [Subsection 2.2]

3. Human actions and inactions. Human actions or inactions may contribute to accident initiation. They may also affect the progression of an accident in both favorable and unfavorable ways. Possibly complex actions extraneous to, or in the absence of, specified procedures are particularly difficult to predict. [Subsections 2.1, 2.2]

4. Progression of core damage following initial loss of intact geometry. Configuration, heatup rate (due to oxidation), heat loss rate, and hydrogen evolution are significantly uncertain, including effects resulting from coolant injection into an overheated core. [Section 3]

5. In-vessel core-melt/coolant interactions, including the possibility of a steam explosion of magnitude sufficient to fail containment. Core-melt/coolant interactions can affect the particle size, temperature and position of debris, and hydrogen production. A powerful enough steam explosion could result in simultaneous breach of containment and release of fission products to the offsite environment. The probability

of such an event is uncertain (unless no water is present in the reactor vessel). [Subsections 3.4, 3.5, 6.3]

6. Location and nature of ex-vessel core debris. The behavior of melt and core debris after leaving the vessel affects pressure spikes, ex-vessel steam explosions, aerosol and fission-product evolution, hydrogen production, and debris coolability. [Subsections 4.1, 4.2]

7. Magnitude and timing of pressure source caused by ignition of flammable or detonable gases in containment, and consequent release of heat. Hydrogen is evolved during the oxidation of core zirconium and other metals, and hydrogen and carbon monoxide can be released by core/concrete interactions. For many plants and accident sequences, the pressures accompanying combustion of these gases are uncertain. [Subsection 5.1]

8. Magnitude of containment pressure sources due to steam and noncondensable gases, including those from steam spikes, core/concrete interactions, and gas heating by hot aerosols. Accumulation of noncondensable gases and steam and heating by hot aerosols can threaten containment integrity owing to high temperatures and pressures. [Subsection 5.2]

9. Effects of severe accident conditions, including aerosol deposition and hydrogen burning, on engineered safety features and other required systems. Certain equipment needed to arrest the progression, or mitigate the effects, of severe accidents may be rendered less effective or ineffective because of deflagrations, radiation, steam, atmospheric temperatures, aerosols, etc. produced by the accident. [Subsections 2.1, 5.1, 5.2, 5.3]

10. Containment breach pressure and size due to quasi-steady overpressure. It is difficult to predict the pressure that will result in containment building failure and the resulting equivalent size of the breach to the environment. Stylized and unjustified assumptions are often made in the absence of detailed structural evaluations. [Subsection 6.5]

11. Release of fission products into the containment atmosphere. Our understanding of release and transport phenomena in the RCS, and of ex-vessel release phenomena, is at a developmental stage. This leads to large uncertainties in the magnitude and timing of FP and aerosol releases a) from the RCS and b) via ex-vessel processes. [Subsections 7.1, 7.2, 7.3]

12. Attenuation of fission products in containment. Despite our improving ability to predict aerosol behavior, uncertainties in the inputs (releases and atmosphere conditions) to aerosol calculations make the quantities of FPs airborne at any given time substantially uncertain. Releases from containment are

made more uncertain by possibilities for effects during discharge from the plant ranging from extensive retention to substantial re-entrainment. [Subsections 7.4, 7.5]

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2. ACCIDENT SEQUENCE DEFINITION AND QUANTIFICATION

The nuclear industry and the Nuclear Regulatory Commission (NRC) are actively conducting Probabilistic Risk Assessment (PRA) and PRA-related studies which address the definition of severe accidents (alternatively called accident sequences) and the likelihood of their occurrence. Examples are the Industry Degraded Core Rulemaking (IDCOR) Program [1] and various NRC programs such as the Severe Accident Risk Reduction Program (SARRP),[2] Severe Accident Sequence Analysis (SASA) Program,[3] Accident Sequence Evaluation Program (ASEP),[4] and the Risk Methods Integration and Evaluation Program (RMIEP). Several plant-specific risk assessments, sponsored by both the NRC and industry, have been published. Some examples of these assessments are given in Table 2-1.

As the emphasis on the study of severe accident sequences has increased, the importance of identifying and quantifying the uncertainties in the analyses has become clear. The use of PRAs and other accident sequence studies will be limited if no indication is given as to how the results might vary as a result of the uncertainties that exist in the data, models, and methods. The objective of this section is to identify and describe the uncertainties in the accident sequence identification and quantification process.

An evaluation of severe accident sequences includes both qualitative and quantitative analyses. The qualitative analysis includes the identification and definition of the events that can initiate an accident and the subsequent combinations of failures necessary to cause damage to the reactor core. The quantitative part involves gathering and combining data and other information to estimate individual component failure and human error probabilities and the frequencies of accident sequences.

This section is divided into two subsections, accident sequence definition and accident sequence quantification. The first subsection includes a discussion of the uncertainties in the identification of events that initiate accidents, in the development of accident sequence event trees that delineate unique accident scenarios, and in the models used to characterize how components fail and humans err. The accident sequence quantification subsection describes the uncertainty in collecting and interpreting initiating events, component, and human reliability data and in using the data to evaluate the event trees and fault trees.

2.1 Accident Sequence Definition

The objective of the accident sequence definition task of a PRA is to define a comprehensive set of accident sequences that encompasses the effects of all realistic and physically possible accidents involving the reactor core. Accident sequence definition can be divided into five activities: initiating event identification, accident sequence delineation, success criteria definition, system modeling, and human reliability modeling. Uncertainties associated with each of these activities are discussed below.

Table 2-1 Published probabilistic risk assessments for U.S. light water reactors

Plant	Vendor	Type	Sponsor	Uncertainties Addressed?	External Events Analyzed?
Arkansas Nuclear One 1	B&W	PWR	NRC	yes	no
Big Rock Point	GE	BWR/1	Utility	yes	yes
Browns Ferry 1	GE	BWR/4	NRC	no	no
Calvert Cliffs 2	CE	PWR	NRC	no	no
Crystal River 3	B&W	PWR	NRC	no	no
Grand Gulf 1	GE	BWR/6	NRC	no	no
Indian Point 2&3	W	PWR	Utility	yes	yes
Limerick 1	GE	BWR/4	Utility	yes	no
Millstone 1	GE	BWR/3	NRC	no	no
Oconee 3	B&W	PWR	NRC	no	no
Peach Bottom 2	GE	BWR/4	AEC	yes	yes
Sequoyah 1	W	PWR	NRC	no	no
Shoreham	GE	BWR/4	Utility		
Surry 1	W	PWR	AEC	yes	yes
Yankee Rowe	W	PWR	Utility		
Zion 1&2	W	PWR	Utility	yes	yes

NRC - Nuclear Regulatory Commission
PWR - Pressurized Water Reactor
BWR - Boiling Water Reactor
AEC - Atomic Energy Commission

GE - General Electric
CE - Combustion Engineering
W - Westinghouse
B&W - Babcock and Wilcox

2.1.1 Initiating Event Identification

The first major step in defining accident sequences is the identification of initiating events. An initiating event can be defined as any incident that leads to a demand for safe shutdown systems such as the reactor protection system, auxiliary feedwater system, or emergency power systems. By definition then, an initiating event is the beginning point in a sequence. Hence, a comprehensive list of accident-initiating events must be compiled to ensure that all important sequences are identified. Initiating events can be component failures such as a pump stopping or a valve failing to open, or stem from outside disturbances such as a loss of off-site power or an earthquake. A partial list of initiating events, condensed from Reference 5, is given in Table 2-2. These initiating events are representative of those events that have occurred at nuclear plants or that may reasonably be expected to occur during a plant's lifetime. This list does not include other more "rare" events such as large pipe ruptures, earthquakes, hurricanes, and floods which might damage a plant.

A major uncertainty associated with the identification of initiating events is completeness; has the analyst identified and included all of the initiating events that might significantly contribute to the frequency of core melting or consequences of a reactor accident? A lack of completeness can result from the defined scope of a PRA, from the level of detail employed by the analyst, or from a lack of adequate information.

In the Reactor Safety Study (RSS),[6] most initiating events were grouped into six classes. These classes were large, medium, and small loss of coolant accidents (LOCAs), accidents initiated by a loss of offsite power, accidents initiated by a loss of feedwater, and a general class for all other transient initiators which demand a reactor trip but do not otherwise affect the reactor. In addition, a few special events, such as a reactor vessel rupture and the interfacing systems LOCA, were evaluated. The interfacing systems LOCA involves a failure of valves at the high-to-low-pressure interface in the reactor coolant piping that leads to an unmitigable loss of coolant from the reactor vessel. The RSS found that small LOCAs and transient-initiated events (e.g., a loss of offsite power) dominated the estimated core melt frequency and risk at the two plants studied. Events such as earthquakes, fires, and floods were only superficially treated in the RSS and were deemed not to be significant compared to other initiators.

Later studies, such as those done in the Interim Reliability Evaluation Program (IREP) sponsored by the NRC, recognized that other initiating events could be important. In particular, initiating events caused by partial or complete losses of support systems (i.e., systems which provide power or cooling water to the main safety systems) were found to require special consideration because they could not only lead to a reactor shutdown, but also degrade the operability of certain safety systems. For example, in the Arkansas Nuclear One-Unit 1 PRA,[7] a single failure of a valve in a service water system was

Table 2-2 Partial list of initiating events

Initiating events for PWRs

1. Loss of RCS flow (one loop)
2. Uncontrolled rod withdrawal
3. Problems with control-rod drive mechanism and/or rod drop
4. Leakage from control rods
5. Leakage in primary system
6. Low pressurizer pressure
7. Pressurizer leakage
8. High pressurizer pressure
9. Inadvertent safety injection signal
10. Containment pressure problems
11. Chemical and Volume Control System malfunction--boron dilution
12. Pressure, temperature, power imbalance--rod-position error
13. Startup of inactive coolant pump
14. Total loss of RCS flow
15. Loss or reduction in feedwater flow (one loop)
16. Total loss of feedwater flow (all loops)
17. Full or partial closure of main steam isolation valve (MSIV) (one loop)
18. Closure of all MSIVs
19. Increase in feedwater flow (one loop)
20. Increase in feedwater flow (all loops)
21. Feedwater flow instability--operator error
22. Feedwater flow instability--miscellaneous mechanical causes
23. Loss of condensate pumps (one loop)
24. Loss of condensate pumps (all loops)
25. Loss of condenser vacuum
26. Steam-generator leakage
27. Condenser leakage
28. Miscellaneous leakage in secondary system
29. Sudden opening of steam relief valves
30. Loss of circulating water
31. Loss of component cooling
32. Loss of service-water system
33. Turbine trip, throttle valve closure, electro-hydraulic control problems
34. Generator trip or generator-caused faults
35. Loss of all offsite power
36. Pressurizer spray failure
37. Loss of power to necessary plant systems
38. Spurious trips--cause unknown
39. Automatic trip--no transient condition
40. Manual trip--no transient condition
41. Fire within plant

Table 2-2 (Continued)

Initiating events for BWRs

1. Electric load rejection
2. Electric load rejection with turbine bypass valve failure
3. Turbine trip
4. Turbine trip with turbine bypass valve failure
5. Main-steam isolation valve (MSIV) closure
6. Inadvertent closure of one MSIV
7. Partial MSIV closure
8. Loss of normal condenser vacuum
9. Pressure regulator fails open
10. Pressure regulator fails closed
11. Inadvertent opening of a safety/relief valve (stuck)
12. Turbine bypass fails open
13. Turbine bypass or control valves cause increase in pressure (closed)
14. Recirculation control failure--increasing flow
15. Recirculation control failure--decreasing flow
16. Trip of one recirculation pump
17. Trip of all recirculation pumps
18. Abnormal startup of idle recirculation pump
19. Recirculation pump seizure
20. Feedwater--increasing flow at power
21. Loss of feedwater heater
22. Loss of all feedwater flow
23. Trip of one feedwater pump (or condensate pump)
24. Feedwater--low flow
25. Low feedwater flow during startup or shutdown
26. High feedwater flow during startup or shutdown
27. Rod withdrawal at power
28. High flux due to rod withdrawal at startup
29. Inadvertent insertion of control rod or rods
30. Detected fault in reactor protection system
31. Loss of offsite power
32. Loss of auxiliary power (loss of auxiliary transformer)
33. Inadvertent startup of HPCI/HPCS
34. Scram due to plant occurrences
35. Spurious trip via instrumentation, RPS fault
36. Manual scram--no out-of-tolerance condition

found to lead not only to a reactor shutdown but also to degrade the ability of the containment heat removal systems to perform their function. Accident sequences initiated by single support system failures were found to dominate the Arkansas Nuclear One PRA results, contributing over 35% to the estimated total frequency of core melting.

Other PRAs, such as those for Zion and Indian Point, have made significant contributions to the analysis of an initiating event class called external events. External events include many natural phenomena such as earthquakes, tornados, and floods but also include events such as airplane crashes that could affect a nuclear plant. Most external events are considered to be "rare" events, meaning their frequency of occurrence is thought to be low. For example, an earthquake of sufficient magnitude to damage a nuclear plant may be estimated to occur only once in a million years. This has sometimes, as in the RSS, caused the risk from external events to be considered insignificant compared to other initiators which occur more frequently. However, in more recent studies this has not been the case. In the Zion PRA,[8] a more detailed analysis was performed and earthquakes were calculated to be the most important contributor (greater than 50%) to the risk of early fatalities stemming from postulated nuclear accidents. In the Indian Point 2 PRA, external events were very important contributors to both the frequency of core melting (80%) and the risk of early fatalities (60%).[9]

It is clear that the initiating events selected for study in a PRA and the depth to which these events are evaluated will affect the completeness of the results and thereby the associated uncertainty. The previous examples show that PRA results can easily be low by 50% if support system initiators and external event initiators are not evaluated. If future studies include these initiators and others that have been identified as important, the uncertainty in the definition of accident sequence initiating events will be smaller than if some initiators are omitted.

2.1.2 Accident Sequence Delineation

The accident sequence delineation activity in a PRA involves the construction of event trees to represent plant responses to initiating events. An event tree is constructed by postulating the success or failure of each safety-related system in the context of all the boundary conditions established by an initiating event. Only those unique combinations of system successes and failures that have physical meaning are included in an event tree. For example, the failure of one system may preclude the success of another.

While the logical and integrated approach is a strength, there are basic uncertainties with regard to how well event trees are able to represent the actual conditions associated with a plant's design, operation, and response to accident conditions. There are obvious limitations in our ability to faithfully represent the real world by analytical models. As an example, event trees are binary models and tend to show only discrete on-off, yes-no type situations, whereas in reality, a plant response may be in gradations of partial failures or complex events involving degraded system operation. Model uncertainties are acknowledged and addressed by efforts to make models as realistic as possible.

Another basic uncertainty concerns completeness; have all the potentially significant accident sequences been identified and properly characterized? A major cause of this type of uncertainty in the delineation of accident sequences is the lack of knowledge of how reactor systems and operators might respond to accident conditions. For example, in the Reactor Safety Study and other PRAs it was assumed that, given a small LOCA, a reactor would eventually have to go into a recirculation mode of operation where cooling water is circulated from the containment sump, through the reactor, out of the small break in the reactor coolant system, and back into the sump. An accident sequence that involved failure of the recirculation systems after a small LOCA was identified on the RSS event trees for Surry and found to be important, producing about 10% of the total core melt frequency. Current assessments of this accident sequence suggest that it may be possible for the operator to depressurize the reactor and attain a safe condition before the need to enter the recirculation mode is ever required. If this operator action is credible at the Surry plant, then the frequency of core melting could be overestimated by about 10%. This example illustrates how PRA results might be overestimated if the set of accident sequences being evaluated is not complete.

Given the limitations in delineating accident sequences there is little doubt that there are many alternative accident scenarios that will not be identified. Many of these scenarios are probably not important and would not significantly affect PRA results. Some, as in the given example, would reduce the estimated probability of a core melt occurring. Others might make things worse. Until the Three Mile Island accident, the possibility of an operator mistakenly turning off a safety system had not been considered. As additional research and operating experience is gained, the uncertainty in delineating accident sequences should be reduced.

2.1.3 System Success Criteria Definition

System success criteria define how and in what combination systems must perform to successfully respond to an initiating event. For example, given a loss of offsite power at a nuclear plant, operation of two of the three auxiliary feedwater pumps might be required to prevent excessive pressure from building up in the reactor coolant system. If only one or none of the auxiliary feedwater pumps successfully operates, then the pressure would build up and other systems would be demanded. Thus, the success criterion for auxiliary feedwater would not have been met and the accident would continue.

One type of uncertainty in establishing accurate success criteria deals with a lack of understanding of system capabilities during accident conditions. The success criteria used in many PRAs are based on criteria published in the Final Safety Analysis Report (FSAR) for a plant. These FSAR criteria are based on licensing calculations for accidents which do not lead to fuel damage or melting (e.g., a large LOCA with no concurrent failures of redundant systems). Explicit thermal-hydraulic calculations for the accident sequences of interest in PRAs are often difficult to obtain due to a lack of adequate models or restrictions in the scope, schedule, or budget of a program. Phenomena that may occur during an accident and that are not well understood or modeled may impact the success criteria.

An example of an uncertainty in system capability is the number of high pressure injection (HPI) pumps needed to maintain core cooling given a small LOCA combined with a loss of all secondary cooling in a PWR. The FSAR might say that two of three HPI pumps are required after a small LOCA but does not give the criterion for a small LOCA combined with a loss of secondary cooling. Unless further information is available the analyst will have difficulty in rationalizing a criterion to use. Maybe three pumps would be sufficient or perhaps the pressure in the primary might rise above the discharge pressure of the pumps, causing HPI to become completely ineffective. Other examples of uncertainties that involve system capabilities are given in Table 2-3.

A second type of uncertainty involving system success criteria deals with the binary success/failure format used in PRAs. Components modeled in the fault trees are assumed to be either fully operational or completely failed with some probability. Partial successes (or failures) such as a pump running at half speed or a valve failing halfway closed are not generally modeled on a fault tree. Perhaps in the previous example two HPI pumps running at full speed and one pump running at half speed would be sufficient to provide core cooling. A fault tree analyst will consider these partial states and make assumptions as to which binary state (complete success/failure) they conservatively fall into. Partial component successes could be explicitly modeled on fault trees; however, the detailed information needed to do this is generally not available and the size of the fault trees would become excessive if all partial states were modeled.

A final uncertainty concerns the issue of completeness in the definition of success criteria: have all systems which could mitigate or exacerbate an accident been included in the analysis? PRAs have generally not given credit for the operation of some nonsafety-related systems or for systems operating in an out-of-normal mode or configuration. For example, many PRAs have not included the control rod drive system (for BWRs) or fire protection systems (PWRs and BWRs) as possible sources for emergency coolant injection. An example of how this can affect PRA results comes from an analysis of the Grand Gulf BWR.[10] In this report, a mode of core cooling was evaluated which involved connecting a service water system with the reactor coolant system so that water could be pumped at low pressures directly from an outside reservoir to the reactor core. This mode of operation, while feasible, was not included in the original risk assessment of Grand Gulf. Its inclusion led to a reduction of the core melt frequency calculated for Grand Gulf by approximately 50%. It should be noted that the assessment made in the original analysis of Grand Gulf lies

Table 2-3 Examples of uncertainties in system capabilities

- Capability of systems to operate under loss of pump lubrication, oil cooling, or pump room cooling

- Capability of establishing decay heat removal in PWRs using high pressure injection and depressurization with the relief valves (feed and bleed)

- Capability of steam-driven systems, such as the reactor core isolation cooling (RCIC) system in BWRs and the auxiliary feedwater system (AFWS) steam train in PWRs, to operate under loss of reactor coolant system (RCS) integrity conditions. For example, the degree to which the RCIC turbine is starved of steam with a stuck-open relief valve in the RCS, and the effects of these conditions on the efficiency of the RCIC's performance, are uncertain.

- Whether single (or even multiple) trains of high pressure injection system (HPIS) operation can adequately maintain water level and boron concentration in a PWR under anticipated-transient-without-scram (ATWS) conditions, particularly with a stuck-open RCS relief valve

- Capability to use only the condensate pumps in conjunction with steam relief valves and/or steam-generator-blowdown valves to provide successful RCS cooling in PWRs

- Capability of control-rod-drive (CRD) pumps in BWRs to provide sufficient water inventory to avoid core damage under some transient conditions

- Containment blowdown following containment failure, or containment deformation under high pressure conditions, could affect core-cooling piping, equipment, and/or the core-cooling-recirculation-water supply

- Capability of containment fans and H_2 recombiners to perform their functions, once considerable core damage has occurred and aerosols are distributed throughout the containment environment

- Capability of systems to continue to operate, given a H_2 burn or any other sudden pressure spike, such as in-vessel and ex-vessel steam explosions

- Effect of higher-than-design temperature, pressure, and radiation conditions on continued operability of containment systems on the conservative side; i.e., consideration of extra systems should improve the probability of success.

2.1.4 System Modeling

System modeling involves constructing fault trees for each of the important support and safety systems of a nuclear plant. A fault tree is a graphic model of the various parallel and sequential combinations of faults that will result in the occurrence of some predefined undesired (or top) event. The faults can be associated with component hardware failures, human errors, or any other pertinent faults that can lead to the top event. The fault tree approach is a deductive process, whereby the top event is postulated and the possible means for that event to occur are systematically deduced.

It should be noted that a fault tree does not contain all possible component failure modes but includes only the events considered to be significant by the analyst. The choice of faults for inclusion is not arbitrary; it is guided by detailed fault-tree procedures, information on system design and operation, operating histories, input from plant personnel, the level of detail at which basic data are available, and the experience of the analyst. However, the issue of completeness must be addressed and is a source of uncertainty. If all of the important failure modes of a component are not found, the results of a PRA will underestimate the frequency and risk from nuclear accidents.

Most accident sequences of interest in PRAs involve fuel melting, reactor vessel failure, and containment failure. Under these conditions, safety equipment inside containment will be exposed to high pressures, temperatures, and radiation which exceed what the equipment is environmentally qualified for. This makes it difficult for the fault tree analyst to accurately assess how and when a component might fail. Other concerns are the extent and type of debris that fills the containment atmosphere during an accident, the possibility of fires or explosions, and the corrosiveness of the environment around components.

One general class of component failure modes that are hard to identify and evaluate is that of dependent failures. Dependent failure can be defined as a combination of failures whose probabilities are correlated by some physical or environmental condition. Common-cause failures, common-mode failures, and system interactions are all considered to be types of dependent failures. A good example of a dependent failure occurred at the Salem reactor in 1983. Several automatic reactor shutdown breakers which were made by the same company simultaneously failed because of a common inadequacy in the plant's maintenance procedures. Other examples of possible causes of dependent failures include high temperatures, dust, and vibration.

An example of how this uncertainty can affect PRA results can be found in the Zion PRA [8] and a subsequent review of this study performed by Sandia and Brookhaven National Laboratories.[11] In the original Zion analysis it was assumed that a core melt environment would not significantly affect the operation of the containment fan cooling system which helps to prevent post-accident containment overpressurization. Upon review, this assumption was questioned and a sensitivity calculation was performed to estimate its effect on the results. The sensitivity calculations showed that if the fans fail after a core melt, the original number of total early injuries, latent fatalities, and total radiation dose received by the public would each increase by approximately a factor of 3.

2.1.5 Human Reliability Modeling

Human reliability analysis (HRA) is a method by which human reliability is estimated. In carrying out an HRA, it is necessary to identify those human actions that can have an effect (either positive or negative) on system reliability or availability. The most common application of HRA is the evaluation of human acts required in a system context. The consideration of extraneous actions is also important. The person interacting with a system may not only fail to do what he is supposed to do, or fail to do it correctly, but he may also do something extraneous which could degrade the system. The latter is a weakness of HRA. It is not possible to anticipate all undesirable extraneous human actions. The best anyone can do is to identify those actions having the greatest potential for degrading system reliability or availability.

There are two major sources of uncertainty in modeling the occurrence of human errors in the operation of nuclear plants: 1) the inexactness of models of human performance that purport to describe how people act in various situations and conditions and, 2) the identification of all relevant factors that shape human performance and their interactions and effects.

The state-of-the-art of HRA is such that the modeling of human behavior can qualitatively account for its variability and for discrepancies in human response situations, but there are definite limitations in quantifying such models. There are many models of human performance, but few can be used to estimate the probability of correct or incorrect human performance in applied situations. Furthermore, all models, even those that can be applied to HRA (e.g., the models in Reference 12) are themselves abstractions of real-world circumstances. As such, they only partially represent the situations they simulate. In some cases, experimental data have provided strong support for the general form of the models, but in others the forms are still speculative (although based on sound psychological concepts).

Another source of uncertainty, the identification of the factors that shape human performance associated with a task, also involves some abstraction and is subject to some interpretation on the part of the analyst. This is probably the biggest source of error in extrapolating data from other sources to the nuclear power plant. Unless the tasks required in both situations are analyzed in sufficient detail, data from other sources may be misapplied to the tasks performed in a nuclear power plant. For example, a valve restoration task in a chemical processing plant may be superficially similar to an equivalent task in a nuclear power plant, but the Human Error Probability (HEP) from the chemical plant may be based on errors made by people using well-designed checklists, whereas the valve restoration procedures carried out in the nuclear power plant may be performed from memory only. Using the HEP from the chemical plant to estimate the HEP for the nuclear power plant would obviously result in a gross underestimation of the true HEP.

Many of the PRAs published to date have identified at least one or two human errors that contribute significantly to severe accidents. Some examples of important human errors identified in PRAs are given in Table 2-4. Examples of just how important human errors can be in PRA results are found in Reference 13. In this study, the sensitivity of core melt frequencies to changes in the unavailabilities of systems and individual faults was calculated. For the Millstone BWR, it was found that the failure of the reactor operators to manually depressurize the reactor during particular accidents was important. The results showed that a 1% change in the probability of this human error alone would cause a 0.05% change in the total core melt frequency. Therefore, if the uncertainty in the HEP for this human action is many orders of magnitude, the resulting uncertainty in the core melt frequency would be large.

To summarize, the most significant contributors to uncertainty in evaluating the human behavior of nuclear power plant operators is the inexactness of the models and identification of pertinent human performance factors. No abstraction can fully define or account for all the variables in response situations as complex as those found in a nuclear power plant. Furthermore, it is unrealistic to suppose that each model will be applied consistently across all analyses. This lack of consistency is related to the difficulties in performing the necessary analyses of human inputs, mediating processes, and responses so that the relevant performance-shaping factors can be identified and assessed correctly.

2.2 Accident Sequence Quantification

The results of the analyses done to identify accident-initiating events and to develop event trees and fault trees that depict "what could go wrong," and thus result in core damage, are mathematical models. These models express the occurrence rate of an accident, or class of accidents, as a function of initiating event rates, component and system failure probabilities, human error probabilities, and perhaps other parameters (unknown constants). Estimating these parameters and thence the various accident sequence rates provides a basis for comparing and evaluating these sequences. Estimates, though, are imprecise (to varying degrees), so there is uncertainty in the assessments of "how likely" things are to go wrong. Such uncertainty needs to be evaluated to provide a proper frame of reference for comparisons, judgments, or actions that might be taken pertinent to risk.

The imprecision of an estimate depends on the data or other information on which the estimate is based. At the "front end" of a risk analysis (the progression of an accident from initiating event to core damage), the potential for data -- the actual experience of operating

Table 2-4 Examples of human errors in LWR analyses

Item	Comments
Operator failure to manually initiate HPIS for 3 types of conditions: ATWS, feed & bleed following AFWS failure, & small LOCAs where auto initiation conditions aren't reached.	ATWS - HEPs of 0.1 used in ANO-1, Oconee PRAs for high stress. Feed & Bleed - CR-3, Oconee PRAs used HEPs of 1E-2. Value could change depending on ease of initiating feed & bleed. Small LOCAs - Values similar to Feed & Bleed case. How does operator recognize the need for HPIS?
Operator failure to initiate & maintain recirculation cooling following LOCAs (large to small)	Many examples available including switch to recirculation prematurely (CR-3, 5E-2), switch too late (CR-3, 3E-3; Oconee, 3E-3; Surry, 3E-3), errors during switching process (CR-3, 8E-2), or later failures such as inadvertently shutting off pumps (CR-3, 5E-2) or failure to realign to hot legs in 24 hours (Sequoyah, 6E-3; Surry, 3E-3).
Operator failure to successfully depressurize and cooldown plant following a small LOCA so that there is no need to "go to recirculation cooling."	How does operator recognize the LOCA? What do procedures say to do? What are likely failures and their probability?
Operator failure to recognize stuck open PORV/SRV and close block valve.	Calvert PRA used 0.1. TMI accident had this failure. How does operator recognize problems & perform action?
Operator failure to extend battery life, water sources, and be prepared for local manual system operations during extended station blackout.	What is variability in depth of procedure/training on these issues? What are likely failures and their probabilities?
Operator failure to manually depressurize primary system when high pressure injection systems have failed (BWRs).	How is high pressure cooling failure recognized? Will there be a reluctance to depressurize?

Table 2-4 (Continued)

Item	Comments
Operator failure to perform correct actions following an ATWS such as initiating emergency boration systems.	What are current procedures/training for handling ATWS? Is it clear when emergency boron should be initiated?
Operator errors following component test or maintenance.	Examples include miscalibration of Reactor Protection System. Are there large variations in test and maintenance procedures?
Operator failure to perform recovery or repair actions to increase system availability.	What is variation in the depth of operator training and/or procedures on this issue?

nuclear plants -- is much greater than it is for post-core-melt phenomena and consequences. That is, a wide variety of initiating events have actually happened and the performance of many components and systems in operating plants has been observed and recorded, whereas data pertaining to post-core melt phenomena are available only (fortunately) from reduced-scale experiments, if at all. The availability of data and the relative simplicity of accident sequence models (compared to post-core-melt models) provide an excellent opportunity for a quantitative assessment of the uncertainty in estimated accident sequence rates. A data-based assessment of uncertainty also provides guidance for subsequent data collection and uncertainty reduction. However, there are gaps in the data, such as with respect to human errors, and concerns about data quality and applicability. Also, available data may yield quite imprecise estimates, so all is not favorable. The following subsections evaluate the present situation with respect to estimating initiating event rates, component failure probabilities, and human error probabilities.

2.2.1 Initiating Event Frequency Estimation

In accident sequence models, initiating events are generally assumed to occur at a constant rate, over time. This is primarily an assumption of convenience; without it one would have different estimated risks year by year. Available data, such as in EPRI NP-2230,[5] support this assumption in some cases, contradict it in others. Over all "transients", there is evidence of a decrease in occurrence frequency over the first four years of operation. For the rarer events, not enough data exist to show much of a trend. With respect to possible aging effects, there are not yet enough end-of-life data to assess the error in estimates based on the assumption of constant initiating event rates, if in fact there is an aging trend. (On the other hand, if there is an initial decreasing trend in initiating event frequency, data over the early years of plant life, which is what we mostly have now, would lead to overestimates of current occurrence rates.)

Given the assumption of a constant occurrence rate, the data at a plant of interest may still be too limited to yield usefully precise estimates. For example, suppose a three-year-old plant has had no large LOCA, small LOCA, or loss-of-offsite power. Such data provide almost no information about differences among the occurrence rates of these events, while other considerations would suggest possibly major differences. These other considerations include data from other plants. Data from other plants can be directly used--that is, the occurrences and the operating times from these plants can be added to those of the plant of interest--if there is reason to assume the occurrence rate is not only constant over time, but also across plants. Otherwise, an analysis needs to be done that reflects plant-to-plant variation. The data in EPRI NP-2230 again support the assumption of equal initiating event rates across plants in some cases, contradict it in others.

For example, Table 2-5 is an excerpt from EPRI-NP 2230 and shows the occurrences of "Turbine Trip, Throttle Valve Closure, EHC (electro-mechanical hydraulic control) Problems" for seven plants over nine years of operation.[5]

Table 2-5 Selected initiating event annual occurrences: Turbine trip, throttle valve closure, EHC problems*

	Year of operation									
Plant	1	2	3	4	5	6	7	8	9	Total
1. Yankee Rowe	0	0	4	2	0	0	0	0	0	6
2. Ind. Pt. 1	2	1	1	3	3	4	1	0	0	15
3. San Onofre	0	1	1	4	1	1	0	1	1	10
4. Haddam Neck	2	4	1	1	0	0	0	0	1	9
5. Ginna	1	1	0	0	2	1	0	0	0	5
6. Pt. Beach 1	4	1	0	0	1	0	0	1	0	7
7. Robinson	6	3	3	3	0	1	3	3	4	26
TOTAL	15	11	10	13	7	7	4	5	6	78

*Source, Reference 5, page A-37

A statistical analysis of this table shows fairly strong evidence of nonconstant occurrence rates over time and among plants. That is, the differences among plants and years are larger than would be expected, by chance alone, under the assumption of a single occurrence rate. If these data were to be used to estimate the occurrence rate of this transient at a new plant, these differences would have to be accounted for in assessing the uncertainty of that estimate. The uncertainty would be much greater than that obtained from the combined data of 78 occurrences in 63 reactor-years and the assumption of a constant underlying occurrence rate. To illustrate, suppose that the yearly trend is ignored and that the objective of the analysis is to estimate the occurence rate at a new plant. An analysis of variance and a statistical prediction analysis leads to an approximate 90% statistical confidence interval on the rate of 0.3 to 3.5 occurrences/yr. The pooled data, and the assumption that all plants have the same occurrence rate, would lead to an (unrealistically narrow) rate of 1.0 to 1.5 occurrences/yr. Thus, plant-to-plant differences, which are treated as "random variation" in the statistical prediction analysis, have a sizable effect, in this case, on the uncertainty with which an occurrence rate for a new plant can be estimated.

For the case of a constant occurrence rate and pertinent data of n occurrences in T reactor-years, the conventional statistical estimate of the underlying occurrence rate is n/T. For example, if the analysis objective was to estimate the Ginna turbine trip rate, the Table 2-5 data lead to n/T = 5/9 = 0.56/yr. This just a point estimate of the underlying rate, which could in fact be somewhat, or substantially, different from 0.56/yr. Statistical confidence limits identify a plausible range for this underlying rate. In particular, given data of n occurrences in T years, the upper 95% statistical confidence limit on the underlying annual occurrence rate is $\chi^2(2n + 2; 95)/2T$, where $\chi^2(f, c)$ denotes the cth percentile of the chi-squared distribution with f degrees of freedom. For the Ginna data, this limit is equal to 1.2/yr. What this confidence limit means is that if the underlying occurrence rate were above 1.2/yr, the observed data would be fairly unlikely--the chance of 5 or fewer occurrences in 9 years would be 5% or less. Thus, values of the rate above the limit are inconsistent with the data (to the extent indicated). The lower 95% statistical confidence limit on the underlying rate is $\chi^2(2n, 5)/2T$. Values of the rate below this limit, which is equal to 0.2/yr for the Ginna data, are similarly inconsistent with the data. If the true rate were 0.2/yr or less, the chance of 5 or more occurrences in 9 years would be 5% or less. By convention, the interval between the lower and upper 95% confidence limits is called a 90% statistical confidence interval. These confidence limits are given in Reference 14 and chi-squared tables are given there and in many other books.

The relative precision with which an initiating event rate can be estimated is indicated by the ratio of the upper 95% limit to the lower 95% limit. This ratio is a function only of n (T cancels out), so relative precision is controlled by n. Table 2-6 gives the upper and lower 95% statistical confidence limits, in units of 1/T. Also tablulated is the ratio of the upper limit to the lower limit. (The limits have been rounded so their ratio is not always equal to the

Table 2-6 Statistical confidence limits on a constant occurrence rate, in units of 1/T

n	Lower 95% Limit	Upper 95% Limit	Ratio
0	--	3.0	--
1	0.05	4.8	92
2	0.4	6.3	18
3	0.8	7.8	9.5
5	2.0	10.5	5.3
7	3.3	13.2	4.0
10	5.5	17.0	3.1
15	9.3	23.1	2.5
20	13.3	29.1	2.2
30	21.6	40.7	1.9
50	39.0	62.2	1.6

tabulated ratio.) Figure 2-1 gives a plot of this ratio as a function of n and thus shows graphically how precision increases with increasing n. Note that one occurrence yields essentially two orders of magnitude of uncertainty (as quantified here), three yield about one order of magnitude, etc.

It should be added that any statistical analysis is conditional on the assumption of "good" data. Uneven or poor data quality introduces additional uncertainty that is not quantifiable. For example, inaccurate or inconsistent reporting of initiating events would vitiate any analysis of the data in Table 2-5.

A popular way of expressing subjective (as opposed to statistical) uncertainty is by way of a lognormal degree-of-belief distribution for a parameter, such as an occurrence rate. This distribution is often characterized by a median and an "error factor." The error factor squared is the ratio of the 95th to 5th percentile of a lognormal distribution. Though statistical confidence limits and subjective probability limits are not at all the same concepts, one can see from Table 2-6 that an error factor of 10 corresponds roughly to data of one occurrence in T years; an error factor of 3 corresponds to essentially three occurrences. This correspondence is an aid in interpreting subjective error factors. For example, consider a nominal subjective estimate of 10^{-4}/yr with an error factor of 10. To obtain a comparable assessment based on a statistical confidence interval would

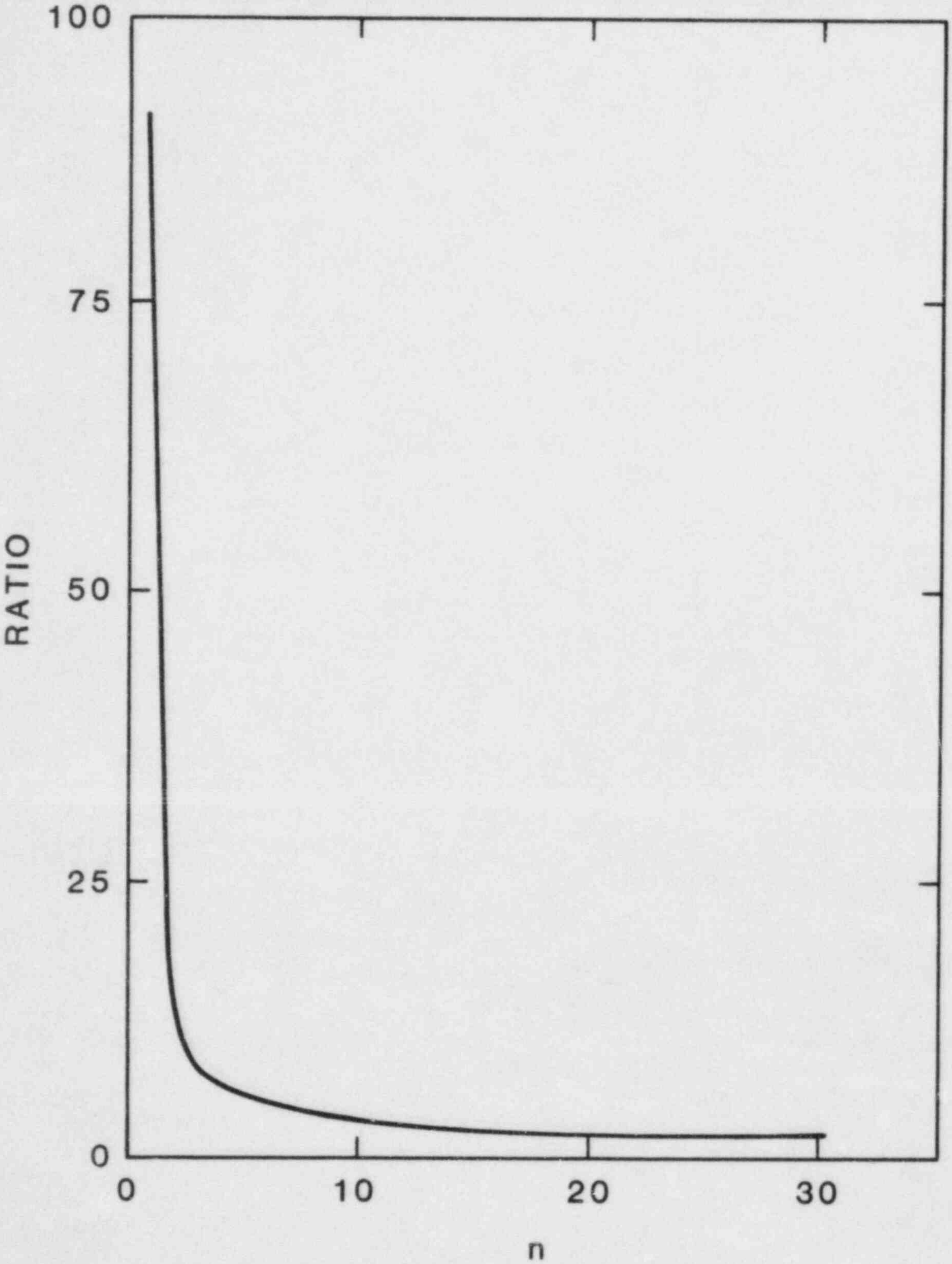

Figure 2-1 Ratio of upper to lower 95% statistical confidence limits on an occurrence rate, given n occurrences of the event.

require data of one occurrence in about 10 000 years. That is, one would need such data for a statistical uncertainty assessment to match this subjective assessment.

For the case of n = 0 occurrences, all that can be inferred statistically is an upper bound. If an event has not occurred, then there is no statistical basis to rule out the possibility that the underlying rate is actually zero. Thus, zero would be the lower bound, which means relative precision is undefined. The conventional statistical point estimate of the occurrence rate, n/T, without modification, equals zero in this case, which of course is an imprudent value to assume in a PRA. Some indication of the information provided by data of no occurrences in T years and the assumption that the underlying rate is constant is given by the upper 95% statistical confidence limit of 3/T and the upper 50% limit of 0.7/T. However, a major uncertainty in any assessment for an event which has not occurred is the specification of the denominator, T. For example, in estimating the large LOCA rate for a particular plant, what experience should be used: U.S.? World-wide? Vendor-specific? ... Pooling experience over any set of reactors carries the (uncertain) assumption of a constant occurrence rate over that set of reactors. Data of no occurrences provide no guidance on what experience is relevant, so such a choice has to be based on other considerations which also may be quite uncertain.

Estimates of initiating-event rates in published PRAs have not been wholly statistical in nature, but, because of data limitations or unavailability and analyst philosophy, have been based at least in part on analyst judgment and beliefs. Reactor Safety Study [6] estimates, which have been used repeatedly, were based on an informal amalgamation of data and judgment. More recent estimates have been based on a formal Bayesian analysis, whereby judgment is expressed probabilistically (the "prior" distribution), then merged with data via Bayes's Theorem to provide a "posterior" probability distribution representing the analyst's so-called "state-of-knowledge," or "degree-of-belief" about the occurrence rate of an initiating event. A Bayesian analysis can reduce the apparent uncertainty in an estimate, but it can also increase uncertainty because it introduces another source of uncertainty: analyst-to-analyst differences.

Consider the case of a large LOCA, an event which has never happened. In the Bayesian analyses done for the Zion and Indian Point risk assessments,[8 9] the experiential data cited in each case were zero occurrences in 131 reactor-years (U.S. PWRs). These data alone yield an upper 95% statistical confidence limit on the large LOCA rate of 0.023/yr. After incorporating their prior distribution, the Zion authors obtained a posterior 95th percentile of 0.0036/yr. To obtain this value as a statistical 95% confidence limit would require no occurrences in 833 reactor-years. Thus, from a statistical standpoint, the assumed prior distribution (not disclosed or substantiated in the report) effectively added 702 LOCA-free years to the data base. The same group of analysts (but a different individual), in performing the Indian Point study, chose a different prior distribution and obtained a posterior 95th percentile on the large LOCA rate of 0.0063/yr for Unit 2. This corresponds to adding 345 LOCA-free years to the recorded experience of 131 reactor-years. Thus, analyst differences, even within the same team, amounted to a factor of two in this case. Independent subjective estimates of rare event probabilities can differ even more, as experience with earthquake frequency estimation shows.

The occurrence rates of extreme natural hazards, such as earthquakes and hurricanes, are extremely difficult to estimate because of the relative briefness of recorded history and the lack of understanding of the physical laws that govern their occurrence. Thus, "informed opinion" has been called upon for estimates. For example, Okrent polled seven experts in the field of earthquakes and obtained independent estimates of the occurrence rates of large earthquakes at a variety of sites.[16] The results showed considerable differences of opinion, often spanning two to four orders of magnitude. For example, for one specified large earthquake (MMI VIII) at the Pilgrim site in Massachusetts, the estimated annual rates of the five experts responding ranged from 10^{-7} to 2×10^{-3}/yr.

Another example of earthquake estimation uncertainty is provided by the Indian Point analysis.[9] Two consulting firms provided families of seismicity curves (exceedance frequency vs. peak ground acceleration). A major uncertainty concerns the maximum possible ground acceleration. The curves provided (see Figure 2-2, which is Figure 7.2-4 in volume 10 of the Indian Point Probabilistic Safety Study) indicate opinions on the maximum acceleration that ranged from 0.25 to 0.8 g. The estimated occurrence rates of these maximum earthquakes were about 10^{-5}/yr. The curves also indicate that for an event yielding 0.4 g acceleration, the estimated occurrence rate ranges from zero to about 10^{-4}/yr. Thus, again, analyst uncertainty is substantial. However, in the Indian Point analysis the different curves were assigned probability weights, shown in Figure 2-2, so that analyst uncertainty was treated as random variation.

2.2.2 Component Failure Probability Estimation

Estimating accident sequence rates requires estimating the probabilities of the combinations of component failures that lead from the initiating event to core damage. The uncertainty of component estimates can have a substantial impact on the uncertainty of an accident sequence estimate. For example, because system failures often involve dual failures, one order of magnitude uncertainty for a component becomes two orders for the system.

As is the case with initiating event rates, different PRAs have used different component estimates based on different sources of information. The Reactor Safety Study used a combination of nuclear plant data, other data, and judgment.[6] The IREP analyses generally used the RSS estimates, modified in some cases by findings from data collected after the RSS.[17] The attendant subjective uncertainty was expressed as lognormal probability distributions. That is, analyst "degree-of-belief" about a component failure probability was expressed as a lognormal probability distribution characterized by a median and

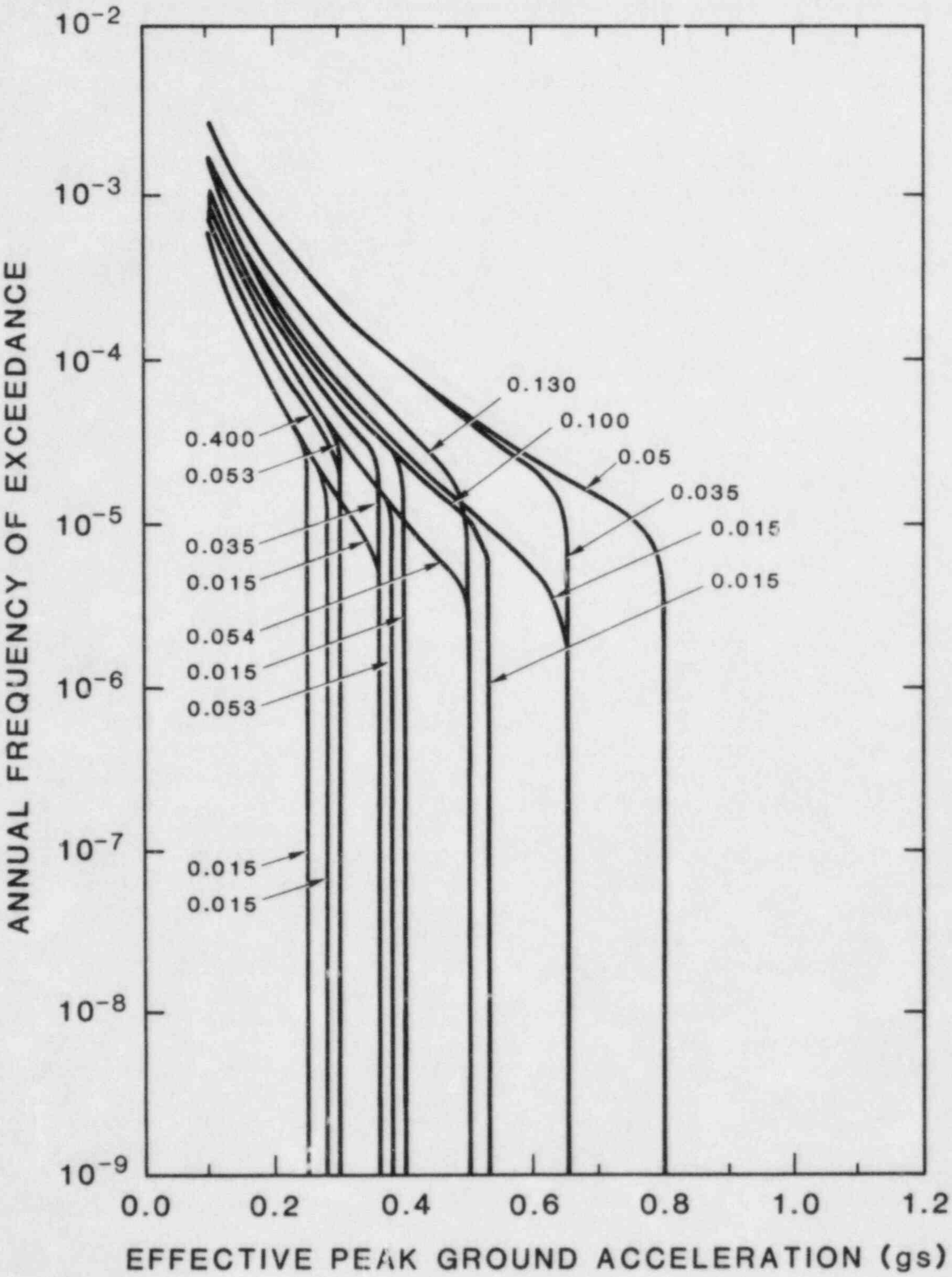

Figure 2-2 Possible seismicity curves for Indian Point (reproduced from Reference 9)

an "error factor" (ratio of the distribution's 95th percentile to its median).

The type of uncertainty meant to be covered by an error factor is not always clear. One might be considering differences in component performance from plant-to-plant, or performance differences among components of the same type within a plant, or performance differences of a single component in different environments, or some combination of these and other sources of uncertainty. Such ambiguity makes it difficult to interpret uncertainty assessments of the type described. However, even if the sources of uncertainty are clearly specified, the assumption, say, that the plant-to-plant distribution of component failure probabilities is completely known, is unwarranted. Thus, the interpretation that must be attached to the RSS and related uncertainty distributions is that of subjective uncertainty. The failure probability of all components within a given class, at a plant, is assumed to be an unknown constant, p. Subjective uncertainty about p is then expressed as a lognormal distribution. Such an expression may be quite arbitrary and of little force for others who don't share the same (quantified) degree-of-belief.

The clear need for better-substantiated estimates of component failure probabilities has led to NRC- and industry-sponsored data collection efforts. Estimates based on licensee event reports (LERs) have been published in a series of reports by INEL.[18-20] A sample is given in Table 2-7. Because the data are available plant by plant, it is possible to assess plant-to-plant variation. Note that in Table 2-7 estimated failure probabilities (the last column) range from 0.0012 to 0.061. It should also be noted, though, that there are differences in reporting requirements and diligence among plants, so one cannot tell the extent to which observed differences reflect performance or reporting differences. Another, so far unquantifiable, source of uncertainty in the LER data is the fact that the denominators--the number of attempts to start a pump, for example--are not known from plant records, but must be estimated. Typically, one might assume monthly tests in the period spanned by the data and then estimate the demands accordingly. Demands on other than monthly tests would not be counted, though presumably failures in such cases would be reported.

A more recent data collection and analysis effort has been that of the In-Plant Reliability Data System (IPRDS).[21 22] In this program, maintenance records have been collected and analyzed by NRC contractors. This data system is closer to actual plant experience and removes some of the uncertainty of LER reporting. The uncertainty of estimated denominators remains, however. As of yet, no PRAs have used the IPRDS. Neither have they used the NPRDS (Nuclear Plant Reliability Data System) which is a voluntary reporting system and hence very uneven in reporting quality.

Some PRAs have performed Bayesian analyses using plant-specific data (retrieved from LERs and plant operating logs) and subjective prior probability distributions based somewhat on the RSS subjective distribitions and the industry-wide LER summaries. This is the approach of the Zion and Indian Point PRAs.

Table 2-7 Sample Licensee Event Report Summary, excerpted from Reference 18

GENERAL ELECTRIC — VALVE--OPERATOR (MOTOR)--FAIL TO OPERATE

	Component		Population		Estimated
Plant	Number of Valves	Number of Demands per Valve	Total Number of Failures	Total Number of Demands	Failure Probability (Failures/Demand)
BF1	66	20	5	1260	4.0E-03
BF2	63	20	2	1260	1.6E-03
BF3	63	18	4	1134	3.5E-03
BR1	72	17	7	1224	5.7E-03
BR2	72	20	4	1440	2.8E-03
CD1	57	20	11	1140	9.6E-03
DA1	67	20	5	1340	3.7E-03
DR2	51	20	0	1020	8.8E-03
DR3	51	20	5	1020	4.9E-03
EN1	68	20	7	1360	5.1E-03
EN2	67	10	10	670	1.5E-02
FP1	10	20	22	360	6.1E-02
NI1	34	20	1	680	1.5E-03
MO1	47	20	6	940	6.4E-03
NM1	21	20	1	420	2.4E-03
OC1	42	20	1	840	1.2E-03
PB2	58	20	4	1160	1.4E-03
PB3	58	28	8	1160	6.9E-03
PI1	47	20	5	940	5.3E-03
QC1	57	20	6	1140	5.3E-03
QC2	57	20	6	1140	5.3E-03
VV1	56	20	6	1120	5.4E-03
TOTALS			135	22768	5.9E-03

Not surprisingly, the variety of approaches to estimation and the variety of information used has led to a variety of estimates and assessed uncertainties. A selection of these is given in Table 2-8. For pumps and valves, the nominal values are all within an order of magnitude of each other. As mentioned above, such uncertainty in component failure probabilities can translate into up to two orders of magnitude in a sequence estimate. Also, note that actual data tend to suggest higher probabilities than the RSS and IREP estimates.

2.2.3 Dependent Failures

Severe accidents generally require multiple component failures. If the occurrences of multiple failures, say of redundant pumps, are statistically independent, then failure probabilities can be multiplied in the mathematical model of the accident sequence. However, concern about the possibility of nonindependent failures exists. Some dependencies, such as the dependence of components on common support systems, are generally explicitly modeled, but other subtle dependencies may remain. Thus, estimating the probability of multiple failures may involve more than just multiplying estimated probabilities.

One model for dependent failures is referred to as the beta-factor method.[23] Consider the case of failure of two pumps and let p denote the failure probability of a single pump. Then the beta-factor model for two-pump failure is

$$P(\text{two pumps}) = p(p + \beta).$$

Some attempts have been made to estimate values of β from LER data.[24 25] These estimates are based on industry-wide experience and their applicability to plant-specific dependencies is a source of uncertainty. Other estimates have been primarily subjective. For example, the Zion and Indian Point analysts assumed a lognormal state-of-knowledge distribution for β, with a mean of 0.014. The point to be made, though, is that the modeling of dependent failures and the estimation of parameters in those models is presently an important source of uncertainty.

2.2.4 Human Error Probability Estimates

There are three main sources of uncertainty in estimating human error probabilities:

1. The unavailability and quality of data on human performance in nuclear power plants.

2. The inherent variability of human performance by one individual, as well as across different individuals.

3. The use of "informed opinion" to provide subjective estimates. The following discussion of these sources of uncertainty is adapted from the PRA Procedures Guide.[26]

Table 2-8 Selected estimates of component failure probabilities

Component/Event		RSS[6]	IREP[17]	Zion[8]	IP-2[9]	IP-3[9]	LER[18 19 20]	IPRDS[21 22]
Pumps Motor-Driven/	med.*	10^{-3}	10^{-3}	$6x10^{-4}$	$6.4x10^{-3}$	10^{-3}	$3x10^{-4}$	$5.3x10^{-3}$
Fail-to-operate	(EF)*	(3)	(10)	(2.5)	(2)	(3)	---	---
Valves, MOV/ Fail-to-	med.	10^{-3}	10^{-3}	$1.5x10^{-3}$	$2.8x10^{-3}$	10^{-3}	$4x10^{-3}$	$6.4x10^{-3}$
operate	(EF)	(3)	(10)	(1.3)	(2)	(4)	---	---
Diesel Generators/	med.	.03	.03	.018	.012	.013	.044	---
Fail-to-start	(EF)	(3)	(3)	(1.4)	(2)	(2)	---	---

*The RSS and the IREP Procedures Guide authors expressed their nominal failure probabilities and attendant uncertainty in terms of lognormal distributions characterized by a median and an error factor.[6 17] The Zion and Indian Point authors summarized their posterior distributions by their mean and variance.[8 9] These moments have been equated to the corresponding lognormal moments in order to obtain approximate medians and error factors, for the sake of comparison. The LER and IPRDS nominal values are the overall observed failure frequencies.[18-22] Because the analyses performed in the appropriate LER are based on the assumption of no system-to-system or plant-to-plant variation, which can be seen to be badly violated, they understate uncertainty, so no approximate error factors or statistical confidence limits are given here.

The first source of uncertainty, the shortage of human-performance data specific for nuclear power plants, is the most critical. Historically, such data have not been collected on a scale large enough to establish a data base for operations in nuclear power plants. Presently, some data have been collected from control room simulators [27] and this could be a valuable future source of information, if realistic experiments can be run. Because of the lack of data, most available estimates of human-error probabilities involve extrapolation from other sources of information. These sources include (1) the collective judgment of experts (i.e., people with expertise on the performance of the tasks being evaluated) who may directly or indirectly assess error probabilities, (2) the human-performance models and the associated derived estimates from sources like Reference 12, and (3) data gathered on operationally similar tasks. For example, the actions involved in closing a valve, as specified in a set of procedures, often will be very similar whether the actions are performed in a chemical processing plant or in a nuclear power plant. Such data from similar tasks can be extrapolated or modified to account for dissimilarities in the situations. This extrapolation is subject to error itself, but represents the best approximation available.

A second source of uncertainty is the inherent variability of human performance due to individual differences, both within and between the people whose performances are being assessed. Even if one had a large amount of excellent-quality human-performance data collected for years on all nuclear-power-plants tasks, this variabiilty would contribute to the uncertainty in a human-reliability analysis. A human-reliability analysis does not attempt to estimate the performance of one known person; instead, the analyst's estimates have to account for the fact that any given task may be performed by any one of many individuals, each of whom may vary somewhat in his reliability from day to day or even within a day.

As mentioned above, the lack of data means that subjective estimates may be necessary, with all the uncertainty that entails. The situation is much the same as for subjective estimates and uncertainty assessments for component failure probabilities, but with the additional problem that there are no data available (yet) to calibrate the judgments or modify them via Bayes's Theorem, or otherwise. Furthermore, the intricacies of human actions and the differences in analyst understanding and perception of these actions mean that considerable differences in subjective estimates may result. For example, in a current NRC-sponsored project on psychological scaling (the program plan is Reference 28, but the results are not yet published) nineteen BWR trainers were asked to assess the following event:

> A station blackout including total failure of the diesel generator system has just occurred. After the first immediate steps have been taken, the emergency procedures are referenced. What is the likelihood that the operator will attempt to restore off-site power before he attempts to restore power using the diesel generators?

The nominal values given ranged from 10^{-5} to 0.5. The trainers were also asked to consider the variability of human performance and situational factors and give lower and upper 95th percentiles for the probability of this error. The range from the lowest lower bound to the highest upper bound was 10^{-6} to 1.0. Such uncertainty is not necessarily typical (though this example was selected without first considering the results) but does illustrate what can happen with subjective estimation.

2.2.5 Accident Sequence Estimates

Imprecision in the estimates of initiating event rates, component failure probabilities, and human error probabilities translate into imprecision in estimating the occurrence rates of accidents, or classes of accidents. This imprecision of the "bottom line" estimates is of interest as an analysis summary and as an input to post-core-damage analyses. An example of sequence uncertainty is provided by the Sandia reviews of the Zion and Indian Point studies.[11 15] These reviews included a statistical assessment of the uncertainty associated with various accidents, classified into five damage states. The results of this assessment are shown in Table 2-9. Fairly extensive data went into this assessment, yet it is clear that substantial uncertainty still exists in estimating the frequency of severe accidents.

Table 2-9 Statistical estimates of plant damage state annual frequencies*

Plant Damage State	Zion [11]		Ind. Pt. 2 [15]		Ind. Pt. 3 [15]	
	L95	U95	L95	U95	L95	U95
Early Core Melt With Containment Cooling	2(-5)	2(-3)	2(-8)	3(-4)	0	6(-4)
Early Core Melt Without Containment Cooling	1(-8)	3(-5)	1(-9)	2(-6)	1(-9)	2(-6)
Late Core Melt With Containment Cooling	3(-8)	3(-5)	0	5(-4)	0	5(-4)
Late Core Melt Without Containment Cooling	---- No Statistical assessment ----					
Containment Bypass	0	1(-7)	0	2(-7)	0	2(-7)

* Tabulated values are lower and upper statistical 95% confidence limits on the annual occurrence rates of accidents resulting in the specified damage. Externally-initiated accidents are excluded. An abbreviated notation is used: 2(-5) means 2×10^{-5}/yr.

2.3 Summary

Identifying accident sequences and estimating their frequencies is an important facet of severe accident analysis. The accident sequences identified as important in a risk study define the initial boundary conditions for the core melt, containment, and consequence analyses. The estimated frequencies of these accident sequences directly affect the ultimate estimated risk results (e.g., the frequency of early fatalities). Therefore, uncertainties at this stage of an accident sequence analysis are important.

There are many sources of uncertainty in the identification and quantification of PRA accident sequences. The primary categories of uncertainties described in this section relate to the models used in the analyses, completeness, and the estimation of accident sequence frequencies.

Model uncertainties occur because of limitations in the ability to faithfully represent the real world by analytical models. An example is the binary nature of event trees and fault trees, whereas actual plant responses may involve partial failures or degraded system operation. This uncertainty can cause PRA results to underestimate or overestimate the probability of an accident occurring. Model uncertainties are acknowledged and addressed by efforts to make models as realistic as possible. Conservative assumptions and models (i.e., those which may deliberately lead to an overestimation of risk) are often used in PRAs when more exact information is unavailable.

One of the most important uncertainties in the delineation and quantification of accident sequences deals with the completeness of the models. Completeness uncertainties include:

1. Initiating events: Is the list of initiating events complete and exhaustive?

2. Component failures: Are all of the significant contributors to component failures properly identified?

3. System interactions: Are all physical and environmental interactions between systems properly accounted for?

4. Accident sequences: Are all potentially significant accident sequences identified and properly characterized?

5. Operator actions: Are actions performed by the reactor operators which either mitigate or exacerbate an accident accounted for in the models?

Table 2-10 summarizes the modeling and completeness uncertainties in identifying and modeling accident sequences.

Given that a model for an accident sequence is specified, there can be large uncertainty in the estimates of parameters used by the model.

The sources of parameter estimation uncertainty include (1) the amount of data, (2) the diversity of data sources, and (3) the accuracy of data sources. The use of subjective estimates, in the absence of data, is another substantial source of uncertainty. Data-based uncertainty best yields itself to quantification via statistical techniques. Present results suggest that statistical uncertainty pertaining to the core melt frequency spans three to four orders of magnitude. Table 2-11 summarizes the uncertainties that affect estimating accident frequencies. Tables 2-10 and 2-11 show that uncertainties at the front end of accident sequence analyses essentially carry through to subsequent stages and final risk estimates.

Table 2-10 Summary of accident sequence definition uncertainties

Uncertainty	Implications	Comments
Identification of all important initiating events	Limits accidents to be considered in subsequent analyses.	Sequences contributing to the core melt frequency and/or risk may be missed.
Delineation of all important accident sequences	Limits accidents to be considered in subsequent analyses.	Binary nature of event trees adds uncertainty as does an inadequate knowledge of how systems and operators interact during accidents. Impacts core melt frequency and risk calculations.
System Modeling	Affects choice of accident sequence models.	FSAR criteria are based on accidents which do not lead to core melting and may be overly conservative or nonconservative. Binary nature of fault trees limits definition. Inadequate understanding of system capabilities adds uncertainty. Incomplete understanding of how components might fail adds uncertainty.
Modeling of human actions	Affects the way in which human interactions are considered in accident sequence models.	The wide variety and complexity of human personalities and conditions which affect human performance makes it difficult to accurately model human actions.

Table 2-11 Summary of uncertainties in estimating accident frequencies

Uncertainty	Implications	Comments
Initiating event rates	Affects accident occurrence rates and consequence frequencies	Industry-wide data provide a statistical evaluation of this uncertainty. Largest uncertainties are associated with "rare" initiating events
Component failure probabilities	Affects accident occurrence rates and consequence frequencies	Plant-specific and industry-wide data can be used to evaluate these uncertainties. The use of subjective estimates can add additional uncertainty
Human error probabilities	Affects accident occurrence rates and consequence frequencies	Estimates and assessed uncertainties are now primarily subjective

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3. CORE DAMAGE AND RELOCATION IN THE REACTOR COOLANT SYSTEM (RCS)

The severe light-water reactor (LWR) accidents treated in this section involve the overheating of the reactor core and subsequent progression to severe damage states in the absence of actions that arrest the damage and return the core to a cooled state. Incidental discussion is included which treats actions designed to arrest the damage, but this is not the main focus of the discussion.

The core damage process is important beyond its marking the onset of a severe accident; overheating, accompanied by oxidation of the cladding, can lead to distortion and breach of the fuel, loss of core geometry, production of very high temperature gases (including hydrogen), thermal degradation of structures in the reactor pressure vessel, and the potential for contact between molten materials and water.[1] These, in turn, are important because

- the rates of core heating and the temperatures attained bear importantly upon the mobilities and release of fission products from the fuel (Subsection 7.1),
- the very high temperature gases influence strongly the flow velocities, heat transfer, and turbulence levels that govern fission-product and aerosol transport and retention within the vessel and RCS (Subsection 7.2),
- the high temperatures of the core, its loss of geometry, and the degradation of in-vessel structures yield the possibility of significant in-vessel melt-water interactions (Subsections 3.4 and 3.5),
- the evolved hydrogen can escape the RCS, where its combustion can pressurize and heat the containment (Sections 5 and 6), and
- the melting and downward relocation of core materials can breach the pressure vessel resulting in a discharge of high-temperature melt, fission products, and aerosols into the containment, where they may interact with the atmosphere, water, and/or concrete (Sections 3.4 and 4). The characteristics of these discharges, which are determined by in-vessel processes, are among the most important contributors to uncertainties in containment failure mode and timing (both "early" and "late"), and to uncertainties in the release of radioactivity from the plant (Sections 4, 5, 6, and 7).

Although the accident sequences underlying the treatment given here involve shutdown (scram) of the fission process, so that decay of fission products drives the initial heating of the reactor core, it is recognized that certain sequences have been postulated in which the reactor's shutdown system fails and core heatup is driven by full or partial fission power. The treatment of such transients is complicated by the necessity of considering reactivity effects (i.e., neutronic feedback) in a space- and time-dependent manner. However, many of the phenomena and attendant uncertainties discussed herein also apply, although probably with different time scales, to these transients.

In a large-break loss-of-coolant accident (LOCA), a large amount of coolant is discharged from the break, a large reduction in pressure occurs, and the core is rapidly uncovered. However, the risk from severe accidents in LWRs is considered to be dominated by transient and small-break loss-of-coolant sequences in which the reactor core is uncovered by quasi-steady boiloff or flashing of coolant, heats up, and ultimately melts, with a significantly slower loss of pressure. Because of the need to compress the discussion of in-vessel and RCS processes, only the latter scenario is considered in this section. (Other sequences can differ in timing, rates and extent of heating and oxidation, thermal-hydraulic conditions--including the presence of residual water in the lower plenum--and in other ways. However, many of the processes discussed, and the associated uncertainties, will exist in these sequences.)

From the viewpoint of severe core damage, the processes in pressurized-water reactors (PWRs) and boiling-water reactors (BWRs) should be somewhat similar, although core-wide natural circulation (Subsection 3.2 below) in BWRs during the initial damage phase will be precluded by the bundle shrouds. Also, the additional metal in the bundle shrouds and control blades of the BWR provides increased barriers to radiation heat transfer and additional surfaces for oxidation reactions. These differences are not treated explicitly herein.

This section discusses the known processes that play important roles in severe core damage and the attendant uncertainties that have so far been identified. The discussion is divided into four parts, corresponding to four chronological phases of the core-damage process identified in Table 3-1.

Table 3-1 Phases of core damage during boiloff

Phase	Starting Condition	Approximate Duration (min)	Subsection of This Report
1	Core uncovering begins	10	3.1
2	Hottest fuel attains 1300 K	5	3.2
3	Hottest fuel attains 2000 K	10-25	3.3
4	Fuel discharged to lower plenum	0-80	3.4, 3.5
Ex-Vessel	Vessel breach (Section 4)	--	--

Each phase in Table 3-1 begins with the indicated starting condition and terminates with the starting condition of the following phase. The discussion of severe core damage concludes with breach of the reactor vessel and discharge of core materials into the containment environment. The reasons for the division into the phases and the choices of starting (ending) events in Table 3-1 are given in the discussion of each phase.

The durations of the phases given in Table 3-1 are intended to provide a general orientation; they are necessarily approximate and incorporate ranges of values (Phases 3 and 4), both because the table applies to a range of sequences with different conditions and because of the attendant uncertainties.

One element in the consideration of severe core damage is the potential for the reintroduction of coolant into the damaged area (as occurred at TMI-2). Although traditional treatment of accidents in probabilistic risk assessments (PRAs) usually involves the assumption that core-coolant injection, once lost, is not regained (i.e., that coolant is not injected into a damaged core), injection into a damaged core is likely under certain circumstances (for example, when lost electrical power is restored). If water is reintroduced early enough, the configuration of the fuel rods differs little from the original geometry, and the temperatures of the fuel and cladding are only modestly above operating levels. Cooling of the core under these conditions is reasonably assured. However, reintroduction of coolant at later times creates conditions under which the resultant outcome is significantly uncertain. Uncertainties regarding core behavior during coolant reintroduction are discussed for each phase (other than Phase 1) in Subsections 3.2 through 3.4.

The core-damage process[2] is a progression of coupled events and interactions such that uncertainties in earlier parts of the process tend to be amplified into greater uncertainties in later parts. Thus, uncertainties during the later phases of damage are generally greater than those concerning the early development of damage.

3.1 Phase 1

Phase 1 begins with the start of boiloff of water from the core region and ends when the highest temperature fuel in the core attains 1300 K. This choice is based on 1300 K marking the effective onset of significant oxidation of Zircaloy cladding by steam.

Fuel temperatures at the beginning of Phase 1 are close to the system saturation temperature because heat fluxes to the coolant are small (giving small temperature gradients), and the fuel-temperature elevations corresponding to the expected coolant heat transfer regime (nucleate boiling) are quite small. Damage to fuel in the part of the core covered by coolant is not expected.

During the uncovering of the core, the fraction of the core-decay power that is utilized to vaporize the water is reduced as the water level is lowered. To a first approximation, this leads to an exponential lowering of the level, such that it takes about 1.7 times as long to uncover the second third of the height as it did to uncover the first third of the height. Analytic approximation of the boiloff process is discussed in Reference 3; a detailed treatment of the local heat transfer, two-phase mixture dynamics, and coupling with reactor coolant systems requires use of complex computer models. However, uncertainties regarding the boiloff process are not currently regarded as significant to the core-damage process.

Because of low gas flow rates, the cooling of the fuel in the uncovered part of the core by the flow of steam generated by boiloff is relatively ineffective, and the initial temperature rise in the uncovered fuel is probably well approximated during Phase 1 by adiabatic absorption of the fission-product-decay energy.[3]

During boiloff, fission-product-decay power is the dominant driving force for water loss from the primary system as well as for core uncovering. Uncertainties in predicted behavior and timing can arise from inappropriate assumptions regarding irradiation time, neutron capture, and heavy-element decay. Also, approximately half of the decay energy is in the form of gamma rays and the escape of this form of energy from the core boundaries and its direct deposition in surrounding structures[3] is often ignored in thermal analyses, introducing some uncertainty into the magnitude of predicted temperatures in the fuel rods near the core boundary.

None of the uncertainties in Phase 1 significantly affect the hydrogen evolved during the accident, and except for relatively minor timing differences, little uncertainty is introduced into the fission product source term or containment integrity considerations.

3.2 Phase 2

The start of Phase 2 (Table 3-1) is denoted by the initiation of significant cladding oxidation, which begins at about 1300 K. This phase is particularly important to severe core damage because

- The oxidation of zirconium is highly exothermic (approximately 6.5 MJ/kg Zr reacted),
- The reaction rate increases strongly with cladding temperature, and
- The gaseous reaction product is hydrogen.

Although a considerable amount of data on oxidation-reaction kinetics exists, there remains some controversy as to the validity of more recent data compared to the most-commonly-used Baker and Just formulation.[4] It is generally believed that the reaction is limited by oxygen diffusion through the ZrO_2 film and underlying metal. Mass-transfer effects in the hydrogen-steam boundary layer adjacent to the surface could also influence the access of water molecules to the surface.

Most oxidation experiments have been performed at a limited range of temperatures and in steam-rich environments. Extrapolation of the kinetics to higher temperatures and high hydrogen concentrations may not be valid. The most recent measurements made at higher temperatures include the work of Urbanic and Heidrick in 1978 which extended to 2123 K,[5] and the work of Leistikow and Schanz up to 1873 K at Kernforschungszentrum Karlsruhe (KfK).[6]

Although some of the uncertainty in predicted oxidation behavior is due to differences in kinetics formulations, other aspects of the phenomena yield even greater uncertainties. These include

- Uncertainties regarding the transfer and transport of the reaction energy near the oxidation zone,[7]
- Feedback to the oxidation reaction by increased steam flow resulting from transfer of oxidation energy to the coolant,[3] and
- Changes in interfacial area (steam/cladding) due to cladding deformation, rupture, and relocation.[7]

Heat is removed from the reaction site by transport in hydrogen, and by inward and axial transfer to the metal substrate and then to the fuel. Outward heat transfer is by convective and radiative loss to the flow stream and surroundings.

Gas movement in the core and plenum regions of PWRs above the two-phase mixture is driven almost entirely by natural convection (buoyancy) forces during high-pressure accidents.[8-10] Numerical demonstration of the potential magnitude of these forces is provided by Table 3-2. The table gives Rayleigh numbers (Ra) and convective velocities (V) corresponding to assumed wall and gas temperatures for both steam (St) and hydrogen (H_2), at two pressure levels. The wall temperatures correspond to the saturation temperatures at the given pressures. Two levels of elevated gas temperatures are given. In evaluating the Rayleigh numbers, a characteristic length of 3 m has been used (as a typical height), and the gas properties (except β) have been evaluated at the mean film temperature, $(T_{wall} + T_{gas})/2$. The estimated velocity is based on the theory that the square root of the Grashof number constitutes a special case of the Reynolds number.

Because of the very large Rayleigh numbers (even at fairly modest ΔTs), and the large corresponding velocities relative to the small velocities of steam flow arising from boiloff of the core water (cm/s or smaller), the transport of core energy and the heat and mass transfer (fission products) from the gas to the surroundings will be completely dominated in PWRs by the buoyancy-driven components of the flow field. This transport may be enhanced by direct heating of the gas phase by suspended gaseous and aerosol fission products.

With the exception of the 3-dimensional-fluids version of TRAC and the code discussed in Reference 9, no current severe-accident codes treat the buoyancy-driven large-scale recirculating flows discussed

Table 3-2 PWR natural convection parameters

Pressure	Gas	T_{wall} (K)	T_{gas} (K)	ΔT (K)	Ra	V (m/s)
6.9 MPa (1000 psia)	St	558	842	284	7.4×10^{13}	3.1
	St	558	1631	1073	1.8×10^{13}	4.4
	H_2	558	842	284	1.3×10^{12}	3.1
	H_2	558	1631	1073	6.2×10^{11}	4.4
13.8 MPa (2000 psia)	St	609	791	182	2.5×10^{14}	2.6
	St	609	1580	971	6.8×10^{13}	3.3
	H_2	609	791	182	3.8×10^{12}	2.6
	H_2	609	1580	971	2.3×10^{12}	3.3

above. As a result, most current (1983) predictions of the rates of core heating, extent of oxidation, and core heat transfer are likely to be in error significantly. (For example, the authors of Reference 9 predict significantly larger rates of oxidation.) This error is also expected to lead to considerable uncertainty in predictions of in-vessel retention of fission products and aerosols (see Reference 10).

When the reaction zone attains temperatures above 1600 to 1800 K, the oxidation rate becomes so large that nearly all the available steam is reacted (for typical boiloff sequences).[11] Under these conditions, any mechanism which increases the rate of steam generation can result in an increase in energy generation rate and hydrogen release due to increased oxidation. The injection of coolant into the vessel during this phase of the sequence (in an effort to terminate the damage) could precipitate an acceleration in core damage and hydrogen release due to increased steam generation caused by coolant boiling on newly reflooded core surfaces.[12] If increased oxidation results in even small increases in the fraction of the oxidation energy going into the vaporization of coolant, the feedback will cause a nonlinear increase in the rate of energy generation.[3]

An assessment of conditions necessary to cause the termination of the accident due to coolant injection requires a determination of the rate of cooling afforded by vaporization of the injected coolant relative to the rate of heating resulting from the exothermic oxidation of zirconium.

To a first approximation, the cooling rate Q_B' is given by the surface integral

$$Q_B' = \iint_{S_B} q_B dS \tag{3-1}$$

where q_B is the boiling heat flux over the surface S_B. This will be strongly affected by the boiling regime(s) (nucleate, film, etc.) present over S_B. S_B is, in general, time dependent, due to deformation of the fuel.

The heating rate Q_X' is approximately

$$Q_X' = \mathrm{MIN} \left[\begin{array}{l} {}_{S_X} q_X dS \\ \dfrac{Q_B'}{h_{fg}} R_X \end{array} \right. \tag{3-2}$$

where q_X is the heat flux due to the oxidation reaction over the oxidation surface S_X, h_{fg} is the enthalpy of vaporization for water, and R_X is the heat of reaction referenced to water (16.5 MJ/kg). The rate of hydrogen evolution differs by only a constant from Eq. 3-2. The latter approximation is introduced by assuming, in the term containing Q_B', no delay time for transport of steam from the boiling site to the site of the chemical reaction.

The magnitude of q_X depends upon the local oxide barrier thickness and strongly upon the local surface temperature. The enthalpy of vaporization h_{fg} decreases strongly at high pressure. (It should be noted that the surface being cooled by injected coolant, S_B, is not necessarily identical with the surface undergoing oxidation, S_X, so that questions of net energy increase or decrease must be posed over suitable volumes of the reactor core.)

Instantaneous termination of the accident is favored when $Q_B' >> Q_X'$. Integration over time, with consideration for the contribution due to decay heating and various loss terms considered here to be small relative to Q_B', is necessary to determine the actual outcome.

Examination of Eq. 3-2 shows that, if all the steam generated by the boiling is reacted,

$$\iint_{S_X} q_X dS = \frac{R_X}{h_{fg}} \iint_{S_B} q_B dS, \tag{3-3}$$

where R_X/h_{fg} varies between 7.3 at 0.1 MPa and about 19 at 17.0 MPa. Eq. 3-3 thus illustrates the difficulty of terminating the accident in the presence of substantial oxidation, and the reason why injection into an overheated core must be regarded as producing significantly uncertain cooling (or heatup) and hydrogen evolution.

Finally, observation of zirconium-burning tests shows clouds of smoke issuing from the test chamber, indicating that large quantities of aerosols may be generated during the oxidation accompanying severe core damage.[13] Aerosol processes are discussed in Sections 5 and 7.

Although cladding melting is excluded during this phase (by limiting temperatures to 2000 K or less), several types of cladding deformation and rupture can occur, including ballooning, oxidation-induced deformation,[13] and brittle fracture. The cladding is simultaneously subjected to thermal transients and may be subjected to stresses resulting from the increasing internal pressure (relative to the external environment) of the initial fill-gases and fission gases in loss-of-coolant accidents (LOCAs). Prior to cladding rupture, ballooning of the cladding is expected at low primary-system pressures.[14] The temperature and pressure at which ballooned Zircaloy-4 cladding bursts in a steam environment has been studied, and it has been found that, even at low (initial) internal pressures, cladding usually bursts at temperatures below 1473 K (1200°C).[14] Uncertainties in cladding-rupture conditions and timing do not strongly influence fuel-damage behavior, but may be important for the release of fission products (see Reference 7).

Embrittlement and spallation of ZrO_2 from the surface of the cladding as oxidation proceeds may result in weakening of the fuel rods, may alter the kinetics by exposing fresh surfaces to steam, and/or may result in the formation of rubble beds with the potential for blockage of flow passages. These mechanisms, which have been extensively investigated, can cause increases in the cladding-surface area exposed to steam and can thus increase the oxidation rate during conditions in which the reaction is not steam-starved. However, because of the probable prevalence of steam-starved conditions during this phase of the sequence, uncertainty regarding the geometrical condition of the cladding is probably of second order in importance to damage progression during Phase 2.

Figure 3-1 illustrates a calculation of the thermal behavior of fuel during this phase.[11] The calculation is one-dimensional, and does not account for the buoyancy-driven flow discussed above. It is included here to illustrate the tendency of the oxidation process to dominate the temporal and spatial behavior of fuel temperature. The calculated behavior is characterized by smooth temperature profiles, which resemble the axial power profile, until the onset of significant Zircaloy oxidation which occurs at the location of the highest axial temperature, approximately 2 m above the bottom of the core. As oxidation power continues to develop at this location, a sharper temperature profile develops that is characteristic of a distinct oxidation front. On the steep upstream side of the front (toward the base of the core), the oxidation increases rapidly with distance upward; on the more gentle downstream slope, the oxidation is reduced by depletion of steam (the peak is less pronounced if substantial hydrogen blanketing is assumed in the models).

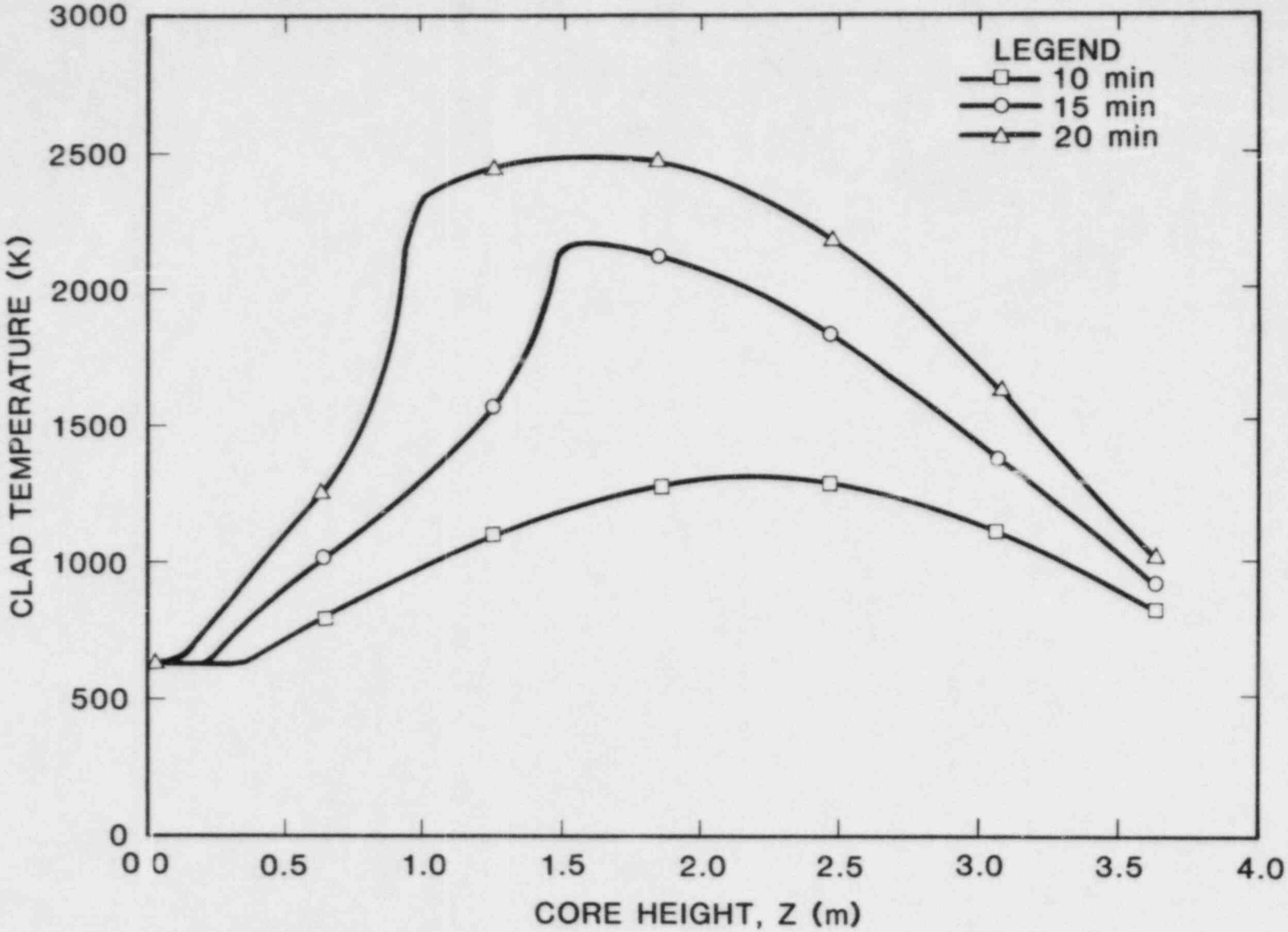

Figure 3-1 Calculated axial cladding temperatures at three different times following start of core uncovering for a TMLB' sequence in a PWR [11]

As the elapsed time from the start of core uncovering increases, not only do the size and temperature of the high-temperature zone increase, but the position of the oxidation front moves upstream (downward), and the temperature gradient at the leading edge increases. It is this characteristic temperature profile that provides conditions from which physical damage and relocation of the fuel-rod materials develop.

System pressure and decay-power level have a weak phenomenological influence in that they basically affect only the timing of the sequence. However, the development of the oxidation front depends strongly upon the rapid increase in the fuel-rod and steam temperature with distance above the water (mixture) interface. The rate of increase of the steam temperature is augmented by the positive slope of the decay-power profile in the lower half of the fuel rod. Downstream, near the top of the core, reduced power and reduced oxidation result in relatively slower temperature increases. This lower temperature portion of the core cools the flowing, hydrogen-dominated coolant to relatively modest temperatures as it exits the core at the top. It should be emphasized, however, that the dominating influence of buoyancy-driven flow discussed above is not considered in this calculation. Consideration of this flow would alter both the temporal and spatial aspects of the temperature behavior, although the influence of oxidation would be expected to remain strong.

The relatively short duration shown in Table 3-1 for Phase 2 is based on calculations which indicate average temperature rise rates in excess of 2 K/s in regions undergoing vigorous oxidation.[11] Uncertainty in calculations of this kind is introduced because multidimensional analysis might show cross-flow of steam from the core periphery towards the (hotter) central region. Thus steam-starved conditions (as predicted by one-dimensional treatments) may be incorrect, and damage may occur more rapidly than currently predicted. The short interval between the inception of significant oxidation and the threshold for melting cladding--in the highest temperature zone of the core--suggests the acceleration in damage which is caused by the oxidation reaction. However, because of (1) the variation in local decay-power levels, (2) the time-dependence of the core-uncovering process, and (3) the effects of heat transfer and convective transport within the core and to the core periphery, oxidation-damage processes in different parts of the core advance over a much broader time scale than is indicated by the nominal 5-minute interval associated with the peak damage location. This aspect of core behavior is discussed further in Subsection 3.3.

Because low-melting-point silver-indium-cadmium alloys are often employed as neutron-flux-control (absorber) materials in LWRs, the possibility exists for formation of significant molten quantities of these materials at the temperatures attained during Phase 2. Whether or not such melts would be retained by their cladding under these conditions is not known. It is thus uncertain when, and how coherently, the many tons of these liquids might pour into the core, perhaps forming aerosols before contacting water or structures below the core.

If water is reintroduced to the core zone (reflooding) during Phase 2, it is possible that the core-damage processes may be initially accelerated (and hydrogen generation increased) by cladding oxidation resulting from the additional steam generated during the cooling of overheated fuel. Considerable fracturing (during reflood) of cladding embrittled during oxidation is expected, as well as formation of fairly coarse rubble (fractured cladding, fuel, and control materials) within the central region of the core (as at TMI-2). Based on the TMI-2 experience, it is possible that the rubble beds formed can be maintained in a cooled condition, terminating the accident during this phase. However, the question of establishing long-term cooling of a reflooded core which has undergone severe damage must still be considered as highly uncertain (see Section 4). Additional aspects of rubble-bed cooling are discussed in the following sections.

3.3 Phase 3

Phase 3 begins with Zircaloy melting at the peak damage location as it attains a nominal 2000 K (Zircaloy-4 melts at 1993 K), and extends to the time that core material issues into the lower plenum of the reactor vessel--the interval spanning maximum fuel and core damage. Especially during Phase 3, there is a strong coupling (more forward than reverse) between fuel damage processes and the associated processes governing the release, chemistry, and transport of fission products (Section 7). (The reverse coupling is weak because fuel damage during this phase is driven approximately equally by oxidation power and by decay power.) The reader should note the strong forward coupling and observe the concomitant coupling of uncertainties detailed below and in Section 7.

When the local temperature of the fuel reaches the Zircaloy-melting temperature, flow of metallic cladding beneath the oxidized layer will occur, which should provide significant contact between the melted metal and the UO_2 fuel. Recent experiments have provided dramatic evidence of the interaction that can subsequently occur between molten Zircaloy-4 and solid UO_2. In one series of laboratory experiments, UO_2 crucibles holding molten Zircaloy at temperatures between 2073 K and 2273 K (in argon atmospheres) were rapidly destroyed by the dissolution of solid UO_2 in molten Zircaloy.[15] In another laboratory experiment, electrically-heated fuel-rod simulants in steam were massively liquefied and relocated when the oxidation-driven 9-rod-bundle temperature approached (a measured) 2300 K.[16] The rapid kinetics observed in these experiments imply that the reactor core may be quite vulnerable to destruction at these temperatures.

This process, in which Zircaloy reduces UO_2 to form a homogeneous (U, Zr, O) melt at low oxygen concentrations or a heterogeneous (U, Zr, O) melt containing UO_2 particles at high oxygen concentrations,[15] provides a powerful mechanism promoting the destruction of fuel-rod geometry at temperatures slightly above the Zircaloy-melting temperature but far below the melting point of UO_2. This process is referred to herein as fuel "liquefaction".

Apparently, the rapid disintegration of the solid UO_2 is due initially to the formation of liquid uranium preferentially along UO_2 grain boundaries near the UO_2-Zircaloy interface, causing a loss of cohesion. The uranium results from reduction of the UO_2 by the Zircaloy. In addition to destroying the UO_2 matrix, these processes will apparently also accelerate the release of fission products from the fuel (Section 7).[15
It is thus presumed that significant liquefaction of fuel can occur at local temperatures between 2000 and 2300 K, resulting in downward flow of the liquid. Because of the much lower temperatures existing at lower levels in the core (Figure 3-1), freezing of the liquid will tend to be promoted. However, a countering tendency is provided by the possibility of accelerated oxidation as high-temperature, liquefied metal flows downward into a steam-rich region. The balance between these countering tendencies will determine the extent to which material solidifies above the bottom of the core and, by this means, the tendency to develop a large, coherent melt zone. The size of such a melt zone has implications for melt-water interactions (Subsections 3.4 and 3.5).

Calculations indicate that, without additional oxidation, a considerable margin exists to produce rapid freezing of the liquefied fuel--and thus significant core blockage--even if freezing requires the transfer of the full UO_2 latent heat of fusion (270 kJ/kg).* But energy added by oxidation could reverse this conclusion, and the temperature range of the liquefied material (2000 to 2300 K) favors high specific (per unit surface area) oxidation rates. The principal uncertainties in the process are the magnitude and time-dependence of the exposed surface area and the oxidation kinetics above 2100 K.

The power distribution within the core will (very approximately) determine the decay-heat-driven rise in temperature from the initial conditions to temperatures that permit rapid oxidation. The power distribution can therefore be used to provide a rough idea of the degree of coherency in core degradation.

Figure 3-2 and the associated Table 3-3 show the power distributions in the TMI-2 core prior to the 1979 accident.[17 18] These show that approximately 50% of the core, located within about 80% of the full radius and to within about 0.5 m of the top and bottom, has decay-power levels within 20% of a nominal value. The remaining 50% of the core is at significantly lower decay-power levels. This suggests that the initial core degradation and perhaps slumping might be limited to the central 50% of the core. Some of the outermost fuel rods may not

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* Based on the description of the liquefaction process given at the beginning of this section, use of this value of latent heat is inappropriate. A more nearly appropriate value might be the Zircaloy latent heat of fusion smeared over the entire Zr-U-O mass (about 50 kJ/kg). This would provide an even greater margin favoring freezing.

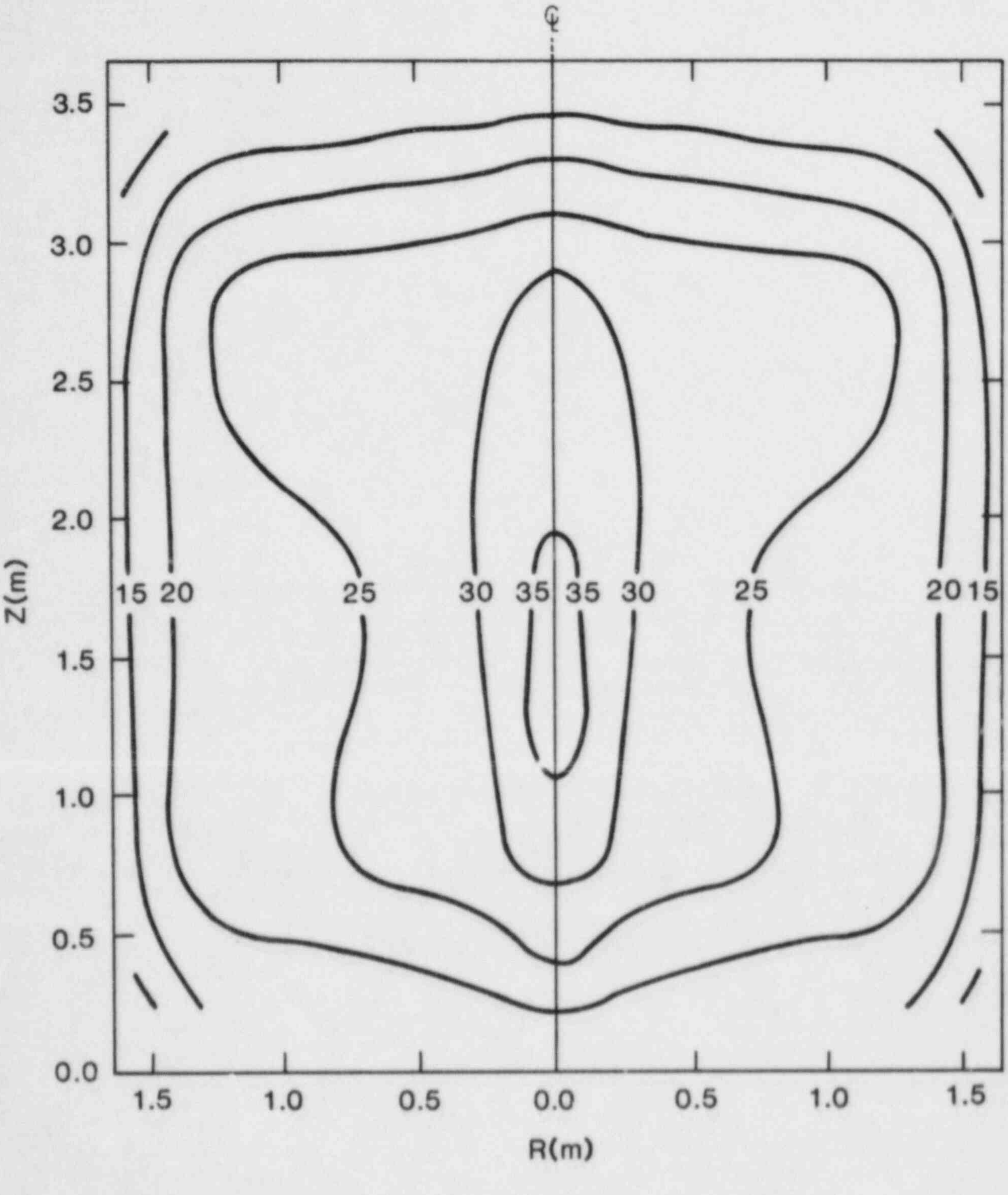

Figure 3-2 Distribution of fuel rod rating in kW/m in the TMI-2 core based on values in References 17 and 18.

Table 3-3 Axial and radial variations of fuel rod rating in TMI-2 core (in kW/m) for 98% full power (2720 MWt) based on values from References 17 and 18

Z (m)	Radial region 1	2	3	4	5	6	7	8
3.40	17.6	15.7	15.7	14.7	13.4	13.9	11.5	7.2
2.88	30.0	27.4	27.0	26.7	25.5	25.6	20.9	13.9
2.35	33.6	30.2	29.3	27.7	25.2	26.1	22.1	15.4
1.83	35.2	30.3	29.1	25.9	21.6	23.2	21.2	15.0
1.31	36.1	30.0	28.5	26.0	22.0	23.6	21.4	15.0
0.78	33.7	28.0	26.7	25.8	23.8	24.8	20.9	14.3
0.26	22.0	18.6	18.0	17.3	16.7	17.0	13.9	9.2

attain temperatures representative of severe damage during the times corresponding to the phases used in this section because of their low power levels and their location adjacent to surrounding structures, but this is somewhat uncertain. The degree of coherency in core damage affects both the course of the accident--for example by its influence on the magnitude of a possible steam explosion (Subsection 3.5)--and the rate of release of fission products and aerosols from the core (Subsection 7.1).

Even in accidents, like a TMLB' in a Westinghouse PWR, in which secondary-side inventory is depleted at the time of core damage, gaseous natural convection between the vessel and the upflow-half of the primary side of the steam generators is favored. Because of potential loop seal and downcomer blockage, this convection is most likely required to traverse the hot leg piping, and consists of displacement of cooler steam/hydrogen in the generator tubes (cooled by contact with the large mass of steel) by warmer steam/hydrogen migrating into the hot legs from the reactor vessel (and pressure-driven into the pressurizer leg during PORV operation). The great height of the steam generator tubes (18 m) provides a large driving force: The small diameter of the tubes (22 mm) hinders the flow due to high frictional resistance and direct interference caused by countercurrent velocities. The lack of secondary-side inventory means that the tubes will steadily rise in temperature (thus reducing the convective driving force) as the convection proceeds.

To the extent that the convection is effective, it will provide a sink for fission products. Based on a 3260-tube generator with 18 m of upflow, and 22-mm-tube diameter with 1-mm wall, the generators in the Surry plant, for example, contain about 1.3 x 10^5 kg of "effective" steel. Assuming that the total Cs + Te + I core inventory is uniformly deposited throughout the tubes, the equivalent heat flux is 0.5 kW/m^2, and the lumped long-term rate of temperature rise is about 0.1 K/s.

The above calculation overestimates the heating effect because heat will be lost by thermal radiation and convective heat transfer to cooler components, and by gamma (primarily) and beta emissions not captured within the tube walls.

The effectiveness of this heat sink decreases strongly as the tubes heat up (approximately as the 4/3 power of the temperature difference so that halving the ΔT reduces the convective heat flux to less than 40% of its original value). The process is little aided by condensation of steam, for two reasons. First, the tubes will be initially only slightly subcooled (~50 K), and this will soon vanish. Second, condensation will increase the local partial pressure of the noncondensable component (hydrogen), resulting in drastically reduced condensation rates.

Thus, heating rates from deposited species can potentially become significant with time by reducing natural convection, reevaporating deposited species and inhibiting further deposition, and failing structures. Many of the reactor coolant system structures will fail well before they melt (M.P. ~1400°C); effective natural convection may be inhibited when the structure attains a temperature of approximately 1000°C.

The equilibrium solubility of UO_2 in oxygen-saturated α-Zr (at about 2300 K) is uncertain, but lies between 8 and 20 mole percent, according to Hofmann et al.[15] At lower oxygen concentrations, considerably more UO_2 is dissolved. At roughly 2700 K and high oxygen concentrations, an equilibrium 85 mole percent UO_2 is dissolved. (Equilibrium conditions are not expected, in any case.) Also, the extent of destruction of the UO_2 matrix may be only qualitatively related to the fraction dissolved in the melting zirconium. Thus, the degradation from rod geometry to a relocated, partially liquefied mass as temperatures increase involves complex chemical and physical processes. Material motion and resulting configuration(s) cannot be predicted at present, constituting a major area of uncertainty in assessing LWR core-damage progression.

If water is reintroduced into the lower core during Phase 3, acceleration of core damage may occur, because

- A large fraction of the core has achieved elevated temperatures,
- The quantity of cladding oxidized is relatively modest because of low steam evolution resulting from boiloff prior to reintroduction of water,
- The relatively intact core geometry (assuming only modest degradation at this time) provides uninhibited access of steam to most of the cladding, and

* Reflooding of hot fuel in the lower part of the core will produce copious amounts of additional steam.

For example, if 1/10 of the core is suddenly covered by water at atmospheric pressure, the assumption of a critical heat flux of 3 MW/m^2 gives an instantaneous heat transfer rate of 1.5 GW. The resultant steaming rate, if totally reduced by the oxidation reaction, gives a peak power addition of about 10 GW (3 times nominal <u>operating</u> power for a roughly 3000 MWt plant).

Acceleration of oxidation associated with reintroduced coolant might, given these assumptions, add tens of gigajoules of energy to the system in a short time and evolve large quantities of hydrogen, because of the rapid oxidation kinetics at temperatures of 2000 K and above, and the modest energy used to increase the coolant temperature and vaporize it (see description in Subsection 3.2). Because the energy required to destroy the entire core geometry at these temperatures may be as little as 6 GJ, it is possible to postulate massive destruction of the core in a very short time following the reintroduction of water. An attendant possibility is one or more steam explosions caused when hot, liquefied fuel falls into the pool of reflooding water. (Steam explosions are discussed below in Subsections 3.4 and 3.5.) The actual scenario is quite uncertain, producing significant uncertainty in all subsequent events and processes that are affected.

If much of the steam generated is not reacted, the reintroduction of sufficient water should halt the heatup and result in a cooling of the core. This requires, in addition to quench, either reestablished loop flow (forced or natural convection in the primary system) or local bed convection. Cooling by local convection in the bed, as well as by reestablished loop flow, depends upon the size and characteristics of the rubble and the coolant-volume fraction,[19] and requires that a long-term heat sink be available for the energy removed from the bed. Little is currently known about the rubble configuration that may be obtained following reintroduction of water, although experiment programs to produce this data have been established.[20]

Assuming water is not added and that the previously given description of space-dependent, initial core liquefaction holds (there is significant uncertainty in this), a region of slumped fuel is envisioned to initiate in the radial and axial center of the core and expand radially to about 80% of the core radius. This slumped fuel could possibly form a temporary, fairly dense blockage above a height of 0.5 to 0.8 m in the core (because of the relatively low temperature of this base part of the core), but it might also begin to stream through this region into the lower plenum if not halted by freezing. The question of freezing versus penetration is quite uncertain and is important because it affects the magnitude of resulting melt-water interactions (Subsections 3.4 and 3.5).

Heating the central half of the core to a temperature at which it may initially liquefy requires energy equivalent to oxidation of about 15% of the core zirconium; liquefaction may require an additional 1 to 5%, depending on "latent heat" assumptions. From this it might be assumed that roughly 20% of the core zirconium would be oxidized at the time of significant core degradation. The presence of significant natural circulation in a PWR makes this assumption quite uncertain, however. At the conclusion of the postulated initial slumping and freezing stage, the rate of oxidation of the central 50% of the core would probably decrease significantly because of the density of blockage and reduced interfacial area for reaction.

A central (incomplete) blockage would redirect steam flow outward in the lower part of the core of an open lattice (PWR) core, producing two possible alternatives; if the fuel rods have not yet attained temperatures capable of supporting rapid oxidation, they may be cooled by the additional flow, but if the rods are hot enough, they may rapidly oxidize. Meanwhile, slumped and frozen fuel would begin to reliquefy as it reheated, primarily by decay heat (reduced by fission products lost during initial liquefaction). Because it appears unlikely that further downward flow of this fuel would be halted (refrozen) by the reduced thermal mass of cool fuel and structure below it, breach of core-support structures and discharge of melt into the lower plenum would occur shortly thereafter. Note, however, that much of this scenario is quite uncertain.

If a massive breach in the core supports should occur, 50% or so of the core could suddenly plunge into the plenum water. For the streaming scenario on the other hand, Bracht has calculated a maximum liquid-flow rate outward from a single fuel-assembly opening of 940 kg/s with a mean value of 470 kg/s, implying that several minutes would be required for the discharge of 50% of the core through the opening.[21] A streaming discharge simultaneously through several openings would reduce the duration accordingly. Which mode might occur is clearly uncertain.

The descriptions given of Phase 3 have been based on analyses of the early stages of core damage, current estimates of severe-damage phenomena, and significant extrapolation of the current data base. Several of the more significant uncertainties are summarized in Table 3-4, together with the implications of these uncertainties that are currently foreseen.

3.4 Phase 4

Phase 4 begins with the discharge of core materials into the lower plenum. The progression treated here assumes that water exists in the plenum, although this will not be true for some sequences. Based upon the Phase 3 description, the melt might issue from the core region in three possible modes:

(1) In a narrow continuous stream over a period of fractions of minutes to several minutes,

(2) In a narrow discontinuous stream, or streams, distributed over a longer period of time, and

Table 3-4 Phase 2 and 3 uncertainties

Uncertainty	Implications	Comments
Natural/forced convection in reactor coolant system	Pressure/heat sink behavior: transport/deposition/resuspension of fission products	Design- and sequence-detail-dependent: e.g., location of break in LOCA, PWR vs BWR
Zr oxidation kinetics at T > 2000 K	Influences core heating rate, fission product release	No data above 1800°C: extrapolation of kinetics produces very high rates at T >2000 K
Liquefaction temperature	Temperature of liquid, energy level for steam explosion, timing	Might make 80-GJ difference in core energy and 45-min time difference
Natural convection in core/plenum volume (primarily in PWRs)	Energy and aerosol distributions, heat and mass (fission products) transfer	Greater energy at core slump, oxidation may go more nearly to completion (50+ %), fission products and aerosols may distribute differently
Effect of coolant injection on core when significantly above Zr oxidation threshold	Possibility of accident termination by establishment of cooled rubble bed: damage acceleration could induce large-scale rapid core liquefaction: energy level for steam explosion	Possible accelerated meltdown vs benign termination: enhanced H_2 evolution might correspond to near-completion of Zr oxidation, but is highly uncertain
Oxidation and conduction freezing of (U,Zr,O) liquid as it flows into steam-rich zone	Addition of energy could more than offset conduction freezing leading to continued flow (little freezing) Affects potential for blockage and hence scale of possible steam explosion(s)	Streaming vs freezing (core blockage) behavior uncertain: thermal conductivity, viscosity, and Prandtl number of liquid unknown: affects interactions in lower plenum
Heating of surfaces by deposited fission products	Reduction of natural convection, inhibition of deposition, revaporization of fission products, thermal heatup to failure of structure	Susceptible surfaces include core barrel and steam generator tubes in PWR, upper internal structure in BWR
Extent of UO_2 "dissolution" in low temperature (U,Zr,O) liquid	UO_2 not dissolved could absorb ~0.5 MJ/kg in additional sensible heat plus ~0.3 MJ/kg in latent heat of fusion at melting point. Energy levels and timing affected	If much of core remains structurally stable until very high (> 2600 K) temperatures are attained, melt/water interactions in lower plenum could be more energetic
Failure (breach) of core support structures	Scale of breach and timing relative to core damage affects rate and amount of energy added to water in lower plenum (steam explosion influence)	Local vs. global failure mode uncertain

(3) In a relatively massive, coherent discharge occupying a few seconds or less (this is probably less likely in a BWR because of the method of core support).

The temporal distribution of the discharge is related to the three modes listed and to the level of damage achieved (fraction of core liquefied). This is true because the rate of formation of liquefied fuel is slow compared to all but the very slowest discharge rates. Thus, if a large fraction of the core is liquefied at the onset of discharge, a larger amount might be discharged; conversely, if only a small fraction is liquefied at the onset of discharge, a smaller amount might be discharged (corresponding to Mode 1 or 2 above).

Experiments have shown that, under certain conditions, high temperature melts explode upon water contact.[22] Such a "steam explosion" results from transfer of thermal energy from the molten mass to water on a time scale so short (approximately 1 ms) as to produce effects associated with chemical explosions. Industrial experience with such explosions has shown them to be often powerfully destructive.[23]

The four major stages of such an explosion have been identified as

(1) Initial coarse pre-mixing without large heat transfer, generally implying stable film boiling,

(2) Destabilization of film boiling, either spontaneously or from an external pressure pulse (triggering), leading to small-scale mixing and rapid heat transfer in a local region,

(3) Propagation of a zone of rapid heat transfer through the coarse mixture, which may develop into a propagating detonation, and

(4) Explosive expansion driven by steam at high pressure.

Although based on observation of interactions within a range of material pairs, such evidence as is available for reactor materials, including simulants of LWR materials, is consistent with this description.[21 24]

The Gittus committee assessed the present state of knowledge concerning in-vessel melt-water interactions, and proffered four major scenarios:[25]

(1) A series of relatively low-yield steam explosions continuing until the whole of the molten mass of fuel has been fragmented or all of the water evaporated,

(2) A low-yield steam explosion which stimulates a large steam explosion involving a significant fraction of the melt,

(3) A single, large steam explosion, and

(4) No steam explosion but violent boiling, which may or may not quench the debris, depending on the quantity of water available and the agglomeration of the debris.

Because of the resultant disruption (and possible dispersal) of internal structures and residual core materials, the occurrence of even a relatively low-yield steam explosion will qualitatively alter the subsequent progression of damage.

Scenario 2 or 3 (above) could possibly cause

(a) breach of the reactor vessel, or

(b) breach of the reactor vessel and generation of containment-failing missiles.[25]

Either (a) or (b) would completely alter the course of the accident, particularly the second which would permit ejection of fuel and fission products from the reactor vessel and the nearly simultaneous venting of the containment. The uncertainties in assessing these possibilities are discussed in Subsection 3.5 below. These two possibilities, (a) and (b), account for the minimum duration for Phase 4 given in Table 3-1. If no steam explosion occurs, Scenario 4 (above) is assured.

In the event that the vessel is not breached by a steam explosion, a fraction of the core melt may be quenched. For core fractions equaling or exceeding the values in Table 3-5 (or smaller fractions for less water), the quenching will vaporize all of the water in the plenum.

Table 3-5 Fractions of core mixture* which can be quenched in below-core water**

	Atmos	5.5 MPa	11 MPa	17 MPa
ΔT = 1500 K NO FREEZE	0.79	0.44	0.31	0.17
ΔT = 2000 K NO FREEZE	0.59	0.33	0.23	0.13
ΔT = 2500 K FREEZE	0.37	0.21	0.14	0.08

*10^5 kg UO_2 + $2(10^4)$ kg Zr + 10^4 kg Steel

**in 29 m^3 of Water

If excess melt over that which can be quenched is deposited in the plenum, it will begin heating the reactor vessel wall immediately. The fraction quenched by the vaporization of the residual water will subsequently begin reheating, but will require 20 to 40 minutes to attain temperatures that augment the attack on the pressure vessel.

The table indicates the limited capacity for the formation of quenched debris in the lower plenum. (The capacity is further reduced if the inventory of residual water is reduced below 29 m^3.) The implication for debris cooling and vessel attack is discussed below.

To this description must be added the contribution from the accompanying chemical reaction. As noted above in Subsection 3.3, quantities of unoxidized zirconium are likely to be involved in the core-liquefaction processes. Mixing of this metallic phase at high temperatures with the water (and steam) in the lower plenum will promote rapid oxidation of the zirconium, depending primarily upon the degree to which fragmentation of the melt provides large increases in the interfacial area for interaction. In the case of a steam explosion, the resulting fine fragmentation in the presence of a steam environment should promote oxidation and there is some experimental evidence of this occurring.[26] Regardless of the exact outcome, the addition of reaction energy and liberation of a quantity of hydrogen by the oxidation of zirconium during the melt-water interaction phase seems likely.

If the quantities of core melt deposited into the lower plenum are quenched (Table 3-5), it is possible that the resulting bed of core rubble might be cooled over the long term, resulting in the termination of the accident sequence without vessel breach, release of additional fission products from the reactor vessel, and subsequent additional damage. In addition to a coolable bed configuration, this requires a supply of water to keep the rubble submerged and a transport path and heat sink that can remove and absorb the energy from the system on a continuing basis. Conditions leading to a coolable configuration would appear to be restrictive because of the apparent likelihood of the occurrence of a deep bed, possibly with stratified particle sizes, and diameters and coolant fraction small enough to produce dryout in the bed even after it is initially quenched.[27] Lack of a continuous source of appropriate amounts of coolant may also restrict the attainment of a coolable state. If dryout occurs, the rubble will remelt in 20 to 40 minutes, as indicated above.

The vessel failure (breach) mode may depend on the excess of internal pressure above containment pressure.[2] Two modes of failure identified in Reference 2 were

(1) At low relative pressure, combined melting and high temperature weakening accompanied by large deformation, leading to a breach in the bottom of the lower head, and

(2) At high relative pressure, meridional cracking with little melting or deformation.

The high-pressure failure could develop below or above the bulk of the debris and is caused by a combination of thermal stresses, pressure stresses, and material weakening (at temperature). The time-to-failure identified for Mode 1 was 30 minutes; that for Mode 2 was "short." As noted above, removal of the lower head by a steam explosion is possible; removal would be prompt.

Estimates for time-to-failure identified in Reference 28 vary typically from 22 minutes to 40 minutes.

To these should be added the recently suggested mechanism of melt failure of instrument-penetration welds in the Zion reactor vessel.[29] The time-to-failure identified for this mode is typically 5 to 7 minutes, independent of relative pressure. These diverse modes and durations illustrate the broad range of vessel-breach concepts and the lack of agreement currently characterizing this aspect of accident analysis. This broad range of potential vessel-breach modes provides a wide range of subsequent behavior, as is discussed in Section 4.

The 80 minutes given in Table 3-1 results from combining the maximum estimated time-to-breach for the reactor vessel (40 minutes) with a scenario in which the core material deposited in the lower plenum is initially quenched (without a vessel-failing steam explosion), and must subsequently reheat to produce vessel failure.

One of the most significant aspects of the uncertainty in the vessel-failure mode is the effect upon the subsequent blowdown during pressurized sequences. Large blowdown flow rates and pressure drops may strongly influence the resuspension and subsequent redistribution of fission products. This is discussed in Section 7.

Table 3-6 summarizes the major uncertainties occurring during Phase 4.

3.5 In-Vessel Steam Explosions

This subsection discusses the uncertainties in the predictions of the effects of steam explosions that might occur if liquid fuel pours into residual water in the bottom of the reactor pressure vessel (RPV). It considers how energetic such explosions might be, whether or not they can be triggered, and uncertainties in the magnitude of explosions required to fail the RPV or the containment.

Both liquid water and liquid fuel are needed for a steam explosion. For some large LOCA sequences, all the water in the RPV may be rapidly boiled off or blown out; so our ability to predict whether there will be water in the lower plenum depends on uncertainties in dynamic blowdown calculations.

The capacity of a steam explosion to do damage depends on, among other things, the kinetic energy developed during the expansion phase. Limits that can be placed on this energy are therefore considered here. The maximum kinetic energy is the product of the mass of molten material mixed with water at the time of the explosion, its heat content per unit mass above the water temperature, and the efficiency of conversion of this heat into work (the conversion ratio).

To participate in an explosion, melt has to pour from the core region and then mix with water. A relatively large pour is required if the RPV is to be threatened. (Based on high values of heat content, 1.6 MJ/kg, and conversion ratio, 16%, discussed below, at least 4000 kg of melt would be required.) Thus uncertainty in the discharge

Table 3-6 Phase 4 uncertainties

Uncertainty	Implication	Comments
Occurrence of steam explosion(s) and yield	Possible failure (breach) of containment and/or reactor vessel accompanied by dispersal of finely fragmented core materials	See Table 3-7
Extent and rate of oxidation during melt-water interacton	Amount and rate of hydrogen liberated; amount of energy added to core melt	Oxidation of Zr may or may not approach completion; significant effects on some containments are possible if hydrogen ignites--see Sections 5 and 6; energy addition influences vessel breach and steam explosion (if any) yields
Formation of coolable rubble bed in lower plenum	Possible termination of accident sequence without vessel breach	Requires fluid energy transport path and heat sink
Mode and timing of (nonexplosive) vessel breach	Variation in time-to-breach (0 to 80 min); type of discharge of core materials; possible melt-water interactions including steam explosions (accumulator discharge, cavity water); aerosol generation and fission-product redistribution on blowdown	Mode of vessel breach affects rate of melt ejection, type of cavity interaction, and blowdown behavior in pressurized sequences--see Section 4 for effects. Latter item influences possible resuspension of fission products and redistribution into containment (Section 7)

mode affects the possibility of vessel failure. In addition to the mechanisms described above, a large discharge might be stimulated by an initial small steam explosion disrupting the lower core plate and refrozen crust holding up a large pool of liquefied core material.

Mixing between melt and water will be promoted by gravity causing the melt to flow through the water; by hydrodynamic breakup processes such as the Rayleigh-Taylor instability; by unordered velocities near the interface between the two liquids (which will be separated by steam in film boiling); and by early steam explosions if there is more than one. It will be hindered by gravitational settling-out and the disruptive effect of steam upon the mixture. It has been argued that the latter effect can now be used to place an upper limit, ranging from about 100 kg to about 2000 kg, on the mass of melt mixed.[30-33] However, current quantitative formulations of this argument are one-dimensional and time-independent, whereas the experimentally observed mixing process is manifestly two- or three-dimensional and time-dependent.[24 34] There is some experimental evidence of substantial underprediction of the extent of mixing that can occur.[35 36] In particular, transient mixtures lasting for a second or so, which would be unstable in a steady state, may be possible. It is reasonable to conclude that while mixing limitations caused by steam production may be established by further research, current uncertainties mean that no such limit can now be imposed.[25]

The next requirement for a steam explosion is the occurrence of a trigger. This term encompasses both the spontaneous initiation of a detonation at some place in a premixture and external initiation by, for example, a hammer blow or chemical explosion. Detailed processes involved in triggering are not well understood; it is possible that a range of processes may compete, with different ones dominant in different circumstances.

Experiments show that at low ambient pressures--up to, say, 0.5 MPa--spontaneous triggering of corium simulant or iron-alumina thermite in water occurs either within the body of the mixture at an apparently random time, or at the contact site within the first 30 ms of contact between hot melt and the steel container.[24] This effect, therefore, puts a geometrical limit on the amount of melt that can participate in an explosion in a reactor accident when melt pours into water, because an explosion will occur at, or before, or only shortly after, contact between melt and any large steel components. This restriction will not apply with such strength to any explosions after the first, since at that point it is not certain that the original arrangement of structures will be maintained, and the mixing time scale may be comparable with the 30-ms maximum delay time instead of being on the order of 1 s as in the case of a large pour.

At higher ambient pressures, spontaneous triggering does not occur (or only rarely, if at all), and an external trigger is required. Until recently, all the (few) data were consistent with a steady increase in the threshold trigger strength required with ambient pressure up to 4.0 MPa, the highest pressure attained in controlled experiments.[24 37 38] However, this monotonic relationship is called into question by recent experiments at Sandia National Laboratories, in which the required trigger for small drops of iron oxide increased, decreased, and increased again with increasing pressure.[39 40] Complete uncertainty exists in predictions of the triggers required at the pressures of interest in reactor accidents (up to 17 MPa), because neither the characteristic of a trigger pulse that determines its strength in this context, nor the processes that control initiation of a detonation are known. A complementary difficulty is that it is not known what external triggers will be available, or with what frequency, in an accident. Although the energy in some laboratory chemical explosive triggers (roughly 3000 J)[24 38] is towards the upper end of the range of credible triggers from falling objects in an accident, the magnitude of the energy alone may not be a good measure of the effective perturbation at the point of initiation of a steam explosion.

At atmospheric pressure there is some indication of a mass threshold for spontaneous triggering; if this extends to higher pressures, then the pressure above which spontaneous triggering does not occur might be higher for higher masses.

The consequence of these uncertainties in triggering at high pressure is that the range of possibilities is very wide. On one hand, explosions may never occur or only with low probability per core-melt accident. In this case, the geometrical limitation applicable at low pressures would not apply, because triggering would be expected to occur at random times during a pour. On the other hand, effective triggering might have a high probability.

Analyses of steam explosion experiments compare the heat in the fuel (over the temperature of the water) with the explosive energy; for the corresponding accident calculations, the heat content of the melt is required. For these scoping calculations, a base water temperature of 400 K will be used. In Subsection 3.3 above, the lowest temperature for liquefaction of core materials was estimated at 2000 K, when liquid Zircaloy begins to destroy solid UO_2. Using a specific heat of 500 J/kg•K for solid UO_2, and neglecting a small contribution from the latent heat of Zircaloy, gives 0.8 MJ/kg in the melt over 400 K as an estimate of the lower limit of the melt's heat content. An upper limit can be estimated by considering UO_2 heated up to its melting point, about 3100 K, and melted with latent heat 0.27 MJ/kg. This then implies a total latent plus sensible heat above 400 K of 1.6 MJ/kg. Subsection 3.3 above discusses the range of possible quantities of melt that might accumulate in a pool before discharging into the lower plenum. It is reasonable to suppose that there is a correlation between the size of this pool and its heat content per unit mass, because as time proceeds and the pool grows larger, it will (being internally heated by fission products that have not evaporated) get hotter.

The product of the amount of melt mixed with water when (and if) a steam explosion occurs and the melt's heat content is the amount of heat available to drive the explosion. The uncertainty in the fraction of this heat that is converted into kinetic energy, or the conversion ratio, is now considered. Experiments involving 1 to 20 kg of iron-alumina or corium simulant thermites have mostly yielded conversion ratios in the range 0 to 3%.[24 41] Some experiments[42] where the conversion ratio was less than 0.1% probably involved partially-frozen melt. In addition, the Sandia Fully Instrumented Test Series experiments have measured increases in internal gas energy in the closed vessel of up to 10% of the heat in the melt. This is attributed to steam produced in the explosion.[43] Under different geometrical conditions, the steam produced may have done more or less work, although not up to the whole 10% available on energetic grounds alone.

Extrapolating from conversion ratios measured in 10-kg experiments to explosions involving on the order of 10^4 kg of melt is a major source of uncertainty, because of our limited understanding of the processes governing fine fragmentation and heat transfer. Two extrapolations that have been made are considered here to give an indication of the range of possibilities. The first extrapolation assumes a constant conversion ratio and is used by Mayinger, for example, to advocate a maximum conversion ratio of 1%.[44]

The second extrapolation assumes fine fragments of constant size. Calculations in the Los Alamos section of the ZIP study[45] used this method with parameters chosen to be consistent with an experiment in which 9.3 kg of iron-alumina thermite gave a conversion ratio of 0.43%.[41] This was modeled with the SIMMER code. Using the same parameters to describe the explosion, the code then calculated conversion ratios of 3.7 to 14% for explosions with 10^4 and 2×10^4 kg of corium in an RPV. Two reasons were given for this increase with mass. First, in the experiment calculation, steam had expanded through overlying water which cooled it, whereas in the reactor calculation it had expanded through overlying melt which heated it. Second, the expansion timescale of the reactor calculation was about 10 times that in the experiment calculation, which allowed more heat transfer. These SIMMER calculations have been criticized and cannot be said to provide realistic simulation, particularly because the choice of a single, unvarying particle size is an oversimplification. However, they show the potential for effects which may cause conversion ratios to increase with melt mass.

An approximate upper limit on the conversion ratio can be determined from thermodynamic considerations; Swenson and Corradini have calculated values in the range of 2.7 to 17.9% for explosions in a PWR pressure vessel.[46] The lower values occur when the mass of water participating in the explosion is larger than the mass of melt, the highest ones where the melt mass is about four times the water mass. Generally higher values (2.3 to 22.2%) have been obtained by McFarlane,[47 48] who differs from Swenson and Corradini primarily in not assuming an initial void fraction in the fuel-water mixture.

Existing calculations do not, then, provide a hard upper limit for the conversion ratio of a steam explosion involving on the order of 10^4 kg of melt. For the purpose of the present scoping calculations, 16% will be used as a representative figure; this is consistent with References 25 and 31.

The kinetic energy generated in an explosion will be directed upwards and also downwards if the vessel bottom fails, as predicted if the explosion energy exceeds a threshold in the range of 1.0 to 1.5 GJ.[45] The partitioning will be roughly in inverse proportion to the masses above and below the middle of the exploding region; so in the case of explosions involving a substantial fraction of the core, the partitioning will be roughly equal. The uncertainty in this would be larger for smaller explosions that, depending on the trigger time and site, might occur above or below a substantial majority of the core mass.

Some dissipation of the kinetic energy of the resulting upward-moving slug of core debris and water will occur in the upper internal structure; an upper bound for this may be estimated by multiplying the maximum force that the structure holding the upper core plate down can sustain while being crushed, by its vertical length. For a PWR this will be about 250 MJ, at most. Swenson and Corradini assume that the core and upper internal structure can successively dissipate up to 50% and 90% of the slug energy,[46] which can be much greater than our upper bound estimate. Their assumptions imply retarding forces much greater than the yield strength of these structures, however.

Various calculations indicate that the bolts holding the vessel top head down will fail on impact of a slug of energy greater than a threshold in the region of 1.0 to 1.5 GJ.[25 49] If the bolts fail, the slug's momentum will be shared with the vessel top head, a mass of about 7×10^4 kg for a PWR. If a perfectly inelastic collision is assumed, the unimpeded rise height can then be calculated. If the slug rebounds, the rise height will be greater. It is often conservatively assumed that any impact of the top head on the containment roof implies failure; a non-zero closing speed would in practice be required.

The uncertainties described above are listed in Table 3-7. This is not an exhaustive compilation, and the parameter ranges listed are not necessarily all strict upper and lower bounds.

Two examples chosen from within these ranges of possibilities are now examined to explore the effect of combining these uncertainties.

First, suppose 10^4 kg of melt, with heat content 1.2 MJ/kg, explodes with 2.5% efficiency. Then the kinetic energy produced is 300 MJ. This is below the lower limits of the thresholds for vessel failure at either end, and so vessel and containment remain intact.

An example of a larger explosion within the bounds of possibility discussed above would be 4×10^4 kg of melt, with heat content 1.5 MJ/kg exploding with 8% efficiency. Then the resulting kinetic energy would be 4.8 GJ. This implies vessel base failure and, assuming equipartition of energy up and down, the upward-moving slug has 2.4 GJ of kinetic energy which, allowing for dissipation of 2×10^8 J in upper internal structure, implies bolt failure for any threshold in the range quoted above. Assuming the slug's mass and that of the top head are both 7×10^4 kg and a completely inelastic collision, momentum

Table 3-7 In-vessel steam explosion uncertainties

Uncertainty	Implications	Comments
Fraction of core molten at time of largest explosion	Mass of melt available for steam explosion	Limited by crust thickness that can support melt pool. Maximum ≈ 0.75
Fraction of melt mixed with water at time of largest steam explosion	Heat available for transfer to water	Unknown
Heat content of melt	Heat available for transfer to water	Approximate range 0.8 - 1.6 MJ/kg
Volume of water mixed with melt at time of largest steam explosion	Capacity of water to accept heat	Up to volume of lower plenum, 29 m^3 for a 3000 MWth PWR
Conversion ratio	Kinetic energy in explosion	Thermodynamic limit about 16%. Experimental values up to 3%.
Trigger probability	Likelihood of explosion; fraction of melt mixed	Low pressure: likely, but not certain. High pressure: probability unknown
Threshold explosion energy for vessel bottom failure	Mitigation by venting	Approximate range 1.0 - 1.5 GJ[45]
Fraction of kinetic energy dissipated by bottom failure	Energy of slug	
Energy absorption in upper internal structure	Mitigation of upper head impact	Up to about 250 MJ
Failure threshold for all bolts	Large missile formation	Approximate range 1.0 - 1.5 GJ[49]

will be equally shared, giving the combined missile (head and slug) a kinetic energy of 1.1 GJ and the head alone, 550 MJ. Further progress of the missile will be plant-specific, because of the various obstacles to upward movement that may be present. As an example, an inelastic collision with a missile shield of mass 7 x 10^4 kg would halve the missile's momentum, leaving kinetic energy of 137 MJ. If the vessel head then rises 50 m to the top of the containment dome, this takes up 35 MJ of gravitational energy, implying an impact speed of 50 m/s. Concrete penetration formulae indicate that penetration of a dome 0.81 m thick will occur at impact speeds above 30 m/s.[50] This example therefore implies containment failure.

Thus the combined uncertainties in this description of in-vessel steam explosions admit the possibilities that containment failure may or may not occur. From the discussion above it cannot be determined whether such failure can in practice occur, or if so under what initiating conditions (particularly primary system pressure), or with what conditional probability per core-melt accident.

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4. CORE-MATERIALS INTERACTIONS IN CONTAINMENT

This section addresses the ex-vessel behavior of the core materials. The behavior includes the release from the vessel, dispersal in the containment building, production of both radioactive and non-radioactive aerosols, generation of steam and hydrogen, and melt attack of the containment concrete and associated gas production. These activities are important in that they affect the potential for containment failure and the amount of radionuclides airborne at failure.

The ex-vessel behavior of the core materials may be divided into three phases, which usually occur sequentially in time. (Note that these phases are different from the ones defined in Section 3.) Phase 1 is the exit of core materials from the pressure vessel and involves the dynamics of melt release from the vessel. Phase 2 is the initial interactions of ex-vessel molten materials with water (if water is present). It includes initial melt fragmentation (or simple melt penetration of the water) and possible subsequent steam explosions. Phase 3 is the long-term interaction of ex-vessel core material with water or concrete or both. It includes the ultimate coolability of particulate debris and the interaction of melt or particulate debris with concrete.

In Phase 1, the manner of melt release from the vessel is important because the melt release process affects how the melt is dispersed, which can strongly influence the size of a pressure spike (which can threaten containment), and because it can generate large amounts of aerosols. In Phase 2, steam explosions within the reactor cavity can drive water from the cavity and alter the size of the pressure spike. Steam explosions also have the potential to damage containment structures and heat removal systems. They can also fragment and disperse the debris, affecting ultimate coolability. All of the consequences influence the potential for containment failure. In addition, steam explosions generate aerosols, which affect the amount of suspended radionuclides.

In Phase 3, debris can be either particulate and cooled in water or molten and attacking concrete or even hot dry particulate attacking concrete. Melt/concrete interactions produce copious quantities of both inert and radionuclide aerosols. The radionuclide aerosols from melt/concrete interactions dominate the radiological source term after several hours past vessel failure. In addition, steam, hydrogen, carbon dioxide, and carbon monoxide are produced. These gases can pressurize and threaten containment. Also, the hydrogen and carbon monoxide can burn and further pressurize containment. Finally, molten materials can threaten containment more directly by penetrating the concrete basemat. Coolable particulate debris in water does not produce aerosols or penetrate concrete, but it does produce steam which can pressurize containment more rapidly than a melt/concrete interaction if containment heat-removal devices are not operating.

4.1 Phase 1: Exit of Core Materials from the Pressure Vessel

The various modes of release of core materials from the pressure vessel may be grouped into four classes (see Figures 4-1 through 4-4):

- A pressure-driven melt jet
- A gravity-driven drop of a large melt mass
- Release due to a massive failure of the vessel bottom caused by an in-vessel steam explosion
- A continuous dripping of core materials not involved in the initial release

(Note: The figures are, of necessity, plant specific, but the phenomena to be discussed apply to many plants.)

4.1.1 Pressure-Driven Melt Jet

A high-pressure failure of the the pressure vessel is usually associated with a small-break LOCA, or a transient with a subsequent loss of heat sink.[1] The location and nature of the breach are uncertainties discussed in Subsection 3.4 and Reference 2. The Zion Probabilistic Safety Study (ZPSS) suggests that the most plausible breach mode for the Zion PWRs is along the welds where instrumentation penetrates the vessel bottom.[1] If the breach first occurs below the molten materials (e.g., at instrument penetrations), the melt will be ejected forcefully by the high pressure within the vessel. (See Figure 4-1.) Important uncertainties are the amount of material that forms aerosols, the amount that transfers heat rapidly to the containment atmosphere, the amount that is oxidized, the amount of hydrogen generated, and whether local hydrogen burning occurs even though steam inerting may occur elsewhere in containment. These uncertainties strongly affect the potential for early containment failure by overpressure as well as the radiological source term at that time.

Tarbell et al have performed high-pressure melt ejection experiments to determine the nature of the melt as it is ejected.[3-10] These experiments involved from 2 to 10 kg of aluminum oxide and iron melt at about 2700 to 3200 K ejected by 1.0 to 17.3 MPa of pressure. Considerable aerosol resulted from these tests, estimated to be about 1% of the melt mass ejected. Two distinct sizes of about 0.7 μm and 5 to 30 μm in diameter have been noted. These sizes may be caused by dissolved gases coming out of solution and sparging violently through the melt. The smaller aerosols might then result from the condensation of volatile metals or oxides or both sparged by the gases. These small aerosols could then be selectively made of fission products because many fission products are semivolatile metals. The larger aerosols result from the hydrodynamic breakup of the jet.

In an LWR accident, dissolved hydrogen would cause a result similar to that seen in the experiments.[4] Another possibility is that steam

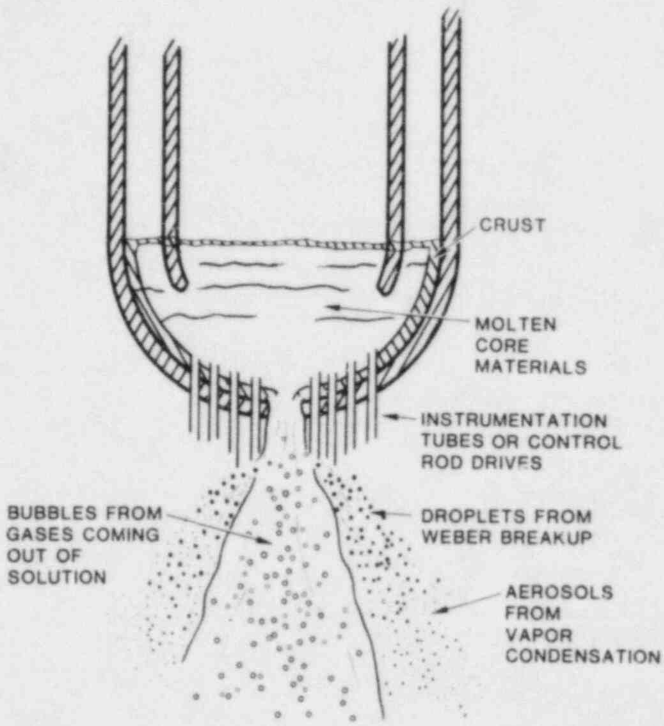

Figure 4-1 High-pressure melt release from bottom of pressure vessel.

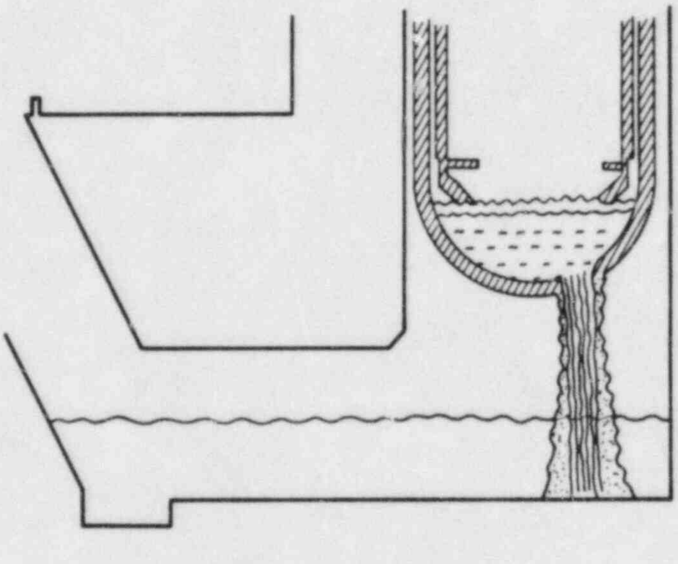

Figure 4-2 Low-pressure melt release from bottom of pressure vessel (Zion-type reactor cavity).

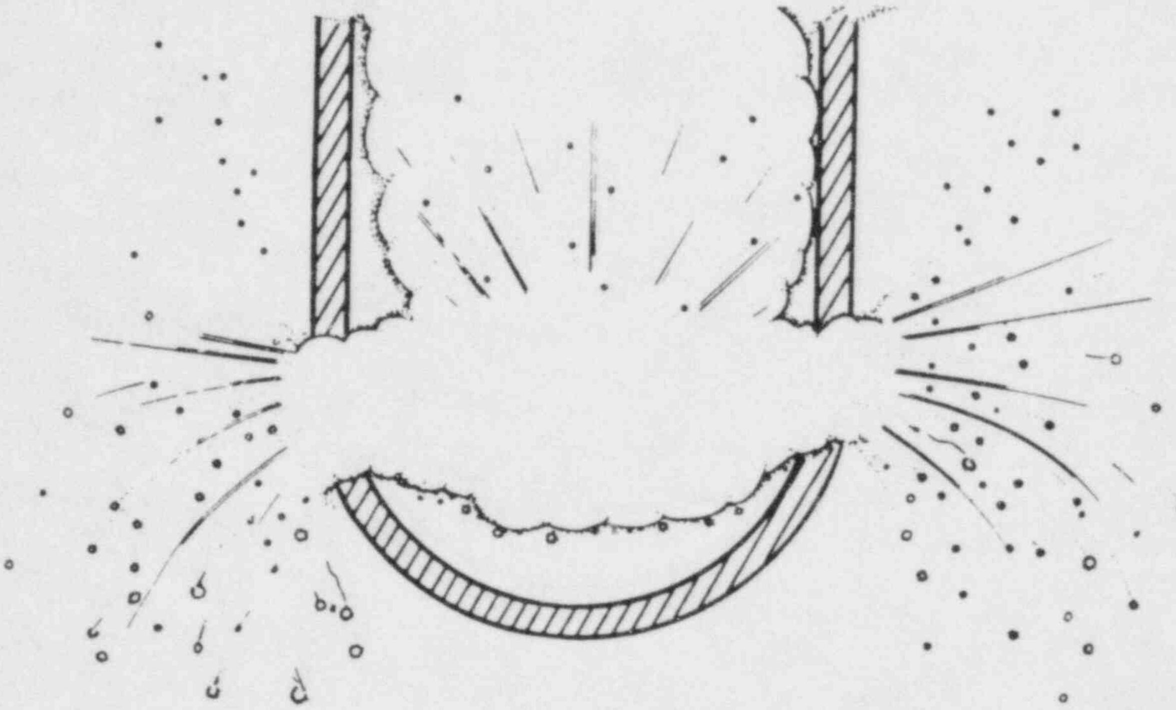

Figure 4-3 Vessel failure from steam explosion.

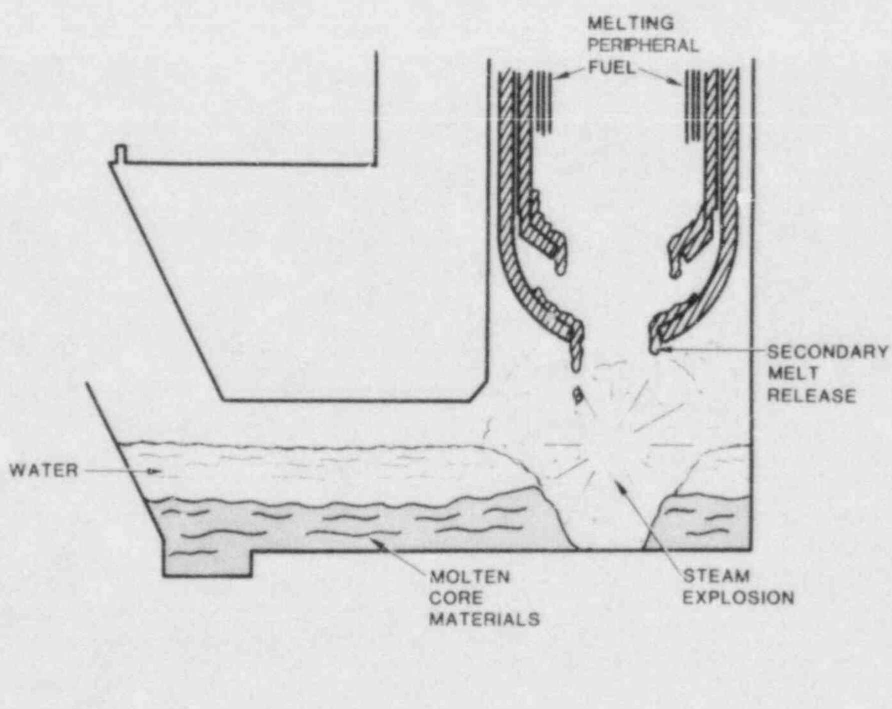

Figure 4-4 Secondary melt release causing a steam explosion in a water pool over a molten core pool (Zion-type reactor cavity).

and hydrogen above the melt pool in the RPV may be expelled at the same time as the melt by penetrating the melt before it is all ejected.[11] Thus, high-pressure melt ejection would be expected to create aerosols, but the total amount is very uncertain. If the scaling is linear with mass, it would be about 1000 kg. The effect of such an aerosol source could be either beneficial or detrimental. If containment failure occurred immediately because of an associated pressure spike, the released radioactivity would be greatly increased by the ejection aerosol source. However, if containment failure were delayed, the ejection source would help to agglomerate and settle out the highly radioactive aerosols.

An important aspect of high-pressure melt ejection is the potential for direct transfer of much of the stored heat in the melt to the containment atmosphere. Such heat transfer will result in a greater pressure increase than if the same amount of heat goes into producing steam. In this case it is not the aerosols that are of interest, because they comprise only a small fraction of the melt (1% in the experiments). Rather, it is the small fragments about 0.1 to 10 mm in diameter that are of concern, because they comprise a large fraction of the ejected melt in the experiments and can transfer heat to the atmosphere rapidly. Recently, Tarbell et al have observed the effects of such a direct heat transfer.[9 10] Ten kg of melt ejected at 10.8 MPa of pressure transferred enough energy to the atmosphere of the experimental facility that it lifted the 6-ton building through 0.7 meters.

An additional feature of the ejecta is that they are composed partly of hot particles which may contain unoxidized metal (zirconium or steel). These hot particles may ignite the subsequent flow of hydrogen streaming out of the vessel breach. (The metal surfaces of helical hydrogen igniters induce ignition in a 6% hydrogen:air mixture when the surface temperature exceeds about 800 K.[12]) In addition, metallic particles may oxidize in steam and produce more hydrogen (and heat). They might also oxidize directly in air. This would liberate heat equivalent to the simultaneous reduction of steam by metal to hydrogen and the combustion of that hydrogen in air. All of these actions will increase the containment pressure and the potential for an early containment failure. However, the size of the pressure increase from these mechanisms is unknown because the size of the particles is uncertain and the likelihood of hydrogen burning is unknown. This pressure has the potential to fail containment early and release the large amounts of radionuclides that are suspended in containment at the time of vessel breach.

The flow patterns and destinations of debris from high-pressure melt ejection obviously depend on the containment geometry. For Zion, the ZPSS suggests that, after the melt is deposited in the reactor cavity, high-velocity steam and hydrogen will remove the melt from the cavity by droplet entrainment, sweepout, and large wave formation.[1] (See Figure 4-5.) Some experiments have been performed to investigate sweepout. Experiments by Spencer and Bengis involved a 1:40 linear scale model of the Zion plant cavity with nitrogen gas flowing over water or Cerrelow (molten metal) to simulate high-velocity steam and

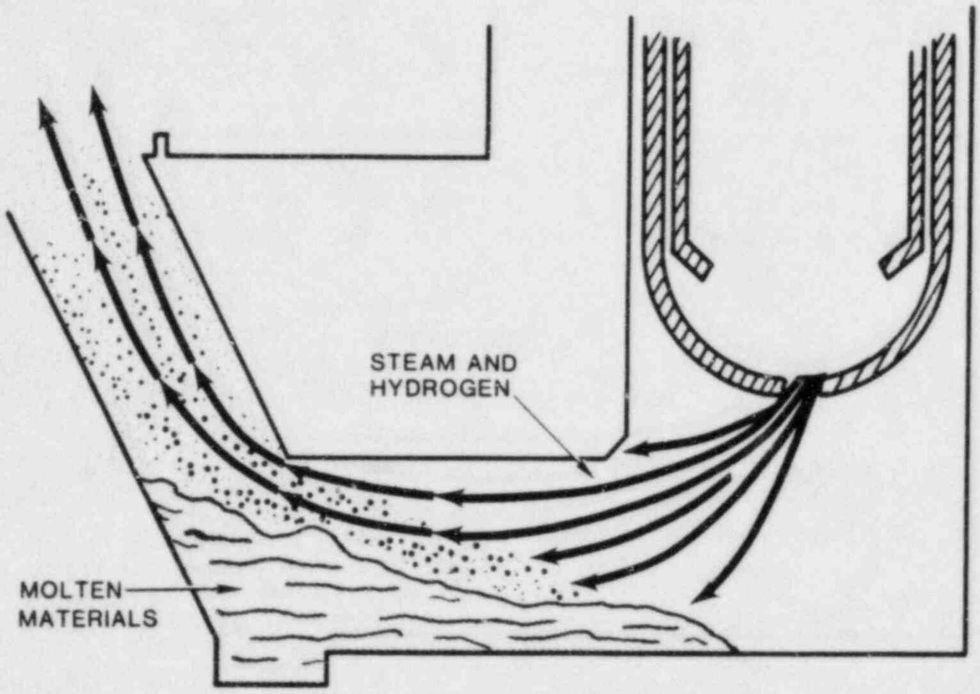

Figure 4-5 High-pressure sweepout of melt by droplet formation and entrainment (Zion-type reactor cavity).

hydrogen flowing over molten core materials.[13] Entrainment and sweepout of the water and Cerrelow were achieved with gas velocities of 20 and 8 m/s, respectively. This agreed with predictions based on Weber breakup of the melt. Extrapolation to the full-scale Zion PWR containment geometry results in predictions of melt sweepout if the hole in the pressure vessel is greater than about 30 cm in diameter. Thus, the uncertainty in melt sweepout and dispersal is linked to vessel hole size. On the other hand MacBeth, using a 1:25 model cavity of a different geometry in which compressed air was blown over water or organic liquids, found that entrained liquid impacted the far wall (which was vertical) of the instrument tube tunnel and flowed back into the cavity.[14] Thus actual plant geometry may affect which uncertainties are important for the melt dispersal process.

The sweepout process has the potential for producing large amounts of steam rapidly if it scatters the debris into water pools. In addition, if the melt is fragmented as it is swept out, all of the pressurization modes described above are again of concern. (The diameter of the fragments produced by flowing gases is inversely proportional to the square of the velocity, and the velocity is proportional to the square of the hole diameter in the pressure vessel, according to Reference 3. So, here is another instance where the uncertainty in the vessel hole size has important effects.)

The above discussion on the consequences of melt dispersal depends on the melt being dispersed as fragments. Whether or not this is the case depends on the structural geometry below the RPV (which is known and which depends strongly on reactor and containment type) and melt dispersal behavior (which is not known). For PWRs fragmented melt may impact the far wall of the instrument tube tunnel and not be dispersed into the air, as in MacBeth's experiments.[14] Conversely, the high-velocity gases may reentrain the impacted melt layer on the wall (depending on its velocity). It is within the range of present uncertainty that the melt dispersal (if there is any) may occur as a flowing of the melt up and out of the keyway (driven by gas flow). Another possibility is that the melt may be entrapped by containment structures and not dispersed as fragments into the atmosphere.[15] In these cases, a major pressure spike will not occur from the sweepout process (but may occur from the initial jetting of the melt). However, the important consequence of this scenario is a reduction in the melt/concrete interaction because the melt is spread out. This also depends on the amount of melt ejected which is affected by uncertainties in core melting processes. A reduction in melt/concrete interactions will reduce the ex-vessel radionuclide aerosol and inert aerosol production rates. It will have beneficial or detrimental consequences depending on the radionuclide inventory in the melt and the containment failure time. In some plants, wide dispersal of debris could plug the containment sump pump.

Even though the geometry in BWR containments is very different from that in PWRs, uncertainties in the destination of melt following high-pressure melt ejection from the RPV may have qualitatively similar effects on risk. For example, in a BWR Mark I containment, if debris follows the flow of steam down the vent lines to the toroidal suppression chamber and is deposited in the vent lines, they will be thermally attacked. Their failure would prevent fission-product retention in the suppression pool. In a BWR Mark II containment if debris remains in the pedestal cavity (directly below the RPV), containment failure by late overpressure is predicted.[16] However, if it is dispersed, either into water, generating steam, or so that the atmospehre is directly heated, early overpressure failure may occur.

4.1.2 Gravity-Driven Drop of a Large Melt Mass

If the vessel is breached by molten materials at low pressure, the melt essentially falls by gravity from the vessel. A breach at the base of the melt will allow a large release at one time. (See Figure 4-2.) A breach near the top of the melt will cause a slower release as the melt spills out the side of the vessel, possibly cutting a channel. In both cases, no immediate dispersal occurs upon exiting. Thus, there are no really important uncertainties in release behavior once a low-pressure failure is assumed. The molten materials simply fall. An uncertainty of intermediate importance is the amount of steel added to the melt before breach. The added steel will reduce the temperature of the molten materials and thus reduce the ex-vessel aerosol generation rate and affect the radiological source term.

If there is no water below the vessel, the melt spreads out on the floor after dropping, and melt/concrete interactions begin. This will be described in Subsection 4.3. However, in many cases water will be present below the vessel. This is because a large amount of water must be lost from the vessel (into containment) in order to allow the meltdown of an LWR core. Additional water from emergency cooling storage tanks may add to the total on the containment floor. The presence of water below the vessel will depend upon containment geometry (e.g., whether curbs prevent flow into an instrument tunnel) and may depend on whether containment sump pumps are working. The process of melt quenching in this water will be addressed in Subsection 4.2.

4.1.3 Vessel Failure from Steam Explosion

Subsection 3.5 notes the possiblity of massive failure of the vessel bottom from an in-vessel steam explosion. (See Figure 4-3.) The core materials which exit the vessel in this case will be fragmented from the steam explosion. (However, they may not all be frozen since Subsection 3.4 notes that insufficient water exists in the lower plenum to completely quench the melt.) Steam explosion debris can range in size from one micrometer to several centimeters.[17 18] Some of this debris will be dispersed by the high-pressure jetting of steam and hydrogen following the vessel failure. This prefragmented debris will then be quenched further in water on the containment floor and will generate steam rapidly. There is considerable uncertainty in whether such a large in-vessel explosion will occur, the amount of material so dispersed, the area over which it is dispersed, and the resulting particle size distribution. A major consequence of this melt release mode is the possibility of the resuspension of large amounts of radionuclide aerosols that may have settled inside the RCS. In addition, more aerosols may be generated from oxidation during the explosion. Both of these activities affect the radiological source term. They are discussed further in Subsection 7.1.3.2.

4.1.4 Gradual Secondary Release

Following all three of the release modes just described, a sizeable fraction of core materials may remain unmelted in the core region.[1] This material is typically near the radial edge of the top of the core and could comprise as much as 30% of the core. It has a lower power density than the core center (see Figure 3-2) but, without coolant, much of it will eventually melt, especially as the core barrel heats up. This material will melt and drop out of the RPV in small amounts over a period of hours. (See Figure 4-4.) This mass will be added to debris already below the RPV. If there is water below the RPV, this mass will almost certainly induce steam explosions.[18 19] The effect of these small steam explosions will be described in Subsection 4.2.2. If there is no water below the RPV, the dripping mass will add to the melt/concrete interaction. The additional mass is not important, but the radionuclide inventory in the dripping material will enhance the radiological source term at late time, when it might otherwise be very low because of aerosol settling.

4.1.5 Summary of Phase I Uncertainty

Of all the uncertainties in the four modes of melt release listed, the uncertainties in the high-pressure melt release are considered to be the most important. This is because high-pressure ejection is associated with a TMLB' accident, which is risk dominant in many PWRs. In addition, the uncertainties in melt behavior directly impact both the containment failure potential (via a pressure spike) and the radiological source term (via the copious amounts of ejection aerosols generated). Indeed, high-pressure melt ejection is a possible mechanism for inducing containment pressurization and hydrogen ignition simultaneously with vessel failure, just when the highly radioactive in-vessel aerosols are released from the vessel. The Phase I uncertainties are listed in Table 4-1 of Subsection 4.4.

4.2 Phase 2: Initial Interaction of Ex-Vessel Melt with Water

The major possible results of the initial interaction of ex-vessel molten materials with water are (1) a steam/pressure spike, (2) a steam explosion, (3) formation of coolable or noncoolable debris, and (4) formation of an unfragmented pool of melt under water. The first result directly impacts the potential for early containment failure (at which time the amount of airborne fission products is particularly high). The second result can damage containment cooling systems, generate hydrogen, and generate additional radionuclide aerosols. The third and fourth results affect the generation rate of ex-vessel aerosols, the long-term pressurization rate of containment, and the melt attack on the basemat.

4.2.1 Pressure Spike in Containment

A concern in quenching of a large melt mass is the rapid generation of steam and hydrogen. This concern is present whether or not a steam explosion occurs. If the steam generation rate exceeds the condensation rate, containment pressure will increase until the steam generation rate is reduced and condensation can reduce the pressure. The magnitude of the pressure spike produced by this process depends on the steam generation rate, total amount of steam generated, and condensation rate. A simultaneous hydrogen burn (e.g., from a local high-concentration region which is not steam inerted) might add to the pressure spike.

A molten mass at 2573 K (2300°C) composed of 100% of the core UO_2 and Zr plus an equal mass of structure will cause a steam spike of 345 kPa (50 psi) in the Zion containment (a large, dry PWR) if it is completely quenched faster than the condensation rate on containment structures.[20] This spike is added to the pressure already developed by the release of steam and hydrogen from the vessel during core meltdown. For Zion, this comes close to the containment failure pressure of 1 MPa (absolute) calculated in the ZPSS.[1] (Uncertainty in containment failure pressures is discussed in Subsection 6.5.) The amount of water required for such a quench is 135 000 kg (which is enough to fill the Zion plant reactor cavity and keyway to the tunnel roof). Thus, water starvation must also be considered in these calculations.

The steam and gas generation rate depends on the mass and temperature of the melt and on the nature of the quenching process. The uncertainty in the melt mass is propagated from uncertainties in in-vessel phenomena. A large-scale steam explosion would create the steam faster than it could condense but might not fully quench the debris unless it dispersed it into water pools on the containment floor. Fragmentation without a steam explosion would yield a smaller rate of vapor generation because of the larger average size of the fragments. The rate from the quenching of a hot, dry particle bed with water from above is about equal to the dryout heat flux for that bed.[21] If the debris is spread over a large area, this quenching rate could be quite large. Another type of pressure spike comes from the dispersal of debris into the containment atmosphere (as described in Subsection 4.1.1). In this case, the hot debris particles will heat the atmosphere directly. For a given amount of hot material, this process generates an even larger pressure increase than does evaporating water. This is because, with small enough particles, the heat will go quickly to the atmosphere, and the pressure increase from heating the existing steam will be greater than from generating new steam by boiling liquid water.

A final source of pressure-spike concern is from a hydrogen burn concurrent with a high-pressure vessel breach. Even though hydrogen concentrations in the entire containment building may be too low to burn, or the presence of steam may prevent ignition, the concentration in the reactor cavity during vessel blowdown may be high enough to burn (if not inerted), and the hot aerosols produced by the melt ejection process may form a convenient ignition source. Additionally, hydrogen burning may occur outside the flammability limits--defined as conditions in which a flame just propagates indefinitely--but will be local to the ignition sources and hence, in general, incomplete.[22 23] However, if hot aerosols or particles form a distributed ignition source in containment after vessel blowdown, a significant amount of burning may occur despite inerted conditions. Thus a hydrogen burn may occur at vessel breach and add to the pressure spike from vessel blowdown, melt-water interactions, and direct atmosphere heating. In combination, these processes may yield a high-pressure spike that could lead to early containment failure coincident with the release of radioactive in-vessel aerosols and melt ejection aerosols. The uncertainty in whether such an event occurs stems from uncertainties in both melt ejection behavior and melt quench behavior.

4.2.2 Ex-Vessel Steam Explosions

The potential effects of ex-vessel steam explosions include damage to containment structures and containment cooling systems, the generation of hydrogen, and the production of additional radiological aerosols.

A large ex-vessel steam explosion might occur when a large amount of molten material drops into a pool. Conversely, one might occur when a pool of water on top of a molten pool is triggered into an explosion. The dropping of a large melt mass into a water pool (as in a low-pressure vessel breach) will be considered first.

4.2.2.1 Steam explosion from dropping a large molten mass into a water pool

Many melt-quench experiments have been performed in the investigation of steam explosions (see Subsection 3.5 and Reference 18). The largest including oxides have consisted of 10 to 20 kg of molten oxides dropped into water.[17-19] In most of the cases a steam explosion has resulted. Yet some have suggested that a large steam explosion will not occur when melt contacts water outside a reactor vessel.[1] The suggestion is that when very large masses (e.g., 50 000 kg for half of a core) are involved, the melt will not fragment sufficiently at first to establish the water-melt mixture needed for a large steam explosion.

Henry and Fauske have proposed a model for fragmentation of melt in water based on limiting the quenching rate by the flat-plate critical heat flux for pool boiling.[24] Both the Gittus report and Corradini note that quenching is very dynamic and far from the steady-state assumptions used in critical-heat-flux modeling.[25 26] Furthermore, when the melt falls into a pool, the core materials are quenched by water streaming in from below, and pool boiling arguments are irrelevant. Corradini proposes an alternative fragmentation model based on fluidizing the particles with a vapor source from below. Both fragmentation models are simple and based on steady-state criteria, and thus are subject to considerable uncertainty.

The particle diameter predicted by the Henry-Fauske model agrees well with the one quench experiment reported by Benz et al, in which both the initial conditions and the resulting average particle size are given.[27] However, the Henry-Fauske model predicts pre-explosion particle diameters from the Fully Instrumented Test Series (FITS) experiments that are ten times larger than observed.[28] The diameter predictions of the Corradini model are about two times larger than observed.

Major differences occur when the two models are extrapolated to reactor-sized melts. For 20 000 kg of melt (about 20% of a PWR core), the Henry-Fauske model predicts a "particle" diameter of 74 cm (i.e., essentially no fragmentation will occur).[24] In contrast, the Corradini model predicts a particle diameter of 3.5 cm (assuming a 5-m-deep pool of water in the reactor cavity).[29] Such large differences in predictions indicate the large uncertainty in this area. Indeed, the uncertainty range is even larger than the two models indicate, because both models overestimate the FITS debris size.

If the melt is fragmented to centimeter-sized particles and mixed within the water, a steam explosion is likely (see Subsection 3.5) because of the general likelihood of spontaneous triggering in a large system (especially when the mixture contacts the cavity floor). Thus, because of the uncertainty in large-melt fragmentation, it is unknown whether a large steam explosion will result from the drop of a large molten mass into a water pool.

4.2.2.2 Steam explosions from water on a molten mass

There are several scenarios that would result in a water pool on a large mass of molten materials. First, during a low-pressure melt release, molten materials may drop by gravity into a pool of water below the vessel. One possibility is that a small steam explosion occurs from the initial melt-water contact, which blows away the water. This is followed by the formation of a molten pool, which is then covered by the return of the water that was blown away. Another possibility is that the melt penetrates the water and forms a molten pool below the water. Water expulsion and return might also occur after a high-pressure vessel breach.

It is uncertain whether a large steam explosion will occur starting with a water pool on top of a large molten pool. Whether or not a steam explosion occurs may depend upon the melt and water temperatures, the presence of noncondensable gases (suppressive), the presence of a crust between the melt and water (suppressive), and the presence of external triggers. Noncondensable gases from core-concrete interactions will be present. A crust, whose steady-state thickness would be of order 1 cm, may be unstable against cracking caused by liquid motions enhanced by gas bubbling (especially in a pool several meters across). The two liquids may intermix due to gas bubbling and steam production.

When Tarbell et al added water to the top of a 20-kg iron-alumina melt at 2700 K, film boiling on the melt appeared to occur, followed by nucleate boiling as the melt cooled.[4] No steam explosion occurred, although the melt and water appeared to mix spontaneously. A crust did not appear to form. Greene et al poured saturated water on top of liquid bismuth, lead, and Wood's metal in an "unmixed pool geometry".[30] Interfacial mixing occurred followed (always with bismuth, sometimes with lead and Wood's metal) by a steam explosion. Evans et al observed one steam explosion in two tests when they poured water onto a molten pool of iron-alumina/melt in graphite crucibles where no noncondensable gases would be expected.[31]

Another consideration in molten-pool/water-pool interactions is the possibility of small "external" steam explosions in the overlying pool. (See Figure 4-4.) Roughly 10 000 kg of core materials may continue to melt and fall from the reactor vessel after the initial, primary release of materials. Experiments show that, when a 10-kg mass of molten oxide drops through a water pool, it fragments, and a steam explosion is very often triggered when the melt contacts the pool bottom.[18] (These steam explosions were enough to blow the debris 10 m into the air in the experiments.) Melt amounts of approximately 10 kg could certainly be envisioned to fall from the vessel during the slow melting of 10 000 kg.

If these secondary melt releases caused steam explosions in the water pool above the primary melt pool, the pressure generated could push the melt down strongly at one location, causing it to rise up strongly in another. (A steam explosion with 10 kg of melt and a conversion ratio of 1% releases enough kinetic energy to raise 50 000 kg through 0.2 m.) In this fashion, mixing of the melt and the water might be induced. An external, mechanical trigger induced a steam explosion between liquid steel and water separated by a crust of slag in the Appleby-Frodingham accident.[32] Apparently the trigger broke the crust. Indeed, the occurrence of an initial steam explosion from the secondary melt might disrupt any crust, mix melt with water, and collapse the film boiling in the melt-water mixture inducing a large steam explosion.

4.2.2.3 Consequences of a large ex-vessel steam explosion

The above discussions lead to the conclusion that large ex-vessel steam explosions are within the current range of uncertainty. A large steam explosion will fragment the debris into much smaller sizes and disperse it. The aerosols generated will be both an airborne radiological source and a potential source for damage of containment cooling systems. The debris dispersal will affect the ultimate coolability (Subsection 4.3). Steam will be generated both during the explosion and when the dispersed debris falls into water on the containment floor. This will induce a steam spike that will load containment. Hydrogen will also be generated during the explosion. (Preliminary results from the FITS tests indicate about 20% to 30% of the available metal produced hydrogen during the explosions, compared with about 5% to 10% without an explosion.[33]) More hydrogen will be generated when the fine metallic debris lands in water.

The possibility of direct explosive damage also exists: the geometry is similar to that in the Quebec iron foundry accident in which a steam explosion caused by 45 kg of molten steel cracked a concrete basemat 0.5 m thick.[34] Reactor accidents will involve up to a thousand times more melt mass. It is very uncertain how the potential damage will scale. An example of the possibility of direct damage is to concrete walls 1.4 m thick in the pedestal region of BWR Mark II containments, directly below the RPV. Concrete structures immediately below the RPV in PWR containments are thicker, typically about 3 m, however. Finally, missiles may be generated which may damage the containment or the electrical or hydraulic functions of the containment cooling systems. It is sometimes difficult to identify a damaging missile, however. Such identification would in general be plant specific. One possibility is the RPV. For example, a steam explosion like the smaller one envisaged in Subsection 3.5, involving 10 000 kg of melt and yielding 300 MJ of kinetic energy might occur below the RPV in the water-filled reactor cavity of a large dry containment PWR. If the mass of water in the cavity, tamping the explosion, were half the mass of the RPV, the RPV would take up about one third of the explosion energy, 100 MJ. For an RPV mass of 3×10^5 kg, such initial kinetic energy in upward motion would lead to a rise height of 30 m, if the vessel were unrestrained. All of these consequences of a steam explosion are not well determined and represent uncertainties in a severe accident sequence.

4.2.3 Formation of Coolable or Noncoolable Debris

The initial interactions of molten materials with water will determine whether the materials are fragmented and quenched or remain molten. If fragmentation occurs, the fragment size distribution, amount of dispersal, and degree of stratification in the resulting particle beds are very important to the question of debris coolability. As shall be seen in Section 4.3, the debris characteristics are the major uncertainty in determining debris coolability. Debris coolability is important because it affects the long-term containment pressurization rate, and it precludes ex-vessel aerosol formation from melt/concrete interactions.

4.2.4 Consequences of No Fragmentation

Without fragmentation, the melt is assumed to penetrate the water and begin attacking the concrete (Subsection 4.4). Large masses of steam or hydrogen are therefore not generated in the quench process. The melt generates a crust, and the water boils above the crust. Under steady conditions the crust would be about 1 cm thick on a pool several meters across. The stability of such a thin crust is uncertain. If the crust collapses, fragmentation and quench questions are raised again, in a different geometry. The outcome will affect ultimate coolability (Subsection 4.3).

4.3 Phase 3: Long-Term Interactions of Ex-Vessel Core Materials with Water or Concrete

The outcome of Phases 1 and 2 is either a bed of hot particulate debris or a molten pool of core materials. If water is available, the particulate debris may be coolable. If water is not available, or if the debris is not coolable, the hot debris will attack and penetrate the concrete. This subsection discusses the long-term interactions of debris with either water or concrete.

4.3.1 Particulate Debris Coolability

Particulate debris in many cases will begin hot and dry, because the processes of fragmenting and freezing molten materials produce a large vapor flux that will remove liquid from the debris. The water that was blown away may return quickly, however, and attempt to quench the hot debris. Conversely, the fragmentation process may evaporate all available water, and quenching may occur when water is reintroduced by safety systems.

The quenching of hot particulate debris by water is a dynamic process, involving the counterflow of liquid and vapor, the heating of the dry particles by decay heat, and the heating of the concrete below the debris. The first step in understanding the quench process is to determine under what conditions the debris can be quenched. For some bed configurations, the decay power in the debris will prevent quenching. Indeed, even if the debris were initially cool and water already present, the decay power could be strong enough to cause the bed to become dry, even while the overlying pool of water was maintained.

Heat removal in initially cool debris submerged in water may proceed by conduction, single-phase convection, or boiling. Conduction and single-phase convection (by natural circulation only) are usually inadequate to remove the decay heat from the debris. In such a case, boiling of the water will occur. The steam rising out of the debris will restrict the flow of replenishing water from the overlying pool. If the decay-power level is large enough, the vapor flow will slow the liquid flow sufficiently to evaporate all the liquid before it reaches some parts of the debris. (Numerical examples will be given shortly.) Local dryout of the debris will occur in those regions. The power required just barely to cause a dry zone in a submerged bed is called the incipient dryout power for that bed. If the decay power in the debris exceeds the dryout power, and if it starts out hot and dry, it cannot be quenched with water added from above. If the debris starts out wet, parts of it will dry out. In both cases, the dry zone will heat up and attack the concrete.

Power levels only moderately greater than the incipient dryout power will often cause a large fraction of the bed to become dry (if it starts out wet) or remain dry (if it starts out dry). Decay-heat-removal capability in the dry region of a debris bed is much less than in the boiling zone. Because of the low thermal conductivity of dry debris, the low efficiency of radiation at low temperatures, and the low vapor-flow rates expected, high temperatures can be achieved over short distances; initially much of the dry zone can heat at near-adiabatic rates. Thus, dryout marks a sharp change in the coolability of debris and indicates the potential for prolonged thermal attack on the concrete.

There are well-established models for predicting the coolability of particulate debris (described in Reference 35). The dryout model developed in Reference 35 predicts most of the experimentally determined dryout powers within a factor of two. Thus, the major uncertainty in debris coolability presently lies not in the modeling but in knowing the bed configuration. However, modeling uncertainty still exists in some regimes of interest for LWRs, including deep stratified beds, beds with large spans in particle size, beds with highly irregular particles, and beds with potential for horizontal liquid flow. The dryout power depends on the debris thickness, the average particle diameter, whether the debris is stratified (as it might be after settling through water or after a steam explosion), and boundary conditions such as the porosity of its support structure. Generally, uniformly mixed debris with large particles in a thin pile is the most coolable.

For example, if 100% of the core of a typical 3000-MW plant is assumed to be spread over the area of a reactor cavity of area about 50 m^2, the heat flux from the debris at two hours past shutdown with 25% decay-heat reduction due to loss of volatiles would be 500 kW/m^2. Assuming a 40% porosity, the debris would be about 40 cm thick. If the debris were uniformly mixed, fairly large particles (2.2 mm average or more) are required to avoid dry out.[35] If only 50% of the core is used and a porosity of 50% is assumed, smaller particles (0.40 mm or more) could avoid dryout. On the other hand, if the debris in the latter case consisted of a spectrum of particles ranging from 0.04 to 4.0 mm (such as might come from steam explosions), and if the debris were completely stratified (i.e., monotonically decreasing particle diameter with increasing elevation in the bed), the model predicts that the debris bed must be thinner than 5 cm to avoid dryout. This illustrates the potentially strong effect of stratification on coolability.

Average particle sizes from steam explosions are often in the range of 0.4 mm. Thus the uncertainties in fragmentation during steam explosions span the range of coolable to noncoolable debris. In addition, steam explosions tend to disperse debris. This results in shallower debris and allows coolability with smaller particles, if the debris were uniformly mixed. However, debris falling after dispersal will be somewhat stratified, which will reduce coolability. Finally, the average particle size from a steam explosion tends to decrease as the efficiency of the explosion increases. All these uncertainties associated with debris generated by a steam explosion illustrate why the initial debris configuration is the major uncertainty in particulate debris coolability. It also illustrates the need for melt quenching experiments at scales up to about 10^4 kg.

4.3.2 Quenching of Hot, Dry, Particulate Debris

Steady-state dryout models describe the maximum amount of heat or vapor flow that can be removed from a particle bed while still allowing liquid to enter the bed. They can thus predict whether or not debris can eventually be quenched, assuming that no bed alterations occur during the quenching process. The dryout models also provide a good first estimate of the rate at which a quench front will progress downward. The average heat removed from the bed during quenching is related to the dryout flux, because the liquid entering the bed is restricted by the vapor leaving it in the quench process, just as it is near incipient dryout.

An important phenomenon noted in some recent quench experiments is the rapid quenching of a central column of the debris by a finger of liquid, while the surrounding annulus remains hot.[21] After the central finger reaches the bed bottom, the water quenches the annulus from the bottom upward. The time to quench the entire bed is approximately equal to the sensible heat in the bed divided by the dryout flux times the bed area. However, the central quench occurs in about one third of the total quench time. The central quench time is important because it marks the halt of heating of the concrete by the unquenched debris. (The detrimental effects of heated particulate debris on concrete are discussed in Subsection 4.4.6.) There is some uncertainty in the universal occurrence of a quench finger because such a finger was not observed with small particles in a fission-heated UO_2 coolability experiment.[36]

Another feature of the two-stage quench process is that while the fingering quench is occurring steam is passing the superheated debris (which is being kept hot by decay heat). If the temperatures are high enough, metal in the debris will be oxidized during this process.

This will add heat and hydrogen to the system (as during the in-vessel core degradation process) and will reduce the quenching rate.

During the quench process, the debris is being heated by decay heat. If there is no steam source below the dry region, heat transfer in that region will be primarily by conduction and radiation. Initially conduction dominates, but at higher temperatures radiation between the particles increases the effective conductivity by several times. Because of the low thermal conductivity of dry particulate debris (lower by many times than oxide alone), much of the dry zone will heat adiabatically. The quench rate will slow considerably as it enters levels with hotter particles. The debris near the concrete will deliver some of its heat to the concrete and begin to melt it. As will be described in Subsection 4.4.6, experiments have shown that the hot particles sink into the melted concrete and may become inaccessible to quench water.[37] Thus, the rapid fingering quench, if it occurs, is important for halting this process.

4.3.3 Permanent Particulate Debris Coolability

A final concern in quenching particulate debris is keeping it cool after it is quenched. As long as a water supply is maintained and the bed structure is not changed, a fully quenched bed should remain cool, and attack of the concrete will be avoided. However, if the coolable debris bed is formed from the initial release of melt from the reactor vessel, other debris formed from the secondary (continuous) release of residual core materials may cause dryout and attack the concrete. The slow secondary release may comprise, over a period of hours, 10 000 kg of melt (within an order of magnitude). If the release comes in drops of about 10 kg each, there may be many hundreds of "small" steam explosions, depending on the ambient pressure and the void fraction in the pool. (A steam explosion resulting from 10 kg of melt is small compared to one involving 100 or 1000 kg, but is still quite powerful.) Assuming the explosions are small enough to allow the water to remain in the reactor cavity, each event will produce fine debris that will settle on top of the coolable debris, creating a stratified bed.

Models and experiments for stratified beds indicate that they are not easily cooled. In deep beds, the dryout criterion is established by the top layer, which in this case is composed of small particles. If the debris is essentially one-dimensional, the top layer will inhibit the downward flow of water sufficiently to cause nearly the entire bed to dry out, even though it had previously been cooled. However, in a two-dimensional bed there may be regions where the liquid can bypass the small-particle layer, travel horizontally, and help cool the debris below the layer. A model that allows liquid flow from below is a start for describing this situation, but a two-dimensional model is really needed.

4.3.4 Molten Materials Coolability

The coolability of molten materials by water is another source of uncertainty. If water is added to the top of a melt, a steam explosion may result immediately. This would fragment the materials and lead to a particulate coolability problem. On the other hand, a crust may develop during the introduction of water which may protect the melt. However, under steady conditions the crust would be about 1 cm thick. Whether such a crust is stable on a molten fuel pool several meters across is questionable. If not, fragmentation and quench questions are raised again.

Theofanous has suggested that the gas generated during a melt/concrete interaction would break up the melt and allow overlying water to enter and fragment the melt into particles which are just the right size to avoid dryout.[38] Such a fortuitous circumstance needs experimental verification.

As described in Subsection 4.2.2.2, when Tarbell et al quenched a 20-kg iron-alumina melt at 2700 K with water, the melt quenched very slowly.[4] When Evans et al performed a similar quench, a steam explosion occurred.[31] Neither experiment agreed with the Theofanous hypothesis.

4.3.5 Melt/Concrete Interactions

In the event there is no water below the vessel when the core melt materials are released, or if the water is blown out, or if the melt penetrates the water without fragmentation and remains there stably, or if particulate debris dries out and remelts, then attack of the concrete by the melt will occur. Hot particulate debris can also attack the concrete, either in the absence of water or during bed dry-out. The four major areas of concern in the interactions with concrete are

- The generation of aerosols (both radioactive and not),
- The generation of combustible noncondensable gases (hydrogen and carbon monoxide),
- The generation of steam and carbon dioxide, and
- The axial and radial penetration of the basemat by the melt.

Aerosol generation rates, including those for refractory fission-product aerosols, dominate the determination of the amount of suspended fission products several hours after vessel failure. Uncertainties in the generation rate lead directly to uncertainties in the radiological source term at late times. The generation of combustible gases can help cause containment failure by increasing the amount of material available for a burn. The generation of noncombustable gases can lead to long-term pressurization of containment. But because concrete absorbs much of the decay heat, the steam generation rates are generally much less than from boiling water in coolable debris (by an order of magnitude). The most important aspect of gas generation rates is that it affects the ex-vessel aerosol generation rate both directly and indirectly by affecting the melt temperature.

The axial penetration of the melt into the basemat can lead to a breach of containment. Such a breach might preclude containment failure by overpressure but nonetheless would release radiation into the ground. The radial penetration of the melt in a BWR Mark III containment could cause bypass of the suppression pool. In some BWRs, thermal attack on the pedestal supporting the RPV (either by direct contact with melt or by radiative heat transfer causing spallation from thermal stressing[39]) might cause the RPV to collapse. This could cause containment failure due to pulling on pipes.

Large-scale melt/concrete interaction tests (using 200 kg of molten steel) have shown that gas can be generated from the interaction in a variety of ways.[40] The heat will vaporize the water in the concrete, driving it both down into the porous concrete and up into the molten pool. The heat will also release chemically constituted water and produce carbon dioxide in any carbonates (e.g., limestone) that are present. These gases pass through the melt rather than around it. Steam passing through metallic portions of the melt will oxidize the melt and generate hydrogen gas. Carbon dioxide can be reduced to carbon monoxide.

One area of uncertainty in determining the melt temperature, gas generation rates, and melt penetration rate is the heat transfer rate from the melt to the concrete surface.[25] A second area is the rate of heat transfer through the concrete (against the steam and carbon dioxide flow). A third area of uncertainty is the heat transfer coefficient between the metal and oxide phase in the melt[41] (which are generally assumed to be separated into two different layers). A fourth area is the heatup of concrete above the pool, with the associated gas generation, concrete spallation, and reduced radiant heat loss from the melt. A fifth area is the effect of a crust or water pool above the melt (assuming no explosion is induced). In turn, the gas generation rates strongly affect the aerosol production rate.[42] The aerosol density above a melt pool determines whether thermal radiation from the pool is transmitted to structures or absorbed by the aerosol, thereby heating the containment atmosphere. Heating the atmosphere will reduce the humidity which may strongly alter aerosol shape factors, and thus the amount of suspended radionuclides. Containment pressure is also affected.

Core concrete interactions are modeled in the CORCON-MOD1 code.[43] Its performance is indicative of the amount of uncertainty in melt/concrete interactions. An extensive sensitivity study and an assessment of the code may be found in Reference 44. A fuller assessment of the modeling appears in References 45 and 46. A new version of the code is forthcoming.[47]

The best way to judge the ability of CORCON to accurately model melt-concrete interactions is by comparison to experiment. Powers dropped 205 kg of steel onto Clinch River limestone concrete and inductively heated the steel at about 400 W/kg for 1.1 and 1.5 hours in tests CC-1 and CC-2, respectively.[48] The results of the experiment melt temperatures are compared with CORCON predictions in Figures 4-6 and 4-7.[49] In spite of the modeling uncertainties discussed, CORCON

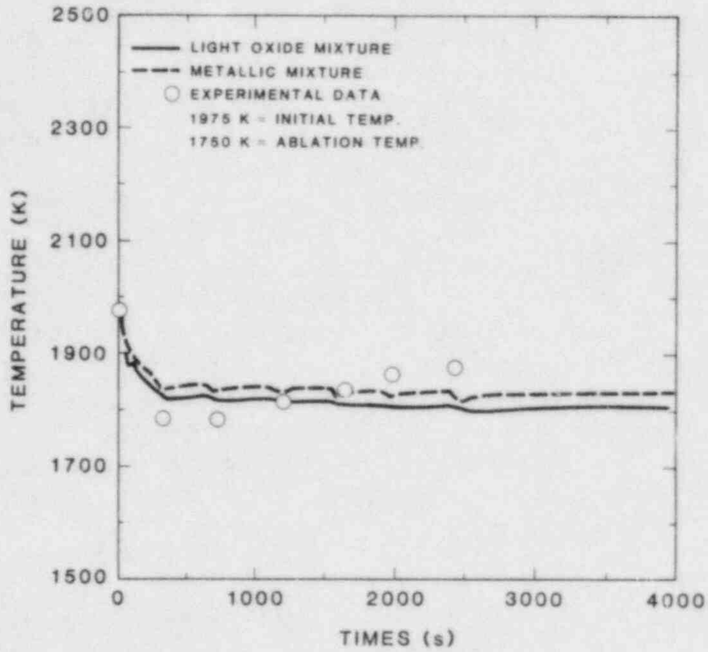

Figure 4-6. Comparison of CORCON-predicted and measured melt temperature histories for test CC-1. [49]

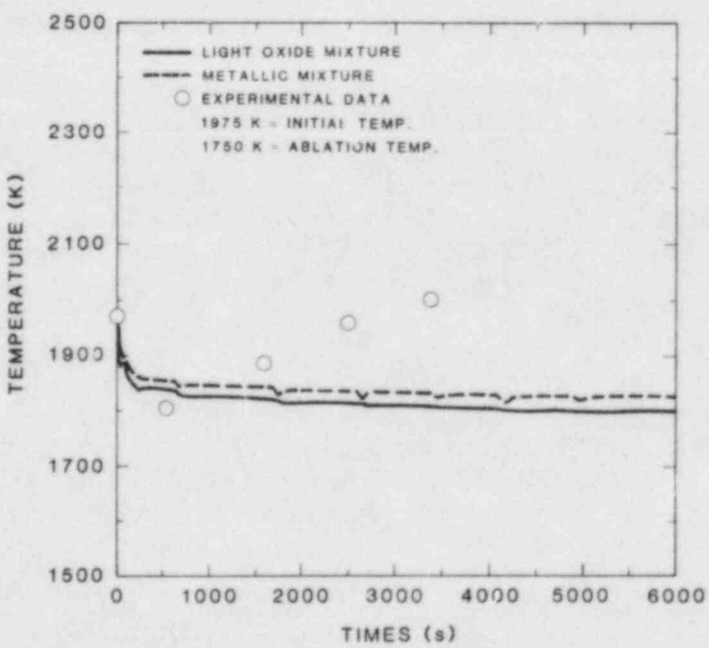

Figure 4-7. Comparison of CORCON-predicted and measured melt temperature histories for test CC-2. [49]

yields what appears to be reasonable agreement with the melt temperatures. However, melt/concrete aerosol generation rates increase exponentially with the melt temperature, and this directly affects the radiological source term at late times. So accurate modeling of the melt temperature is necessary. Of particular concern is the increase in observed temperatures with time, but no increase was predicted by CORCON. If this trend continues, CORCON would cause predictions of aerosol generation rates that are over ten times too small. So the melt temperatures predicted by CORCON are one area of uncertainty which has a strong and direct link to an important safety parameter: the radiological source term.

The major outputs of the CORCON code are the melt temperatures, the generation rates of various gases, and the axial and radial penetration of the melt into concrete. The gas-generation rates and melt temperatures determined by the CORCON code are used in the recently developed VANESA code[49-51] to determine the fission-product aerosol and inert aerosol generation rates. The aerosol average diameters and density are also determined. VANESA also estimates the amount of aerosol retention in a pool of water overlying the melt (assuming no violent interaction occurs), although these amounts are uncertain. The VANESA code is new and not many data are available for validation; an estimate of the uncertainty in its predictions is hard to make. Areas of concern which affect the radiological source term are aerosol generation rates, aerosol composition (especially refractory fission products), average aerosol diameter, and aerosol retention in an overlying pool.

4.3.6 Penetration of Hot Particulate Debris into Concrete

Recent experiments at Sandia National Laboratories have shown that timing may be an important aspect of ex-vessel particulate debris coolability.[27] The energy required to quench the melt may boil away all the quench water below the vessel. This would leave hot particulate debris on the concrete. If there is a delay in introducing water back into the debris, the hot particles may interact with the concrete. In the Sandia experiments, the heat-generating dry debris melted the concrete before the debris particles themselves melted. The concrete was then driven up into the debris by the steam and CO_2 generated in the concrete.

When water was introduced, it penetrated only to the top of the concrete and solidified the concrete there. The heat-generating debris below the crust was inaccessible to the water. There is some uncertainty as to whether this process will occur at a larger scale. If it does, it will impact the generation of gases from the concrete and the penetration potential of debris into the basemat.

4.4 Summary

A summary of the major sources of uncertainties is given in Tables 4-1 through 4-3.

Table 4-1 Major uncertainties in melt release from the pressure vessel

Uncertainty	Implications	Comments
High pressure jet:		
Direct containment heating Aerosol generation Hydrogen generation and ignition Melt dispersal	Affects early containment failure, airborne radiological source term, and ultimate debris coolability	Small experiments (20 kg) confirm existence of all features listed, but scaling to 50 000 kg yields large uncertainties
Low pressure drop:		
Discharge rate	Affects potential for ex-vessel quench and steam explosions	Uncertainties not important
After in-vessel steam explosion:		
Location of debris Size of debris	Affects ultimate debris coolability	Missiles from the explosions might fail containment or damage systems (see Section 3.5)
Secondary drops:		
Amount vs. time	Might trigger a large steam explosion in a water pool over a molten pool; may add to radiological source term at late times	Up to 30% of core may be involved

Table 4-2 Major uncertainties in ex-vessel melt initial interactions with water

Uncertainty	Implications	Comments
Initial fragmentation		
Fragment sizes Mixing Steam generation Hydrogen generation	Containment pressurization, steam explosion potential, debris coolability, burn potential	Maximum mixable amount is ~100 to ~30 000 kg, depending on choice of model
Ex-vessel steam explosion		
Probability of occurrence Energy conversion ratio Steam amount Hydrogen amount Final particle size Dispersal amount Aerosols Missiles Impulse	Containment pressurization, airborne radiological source, debris coolability, burn potential, missile impact on cooling systems, direct damage to containment structures	Much uncertainty. See Subsections 3.5, 4.2.2, and 6.3; scale effects important
Pressure spike		
Steaming rate Hydrogen rate Hot airborne particles	Containment pressurization	Occurs at vessel failure; air-particles important
Molten pool quenching		
Crust stability	Coolability, steam explosion potential	Large-scale instabilities possible

Table 4-3 Major uncertainties in ex-vessel core material long-term interactions with water and concrete

Uncertainty	Implications	Comments
Particulate debris coolability		
Particle sizes Debris thickness Stratification	Affects ultimate termination of the accident, or remelt and attack of basemat with gas generation	Primary uncertainty is in debris configuration, not in coolability modeling
Hot particulate quenching	Concrete attack by hot particle	Particles sink into molten concrete; may be inaccessible to coolant
Average quench rate Fast finger quench rate		
Effect of debris from secondary release on permanent coolability	Dryout and concrete attack	Adds particles and stratification; reduces coolability
Mass Particle size		
Molten pool under water		
Crust stability Water interactions	Ultimate termination of the accident, or remelt and attack of basemat with gas generation, water, filters, FPs, and aerosols	Induced mixing from secondary melt release possible
Melt temperature vs. time	Affects aerosol generation rate	Current code (CORCON-MOD1) fails to model increasing temperature with time as seen in experiments
Gas and vapor generation rate		
Steam rate Hydrogen rate CO rate	Affects aerosol generation rate, containment pressurization, and burn potential	Depends on both steam release from concrete and interaction with metals in the melt
Aerosol generation rate		
Composition Sizes	Dominate radiological source term at late time	Recent code (VANESA) calculates rates, but is not validated
Melt penetration rate	Affects breach of containment	
Hot particles on concrete		
Penetration rate	May yield nonquenchable debris	

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5. CONTAINMENT FLUID PHYSICS

Knowing how containment will fail is crucial in estimating the potential consequences of a given accident sequence.[1 2] The basic ways in which the containment boundary can be breached are discussed in Section 6 on containment breach and bypass.

In this section we consider one facet of the overpressure scenarios, namely, the containment loadings produced by the containment atmosphere in the course of an accident. There are two types of overpressure scenarios: those stemming from combustion events, and other noncombustion scenarios. Each of these is discussed in further detail in the following subsections.

5.1 Combustion Events

Potential sources of combustible gases within containment include radiolysis of water, corrosion of various materials in containment, metal oxidation reactions, particularly zirconium-steam and steel-steam, and the interaction of the molten core debris with concrete. The combustible gas produced is hydrogen, except for the interaction of the core debris with concrete, which can produce hydrogen, carbon monoxide, and possibly some methane and other combustible gases.

Combustion of hot, metallic aerosols (of zirconium or steel) produced by steam explosions or high-pressure melt ejection may add to the threat to containment. Direct aerosol combustion has not been considered in safety studies to date, although the process of zirconium combustion in air would liberate heat equivalent to the simultaneous reduction of steam by zirconium to hydrogen and combustion of that hydrogen in air, processes that are routinely considered in LWR safety studies because of their importance as heat sources. The reason, of course, is that it has not generally been thought possible that sufficient zirconium could be involved in such a reaction to add significantly to the threat to containment. Recent small-scale studies of the high-pressure melt ejection phenomenon suggest, though, that aerosolization of zirconium remaining in the melt upon discharge from the vessel could be quite extensive.[3] The potential importance of this phenomenon is uncertain: some argue that its potential to threaten containment has not been convincingly demonstrated. This subsection concentrates on uncertainties associated with hydrogen burning, which has been predicted to be an important contributor to risk for some plants, and has to date been more thoroughly investigated.

While radiolysis and corrosion can produce significant amounts of hydrogen, they do so over a period of days or weeks, as opposed to the period of minutes during which oxidation and core debris-concrete reactions will dominate as sources of combustible gases in a severe accident. These combustible gases would eventually be released to the containment atmosphere. Unless the containment atmosphere were made inert in some way, the mixture of combustible gases with the air in containment could result in an ignitable mixture. If the mixture ignites, it can burn in four basic ways: a diffusion flame from the hydrogen-steam jet, an ordinary deflagration if the combustible gas has had time to mix (at least partially) with the containment atmosphere, an accelerated flame, or a detonation. The resultant energy release could fail containment or some of the safety-related equipment therein. In order to estimate the contribution of such an event to the overall risk of the plant, the timing, magnitude, and nature of the overpressure must be known. This information depends upon many factors, including

- Location and rate of gaseous discharge into containment,
- Mixing of the containment atmosphere,
- Ignition, propagation, and completeness of combustion,
- Mode of combustion,
- Flame speed, and
- Heat transfer processes.

Major uncertainties exist in at least some of these areas; these uncertainties are discussed in this section.

Mass flow rates include not only those of combustible gases, but other noncondensables and steam as well, because these will also influence the combustion characteristics. The rate of gaseous discharge into containment is discussed in detail in Sections 3 and 4. The remaining topics are discussed here in detail. The implications for containment failure are discussed in Subsection 6.6; those for FP and aerosol behavior in Subsection 7.4.1.

5.1.1 Location of Gaseous Discharge

The location of the release of gases to containment is dependent on more than just the accident sequence. For instance, in loss-of-coolant sequences, the release from the primary system will be at the break location, which can vary widely. The location of the release can, in part, determine the mode of combustion. If an ignition source is present near the source of hydrogen, the gas may burn continuously in the form of a turbulent diffusion flame. If there is no local ignition source, the hydrogen may build up to very high local concentrations that could lead to a local detonation. However, if sufficient mixing mechanisms are present so that concentration gradients are minimized, the combustible gases may accumulate until some global form of combustion takes place. It is not always clear which mode of combustion would be the least desirable, especially if the location of the local burns is unknown. Thus, the location of the gaseous discharge can affect the progression and characteristics of the accident.

5.1.2 Mixing

Gases, when introduced into containment, will tend to mix uniformly throughout the accessible volume. The driving forces for this mixing include diffusion and convection, both natural and forced. Molecular diffusion is always active when concentration gradients are present, but it is slow to act compared to the convective modes of mixing. Natural convection will almost always be active in containment and would tend to mix the gases. Recent work with a containment response code seems to indicate that natural convection mixing does have a strong effect on the results predicted for hydrogen combustion.[4] Forced convection will exist locally in the vicinity of the steam and hydrogen jet exiting the reactor coolant system and globally if fans and sprays are in operation. Forced convection will induce turbulence which can greatly enhance the combustion process.

Because of the complexities of the mixing phenomena and of the containment geometry, all the available models capable of treating containment mixing are greatly simplified. There are complex codes developed specifically to predict mixing, but they do not include other phenomena such as combustion. Most of the containment thermal response computer codes use a coarse nodalization which does not include natural or forced convective mixing within a node, but instead treats them as uniformly mixed.[5 6] For these physically large nodes, the validity of the "uniformly mixed" assumption is questionable. Local concentrations (such as near a deliberate ignition source or heat transfer surface) may be grossly different from those in the bulk atmosphere.

The neglect of mixing phenomena (other than bulk intercompartmental flow) in most containment thermal response codes currently used leads to deficiencies in their predictive capabilities. In a study of a Mark III BWR containment, the predicted peak pressures and the number of burns were shown to be strongly dependent on the compartmentalization model used, as depicted in Figure 5-1.[7] Similar analyses for an ice-condenser containment also show a strong dependence on the choice for compartmentalization.[8] This dependency indicates the importance of an intelligent selection of compartments for the numerical modeling of the containment atmosphere. While large differences in the predicted results can be found with different degrees of compartmentalization, a sufficient understanding of this effect now exists so that little uncertainty should be introduced when an intelligent selection for compartmentalization is employed.

5.1.3 Ignition, Propagation, and Combustion Completeness

The predicted combustion overpressure depends strongly on the amount of combustible gas predicted to be burned, and hence on the specific criteria used to predict ignition, to predict flame propagation from the initial compartment to adjacent compartments, and to predict completeness of combustion.

The ignition of hydrogen-air mixtures requires very small amounts of energy, of the order of millijoules.[9] Small static electric sparks can set off combustible hydrogen-air mixtures. We have little information on when such an ignition might occur. If flame igniters are used, the picture is clearer. The main variables are hydrogen mole fraction and igniter location. The mixture can be made inert if the

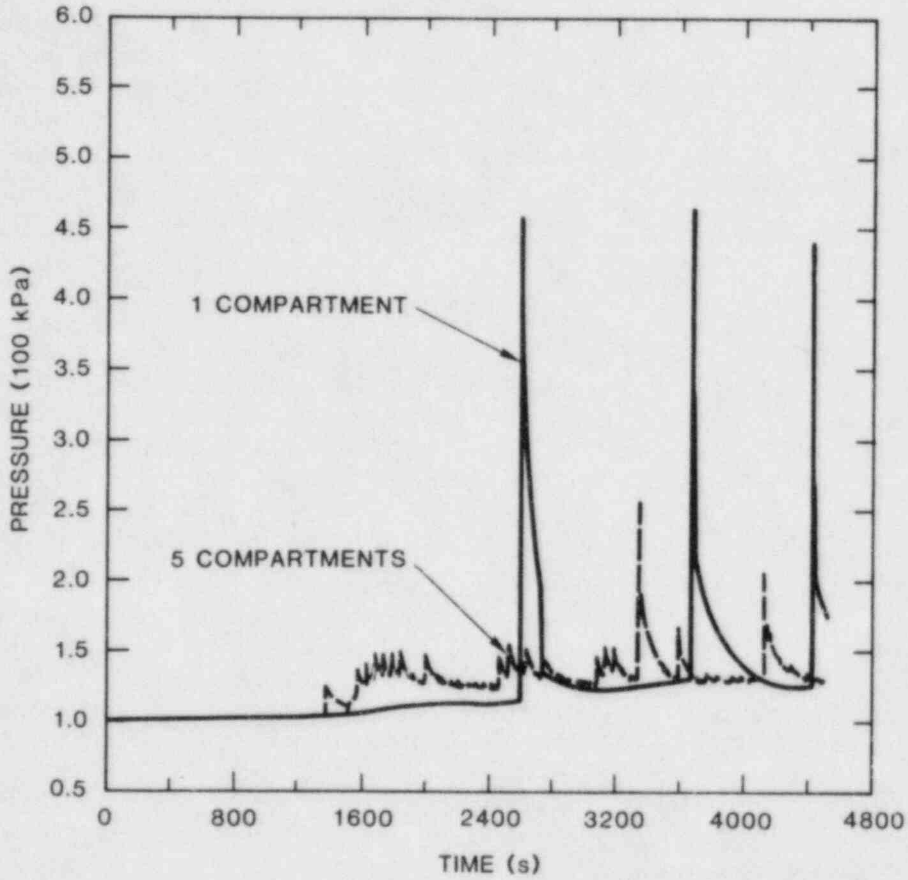

Figure 5-1 Effect of compartmentalization on peak pressure prediction.[7]

oxygen mole fraction is reduced enough, or if the steam mole fraction is increased sufficiently.[10 11] Slight variations in these limits of inertness could have a large influence on the course of the accident. Although a range of values has been established by experimental results, the values are not definitive enough when sensitivity to the parameters is considered. The effects of initial gas temperature and pressure are negligible in the range of interest. If the mixture is not made inert, ignition by glow plug or spark igniters is fairly certain when the hydrogen mole fraction reaches 10%, and is likely down to 8% when igniters are located at the bottom of the mixture such that the flames can propagate upward.[12] Figure 5-2 illustrates the difference in predicted pressures from burns based on ignition criteria of 8% and 10%. Even though the difference in peak pressure may seem small, it can determine whether containments with low failure thresholds will survive. In predictions by models using homogeneous atmospheres, a major uncertainty arises when the ignition test is applied to the homogeneous conditions within the compartment, rather than to the unknown local conditions that would actually exist at the igniter location.

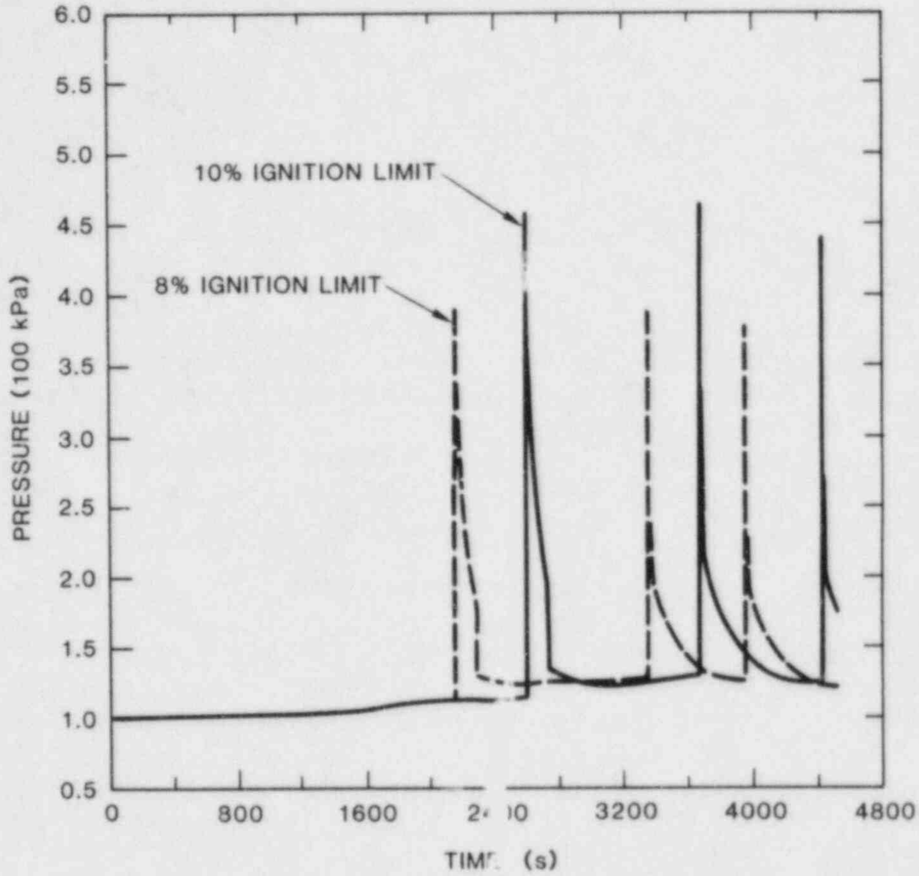

Figure 5-2 Effect of ignition limit on peak pressure prediction.[7]

The mass of combustible gas burned in any multicompartment model strongly depends on whether the burn propagates from the initial compartment (where ignition occurred) to adjacent compartments. For models in which a burn in a compartment is allowed only if the ignition criterion is met,[13] there will be only single-compartment burns, unless two compartments simultaneously meet the criterion. Physically it is expected that once a burn starts, it will propagate into adjacent volumes if the conditions there meet some propagation criterion generally less strict than the ignition criterion.[10 11] Propagation between compartments has been found to be dependent upon the physical location of the neighboring compartment relative to the compartment containing the burn and upon the concentrations of combustible gases in the neighboring compartment.[5 11] Here mixing strongly affects combustion. If mixing is rapid, such that concentrations are nearly the same throughout containment, burns will propagate, giving a near-global burn and a high peak pressure.[5 14] If mixing is slow, the burns are likely to take place in one or only a few compartments, giving lower pressure rises.[7]

Experiments performed on mixtures under marginally-combustible conditions, such as low hydrogen concentrations or high steam concentrations (see Figures 5-3 and 5-4 for details), show that not all of the available hydrogen will burn under such conditions, because lower pressure rises are observed than would be expected from complete combustion.[15 16] However, these results are for quiescent conditions and simple geometries. Containment geometries are far from simple, and intense turbulence would be expected to be present in the atmosphere, which might render the combustion process nearly complete.

The consideration of the combustion of hydrogen/carbon-monoxide mixtures should not add any major new uncertainties. The flammability limits of such mixtures in air have been measured, and follows fairly closely to the empirical Le Chatelier equation for the flammability limits of combustible gas mixtures.[17] The flammability limits of such mixtures in an air/steam atmosphere are less well known. Procedures for computing these flammability limits that should be reasonably accurate are given in Coward & Jones[17].

Gas mixtures containing combustible gases outside their flammability limits can be burned in regions near ignition sources, but the flames will not propagate far from the ignition source. This is the principle behind the hydrogen recombiners used in containments.

There are several possible effects of aerosols on hydrogen combustion. These include catalysis, radiative absorption, photochemical and other solid-state reactions on aerosol particles, cooling of the reaction via the thermal capacity of the aerosol, and localized modification of the burning gas mixture by aerosol decomposition (e.g., thermolysis of carbonate aerosols to produce CO_2). Few of these effects have been investigated experimentally or considered in analyses.

Preliminary experiments at Sandia National Laboratories suggest that aerosols may have a significant influence on the course of hydrogen burning.[18] In one experiment, large concentrations (~100 g/m^3) of iron oxide (Fe_2O_3) aerosol appeared to reduce the peak pressure attained during hydrogen combustion. No such effect was observed in an experiment with similar concentrations of aluminum oxide aerosols. The phenomena responsible for these effects, and the significance of the uncertainty they introduce, have yet to be evaluated. The effects of hydrogen burns on FP and aerosol behavior are discussed in Subsection 7.4.1.

5.1.4 Flame Speed

The flame speed during combustion will influence the associated pressure rise. Figure 5-5 represents the effect of flame speed on peak pressure for a given volume.[7] If the flame speed is rapid, there will be little time for heat transfer from the gas to the walls or spray droplets, and the combustion will be nearly adiabatic. Laminar flame speeds for hydrogen are slow, a few meters per second. Combustion experiments have shown, however, that the flame speed changes dramatically when the mixture is agitated, as illustrated in Figure 5-6.[7] Considering the probable effects of turbulence and flame

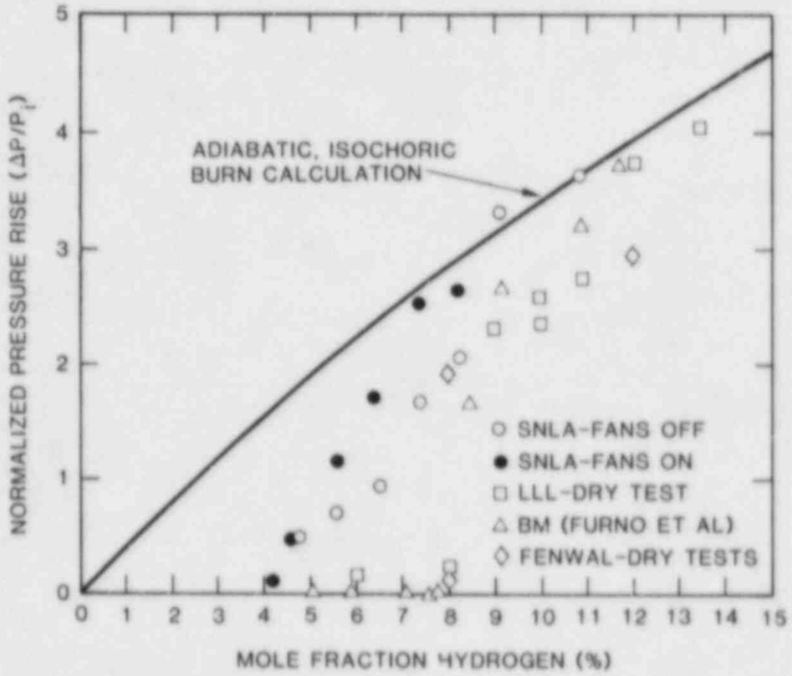

Figure 5-3 Effects of initial hydrogen concentration on completeness of combustion.[15]

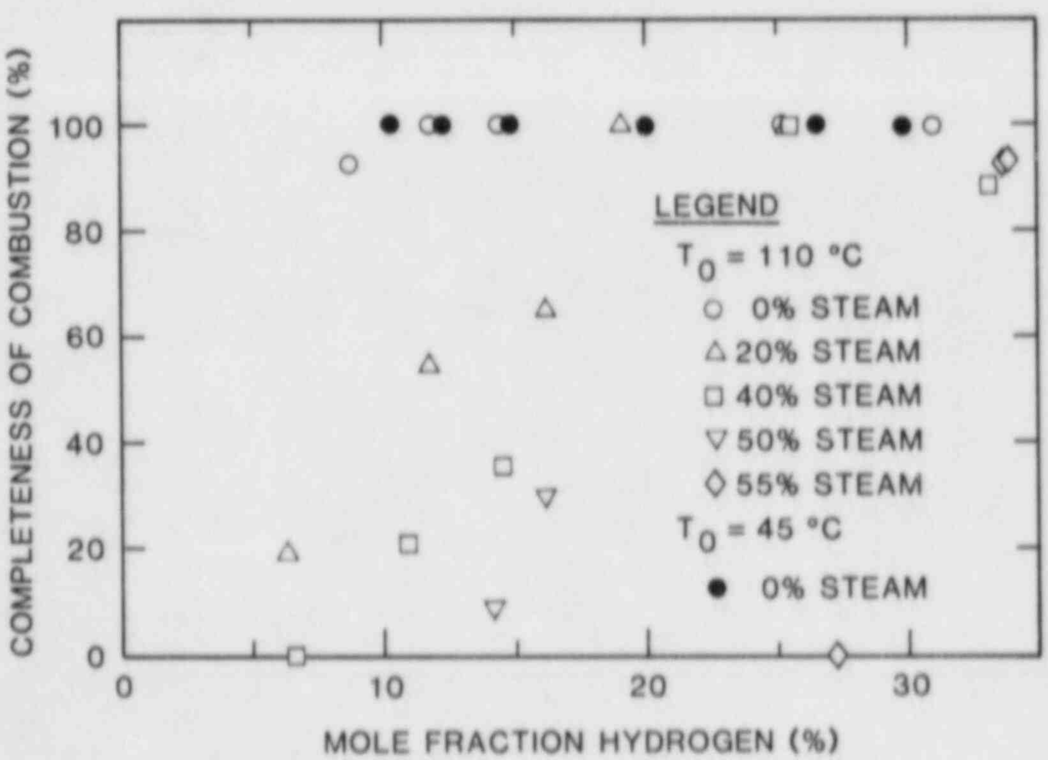

Figure 5-4 Effects of steam concentration on completeness of combustion.[16]

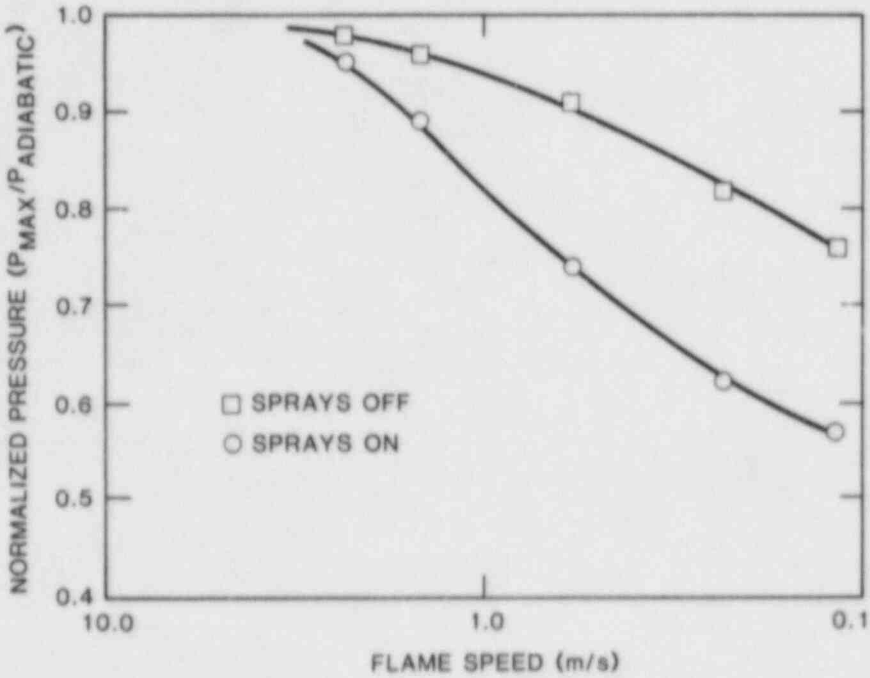

Figure 5-5 Calculated effect of flame speed on peak pressure.[7]

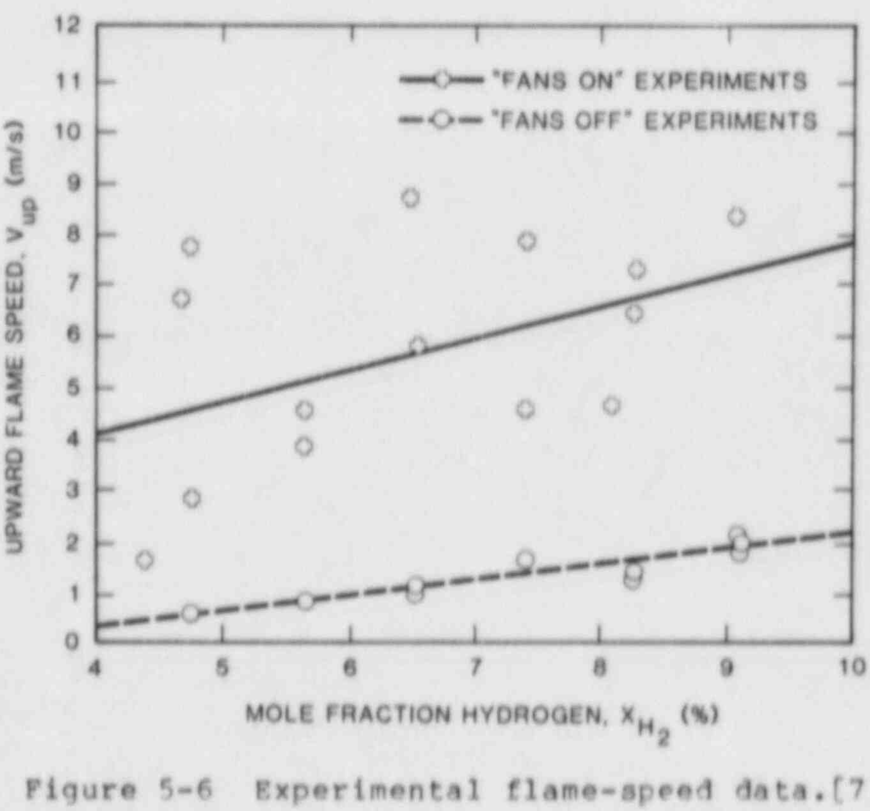

Figure 5-6 Experimental flame-speed data.[7]

acceleration due to flame folding around obstacles within containment, the effective flame speed may be much higher. Considerable uncertainty exists in modeling the flame speed for calculational purposes. A major portion of this uncertainty is generally attributed to inadequate modeling of the effects of turbulence.

5.1.5 Heat Transfer

In the process of combustion, the heat liberated from the chemical reaction is deposited with the reaction products. This heated gas produces an atmospheric pressure increase that can threaten containment. Thus, the more heat that can be transferred away from the atmosphere, the lower the peak pressure will be. There are three modes of heat transfer which can be of importance in this situation: radiation, convection, and evaporation/condensation.

Radiative heat transfer from hot steam, or other potential radiators such as carbon dioxide, dominates the heat transfer from very hot combustion gases when there are no containment sprays in operation. The theory of radiative heat transfer from high-temperature molecular radiators is well developed.[19] The main error in most containment thermal analysis codes in computing radiation heat transfer is a failure to apply present-day developments in this area. Some of the more recently developed codes do utilize this information and calculate the radiative heat transfer quite accurately.[5] In these codes, the major uncertainties in radiation heat transfer to the walls arise from the lack of knowledge of the distribution of temperature and steam in the compartments.

Heat transfer by convection results when the hot gas comes into contact with cooler bodies within containment. These mechanisms are a factor even in noncombustion scenarios and are discussed in Subsection 5.2.

If containment sprays are operating during a burn, heat transfer to the spray drops will dominate the heat loss from the gas. Figure 5-5 shows that for realistic flame speeds the sprays can reduce the peak pressure resulting from a burn by as much as 25%. Models for this effect exist in several containment thermal response codes but are simplistic, considering only convective heat transfer to isolated drops falling through a quiescent atmosphere. The effects of drop agglomeration, gas motion caused by the burn, the interaction between drops, and the radiative heat transfer process are not considered. The effects of aerosols on radiative heat transfer are likewise not considered, though their potential to block heat transfer has been observed.[14]

5.1.6 Detonations

In the event of a detonation, shock waves will be generated, and pressure spikes will be produced, giving dynamic loads on containment in addition to static loads. It was formerly believed that detonations were impossible in hydrogen-air mixtures if the hydrogen mole fraction was below about 18%, the "detonation limit." It is now known that detonations are possible in mixtures of as low as 14% hydrogen, and possibly lower.[20] For a marginal mixture, the possibility of propagating a detonation down a tube, or other geometry, increases with increasing geometric size. There is a reasonably well-developed ability to determine the dynamic loads that would be caused by a detonation. The major uncertainties with regard to detonation are whether a detonable mixture will be present, what composition and volume it might have, and if the mixture will detonate rather than just deflagrate. A detonation can be initiated directly by an energetic source such as a strong spark, or a deflagration can accelerate and transition into a detonation.

5.1.7 Highly Accelerated Burns

For certain geometries, experiments have shown that ordinary deflagrations in confined geometries or in the presence of obstacles can accelerate to very high flame speeds and even undergo a transition to detonation. In this case, pressure rises are so fast that the assumption of a quasi-static containment loading is no longer valid. Rather, a complete dynamic analysis, similar to that required for the treatment of detonations, is required.

Experiments to date indicate that the mechanisms governing flame acceleration may represent a precarious balance of positive and negative factors associated with flame folding and turbulence. Two positive factors that lead to an increase in burning rate are the increase in flame area due to folding, and the increase in the local burning velocity of the folds associated with fine-scale turbulence. The increase in flame area is a result of the gas flow ahead of the flame being greatly perturbed by the presence of obstacles. In the absence of negative factors, the volumetric burning rate would continue to increase until a transition to detonation occurred. The negative factors that lead to a decrease in the burning rate are reaction quenching due to excessive flame stretching, and rapid cooling due to turbulent mixing. In addition, venting from a confined geometry tends to inhibit flame acceleration by greatly reducing the convective motion of the gas ahead of the flame. If the negative factors are strong enough, the flame may be quenched; if they are of intermediate strength, then a steady flame can be produced which burns much faster than ordinary deflagrations, but does not undergo a transition to detonation.

The lack of understanding of all the physical phenomena involved in these processes gives rise to one of the larger uncertainties associated with combustion, namely, will an ordinary deflagration accelerate or even undergo a transition to detonation?

5.1.8 Diffusion Flames

When hydrogen leaves the primary system, it can burn in the form of a turbulent diffusion flame. Ignition may occur from an accidental source, or from autoignition if the jet temperature is above approximately 875 K (600°C). Similarly, the hydrogen plume coming from the suppression pool in a BWR may burn in this manner. The continuous burning of hydrogen in this fashion prevents the buildup of hydrogen and reduces the danger of failing containment by overpressurization during some global form of combustion. However, the very hot diffusion flame will subject equipment in or near it to high heat loads, which may cause subsequent failure of critical engineered safety features (ESFs).

Not yet discussed to this point is the impact of combustion on systems within containment. During a combustion event, severe thermal and pressure loads will be imparted to equipment located within containment. It is conceivable that a burn could occur that would not threaten the containment structurally, but that would render some critical containment systems inoperative. Currently, these effects are not included in PRAs, but they could play a very important role in determining the overall risk of a plant. Research is underway in this area, in a program entitled "Hydrogen Burn Survivability."[21]

5.2 Non-Combustion Overpressure

The previous subsection dealt with the overpressure scenario resulting from combustion events. Other overpressure scenarios are possible in which the containment is threatened structurally by overpressure. This type of overpressure can be classified into three basic types, those due to

- Slow gas accumulation,
- Rapid steam generation, and
- Direct heating of the atmosphere.

In the case of slow gas accumulation, the containment heat removal systems have been rendered inoperable for some reason, so that the only heat sinks available to containment are the thermal masses therein. The rate of pressurization and its associated uncertainties thus depend solely upon the discharge rates of steam and noncondensables into containment and the rate of heat and mass transfer from the atmosphere to the passive heat sinks.

The rapid steam generation category covers the situation in which steam is introduced into containment at a very high rate as a result of depressurization of the primary system, and the interactions of the core debris and water in the reactor cavity after reactor vessel head failure. Conceptually, although improbably, this input of steam can be so rapid that the capacity of the containment heat removal systems is exceeded, and the containment is failed. Obviously the performance of the containment heat removal systems at off-design conditions will have an effect here, but the uncertainty in their performance is probably overshadowed by the uncertainty in the steam source term. This source term uncertainty is discussed in detail in Sections 3 and 4.

Direct heating of the atmosphere by particles and aerosols can produce overpressure by heating of the noncondensable gases. To date, PRAs have typically ignored these effects in the thermal-hydraulic calculations for the containment atmosphere.

From the preceding discussion of the types of noncombustion overpressure, it can be seen that the performance of the containment heat-removal systems is crucial to containment integrity. In general there are six systems for containment heat removal:

- Containment sprays
- Fan coolers
- Sump or suppression pool residual-heat-removal systems
- Suppression pool
- Ice condenser
- Passive heat sinks

These are discussed in detail in the following subsections.

5.2.1 Containment Sprays

The containment spray system can operate in two modes: injection or recirculation. In the injection mode, water is taken from a storage tank outside of containment. It is pumped into containment and through the spray nozzles at the top of containment, where the water is injected as small drops which then fall downward through the containment atmosphere. When the water in the storage tank is depleted, the intake for the spray pumps is switched to draw water that has accumulated in the containment sump. In the recirculation mode, the water is sometimes cooled by pumping it through a heat exchanger before injecting it through the spray nozzles.

As the droplets fall through the warmer containment atmosphere, they can remove sensible as well as latent heat. The removal rate will be a function of the temperature, droplet diameter, fall time, and mass flow rate of the drops, as well as the temperature and composition of the atmosphere. Accurate knowledge of the heat removal and condensation rates is important primarily when considering "pressure spikes" (see Section 4). In order to predict the impact of the sprays on containment pressure, a heat and mass transfer correlation must be developed to describe this process. Such a model has been developed for spherical drops in a stationary atmosphere with constant and homogenenous properties.[5 6] However, the real situation consists of a nonstationary atmosphere, with time- and space-dependent properties, in which large numbers of drops are falling and most likely agglomerating. The "constant and homogeneous property atmosphere" assumption probably produces relatively small errors for most situations, except for rapid events such as combustion. The error in this situation has not been quantified. The neglect of agglomeration of drops could lead to an overprediction of the heat removal capability of the sprays, because larger drops have a smaller surface-to-volume ratio, larger terminal velocities, and correspondingly shorter residence times in containment. However, this effect has been estimated to be on the order of 10% with respect to temperature-pressure response.[22]

The temperature of the spray droplets is either the temperature of the water in the storage tank during injection or the temperature of the sump water (after passing through the spray heat exchanger for recirculation). Variations in the storage tank water temperature would be observed as a function of the time of year. In the recirculation mode, the heat removed by the heat exchanger will be a function of the sump water temperature, the temperature of the water on the secondary side of the heat exchanger, and the flow rates. The most important parameter here would be the secondary water temperature, which would have seasonal variations. These fluctuations would affect heat removal capabilities and should be considered in the overall uncertainty but are probably not as important as other factors.

5.2.2 Fan Coolers

Some containment designs utilize fan coolers. A fan cooler uses a recirculating forced flow of the containment atmosphere past cooling coils in an effort to condense the steam and thus maintain an acceptable containment pressure. This type of operation is well described by experimental data. However, the air also passes through a series of filters in the fan cooler units. During a severe accident, the atmosphere is likely to be laden with aerosols, which could lessen the effectiveness of the cooling coils and clog the filters, reducing the cooling capability of the system.

Part of the fan cooler system includes ducting for gathering the intake flow or distributing the discharge flow. When these units are operating, considerable turbulence will be generated by the flow through these ducts. If combustion propagates into the ducts, this confined geometry along with the turbulence could provide a mechanism for the transition from a deflagration to a detonation. Such a detonation could render the fan cooler system inoperative from then on. Thus, the major uncertainties with respect to fan cooler performance are whether the unit will survive and continue to perform as designed in the presence of high atmospheric aerosol loadings and/or combustion.

5.2.3 Residual-Heat-Removal (RHR) Systems

Some containment designs employ RHR systems that circulate liquid water from containment through a heat exchanger and back into containment as a means of heat removal. This is another system that is dependent upon the inlet conditions on the secondary side of the heat exchanger, but it is a process that is well defined and unlikely to give rise to dominant uncertainties. The largest source of uncertainty about the performance of this system would be that associated with the failure criterion for the pumps. Present PRAs only consider pump failures due to depletion of the intake water supply. Other possible reasons for pump failure include

- Debris entrainment in the intake water following an energetic reaction between the core debris and water in the reactor cavity,

- Overheating of the pump room after a failure in the heating, ventilating, and air-conditioning (HVAC) system, and

- Cavitation within the pump due to the intake water being at or near saturation conditions.

5.2.4 Suppression Pool

BWR containment design utilizes a suppression pool to connect two containment compartments. The original design called for the blowdown from a design-basis LOCA to be forced through the suppression pool that would condense the steam, thus maintaining an acceptable containment pressure. Reduced-scale experiments by the vendor have yielded a good data base for this type of accident. However, flow rates much larger than those from a design-basis LOCA are obtainable in a degraded-core accident that produces combustion in one compartment. How the pool will behave under these conditions is unknown. Questions arise relating to pool integrity and/or steam scrubbing under such conditions and as yet remain unanswered. This gives rise to large uncertainties with respect to the progression of accidents involving large combustions.

In BWRs, the primary system safety relief valves exhaust, via piping, into the suppression pool. Operating experience and subsequent vendor testing have shown that severe loadings can be produced on the suppression pool walls when steam and air are discharged through the suppression pool.[23 24] These loads arise from condensation instabilities and seem to be most dependent upon pool temperature and design of the discharge piping. Although steps have been taken to correct this problem, some uncertainty still exists in relation to pool integrity during actual accident conditions which would probably include elevated pool temperatures.

5.2.5 Ice Condenser

Still another containment design utilizes a device, known as an ice condenser, between two separate containment volumes. Recirculation fans force flow from one compartment, through the ice condenser, into the second compartment, and back to the first. There are one-way doors on the ice condenser and fans to prevent backflow through these devices. The ice condenser consists of many containers (known as baskets) of ice over and through which the flow is forced. The intent of the design is to condense the steam in the atmosphere onto the ice, thus scrubbing out the steam and maintaining an acceptable containment pressure.

Like the suppression pool, the ice condenser was designed to handle the design-basis LOCA. However, postulated severe accidents for ice-condenser containment designs often involve an extended release of steam and hydrogen and subsequent ignition of a combustible atmosphere. These combustion scenarios produce much larger flow rates through the ice condenser than designed for,[8] and the ice-condenser performance is not well characterized for these conditions.

Uncertainties with respect to steam scrubbing capabilities and time to completely melt the ice can significantly alter the progression of the accident and the associated risk. These uncertainties are difficult to quantify, because only limited experimental data are available even for the design conditions,[25] and none is available for off-design conditions. The complex geometry of the ice condenser makes it difficult to justify the extrapolation of any existing experimental data of similar nature to the problem at hand. There is also uncertainty associated with the survivability of the intercompartmental fans and one-way doors during those combustion events. Large pressure differences between the compartments would occur during combustion that could render the fans or one-way doors inoperative. All of these uncertainties lead to a large uncertainty in the outcome of any accident involving large-scale combustion in an ice-condenser equipped containment building.[8]

5.2.6 Passive Heat Sinks

As discussed earlier, if all the designed heat-removal systems fail, the only other source of heat removal is the passive heat sinks of the containment structure itself. Any containment has a large number of passive heat sinks associated with it because of its massive structure. The rate of heat removal by these masses is initially limited by convective/condensation heat transfer from the atmosphere to the surfaces. Later, as the masses heat, conduction in the bodies will dominate. (Radiation from the atmosphere has negligible effects in these noncombustion scenarios.)

To predict the heat-removal rate from containment, the containment geometry and construction must be well known. More importantly, convective/condensation heat transfer coefficients are needed, and it is these that are the most difficult to predict. Predicting the heat transfer rates for free convection condensation in the presence of noncondensables in simple geometries is complex enough by itself, but the containment geometry is by no means simple and complicates the problem even further.

This uncertainty will have an impact for those accident scenarios in which all containment cooling is lost, and the resultant steam partial pressure (possibly in connection with some noncondensable partial pressure) rises high enough to fail containment. Specifically, the time to containment failure will be altered. The rate of condensation can also impact the rate of removal of fission products from containment, as discussed in Section 7 of this report.

5.3 Aerosols

In the past, airborne aerosols have generally not been treated in calculations of the thermal-hydraulic behavior of the containment atmosphere. Rather, they were only considered in the calculation of fission-product transport and release. But the aerosols do have an effect on the thermal-hydraulic response, and decoupling these effects can lead to errors in the predicted results, especially in those accident scenarios where the only effective aerosol-removal mechanisms are agglomeration and gravitational settling (e.g., TMLB'). Recently the need to couple the thermal-hydraulic and aerosol behavior has been recognized and has been incorporated into the containment atmosphere analysis code CONTAIN.[26] Uncertainties in the thermal-hydraulic response that are due to aerosol phenomenology are discussed here and in Subsection 5.1. Uncertainties in the aerosol source term as related to fission-product transport are discussed in Section 7.

The solid aerosols in the atmosphere are generated from the molten core debris and potentially have a portion of the decay heat associated with them. This leads to a volumetric heat source in the atmosphere itself. Although this amount of heat in itself is small, it can have a significant effect on the pressure-temperature response when operating at or near saturation conditions, because solid aerosols serve as potential condensation sites in a saturated atmosphere. These solid aerosols will also plate out on the various heat sinks over time, carrying condensate out of the atmosphere and depositing it on the heat transfer surfaces. If condensate does not wash these aerosols off, the heat source associated with them will alter the surface temperature of the heat sink, which will affect the rate of heat removal by the heat sink. The solid aerosols will also settle out of the atmosphere onto various components of the ESFs (e.g., ice condenser, fan cooler filters, electrical equipment). How these ESFs will perform under these high aerosol loadings is unknown.

Another type of aerosol that must be considered is small droplets of water that are apt to be present in the containment atmosphere. These will be created during blowdown of the primary system, interactions between the hot core debris and water in the reactor cavity, and condensation of water vapor out of a supersaturated atmosphere. These aerosols can significantly affect the thermal-hydraulic response of the containment atmosphere. These small droplets represent a heat source/sink which tends to dampen out fast pressure-temperature excursions. These, with the solid aerosols, also change the properties (e.g., heat capacity, thermal radiative transmittance) of the atmosphere. This effect would be most noticeable during combustion events when large amounts of energy are deposited over short periods of time.

The largest uncertainty in this area is most likely that associated with the performance of the engineered safety features. If the aerosol effects are detrimental enough, then the entire progression of events during the scenario could be changed and more serious results might be realized than would be expected without considering the aerosol effects.

5.4 Liquid Levels in Containment

The successful operation of certain ESFs (e.g., the recirculation mode of containment sprays) requires a source of liquid water within containment. For a PWR, this source is the containment sump, and in a BWR it is the suppression pool. The quantity and thermodynamic state of this water can affect the performance of the ESF. For instance, if insufficient water is available, or if the water is not cooled to an acceptable temperature, the pumps drawing on this water may experience cavitation and fail. If too much water accumulates in containment, other questions arise as to the performance of an ESF that may have components immersed in water. The amount of water in the reactor cavity is also important with regard to core debris-water interactions, as discussed in Section 4.

The location, level, and temperature of the liquid water in containment depend strongly upon containment geometry and previous ESF operation. Some uncertainty arises in this area because of uncertainty in the containment geometry, namely, the capacities of different compartments within containment and possible liquid flow paths between compartments. However, this uncertainty can be resolved by a detailed examination of the containment and writing the appropriately descriptive numerical models for use in calculations.

The largest uncertainty in this area can be attributed to our lack of understanding of how the condensate will be distributed on the heat transfer surfaces. Large areas exist within a containment (e.g., floor grates, support structures) which will be covered by a liquid film of condensate. No good models or integrated experiments exist which reflect on how this condensate film behaves; that is, how much condensate will remain on the surface and how much will drain off. Thus, a good description of where the condensate resides is not possible. This impacts upon the subsequent heat transfer processes to these surfaces which can be of large importance during rapid events such as combustion or steam spikes.

5.5 Summary

Table 5-1 summarizes the important uncertainties discussed in this chapter.

Table 5-1 Summary of important uncertainties related to containment fluid physics

Uncertainty	Implications
Combustion Processes	
Location and magnitude of gaseous discharge	Affects rate of pressure increase for steam overpressure and peak pressures from hydrogen burn. Also affects mode of combustion (e.g., diffusion flame, deflagration)
Mixing	Affects timing of burns, propagation between compartments, and mode of combustion.
Ignition propagation and combustion completeness	Affects timing of burns, quantities available to participate in burns, and amount actually burned. A large part of this uncertainty arises from the effects of turbulence in the atmosphere.
Flame speed	Determines which heat transfer mechanisms will be important during burn.
Detonations	Detonations are relatively well understood; major uncertainty here is related to predicting detonable concentrations and sufficiently intense ignition sources.
Highly accelerated burns	The effects of confinement and obstacles lead to largest uncertainty here. Ordinary deflagration could accelerate to the point that the quasi-static treatment of containment loading would be inadequate.
Heat transfer processes	
Free/forced convection	Governs maximum heat removal rate when all containment cooling is lost.
Radiation	Can alter peak pressure from burn and peak surface temperatures.
Evaporation/condensation	Depends strongly upon atmospheric conditions. This mechanism not modeled well in the presence of noncondensables. Greatest impact on those scenarios in which all active containment heat removal mechanisms are unavailable.

Table 5-1 (continued)

Uncertainty	Implications
Engineered safety systems	
Sprays	Sprays could impact thermal radiation processes during burn.
Fan coolers	Performance could degrade in presence of high aerosol concentrations.
Suppression pool	Larger-than-design-basis flow rates could pass through pool with little or no steam scrubbing.
Ice condenser	Larger-than-design-basis flow rates could pass through condenser with little or no steam condensation. Questions arise about performance in presence of high aerosol concentrations.
Passive heat sinks	Characterization of available passive heat sinks in containment impacts heat removal rate when all containment cooling is lost. Plate out of aerosols could alter surface temperature of heat sinks. Largest uncertainty is associated with actual vs. predicted atmospheric conditions and modeling of condensation heat transfer.
Aerosols	
Heating of atmosphere	Represents direct source of heat to containment atmosphere. Can be important when atmosphere is near saturation.
Alteration of atmospheric properties in containment	Can alter thermal radiative transport properties, heat capacity of atmosphere, burn characteristics, etc.
Impact on equipment	Could lead to equipment failure or degraded performance.

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26. K. D. Bergeron et al, "An Overview of the CONTAIN Code for Severe Accident Analysis," Vol. 3, pp 701-705 in Proceedings, USNRC Eleventh Water Reactor Safety Research Information Meeting, USNRC Conference Proceeding NUREG/CP-0048, Gaithersburg, Maryland, October 1983.

6. CONTAINMENT BREACH AND BYPASS

This section discusses the uncertainties associated with the various modes of breach or bypass of containment. The word "mode", as used here, encompasses considerations of cause, timing, structural mechanics, location, and the flow path for discharge of material through the containment barrier. The subsections are arranged by cause as follows:

- 6.1 Containment Bypass
- 6.2 External Events
- 6.3 In-Vessel Steam Explosions
- 6.4 Containment-Isolation Failure
- 6.5 Overpressure Attributable to Gas Accumulation
- 6.6 Overpressure Attributable to Combustion Events
- 6.7 Temperature-Induced Failure
- 6.8 Basemat Melt-Through

Direct containment failure because of vacuum, pipe whip, jet impingement, or internally generated missiles (except in the case of an in-vessel steam explosion) should be precluded by conventional design practice and will not be considered here. Also, this section does not consider unique failure modes that could be introduced by future mitigation features, such as filtered venting systems.

Two basic types of uncertainty are discussed in this section: (1) uncertainty with respect to whether the containment will be breached or bypassed and, if so, with what probability, and (2) uncertainty regarding the degree of containment breach or bypass. If containment breach or bypass can be precluded, the offsite radiological consequences will be negligible. On the other hand, if containment breach or bypass does occur, the offsite radiological consequences will depend strongly on the nature of the breach or bypass--its cause, timing, location, and the flow path through the containment barrier. Table 6-1 indicates how these general breach or bypass characteristics could influence the relative radiological consequences.

6.1 Containment Bypass

Containment bypass involves the discharge of materials (steam, hydrogen, fission-product gases, and aerosols) from the primary coolant system through the containment barrier without passing through the containment atmosphere. As typically discussed in PRAs, containment bypass is the result of an initiating event. Containment bypass at later stages of the accident is also conceivable. Two examples of containment bypass will be used to illustrate some of the uncertainties in the pathway which may lead to corresponding uncertainties in radiological consequences. Uncertainty regarding the frequencies of initiating events is discussed in Subsection 2.2.1.

6.1.1 PWR V-Sequence LOCA

The V-sequence interfacing-system LOCA is caused by the failure of the valves (in series) that isolate the low-pressure injection system

Table 6-1 Effect of breach or bypass characteristics on offsite radiological consequences

Breach or Bypass Characteristic	Effect on Offsite Radiological Consequences
Cause	For containment breach or bypass modes which involve forced expulsion of material through the containment barrier, offsite radiological consequences will tend to increase with the magnitude of the driving force.
Timing	An earlier containment breach provides more time for radionuclide release to the environment, less time for removal in the reactor coolant system or containment, and less time for sheltering and evacuation.
Location	A containment breach or bypass which provides a direct pathway for radionuclide transport to the outside atmosphere would yield higher offsite radiological consequences than an otherwise equivalent breach or bypass which resulted in radionuclide holdup and plate-out in surrounding buildings.
Flow Path	Larger flow resistances give smaller leakage rates. Wet flow paths should remove more radionuclides than dry flow paths. Intermediate-size openings may arrest containment-pressure buildup which could otherwise lead to larger openings.

(LPIS) in the auxiliary building from the high-pressure reactor-coolant system inside containment. The postulated isolation-valve failures would result in flow from the reactor-coolant system into the LPIS, initially through the LPIS relief valves.* Should the backflow through the failed valves exceed the capacity of the LPIS relief valves, breach of the LPIS pressure boundary due to overpressure or dynamic loading beyond the design basis could occur. Thus, a V-sequence LOCA could result in the discharge of primary coolant directly to the auxiliary building.*

Core damage in a V-sequence LOCA could, in general, only be prevented if the human operators acted promptly to isolate the discharge flow path by closing the appropriate motor-operated valve. If this valve were not closed promptly, the motor for the valve operator could overheat because it would be located in the vicinity of the postulated

*In some plants flow from the LPIS relief valves is returned via a collection header to containment.

break in the auxiliary building. The emergency core cooling (ECC) pump motors could also fail due to steam flooding unless prompt action were taken. An astute operator might diagnose the V-sequence LOCA by its distinct signature--no initial change in containment parameters but sharp increases in the auxiliary building pressure, temperature, and radiation levels. The larger the flow area, the less time would be available for the operator to isolate the break.

The RSS assumed that a V-sequence LOCA would "almost surely" lead to total LPIS failure.[1] The RSS postulated a 15-cm (6-inch) diameter break in the LPIS pressure boundary as a result of the LPIS isolation-valve failures. With a total LPIS failure, even if ECC injection from the refueling water storage tank were to function properly, switchover to ECC recirculation would not be possible because the LPIS pumps are required to take suction from the containment sump which would be empty.

Depending on the nature of the LPIS isolation-valve failures and the details of the ECCS design and layout, it is conceivable that backflow through the LPIS isolation valves could be accommodated by the LPIS relief valves. If not, small breaks in interconnecting process or instrument lines could well occur before a larger (e.g., 6-inch diameter as per the RSS) process-line break. It is also conceivable that only one LPIS train, the one in which the break occurred, would fail. To test such hypotheses would require detailed design information and thermal-hydraulic and fracture analyses. Also, the possible effects of steam flooding on the operability of ECCS pump motors and valve operators in the auxiliary building would have to be assessed.

Given a V-sequence LOCA that cannot be isolated, the area and point of discharge for primary coolant flow into the auxiliary building would determine the rate of primary coolant loss, the time available for evacuation and sheltering, and the degree of in-plant holdup or plate-out of fission products.

At some point in a severe V-sequence LOCA, certainly by the time of vessel breach, materials would be discharged to the containment atmosphere. The size of the opening into the auxiliary building would then influence the degree of holdup and plate-out of radionuclides in the containment atmosphere. Containment overpressurization due to accumulation of gases or combustion events would still be possible for a small enough opening (see Subsections 6.5 and 6.6). Also, containment breach due to in-vessel steam explosions, high containment temperature, or basemat melt-through cannot be precluded in a V-sequence LOCA.

6.1.2 PWR Steam-Generator-Tube Ruptures

Steam-generator-tube ruptures (SGTRs) provide a pathway for flow of material from the primary to the secondary side of the reactor coolant system. From the secondary side material may be discharged directly to the environment via the main steam relief valves. Containment bypass via SGTRs could lead to severe radiological consequences should additional events lead to core degradation. As in the case of the

V-sequence LOCA, details regarding the flow path, in particular the break area and location, could strongly affect the timing of events, the amount of material released to the environment, and the occurrence of containment breach because of overpressurization. Considerations of a wet versus dry pathway and plate-out in the steam generators could be particularly important for severe accidents involving SGTRs.

6.2 External Events

External events such as earthquakes, plane crashes, or terrorist attacks could lead to containment failure. Ignoring design or construction faults, the severity of natural events such as earthquakes would have to exceed the corresponding design bases (e.g., the safe-shutdown earthquake(SSE)). SSEs typically have ground accelerations in the range of 0.1 to 0.25 g. For some plants it has been argued that an earthquake exceeding the SSE in intensity would not result in direct containment failure but would weaken the structure, making it more susceptible to subsequent overpressurization.[2 3] However, presumably earthquakes of some intensity higher than that of the SSE could produce openings that are large when compared with the allowable leak rate, thus effectively failing containment. In the absence of analysis and model testing to evaluate the sizes of openings produced by earthquakes of different strengths, the earthquake strength that fails containment is uncertain. It is reasonable to assume that external events capable of directly inducing containment failure or weakening the containment would constitute the initiating events for the resulting severe accident. Thus, uncertainties regarding timing are relatively unimportant for such containment failure modes. An exception is the uncertain delay before any earthquake aftershock which might further weaken or fail containment when under severe accident pressure loads.

Uncertainties in the effects of external events can make a small or large contribution to risk uncertainty because the frequency of external events, such as large earthquakes, can vary substantially between sites.

6.3 In-Vessel Steam Explosions

Subsection 3.5 discusses uncertainties regarding the occurrence of in-vessel steam explosions and the likelihood of generating from such an event a missile with sufficient energy to penetrate the containment. Despite the advances in our knowledge over the last ten years, it concludes (for a PWR) that current uncertainties make it impossible to determine whether containment failure by in-vessel steam explosions can occur, and if it can, under what initiating conditions (particularly primary system pressure) or with what conditional probability, given core melt. This means that bounds narrower than 0 to 1 cannot be placed on this probability. Similar analysis has not been performed for BWRs, in which the structures within and below the core differ significantly from those in PWRs. Such analysis may or may not lead to the same conclusions for the possibility and probability of containment failure as for PWRs.

6.4 Containment-Isolation Failure

6.4.1 Nature of Containment-Isolation Failure

Most potential modes of containment breach are the result of loads (e.g., pressure, temperature, seismic motion, or missiles) imposed on containment during accidents. However, containment can also be breached through (a) preexisting openings which cannot be completely isolated following the accident, either because of the nature of the opening or because of failures in systems designed to accomplish containment isolation, or (b) openings which are introduced during the accident because of failures of the containment-isolation systems. The term "containment-isolation failure" is used here to describe such openings collectively.

The potential for containment-isolation failures is limited in several ways. In accordance with the requirements of the Code of Federal Regulations (10CFR50), fluid-bearing lines that penetrate containment are equipped with isolation valves. The containment-isolation system is designed to shut isolation valves automatically in nonessential lines* upon receipt of an ESF-actuation signal. Personnel hatches, equipment hatches, and fuel-transfer-tube doors are provided with double O-ring seals. Opening of such doors during normal plant operation is subject to strict administrative controls. Integrated containment-leakage-rate testing is conducted to ensure the pressure-retaining capability of containment prior to initial plant operation and periodically thereafter.

A breach because of containment-isolation failure would very likely exist from the time of the initiating event, or shortly thereafter, when automatic actuation of containment-isolation systems occurs. Therefore, timing is not a key area of uncertainty for this failure mode. There are much larger uncertainties associated with the location and area of the breach. Some plants have equipment hatches or auxiliary personnel locks that lead directly to the outside; however, most containment penetrations lead from containment into the surrounding buildings. In most plants the equipment-hatch, personnel-lock, purge-air, and supply-air penetrations are installed at elevations higher than penetrations for most liquid-bearing lines. Leakage through penetrations at higher elevations could result in a more direct pathway to the environment.

The probable location of an isolation failure can be accident-specific, depending on signals generated to actuate isolation and on the availability of motive power. The larger isolation valves are typically air-operated and would fail closed upon loss of power. However, there are also motor-operated containment-isolation valves that require power for closure.

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*Essential isolation valves, for example those in emergency core cooling lines, must be open.

Isolation failure could occur because of mechanical failure to fully close or properly seat valves or doors. In some cases, failure to close or properly seat a containment-isolation valve may not result in a significant pathway for leakage from the containment atmosphere. Leakage could, for example, be precluded by a redundant isolation valve or by the lack of a direct connection between the containment atmosphere and the inside of the pipe or duct.

6.4.2 Leakage Rates

Direct flow from the containment atmosphere through an inadequately sealed door, isolation valve, or other opening could easily lead to substantial leakage. For example, design leakage of 0.1 volume percent per day from a typical, large-dry PWR containment at its design pressure requires an opening only a few millimeters in diameter.

If the breach due to containment-isolation failure is sufficiently large, the resulting leakage rate may limit the pressure buildup in containment. For example, the volume leakage rate L, measured in units of the containment volume, required to prevent further increase in the containment pressure because of steam generation from a coolable debris bed can be found from the expression

$$L = Q_D/(V \cdot \rho \cdot h_g)$$

where Q_D is the decay power, V is the containment free volume, and ρ and h_g are the density and heat of vaporization of saturated steam, evaluated at the containment temperature. Values for a typical, large-dry PWR at 422 K (300°F) are:

$$Q_D = 40 \text{ MW}$$
$$V = 5.66 \times 10^4 \text{ m}^3$$
$$\rho = 2.476 \text{ kg/m}^3$$
$$h_g = 2.114 \text{ MJ/kg}$$

In this example, L is approximately 1000 volume percent per day; this leakage rate would require an opening roughly 0.2 m (8 in) in diameter, based on Napier's equation for critical flow of saturated steam through an orifice, w = CAP/687, where w is the rate of steam flow in kilograms per second, C is the coefficient of discharge (say, 0.6), A is the area of the orifice in square meters, and P is the pressure in Pascals.[4] The RSS used a leakage rate 10 000 times the design leakage rate for accidents involving PWR containment-isolation failure. This represents a leakage rate of 1000 volume percent per day at the containment design pressure.

To arrest containment-pressure buildup, less leakage would be required for a lower decay power, a larger free volume, or a higher temperature (and associated pressure). The above expression also implies that leakage through the breach is the only means of removing energy from the containment atmosphere. In reality, even with failure of the containment-heat-removal systems, a fraction of the decay power would be transferred to the passive heat sinks in contact with the containment atmosphere. In the case of core-concrete interactions, even less leakage would be required to arrest containment-pressure buildup, because a substantial fraction of the decay power would be imparted to the concrete. For example, according to the RSS, overpressurization because of the gradual buildup of steam or noncondensable gases would be precluded in a typical, large-dry PWR by an opening of 0.09 m (3.6 in) in diameter (i.e., one that permits leakage of 200 volume percent per day).[1]

The rate of energy addition to the containment atmosphere during rapid vessel depressurization, ex-vessel steam explosions or steam spikes, or combustion events could greatly exceed the decay power. Much larger leakage rates through correspondingly larger breaches would be required to affect significantly the containment pressure response to such events. Thus a significant breach because of containment-isolation failure may not strongly influence the timing or consequences of a subsequent larger breach attributed to rapid overpressurization. Containment breach because of overpressurization is discussed further in Subsections 6.5 and 6.6.

Some BWR sequences in which decay heat is transferred from the core to the suppression pool by the ECCS, and suppression-pool cooling fails, would be mitigated by a containment leakage of an appropriate size. Without leakage, containment failure by steam overpressure would occur after one day, probably failing the ECCS and leading to core melt. An appropriate containment leak would allow the suppression-pool water to boil off over about 10 days and, if the pool could be replenished in this time, the core would remain cooled.

Another uncertainty associated with the breach diameter arises when high aerosol concentrations are present in the containment atmosphere. Morewitz cites experimental data which indicate that when 10 to 70 mg of aerosol have entered a 1-mm-diameter capillary, the capillary will plug.[5] Thus aerosols could decrease or actually arrest containment leakage.

6.4.3 Conditional Probability

Where sufficient data exist, the uncertainty in the conditional probability of containment-isolation failure should be smaller than the uncertainties in the probabilities of other containment-failure modes, which are dominated by modeling uncertainties. As explained in Subsection 6.4.1, containment-isolation failure can be caused by preexisting openings or by failure of the containment-isolation systems. This subsection shows how the uncertainty in the probability of isolation failure by preexisting openings depends on the leak size. The potentially more complicated problem of the probability of openings caused by failure of isolation systems is a matter of plant reliability, of the kind generically discussed in Section 2. It is not considered here.

Because small preexisting leaks are more likely than large ones, there are more data and less uncertainty for the probability of a small leak. The data in Table 6-2, which show containment unavailability due to leakage exceeding allowable leak rates, were compiled by Weinstein from licensee event reports and other sources for LWRs in the USA up to 1980.[6] The allowable leak rate is typically 1 volume percent per day for BWRs and 0.1 volume percent per day for PWRs.[7]

Table 6-2 Leakage in excess of allowable leak rates[6]

	BWRs	PWRs
Experience (in plant-years)	234	300
Number of Failures	50	25
Containment Unavailability	19%	5%

Because these unavailabilities come from a reasonably large data base, they are unlikely to change by as much as a factor of two in the future, unless positive steps are taken to reduce them. Specific features of some plants--for example, maintenance of containment at a continuously monitored subatmospheric pressure--would make the generic figures inapplicable and the relative uncertainty, from a smaller data base, larger. The unavailabilities given in Table 6-2, being the fraction of operating time that the containment is not leak-tight, are therefore estimates of the probability of preexisting containment leakage exceeding the allowable rate at the time of an accident.

Exceeding the allowable leak rate is not necessarily a relevant criterion for containment unavailability in risk calculations, however. As explained in Subsection 6.4.2, leakage can mitigate overpressurization if the leak rate exceeds about 1000 volume percent per day. This factor is clearly plant-specific (the RSS estimated that a leak rate between 100 and 200 volume percent per day would prevent overpressure failure for Surry[1]). The probability of such large leaks is much smaller and more uncertain than that of all leaks exceeding the allowed rate, as Table 6-3 shows.

Because of the paucity of data (from a total of 3 failures), the uncertainty in the probability of a preexisting leak in excess of about 300 times the allowable rate is substantial. The uncertainty increases with increasing leak rate for leak rates between those in Table 6-2 and 6-3.

Table 6-3 Leakage in excess of 300x allowable leak rates[7]

	BWRs	PWRs
Experience (in plant-years)	171	236
Number of failures	2	1
Actual leak rate ÷ allowable leak rate	~400	~900
Containment unavailability	0.15%	0.07%

Another source of uncertainty in leakage probabilities is the possibility that failures have been omitted from the data base used. Weinstein regards his estimates of availability as upper limits because

- Not all failure data have been reviewed and
- Not all failures have been discovered.[6 7]

In addition to the possible mitigation of overpressure failure, another potentially important effect of containment leakage is the release of radioactivity through the leak in the event of a severe accident. Whether this makes an important contribution to the radioactive source term to the environment, and hence to risk, will depend upon

- Whether containment failure occurs by some other means and
- The inventory of airborne fission products in containment.

Isolation failure is obviously unimportant if some larger break occurs. However, if isolation failure provides the main escape route for fission products to the environment, then the large uncertainty in the inventory of airborne fission products in containment, discussed in Subsections 7.3 and 7.4, causes large uncertainty in the minimum leak size that would be important. Thus, even though the probability of preexisting leak sizes smaller than about 100 times the allowed leak rate are relatively well known from data, there is large uncertainty in the probability of a leak larger than the minimum size needed to generate a significant off-site radioactive source term.

6.5 Overpressure Attributable to Gas Accumulation

Accident sequences involving containment failure because of internal overpressure are dominant contributors to estimated public risk, according to several PRAs.[1 8-11] Containment internal overpressure could occur in various ways. Those discussed in this subsection involve the accumulation of steam or noncondensable gases. Containment internal overpressure could also occur because of combustion events: deflagrations or detonations. Uncertainties regarding containment failure attributable to combustion events are discussed in Subsection 6.6. Except in the case of detonations, the containment structural response is essentially static (faster than the pressure buildup). This permits the analysis of uncertainties in the containment structural response separately from uncertainties in the pressure history.

In the absence of containment bypass or a large breach, the containment pressure tends to increase as steam and noncondensable gases are produced. If containment-heat-removal systems are available, such gas production does not pose a threat to containment. However, if containment-heat-removal systems fail, the accumulation of gases can result in pressures that challenge containment integrity. Whether and when containment fails depend on the pressure difference across its boundary; this also may affect the nature of any failure, because it is possible in certain cases that cracks in the steel liner of concrete containments will occur at lower pressures than will gross failure. Subsection 6.5.1 discusses uncertainties in calculations of pressures within containment, which arise from uncertainties identified in Subsections 4.2 and 5.2. Then, uncertainties in failure pressures and failure modes of containments are examined in Subsection 6.5.2. Finally, the effect of these uncertainties on containment-failure probabilities and risk is discussed in Subsection 6.5.3.

6.5.1 Sources of Steam and Noncondensable Gases

There are various sources of steam and noncondensable gases that could contribute to internal overpressure when containment-heat-removal systems fail. Blowdown of primary or secondary coolant can occur at various rates depending on the accident sequence. In "feed and bleed" situations, steam could be continuously added to the containment atmosphere while maintaining a covered core. Containment failure because of steam accumulation could then, in some plant designs, cause "feed and bleed" failure which, in turn, would lead to core degradation. In the majority of postulated severe accidents, failure of core cooling would precede containment failure, and containment overpressurization (from accumulation of gases) would not occur prior to vessel breach and the discharge of molten material into containment. At the time of vessel breach, steam and noncondensable gases could rapidly be added to containment as a result of vessel depressurization and the quenching of the discharged melt by water. This would result in a so-called "steam spike". Subsequently, a more gradual pressure rise could result from the release of steam generated by debris-bed cooling, or the release of steam and noncondensable gases as a result of melt/concrete interactions. Figure 6-1 illustrates a characteristic "steam spike" (at the time of bottom head failure) followed by a long-term pressure buildup because of core/concrete interactions for a particular PWR accident, based on MARCH calculations.[12] This kind of calculation, from which the likelihood of containment failure is often assessed, is based upon many assumptions, both pessimistic and optimistic.[1 8-11] These make the uncertainty difficult to assess.

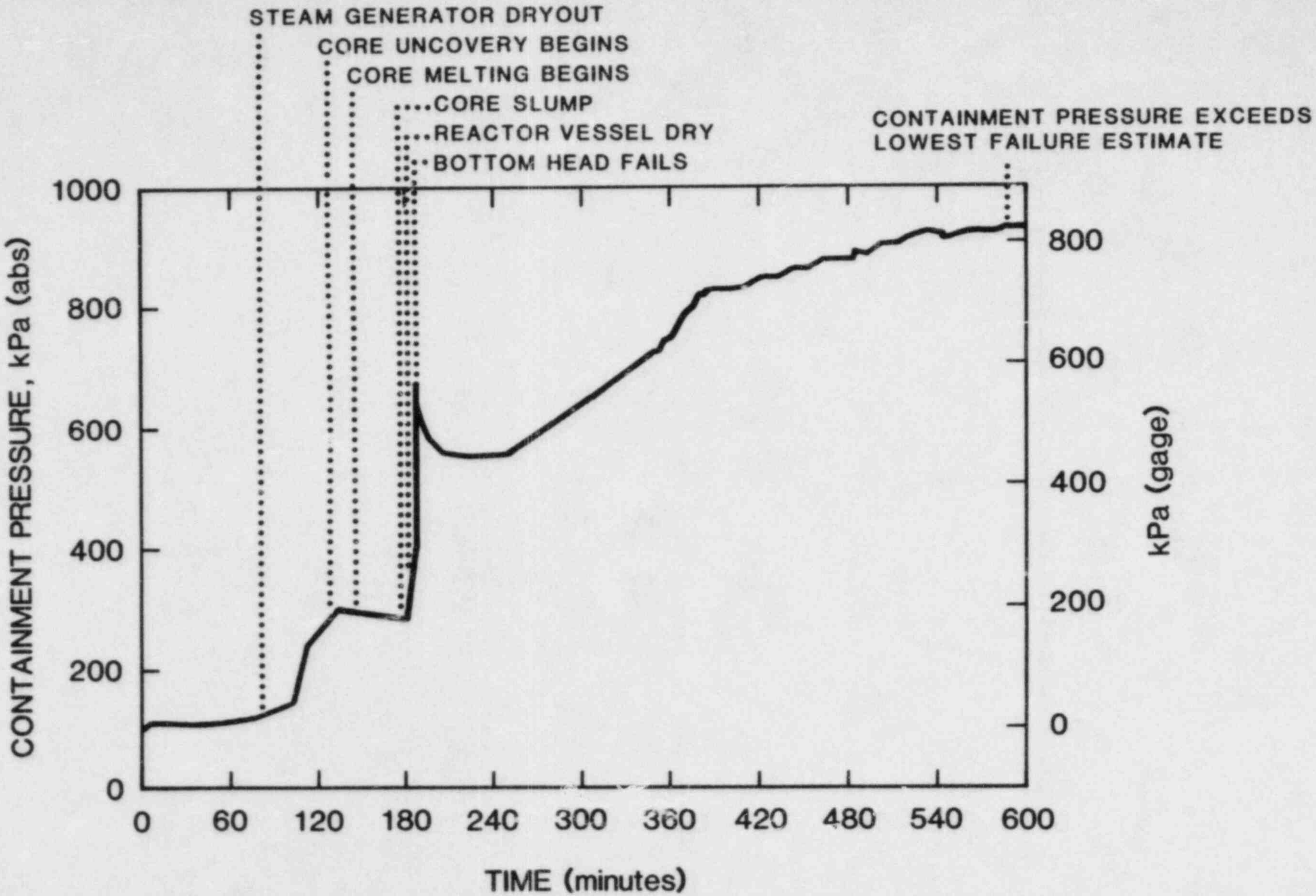

Figure 6-1 Containment building internal pressure versus time for Zion 1 TMLB', MARCH base case calculations.[12]

This is particularly true of calculations of peak pressures of steam spikes produced at vessel failure because of the wide range of possible phenomena. As discussed in Subsection 4.2, these include steam explosions of uncertain size, debris beds confined in the reactor cavity or distributed over the containment floor, and, for high pressure sequences, the interaction of a high-speed jet of melt with water, and the possible production and subsequent oxidation of hot aerosol.

6.5.2 Failure Modes and Threshold

It is difficult to generalize about the failure pressures of containment buildings because of the wide variety of LWR containments in the US. Reference 13 classifies the different types and provides illustrations. To our knowledge, all US containments are either steel or steel-lined concrete. Calculations and estimates of the failure pressures of containment buildings, discussed in Subsections 6.5.2.1 and 6.5.2.3 below, predict failure at gage pressures of 1.4 to 3 times the design pressure for steel-lined concrete containments and 2 to 10 times the design pressure for steel containments. The predicted failure pressures are higher than the design pressures for at least three reasons:

- Design codes ensure, with a high degree of confidence, that containment buildings withstand proof tests (typically at 115% or 125% of the design pressure) and design-basis accidents (up to the design pressure) throughout their design life. The design codes include substantial margins of safety because failure pressures could not be predicted accurately when the codes were written.[14 15]

- Materials used may have better properties than those specified in the procurement documents.[14 15]

- In some cases it is easier to construct a stronger structure than one that just satisfies the design requirements.

Current uncertainties in failure pressures exist because the processes leading to failure cannot be predicted in detail. It is generally possible to identify gross failure modes, with corresponding pressures, above which the containment will certainly fail; however, it has not been feasible to analyze local failure modes exhaustively to establish the lowest pressure at which a particular containment will fail. This is because such analysis would have to cover many potential failure sites and modes and would require knowledge of the actual processes appropriate to each. Moreover, although structural deformations can be calculated and structural failures can be estimated, the use of structural deformation information to estimate when the functional failure of the containment occurs (such as leakage around seals) is not possible with the current state of the art.

6.5.2.1 Concrete containment

The factors causing uncertainty in the failure modes and pressures of concrete containments are reviewed in Reference 14. The RSS employed two groups of analysts to calculate the failure pressure of the Surry containment, which consists of reinforced concrete.[1] Their results are described in Appendix E of Appendix VIII of Reference 1 and are listed, with the different assumptions used, in Table 6-4 of this report. All the criteria used in these analyses relate to gross failure, rather than to local failure associated (for example) with penetrations. There is agreement between the authors of these calculations that gross failure of concrete containment will occur at a pressure slightly above that causing the last main load-bearing structural element to reach its yield stress, because thereafter small increases in pressure (and hence stress) will lead to large deformation of the whole structure. The variation among the results for Surry in Table 6-4 arises from different identifications of the last effective element. Thus, if all the rebars are effective at yield, failure occurs at a gage pressure of 517 kPa. (Note: All pressures cited in this subsection are gage pressures.) The behavior of the steel liner is unknown, and there currently is a considerable amount of technical discussion and controversy on this issue. The liner will contribute to the strength if it does not tear; Mast makes a small allowance (up to 552 kPa) for this,[1] but a full contribution of the liner yield strength would give 634 kPa. This seems unlikely because of the possibility of tearing at penetrations or at the base of the walls. However, Sampath et al argue that spallation of concrete from the outside will relieve the outer reinforcement from load bearing.[1] This means that only the inner rebars will be effective, reducing the failure pressure to 439 kPa, including the whole strength of the liner. Earlier tearing of the liner would further reduce this value. It is not clear, therefore, that the 1-std-dev range finally adopted, 536 ± 103 kPa, accommodates all possible gross failure modes. The 2-std-dev range (listed in Table 6-4) probably does, however. In addition, locally initiated failures (for example, at the base of the wall, at penetrations by movement of concrete relative to pipes or differential straining in concrete near additionally reinforced areas, or at defects such as voids) might occur at lower pressures than gross failure. There are many (~ 150) penetrations in a reactor containment building, and, while some components (such as equipment hatches) may be relatively easily analyzed, the consequent strains in the neighboring wall will be more difficult to predict.

The main load-bearing elements in a prestressed containment building are about 500 steel tendons. Three examples are listed in Table 6-4. The Oconee analysis is mainly an extrapolation from the RSS in spite of the different structure.[9 16] The Calvert Cliffs analysis used hand calculations for the yield point and ultimate strength of the tendons.[10 16] The lower limit of the margin (ratio of calculated failure pressure to design pressure) is higher than in the RSS because a strength degradation mode like concrete spallation has not been identified. The Zion analysis shows that, if it is assumed that the system behaves in a grossly elastic-plastic manner, a hand calculation leads to a high failure pressure with low uncertainty.[2 17] The small range adopted is intended to account for uncertainties in material properties alone. It includes no allowance for uncertainty in the failure process. However, all the caveats about local effects noted above and in Reference 14 apply. Also, when a 1/14-scale model

Table 6-4 Calculated failure pressures for concrete containments

Plant	Containment Type	Design Pressure* (kPa)	Analyst (Reference)	Failure Criterion	Failure Pressure* (kPa)	Margin Failure Pr/ Design Pr
Surry	Reinforced concrete, steel lined	310	Mast [1]	Rebar yield	517	1.7
				Rebar yield + liner strength (part)	552	1.8
				Rebar ultimate	827	2.7
			Sampath et al [1]	Part rebar + liner yield	439	1.4
				Part rebar + liner ultimate	603	1.9
			Summary [1]	Rebar + linear yield	634	2.0
				Adopted range (2 std dev)	586 ± 207	1.9 ± 0.7
Oconee	Prestressed reinforced concrete, steel lined	407	Kolb et al, RSSMAP [9 16]	Estimate (2 std dev)	814 ± 276	2.0 ± 0.7
Calvert Cliffs	Prestressed reinforced concrete, steel lined	345	Hatch et al, RSSMAP [10 16]	Gross yield	690	2.0
				Ultimate	965	2.8
				Adopted range (2 std dev)	827 ± 138	2.4 ± 0.4
Zion	Prestressed reinforced concrete, steel lined	324	Walser [17]	Tendon yield (1% strain)	827	2.6
				Tendon yield + liner	927	2.9
			ZPSS [2]	Adopted range (2 std dev)	926 ± 53	2.86 ± 0.16

*Gage pressure

of a prestressed containment was tested to failure, bowing of a vertical buttress holding the hoop tendon anchors caused cracking, loosening some anchors which overstressed another tendon, which failed.[18] This kind of interactive effect is not modeled in the hand analyses [2 17] but should be considered in order to ascertain, for example, that it does not become more important at full scale.

Part of the margin calculated for prestressed containments arises from the excess of the mean tendon steel strength, as measured in tests, over the specified value. Because of (1) the series-parallel combination of elements of steel wire in a network of many tendons, (2) the statistical variation in material properties, and (3) the likelihood that failure will occur at the weakest point, use of the mean measured tendon strength may lead to an overestimate of containment ultimate strength.[19] Harrop has shown that, for ungrouted tendons, it is not unduly conservative to use the minimum specified strength of the tendon material in calculations to predict the ultimate failure pressure of a posttensioned concrete containment.[20]

6.5.2.2 Steel containment

It might be thought, because of the presence of only one material, that the failure pressure of a steel containment would be easier to calculate than that of a concrete one. At least some of the same problems are present, however. These are illustrated in Table 6-5, where different estimates for the failure pressure of the Sequoyah 1 containment are compared. The main source of variation between the calculations compared in the Greimann et al review is the use of different analytic approximations for treating the stiffeners.[21] The finite-element calculations generally give higher results.[21 22] The recent analysis of the Watts Bar steel shell containment examined potential failure locations in considerable detail and concluded that a realistic range for the capacity of the Watts Bar containment would be between the point of initial yielding of the cylindrical steel shell (827 kPa) and the point at which buckling of the equipment-hatch door would occur (965 kPa).[23 24] The high margin of safety at this plant arises from the use of thicker plate following difficulties in construction of the relatively thin Sequoyah 1 containment. Yielding of the steel shell would not result in failure due to breach of the shell itself. Rather, such yielding introduces the possiblity of failures due to structural interactions at penetrations caused by excessive shell deformations. The fractional variations of the margins calculated by finite-element codes are somewhat less than the uncertainties indicated for Surry in Table 6-4.

6.5.3 PRA Approach

This discussion of the failure pressure of containments shows that uncertainties are probably dominated by local effects that influence the actual processes leading to functional failure. These are not included in calculations that assume the structure behaves in an axisymmetric, elastic-plastic manner. Such calculations therefore understate the uncertainty. However, it is difficult to provide a justified uncertainty range for a given plant. An upper limit may reasonably be assigned at the point when all the structural materials, assumed to act in an axisymmetric elastic-plastic manner, have substantially yielded. It is more difficult to provide a lower bound. One way of doing this would be to identify all potential failure modes, both global and local (for example, at penetrations) and the possible ranges of their failure pressures. These ranges would include uncertainty due to expected variations in workmanship, material properties, weather conditions, and evaluations of the processes leading to functional failure. The lowest of these lower limits would then be the overall lower bound of the failure pressure. Identification of potential failure modes is subject to completeness uncertainty, however. This could be reduced by more model testing, which would also assist in improving analytical capabilities to describe deformations as a function of pressure and, with greater difficulty, the processes of functional failure. In the absence of detailed structural analysis, including identification and evaluation of all potential failure modes, the proof pressure would be the justified lower bound. Even this value could be too high, however, if temperature effects in accidents weaken the containment between the design pressure and the proof pressure (to which the containment was tested at ambient temperature).

Figure 6-2, a composite reconstructed from more than one source, illustrates containment structural failure information as frequently displayed in PRAs.[1 2 9 10] The figure purports to give cumulative distributions of probability for several containments, as functions of internal pressure. However, it is immediately clear that what is given is not probability in the sense of the fraction of a large number of containments of a particular design expected to have failed below a given pressure (a frequentist probability), because in practice data and models do not exist that would justify such a probability distribution for any existing containment design. Rather, what is plotted in the figure is someone's degree of belief, expressed as an increasing fraction of 1, that a certain containment will fail at a given value of pressure--the subjective probability as defined in Reference 26. Thus, the ordinate in the figure should be labeled "Subjective Probability" (a good general practice).

Subjective probability assignments are not unique because they reflect the experience, attitudes, and other influences upon those making them. Thus, their use in an analysis (like a PRA) causes the results to be nonunique, and controversy concerning their use in PRAs exists.[27-29] Identification of those aspects of an analysis subject to subjective probability limitations would enhance the understanding of the bases of the results and their applicability and reliability in decision-making. In the usual situation, where little or no supporting analysis or testing is undertaken, stylized assumptions regarding the form of the subjective probability distribution curve (Gaussian), the "median" failure pressure (twice the design pressure), etc., are often made. Conclusions depending on such assumptions will be correspondingly uncertain. This uncertainty can be evaluated by the use of a range of possible subjective probability distribution curves.

Table 6-5 Calculated failure pressures for steel containments

Plant	Containment Type	Design Pressure* (kPa)	Analyst [Reference]	Failure Criterion	Failure Pressure* (kPa)	Margin Failure Pr/ Design Pr
Sequoyah 1	Cylindrical steel with circumferential and vertical stiffeners	74	Carlson et al, RSSMAP [8]	First yield Ultimate strength Adopted range (2 std dev)	165 207 207 ± 41	2.2 2.8 2.8 ± 0.6
			R&D Associates [21]	First yield without stiffeners†	276	3.7
			NRC Research [21]	Yield of vertical stiffener acting as beam†	345	4.7
			Franklin Research [21]	Membrane yield†	352	4.8
			Offshore Power Systems [21] See also Ref 25	Yield (smeared stiffeners)†	365	4.9
			Greimann et al [21]	Stiffeners yield (2 std dev range)†	414 ± 110	5.6 ± 1.5
			Foster and Stone (TVA) [22]	First yield 166% of yield strain (min. actual strength)	345 400	4.7 5.4
Watts Bar	Cylindrical steel with circumferential stiffeners	93	Jung [23 24]	General yield of cylinder Buckling of equipment hatch Range	827 965 896 ± 69	8.9 10.4 9.6 ± 0.7

*Gage pressure.

†Values adjusted by Greimann et al [21] to use mean actual strength.

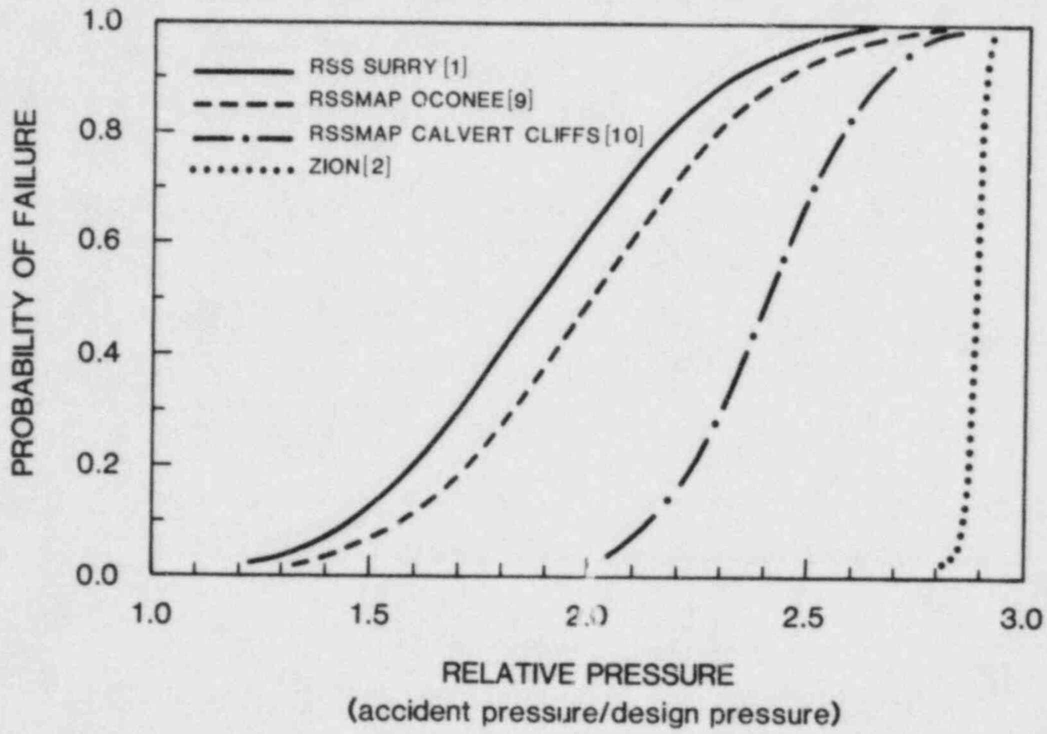

Figure 6-2 Subjective containment structural failure-probability distributions utilized in PRAs. [16]

The uncertainties in modeling local failure processes cause not only the uncertainty in failure pressure discussed above but also uncertainty in the size of the breach. It has been suggested by Cybulskis that actual containment failures could be characterized by numerous small openings that would reseat as the pressure was relieved, resulting in significant fission-product retention.[16] Experiments commissioned by the Atomic Energy Control Board of Canada (AECB) on a concrete containment model subjected to gradual pressurization showed that cracking occurred at a significantly lower pressure than catastrophic failure of posttensioning tendons. Thus, such a structure may be expected to leak gradually before gross failure (unless additional effects due to the energetic venting of steam under pressure, absent in the model which was pressurized by liquid water in a vinyl liner, are important).[18] However, such gradual leakage seems less likely for steel or steel-lined containments (unless the liner fails when the concrete cracks), or for any containment subjected to rapid overpressurization from a steam spike or combustion event. As discussed in Subsection 6.4, a relatively large area (up to 0.2 m^2) would be required to arrest pressure buildup attributable to the addition of reasonable fractions of the decay power to the containment atmosphere in a gradual manner. For transient events such as rapid quenching of melt discharged from the reactor vessel, or combustion events, an opening at least an order of magnitude larger would be required to reduce peak pressures by only about 10 percent. It seems unlikely that cracks could occur in the steel liner or shell, in penetrations, or in associated welds or seals, in a manner which would permit re-seating. These uncertainties in the size and shape of the breach, and in whether failure would be catastrophic or gradual, strongly influence the release of fission products, as discussed in Section 7 below.

Uncertainties discussed up to this point include the containment-failure pressure, the location of the failure, and the associated flow path. Such uncertainties have potential importance to uncertainty in risk. Spulak performed a sensitivity study in which he estimated risk as a function of mean containment-failure pressure based on PRAs for Oconee and Grand Gulf.[30] The standard deviation in the mean failure pressure was assumed to be proportional to the mean (gage) failure pressure, with a proportionality constant of 0.2. Containment-overpressure failures were assumed to be catastrophic, creating a direct pathway to the outside environment. The results indicate that the risk calculated by both PRAs is sensitive to the mean containment-failure pressure. The risk for both plants decreased by about a factor of three as the mean failure pressure was varied from the design pressure to a value high enough to preclude overpressure failure. Although these calculations display the broad character of the dependence of risk on failure pressure, they also contain uncertainties which make the results uncertain in several ways. The possibility that the standard deviation of the failure-probability distribution is narrower than 0.2 of the mean may make the dependence on the mean sharper, as pointed out by Cybulskis.[16] Uncertainties in the peak-pressure calculations in the various accident sequences make uncertain the pressure value at which risk is most sensitive to pressure. Finally, the magnitudes of the radioactive source terms corresponding to each sequence and release category are uncertain, and this makes uncertain the overall range over which the risk can vary as a function of failure pressure. The combined effect of these factors is that the sensitivity of risk to failure pressure may be more or less important than indicated by Spulak's calculations.[30]

The level of effort appropriate for reducing uncertainty in the failure pressure and breach characteristics depends on the nature of the accident sequences found to be important. If accidents involving loss of containment-heat-removal systems and resulting in containment-overpressure failure are important, it may be more appropriate to reduce the uncertainty in the time required for equipment recovery than to characterize precisely the failure pressure and breach characteristics.

6.6 Overpressure Attributed to Combustion Events

Containment overpressure can occur due to combustion events--deflagrations or detonations. In the case of deflagrations, the containment structural response will be static, and the structural-response uncertainties discussed in Subsection 6.5 will be directly applicable.

Again, the importance attached to reducing such uncertainties will be plant-specific. If accidents involving deflagrations dominate perceived risk, it may be more useful to reduce uncertainties in hydrogen and carbon monoxide generation rates, ignition criteria, burn velocities, and extent of combustion, than to characterize precisely the failure pressure and associated breach characteristics.

In the case of detonations, depending on details of the plant design, the containment could be threatened by detonation waves, detonation-induced shock waves, or missiles. For detonation events, the response of the containment structure depends on the initial impulse and the residual overpressure. The load characterization on the wall of the structure can be quite complex, depending on the initiation location, internal wall geometry, and the nature of the combustible mixture. A well-established computer code, CSQ, which solves continuum-mechanics problems for two-dimensional motion, has been used to analyze detonations of a dry hydrogen-air mixture in a large, dry containment building (Zion) and in subcompartments of an ice-condenser containment (Sequoyah).[31] For details regarding the analyses and associated uncertainties, see Reference 31. It will suffice here to note that detonations require relatively high concentrations of combustible gases and oxygen. The potential for achieving such concentrations and initiating a detonation appears to be low, based on existing analyses of light-water reactors. Should this perception be altered in the future, uncertainties in containment response to detonations should be examined in detail.

Because of the variety of containment types and strengths and the uncertainties in the production and combustion of flammable gases, the possibility of containment failure is highly plant specific. Camp has described how the different types of combustion processes may affect different types of containment in severe accidents.[32] Table 6-6 summarizes his conclusions as to the possibility of containment failure. The quantities of hydrogen that need to burn in order to fail containment, and the corresponding extent of metal/water reaction are derived by methods set out in Reference 33. Specific calculations are referenced in Table 6-6. In some cases, different containments in the same classes as the reference plants listed may have properties, such as failure pressure, sufficiently different to change the conclusions listed.

BWR containments of Mark I and II are not at risk from combustion because they are inerted. Table 6-6 shows that BWR Mark III and PWR Ice Condenser containments may be at risk because of uncertainties in the extent of hydrogen production and the burning processes. For large dry containments the possibility of direct failure due to combustion appears to be small, although this depends to some extent upon the uncertain ability of containments to withstand pressures substantially higher than the design pressure (discussed in Subsection 6.5). Indirect containment failure, caused by damage to equipment--particularly fan coolers--by local detonations cannot be ruled out.

6.7 Temperature-Induced Failures

Temperature-induced failures of gaskets or seals could result in containment breach during some postulated severe accidents. Gaskets are used in conjunction with the equipment hatch, personnel-lock, and

Table 6-6 Possibility of containment failure due to combustion

Containment Type (Reference plant, assumed failure pressure)	Hydrogen burning modes	Extent of clad-water reaction necessary (or equivalent flammable gas production)	Possibility of containment failure	
			Without igniters	With igniters
BWR Mark I and II	None - containment inerted	Any	None	
BWR Mark III (Grand Gulf, 386 kPa gage)[35]	Global burn of less than 6-8% H_2 mole fraction	Less than 15-20%	None - insufficient to fail containment	
	Global burn of more than 6-8% H_2 mole fraction	More than 15-20% (Well within possible range)	Highly probable	None
	Stable diffusion flames	Any	Direct - none Indirect - via possibility of equipment damage	
	Local detonations	Unknown	Possible	Possible
	Large burns due to: high H_2 release rates at vessel breach, local inerting followed by remixing, slow mixing due to temperature inversions or elevated initial pressures if suppression pool cooling compromised	15-20%	Possible	Possible
PWR Ice Condenser (Sequoyah, 249 kPa gage)[36]	Global burn of less than 6-8% H_2 mole fraction	Less than 20-25%	None - insufficient to fail containment	
	Global burn of more than 6-8% H_2 mole fraction	More than 20-25% (well within possible range)	Highly probable	None
	Stable diffusion flames, lower compartment burns or upper plenum and ice condenser burns.	Any	Direct - none Indirect - via possibility of equipment damage	
	Local detonations particularly if fans fail and containment atmosphere is not well mixed	Unknown	Possible	Possible
	Burns in upper compartment	More than 20-25%	Possible	Possible

Table 6-6 Possibility of containment failure due to combustion

Containment Type (Reference plant, assumed failure pressure)	Hydrogen burning modes	Extent of clad-water reaction necessary (or equivalent flammable gas production)	Possibility of containment failure	
			Without igniters	With igniters
PWR Large Dry (Bellefonte, 900 kPa gage*)	Global burn of less than about 10% H_2 mole fraction	Less than about 95%	None - insufficient to to fail containment	
	Global burn of more than about 10% H_2 mole fraction	More than about 95% (Possible because of possible contribution from steel-steam reactions, core-concrete interactions, etc.)	Unlikely because if power is on, pre-ignition is likely before 10% mole fraction of H_2 is reached and if power is off, loss of fans and sprays probably implies at least partial steam inerting.	
(740 kPa gage or less*)	Global burn of about 8% H_2 mole fraction	More than about 75%	Likely	Unlikely
(900 kPa gage*)	Local detonations	Unknown	Direct - probably none Indirect - via possibility of equipment damage	

*The design pressure of the Bellefonte containment is 345 kPa gage.

fuel-transfer-tube doors. Electrical-penetration assemblies and some isolation valves use sealing materials which are susceptible to failure after prolonged exposure to high temperatures at pressures below the containment-failure pressure. For example, it has been suggested that temperature-induced failures of electrical-penetration assemblies would occur prior to overpressurization for several postulated severe accident sequences at Browns Ferry, a BWR with a Mark I containment.[34]

Tests of typical penetration assemblies beyond their temperature and pressure-qualification conditions are not, in general, available. By definition, qualification tests are performed to show that a component will be functional when subjected to the test conditions. They are not intended to determine fragility levels. It may be possible, based on knowledge of the seal or gasket material and typical qualification-test data, to estimate the temperature at which failure of a given penetration assembly would become possible. The use of such a threshold to predict a seal or gasket failure would involve some uncertainty, because the actual failure point might also depend on the temperature history, the differential pressure across the seal or gasket, the radiation exposure, etc. Finally, estimating the temperature of the seal or gasket in some complex penetration-assembly geometries might require some detailed analyses.

Temperature-induced failures would be most likely in accidents which also had the potential for overpressure failure because of gas accumulation. Temperature-induced failures might also result if stable diffusion flames occurred instead of deflagrations. In essence, temperature-induced failure is a competing mode of failure which could lead to earlier breach. It should be possible to bound the location and flow areas associated with temperature-induced failures. This could be useful in estimates of offsite consequences.

6.8 Basemat Melt-Through

Uncertainties associated with basemat melt-through are discussed in Subsection 4.4 under core/concrete interactions. Conventional logic suggests that basemat melt-through is preferable to other modes of containment failure because the other modes create direct pathways to surrounding buildings or the outside atmosphere. This logic is valid only if basemat melt-through provides a path for depressurization through the underlying foundation material, thereby precluding a direct pathway to the outside atmosphere. There exists considerable uncertainty as to whether such depressurization will actually occur.

6.9 Summary

A summary of the major sources of uncertainties affecting containment breach and bypass is provided in Table 6-7.

Table 6-7 Major uncertainties regarding containment breach and bypass

Uncertainty	Comments
Containment Bypass	
Location of breach	Affects steam flooding in V-sequence, wet versus dry pathway for SGTR.
Area of breach	Affects time available for operator action. Multiple SGTRs have not been investigated.
Likelihood of successful operator action to isolate	
Effects of steam flooding in V-Sequence on operability of isolation valve and redundant ECC train	
Circumstances under which SGTR could be induced during other accidents	
External Events	
Frequency of external events which could directly compromise containment integrity	See Section 2
In-Vessel Steam Explosion	
Occurrence of high yield steam explosion	See also Table 3-6
Slug energy dissipated by upper internal structures	
Containment Isolation Failures	
Location of breach	To surrounding buildings or directly to atmosphere
Area of breach	Seal leakage to open penetration

Table 6-7 (Continued)

Uncertainty	Comments
Impact of isolation failure on occurrence and timing of overpressure failure due to gas accumulation	Strongly depends on passive heat sink available within containment
Clogging of leakage pathways by aerosols	Less significant if sprays operate
<u>Overpressure Attributable to Gas Accumulation</u>	
P-T Loadings	See Section 5
Containment Failure Pressure--	
Point at which last main load-bearing structural element reaches its yield stress	
Effects of concrete spallation	
Effects of liner plate cracking	
Effects of locally isolated deformations	
Appropriate material properties (e.g., tendon strength)	Use of mean properties may be nonconservative
Likelihood of benign failure arresting pressure buildup	
<u>Overpressure Attributable to Combustion Events</u>	
Deflagrations	
P-T loadings	See Section 5
Containment failure pressure	

Table 6-7 (Continued)

Uncertainty	Comments
Detonations	
Likelihood of achieving and igniting detonable mixtures	See Section 5
Characterization of impulse and residual pressure loads on containment wall	
Temperature-Induced Failures	
Integrity of penetration seals given high P, T, radiation history of severe accidents	
Basemat Melt-Through	
Occurrence and rate of core-concrete interactions	See Section 4
Rate of depressurization of containment due to basemat melt-through	

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7. IN-PLANT RELEASE AND TRANSPORT OF RADIONUCLIDES

The radiological consequences of a reactor accident are determined by the magnitude and characteristics of the radioactivity release, or "source term," from the plant, as well as by the relevant off-site conditions. Important characteristics of the source term, other than the magnitude of the release of the many radionuclides involved, include the physical and chemical nature of the released species, release timing and duration, and thermal-hydraulic features of the accompanying gas discharge (sensible heat, speed and direction of ejection, etc.). The aim of studies of severe accident phenomena is to provide an improved understanding of accident progression and hence, ultimately, of the radiological source term. Other sections of this report have described important uncertainties in the phenomena that govern accident progression and affect radionuclide release and transport. Section 7 discusses the release and transport phenomena themselves.

The first five subsections discuss successive stages of fission-product* and aerosol release and transport. Most of the discussion of actual FP and aerosol phenomena is presented in the first three subsections, which deal respectively with in-vessel release, transport and retention in the reactor coolant system (RCS), and ex-vessel release. Subsections 7.4 and 7.5 discuss the uncertainties in FP behavior in the containment atmosphere and during the discharge from the plant to the environment, which involve phenomena essentially similar to those pertinent in-vessel.

Subsection 7.6 is devoted to the treatment of source terms adopted in reactor safety studies to date, and the uncertainties therein. A summary, Subsection 7.7, collects together the chief uncertainties in fission-product and aerosol behavior, and discusses their implications for source terms from the reactor plant.

7.1 In-Vessel Release from Degraded Cores

This subsection opens with an overview (Subsection 7.1.1) of release phenomena and the methods used to describe them in reactor safety studies. Its largest component (Subsection 7.1.2) is devoted to the discussion of uncertainties associated with vaporization phenomena, which are judged to be the key contributors to the in-vessel release of fission products and of structural, cladding, and control-rod materials from degraded cores. A shorter subsection (7.1.3) discusses other processes that may, in some circumstances, have a significant impact on in-vessel release.

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*The term fission product (FP) is used throughout Section 7 to refer to radionuclides in general (i.e., including activation products, daughter products, etc.)

7.1.1 Overview of Release Processes

In the terminology of the Reactor Safety Study (RSS),[1] in-vessel releases may be regarded as the sum of two components:

1. A "gap" release--the release upon the initial clad rupture of some fraction of that portion of the volatile FP inventory present in the fuel-pellet cladding gap in gaseous form, and

2. A "melt" release--the further release which occurs as the core heats to melting and becomes molten.

The RSS specified release fractions of each radionuclide that would be associated with each of these components. For simplicity, and in recognition of the substantial uncertainties involved in defining these release fractions, set values were used for generic application to all core-damage conditions resulting in meltdown.

The gap releases calculated in the RSS and other safety studies are dependent upon fuel burn-up and reactor operating conditions, but typically vary from a fraction of a percent to a few percent of the fuel inventory of noble gases (xenon, krypton) and of the most volatile fission products (cesium, iodine, tellurium). The gap release constitutes only a small portion (at most a few percent) of the overall FP release in severe accidents, and can be defined reasonably accurately using available, experimentally well-validated models of fuel behavior. It is therefore not judged to contribute significantly to severe-accident uncertainty, and is discussed no further in this document.

Although accounted for in the RSS and subsequent PRAs by an instantaneous discharge of certain fixed fractions of FP inventory into the containment atmosphere, the "melt release," which commences immediately after the clad failure, will actually continue throughout the in-vessel phase of an accident. The timing of this, the major portion of the in-vessel release, has major implications for the subsequent behavior of radionuclides; some of the more obvious are:

1. Early release of radionuclides precludes the possibility of those radionuclides being trapped, long-term, in the fuel or melt when it is ultimately quenched.

2. Release of radionuclides late in the in-vessel stages of an accident, or ex-vessel, may preclude the possibility of substantial retention in the reactor coolant system (RCS).

3. The concurrent presence in the RCS of substantial quantities of aerosols, along with condensable, vapor-phase fission-product species, may provide an alternative to RCS surfaces for condensation of those species, thereby altering their deposition characteristics.

For these and other reasons, interest in release processes in-vessel has focused since the RSS on providing improved descriptions of the magnitude and the time-dependence of releases both of fission products and of nonradioactive aerosols. Many experiments have been performed, over wide ranges of conditions, in order to establish quantitative models of FP and aerosol-release processes. The state of the art in this respect was reviewed by the USNRC in 1981,[2] leading to the widespread adoption of a model in which fractional release rates of FP and other materials were correlated simply with temperature. The basis of, and uncertainties inherent in, both this type of model and the vaporization process itself are discussed in Subsection 7.1.2.

The qualitative features of release from overheated core materials discovered in experiments performed to date may be summarized as follows:[2]

1. Release rates of all species are strongly dependent on temperature.

2. Volatility is an important factor governing release; thus, in experiments involving progressive fuel heating, it is generally observed that species such as iodine, cesium, and tellurium are released earlier and in larger quantities than refractory materials (e.g., lanthanides and actinides), while there is a group of "medium volatility" fission products (e.g., antimony, silver, molybdenum, barium, strontium) exhibiting intermediate behavior.

3. Release rates of volatile FPs measured in several experiments suggest that, in accidents involving severe core damage, the release of these species during the in-vessel phase of an accident may proceed essentially to completion.

4. The potential exists, based on a very small number of small-scale results, for very large quantities of aerosol to be formed from the more volatile constituents of clad, structural, and control-rod materials during the in-vessel phase of an accident (e.g., tin from Zircaloy; iron, chromium, and manganese from steels and Inconels; and silver, cadmium, and indium from control rods).

Table 7-1 places these observations in the context of the accident progression discussed in Section 3.

Vaporization is the chief process believed to be responsible for the release of FP and other aerosol species from a degraded core. The uncertainties associated with vaporization and aerosolization rates in-vessel are discussed in Subsection 7.1.2. There are other phenomena, discussed in Subsection 7.1.3, which could have a significant effect on release under certain conditions. These include energetic phenomena (such as Zircaloy oxidation and melt/water interactions) with the potential for mechanical fragmentation and associated aerosol formation, as well as leaching, which could in some cases lead to substantial releases of FPs into water.

Table 7-1 Phases of core degradation

Phase	Starting Condition	Anticipated Releases
1	Core uncovering	Insignificant
2	Hottest fuel @ 1300 K	- Gap releases - Increasingly rapid cesium, iodine, tellurium, xenon, and krypton release - Small releases of medium volatiles
3	Hottest fuel @ 2000 K	Increasing, rather uncertain (+/- order-of-magnitude) release rates of volatile and medium volatile species
4	Fuel discharged to lower plenum	Potential for (a) leaching, and (b) energetic interactions leading to gas-borne release

7.1.2 Vaporization Rate Uncertainty

7.1.2.1 Magnitude of uncertainties

As stated above, vaporization, possibly followed by condensation into or onto aerosols, is expected to be the most important phenomenon giving rise to gas-borne fission products and aerosols in the reactor coolant system. A substantial body of experimental data exists on the releases that have resulted from prolonged (several minutes and upwards) heating of irradiated fuels and simulants. Almost all of these experiments have measured releases of cesium, iodine, and tellurium; a much smaller number have measured the (generally much smaller) releases of lower volatility fission products (e.g., antimony, silver, molybdenum, ruthenium, barium, strontium) and actinides. Data on release of control-rod and structural materials are even more scarce.

Experiments have typically been performed under widely different conditions of pressure, carrier gas, temperature, fuel condition and composition, gas flow rates, and more. Two factors have emerged from attempts to draw this data together into a coherent model of release from degraded cores:[2]

- There is, in general, an increasing trend of release rates with temperature and
- When correlated with temperature, the data exhibit a very large amount of scatter.

Figure 7-1 shows the release rate data for iodine that were collected in Reference 2 plotted along with some more recent data as a function of temperature. The data is presented in the form of a fractional release rate coefficient (k) which relates the release rate $\dot{Q}$ to the remaining inventory Q of a species in an element of the core ($\dot{Q} = k\ Q$). The simple model proposed in Reference 2, in which these fractional release rate coefficients are correlated with temperature, forms the basis of computer codes for estimating release histories of fission products and nonradioactive aerosols from degraded cores. These codes were developed by the USNRC[3] and the UKAEA.[4] However, the IDCOR program has developed a different approach.[5]

In order to place this data in perspective, it should be borne in mind that the duration of in-vessel releases (approximately between core uncovering and vessel failure) will be from a few minutes to at most around 100-200 minutes. Thus in order to obtain "significant" releases* of volatile radionuclides it is necessary for the fractional release rate coefficients plotted in Figure 7-1 to attain values on the order of 10^{-3} or 10^{-2} min^{-1} or higher. When values lower than these are excluded from the figure, it becomes difficult to see any clear trend in the data--in particular, the data at very high temperatures are very sparse.

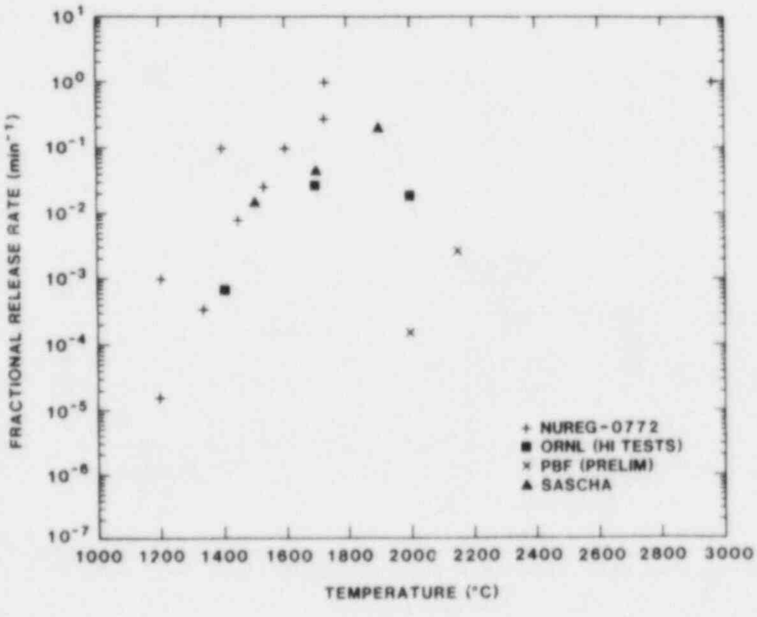


Figure 7-1 Release rate data for iodine.

*Significant releases are loosely defined as around 10% or more, based on reported calculations of the sensitivity of the radiological consequences to the magnitude of releases consisting primarily of cesium, iodine, and tellurium.[6]

It is clear from this figure that iodine release rates calculated using models assuming a simple correlation between release rate and temperature are subject to at least plus or minus order-of-magnitude uncertainty. The data for other volatile FPs (tellurium and cesium) support a similar conclusion. The data acquired since publication of Reference 2 have not permitted that report's estimate of order-of-magnitude release-rate uncertainty to be narrowed.

With regard to lower volatility fission products and nonradioactive aerosol materials (from cladding, structure, and control rod components), far fewer data are available. The uncertainty inherent in applying the model of Reference 2 to these materials is therefore, as stated in Reference 2, at least as great as that for the volatile fission products--i.e., the fractional release rate coefficients to be used are uncertain by at least plus or minus an order of magnitude.

The implications of order-of-magnitude uncertainties in release-rate coefficients are discussed in Subsection 7.1.2.2. Sources of these uncertainties, and the means by which they might be reduced, form the subject of Subsection 7.1.2.3.

7.1.2.2 Implications of release-rate uncertainties

This subsection demonstrates the impact of order-of-magnitude variations in release-rate-coefficient values (around a "base case" currently in use in USNRC source-term studies[3]) on in-vessel release histories, with the aid of a series of calculations using a computer code based on the CORSOR algorithm.[3] This method uses the release-rate-coefficient approach developed in Reference 2. Release histories of three groups of materials are considered:

1. Volatile fission products (cesium, iodine, tellurium, etc.)

2. Medium-volatile fission products (barium, molybdenum, strontium, ruthenium, antimony, etc.)

3. Total aerosol mass

Release coefficients are not the only parameters that influence these histories; using the current model (based on the CORSOR code developed at BCL[3]), the core thermal history is the other key factor. Sensitivity of release histories to coefficient values therefore needs to be investigated across a range of thermal histories corresponding to that possible in light of the uncertainties discussed in Section 3. Such sensitivity studies, albeit across some subset of that range, have been performed as part of the QUEST program at SNLA,[7] and many of the results quoted here are excerpted from that study.

The basis for order-of-magnitude (or greater) variation in release-rate coefficients was outlined in Subsection 7.1.2.1; a brief explanation must be given of the range of thermal histories used in these calculations. Thermal histories were calculated using the MARCH code.[8] Three histories were used in these calculations, all performed for a TMLB' accident at the Surry plant:

1. A "base case" using inputs corresponding to the most recent USNRC source-term studies;[3]

2. A "high case," in which MARCH input parameters were adjusted so as to prolong the time spent by the core at high temperatures; and

3. A "low case," in which input parameters were adjusted so as to minimize temperatures during and duration of the in-vessel phase of the accident.

Further details of these calculations are described in Reference 7. One other parameter concerning thermal history has been varied; it is the maximum temperature specified in the CORSOR algorithm, above which temperatures obtained from MARCH are assumed to be spurious and to which such temperatures are reduced. The first two cases (base and high) used a value of 2760°C, the third (low) case used a value of 2300°C for this parameter.

The results of the calculations are presented in Figures 7-2 through 7-4 and Tables 7-2 through 7-4. The results are discussed below for each of the three groups of species considered.

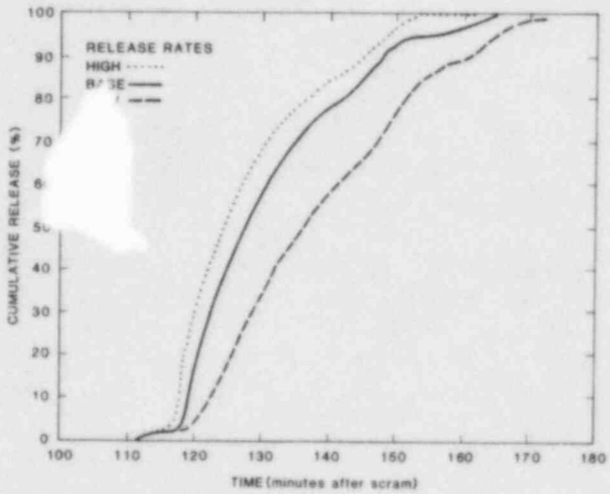
Volatile Fission Products

Figure 7-2 shows the sensitivity of iodine release history to release-rate-coefficient variations for each of the three thermal histories considered. Table 7-2 summarizes the total releases during the in-vessel phase for each of the 9 combinations of release-rate and thermal-history assumptions for iodine and the other radiologically important volatile FPs, cesium and tellurium.

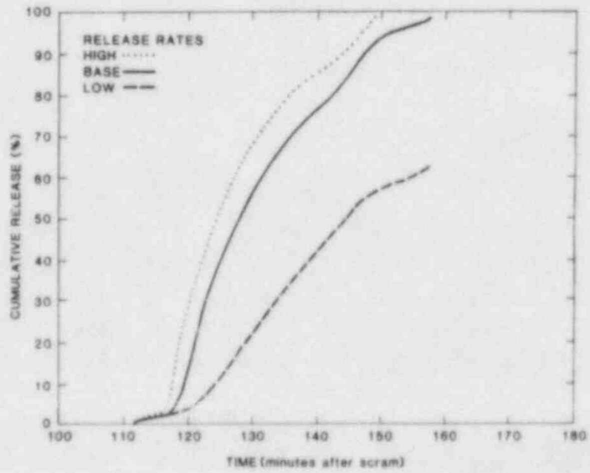
For each species, the results are qualitatively similar. There exists a portion of the uncertain parameter space in which in-vessel releases are close to unity, and a portion in which they are not. While the upper values are clearly the maxima attainable, the lower values do not represent extreme bounds; neither the thermal history nor the release rate coefficient ranges have been "pushed" to the lowest defensible values. It does not require both low release rates and low thermal histories to reduce release fractions significantly. Either alone may effect a substantial reduction, while together the effect is very large. These results are specific to a TMLB' accident at the Surry plant, but their qualitative features are probably of general applicability.

Given the present state of knowledge, the key implication of these results is that we cannot be assured that the bulk of the volatile fission products will be released during the in-vessel phase of a TMLB' accident at the Surry plant. This uncertainty can probably be extended to many, if not most, severe LWR accidents. The implications for release from the plant depend on the subsequent fate of the melt and the containment. Some of the more obvious possible implications are:

(a)
"High" thermal
history



(b)
"Base" thermal
history



(c)
"Low" thermal
history

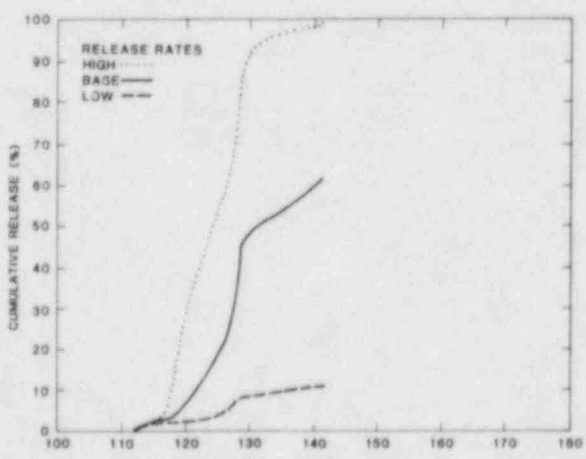


Figure 7-2 Iodine release histories for different release rate/thermal history combinations.

Table 7-2 Sensitivity of volatile fission products in-vessel release fractions to release rate and thermal history assumptions

(a) Iodine

Thermal History	Release Rate Coefficients High	Base	Low
High	1	1	0.99
Base	1	0.99	0.62
Low	1	0.62	0.11

(b) Cesium

Thermal History	Release Rate Coefficients High	Base	Low
High	1	1	0.99
Base	1	0.99	0.64
Low	1	0.63	0.14

(c) Tellurium

Thermal History	Release Rate Coefficients High	Base	Low
High	1	0.90	0.78
Base	1	0.33	0.22
Low	0.87	0.03	0.02

1. If ex-vessel release (during melt discharge, or via steam explosions or core/concrete interactions) can be avoided, the release of volatile radionuclides into the containment atmosphere or from the plant may be substantially reduced in comparison with WASH-1400 assumptions.

2. If in-vessel releases of volatile species fall in the lower end of their possible ranges, there will be no possibility of large attenuation factors within the reactor coolant system, because no matter how efficient the deposition and retention processes therein, substantial fractions of these species could still be trapped in the melt and core debris when it exits the reactor vessel.

Table 7-3 Sensitivity of medium-volatile fission products in-vessel release fractions to release rate and thermal history assumptions

(a) Barium

Thermal History	Release Rate Coefficients High	Base	Low
High	0.99	0.58	0.09
Base	0.78	0.18	0.02
Low	0.18	0.02	2E-03

(b) Molybdenum

Thermal History	Release Rate Coefficients High	Base	Low
High	0.89	0.22	0.02
Base	0.58	0.09	0.01
Low	0.17	0.02	2E-03

(c) Antimony

Thermal History	Release Rate Coefficients High	Base	Low
High	1	0.96	0.42
Base	0.96	0.50	0.07
Low	0.46	0.06	7E-03

(d) Ruthenium

Thermal History	Release Rate Coefficients High	Base	Low
High	0.26	0.03	3E-03
Base	0.07	7E-03	7E-04
Low	5E-03	5E-04	5E-05

Table 7-4 Sensitivity of in-vessel iron release to release rates and thermal history assumptions. All releases are expressed in kilograms.

Thermal History	Release Rate Coefficients High	Base	Low
High	18 000	2300	240
Base	6100	660	66
Low	720	73	7.3

3. If containment survives, or retains its integrity long enough beyond the releases (whether from in-vessel or ex-vessel processes) into its atmosphere of volatile radionuclides for those species to have been in large measure depleted from the atmosphere, then uncertainties in in-vessel release and timing will have a relatively small direct influence on long-term airborne radioactivity levels in containment.

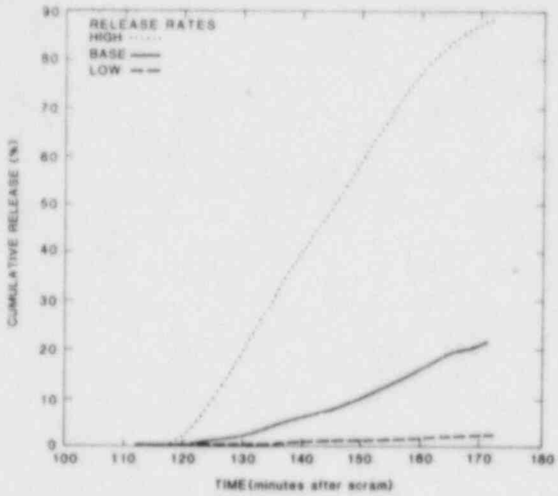
Medium-Volatility Radionuclides

Figure 7-3 is the analogue of 7-2, excepting only that it refers to fission-product molybdenum. Table 7-3 summarizes the total in-vessel release of barium, molybdenum, antimony, and ruthenium for each of the nine thermal history and release rate assumptions considered.

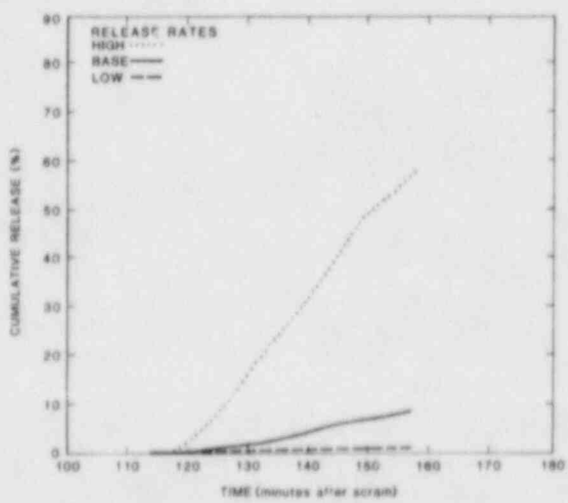
The qualitative results are similar to those for the volatile nuclides. In each case, there is a part of the parameter space leading to substantial (in some cases virtually complete) release fractions, while in other parts of the space release fractions are very small. The transition between the two varies from nuclide to nuclide; it is "easier" to postulate high releases for antimony and barium than for ruthenium. In general, thermal history and release rate assumptions must be biased farther towards the "high" end of their ranges to produce large releases than is the case for the volatile species cesium, iodine, and tellurium.

The chief implication of these results lies in the potential for larger releases than were considered in the RSS, and hence in the potential (given little retention in RCS and containment) for more radiologically damaging source terms than have been considered in off-site consequence studies to date. Again, the results are specific to a TMLB' accident at Surry, but can probably be generalized in a qualitative sense. Likewise, the uncertainties will have little direct impact on source terms if most of the materials released in-vessel are in any case trapped within the plant.

(a)
"High" thermal
history



(b)
"Base" thermal
history



(c)
"Low" thermal
history

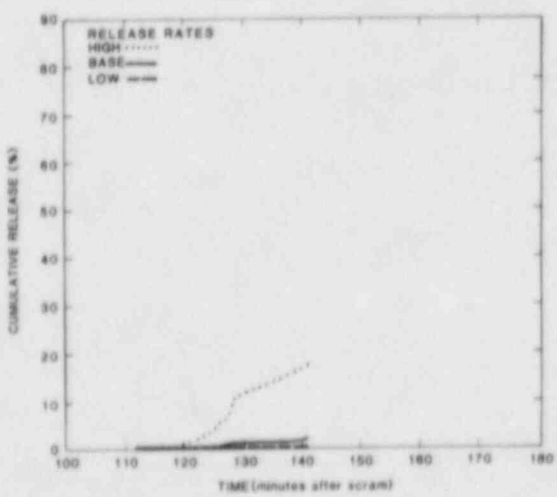
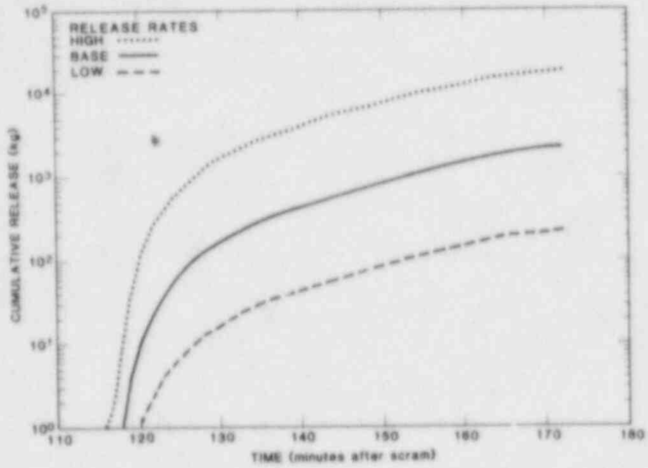
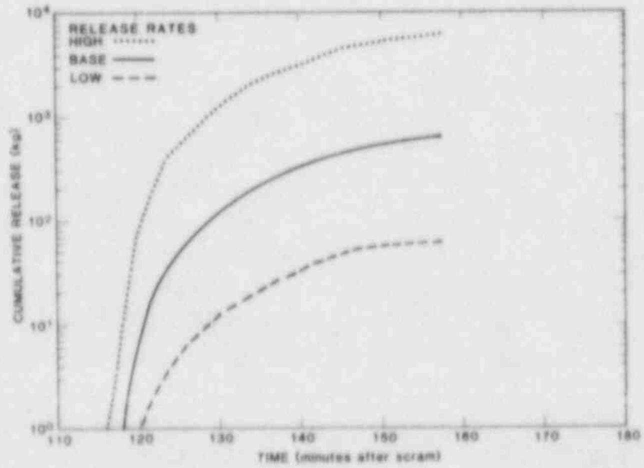


Figure 7-3 Molybdenum release history for different release rate/thermal history combinations.

(a)
"High" thermal
history



(b)
"Base" thermal
history



(c)
"Low" thermal
history

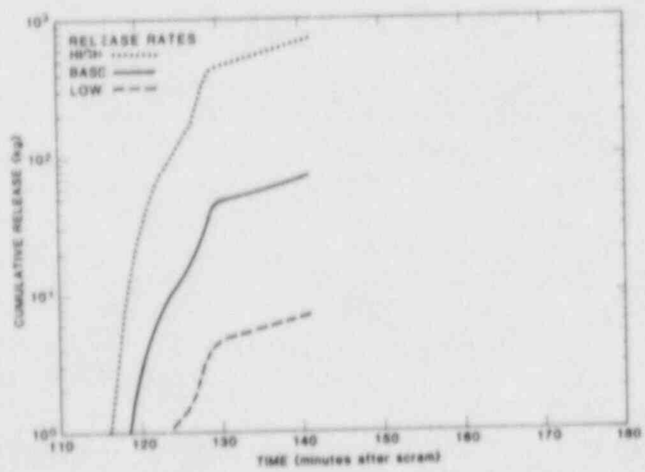


Figure 7-4 Iron aerosols release history for different release rate/thermal history combinations.

Aerosols

The release history of iron is used to illustrate the release histories of nonradioactive aerosols. Figure 7-4 shows the release histories (expressed as absolute mass) of structural iron corresponding to order of magnitude release rate variations for each thermal history. Table 7-4 summarizes the total cumulative in-vessel iron release data for each combination of release rate and thermal history assumptions.

The range of releases is very wide--again, this cannot be generalized and the range in no sense represents bounds. The important general, qualitative point is that the range is large.

Large masses of aerosol in-vessel may either mitigate or exacerbate releases of radionuclides from the plant. Mitigation could occur by

1. Enhanced agglomeration and settling of aerosols and associated radionuclides within the RCS, or

2. A similar effect, but on a longer time scale, in the containment atmosphere.

On the other hand, exacerbation could be caused by

1. Trapping on aerosols (which may remain airborne or deposit and later be resuspended) of condensable fission-product species (e.g., Te, CsOH) which might otherwise be trapped semipermanently (i.e., on a time scale much longer than the accident duration and any releases from the plant) on RCS surfaces, or

2. Increased threats to safety-related equipment (e.g., coolers, filters, or sprays if operating) arising from high particulate loadings in the containment atmosphere.

Of all the release ranges predicted using the CORSOR algorithm, these are the most difficult to defend. The high values (several tons) appear intuitively unlikely; simple bounding calculations based on limitations imposed by mass transfer into the gas phase have resulted in upper bound estimates around 100 g/s of structural material vaporization for an entire LWR core, suggesting an upper bound for overall releases on the order of hundreds of kilograms rather than several tons.[4] The calculations quoted in Reference 4 do, though, lend tentative support to the lower bounds estimated here.

The implications of the large uncertainty in aerosol release are difficult to judge in view of the conflicting "good" and "bad" effects of large aerosol loads. Once again, this uncertainty will be immaterial if containment integrity is assured. The absence of a large in-vessel aerosol source might preclude the trapping of aerosol fission products inside the RCS by agglomeration and settling (the chief aerosol retention mechanism identified in Reference 3), but might permit more efficient trapping of volatile species on RCS surfaces. The complexity of the aerosol and vapor phenomena in the RCS make the effects of a reduction in this uncertainty very difficult to evaluate. Structural aerosol release, though, is an area where it should be possible to narrow the present very wide uncertainty range with the aid of more mechanistic bounding calculations.

7.1.2.3 Vaporization rate uncertainties--discussion

The available data on the release rates of fission products from UO_2 show a wide scatter of results. The ranges are wide enough to lead to potentially very important uncertainties in the in-vessel releases of fission products and nonradioactive aerosols. Recent and current experiments do not appear to be narrowing the range of release rates we can anticipate in severe accidents. Given these three factors, it is reasonable to consider whether or not the uncertainty in in-vessel release predictions can readily be reduced. This section provides a brief discussion of the mechanism of release by vaporization* followed by a discussion of the feasibility of reducing release rate uncertainties. A fuller discussion of vaporization processes and rates is given in Reference 9.

Like many of the chemical processes that affect radionuclide release and transport, vaporization is a heterogeneous phenomenon--that is, it takes place at the boundary between two distinct phases. As such, it is susceptible to rate control by any or all of a large number of processes, which may conveniently be grouped as

1. Transport and reaction in the condensed phase,
2. Heterogeneous reaction at the phase boundary, and
3. Transport and reaction in the gas phase.

The driving force for vaporization is in general provided by the existence adjacent to the condensed phase surface of an equilibrium vapor pressure (controlled by step 2 above) of the evaporating species greater than its vapor pressure in the bulk gas. The rate of evaporation may then be controlled by transport, generally by some mixture of molecular and convective diffusion, of the gaseous species away from the surface into the gas (step 3 above). However, the rate of the transition between condensed and gaseous phases may not be fast enough to maintain the equilibrium vapor pressure adjacent to the surface, in which case the vapor pressure next to the surface will drop and step 2 above will also play a limiting role. The third possibility (step 1 limiting) arises if the activity** of the condensed phase species at the boundary falls as a result of evaporation at a rate greater than

---

*This should not be confused with the RSS usage of the term "vaporization release" that is applied solely to the release arising ex-vessel from core/concrete interactions.

**"Activity" is used here in the chemical thermodynamic sense: as a measure of the effective concentration of a material in the phase under consideration.

that which can be balanced by transport through the condensed phase to the interface.

Different processes may be responsible for rate limitation under different conditions. At lower temperatures, we can be confident that condensed phase transport is limiting, for even the very volatile noble gases are released at rates enormously lower than would be expected if they could sustain anything approaching their equilibrium vapor pressures at fuel surfaces. Preliminary calculations have suggested that gas-phase transport may be an important limiting factor in some conditions, at least for structural and control-rod materials releases.[9 10] Because both condensed phase diffusion rates and vapor pressures will generally rise faster with temperature than will gas-phase transport rates, limitation by gas-phase transport would be expected to become relatively more likely at higher temperatures.

Moreoever, the different processes described above are each subject to control by different factors. Table 7-5 summarizes the major dependencies anticipated for each of the three groups of processes described above on some of the more obviously important parameters (not an exhaustive list).

In view of the large number of parameters capable of influencing release rates, yet not accounted for in the simple temperature correlative model described and used in Subsection 7.1.2.2, it seems reasonable to assume that the scatter in experimental data when correlated with that model arises not from random variations in release rates but from systematic differences between experiments with regard to the other potentially important parameters. Reference 2 went some way towards explaining anomalously high and low results in terms of such differences, but no systematic effort has been made to incorporate all of these effects into a mechanistic release model that would enable uncertainties in release rates to be reduced. Reference 2 discussed this problem, but concluded that the quality and extent of the available data would not support such an effort. More mechanistic scoping calculations of release phenomena are, though, beginning to be developed.[11]

In the absence of a quantitative formulation of the effects of the processes and parameters outlined in Table 7-5, it is natural to inquire as to whether or not we can make at least qualitative predictions as to the effects of various parameters. For example, in a transient accident in a PWR with the RCS at high pressure, we can be reasonably assured of relatively low net upward gas flow rates and high system pressures in the core throughout the in-vessel release phase of the accident. Would such conditions cause higher or lower release rates than those predicted using current models?

Table 7-6 summarizes, in a qualitative manner, the effects that changes in each of the parameters listed in Table 7-5 would have on each of the processes involved. There are some cases in which effects are clear: increasing temperature will always favor release, as will

Table 7-5 Dependence of release-rate processes on system parameters

Parameter	Process Influenced: Condensed Phase Transport	Heterogeneous Chemistry	Gas-Phase Transport
Temperature	X	X	X
Pressure		X	X
Condensed phase composition	X	X	
Condensed phase effective surface area	X	X(1)	
Fuel history and condition	X	X	
Condensed phase geometry	X	X(1)	X
Gas composition		X	X
Gas flow rates			X

X Denotes a significant influence--probably order of magnitude or greater for at least some species over the range of parameters pertinent to severe LWR accidents.

(1) Denotes an effect on kinetics only--other parameters affect thermodynamics also.

increasing surface-area-to-volume ratios of the condensed phases. There are several other cases, though, where effects could lie in opposite directions for two different processes, or for a single process but for two different species. For example, increasing steam pressure and steam/hydrogen ratios would favor evaporation of a metal oxide (MO_x) to a vapor-phase hydroxide:

$$MO_{x(c)} + y \cdot H_2O_{(g)} \rightleftarrows MO_{x-y}(OH)_{2y(g)}$$

$$\text{e.g., } MoO_{3(c)} + H_2O_{(g)} \rightleftarrows MoO_2(OH)_{2(g)}$$

but would hinder reduction of such an oxide to a metal vapor:

Table 7-6 Qualitative effects of changing system parameters on release-rate processes

Parameter Change	Process Influenced		
	Condensed Phase Transport	Heterogeneous Chemistry	Gas-Phase Transport
Increased temperature	+	+	+
Increased pressure		+/-	-
Condensed phase composition, e.g., increased O_2 potential	+/-	+/-	
Increased effective surface area of condensed phase	+	+	
Fuel condition, e.g., increased burn-up & cracking	+	+/-	
Condensed phase geometry, e.g., increased surface/volume ratio	+	+	+
Gas composition, e.g., increased H_2/H_2O ratio		+/-	+
Increased gas flow			+

KEY

- \+ Denotes rate increase
- \- Denotes rate decrease
- +/- Could go either way, depending on the species considered and other conditions

$$MO_{x(c)} + x \cdot H_{2(g)} \rightleftarrows x \cdot H_2O_{(g)} + M_{(g)}$$

$$\text{e.g.,}\ BaO_{(c)} + H_{2(g)} \rightleftarrows Ba_{(g)} + H_2O_{(g)}$$

In such cases, no generalizations are possible.

Systematic evaluation of release mechanisms and likely rate-controlling processes might enable a considerable advancement to be made in our ability to make such qualitative predictions, without recourse to further experimentation. For example, careful evaluation of the available data might enable regions to be defined in which each of the three processes (condensed phase transport, heterogeneous reaction, and gas-phase transport) of Tables 7-5 and 7-6 are dominant. The regions may be very different for different chemical species. Another example might be in determining the effects of chemical changes in the system on the phase equilibria. If key reactions involving a phase transition can be defined for all of the condensed and vapor phase species of interest, then it will become a simple matter to predict the effects of pressure and gas or solid composition on vaporization rates.

It must be emphasized that many of the parameters in Tables 7-5 and 7-6 are themselves subject to appreciable uncertainty. In order to narrow release rate uncertainty it is therefore necessary both

a) to determine their effects on release rates, and

b) if (a) is significant, to narrow uncertainty in the other controlling system parameters.

Thus it seems unlikely that release rate uncertainties can be narrowed in the near future using existing models and data. There is, though, no reason to believe that these uncertainties cannot be reduced. If their reduction is felt to be desirable, then two approaches might be adopted:

1. An analytic investigation into the bounds on release rates imposed by the factors outlined in Table 7-5 could be undertaken. It is possible that in some cases, if not many, (particularly those of structural and control-rod aerosol release) that this alone would lead to an enormous narrowing of the existing bounds on release rates.

2. A longer term, systematic experimental program could be established to investigate release rates and mechanisms. Regions of dominance of the different processes in Table 7-5 could be defined (at least in part based on existing knowledge) and systematic investigation into the effects of these and other potentially important parameters could then be performed.

In summary, uncertainties in release rates due to vaporization arise both from the limited understanding of the release processes themselves and from uncertainty in the system boundary conditions (temperatures, pressures, etc.) prevalent in severe accidents. With the present state of knowledge it is not possible to make generalized predictions as to the likelihood that different system parameters will lead to releases in different parts of the possible ranges.

There is appreciable scope for reduction of in-vessel release uncertainty, both by

1. A systematic analytic investigation of release phenomena in a mechanistic manner, and

2. Experimental programs aimed at an improved mechanistic understanding of release phenomena and the effects of a wide range of system parameters thereupon.

7.1.3 Release Phenomena other than Vaporization

Although vaporization from overheated core materials is expected to be the most important mechanism of radionuclide release in-vessel, there are other release processes that could be significant in particular circumstances. Two such are considered here: leaching into water, and fuel dispersal by mechanical phenomena.

7.1.3.1 The leaching process

Leaching of radionuclides from UO_2 or frozen mixed melts into water may occur either as the result of core reflood or when the molten core slumps into the lower plenum (Phase 4, in the terminology of Section 3). In either case, the contact between hot core materials and water may result in an energetic melt/water interaction, which by dispersing the core materials and/or fracturing the debris may enormously increase the surface area available for (and hence the rate of) leaching. In either case, the configuration and quenchability of the core are key contributors to uncertainty as to the quantity of radionuclides leached and the rate at which they leach into primary-system water. There are some accident sequences (e.g., a large break with failure of ECCS) where the circuit is essentially dry; in such situations leaching is clearly unimportant.

The importance of leaching will depend not only on the quantities of radionuclides leached into primary-system water but also on the subsequent fate of that water. If the water is ejected fro the reactor vessel at low pressure as a liquid, or if it evaporates and leaves the FPs contained within it plated on the vessel walls, it may provide a permanent sink for FPs. On the other hand, if it is ejected from the circuit at high pressure, or if the plated FPs become gas-borne as surfaces heat up after evaporation of the water in-vessel, it may add a new and different FP source term to the containment atmosphere. Even if it does not provide a permanent sink for radionuclides, leaching into water may have an important impact on both the timing and the form of the release from the vessel into the containment. The ultimate disposition of residual water in the vessel is a contributor to uncertainty in the impact that the leaching release will have.

The actual process of leaching will be very much analogous to that of vaporization, except that the gas will be replaced by liquid water and the vaporization and gas-phase mass-transport processes by dissolving and liquid-phase transport, respectively. Generally speaking, fuel-debris temperatures will be relatively very low, so that transport through the condensed phase will be very slow. This may be compensated for, though, by an enormous increase in effective surface area, not only by macroscopic debris fragmentation during an FCI but also by extensive microscopic fracturing of surfaces. To a very crude approximation, those FPs that display a strong affinity for the solid, oxide phase rather than for the gas (lanthanides, actinides, and some transition elements) will also be very insoluble in water, so that the FPs that dominated the gas-borne release earlier in the accident will also tend to dominate the leaching release. Experiments on the leaching of irradiated fuel (albeit aged and under much less extreme temperature conditions) have consistently resulted in the dissolving of only trace quantities of species other than cesium and strontium isotopes.[12] The possibility remains, though, that some species not readily volatilized may possess appreciable solubility under reactor-accident conditions. Solution chemistry will contribute to uncertainty in the leaching release, but it is not regarded as a major contributor to the source term uncertainty.

7.1.3.2 Mechanical disruption and/or fragmentation of core materials

There are at least two possibilities for energetic and mechanically disruptive processes to occur in the reactor vessel during the course of an accident involving severe core damage. First, there is the possibility of energetic melt/water interactions, both upon reflood of the core and upon slumping of the core into a water-filled lower plenum. Second, the oxidation of Zircaloy cladding may be so rapid and energetic as to generate particles of fuel and clad as a smoke. The impact of these processes on source terms is potentially twofold: First, they may directly render large quantities of material gas-borne; and second, by changing the configuration of the fuel, they may have a dramatic impact on the rates of vaporization from it.

Reflooding of the core during Phase 3 of fuel damage, based on the experience of the TMI-2 accident, appears likely to lead to fairly coarse fragmentation of fuel and cladding by thermal shock, and therefore is not a major source of aerosolized material. Caution must be used, though, in interpreting a single result in this way; with the possibility of both fragmentation and enhanced Zircaloy oxidation rates through an increased supply of steam, there is clearly the potential for very violent interactions during reflooding.

The bulk of LWR-related FCI work has focused on the problem of estimating the energy that will be released as mechanical work when the molten core slumps into residual water in the lower plenum. It appears that this may be to some extent a stochastic variable; the factors influencing it are discussed in Subsection 3.4. The interaction may be extremely energetic, even if not sufficiently so as to fail the reactor vessel, and may produce a large, virtually instantaneous source of fuel, FP, and structural-material aerosols and vapors, both of volatile and nonvolatile species.

The extent and nature of direct aerosolization by melt/water interactions (steam explosions in particular) has never been measured. Several measurements of debris sizes produced by steam explosions have suggested that the gas-borne debris will be in the form of rather large particles.[13] More recently, though, UKAEA investigators discovered some 1% of the melt mass suspended as a fine colloid (number distribution peaked at sizes around 1 μm) in water examined after a steam explosion experiment.[14] The general absence of data on the effects of steam explosions on FP and aerosol release, and the difficulty in extrapolating that available into the regimes of interest, leads to a large and potentially important uncertainty in this area.

As regards the other effect mentioned above, of increased release rates arising from finer fragmentation of the fuel materials, there are experimental data obtained in the Power Burst Facility which suggest that this effect may be substantial.[15] Again, though, the data are very sparse, and it is difficult even to scope out the range of effects this might have. Generally, though, melt/water interactions will affect FP release by some at present uncertain balance between

- heating or cooling the fuel (itself subject to gross uncertainty - see Subsection 3.3),
- altering its configuration, and
- providing new mechanisms for release.

As discussed in Section 3, the reaction of Zircaloy cladding with steam is very exothermic and becomes rapid enough by about 1575 K (1300°C) to be the driving process in core degradation. At higher temperatures, the reaction becomes extremely rapid; given an adequate supply of steam, its rate is comparable to that of Zircaloy with air. This comparison suggests that the reaction of Zircaloy with steam will be of the nature of a burning process. The extent to which this might contribute directly to FP and aerosol release in-vessel is uncertain. Certainly, the mechanics of Zircaloy melting and liquefaction may have a dramatic effect on the fuel and core configuration (Section 3); thus Zircaloy reaction uncertainties will make at least a significant indirect contribution to release uncertainty.

7.1.4 In-vessel Release Phenomena--Summary

Releases of fission products and other aerosols in-vessel are subject to substantial uncertainties stemming from both a) uncertainties as to evaporation rates from overheated core materials and b) the extent of release via other mechanisms (violent Zircaloy oxidation or melt/water interactions). The contributors to these uncertainties are summarized in Table 7-10, presented as part of the summary, Subsection 7.7. The major implications of these uncertainties are as follows:

1. Substantial fractions (tens of percent or more) of the inventory of volatile radionuclides (cesium, iodine, tellurium) may remain in the core during the in-vessel phase of an accident. Substantial quantities of these FPs may therefore be available for release ex-vessel, without hold-up in the RCS.

2. In-vessel evaporation of medium volatility radionuclides (e.g., strontium, barium, antimony, molybdenum, ruthenium) may lead to substantially larger releases of these materials than have been considered in PRAs to date.

3. Energetic events (melt/water interactions, Zircaloy combustion) may add to uncertainties in evaporation rates by altering dramatically the configuration of the core and/or melt. Further, they may provide a mechanism directly to aerosolize refractory materials, including highly radioactive lanthanides and actinides, which would not be vaporized to a significant extent.

4. The range of possible in-vessel releases (via vaporization) of nonradioactive aerosols is very wide. This has major implications both for RCS retention of fission products and for subsequent aerosol and FP loads discharged from the RCS.

7.2 Transport and Retention in the Reactor Coolant System

This subsection will be divided into three parts, dealing with three major groups of factors which together determine the extent of radionuclide retention in the reactor coolant system (RCS):

1. System thermal-hydraulic "boundary conditions": temperature, pressure, flow rate histories, etc.

2. Fission product chemistry: For those materials released in vapor form and not immediately condensed to aerosols, this will determine the partitioning between gas phase, aerosols, and RCS surfaces.

3. Aerosol behavior: Many fission products will be present as particulates; their behavior and that of condensable (or otherwise trappable) vapor FPs will be closely tied to the behavior of the entire aerosol mass, including that of nonradioactive materials, in the RCS.

These three areas are discussed in turn, along with four important areas of overlap.

7.2.1 Thermal-Hydraulic "Boundary Condition" Uncertainties

Many of the uncertainties relating to gas flow and heat transfer in the RCS have already been discussed in Section 3. This section discusses the implications for fission product retention in the RCS of two important such groups of uncertainties. The first concerns the flow pathways by which gases, liquids, and associated fission products leave the RCS. The second concerns heat and mass transfer to RCS surfaces. It discusses the implications both of fission-product deposition for surface heating and of the important natural circulation effects discussed in Section 3 for FP deposition and evaporation rates.

7.2.1.1 RCS fluid pathways

It is clear that the locations of breaks in the RCS will be among the chief factors determining flow paths out of the system. Given exact knowledge of break locations, there may still be some uncertainty, for example, in PWRs, as to the fraction of the flow through the break which arrives there via the broken loop and that which arrives via intact loops.

These matters have received significant consideration recently in the contexts both of thermal hydraulics and of FP transport.[3 16 17] At present, systems analyses used in PRAs do not provide conditional probabilities of different system break locations; this alone may give rise to appreciable uncertainty when consideration of FP and aerosol retention in reactor coolant systems is introduced into risk analysis.

A few specific examples of pathway uncertainties are listed below:

- What are the frequencies of LOCAs for different points in the RCS?
- Will relief valves stick open or reseat in overpressure accidents (e.g., some transients) - and might their sticking closed lead to alternate modes of pressure relief (rupture disks or RCS failure)?
- What is the probability that a transient overpressure will induce a system break?
- What fraction of flow to a (PWR) break reaches it via intact loops?
- For interfacing systems LOCAs, does the break occur upstream or downstream of the large surface areas presented by certain auxiliary systems--for example, the residual-heat-removal (RHR) system?

The IDCOR program has delineated, in a systematic manner, a large number of possible in-plant fission-product transport pathways.[5] Such analyses are essential precursors to efforts to reduce flow pathway uncertainty.

In addition to these pathway uncertainties, which are all associated with the time periods corresponding to the loss of coolant and core uncovering, there are uncertainties associated with gas, water, and melt transport out of the RCS at later stages of the accident, after substantial core melting has occurred (i.e., at Phase 4 and beyond, in the nomenclature of Section 3). Two examples are

- Vessel failure by in-vessel steam explosions (discussed in Subsection 3.5) and
- Vessel melt-through, when the steel RPV is attacked by molten or unquenched core debris.

These examples involve uncertainty in the timing and nature of vessel breach and hence, in the nature and magnitude of discharges from the RCS.

7.2.1.2 RCS heat and mass transfer

Recent studies have examined the nature of RCS flows during severe accidents in some depth.[7 16 18] It appears that, for many accidents of interest, flow regimes will be dominated by natural, rather than by forced convection. The corollary is that, in general, conditions in the RCS will differ from those assumed in typical current models in several important respects:

- Gas velocities over surfaces will be considerably higher,
- Heat and mass transfer will be substantially more rapid, and
- RCS gas flows will be turbulent, rather than laminar.

These observations are all more applicable to PWRs than BWRs (see Section 3). Some of the important uncertainties that such conditions introduce into fission product transport analyses are discussed below. A fuller discussion can be found in Appendix B to the QUEST study.[7]

Enhanced gas movement around the RCS will have effects that may either enhance or diminish FP and aerosol retention in the RCS. On the one hand, there will be an increased tendency for FPs to be transported to cooler parts of the system, where the potential for vapor deposition and sorption is greatest. On the other, cooler parts of the system will be heated up more rapidly, so that long-term deposition may be precluded, and revaporization may be possible.

Enhanced heat and mass transfer coefficients will have a similar, two-edged effect. The under-estimation of deposition rates calculated for condensable fission products caused by the neglect of natural convection effects is very marked (2-3 orders of magnitude or greater[7]). The same applies, though, to heat transfer rates, and the combined effects of enhanced convective heat transfer to surfaces and heating by the decay of deposited fission products may be sufficient to preclude long-term condensation of compounds such as CsI and CsOH on RCS surfaces.

Enhanced RCS atmosphere turbulence will have a marked effect on aerosols. Certain deposition rates (due to turbulent diffusion and inertial deposition) will be increased. The most marked effect, though, may be on turbulent agglomeration[7]--rates of particle growth and settling may be greatly enhanced.

In general, the consideration of these phenomena, without the existence of consistent models to evaluate their many and intimately coupled effects, serves to widen the range of possibilities associated with FP transport and retention within the RCS. For example, for a TMLB' accident at the Surry plant, natural circulation effects within the reactor vessel have been estimated to have the potential to enhance mass-transfer-limited vapor-deposition rates (and hence upper bound retention fractions for volatile FP species) by two to three orders of magnitude.[7] However, the same effects responsible for enhanced mass transfer will produce enhanced heat transfer, so that a) temperatures will tend to be equalized more rapidly, removing the driving force for these natural convection effects, and b) deposition surfaces will heat up more quickly, lowering the potential for volatile species deposition.

This example illustrates the order-of-magnitude effects that these phenomena may have on retention in the RCS. While such scoping calculations lead, in the short term, to the widening of uncertainty ranges associated with FP retention in the RCS, they should provide the stimulus for the development of improved models which, in the longer term, should afford genuine reductions in those uncertainties.

7.2.2 Fission Product Chemistry (Molecular Species)

The volatile fission products cesium, iodine, and tellurium have always been regarded as among the most important contributors to radiological source terms from LWRs. The chemical compounds that these elements are expected to form under severe accident conditions will be gaseous at the temperatures of the hotter parts of the RCS (particularly in the vessel region). As these compounds are carried over cooler surfaces they may condense, either onto RCS structures or onto gas-borne particulates. Even at higher temperatures, too high for condensation, there are a number of possible chemical reactions that may lower these species' vapor pressures over RCS surfaces, and lead to their deposition thereupon.

Vapor deposition involves a heterogeneous reaction, exactly analogous to those discussed in Subsection 7.1.2.3 in the context of release by vaporization. The driving force for deposition is provided by thermodynamics favoring the condensed phase, rather than the gaseous phase. The rate of deposition may be controlled by transport through the gas, by the heterogeneous process itself, or by physical or chemical processes in the condensed phase. A particularly important issue concerning kinetics is the rate of deposition onto RCS surfaces relative to that onto aerosols. Unless aerosol concentrations are very small, gas-phase mass transfer to aerosols will generally be much more rapid than that to surfaces. On the other hand, aerosol temperatures will generally be rather higher, and their compositions will be different from those of the RCS structures. Further discussion of the vapor/aerosol reaction is provided in Subsection 7.2.3.3.

The chemistries of the gaseous compounds of iodine, cesium, and tellurium likely under severe accident conditions are discussed below. The implications for both short- and long-term retention of these fission products in the RCS are discussed. The discussion is biased towards interactions with RCS structural materials. Reactions with aerosols are also uncertain; their implications are described in Subsection 7.2.3.3.

7.2.2.1 Iodine chemistry

Cesium iodide has been predicted to be the most likely chemical form of iodine to exist in severe accident conditions in the RCS.[2 19 20] Recent experiments, though, suggest that the possibility of formation of more stable cesium compounds, accompanied by a volatile iodine species, should not be ruled out.[21 22] The more volatile iodine compounds include HI, I_2, FeI_2, and various zirconium and transition metal iodides--though these may react with structural alloys to form less volatile nickel and chrome iodides. The discussion here, though, centers on CsI as the most probable chemical form.

A number of experiments have been performed on the interaction of gaseous CsI with RCS structural alloys (principally Inconel-600 and 304-stainless steel).[21 23 24] The results are best explained simply by the condensation and evaporation of pure CsI. If there is any interaction with structural alloys, it does not appear to alter significantly the vapor pressure of CsI over those surfaces.

In the presence of other compounds (e.g., boric acid, silver) though, there have been some observations of CsI decomposition.[21 22 25] In each case, a more volatile iodine compound was apparently formed. The implication is that if CsI interacts chemically with surfaces, the resulting iodine compound will be as volatile or more volatile than CsI.

Figure 7-5 shows the vapor pressure of CsI as a function of temperature. (The vapor pressure of hydrated CsI vapor species has been neglected; the curve in Figure 7-5 may therefore understate the total vapor pressure of CsI. It is felt to be adequate, though, to illustrate the point being made.) For comparison, the vapor pressure implied by vaporization of the whole-core inventory of iodine as CsI into a typical Westinghouse PWR RCS is also shown. This is calculated simply from the ideal gas law ($PV = nRT$), with n corresponding to the ORIGEN[26] iodine inventory for the Surry plant at full cycle (12.1 kg[3]) and V taken as 200 m^3.

This figure demonstrates the potential for substantial condensation of CsI in the RCS at temperatures below about 1100 or 1200 K. However, there are two factors that suggest that this retention may be only temporary. First, the heating of the RCS surfaces and the atmosphere by convective heat transfer and fission-product decay may elevate temperatures to levels at which much of the CsI will revaporize. Second, the vapor pressure of CsI even at relatively low temperatures (<1000 K) is sufficient that, given a gas flow containing little or no CsI over surfaces, the evaporation rate of CsI may be rapid compared

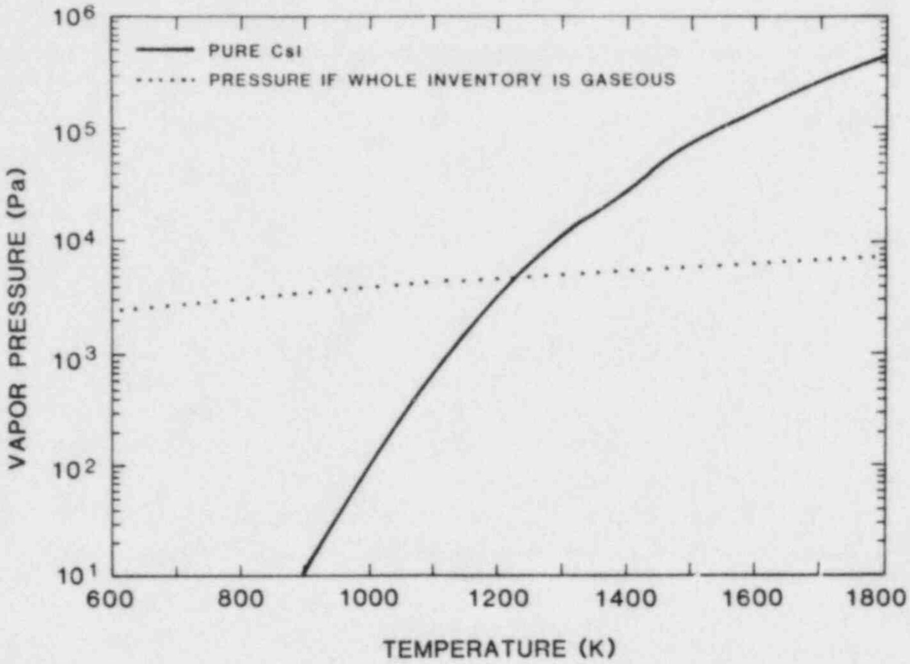


Figure 7-5 Cesium iodide vapor pressure.

to the half-lives of the important iodine isotopes (~8 days for ^{131}I). This may be demonstrated with a simple calculation.

Suppose that the iodine inventory (as CsI) is deposited, within the reactor vessel, over an area, A, of about 1000 m^2. If surface temperatures, T, reach 1000 K, then the CsI vapor pressure over them will reach about 100 Pa. The evaporative flux from the surface , J (gmol/s), may be expressed as

$$J = k_m A (C_w - C_\infty)$$

where k_m is a mass transfer coefficient (m/s) and C is the gas-phase CsI concentration (gmol/m^3) at the surface (w) or in the bulk gas(∞). If we assume a zero bulk concentration of CsI ($C_\infty = 0$) and replace C_w by P/RT (ideal gas), then

$$J = k_m A \frac{P}{RT}$$

$$\approx 12\ k_m \qquad \text{gmol/s} \quad \text{for this example}$$

Lower bound values of k_m, based on laminar flow heat/mass transfer correlations are around 10^{-4} m/s. More realistic values, based on the natural convection phenomena discussed in Subsection 7.2.1, would be of the order of 10^{-2} to 1 m/s. Thus, evaporative fluxes might be expected to lie in a range from roughly 10^{-3} up to 1 or even 10 gmol/s. Compared with a whole-core inventory of less than 100 gmols of cesium iodide, this suggests time scales for evaporation between tens of seconds and about a day.

Of course, there are many parameters which will determine the course of later revaporization and escape of CsI and other FPs from the RCS. The ultimate temperatures attained by the surfaces, and flow conditions over those surfaces, as well as into and out of the RCS, are clearly important. Other possibilities not even considered here include the formation of more or less volatile iodine compounds, or that CsI will deposit on aerosols rather than directly onto surfaces, and may thus a) remain airborne when removed from the vapor phase, or b) be resuspended by purely mechanical forces (fluid shear or surface vibration) at the time of RPV failure.

However, within the limitations of treating iodine in the RCS in terms of vapor phase CsI, it seems reasonable to conclude that

a) Short-term deposition of iodine on surfaces is feasible. The extent is uncertain, and depends primarily on RCS thermal-hydraulics and the partitioning of condensed CsI between aerosols and structures;

b) Longer-term re-vaporization of CsI from RCS surfaces appears likely in the absence of some means of cooling those surfaces (such as secondary side cooling in a steam generator). The extent and timing of later release of deposited CsI will depend on RCS thermal-hydraulic parameters up to and beyond vessel melt-through; and

c) Uncertainties in CsI deposition chemistry are probably relatively unimportant contributors to the uncertainty in iodine retention in the RCS. Uncertainties in thermal-hydraulic parameters and the quantities and behavior of aerosols in the RCS and the vapor-phase reactions of cesium and iodine compounds are more important contributors to that uncertainty.

7.2.2.2 Cesium chemistry

Cesium iodide is predicted to be the most stable vapor phase cesium compound under RCS accident conditions. Its chemistry has been discussed above. However, there is only sufficient iodine in the core to react with some 10% of the cesium. The remaining cesium is expected to exist as cesium hydroxide--or in some cases, as a mixed oxide. Its chemistry will be discussed here in terms of cesium hydroxide.

A number of experiments have indicated that cesium hydroxide, besides evaporating and condensing as discussed for CsI, undergoes a rather slower chemical interaction with RCS structural alloys. This interaction depresses its vapor pressure by about 3 orders of magnitude or more at temperatures around 1000 K. The exact nature of the solid species formed is not well-defined. Some experiments indicate that a solid solution of Cs^+ ions is formed in various surface oxides,[27] while others indicate the formation of well-characterized cesium compounds with trace impurities on the surface.[28] The exact mechanisms and kinetics of the processes responsible for vapor pressure depression are at present uncertain.

The implications of these uncertainties are discussed with reference to Figure 7-6, which like Figure 7-5, shows CsOH vapor pressure in comparison with that anticipated if the whole-core inventory of cesium in a typical PWR were vaporized into a 200 m^3 RCS. The curves allow for the existence of both monomeric and dimeric CsOH vapor, though not for their hydrates--this is an illustrative, rather than an exact, description.

Like CsI, the potential is clear for substantial surface deposition at temperatures below around 1100-1200 K via simple condensation of pure CsOH without chemical reaction. The possibility of revaporization or deposition or both at higher temperatures, though, is dramatically affected by the surface interactions. Even at temperatures up to steel melting (~1700 K), the possibility remains that cesium could remain bound to RCS structures, exerting a rather small vapor pressure over them. The importance of the uncertainty in surface chemistry depends very much on the system thermal-hydraulics. If cool surfaces (<~1000 K) remain readily available throughout the accident, the energetics and kinetics of the processes lowering vapor pressures are immaterial. Uncertainties in the surface chemistry will only be important if

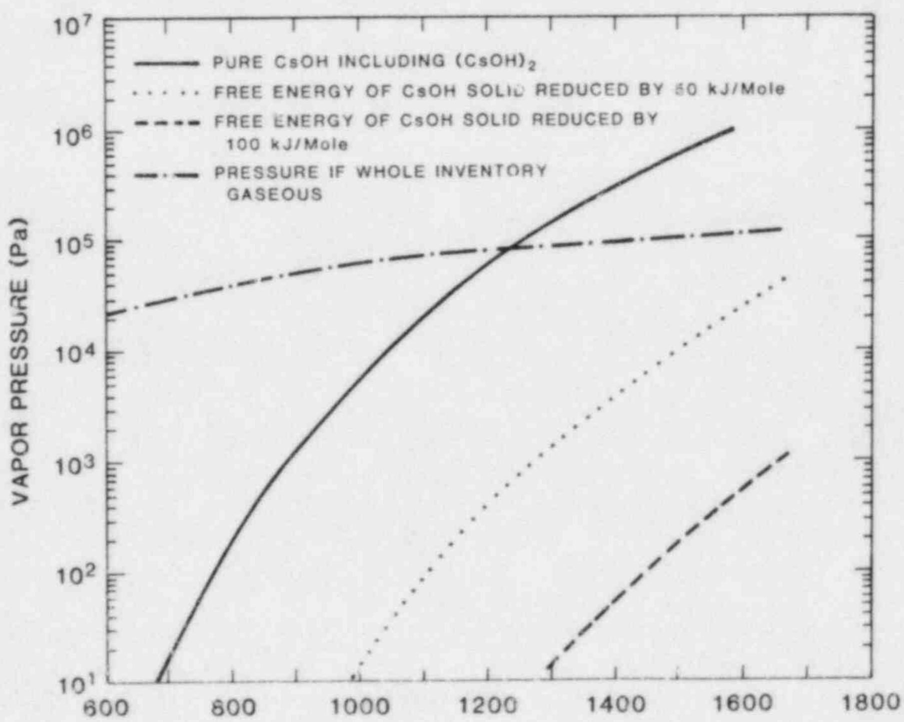


Figure 7-6 CsOH vapor pressure.

a) Temperatures of available deposition surfaces rise into the region well above 1000 K,

b) Reaction with aerosols to form aerosol cesium compounds does not substantially deplete cesium from the gas, and

c) In-vessel release fractions of cesium are high (see Subsection 7.1).

7.2.2.3 Tellurium chemistry

Elemental tellurium (or, more rarely, hydrogen telluride) is predicted to be the dominant form of this chemical species in RCS atmospheres. The available data all indicate that tellurium reacts rapidly with both iron and nickel alloys, with the initial formation of condensed phase iron and nickel tellurides.[24 29] The reduction in vapor pressure caused by these reactions is substantial; thermodynamic calculations for the Fe-Te and Ni-Te systems suggest that it is significantly greater than for CsOH. The Gibbs free energy depression of the Fe/Ni tellurides relative to elemental tellurium is probably of the order of 100 - 150 kJ/mol.

Two effects of the prolonged heat-up of surface tellurium deposits have been noted.[30] First, revaporization is possible--though it is apparently very slow up to temperatures on the order of 1500 K (consistent with vapor pressure suppression as described above). Second, iron tellurides are transformed into more stable chrome tellurides--a process which further lowers tellurium volatility.

Figure 7-7 shows tellurium vapor pressures analogous to those of Figure 7-6 for CsOH. The pressure due to both Te and Te_2 vapors is considered. Again, the set of vapor species considered is incomplete, but this should not affect the qualitative points discussed. The effects are qualitatively similar to those for CsOH: at lower temperatures, simple evaporation and condensation of pure tellurium could lead to substantial retention. At higher temperatures, several experiments have resulted in very rapid reactions removing Te vapor from the gas.[24 29] These results imply that, by reaction with RCS surfaces or with aerosols, Te would be prevented from existing as a vapor. However, more recent experiments in high-temperature steam suggest that Te reactions with RCS surfaces may be very substantially inhibited by the oxide layer on metal surfaces.[31] Thus some caution must be used before treating Te as unlikely to be transported in the vapor phase.

7.2.2.4 Volatile fission product chemistry--summary

Uncertainties in the chemistry of volatile FP reactions with RCS structural materials have been much reduced in recent years. However, several other major uncertainties would need to be reduced in order to utilize this improved knowledge to provide improved (i.e., with reduced uncertainty) estimates of the retention of these materials in

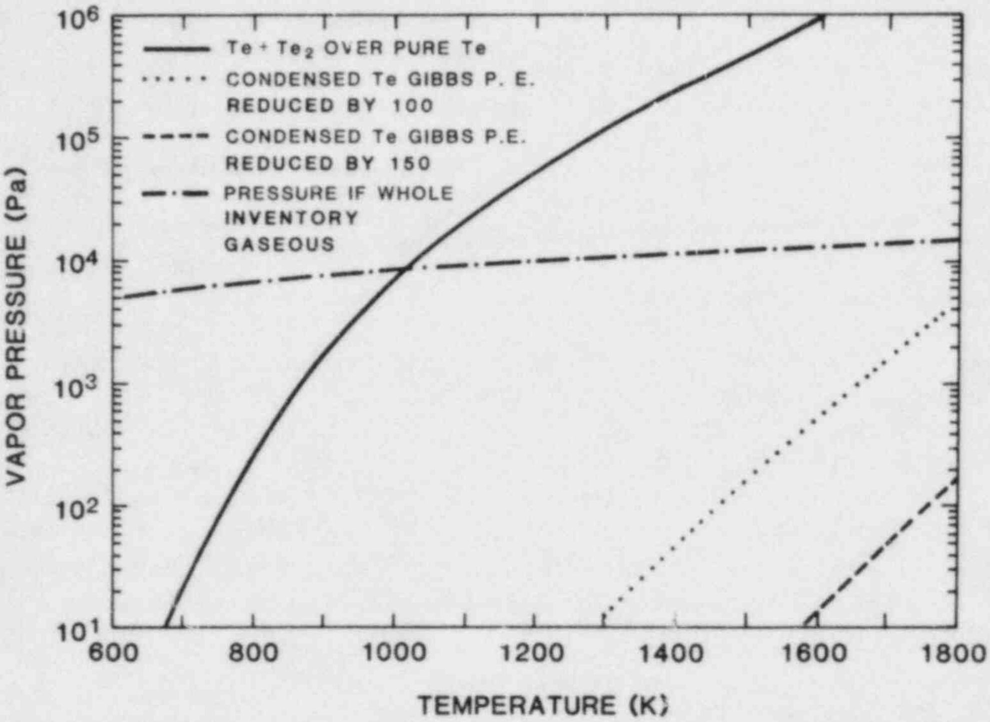


Figure 7-7 Tellurium vapor pressure.

the RCS in severe accidents. These other areas (FP and aerosol release; thermal-hydraulics and heat/mass transfer; aerosol behavior) are discussed in other parts of Subsections 7.1 and 7.2.

There remain areas of uncertainty in fission product chemistry that could, in certain circumstances, contribute significant uncertainty to RCS retention estimates. These include:

- The chemical form in which iodine exists in the RCS, given the potential for CsI to react with several RCS gas/surface combinations,

- The mechanism and kinetics of reactions that lower cesium vapor pressure below that of pure CsOH, and

- The mechanism, thermodynamics, and kinetics of the reactions of cesium and tellurium compounds with RCS aerosol materials. The conditions in which these uncertainties are important are discussed further in the summary, Subsection 7.7.

7.2.3 Aerosol Behavior in the RCS

Along with the volatile fission products discussed in Subsection 7.2.2, it is inevitable that some aerosols, containing both radioactive and nonradioactive components, will be formed by the overheating of fuel, RCS structures, and control rod materials. Uncertainties in the release of aerosols in-vessel were discussed in Subsection 7.1. The transport and retention of fission products is closely tied to that of aerosols. Three aspects of aerosol behavior uncertainty that propagate into RCS retention uncertainty are considered here:

1. Aerosol release, agglomeration, and settling,
2. Particulate deposition on and removal from surfaces, and
3. Particulate interactions with volatile FPs.

7.2.3.1 Aerosol release, agglomeration, and settling

The settling of in-vessel aerosols has been identified as the dominant mechanism for fission product retention in the RCS in recent USNRC studies, at least for some plants and accidents.[3] In the confined volumes of the RCS, high aerosol concentrations may rapidly build up; the total volume of the system is of the order of 200 m^3 for a typical Westinghouse PWR, so that even a few hundred grams of aerosol will lead to concentrations equivalent to a thick fog. The hundreds or thousands of kilograms of structural and control rod materials predicted to be released during some core melts (see Subsection 7.1) would lead to extremely dense aerosols and, presumably, to correspondingly high agglomeration and settling rates.

Several processes are responsible for agglomeration--each corresponds to a source of relative motion between particles (e.g., Brownian, gravitational, turbulent shear) or to the enhancement of attractive or repulsive forces between particles (e.g., electrostatic, diffusiophoretic). Theoretical formulations for the rates of agglomeration between particles of given sizes due to each of these processes have been developed, and have been reviewed by several authors (e.g., Fuchs[30], Hidy and Brock[32], Davies[33], Mercer[34], Friedlander[35]). Agglomeration rates are in general of the form

$$R_{ab} = K_{ab} \cdot n_a \cdot n_b$$

where R_{ab} is the rate of agglomeration of particles of class a with those of class b, K_{ab} is an agglomeration rate constant; and n_a and n_b are the number concentrations of particles of classes a and b respectively. We discuss here the contributors to uncertainty in these agglomeration rates, but note that the implications for time evolution of RCS aerosols do not follow from these in a simple manner. The system is a complex one involving many other phenomena--for example, slower particle growth due to agglomeration may lead to more rapid depletion of the aerosol by diffusion to structural surfaces.

Such an expression forms the basis for Smoluchowski's well-known integrodifferential equation for the time evolution of an aerosol size distribution.[36] This equation is analytically soluble only for certain special cases, and under various restrictive approximations. In practice, any assessment of aerosol behavior in reactor accidents must rely on numerical solution schemes which, because of the marked dependence of agglomeration and removal rates on particle size and other factors, tend to be computationally rather expensive. Uncertainty in the evolution of aerosol size distributions due to agglomeration in reactor accident conditions in general arises from

1. Uncertainty as to the correct form of the agglomeration rate constants for different processes,

2. Uncertainty as to how the rate constants for different processes should be combined,

3. Uncertainties in the parameters input to the expressions used to calculate agglomeration rates ("propagated parameters"), and

4. Errors introduced by approximations made in the numerical solution of the equations describing the evolution of an aerosol with time.

Uncertainty in the rate of depletion of aerosols by settling arises from all these sources, as well as from uncertainties in settling rates themselves. The uncertainties, and their likely relative magnitude under RCS severe accident conditions, are briefly discussed below.

Form of agglomeration rate constants: The accuracy with which agglomeration rates can be predicted depends on the sizes of the particles involved, the process being considered, and the prevailing system conditions. As with many aerosol processes, it is often possible to predict properties more accurately for very large or very small particles (in the continuum and free molecule regimes respectively) than for the intermediate "transition regime" where the particles of interest in severe accidents may often lie. For Brownian agglomeration, the formula proposed by Davies appears to give good agreement between theory and experiment.[37] However, the data collected by Mercer appears to show order of magnitude scatter in Brownian coagulation rates--though Mercer attributes much of the scatter to poor experimental techniques.[34] Uncertainties in accounting for electrical and magnetic effects add to the uncertainty in descriptions of thermal diffusion, especially for non-spherical particles. These effects have generally not been considered in reactor safety studies.

Recent studies have demonstrated the large potential importance of gravitational and turbulent agglomeration relative to Brownian agglomeration in severe accident conditions.[7] In either gravitational agglomeration, or in turbulent agglomeration due to inertia, the efficiency with which particles will collect other particles flowing past their virtual cross-section in the fluid is rather uncertain, because of uncertainties as to the effects of particle morphology, fluid flow around larger particles, and relative particle motion induced by, for example, thermal diffusion or Van der Waals forces. Gravitational collision rates differing by factors of as much as 3 have recently been adopted in different "best estimate" containment code studies,[3 7] and the uncertainty range appropriate for RCS conditions is considerably wider.

These uncertainties may be particularly important in conditions of high temperatures, pressures, and aerosol concentrations, where collision rates will be high and effects of fluid drag on particle motion will be most marked. None of the formulations of agglomeration rate constants have been tested or verified under the temperature, pressure, and gas composition conditions which would prevail in the RCS in a severe accident. The extent of the uncertainty added by extrapolation into untested regimes is unknown. The added uncertainty must, though, be considerable for mechanisms whose rates are described by empirical correlations (e.g., interpolation between continuum and free molecule regimes for Brownian agglomeration rates[37]). Further light may be shed in this area by aerosol experiments being conducted under more prototypic conditions at ORNL[38] and at Marviken.[39]

Rate constants for multiple agglomeration mechanisms: An effective K_{ab} in the presence of several processes leading to agglomeration has generally been obtained by summing the values due to the individual processes.[40] This procedure is known to be incorrect in the case of combined Brownian and gravitational agglomeration[41]--in this instance it can lead to an error of up to a factor of 1.4.[40] Other combinations of mechanisms have not been investigated in detail. Intuitively, though, it seems likely that particle agglomeration rates would not be simply additive; Brownian motion, forces due to turbulence or gravity, and other forces (electrical, magnetic, diffusion currents, etc.) might be expected to have intimately coupled effects on particle motion. The magnitude of the uncertainty introduced by the neglect of these couplings is unclear, but error factors of two or three would not appear unreasonable for situations in which both turbulent and gravitational agglomeration are important.

"Input parameters": these would appear to be perhaps the most substantial contributors to agglomeration rate uncertainties in the RCS. Among the uncertain parameters capable of having large effects on agglomeration rates are

1. Aerosol concentrations: These appear in rate expressions for all the processes as a "squared" term (all agglomeration processes are, almost by definition, second order). As discussed in Subsection 7.1, aerosol evolution rates from the core are subject to orders of magnitude uncertainty--this will propagate directly into agglomeration rate uncertainties via all of the mechanisms described above.

2. Aerosol morphology: Shape factors have been shown in a number of studies to have a very important influence on agglomeration rates.[7 42] Shape factors account for deviations from spherical aerosols; some RCS aerosols may in fact be molten, and therefore not subject to this uncertainty. On the other hand, very high concentrations of solid aerosols are also possible; these would quite possibly produce chains, rods and other agglomerates very far removed from a spherical

shape. An added complexity arises in describing and keeping track of the changing shape of agglomerates of particles--this becomes particularly important in very high concentration aerosols.

3. Aerosol composition: Particle density has a clear impact on particle motion relative to the fluid, and could vary in the RCS from around 2 g/cm^3 up to about 10 g/cm^3, depending on the balance between the lighter structural materials' oxides/hydroxides and the heavier control rod metals in the aerosol. Shape may also be affected--for large portions of the temperature regime of interest, control rod metals would be molten, while structural metal oxides would be solid. Again, additional complexity is introduced by the changes upon agglomeration of the distribution of compositions (as well as other, associated properties such as shape) over the particle size range.

4. Fluid turbulence: In the RCS this is subject to very substantial uncertainty (see Section 3). The effect on agglomeration rates was shown in the QUEST study (Appendix N of Reference 7) to be very marked; individual agglomeration rate constants K_{ab} could be modified by several orders of magnitude, leading to much more rapid agglomeration in situations with highly turbulent atmospheres.

Taken together, it seems clear that these "input parameter" uncertainties could lead to several orders of magnitude uncertainty in particle agglomeration rates. As mentioned above, though, this may not propagate into a similarly large uncertainty in the fate of RCS aerosols.

Numerical methods: A number of algorithms have been developed for the approximate solution of the equations governing the behavior of aerosols of homogeneous properties (morphology, composition, etc.) as a function of size. These have been developed into computer programs such as HAARM-3[43] and QUICK[44]. More recently, algorithms have been developed for the treatment of aerosols of non-uniform composition (e.g., Gelbard and Seinfeld[45]), and computer programs such as MAEROS[46] and MSPEC[47] have become available, implementing multicomponent aerosol approaches. These methods may readily be extended to cover the variation of other particle properties (e.g., shape) with size as agglomeration proceeds. Problems arise from the lack of any physically-based way of determining how these other properties vary with size, composition, and other factors, and from the large computational expense of keeping track of a set of variables as a function of size and other properties. Thus most treatments to date have used simplified assumptions--for example, that used in TRAP-MELT[3 48] and NAUA,[49] that particle composition is the same for particles of all sizes, or that used in MAEROS's applications to date, that morphology is independent of composition.

The degree of sophistication merited in the numerical treatment of aerosols whose properties vary with size is currently uncertain. Consensus seems to be being reached though, as to the desirability of using the so-called "discrete" methods rather than the "method of moments" approach to solving the basic equations. The latter methods, implemented for example in the HAARM-3 code, rely on assumptions as to the form of the particle size distribution--in most cases this is assumed to be log-normal. This assumption enables the equations describing the moments of the size distribution to be closed, and great computational speed to be attained. The approximation works well for many atmospheric aerosols, and indeed for aerosols in general when concentrations are low, and continuous source and steady depletion mechanisms are present. However, such conditions are unlikely in severe reactor accidents. High concentrations alone will lead to size distributions substantially "skewed" from a log-normal or any other formalized distribution. The presence of different aerosol sources at different stages of the accident adds to the non-uniformity of size distributions. The "discrete" approach affords much greater flexibility in the treatment of non-uniform size distributions. For these reasons, the TRAP-MELT code, originally written using the method of moments approach,[48] has more recently been modified to incorporate the QUICK algorithm.[3]

Settling rates: Agglomeration rates are not the only contributors to uncertainty in aerosol settling rates. Although for a particle which is aerodynamically well-defined the terminal velocity in air can be calculated very accurately, the area available for settling in the RCS is uncertain. This is particularly true in the RPV where most such settling is predicted to occur.[3] It seems unreasonable to assume that aerosols just volatilized from an extremely hot core will settle back onto it. On the other hand, degradation of the core might lead to a much enhanced surface area available for settling. Current calculations[3 7] assume the settling area to be unchanged from that of an intact core--substantial deviations could in reality occur in either direction.

7.2.3.2 Particulate deposition on and removal from surfaces

There are several processes other than gravitational settling which may lead to aerosol deposition onto surfaces. The mechanics of Brownian diffusion, inertial transport, and thermophoretic (and other radiometric) effects are discussed exhaustively elsewhere.[9 40] The types of uncertainty involved are similar to those discussed above--uncertainties as to the actual phenomena involved and the mathematical formulations used to describe them, and uncertainties as to the inputs to those formulations. An additional source of uncertainty, the adhesion of particles to surfaces, is discussed separately.

Uncertainties due to understanding of phenomena: As a broad generalization, more accurate correlation between theory and experiment is possible for deposition than for agglomeration experiments. This is because deposition experiments may be performed in much simpler systems--most experimenters have used monodisperse aerosols, thereby avoiding the problems of sampling across a size distribution and having to interpret the results in the face of a size distribution continuously evolving via simultaneous agglomeration and deposition.

Many similar problems remain, though--interpolating into the transition regime between small and large particles, describing particle motion in turbulent flows, extrapolating data into regimes far beyond those investigated experimentally, and/or combining rates due to a combination of processes. As was the case for agglomeration, uncertainty in one process need not lead to a large uncertainty in aerosol behavior, because the real system will involve many complex interactions between the growth and depletion processes.

As an example of a good fit between theory and experiment as regards actually describing the phenomena, Brock[50] discusses the good agreement between theory[51 52] and experiment[53 54] regarding the thermal forces on aerosols in the free-molecule regime (small particles compared to mean free path of gas). He extends the treatment into the transition regime with some success, reproducing a limited number of experimental results within 25%. However, as he points out, the results apply only to monatomic gases, and serve only as a qualitative guide for polyatomic cases (steam/hydrogen mixtures will be the rule rather than the exception in the RCS in severe accidents). The earlier correlation of Brock[55] used in several modern computer codes has been estimated to be accurate to within about a factor of two.[48]

The treatment of aerosol deposition from turbulent flows illustrates some of the difficulties involved in describing aerosol behavior, even in simple systems. Observed deposition rates in turbulent flows are generally greater than those predicted from fluid diffusivities alone, a phenomenon which can be attributed to the enhancement of transport close to a surface by the momentum which particles acquire in the bulk fluid. Earlier models of the role of particle inertia in transport through a turbulent fluid to a surface assumed that, at some specific distance from the surface, the inertia of a particle would be sufficient to carry it to that surface. These models then derived formulae for deposition rates based on turbulent diffusion of particles from the bulk to some "free flight" distance from the wall. Depending on the "free flight" distance used, different models of this type yield rather different results.[34 56 57] Sehmel, reviewing the available data in 1971, deduced an empirical correlation for deposition velocities in turbulent flow for both smooth and rough surfaces across a wide variety of conditions.[58] His assembly of data suggests that his correlations, used "blind", would provide estimates accurate only to within about an order of magnitude (plus or minus).

More recently, the careful experiments of Liu and Agarwal[59] and the theoretical advances made by, among others, Reeks and Skyrme[60] have considerably advanced the understanding of aerosol deposition from turbulent flow. The effects of factors such as surface roughness, particle morphology, and the structure of turbulence in the boundary layer close to the surface are all, though, significantly uncertain. Moreover, the problem of extrapolation into regimes untested experimentally remains, though some high temperature and pressure data in carbon dioxide is available from experiments performed in Advanced Gas-cooled Reactors (AGRs) in the UK.[61] The prediction of deposition rates within a factor of two or three is probably as much as or more than can be hoped for in the RCS at present. The behavior of aerosols in turbulent flow fields remains, though, an active field of research, and significant improvements in understanding of both basic phenomena and effects at high temperatures and pressures can be expected over the next few years.

<u>Uncertainties due to "initial and boundary conditions"</u>: The potential for such uncertainties to propagate into uncertainties in fluid and aerosol properties is indicated by the strong dependencies shown in Table 7-7. The table indicates which aerosol and fluid properties have a strong influence on which type of transport process. As well as the uncertainty introduced into the individual transport processes by uncertain input parameters, there is some uncertainty as to how the different processes couple together. For example, thermophoresis will build up aerosol concentrations in the boundary layer next to a cool surface. This will tend to enhance diffusional transport both to the surface and away from it into the bulk gas.

Table 7-7 Influence of aerosol and fluid parameters on particulate transport processes

	Process Influenced		
Parameter	Diffusional Transport	Inertial Transport	Radiometric (e.g., Thermo-phoretic) Transport
Particle size	X*	X	
Particle material and physical properties (e.g., density, thermal conductivity)	X	X	X
Shape factors	X	X	X
Fluid turbulence	X	X	
Heat flux to surfaces			X
Fluid physical properties (e.g., T, P, E, μ)	X	X	X

* X denotes a significant influence.

Obviously, these "input parameter" uncertainties will only be important when the deposition processes they influence are important. USNRC studies to date[3] with the TRAP-MELT code[3 48] suggest that, for the combinations of thermal-hydraulic and aerosol release assumptions used, many accidents fall into one of two categories, in either of which these mechanisms (and hence uncertainties) are unimportant:

1. Residence time scales in the RCS are so short that no significant deposition occurs, whatever assumptions are made concerning fluid and aerosol properties (e.g., the AB hot leg accident analyzed for the Surry plant[62]), or

2. For longer residence times, aerosol agglomeration, growth and settling provide the dominant mechanism for aerosol removal (e.g., the TMLB' accident analysed, also for Surry[62]).

Thus we cannot state that deposition rates are of general importance. There are, though, plants and accidents in which other phenomena are predicted to be important (e.g., a V sequence at Surry[62]). Moreover, the relative unimportance of these phenomena predicted in the calculations cited may have been an artifact of the particular thermal-hydraulic and aerosol release conditions (in each case selected from uncertain ranges) assumed in the TRAP-MELT input. For example, if, as seems plausible from the arguments advanced in Subsections 7.1 and 3.3, aerosol releases used in the TMLB' calculation were unrealistically high while the degree of turbulence allowed for were unrealistically low, actual agglomeration and settling rates would be lower than those predicted by TRAP-MELT, while rates of thermophoretic and turbulent deposition would be higher. Thus the other removal mechanisms would in reality be more important relative to agglomeration and settling than was indicated by the TRAP-MELT calculations.

Uncertainties due to particle adhesion: Particle adhesion to surfaces leads to two types of uncertainty. First, surfaces in general do not act as "perfect sinks" for particles, so that the rate of deposition on a surface is generally less than the rate of transport to it. A huge literature exists on the problems which this causes in aerosol sampling and measurement, and on the development of tacky coatings to improve adhesion.[63 64] Many of the formulae for deposition velocities due to various phenomena have been derived on the assumption of a unit sticking probability (this corresponds to zero aerosol concentration immediately next to the surface). Sticking probabilities are not necessarily important. As discussed above, deposition processes other than settling may in themselves be unimportant for some cases, in such circumstances the sticking probability is immaterial. In others, the coupling of different transport mechanisms may ensure deposition even if the sticking probability per collision is low. For example, if particles are transported to a surface by inertial or radiometric forces but do not adhere to it, their concentration close to the surface will rise and thus their rate of Brownian diffusion to it will increase.

Second, particles previously deposited on surfaces may be resuspended from them. Possible resuspension mechanisms include recoil from radioactive decay, fluid shear forces at the surface, and surface vibration. In general, resuspension will occur if resuspending forces exceed adhesive forces. The range of possible adhesive forces is very wide. The magnitude of the adhesive force depends on particle size and shape, microscopic surface geometry, and other factors such as the surface tension of any liquids involved. A collection of experimentally determined adhesive forces for a wide variety of surfaces and particles as a function of particle size is shown in Figure 7-8 (from Reference 7). The data span a wide range, from about 10^{-12} to 10^{-4} Newtons per particle.

Superimposed on the figure is a range of feasible shear stress forces, calculated for a high pressure system at 900 K in a 50-50 steam/hydrogen atmosphere on the following basis:

High shear stress:	gas velocity (bulk)	= 100 m/s
	shear stress coefficient	= 0.0112
Low shear stress:	gas velocity (bulk)	= 1 m/s
	shear stress coefficient	= 0.00187

The formula used and its validity are discussed in Appendix E of Reference 7. This range represents a conservatively narrow range of plausible shear stresses over RCS surfaces before and during RCS depressurization. Similar ranges of forces could be calculated due to surface vibrations, or to impulses from radioactive decay; here we illustrate only one possible resuspension mechanism.

Both the range of adhesive forces and the range of forces required for resuspension are very wide. The two ranges overlap considerably, and the resuspension uncertainty associated with high flows is large. High gas flow rates may be associated with events such as steam explosions in-vessel, or with vessel failure at pressure. The lower-range velocities of gas required for resuspension are rather modest in comparison with the flows that would be associated with large-scale failures of the RPV lower head. The feasibility of resuspension of realistic aerosols at low flows in a reactor environment is illustrated, for iron oxide aerosols into carbon dioxide, by one of the concluding experiments in the Windscale AGR.[61] In this experiment, particles of 2 and 5 micron diameter were injected into the reactor circuit at very low flow rates, and particle depletion from the RCS atmosphere was monitored. Upon increasing flow rates (to a few meters per seconds at most), much of the previously deposited material was resuspended into the gas. This is in no way an argument that the same would happen in an LWR circuit in a severe accident, but indicates the potential magnitude of an effect not considered in current analyses.

The implications of the possibility of extensive aerosol resuspension around the time of RPV failure are straightforward. Resuspension provides a means by which aerosol deposition could be bypassed as a mechanism for FP retention in the RCS. Because the extent of volatile FP association with aerosols is uncertain, the aerosol resuspension uncertainty is potentially important for all fission products except the noble gases.

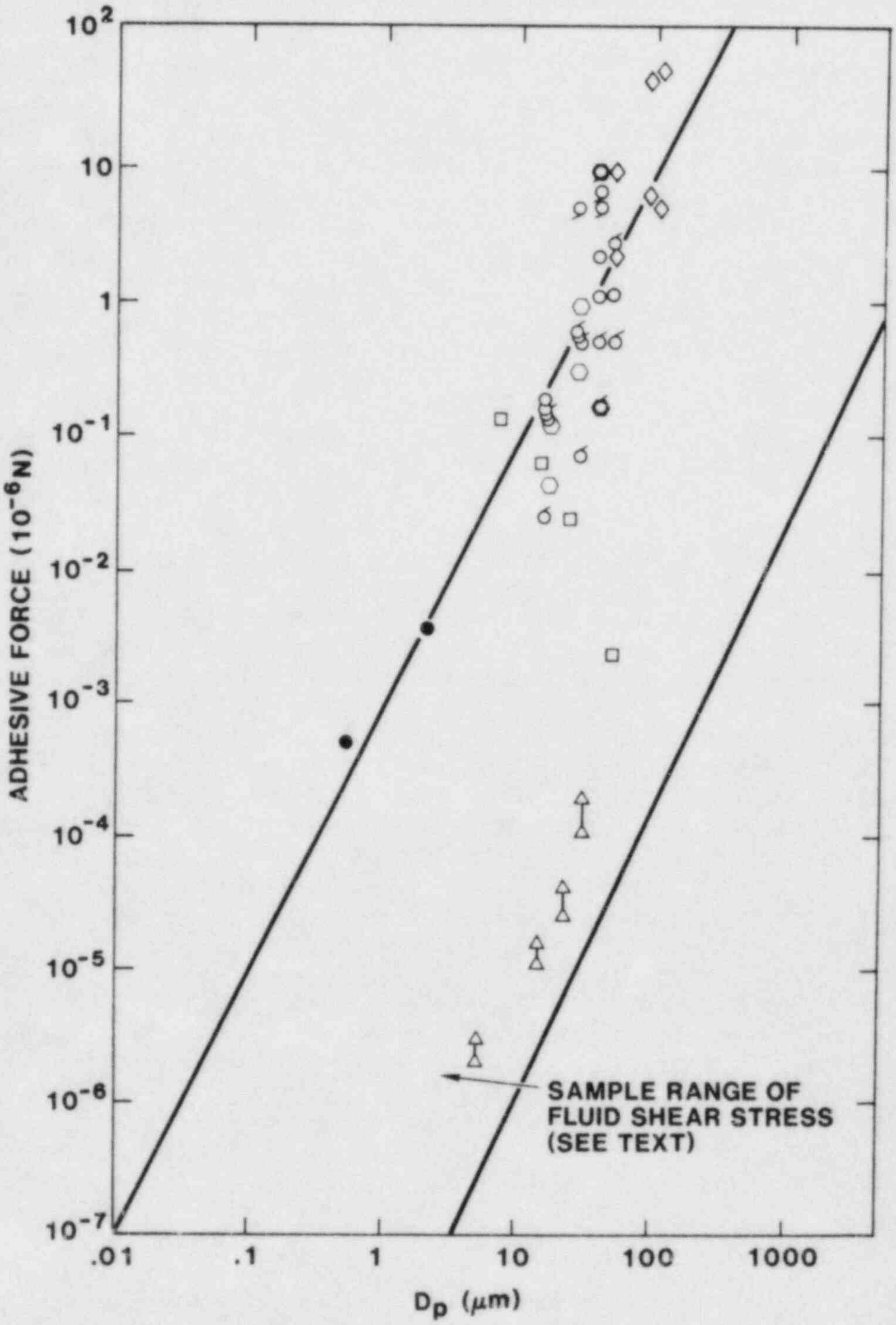


Figure 7-8 Particle adhesive forces (experimental[65-69]) and fluid shear stress range for a sample case (based on Figure E-4 of Reference 7).

7.2.3.3 Particulate interactions with volatile FPs

These provide a special case of the volatile fission product chemistry discussed in Subsection 7.2.2, and are therefore treated only very briefly here.

Just as was the case for reaction of an FP vapor with a structural surface, the rate of vapor reactions with aerosols depends on both the heterogeneous chemistry at a particle surface and on transport phenomena within the gas and the particle. Condensation or a reaction depleting vapor from the gas can only occur if the bulk gas is supersaturated with vapor relative to the vapor's equilibrium vapor pressure over the particle surface. Examples of reactions which might be postulated to lower the equilibrium vapor pressure of fission products over aerosols include the reactions of tellurium vapor with silver and tin[29] and of iodine compounds with silver.[25] The potential for favorable chemical interactions between FP and aerosol species is clear, but few of the possible pairs of systems have yet been investigated.

It is not always possible for supersaturation over a particle surface to occur in a gas which is cooled by contact with other surfaces. Vapor condensation on the vessel walls will occur at a rate paralleled by heat transfer. It is possible that the rate of cooling of gas and aerosol will be slower than the depletion of the FP vapor pressure relative to its equilibrium vapor pressure over particle surfaces--i.e., vapor condensation on the vessel walls may "desaturate" the vapor in the bulk more rapidly than the gas and aerosol are cooled. This partitioning of condensable vapor between walls and aerosol surfaces has been discussed in greater length elsewhere.[70] The important point to note is that, even in the presence of a favorable chemical interaction with aerosols, FP vapor reaction with them will depend on mass transfer and on the FP reactions with other available surfaces.

Uncertainties in RCS thermal-hydraulic conditions and in volatile FP chemistry at both aerosol and RCS surfaces lead to a gross uncertainty as to whether those volatile FPs will deposit preferentially on aerosol or on RCS structural surfaces. This uncertainty has several implications:

1. "Condensable" FPs may remain gas-borne by depositing on aerosols, rather than structural surfaces,

2. Deposition of volatile FPs attached to particulates leaves those FPs subject to resuspension by purely mechanical means, as well as by surface heating, and

3. The additional uncertainty as to the disposition of volatile decay heat sources feeds back into the thermal-hydraulic uncertainties discussed in Section 3.

7.2.4 Fission Product Transport in the RCS- Summary

Our understanding of fission-product chemistry, and in particular of interactions with RCS structural materials, has advanced considerably in the last few years. The mechanisms of these processes are becoming more clearly understood, though in some cases are still rather uncertain (for example, as regards the interaction of Cs compounds with stainless steels[27 28]). Descriptions of the kinetics of these processes, though, and of the interactions of volatile FP compounds with aerosol materials, are at an early stage of their development. Substantial uncertainties remain, too, in the boundary conditions (thermal hydraulics, FP and aerosol release magnitude and timing) governing FP transport and deposition, as well as in our ability to predict aerosol behavior under RCS conditions. Moreover, no calculational framework incorporating self-consistent representations of the several phenomena more recently recognized as important for FP retention in the RCS (natural circulation heat and mass transfer, and redistribution of decay heat loads are obvious examples) is currently available. Thus, large uncertainties remain as to the transport and retention of fission products within the RCS.

The implications of these uncertainties depend very much on the plant and accident sequence under consideration. If residence times in the RCS are very short (e.g., in the case of a large, hot-leg LOCA in a PWR), retention may be insignificant, whatever the assumptions about uncertain phenomena. This would appear to be the case for an AB (hot-leg) sequence at the Surry plant, judging by the very small retention fractions predicted for that accident using the TRAP-MELT code.[3] In such a case, uncertainty in release histories from the RCS to the containment atmosphere derive solely from the release process uncertainties discussed in Subsection 7.1.

For other sequences with longer residence times, such as transients and small breaks, the potential for substantial retention fractions demonstrated in NUREG-0772[2] has been substantiated by more recent TRAP-MELT calculations.[e.g., 3] However, the uncertainty associated with these predictions is large. One simple, physical limit on retention is imposed by the in-vessel release phenomena--material not released from the core during the in-vessel phase of an accident cannot be retained elsewhere in the RCS!

Beyond this, it is difficult to generalize. For example, the natural circulation effects discussed above and in Section 3 may enormously enhance mass transfer to RCS surfaces--on the other hand, they may so accelerate surface heating that revaporization of deposited FP species occurs quite rapidly. The range of possibilities implied by the uncertainties discussed above includes the following:

Aerosols: Substantial uncertainties exist in release rates and timing. Coupled with uncertainties in aerosol phenomena, these lead to large uncertainties in the extent, location, and physical nature (e.g., particle size, liquid/solid mix) of aerosol deposits in the RCS. FPs transported as aerosols might

- Remain gas-borne in the RCS for combinations of shorter residence times and small total aerosol concentrations.

- Be substantially depleted from the gas either during transport along release paths or during longer residence periods in parts of the RCS. The deposited aerosols might then

  - be retained long-term on RCS surfaces, if no large forces are operative to resuspend them. Even in the presence of resuspending forces, strong particle adhesion (e.g., by melting/mixing with other deposits and the surface) may assure long-term retention.

  - be resuspended by high gas flows and vibration associated with vessel failure, or

  - be retained long-term, but re-release more volatile constituents (on an uncertain time-scale) as surfaces and deposits heat up.

Volatile FPs: Even for well-understood deposition mechanisms, such as condensation, the extent of plate-out of these materials in the RCS is very uncertain because of the uncertainties in the thermal-hydraulics that dictate the gas-phase mass transfer conditions which in turn determine deposition rates. The kinetics of other reactions leading to retention and re-evolution of deposited volatile materials are substantially uncertain. So, although substantial short-term condensation of Cs, I, and Te compounds on cooler RCS surfaces is anticipated in many cases, the subsequent fate of those materials includes the following possibilities:

- Long-term retention, even in the event of large temperature rises, because of chemical interactions stabilizing the condensed phase FP species (may be particularly important for Cs and Te compounds);

- Short-term re-evolution by radioactive decay and heat transfer from the gas, or

  - heating of surfaces by radioactive decay and heat transfer from the gas, or

  - chemical reactions liberating volatile FP compounds. (Both of these mechanisms might be particularly important for iodine re-evolution, because iodine has both the highest decay heat load of the FPs concerned, and the greatest potential to form more volatile compounds.)

In addition to these possibilities, volatile FPs could deposit preferentially on aerosols, in which case their behavior would reflect the range of possibilities described above for aerosols.

The overall conclusion to be drawn is that the range of possible releases from the RCS is very large, and is very plant- and accident-dependent. At its widest, the range extends from essentially total release of volatile and other FPs released in-vessel by the time of RPV failure down to long-term retention of all FPs, excluding the noble gases. Short or medium-term retention within the RCS of volatile FPs (cesium and iodine in particular) is also possible. This could have important effects if these species were eventually discharged from the RCS close to or even after the time of containment failure.

7.3 Ex-Vessel Fission-Product and Aerosol Release

There are several potential sources of FP and aerosol loads in the containment atmosphere. The most obvious is that directly injected as gas or water discharges from the RCS. Other potentially important sources identified to date may arise from high-pressure melt streaming, ex-vessel steam explosions, and debris/concrete interactions. Two important uncertainties apply to all of the ex-vessel release processes. First, there is a propagated uncertainty as to what fraction of the inventory of the various radioactive species remains in the melt or debris discharged from the vessel. Second, the fraction of the core and lower structure that is discharged from the vessel is generally rather uncertain (see Section 4).

7.3.1 Release during High-Pressure Melt Streaming

The phenomenon of high-pressure melt ejection (HPME) and dispersal from the reactor vessel in sequences where vessel failure occurs at high pressure was first discussed in the ZPSS.[71] Subsequent experimental investigations have demonstrated the potential for the generation of copious quantities of aerosols during this process, by mechanisms postulated to involve not only vaporization of more volatile melt constituents but also by the mechanical entrainment of tiny liquid droplets into the gas.[72] HPME affords the potential to rapidly render gas-borne a substantial fraction of both the volatile and non-volatile components of the melt. The range of possibilities introduced by this phenomenon is discussed below.

Aerosol production by high-pressure melt ejection has been investigated in a number of tests at Sandia National Laboratories. All the tests used iron/alumina melts. In the earlier tests, aerosols were sampled in the vicinity of unconfined melt jets.[73] The results so obtained provided useful qualitative information as to the nature of the (copious) aerosols formed but, because of the lack of confinement of the experiment, were difficult to quantify. A crude estimate was made that 0.3 to 6% of the 10 kg melt charge was aerosolized from the jets. Two more recent tests confined the aerosols produced in a 45 m^3 chamber;[74] this and future tests should enable much more confident assessments of aerosolized quantities to be made. There still remains, of course, the problem of extrapolating from 10 kg to reactor scale.

Two types of mechanisms appear to be responsible for aerosol production; they lead to very different types of aerosol. The first, which leads to the formation of small particles enriched in the more volatile melt constituents, is vaporization followed by condensation. The uncertainties in vaporization phenomena were discussed in Subsection 7.1. The very effective mass transfer both from a dispersing, fast-moving jet into a gas and within the jet material suggest that highly volatile species (e.g., cesium, iodine, noble gases) remaining in the melt will be released very efficiently by this mechanism. Releases of medium and lower volatility FPs and other materials are more sensitive to jet and melt parameters and are therefore correspondingly more uncertain.

The second type of mechanism involves mechanical fragmentation of the melt. Based on the observed differences between jets pressurized with CO_2 and with N_2, gas solubility in the melt appears to play an important role in jet break-up and fragmentation.[73] Other potentially important mechanisms include pneumatic atomization at the vessel breach, hydraulic break-up at the edges of the jet, atomization upon impact with surfaces below the reactor vessel, and melt or debris entrainment from the reactor cavity. All of the mechanisms of this type would produce an aerosol of rather larger particles, of a composition approximating the melt average. This may provide a radiologically significant contribution of lanthanides, actinides, and other radiotoxic but nonvolatile materials which otherwise might never be released in respirable form.

High-pressure melt ejection uncertainties have several implications. First, HPME provides a pathway for rapid release at vessel breach of volatile radionuclides still trapped in the melt. Second, it provides a mechanism for direct aerosolization of a number of radionuclides of low volatility. Finally, it may add a large source of non-radioactive aerosols to containment close to the time of vessel breach. Because high aerosol loads promote rapid agglomeration and settling, the HPME aerosol may actually deplete FP activity from the containment atmosphere more rapidly than would otherwise be the case. These high aerosol loads may also have important implications for containment atmosphere thermal-hydraulics, as discussed in Section 5.

7.3.2 Ex-vessel Steam Explosions

It was postulated in the RSS that the occurrence of an ex-vessel steam explosion could both fragment the melt into very small particles and expose those particles to the oxidizing atmosphere of the containment. It was further postulated that this process could lead to the release from those melt particles of any constituents volatile in an air atmosphere. These constituents were taken to include not only the FPs previously volatile in the RCS, but also elements possessing volatile forms in an oxidizing atmosphere. In particular, ruthenium was ascribed to the latter category because it possesses volatile higher oxides (RuO_3, RuO_4). This makes an important addition to the source term from the plant, because FP ruthenium (especially ^{106}Ru) makes a significant addition to the radiological consequences of an accident.[6]

The inclusion of substantial fractions of ruthenium in the FP and aerosol loadings predicted for the containment atmosphere from this process has been rather controversial for some time. The experimental data presented in Appendix F to Appendix VII of the RSS clearly show the potential for virtually total release of ruthenium from UO_2 heated in air at temperatures above 1275 K (1000°C) for periods of between 10 and 20 minutes.

This does not necessarily imply, though, that similar releases can be expected in a steam explosion. A very important factor is the time scale for cooling of the particles produced by the explosion relative to that of fission product release. For a single particle traveling through relatively cool air, cooling almost certainly would be so rapid as to prevent the escape of significant quantities of any volatile species. If a large fraction of the melt is dispersed, though, cooling may be relatively slow. An added complication is introduced by the possibility that particles containing large fractions of unoxidized metal may burn in the containment atmosphere. This would presumably lead to the formation of finer aerosols of smoke particles.

If evaporational release from the particles produced by a steam explosion is small, their contribution to release will be limited to that via direct aerosolization. While some feel that this will lead to the formation primarily of very large particles which will rapidly settle out of the containment atmosphere, Bird's results (discussed in the in-vessel context in Subsection 7.2.3) suggest that substantial portions of melt may be converted into rather small particles (on the order of 1% of the melt aerosolized directly with a mean diameter of about a μm).[14]

7.3.3 Core/Concrete Interactions

The interaction of hot core debris with concrete will produce gases (steam, hydrogen, and CO, among others) which, by sparging upwards through the debris, may lead to the release of its more volatile constituents. This phenomenon was acknowledged in the RSS by the provision of a "vaporization release" component of the source term into the containment atmosphere. Again two mechanisms of rendering FPs and aerosol airborne can be envisioned in this process: first, a vaporization mechanism that, combined with nucleation and/or condensation on aerosols, may lead to the formation of rather small aerosol particles rich in the volatile melt or debris constituents; second, mechanical fragmentation processes (e.g., bubble bursting at a melt surface) that in general lead to larger particles of a composition corresponding more closely to the average of the melt or debris.[75]

Uncertainty arises in the contribution to airborne FPs and aerosols due to both of these mechanisms. The vaporization contribution is uncertain because of both external parameter uncertainties in the thermochemistry of the debris constituents and propagated parameter uncertainties in debris composition, quantity, and temperature, and sparge-gas evolution rate, composition, and temperature, as well as inherent process uncertainties in the understanding and modeling capability of the actual release process. The mechanical fragmentation contribution is subject to uncertainties arising from melt composition and both the melt and sparge-gas physical properties.

The debris-coolability questions discussed in Section 4 of this report are clearly of primary importance in determining the magnitude of the vaporization release. If the debris is quenched outside the vessel, there will be no attack on the concrete and no attendant release. On the other hand, if concrete attack does proceed, it seems almost certain that, within the bounds of uncertainty associated with the release processes, volatile radionuclides still associated with the core debris will rapidly be released from it by the sparging action of the gases produced. However, release from the debris does not assure discharge into the containment atmosphere. The sparging gases may pass through 1) cooler regions of debris, 2) a "crust", and/or 3) a water pool before reaching the atmosphere. Significant (but highly uncertain) FP attenuation could occur at any of these three stages.

7.3.4 Ex-vessel Release Processes--Summary

The range of possibilities for the contribution of ex-vessel release phenomena to containment atmosphere loadings of fission products and aerosols is very extensive. At one extreme, there would be essentially no ex-vessel release (if the melt ejection process leads to only a small release and to a quenched configuration of debris ex-vessel). At the other extreme, there could be 1) an essentially total release of the volatile nuclides in the discharged melt or debris which were not released in-vessel (cesium, iodine, and tellurium), plus 2) an appreciable release in volatile form of other nuclides (e.g., ruthenium), plus 3) a large release of aerosol material of something approximating the average composition of the melt or debris. A fuller discussion of ex-vessel release processes and the quantification of several of the attendant uncertainties is given in Reference 7.

The ex-vessel release processes and associated uncertainties have several important implications. Some of the more obvious are described below:

- Volatile fission products retained within melt and debris in-vessel will be rapidly released into the containment atmosphere after RPV breach unless the debris is rapidly and permanently quenced.

- Large additions of volatile, medium-volatile, and non-volatile radionuclides to the containment atmosphere are possible at, or very shortly after breach of the RPV. In the event of early containment failure, these could seriously exacerbate source terms from the plant.

- Large quantities of hot, oxidizable aerosol may be added to the containment atmosphere virtually simultaneously with vessel breach, adding to the threat of containment overpressure failure discussed in Sections 5 and 6.

- Large quantities of non-radioactive aerosol released soon after vessel breach may enhance the removal from the containment atmosphere of the radionuclides that were released in-vessel.

- Substantial quantities of medium- and lower-volatility radionuclides may be released over a long period of time by core/concrete interactions. If the melt is temporarily cooled or quenched after discharge from the RCS, these releases may be delayed for periods of several hours or more after RPV breach. Thus the possibility arises of releases around the time of, or beyond, a delayed containment failure. The environmental source terms arising from such releases could differ qualitatively, both in timing and in the quantities and proportions of radionuclides released, from those considered in PRAs to date.

7.4 Fission-Product Depletion from the Containment Atmosphere

Fission products and aerosols that become airborne in the reactor containment will be removed from the atmosphere by natural settling and deposition processes, as well as by the action of sprays, filters, and other ESFs. The magnitude of the radioactivity release from the containment atmosphere to the environment will depend on

1. The airborne inventory of FPs at the time of containment breach,

2. The time of such failure (which determines the above),

3. Additions to this inventory following containment breach, and

4. The nature of containment breach and the release pathways provided by it.

Uncertainties in the timing and mode of containment failure have been discussed in Section 6. This section discusses uncertainty in the rate at which aerosols and FPs are removed from the containment atmosphere. Subsection 7.5 discusses the implications of release pathways for the mitigation of FP release.

The phenomena governing transport and deposition in the containment are essentially the same as those discussed above for the RCS, with several important qualitative differences. They have been described at length elsewhere.[9 40 76] The discussion here is ordered along the same lines as that in Subsection 7.2; the following groupings of phenomena and the uncertainty they contribute to the time evolution of the containment atmosphere loading are considered:

- Boundary conditions
- Fission-product chemistry
- Aerosol behavior

Uncertainties in FP chemistry and aerosol behavior are perceived to be less important in containment than in the RCS because few FP vapor species are involved and aerosol models and codes are better validated for containment conditions. Most attention is therefore focused on the boundary conditions that affect the phenomena responsible for the depletion of FPs from the containment atmosphere, rather than on the phenomena themselves.

7.4.1 Thermal-Hydraulic Boundary Conditions

The containment fluid physics and associated uncertainties have been discussed in Section 5. Some of their important implications for FP and aerosol behavior are outlined below.

Atmosphere mixing is of great importance for aerosol behavior, because the more concentrated an aerosol the more rapidly will it agglomerate. Thus, calculations performed using multi-compartment representations of the containment geometry tend to produce results rather different from those using single compartment models.[7] The same feature was discussed in Section 5 in terms of its effects on hydrogen burning calculations. Uncertainties here propagate directly into uncertainty in describing the time evolution of the FP and aerosol characteristics in the containment atmosphere.

Mass transport to containment surfaces will be greatly affected by the motion of the containment atmosphere, which will in general be turbulent because of the large Rayleigh numbers associated with the large containment dimensions. Containment gas flows will be affected by such factors as condensation at surfaces, natural convection, and operation of ESFs. Heat transfer and steam condensation, as well as their general effects on flow, have a direct impact on thermophoretic and diffusiophoretic removal rates of aerosols from the containment atmosphere (see Subsection 7.4.3 - Aerosol Behavior).

The presence of saturated steam affects not only flow and heat and mass transfer, but also the nature of aerosols themselves. Aerosols and containment surfaces may compete for condensing steam, though one recent analysis suggests that the potential for condensation on aerosols is rather limited.[70] Steam condensation onto spray droplets, or containment surfaces in general, is also important, because it may greatly enhance (via diffusiophoresis) the rate at which FPs are removed from the containment atmosphere. Uncertainties in steam concentrations and condensation rates therefore have an important influence on FP and aerosol depletion from the containment atmosphere.

Liquid transport pathways: The presence of heat sinks at containment surfaces (and in some accidents, of active heat removal systems) should ensure that, for many accidents, the containment environment would be extremely wet. Containment walls and other surfaces would be expected to be covered in a film of water gradually washing down to pools on the containment floor. Water from pools may be recirculated into the spray system, transported across the containment boundary for heat removal purposes, or, in some plants, injected into the core for emergency cooling. Because airborne FPs will be deposited either onto wet surfaces or pools, it is to be expected that much of the airborne FP loading may become dissolved in, suspended in, or settled beneath water in one form or another. FP transport may thus become closely tied to water behavior (e.g., sprays render some fraction of FPs airborne; RHR systems transport them out of and back into containment; water flashing at containment depressurization may resuspend them). Note that in the TMI-2 accident, an aqueous pathway (the letdown system) was responsible for the only significant transport of non-noble gas radionuclides across the containment boundary.[77]

Hydrogen burns: The effects of aerosols on hydrogen burning were discussed in Subsection 5.3. Hydrogen burns may in turn influence fission product behavior, either by altering the physical or chemical form of airborne materials or by the re-entrainment of aerosols and FPs previously deposited on containment surfaces. Such effects could either mitigate or exacerbate source terms. If containment failure followed rapidly from the burn, either process would be expected to exacerbate the consequences. If, on the other hand, failure was delayed long after the burn, the resuspended or modified materials would in any case be depleted from the containment atmosphere by the mechanisms discussed in this section. Such depletion might be made either slower or faster by the burn. For example, transformation of large CsI particles into CsOH and HI or I_2 would probably make iodine removal slower. On the other hand, transformation of a dilute aerosol of µm-sized CsI particles into HI and CsOH species would probably accelerate iodine removal.

Engineered Safety Features: Two types of uncertainty can be associated with the effects of ESFs on FP and aerosol depletion. First, there is the uncertainty as to the state of the systems-- whether they are operating or not, or whether they are operating at reduced efficiency. Second, there is the uncertainty as to the efficiency with which, for a given operational state, they remove different classes of FPs and aerosols. The latter can affect the former. Ice condensers or fan coolers protected by filters could have their functioning impaired by heavy particulate deposits. Heavily clogged filters may be bypassed. Large loadings of particulate or "crud" in water could lead to damage to recirculating pumps or blockage of spray nozzles.[78]

Contributors to removal efficiency uncertainty include many, if not all, of the basic phenomena described here and in Subsection 7.2 when applied on a local basis to the ESFs. However, because of the limited range of operating conditions of the ESFs, and because in many cases they have been tested for FP and aerosol removal efficiency for many parts of that range, the uncertainties in removal efficiencies will, in general, be relatively small. Thus, uncertainties in ESF operability and degradation are probably more important in general than are removal efficiencies. It is possible, though, to find exceptions-- uncertainties in the efficiency of particulate removal in BWR suppression pools, for example, have recently been the subject of much debate.[79 80]

Releases into the containment: These processes have already been discussed in Subsection 7.3.1. Uncertainties in the timing of such releases relative to containment failure are extremely important, and lead to large uncertainties in the airborne FP and aerosol inventories at the time of containment failure. To give one example, if an aged aerosol containing a high percentage of the cesium and iodine inventory were released from the RCS, it would be depleted more rapidly in the presence of a secondary aerosol source ex-vessel (because of enhanced growth, agglomeration, and settling) than in its absence. The relative timing of releases can therefore be very important. Uncertainties arising from different treatments of aerosols of mixed composition are discussed in Subsection 7.4.3. Also very important are the characteristics of the released materials (chemical form of iodine, aerosol size distributions, and aerosol shape factors). These influence the subsequent FP and aerosol removal from the containment atmosphere, and are themselves subject to some uncertainty (see Subsection 7.2).

Conditions leading to resuspension: The phenomena that might lead to resuspension of deposited aerosol species--shear stress due to gas flow and surface vibration--would not be expected to lead to significant resuspension during normal, stable containment atmosphere conditions, because, in contrast to the situation in the RCS, forced gas flow rates over surfaces are likely to be small. There are several possible circumstances which could provide exceptions to this general rule. One important example is that of a hydrogen burn (which might lead both to very high transient gas flow rates over surfaces and to pronounced vibration of plant components). Others include a rapid depressurization of the containment building (such as might result from a catastrophic above-ground failure of the containment) and high forced flow rates through ESFs (such as fan coolers).

Of particular importance are uncertainties in the fate of radionuclides dissolved or suspended in containment water at the time of such events. A hydrogen burn provides a mechanism by which such species could be rendered at least partially airborne along with the water; the extent of such water aerosolization as a function of burn energy is subject to some uncertainty. Perhaps even more important is the uncertainty in the fate of water-borne species upon sudden depressurization of the containment. Both the water and the containment atmosphere would have very large quantities of energy stored in them (in the former as superheat and the latter as PV work) at the temperatures and pressures associated with containment failure. A rapid depressurization might lead to rapid disassembly of some of the containment internals or extensive flashing of water pools within it, or both. In either case, an important uncertainty would exist as to the fraction of the water and suspended or dissolved FPs which would be ejected airborne into the environment. This is discussed further in Subsection 7.5.1.

7.4.2 Volatile Fission-Product Chemistry

Several features stand out in distinguishing the containment from the RCS with respect to FP chemistry. First, the behavior of most FP species will be governed by that of aerosols, because their vapor pressures at containment temperatures will be too low for them to exist in vapor form. Iodine has often been considered an exception--several species (HI, I_2, organic iodides) would exist in gaseous form in containment. Also, there are other volatile compounds (e.g., those of ruthenium and tellurium) that have generally been neglected. Second, there are the containment surfaces, which will be much cooler, will contain less bare metal (many surfaces are painted), and for the most part will be wet for many accident sequences, so that deposited species will in many cases be either dissolved or suspended in water. Not only are the structural surfaces wet, there may also be spray droplets moving through the atmosphere and scavenging vapor and aerosol species from it. Active devices, such as filters and ice condensers, may be particularly efficient in retaining FP species passed through them. Resuspension may be enhanced during violent events, such as hydrogen burns or the rapid depressurization associated with a catastrophic failure of the containment. Uncertainties associated with volatile FP chemistry are discussed below.

Organic iodides: As discussed above, of the radiologically important FPs only iodine would be anticipated to have the potential to exist in vapor form in the containment environment. The possible alternative forms in which it might exist are CsI (aerosol), HI, I_2, or organic iodides. Of these, all but the organic iodides would be expected to deplete, albeit at different and uncertain rates, from the containment atmosphere until very small airborne levels have been reached. Thus, although it has never been postulated that organic iodides (particularly methyl iodide) could be formed in anything other than very small quantities, the small fraction of iodine present in this form could constitute a major component of the airborne activity in containment long after the release processes have occurred. There is uncertainty as to the mechanisms and rates of organic iodide formation; possible processes involved include reactions with small quantities of gaseous hydrocarbons (e.g., from oil leaks into the RCS or containment) and heterogeneous reactions at painted surfaces. It is difficult to place an upper bound on the fraction of iodine which could exist as organic iodides. Based on empirical observations made to date, as reviewed in Reference 2, it would appear to be a small fraction of one percent. However, as that review indicated, because the conditions responsible for organic iodide formation are not fully identified, the possibility of larger proportions cannot be ruled out.

Aqueous Iodine chemistry: With the possible exception of organic iodides, the vapor phase iodine species (HI, I_2) that might exist in the containment atmosphere will partition between atmosphere and solution or between atmosphere and surfaces with a very marked preference for the condensed phase. Although there is some uncertainty as to the partition coefficients, there can be little doubt that, if thermodynamic equilibrium were attained between gas, surfaces, and solution, the airborne fraction of these iodine species would be very small.

However, there are circumstances in which uncertainties in the kinetics of iodine removal into and release from water may lead to an important uncertainty in the fraction of iodine released from the containment. If containment failure occurs within a few hours of RPV failure, equilibrium may not be attained, and a substantial proportion of molecular iodine may remain gas-borne. If gas containing iodine in either vapor or particulate form is bubbled through water (e.g., in a BWR suppression pool), the decontamination factor (DF) achieved will be rather uncertain (for example, DFs presented recently for a saturated pool for a specific accident sequence ranged from less than five to many orders of magnitude).[81]

Another potentially important uncertainty with regard to iodine chemistry relates to the oxidation and subsequent evaporation of aqueous iodide ions. Although the equilibrium between gaseous or dissolved molecular iodine and more stable, dissolved ionic iodine species (especially I^-) would for most reactor accident conditions be expected strongly to favor the latter, the kinetics of redox reaction systems which we might formalize as

$$I^- + 0.5\ H_2O \xrightleftharpoons{0.25\ O_2} 0.5\ I_2(aq) + OH^- \quad (\rightleftharpoons I_2(g))$$

are significantly uncertain. Thus, if some mechanism exists by which molecular iodine is continuously depleted from the gas over a solution (e.g., by escape from a failed containment) the rate at which further iodine is released from solution is uncertain.

The kinetics and mechanism of aqueous iodide oxidation and evaporation have been the subject of experimental studies for many years. The state of knowledge in this area was reviewed in Reference 2, where it was concluded that "Reaction rates for I_2 hydrolysis are not known to an extent that non-equilbrium concentrations of iodine species can be accurately estimated." More recently, French investigators observed high iodine release rates in experiments where solutions were irradiated.[82] However, Burns et al have explained these results in terms of the experimental system used, and conclude that radiolysis effects will not significantly enhance iodide oxidation in LWR accident conditions.[83] Yet more recent experiments at Oak Ridge National Laboratories appear to confirm the potential for enhanced I_2 release because of radiolysis.[84]

Despite the continued experimental research efforts, uncertainties in aqueous iodine chemistry are still significant. The chief implication of the uncertainty is that, if fresh air is circulated over a solution of iodide ions (e.g., CsI solution), the fraction of the iodine which will oxidize and escape from solution, and the time scale over which it will do so, are substantially uncertain. The possibility of releases sufficient to influence significantly the off-site consequences of a severe accident cannot be ruled out.

7.4.3 Aerosol Behavior

Uncertainties in aerosol agglomeration and deposition phenomena were discussed in Section 7.2.3. Most of the available experimental data on aerosol deposition and agglomeration processes were obtained in air at relatively low temperatures and pressures (close to ordinary atmospheric conditions). Therefore, the uncertainty inherent in applying models of aerosol behavior (several at least partially validated computer codes are available[46 47 49 76]) in the containment is probably smaller than that in the RCS. Moreover, many containment surfaces will be wet, at least for large portions of some accident sequences. Because wet surfaces may reasonably be assumed to act as good aerosol "sinks", it would seem reasonable to assume high sticking probabilities and low resuspension rates at wet surfaces, except perhaps under very energetic containment atmosphere conditions. Still, coagulation and deposition rates due to individual processes are probably uncertain to within plus or minus factors of 2 or 3 in containment conditions, based on the reviews cited in Subsection 7.2.3.[32 33 35]

The characteristics of aerosol releases to the containment atmosphere may contribute significantly to uncertainties in containment aerosol behavior:

- Agglomeration rates are to a first approximation proportional to the square of aerosol number concentration, so that agglomeration and settling rates are rather sensitive to assumptions concerning release rates.

- Different aerosol components may be released at different times during the accident. An example considered in the QUEST study[7] is that of the release of the volatile FPs, Cs and I, around the time of RPV failure, followed by a later release of less radioactive aerosols from core/concrete interactions. The more aged, generally larger particles containing higher Cs and I levels are predicted to be "washed out" rapidly from the containment atmosphere--this effect is missed in calculations which do not track aerosol composition as a function of particle size (e.g., using the NAUA code[49]).

- Aerosol shapes have important effects on behavior, illustrated by recent sensitivity studies using the NAUA[7] and MAEROS codes.[42] In saturated or condensing atmospheres, aerosol particles will probably in any case assume near-spherical shapes,[85] though in drier atmospheres such as that predicted for much of a TMLB' accident at Surry with the CONTAIN code,[7] there is substantial uncertainty as to particle shape and its variation with other properties (size, composition, degree of agglomeration).

Aerosol characteristics, particularly shape, are very difficult to quantify (see Subsection 7.2.3.1). The ranges of possible shape factors indicated by comparisons of experimental data with code predictions made using shape factors to fit the results can be rather misleading. Such studies generally ignore the many other uncertainties in aerosol behavior. The shape factors are often in practice used as universal "fudge factors"--single parameters used to force a fit between experiment and calculation. The values which provide the best fit to data may thus account for rather more than just deviations from spherical shapes. Moreover, the shape factors used to fit data are generally assumed to be constant in time and independent of shape, size, composition, and other potentially important properties. For these reasons, the use of shape factor ranges derived from experimental data may lead to predicted ranges of aerosol behavior that do not accurately reflect the uncertainty due to aerosol shapes. More detailed studies of particle morphology should lead to more accurate identification of the true contributors to uncertainty in aerosol behavior, and thence to clearer resolution of the magnitude of aerosol behavior uncertainties in containment.

Other "boundary condition" uncertainties besides aerosol characteristics also contribute to aerosol behavior uncertainty in containment. Examples include the degree of atmosphere turbulence (the effect of which is illustrated in Reference 7) and the rates of heat and condensing steam transport to surfaces. The heat transfer rate has a direct impact on thermophoretic deposition, which has been estimated to be important for various parameter ranges.[86] Steam condensation has a very large impact, whether it take place onto aerosols (producing growth and enhanced gravitational settling[3]) or onto walls (leading to aerosol removal via diffusiophoresis[87]) and condensation uncertainties are correspondingly important. Uncertainties in containment atmosphere physics (temperature, pressure, composition, flow rates, etc.) will propagate directly into uncertainties in vapor and aerosol deposition rates in general.

7.4.4 FP Depletion from the Containment Atmosphere--Summary

As a general rule, we may state with confidence that, given sufficient time, airborne radioactivity in containment will be depleted to very small levels. The importance of uncertainties in the rate at which this depletion occurs thus depends very much on the timing of releases relative to containment failure. For very short residence times, depletion will in any case be small, while for very long residence times, it will be so substantial that uncertainty as to its extent will have little impact on the consequences of discharging that atmosphere from the plant, because those consequences will in any case be relatively small. In between, there is some "window" of relative timings, and balances between release rates into and gas escape rates from containment, for which the uncertainties discussed in Subsection 7.4 are important.

It would be a very long and complex task to determine for exactly what parameter ranges FP depletion rate uncertainties would be important. We have defined, albeit qualitatively, various regimes in which their importance (or lack of) is clear:

<u>Regimes where depletion rate uncertainties are unimportant</u>

- Major FP releases into the containment atmosphere all take place early in the accident (up to and around the time of RPV failure) and containment fails in a manner so as to depressurize quickly (time scale of less than an hour or so). FP removal from the atmosphere is minimal whatever the depletion rates.

- Major FP releases into the containment atmosphere take place early (as above), and containment failure either (a) does not occur, or (b) occurs after many hours, and in a manner such that reentrainment of deposited FPs is minimal (time scale for depressurization greater than a few seconds, and no long-term evolution of volatile species from solutions). In either case, releases from the plant would be so small that off-site consequences would be expected to be negligible whatever the rate of FP removal from the containment atmosphere prior to failure.

Regimes where depletion rate uncertainties are important

- Containment failure occurs within a few hours from the time of major FP releases into containment. In this case, airborne radioactivity levels may still be sufficient to have significant off-site radiological consequences, and uncertainties in those levels will therefore lead to uncertainty as to those consequences.

- Significant FP release takes place up until or beyond the time of containment failure. Even for long-delayed failures of containment, ex-vessel processes, plus the continued heat-up and discharge of debris and airborne FPs from the RCS, may lead to such releases. In such a situation, the degree of retention in containment will be determined by competition between gas discharge rates from the plant and FP depletion from the containment atmosphere. In many cases, the rates of these conflicting processes will be similar--uncertainty in either would then contribute significant uncertainty to the release from the plant.

- FP release takes place into a pre-failed containment. This may occur either through containment isolation failures or through accidents in which containment fails early in the sequence of events (various ex-plant initiated accidents: some BWR transient sequences). As in the previous scenario, competition between release of gases from containment and FP removal from the containment atmosphere will determine the extent of FP attenuation in containment. For much of the plausible parameter space, uncertainties in depletion rates of FPs from the containment atmosphere will lead to significant uncertainties in the releases from the plant.

In all of these examples, the range of uncertainty introduced by the combination of gas discharge rate from containment and FP depletion rate uncertainties is rather wide. In general, upper bounds on retention will be quite high. The minimum extent of retention requires more careful consideration--for some cases it will be small, but generalizations may be misleading. A sample problem investigated by Stone and Webster Corporation illustrates the effects of a wide range of gas discharge rates (containment hole sizes pre-existing in the Surry containment) for a given set of release and containment depletion assumptions.[88] As might be expected, the degree of retention varied considerably with hole size, showing a minimum where the balance between forcing gas out of containment and depletion most favored FP release from the plant. Although specific to a single combination of plant, accident, and other assumed details of the scenario, the study indicates the type of approach which can be adopted in order to scope out the range of possibilities introduced by these conflicting effects.

These examples illustrate the general difficulty in ranking in order of importance the factors contributing to source term uncertainties. For some highly plausible situations, the uncertainties discussed in this subsection are unimportant, while in others, they are among the key factors contributing to source term uncertainty.

7.5 Discharge of FPs and Aerosols from the Plant

Uncertainties in containment failure modes and associated probabilities have been discussed in Section 6. Uncertainty as to the quantities and characteristics of aerosols and FPs present in the containment atmosphere at the time of failure were described in Subsection 7.4. Subsection 7.5 discusses those features of the processes and pathways by which fluids, FPs, and aerosols actually exit the plant that contribute to uncertainty in the radioactive release to the environment. The discussion is organized into five sections:

- Above-ground releases from pressurized containment
- Above-ground releases from unpressurized containment
- Below-ground releases
- Containment bypass
- Ex-plant mitigation arising from release processes

7.5.1 Above-Ground Releases from Pressurized Containments

The capacity for mitigation or enhancement of the FP and aerosol release during the actual transfer from containment atmosphere to the environment depends very much upon the exact nature of the containment failure process. A so-called "catastrophic" failure (i.e., one involving very rapid depressurization through a large breach or breaches) would lead to little opportunity for attenuation along release pathways. Furthermore, it could lead to entrainment of substantial quantities of radionuclides into the atmosphere from containment surfaces and water pools. It would, though, be associated with a very large and rapid release, affording increased potential for mitigation of early off-site radiological effects through the effects of plume rise. Noncatastrophic failure, on the other hand, could involve very appreciable retention along release pathways, but would entail much lower rates of energy release from containment. Thus, the uncertainty as to the nature of the containment failure process is a very important contributor to uncertainty in both the magnitude and characteristics of the release from the plant.

The nature of containment overpressure failure is an area of current controversy. Some feel that a "catastrophic" failure of concrete containments is highly improbable or impossible. An experiment performed in Canada by the AECB, however, demonstrated the potential for such a failure in a 1/14 scale model of a pre-stressed concrete containment with a vinyl liner.[89] Extensive cracking of the concrete earlier in the pressurization experiment did not provide pressure relief because the liner retained its integrity. It is unclear whether a steel liner in a power plant containment could have the same effect. More recent experiments with models of steel containments have demonstrated, at reduced scale, the potential for violent disassembly of model components associated with rapid depressurization.[90] From the point of view of the radiological source term to the environment, there are four key factors associated with the depressurization:

1. Does it occur in a relatively "benign" manner, allowing gradual pressure relief through cracks or penetrations and preventing pressure build-up to the point where catastrophic failure could occur?

2. Do the pathways by which gas reaches the environment afford the potential for FP and aerosol attenuation, and if so, how much?

3. Will depressurization be sufficiently energetic to entrain substantial quantities of radionuclides deposited in water pools or on containment surfaces into the gas exiting the containment?

4. To what extent will the offsite effects of the release be mitigated by plume-rise, rain-out, or other phenomena associated with the discharge of gases from the containment?

The first of these questions has been addressed in Section 6. The last is considered in Subsection 7.5.5. Retention along release pathways and reentrainment of deposited FPs are discussed below.

7.5.1.1 Retention along release pathways

Two kinds of phenomena are considered here. The first is the actual passage through the containment wall. In the event of a noncatastrophic failure, this may involve gas flow through anything from a series of very narrow, lengthy cracks to one or more of the containment penetrations, which are generally cylindrical pipes of various diameters. Either pathway could afford appreciable FP attenuation along its length. The second is the provision of additional containment and opportunities for FP depletion by the auxiliary building(s) into which the escaping gas may be discharged.

Retention along cracks and pipes is simply a special case of the deposition and resuspension phenomena discussed in Subsections 7.2 and 7.4. Experimental data with particular reference to containment leak pathways has been reviewed by Morewitz.[91] He cites numerous examples of very efficient retention of aerosols and plugging of leakage channels ranging from cracks to large diameter pipes. The data quoted are fitted to a simple model proposed by Vaughn.[92] Despite the apparent consistency of model and data, there are large uncertainties in aerosol retention fractions anticipated in transit through the containment wall. These stem in part from process uncertainties--effects of resuspension and low particle sticking coefficients, particularly when large pressure differentials are involved, and the correlation of deposition with different aerosol transport phenomena. However, the major uncertainties are in propagated parameters--aerosol characteristics, steam fraction in the gas, and, most importantly, the flow diameter or other representative dimensions of the passage. The latter will make the difference between essentially total retention (for narrow cracks) and no retention at all (for rapid flow through short, large diameter pipes).

The second possibility for significant FP attenuation between containment atmosphere and the environment is that gases will be discharged through auxiliary buildings, leading to the possibility of further FP and aerosol depletion in those buildings by all the mechanisms described earlier in Subsections 7.2 and 7.4. This was demonstrated to have the potential for providing substantial source term reductions in the Sizewell source term study.[17] That same study noted that most of the penetrations of the Sizewell containment led into other buildings, and that the auxiliary building seemed the most likely destination of containment gas in the event of containment failure at penetrations or seals. It also raised the consequently very important issue of survivability of auxiliary buildings in the event that the containment depressurized into them. The state of auxiliary buildings following containment depressurization into them is, in general, subject to large uncertainty; for many plants it is probable that auxiliary buildings would not survive the rapid pressure transients resulting from such events.

7.5.1.2 Entrainment of deposited fission products

"Catastrophic" containment failure would not be expected to provide any potential for retention of FPs in passage through the containment boundary. In fact, the rapid dissipation of very large quantities of energy associated with catastrophic failure leads to a large additional uncertainty as to the extent of resuspension of FPs previously deposited in the containment. As discussed in Subsection 7.4.4, the depletion of FP loads in the containment atmosphere is expected to be very substantial several hours after the major releases of activity into that atmosphere. Typical recent calculations, by no means bounding the upper limit of depletion, suggest depletion by three or more orders of magnitude for several accidents in the Surry PWR.[62] FPs are removed from the containment atmosphere by ESFs, gravitational settling, and various plate-out mechanisms on the containment surfaces. Subsequently, they may either be washed down into water pools on the containment floor (perhaps the most likely eventuality for most sequences in view of the extensive steam condensation anticipated both on containment walls and in the atmosphere) or remain attached to other surfaces.

Estimates of the release of radioactivity to the environment associated with a delayed containment failure typically assume that all the deposited FPs will remain trapped within the building during and after failure. In fact, there are two mechanisms by which they may be resuspended. Neither mechanism would be expected to make a significant contribution to FP release except during a very energetic perturbation of the containment atmosphere. Such conditions could only be created by a very rapid containment depressurization, or perhaps by an energetic hydrogen burn.

The first mechanism involves aerosol resuspension via gas shear stresses or vibration, as discussed in the RCS context in Subsection 7.2.3. The second involves the entrainment of small water droplets from water pools during the "flashing" which would be expected to be associated with rapid containment depressurization. Such entrainment is commonly observed when pressurized liquids are allowed to "flash", and can be very extensive (and lead to the formation of very small droplets) even when the liquid superheat is sufficient to vaporize only a small fraction of the liquid.

A number of scoping calculations on the magnitude of these effects have been performed as part of the QUEST study.[7] The results indicate that containment depressurization rate (and thus hole size) is a critical parameter. For the shorter depressurization time scales investigated (a few seconds or less), large resuspension fractions for dry aerosols (up to 90%) and substantial water entrainment fractions (> 50%) can be predicted.

Because the water pools in containment may contain a sizeable fraction of the FP inventory released from fuel (see above), these results imply that resuspension may make a very significant contribution to source terms in cases of late containment failure. However, it is felt to be unlikely that it could lead to source terms comparable to those possible for early containment failure because a) the resuspended or entrained material is likely to consist of rather large particulates which could rapidly settle out in or very close to the plant; b) the very dense, wet aerosols formed could rapidly agglomerate and settle out; and c) the aerosols might be substantially depleted by impaction with the numerous containment structure and equipment surfaces lying in their path from pools in the containment floor or basement to the outside environment.

7.5.2 Above-Ground Failure of Nonpressurized Containments

In some accident sequences, release from an unpressurized containment is the only contributor to environmental release of activity. Examples include sequences in which containment failure occurs prior to core damage (such as certain transient sequences for a BWR, and many accidents initiated ex-plant). These latter, such as earthquakes or aircraft or other missile impact, are postulated to involve events which both fail containment (possibly with the opening of large breaches) and initiate core melting. For many plants, these sequences, even if they lead to severe consequences, may contribute little to risk (and thus to uncertainty therein) because of their low frequency. However, with the tendency to refine systems analyses and to reduce the estimates of the probability of in-plant initiated accidents (whether by improved analysis, for existing plants, or by design for new plants) these sequences are likely to become relatively more important in future PRAs.

It is not only in the above types of sequences that release from an unpressurized containment could be important. Release to the environment may continue beyond the duration of a pressurized failure. Postfailure release may arise either from the continuation of FP release phenomena within the containment or from the resuspension of FPs previously deposited within it. The former may be particularly important if containment failure occurs early in an accident (e.g., due to an in-vessel steam explosion or a pressure spike upon vessel failure) before most or all of the releasable FPs have been lost from the fuel. Possible release processes following containment failure include

- Continued "vaporization" release due to core/debris interaction with concrete in the reactor cavity. Note that the onset of significant core/concrete interactions may be delayed many hours after vessel failure.[7]

- Continued heat-up of debris within the RCS, leading to FP release and possibly to further melt or debris discharge from the vessel. Note that this process could also generate large quantities of gas (through interaction with water or concrete) which could enhance the transport of FPs out of the breached containment.

- Long-term heat-up of the RCS, containment, or ESF (e.g., filter) surfaces due to deposited FPs, leading to the potential for their resuspension and transport into the containment atmosphere.

Having established that there are circumstances in which release from an unpressurized containment can be important, it remains to discuss the uncertainty associated with it. The magnitude of the release will be determined by the relative time scales of the removal of FPs from the containment atmosphere by deposition processes and exchange of that atmosphere with the outside air (or expulsion, driven by gas generation within containment). Although retention in leaky buildings is typically ignored in accident consequence assessments because of the difficulty in justifying estimates of its extent, there is ample evidence that even the simplest shell of a building may afford substantial mitigation of the release. Examples of small-scale accidents illustrating this point have been collected and reviewed by Morewitz.[93]

The time scale of the removal processes within the containment after failure is subject to the same uncertainties discussed in Subsection 7.4. The rate of the expulsion of containment gas will depend on a number of highly uncertain factors. These factors include the breach size and location, the external wind speed and direction, the relative density of gases inside and outside the containment, and the generation rate of steam and noncondensable gases within it. The coupling of all these uncertainties makes it difficult to estimate with any confidence the fraction of the airborne FPs in containment after failure that would be retained within the plant. Note that an uncertainty in containment atmosphere FP depletion rates may, as discussed in Subsection 7.4.4, be unimportant in determining the airborne FP load in containment several hours after release but prior to failure, because that load will in any case be small. However, the same uncertainty may be very important when coupled with exchange with or depressurization to the outside atmosphere on a time scale of the order of an hour or less.

7.5.3 Below-Ground Failure of Containment

Failure of containment by basemat penetration is perceived as a less serious hazard than the above-ground failures, because the radiological consequences of activity release into the ground would be very small in comparison with those of release to the atmosphere. Hence the appreciable uncertainties as to the fraction of FPs trapped in the melt and debris that would be discharged through the penetration, and the fraction of air and water-borne FPs that would be retained en route through it, are not felt to be of major importance. The uncertainty as to whether basemat failure will actually permit pressure relief of the containment atmosphere has been discussed in Section 6. No further consideration will be given to below-ground failures in this section.

7.5.4 Containment Bypass

The accident sequences in this category (V sequences and SGTR are examples for PWRs; steam isolation valve failure is an example for BWRs) all involve the discharge of FPs and aerosols through some necessarily quite lengthy piping into the reactor auxiliary building(s). The uncertainty associated with the actual pathway to these buildings was discussed in Subsection 7.2. Retention along this pathway is subject to all the same uncertainties discussed there. Retention within the auxiliary building(s) is subject to the uncertainties discussed in Subsection 7.4. Particularly important in the case of direct discharge of the RCS contents into the building(s) is the uncertainty as to the impact of rapid pressurization on the building structure. The possibility of partial destruction of the building enormously widens the range of residence times which might be available for FP retention.

7.5.5 Ex-plant Processes

There is the potential for very substantial mitigation of the offsite radiological consequences of a release from LWR plants by processes that, though occurring outside the containment boundary, are associated with the actual release process from containment. These include local deposition, plume rise, and entrainment and mixing in building wakes. The first two in particular afford large potential mitigation; the uncertainty in the extent of that mitigation caused by in-plant processes is discussed below:

<u>Local deposition</u>: This term is used to refer to the phenomena associated with gaseous discharges from the plant that, it has been hypothesized, may act to enhance radionuclide deposition on the ground close to the plant. Two such phenomena are currently being investigated. The first is rainout, the process by which steam cooled by discharge from the containment is postulated to condense, causing localized rainfall through the cloud of discharged radionuclides, and washing out some fraction of those radionuclides close to the plant.[94] The extent of the depletion of radioactivity from the cloud will depend on, inter alia, the containment atmosphere conditions (fraction of steam, temperature, pressure), the external air temperature and relative humidity, and the rate of entrainment of air into the discharged gas. Three major uncertainties are associated with the effect:

- What fraction of the steam will condense?
- Will the droplets so formed agglomerate and rain out, or will they evaporate upon continuing air entrainment?
- With what efficiency will radionuclides be washed out by any rainfall through the cloud?

The second phenomenon being investigated is the enhancement of aerosol agglomeration by turbulence in the gas plume leaving the containment.[95] By increasing aerosol particle sizes in the cloud, this could have a significant enhancing effect on deposition close to the plant. Like the rain-out phenomenon, the extent of enhanced depletion is subject to large uncertainties because of, among other process and parameter uncertainties, gross uncertainty in the mode of discharge and characteristics of gas released from the containment.

<u>Plume rise</u>: Several studies have demonstrated the potential impact on atmospheric dispersion and radiological consequences of reactor accidents of the upward transport of radioactive gas plumes leaving a plant.[6 96] Such transport may be caused either by the momentum of the released material or by the buoyancy due to the sensible heat of the discharged gases. Among the key factors governing the height of plume rise, and thus its impact on accident consequences, are

- The momentum of the discharged gases,
- The angle at which they are discharged,
- Their sensible heat, and
- The rate of energy release from containment.

All of these factors depend critically on the mode of containment failure and gas discharge, uncertainties that will accordingly propagate into atmospheric dispersion uncertainties, as well as into the actual release process uncertainties.

7.5.6 Discharge of FPs and Aerosols from the Plant--Summary

Depending on the size and location of containment failures, the magnitude of the release of radioactivity from the plant may be very different from the quantity airborne in containment at the time of failure. The characteristics of the source term other than magnitude (timing, duration, properties of the released gas plume) that influence off-site consequences are also subject to very broad uncertainty ranges as a result of uncertainties in the discharge processes.

The most obvious process capable of modifying radioactivity releases during discharge is retention along the leakage pathway. If this involves numerous small cracks in concrete, or some long and tortuous route through auxiliary buildings, very substantial source term mitigation might be anticipated. Thus, even a containment failure while large quantities of radioactivity were airborne within it need not necessarily lead to a large and potentially hazardous release from the plant.

However, mitigation of releases is not the only possibility. If a containment fails through overpressure, and depressurizes on a time scale of a few seconds or faster, substantial reentrainment into the "containment atmosphere" (if the concept is still appropriate in such circumstances) of previously deposited aerosols and water-borne FPs might be anticipated. Such reentrainment could, at least to some extent, negate the effects of depletion from the containment atmosphere discussed in Subsection 7.4. Although the mode of containment failure required to produce such rapid depressurization might appear extreme, it is still very much within the range of possibilities discussed in Section 6.

7.6 Approaches to Source-Term Estimation in Accident Analyses

The previous portions of Section 7 may give the impression of a bewildering array of uncertainties associated with fission product and aerosol behavior, in the face of which attempts to determine integrated source-term uncertainties might seem enormously complicated. This need not in fact be the case. Subsection 7.6 places some of the phenomena and uncertainties discussed earlier in Section 7 in the context of approaches to source-term estimation that have been used in analyses to date. The summary, Subsection 7.7, then attempts to point out the major implications for severe accident source terms of the remainder of Section 7.

7.6.1 The RSS Approach

7.6.1.1 A description

The Reactor Safety Study acknowledged the extreme difficulty of estimating a deterministic source term for each possible end-point of the many-branched event trees describing the set of accidents it considered.[1] Accordingly, accidents were grouped into cases characterized by particular groups of features of clear importance in determining the magnitude and nature of FP release from the plant. For PWR accidents, 38 cases were chosen to represent the spectrum of releases, 23 were chosen for BWR accidents. A deterministic calculation of the release from the plant was made for each case according to the methodology described below. The outputs of these calculations were then further grouped into a smaller number of release categories (eight for PWRs, five for BWRs) which was judged to adequately represent all of the cases considered. The probabilities that each type of accident initiator would contribute to each release category were calculated, so as to produce a histogram showing the total frequency with which each release category would arise. In an attempt to account for the uncertainty in the assignment of sequences to release categories, 10% of the calculated total frequency of each category was added to that of each of the immediately adjacent categories, and 1% to each of the "next but one" categories (Section 4 of the main report). This histogram is combined with offsite consequence calculations for each release category in order to produce the various frequency/consequence curves which are the result of the PRA.

The methodology for calculating the release from the plant was as follows:

A. The release of FPs from the fuel was considered in four stages:

   1. A gap release at the time of the fuel clad failure,

   2. A meltdown release occurring at the "time of melting",

   3. A vaporization release due to core/concrete interactions in the reactor cavity, and

   4. An oxidation release associated with core dispersal in air following a steam explosion.

   A fixed fraction, independent of accident features, of the current core inventory of each of eight chemical classes of FP (chosen to represent all of the FPs in the fuel) was assumed to be released at each stage. The vaporization release was assumed to be of two hours duration; other releases were assumed to take place instantaneously. For any fission product i, the fraction F_{ij} released during stage j is given by

$$F_{ij} = R_{ij} \cdot X_{ij} \cdot C_j$$

where R_{ij} is the fraction of the core inventory of i remaining in the fuel at the inception of stage j, X_{ij} is the release fraction for i associated with stage j, and C_j is the fraction of the core which has reached stage j. F_{ij}, X_{ij} and C_j may be functions of time.

B. For PWRs, materials released from the fuel were assumed to be released instantaneously into the containment atmosphere (except in the case of containment bypass, where stages 1 and 2 take place directly to the environment). No retention in the RCS was assumed for PWRs. Escape fractions (for FPs other than noble gases) of 1/10 and 2/3 were ascribed to BWR coolant systems for success or failure of the ECC system, respectively.

C. The CORRAL code was used to estimate the airborne inventory of FPs in the containment as a function of time, and thus to predict escape fractions of different FP classes to the environment associated with different containment failure times and modes. It is at this stage of the RSS analysis that differences in accident sequences are allowed to affect release fractions from the plant. Of the 38 different accident cases analyzed for PWRs, those involving core melt thus involved identical release vs. time profiles from fuel for all FPs, combined with different sets of containment conditions and functions. The release categories 1 through 7 are accordingly distinguished only by the differences in containment retention predicted by the CORRAL code.

D. Each release category has associated with it a time from accident initiation at which it was assumed to occur, a duration of release, an elevation (above outdoor ground level), and an energy release rate. These parameters are used as input for calculations of offsite health effects.

7.6.1.2 Uncertainty considerations

Uncertainty had an important influence on the RSS, both on the choice of methodology for performing deterministic source-term calculations and on the philosophy used in grouping release categories and calculating their frequencies.

First of all, many phenomena then known to have a potentially large influence on FP release were ignored on the basis that justification of their effects was not possible. In many cases, this was associated with the philosophy that the report should be able to justify that it had not underestimated release from the plant. This has led to subsequent extensive criticisms that the RSS source terms are in reality of the nature of an upper bound on a wide range of potentially much smaller radioactive releases due to severe accidents.[97] Particular examples of phenomena not taken into account include retention in the RCS and numerous aerosol processes expected to enhance depletion of airborne activity in the containment.

The treatment of uncertainty adopted in the RSS was the subject of some of the most important criticisms of the study.[98] In particular, the rather arbitrary addition to the frequency of each release category of 10% of the frequency of those adjacent to it was heavily criticized. The uncertainty ranges quoted on the final frequency/consequence diagrams (a factor of 4 for consequence magnitude and of 5 for frequency of early and late deaths) were not justifiable by calculations presented in the report. The entire process of grouping accident sequences into release categories relied heavily on the choice of a calculational procedure for estimating releases from the plant that did not distinguish between sequences yielding potentially widely different releases. The justification for this grouping was based in part on the difficulty in distinguishing between different sequences in the light of uncertainties in FP behavior.

The RSS source-term methodology provided a simple way in which to calculate the radioactive release due to a given accident and to avoid the intractable problem of performing such a calculation for each and every one of the hundreds of possible chains of events considered. However, it does not provide a useful means of establishing the uncertainty in such source terms (except insofar as some of its release categories could be regarded as upper bounds for some sequences). Its limitations in this regard stem basically from its choice of a methodology for source-term estimation that does not permit the effects of many important phenomena to be included in the analysis.

7.6.2 Advances Since the RSS

7.6.2.1 The pressure for more realistic source terms

Since the publication of the RSS, there has been heavy pressure on reactor safety analyses both to account for the possibility of containment survival following severe accidents and to adopt a more phenomenologically complete approach to source-term estimation. The Zion PSS was the first PRA to account for the former possibility, though it adopted a source-term calculational procedure essentially unchanged from that of the RSS.[71] Thus, while it provided elaborate (relative to the RSS) procedures for uncertainty analysis relating to accident progression and event frequencies, it failed to account in an adequate manner for uncertainty arising from source-term estimates.

Several authors have highlighted the areas of pessimism inherent in the RSS source-term methodology, particularly since the time of the TMI-2 incident, most notably Levenson and Rahn.[97] They have given some very general estimates of ranges of mitigation factors (compared to the RSS analysis) to be anticipated for different processes in "generic" core-melt accidents (not specific to particular plants and/or accident sequences). Following criticisms that the RSS source terms are unsuited to provide regulatory guidance, there has been a substantial increase in research aimed at permitting more justifiable and realistic source terms. However, USNRC studies of the technical basis for predicting FP behavior and of suitable interim source terms to be adopted pending the results of ongoing research have concluded that there are no grounds for the justification of generalized reduced source terms at present.[2 3 99]

Several proposals have been put forward for revised source terms to be adopted in regulatory processes, and an extensive USNRC effort to re-evaluate source terms using the state-of-the-art tools available is under way.[3] Stone and Webster Engineering Corporation has proposed an interim source term comprising a 100% release of noble gases with 1% release of other volatile species.[100] The German project for nuclear safety (PNS) has recently published source term estimates much reduced from those of the RSS (and quite close to the Stone and Webster estimates).[101] However, these estimates are not applicable to U.S. plants because they apply to a reactor with a double containment and to accidents with a time between FP release from fuel and containment failure of several days.

7.6.2.2 The spectral source-term approach

This is the only new source-term methodology to date to have been applied in a PRA. It was first developed in follow-on studies to the Zion and Indian Point probabilistic safety studies, and has been used to generate "second estimate" source terms in the Sizewell 'B' (PWR) safety study in the U.K.[102] The aim of these second estimates is to provide an illustration of the impact on risk of source-term reductions that are anticipated to be justifiable within the near future, and for which a technical basis, albeit involving some degree of subjective judgement, has been developed.[17]

The spectral source-term approach attempts to specify probabilities that, for a given plant and accident sequence, the RSS source term will be subject to different reduction (or in some cases multiplication) factors. No attempt is made to use elaborate models to calculate "best estimate" retention factors. Instead, very simple hand calculations are used to justify mitigation factors due to events, parameters, or processes not considered in the RSS. For example, FP retention in the RCS is considered to provide a factor of 2 retention for all species (except noble gases) in a cold-leg, small-break LOCA. The CORRAL code is used to estimate containment retention of FPs (despite its appreciable "conservatism" at longer times) but increased containment retention factors are justified using improved estimates of deposition surface areas in the containment. Recognition is given to retention along leak paths. All of these processes are allowed for in a manner intended to be "conservative" (i.e., to overestimate the source term), retaining the philosophy that source terms should be justifiable in public as upper bounds on release from the plant. Subjective judgement (and concomitant uncertainty) enters the calculations when degrees of confidence are assigned to different phenomena options within a given plant/accident description. For example, for a small-break LOCA, different RCS retention factors are predicted for hot-leg, as opposed to cold-leg breaks. In the spectral source term approach, the consequences of each is calculated. Subjective probabilities are then assigned to the two options, so that each can be factored into the risk analysis within the overall heading of "small break LOCAs." A similar approach is used to assign degrees of confidence to different retention estimates, allowing, for example, the assignment of different degrees of confidence to different deposition rate estimates.

Once the full reduction factor/probability histogram has been constructed--this will typically involve tens or hundreds of different reduction factors for a given accident--it is "conservatively condensed" to a more tractable, reduced number of source terms specified, in the Sizewell case, by modification of the RSS source term by factors of 2, 1, 1/2, 1/4, 1/10, and 1/20. All probabilities of factors with intermediate values are added to that of the next higher factor. This diminishes the usefulness of the approach for giving an idea of the possible range of outcomes of an accident, because all smaller releases are "condensed" into the 1/20 category. These new categories and their associated probabilities were used in the Sizewell study in offsite consequence calculations to re-evaluate risks which were first estimated using source terms similar to the RSS source terms.[102] The effect is in general to depress very significantly the probability of large releases, and hence of severe accident consequences.

The spectral source-term approach might at first be regarded as a useful tool for determining bounds on releases in severe accidents. Indeed, its consideration in the accident event trees of many of the possible branches (previously ignored) which have a major effect on FP release is an essential first step in bounding the source term problem. However, in its published applications to date, the approach has in no way been extended to provide a full range of possible source terms from a reactor accident. The principal reason is the retention of the concept that the source term must be a justifiable upper bound in order to win acceptance. The approach does, though, represent an important advance for the uncertainty analyst, in that it has for the first time introduced into the LWR source-term evaluation many important phenomena and processes not previously considered.

In the manner in which it has been applied to date, the spectral source-term approach can actually inhibit the presentation of information on uncertainty. This is because, although it considers a wide range of different outcomes of source-term calculations, it adopts a probability distribution (for which no non-subjective justification is possible) among this wide range in order to condense the possible results into a single frequency/consequence line. During this process, all the information on the effect of the different calculational outputs is lost. In the Sizewell study this is unimportant because its intention is only to provide a risk comparison with the RSS source term (essentially the "spectral" source-term distribution and a delta-like function at a "mitigation factor" of 1 are being compared). For an uncertainty study, a much wider range of distributions (in particular, those giving the smallest and largest releases from the plant) would need to be considered in order to provide an estimate of the range of possible values of risk from a plant.

7.6.3 Mechanistic Source-Term Calculations

Since the time of the Reactor Safety Study, and more particularly since the publication of NUREG-0772, the USNRC has been funding the development of models and computer codes for the mechanistic evaluation of severe LWR accident source terms. The result has been the development of a code suite, illustrated in Figure 7-9, for the calculation of accident thermal-hydraulics and fission product release and transport, from the earliest in-vessel stages of the accident up to the final release from the plant. The code suite has been applied to a number of specific plants and accident sequences, and the results are being published as a report series designated BMI-2104.[3 62 81]

The source term calculational route provided by this code suite includes much more comprehensive representations of severe accident phenomena than have been applied previously in source term calculations. Although there are many areas in which the treatments of phenomena discussed in this document are imperfect or incomplete, it provides a more attractive calculational tool for use in uncertainty and sensitivity studies than has previously been available. Of particular interest is the QUEST program, which has been set up in an attempt to quantify the uncertainties inherent in the new calculational route.[7] The reader is referred to the QUEST documentation for a fuller and more quantitative discussion of several of the areas of uncertainty collated in Section 7.

The QUEST study provides a first attempt, specific to a particular plant and accident sequence, at quantifying the uncertainty present in source term estimates.[7] The study illustrates some important points. Although the SAUNA study has identified many (hundreds) of uncertainties which contribute to source term uncertainty, it may not be necessary to investigate each of these in order to make a useful estimate of source term uncertainty. QUEST investigated a relatively small subset of the identified uncertainties. Judgement was exercized to minimize the set of uncertainties examined while maximizing the source term range calculated by propagating those uncertainties through a limited number of calculations, combining them in different ways.

This approach enables a useful, crude estimate of the magnitude of the source term uncertainty to be made with a relatively small expenditure of effort. The apparent ranking of uncertainties in order of their contribution to the resulting source term uncertainty, though, is specific not only to the plant and accident analyzed, but also to the subset of uncertainties investigated and the combinations chosen to propagate into the result. Other uncertainties and combinations might produce similar ranges.

It is possible that such a study (involving a subset of the contributory uncertainties and a limited number of combinations thereof) would substantially underestimate the source term uncertainty. This would not be so if the estimated range of source terms could not be made wider without violating simple physical principles. The underestimation would in any case be unimportant if further widening of the range of possible source terms would not alter its implications for regulatory decisions. In either case, the estimate obtained from a study of limited scope can, as the QUEST study has shown, be both useful and adequate.

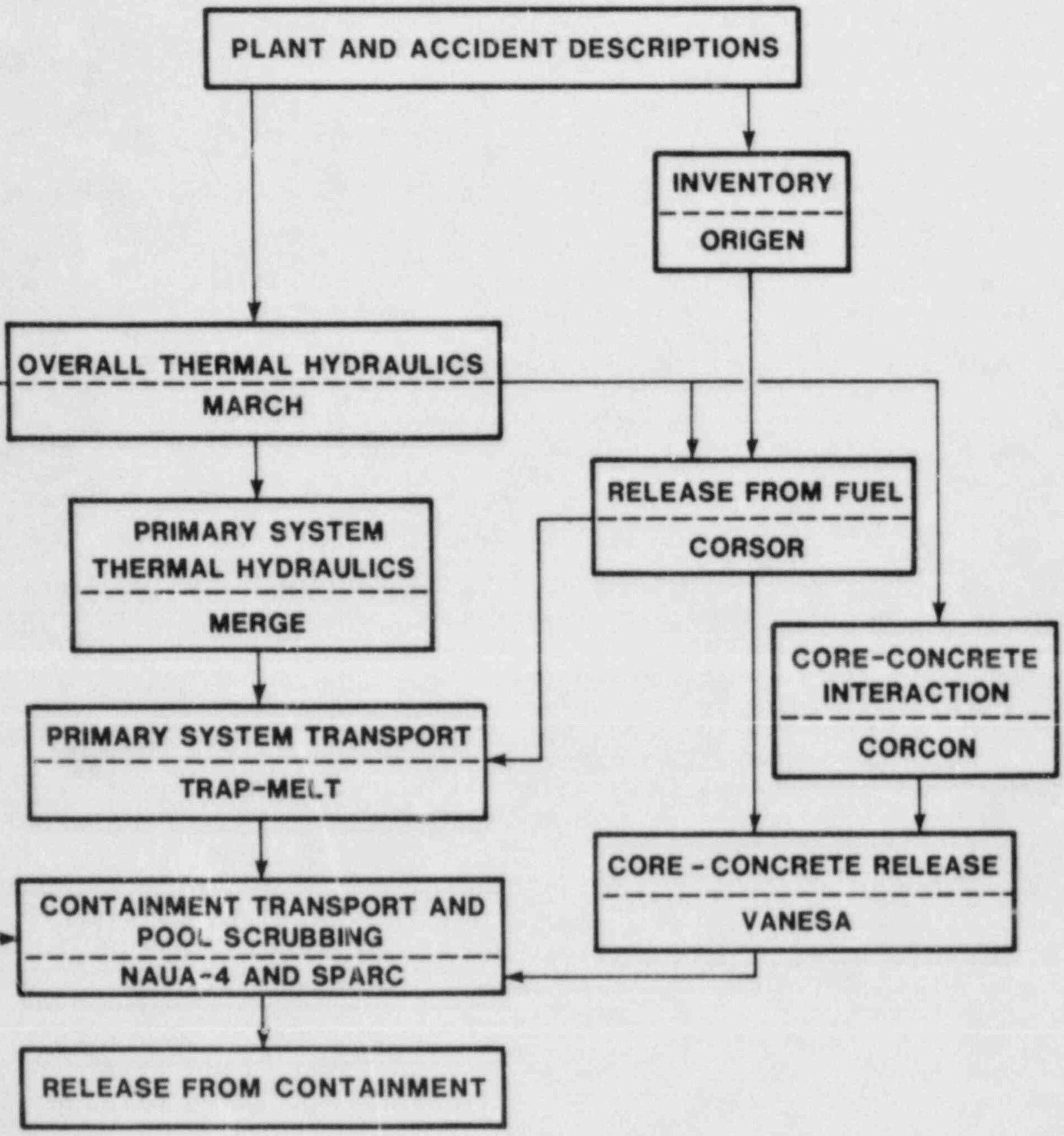


Figure 7-9 Relationships among codes used in USNRC source-term reassessment.

Despite their many limitations, the BMI-2104 and QUEST programs have provided a major step forward in the evaluation of source terms and the associated uncertainties. The utility and limitations of their approaches in these respects are discussed further in the summary, Subsection 7.7.

7.7 Summary

The summary is presented in two parts. In the first, a summary is presented of the major contributors to uncertainty in FP and aerosol behavior identified in Section 7. In the second, the integration of this information into an overall source term uncertainty description is discussed.

7.7.1 Summary of Major Contributors to FP and Aerosol Behavior Uncertainty

This section is further divided into subsections dealing with uncertainties affecting the in-vessel and ex-vessel phases of an accident. In practice, of course, the boundary between the two is not sharp. There may be considerable overlap between the timing of important events in-vessel and those ex-vessel. The distinction, though, affords greater clarity of presentation, and would in many cases correspond to a natural interface between different parts of a source term evaluation or uncertainty study.

7.7.1.1 In-vessel FP/aerosol behavior uncertainty contributors

Table 7-8 presents a "master summary" of the major classes of uncertainty contributing to in-vessel FP release and transport uncertainty. Tables 7-9 through 7-12 provide a breakdown of each of the four major classes of uncertainty listed in Table 7-8.

The tables contain a very large number of entries, each of which is capable of having a significant impact on source terms under certain circumstances (outlined as far as possible in the tables and earlier text). There are some basic, general comments to be made concerning the overall importance of these in-vessel uncertainties.

1. The "source terms" from the RCS of heat, hydrogen, steam, airborne and melt-borne FPs, and aerosols and their characteristics are subject to very substantial uncertainties. The range of possibilities introduced by in-vessel release uncertainties is outlined in Subsection 7.1.4. Their implications, when coupled with RCS transport uncertainties, for ranges of possible releases from the RCS are very dependent on the plant and accident being considered, and are outlined in Subsection 7.2.4.

2. If containment failure is delayed for many hours after releases into the containment atmosphere from the RCS, then in many, if not most circumstances, the fraction of materials released from the RCS and still airborne in containment will be small. In some cases, this may lead to the in-vessel release and retention uncertainties being "washed out" by the ex-vessel processes.

Table 7-8 In-vessel release and transport uncertainty contributors - "master" table

Uncertainty	Implications: For Release	Implications: For FP Disposition in RCS	Implications: For FP Release from RCS
Core and RCS thermal-hydraulics (table 7-9)	Large uncertainty in in-vessel release fractions & quantities	Important uncertainties in: a. fraction of FPs remaining in melt b. fraction deposited on aerosols vs. walls c. fraction of aerosol deposited in RCS	Uncertainty in quantity and form of FP and aerosol released from RCS (airborne and in melt)
FP and aerosol release rates (table 7-10)			
Volatile FP chemistry at RCS and aerosol surfaces (table 7-11)			
Resuspension due to a. surface heating b. high gas flows (table 7-12)	(note - resuspension issues unimportant if no RCS retention)		

Table 7-9 Core and RCS thermal-hydraulics uncertainties
(See Section 3 for sources of these uncertainties)

Uncertainty	Implications	Comments
Gas composition, temperature, flow, and pressure through core	FP, control rod, and structural aerosol release rates	Only temperature allowed for in current models. Large uncertainty (see 7.1.4)
	FP, aerosol retention in core	Special case of in-RCS retention (see 7.2)
Heat and mass transfer to RCS surfaces	Rates of FP and aerosol transport to RCS surfaces	These effects are coupled (see 7.2.1 and Section 3)
	Heating rate of RCS surfaces	
Flow paths through RCS	As above, plus determine accessibility of different plant components	
Melt-water interactions	Extra release source	See Table 3-7 and Section 7.1
	May provide conditions of high flow and vibration favoring aerosol resuspension	
	Vessel failure mechanism	
FP disposition in RCS	Heating of RCS surfaces and thus deposition and revaporization of FPs	See Sections 3 and 7.2.2
Core melt progression	Boundary conditions for release calculations	Geometry, flow regimes increasingly uncertain
Vessel failure	Melt ejection aerosol source	See 7.3.1
	Gas discharge may cause aerosol resuspension	See 7.2.3
	Possible heat-up, vaporization, and FP release after vessel failure	

Table 7-10 Fission product and aerosol release rate uncertainties

Uncertainty	Implications	Comments
Core thermal-hydraulics		See Table 7-9
Rate-controlling factors for evaporation	Determine which uncertainties are important	Probably differ for different plants/accidents
Transport rates through condensed phases	Evaporation rates (if condensed phase transport is limiting)	Depend on fuel conditions. Experimental data show wide scatter
Vapor pressures of FP and other species	Evaporation rates (if condensed phase transport is not limiting)	Involves both completeness uncertainty and lack of thermochemical data
Gas-phase mass transfer	Evaporation rates (if this is limiting)	Most important for high-pressure, low-flow accidents. Dramatic reductions in aerosol release are possible
Energetic events (Zircaloy burning, steam explosions)	Direct aerosolization plus possible enhanced evaporation	Possible mechanism for non-volatile species to become gas-borne
Leaching rates	Extra FPs into RCS water; may become airborne later	Probably not of general importance
		GENERAL COMMENT - ranges difficult to generalize - see 7.1.4 for discussions

Table 7-11 Volatile FP chemistry at RCS and aerosol surfaces uncertainties

Uncertainty	Implications	Comments
Nature of aerosol surfaces	Define possible chemical reaction partners	Some I, CS, Te compounds known to be reactive
FP vapor reactions with aerosol materials	Determine scope for volatile FP deposition on aerosols rather than walls	Most experiments performed on RCS structural materials
Gas, aerosol, and surface thermal-hydraulics	As above	Complicated by FP decay heating, and by heat and mass transfer uncertainties
	Determine potential for short-term condensation/long-term revaporization	
Kinetics and mechanism of Cs high-temperature reactions with steels	Determine potential for "permanent" binding	
Volatile FP vapor forms in RCS	Determine potential for surface deposition	Uncertainties due to e.g. CsI/boric acid reaction; stability of Cs_2MoO_4/Cs_2UO_4/CsOH; radiation effects
		GENERAL COMMENT - ranges difficult to generalize - see 7.2.4 for discussion

Table 7-12 Resuspension due to surface heating and high gas flows uncertainties

Uncertainty	Implications	Comments
Heating rates due to FP decay and convective heat transfer	Determine FP vapor pressures (VPs) over RCS surfaces	
Volatile FP chemistry	As above	Uncertainties for I because of chemical form; for Cs, Te because of uncertain kinetics, mechanisms, and thermodynamics of surface chemical reactions
Extent of volatile FP association with aerosols	Some condensed FP species may still be airborne. VPs may be altered.	Considerable uncertainty even when FP chemistry well-known
	Determines susceptibility to aerosol reentrainment	
Steam explosions	See Table 7-9 - create high flow conditions favoring reentrainment	
RCS flow patterns at and after vessel failure	Failure at pressure may give very high flows, favoring reentrainment	Important for any volatile species attached to aerosols
	Natural circulation after vessel failure may transport revaporized FPs out of RCS	Raises possibility of delayed release relative to containment failure time

3. If containment failure occurs close to or before RCS melt-through, then the importance of in-vessel release and retention must still be gauged in comparison with the potentially important ex-vessel release processes. For example, a large uncertainty in cesium or iodine release in-vessel in an accident affording little potential for retention in the RCS would be immaterial if these species would in any case be released rapidly at or after vessel failure.

7.7.1.2 Ex-vessel FP/aerosol behavior uncertainty contributors

A "master table" of ex-vessel contributors to source term uncertainties is presented as Table 7-13. Tables 7-14 through 7-18 contain a breakdown of each of the major classes of uncertainty listed in Table 7-13.

Table 7-13 Ex-vessel release and transport uncertainty contributors - "master table"

Uncertainty	Implications	
Thermal-hydraulic boundary conditions in containment (Sections 4, 5)		
Energetic events: steam explosions, H_2 burns, HPME, containment depressurization	Threats to containment integrity	
		SOURCE TERMS
Sources of steam, H_2 heat, FPs, melt, etc. from RCS	History of airborne FP and aerosol in containment	
Ex-vessel release phenomena		
FP and aerosol removal from the containment atmosphere		

Again, a large number of uncertainties are present, and each is potentially important for particular conditions or sets of assumptions concerning accident phenomenology. Many of the uncertainties relate to features of accident progression or delineation of system conditions, rather than to the actual behavior of FPs and aerosols. The number of uncertainties involved, and the dependence of the importance

Table 7-14 Source terms from the RCS

Uncertainty	Implications	Comments
Sources of FP and aerosol from RCS		See Table 7-8
Sources of heat, melt, steam, and H_2 from RCS		See Section 3

Table 7-15 Thermal-hydraulic boundary conditions

Uncertainty	Implications	Comments
Heat sources and sinks in containment	Ex-vessel releases: Atmosphere T, P history, Aerosol agglomeration/deposition	See Sections 4 and 5
Steam condensation	a. Aerosol growth b. Diffusiophoretic removal	Major contributor to aerosol removal in several studies to date
Atmosphere turbulence and mixing	a. Rates of aerosol deposition and agglomeration processes b. Rates of aerosol dilution	
ESF operability	a. Effect on containment atmosphere physics (Section 5) b. Direct effect on aerosol and FP removal	Dramatic effects on removal rates possible, but depletion (albeit slower) will still occur without ESFs.
Containment breach size, location, and timing	a. Remaining FPs airborne b. Further retention along leak paths c. Reentrainment of deposited aerosols and water-borne FPs d. Releases after containment failure	See also Section 6 and Subsection 7.5

Table 7-16 Ex-vessel release phenomena

Uncertainty (Contributors listed under Main Headings)	Implications (Refer to Main Heading, not Individual Contributors)	Comments
<u>High-pressure melt ejection</u> (HPME) - Mass, T, and FP inventory of melt - Mode of ejection - Vaporization from jet (Table 7-8) - Mechanical entrainment phenomena	a. Rapid release of volatile FPs not released in-vessel (includes volatiles <u>in air</u>)	Important only if RCS pressurized
	b. Non-volatile FPs may become airborne in significant quantities	Several non-volatile FPs are highly radiotoxic. Small release fraction could significantly exacerbate radiological consequences
	c. Sudden, large heat source to containment atmosphere	
	d. Debris configuration and initial conditions	
<u>Steam explosions</u> - Mass, T, and FP inventory of melt - Distribution of size and energy among melt fragments - Vaporization from particles (Table 7-8) - Rate of cooling of debris fragments	a. Potential for rapid release of volatiles (includes volatiles <u>in air</u>)	
	b. Possibility of direct melt aerosolization	Probably less important than for HPME
	c. Debris configuration and initial conditions	
<u>Core/concrete interactions</u> - Mass, T, FP, steel, Zr inventories in debris - Debris coolability (see Section 4) - Concrete degradation, (rate, gas T, composition) - Vaporization phenomena (Table 7-10) - Mechanical entrainment e.g. bubble bursting	a. Rapid release of volatile FPs	
	b. Additional FPs from "non-volatile" groups	
	c. Substantial quantities of non-radioactive aerosols from steel/concrete	
	d. Possibility of delay of release until long after vessel failure	Possible long-term release mechanism around and after some late containment failures

Table 7-17 FP and aerosol removal from containment atmosphere

Uncertainty	Implications	Comments
Iodine chemical form	a. Organic iodide fraction b. Removal rate	Small, probably not a major risk contributor; effect on on rate - but little effect on ultimate levels
Aerosol deposition phenomena	Rate of aerosol depletion	Factors of <3 uncertainty due to phenomena understanding; initial and boundary conditions more important.
Aerosol agglomeration phenomena	Rate of particle growth and settling	Again, initial and boundary condition uncertainties probably most important.
Steam condensation on aerosols	Rate of particle growth and settling	More substantial uncertainty (or modeling misconceptions)
Relative timing of different releases	Potential for large concentrations of non-radioactive aerosols to "wash out" FPs	Most of the aerosol mass, which controls important agglomeration and settling rates, is due to non-radioactive materials.

Table 7-18 Energetic events in containment

Uncertainty	Implications	Comments
High pressure melt ejection		See Table 7-16
Steam explosions		See Table 7-16
Hydrogen burns (see Section 5)	a. Threat to containment	
	b. Resuspension of deposited or water-borne FPs	
	c. Transformations of chemical form	May, for example, turn CsI to HI - may increase or decrease removal rates - little effect on ultimate extent of removal
Containment depressurization	a. Reentrainment of deposited and water-borne FPs	Probably only significant for very large containment breach
	b. Extent of retention along release pathway	Capable of mitigating worst WASH-1400 source terms

of most of them on the choices made at other uncertain points, illustrate the difficulty of ranking uncertainties in order of importance. One general point to note is that, although for many cases (particularly those involving late containment failure) the ex-vessel processes may appear to be the dominant contributors to source term uncertainty, many of the uncertainties in ex-vessel phenomena can be traced back to boundary condition uncertainties established by in-vessel phenomena. Thus the in-vessel processes may be important contributors to source term uncertainty even when this is not immediately apparent from the tables.

A number of issues have been cited as capable of affording large reductions in the source terms calculated by the RSS. Examples include in-vessel retention, ex-vessel debris coolability, and mode and time of containment failure. Of these, only the containment failure issue stands out as a single uncertainty which, if resolved favorably, could reduce source terms across the entire range of plants, accidents, and assumptions concerning all the other uncertainties. In any ranking of contributors to source term uncertainty, it is thus inevitable that containment failure issues would be at or near the top of the list. Beyond this, ranking of source term uncertainty contributors is extremely difficult.

7.7.2 Integrated Source Term Uncertainty Information

It is apparent from the extent of the uncertainties discussed in Section 7 and elsewhere in this document that the evaluation of uncertainty in radiological source terms is by no means a trivial task. Many contributory uncertainties in accident phenomena must be taken into account. If we regard the source term as the "output" of the problem of accident evaluation, then the parameter space to be spanned in order to determine the range of that output, and to resolve what steps should be taken to narrow that range, is very large. While it is difficult to generalize statements on source term uncertainty across the whole range of plants and accidents of interest, it is obvious from the combined ranges of plant conditions, containment failure modes and times, and FP release and retention possibilities described in this report that the range of radiological source terms which could arise from severe accidents in LWR power plants is very wide.

Such general statements are unsurprising, and, without being made particular and quantified, clearly provide little technical basis for regulatory decisions or for the assignment of research priorities. In order to provide at least some clarification, we continue with a discussion of the implications of the uncertainties discussed in this document for the broad types of source terms used in PRAs to date.

All reactor safety studies performed to date have used a relatively small set of release categories or source terms to characterize the spectrum of possible scenarios for release of radioactivity from LWR power plants in severe accidents. The RSS, for example, used eight categories for a PWR and five for a BWR. The more recent Sizewell study, using a slightly different categorization, used a total of twelve release categories. Viewed in the most simplistic manner, all of these rel e categories or "source terms" can be loosely grouped into two types:

1. Major, early, above-ground containment failures leading to release of most of the core inventories of cesium, iodine, and noble gases, plus significant quantities of other radionuclides, from the plant.

2. Later containment failures, failures below ground, and non-failures, leading to much smaller releases of radioactivity from the plant.

The radiological consequences, for combinations of weather and population distribution where they are significant, of the former type of source term are obviously much greater than those of the second. The contribution of each of the two types to risk depends very much on the weightings given to the different release categories by analysts. In the Reactor Safety Study and several other PRAs, the former type have been risk-dominant. In studies (e.g. Zion, Indian Point) placing greater confidence in containment survivability, the latter, smaller type of source term tends to dominate risk. Uncertainties in both types of source term are clearly of interest, because each dominates some studies. We proceed by describing, insofar as is possible, what are the implications of the uncertainties discussed in this section for the two very broad, loose source term categories.

Early containment failure, large source terms

A number of phenomena have been identified and discussed in this document and elsewhere that are capable of reducing source terms even for situations in which containment failure follows rapidly from, or even precedes, that of the RPV. These include retention in the reactor core, elsewhere in the RCS, in the containment building (if the time scale for release of gas therefrom is relatively long) and along the pathways by which gas leaves the plant. The range of possible mitigation factors is very wide, and depends very much on the mechanics of how the plant responds to the accident. The combination of RCS, containment, and auxiliary building retention could easily mitigate even early releases for some plants and accidents by several orders of magnitude. It is thus possible that even very "severe" accidents (in terms of the extent of core damage and suddenness of containment failure) could, given the right combination of plant conditions and parameter values from within currently uncertain ranges, result in relatively benign source terms from the plant.

In contrast, only the phenomena relating to fission-product release are capable of exacerbating such source terms, for the RSS and subsequent studies have, in their larger source term evaluations, ignored the possibility of any significant retention of radionuclides within the reactor plant. Even so, there are several possible ways by which early, large source terms might be augmented. One is that in-vessel releases of medium and lower volatility fission products could be significantly larger than those used in PRAs to date (Subsection 7.1).

Another is that ex-vessel release processes could add sizeable fractions of the core inventory of medium and lower volatility fission products (e.g., Mo, Ba, Sr, Ru, La, Ce) to the larger source terms considered in the RSS and subsequently. The phenomenon of high-pressure melt ejection (7.3.1) is an obvious example, the wider range of fission products predicted by recent, mechanistic calculations[3 7 62 81] to be released during core/concrete interactions than commonly assumed in PRAs to date is another.

Late containment failure, smaller source terms

A large number of phenomena are capable of enhancing the degree of retention of radionuclides within reactor plants. Many models of containment FP and aerosol behavior predict several orders of magnitude mitigation over and above the values used in the RSS to be afforded by retention in the reactor containment building for many plants and accidents, even in the absence of such factors as ESF operation and steam condensation. Retention along narrow leakage pathways, and in auxiliary buildings, provides further potential for orders of magnitude source term mitigation.

Off-site radiological consequences are, though, more sensitive to source term changes when the source terms themselves are larger, for both early health effects and evacuation/interdiction areas show marked threshold effects - small enough source terms present little hazard whatever the extent of further reduction.[6] Thus, despite the potential for enormous reductions of already small source terms (possibly by many orders of magnitude), it is probably of less significance to focus on the lower limits which might be realized than on the phenomena which might increase these smaller source terms. Uncertainties that might make source terms thought to be small still smaller are probably of less immediate concern than those which might make them larger.

A number of fission product behavior uncertainties have been identified here which could augment source terms arising from containment failures late in the course of an accident. One is the possibility of ex-vessel release phenomena (specifically core/concrete interactions) which may be delayed by temporary quenching of debris ex-vessel, and may release fission products up to and beyond the time of even a failure delayed by many hours. The implications are not readily assessed; the mixture of radionuclides released would probably be quite different, qualitatively, from any investigated in off-site consequence calculations to date. Another possibility is that more volatile radionuclides released early in an accident may be retained temporarily in the RCS. Later, as decay heat causes RCS temperatures to rise, they may be re-evaporated and, if gas flow is established into and out of the RCS, they may be released from it. Both these uncertainties relate very much to the timing of release processes into the containment atmosphere relative to that of containment failure. The FPs released by either process would still be subject to orders of magnitude depletion prior to release from the plant, unless some mechanism existed for driving them out of the containment on a time scale shorter than or comparable to that of removal of fission products from the atmosphere of the damaged containment building.

Another, quite different uncertainty that could lead to exacerbation of small, "late" source terms is that relating to re-entrainment of previously deposited fission products. If containment failure is an energetic process (e.g., caused by a hydrogen burn, or simply if containment depressurizes very rapidly), substantial quantities of fission products previously removed from the containment atmosphere may become resuspended. The fraction that would then leave the plant in respirable form would still be very uncertain, but, particularly in the case of a very rapid (few seconds time scale) containment depressurization, it appears that resuspension could lead to substantial exacerbation relative to the quantities of radionuclides suspended in the containment atmosphere at late times.

7.7.3 Concluding Remarks

This discussion outlines some of the phenomena that contribute to source term uncertainties that are clearly large enough to affect the technical bases of regulatory decisions. To provide more detailed source term uncertainty information, far more information would have to be presented and far more effort expended than that spent so far in identifying important uncertainty contributors.

The resources required for comprehensive source-term uncertainty studies, to provide information as to the ranking of uncertainties as well as the overall magnitude of the source term uncertainty, would be considerable. However, it is strongly recommended that further effort be expended in this field. The computer codes developed by continued USNRC research since WASH-1400 and NUREG-0772 have provided valuable tools, still under development, for more comprehensive incorporation of accident phenomena into source-term descriptions. The QUEST study is providing useful initial estimates of the magnitude of that uncertainty for certain specific plants and accidents. Much scope for progress remains, though, both in model development and in source term uncertainty analysis. Further work on source term uncertainty should both improve the technical basis for regulatory decision making and provide clearer guidance for continued research on source-term related issues.

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8. DISCUSSION

As mentioned in Subsection 1.1, the SAUNA working group was formed to coordinate work on common needs for

1. A consolidated list of severe accident uncertainties,

2. Valid methods for estimating the magnitude of severe accident uncertainties, and

3. Knowledge of what research should be most effective in reducing the severe accident uncertainties.

Most of this report is devoted to the first need--identification of severe accident uncertainties. This section also provides some discussion of the latter two needs and suggests some approaches which may prove useful in satisfying these needs. Insofar as these approaches have not been tested, the discussion is necessarily tentative.

Subsection 8.1 discusses accident characteristics that determine risk. The importance of particular phenomena and uncertainties depends, in large part, on how they influence these characteristics. Subsection 8.2 discusses the ways in which the uncertainties that have been identified thus far in this report contribute to uncertainty in the results of severe accident analysis problems. Subsection 8.3 summarizes what we feel are appropriate interim and long-term activities related to severe accident uncertainty analysis.

8.1 Accident Characteristics that Determine Risk

Section 9.6 of the PRA Procedures Guide reviews the factors that are important in the determination of nuclear reactor offsite risk and discusses the importance of each and the associated uncertainties.[1] Major inputs to the offsite risk calculation are the characteristics of the source term (magnitude, frequency, and duration of release; warning time; particle size distribution and chemistry; etc.). These are the outputs of the onsite portions of the analysis which have been the subject of this report.

Because of the additional uncertainties in the offsite risk calculations (discussed in Reference 2) and the natural division of the risk assessment between onsite and offsite aspects, it is convenient to use the source term characteristics that are important inputs to the offsite consequence analysis as end points for the onsite portion of the analysis. Table 8-1 is taken from the PRA Procedures Guide and lists these characteristics, the sensitivity of risk to them, and their contributions to the uncertainties in risk.[1] Examination of the table suggests three groups of factors that determine the risk:

1. Magnitude of the source term, chemical form, and particle-size distribution for each nuclide of interest,

2. Frequency of occurrence of each important accident sequence and associated release, and

Table 8-1 Source terms: Sensitivities and uncertainties[1]

Parameter or modeling assumption	Quantities most sensitive to parameter or modeling changes	Contribution to uncertainties in CCDFs			Sensitivity studies
		Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage	
Magnitude of source term (modeling and parameter)	Airborne and deposited levels of radioactive material	Major	Major	Major	RSS [3] Wall et al [4] Zion study [5]
Frequency of occurrence of each sequence/category (modeling and parameter)	Frequency of CCDFs	Major	Major	Major	Limerick study [6] Zion study [5] Erdmann et al [7]
Time of release (parameter	Time available for evacuation	Major (except peaks	Low	Low	
Duration of release (parameter)	Plume width Possibility of wind shift or weather change during release	Major (especially peaks)	Low	Low	Griffiths [8] Benchmark exercise [9 10] Zion study [5]
Warning time (parameter)	Time available for evacuation	Major (except peaks)	Low	Low	
Building wake or dimensions of release (parameter)	Airborne concentration near reactor	Low	Low	Low	
Rate of heat release (parameter and modeling)	Height of plume rise	Moderate to major for some sequences (low for peaks)	Low	Low	Russo [11] Russo et al [12] Kaiser [13 14]
Particle-size distribution (parameter and modeling)	Deposition velocity Washout coefficients Dose-conversion factors	Moderate	Moderate	Major	Kaul [15] Benchmark exercise [9 10] Hunt et al [16]
Chemical form (parameter)	Dose-conversion factors Deposition velocity	Moderate	Moderate	Moderate	Hunt et al [16]

3. Time between accident initiation and release, warning time, release duration, and rate of heat release.

The magnitude and frequency of the source term are important for all consequence measures. The chemical form and particle-size distribution are important for land contamination and uptake of deposited radionuclides, whereas timing factors and heat of release are important only for early health effects.

Because these groups of characteristics are all important to the determination of risk but affect the various measures of risk in different ways, they should all be considered in the assessment of the importance of particular uncertainties in the onsite portions of accident analyses.

8.2 Relationships between Uncertainties

This subsection sets out the way in which the uncertainties that have been identified in this report contribute to uncertainty in the results of severe accident analyses. All the relationships (whereby one uncertainty contributes to another) identified here are based on judgement; to provide analysis to demonstrate the importance of all these relationships would be a very large task wholly outside the scope of the current project.

Our intention here is only to identify potentially important relationships (rather than unimportant ones). They can only be potentially important, and not absolutely important, for four reasons: First, whether or not the results of a calculation are sensitive to the value of a particular uncertain quantity will, in general, depend on the values of other quantities that are themselves uncertain. As an example from severe accident analysis, if ex-vessel melt/water interactions produce enough steam to fail containment directly without consideration of hydrogen combustion, then uncertainties in hydrogen formation and combustion are unimportant. On the other hand, smaller steam production and/or higher containment strength, both within their ranges of uncertainty, may either make hydrogen uncertainties crucial to the question of containment failure or, at the other extreme, unimportant again if containment integrity is assured within the whole uncertainty range of hydrogen phenomena. Thus the importance of uncertainties in hydrogen processes may be conditioned on other uncertainties. Second, the importance of a contributing uncertainty will depend upon which quantity is sought as output from a calculation. For example, uncertainty in containment-failure pressure is unimportant if only core-melt frequency is sought, but it may well be important if the frequency of early containment failure is required. Third, if the quantity being calculated is a property of a particular accident sequence, rather than being integrated over all possible sequences, the importance of its various contributing uncertainties will, in general, depend upon which sequence is being considered. Fourth, the relative importance of different contributing uncertainties may be different for different plants.

Because we identify potentially important relationships (ones that we judge to be important for some but not necessarily all severe accident analyses), the number of them set out here may well exceed the number that are actually important when a particular problem is investigated. This should be borne in mind when considering the apparent complexity of some of the relationships set out below.

8.2.1 Framework of Relationships

We wish to construct a framework of potentially important relationships in which uncertainties may contribute importantly to other uncertainties. An appropriate indicator of potential importance is the effect of contributing uncertainties on the risk-determining factors enumerated at the end of Subsection 8.1. When defined in its broadest sense, risk entails a list of all possible events with their estimated frequencies and consequences.[5 17] It therefore contains all the different quantities that might be sought in severe accident analyses and hence will be affected by all the different important contributions of one uncertainty to another.

In order to establish relationships between uncertainties in the three risk-determining factors and the many individual uncertainties identified in this report in Tables 2-3, 2-4, 2-10, 2-11, 3-4, 3-6, 3-7, 4-1, 4-2, 4-3, 5-1, 6-7, and 7-8 through 7-18, we set up three broad classes of uncertainty and twelve key uncertainties. These groupings are used as intermediate stages to discuss the relationships between the individual uncertainties and uncertainties in the risk-determining factors.

The framework used is not claimed to be unique; however, it was designed to display the relationships between the individual uncertainties and the risk-determining factors relatively simply and completely. Use of a simpler framework would entail the risk of missing some relationships.

8.2.2 Classes of Uncertainty

It is convenient to divide accident uncertainties into three broad classes:

- frequency of severe accidents,
- time and mode of containment breach, and
- characteristics of fission products within containment.

("Characteristics" here means quantity, distribution in space, chemical form, and particle-size distribution for each nuclide of interest.)

Table 8-2 shows to which of the risk-determining factors each of these classes contributes uncertainty. Each black dot in the table indicates that the class of uncertainty to its left is judged to contribute directly and importantly to the uncertainty in the risk-determining factors listed above it. Thus, for example, uncertainty in time and mode of containment breach contributes uncertainty to factors in all three lists.

Table 8-2 Contributions to uncertainty in risk-determining factors

Classes of uncertainty affecting the risk-determining factors	Risk-determining factors: Magnitude of source term, chemical form, and particle-size distribution for each nuclide of interest	Frequency of occurrence of each important accident sequence and associated release	Time between accident initiation and release, warning time, release duration, and rate of heat release
Frequency of severe accidents		•*	
Time and mode of containment breach	•	•	•
Characteristics of fission products within containment	•		

*A • indicates that the class of uncertainties at the left of the • is judged to contribute directly and importantly to the uncertainty in the determining factors above the •.

8.2.3 Key Uncertainties

Examination of Sections 2 through 7 of this report indicates to us that the uncertainties affecting the risk-determining factors can be summarized in a small group--albeit each is relatively broadly encompassing--of key uncertainties. They are summarized below, in the order in which they are discussed in Sections 2 through 7. Statements summarizing why we consider each of these uncertainties to be a key uncertainty are also provided, and the subsection numbers in which each key uncertainty is discussed are listed for ready reference.

1. <u>Definition of accident sequences</u>. Identification of sequence initiators, equipment failure modes (including partial failures), dependent failures, and modes of recovery from faults may be incomplete. Also, the definition and the effects of partial attainment of some success criteria are uncertain. [Subsection 2.1]

2. Frequencies of events and probabilities of faults. The limitations of data pertaining to initiating events, component failures, and human errors mean that estimates of the frequency of the combinations of these events in an accident sequence can be very uncertain. Time-dependent effects, plant-to-plant variability, and the use of incomplete or inaccurate data add to this uncertainty. [Subsection 2.2]

3. Human actions and inactions. Human actions or inactions may contribute to accident initiation. They may also affect the progression of an accident in both favorable and unfavorable ways. Possibly complex actions extraneous to, or in the absence of, specified procedures are particularly difficult to predict. [Subsections 2.1, 2.2]

4. Progression of core damage following initial loss of intact geometry. Configuration, heatup rate (due to oxidation), heat loss rate, and hydrogen evolution are significantly uncertain, including effects resulting from coolant injection into an overheated core. [Section 3]

5. In-vessel core-melt/coolant interactions, including the possibility of a steam explosion of magnitude sufficient to fail containment. Core-melt/coolant interactions can affect the particle size, temperature and position of debris, and hydrogen production. A powerful enough steam explosion could result in simultaneous breach of containment and release of fission products to the offsite environment. The probability of such an event is uncertain (unless no water is present in the reactor vessel). [Subsections 3.4, 3.5, 6.3]

6. Location and nature of ex-vessel core debris. The behavior of melt and core debris after leaving the vessel affects pressure spikes, ex-vessel steam explosions, aerosol and fission-product evolution, hydrogen production, and debris coolability. [Subsections 4.1, 4.2]

7. Magnitude and timing of pressure source caused by ignition of flammable or detonable gases in containment, and consequent release of heat. Hydrogen is evolved during the oxidation of core zirconium and other metals, and hydrogen and carbon monoxide can be released by core/concrete interactions. For many plants and accident sequences, the pressures accompanying combustion of these gases are uncertain. [Subsection 5.1]

8. Magnitude of containment pressure sources due to steam and noncondensable gases, including those from steam spikes, core/concrete interactions, and gas heating by hot aerosols. Accumulation of noncondensable gases and steam and heating by hot aerosols can threaten containment integrity owing to high temperatures and pressures. [Subsection 5.2]

9. Effects of severe accident conditions, including aerosol deposition and hydrogen burning, on engineered safety features and other required systems. Certain equipment needed to arrest the progression, or mitigate the effects, of severe accidents may be rendered less effective or ineffective because of deflagrations, radiation, steam, atmospheric temperatures, aerosols, etc. produced by the accident. [Subsections 2.1, 5.1, 5.2, 5.3]

10. Containment breach pressure and size due to quasi-steady overpressure. It is difficult to predict the pressure that will result in containment building failure and the resulting equivalent size of the breach to the environment. Stylized and unjustified assumptions are often made in the absence of detailed structural evaluations. [Subsection 6.5]

11. Release of fission products into the containment atmosphere. Our understanding of release and transport phenomena in the RCS, and of ex-vessel release phenomena, is at a developmental stage. This leads to large uncertainties in the magnitude and timing of FP and aerosol releases a) from the RCS and b) via ex-vessel processes. [Subsections 7.1, 7.2, 7.3]

12. Attenuation of fission products in containment. Despite our improving ability to predict aerosol behavior, uncertainties in the inputs (releases and atmosphere conditions) to aerosol calculations make the quantities of FPs airborne at any given time substantially uncertain. Releases from containment are made more uncertain by possibilities for effects during discharge from the plant ranging from extensive retention to substantial re-entrainment. [Subsections 7.4, 7.5]

Table 8-3 sets out which of the twelve key uncertainties are judged to contribute importantly to the three uncertainty classes of Subsection 8.2.2. Some of the contributions are direct, and some are indirect. An indirect contribution means that the key uncertainty contributes to one or more other key uncertainties that, in turn, contribute to the affected uncertainty class. Indirect relationships of this kind are not judged to be necessarily less important than direct ones. It will be seen that each of the key uncertainties is judged to contribute, directly or indirectly, to one or two of the three uncertainty classes.

Some of the direct relationships in Table 8-3 are obvious (for example, that between frequencies of events and probabilities of faults, and frequencies of severe accidents). Table 8-4 lists some of the less obvious reasons that the direct relationships are judged to be important. It also lists the key uncertainties that provide linkage in cases where relationships are indirect. Generally, only indirect relationships involving one intermediate stage are listed here; a multitude of indirect relationships linked by chains of two or more

Table 8-3 Contributions of key uncertainties to uncertainty classes

Key uncertainties	Classes of uncertainty affected by each key uncertainty: Frequency of severe accidents	Time and mode of containment failure	Characteristics of fission products within containment
1. Definition of accident sequences.	•*		
2. Frequencies of events and probabilities of faults.	•	•	
3. Human actions and inactions.	•	•	○†
4. Progression of core damage following initial loss of intact geometry.		○	○
5. In-vessel core-melt/coolant interactions, including the possibility of a steam explosion of magnitude sufficient to fail containment.		•	•
6. Location and nature of ex-vessel core debris.		•	•
7. Magnitude and timing of pressure source caused by ignition of flammable or detonable gases in containment, and consequent release of heat.		•	○
8. Magnitude of containment pressure sources due to steam and noncondensible gases, including those from steam spikes, core/concrete interactions, and gas heating by hot aerosols.		•	○
9. Effects of severe accident conditions, including aerosol deposition and hydrogen burning, on engineered safety features and other required systems.		○	•
10. Containment breach pressure and size due to quasi-steady overpressure.		•	•

Table 8-3 Contributions of key uncertainties to uncertainty classes (Continued)

Key uncertainties	Classes of uncertainty affected by each key uncertainty: Frequency of severe accidents	Classes of uncertainty affected by each key uncertainty: Time and mode of containment failure	Classes of uncertainty affected by each key uncertainty: Characteristics of fission products within containment
11. Release of fission products into the containment atmosphere.		○	●
12. Attenuation of fission products in containment.		○	●

*A ● indicates that the key uncertainty at the left of the ● is judged to contribute directly and importantly to the class of uncertainty above the ●.

†A ○ indicates a contribution judged to be indirect and important.

Table 8-4 Reasons why key uncertainties contribute to uncertainty classes

Key uncertainties contributing to uncertainty classes	Reasons for contributions to these uncertainty classes: Frequency of severe accidents	Time and mode of containment failure	Characteristics of fission products within containment
1. Definition of accident sequences.	Possibly incomplete identification of sequences		
2. Frequencies of events and probabilities of faults.	Obvious	Possibility of unusually large containment leakage at beginning of accident.	
3. Human actions and inactions.	Direct: Possibilities of causing accident initiation and of preventing progression to a severe accident	Direct: Possibility of manual containment isolation or manual overriding of containment isolation during accident. Indirect* through key uncertainty 5.	Indirect* through key uncertainties 5, 9, or 11.
4. Progression of core damage following initial loss of intact geometry.		Indirect* through key uncertainties 5, 6, or 7.	Indirect* through key uncertainties 5, 6, or 11.
5. In-vessel core-melt/coolant interactions, including the possibility of a steam explosion of magnitude sufficient to fail containment.		Direct: Possibility of direct failure by steam explosions. Indirect* through key uncertainties 6, 7, or 8.	Direct: Deposition and suspension processes in RCS: mode of ejection of debris and FPs from RPV. Indirect* through key uncertainties 6, 9, 11, or 12.
6. Location and nature of ex-vessel core debris.		Direct: Basemat melt-through. Indirect* through key uncertainties 7 or 8.	Direct: Release of FP gases, vapors, and aerosols from debris when melt escapes from RPV and during concrete attack and ex-vessel melt-coolant interactions Indirect* through key uncertainties 9, 11, or 12.

Table 8-4 Reasons why key uncertainties contribute to uncertainty classes (Continued)

Key uncertainties contributing to uncertainty classes	Reasons for contributions to these uncertainty classes: Frequency of severe accidents	Time and mode of containment failure	Characteristics of fission products within containment
7. Magnitude and timing of pressure source caused by ignition of flammable or detonable gases in containment, and consequent release of heat.		Direct: Possibility of overpressure failure. Indirect* through key uncertainty 10.	Indirect* through key uncertainties 9, 10 or 12.
8. Magnitude of containment pressure sources due to steam and non-condensible gases, including those from steam spikes, core/concrete interactions, and gas heating by hot aerosols.		Direct: Possibility of overpressure failure. Indirect* through key uncertainty 10.	Indirect* through key uncertainties 9, 10, or 12.
9. Effects of severe accident conditions, including aerosol deposition and hydrogen burning, on engineered safety features and other required systems.		Indirect* through key uncertainty 8.	Direct: Continued operation of containment sprays and air filters. Indirect* through key uncertainty 12
10. Containment breach pressure and size due to quasi-steady overpressure.		Obvious.	Direct: Passage of fission products out through breach at containment failure. Indirect* through key uncertainty 12.
11. Release of fission products into the containment atmosphere.		Indirect* through key uncertainties 12 <u>and</u> 8†	Direct: Obvious. Indirect* through key uncertainty 12.
12. Attenuation of fission products in containment.		Indirect* through key uncertainty 8.	Direct: Obvious. Indirect* through key uncertainty 9

*Reasons for contributions from one key uncertainty to another are listed in Table 8-5.

†Exceptionally in this case, the simplest relationship identified is linked by a chain of two key uncertainties.

key uncertainties also exist. For example, uncertainty in core-melt progression (key uncertainty 4) contributes to uncertainty in in-vessel core-melt/coolant interactions (key uncertainty 5), which includes uncertainty in in-vessel hydrogen production. This, in turn, contributes to uncertainty in containment pressure sources due to ignition (key uncertainty 7), which contributes to uncertainty in containment failure.

Table 8-5 lists reasons why the relationships between key uncertainties, which produce the indirect relationships listed in Table 8-4, are judged to be important. As an example of the propagation of uncertainty indicated by these tables, uncertainty in progression of core damage (key uncertainty 4) is judged to contribute to uncertainty in the location and nature of ex-vessel core debris (key uncertainty 6) because of the influence of core-melt properties, such as temperature and gas solubility, on melting attack of the RPV and flow of melt through an RPV breach. In turn, the uncertainty in the nature and location of ex-vessel debris propagates into uncertainty in characteristics of fission products in containment because it affects the release of fission products, both when melt escapes from the RPV and during concrete attack and ex-vessel core-melt/coolant interactions. Uncertainty in core-melt progression is also judged to propagate into uncertainty in the characteristics of fission products in containment through key uncertainties 5 and 11, as well as 6.

Combining the information in Table 8-3 with that in Table 8-2 enables us to see which key uncertainties contribute potentially important uncertainty to each of the three risk-determining factors. Key uncertainty 1 (definition of accident sequences) contributes uncertainty to the frequency of occurrence of each important accident sequence and associated release. The other key uncertainties each contribute uncertainty, directly or indirectly, to each of the three risk-determining factors.

8.2.4 Individual Uncertainties

Individual uncertainties that contribute to most of the key uncertainties are listed in tables in Sections 2 through 7. Table 8-6 shows which of these tables lists uncertainties that contribute to each key uncertainty.

8.3 Recommended Additional Work

As was discussed in Section 1, the NRC's Severe Accident Research Plan calls for the identification, quantification, and reduction of uncertainties in the analysis of severe accidents. Because the scope of the working group was limited, this report treats only the identification task, although the need for more thorough study is acknowledged. In particular, the preparation of this report required plausibility arguments and speculative reasoning, which introduce uncertainty into the results. A less speculative, better-justified identification, together with estimation of the magnitude of the uncertainties, should be sought. Although efforts to quantify and propagate uncertainties on a generic basis could be attempted, it is the authors' opinion that

Table 8-5 Reasons why some key uncertainties contribute to other key uncertainties

Key uncertainties* Contributing	Contributed to	Reasons
3	5	Human actions or inactions can affect the possibility of core reflooding.
3	9	Human actions or inactions may switch off or restore ESFs or other systems or fail to do so.
3	11	Human actions or inaction may open or shut valves, releasing FPs from the RCS or failing to do so.
4	5	Temperature and other properties of core melt and its rates of production and discharge affect core-melt/coolant interactions.
4	6	Properties and flow pattern of core melt may affect RPV failure position, time, and size, and hence location of debris ex-vessel.
4	7	Quantity of hydrogen produced during core melting affects possible later hydrogen burning.
4	11	Properties and flow pattern of core melt may affect RPV failure position, time, and size, and hence release of FPs from RCS.
5	6	Dispersal of debris if steam explosion fails RPV.
5	7	Hydrogen production during in-vessel core-melt/coolant interactions.
5	8	Pressure spike if steam explosion fails RPV.
5	9	Effect of missiles on ESFs and other systems if steam explosion fails RPV.
5	11	Release of FPs to containment if steam explosion fails RPV.
5	12	Failure of RPV by in-vessel steam explosion affects conditions (pressure, temperature, steam quantity, etc.) determining FP deposition in containment -- nature of breach affects release of FPs from containment if steam explosion fails containment.
6	7	Production of flammable gases during core/concrete interactions.
6	8	Production of steam during debris quenching; production of steam and gas during core/concrete interactions; direct heating of gas by debris.
6	9	Effect of debris on ESFs and other systems.

Table 8-5 Reasons why some key uncertainties contribute to other key uncertainties (Continued)

Key uncertainties* Contributing	Contributed to	Reasons
6	11	Possible resuspension of FPs previously deposited in containment when debris is released from RPV or during ex-vessel core-melt/coolant interactions.
6	12	Ex-vessel melt/water and melt/concrete interactions affect conditions (pressure, temperature, steam quantity, etc.) determining FP deposition in containment.
7	9	Effects of burning on ESFs and other systems.
7	10	Possibility of ignition leading to containment breach.
7	12	Effects of temperature and pressure changes (caused by ignition) upon deposition and resuspension processes.
8	9	Effect of pressure changes on ESFs and other systems.
8	10	Possibility of pressure increases leading to containment breach.
8	12	Effects of pressure changes upon deposition and resuspension processes.
9	8	Effectiveness of ESFs and other systems will affect containment pressure.
9	12	Effectiveness of equipment (e.g., sprays, filters) in directly removing atmospheric FPs. Effect of continued operation or failure of decay-heat-removal system on containment temperature and pressure, and hence on deposition and resuspension processes.
10	12	Nature of breach affects both attenuation of FPs during release from containment and resuspension.
11	12	Release of FPs into containment affects their attenuation therein.
12	8	Contributions of decay heat to steam production, concrete attack, and gas heating.
12	9	Effect of decay heat in aerosols deposited upon equipment.

*The key uncertainties corresponding to the numbers in these columns are identified at the beginning of Subsection 8.2.3 and in the first column of Table 8-4.

Table 8-6 Contributions of individual uncertainties to key uncertainties

Lists of individual uncertainties contributing to key uncertainties	Key uncertainties affected by listed individual uncertainties
Tables 2-3, 2-10	1. Definition of accident sequences.
Table 2-11	2. Frequencies of events and probabilities of faults.
Tables 2-4, 2-10, 2-11	3. Human actions and interactions.
Tables 3-4, 3-6	4. Progression of core damage following initial loss of intact geometry.
Table 3-7	5. In-vessel core-melt/coolant interactions, including the possibility of a steam explosion of magnitude sufficient to fail containment.
Tables 4-1, 4-2	6. Location and nature of ex-vessel core debris.
Table 5-1 (part)	7. Magnitude and timing of pressure source caused by ignition of flammable or detonable gases in containment, and consequent release of heat.
Tables 4-2, 4-3	8. Magnitude of containment pressure sources due to steam and noncondensible gases, including those from steam spikes, core/concrete interactions, and gas heating by hot aerosols.
Table 5-1 (part)	9. Effects of severe accident conditions, including aerosol deposition and hydrogen burning, on engineered safety features and other required systems.
Table 6-7 (part)	10. Containment breach pressure and size due to quasi-steady overpressure.
Tables 7-8 through 7-12	11. Release of fission products into the containment atmosphere.
Tables 7-13 through 7-18	12. Attentuation of fission products in containment.

several potential pitfalls could be avoided by performing additional studies in a plant- and accident-specific framework. We therefore recommend use of an existing or planned PRA to provide a framework for additional work with the above-discussed goals. To be most comprehensive, additional work would benefit from being combined with an evaluation of the uncertainties in the offsite consequence analysis. This approach has the advantages of specificity (specific plant and sequences), realism (focusing on relevant sequences rather than hypothetical situations), and risk-orientation (frequency and consequence modeling), but would require significant resources. A less comprehensive and hence smaller task is an uncertainty study for a specified subset of a PRA. The QUEST study, being conducted at Sandia National Laboratories, attempts to estimate the uncertainties in fission product source terms for specific plants and sequences.[18] The QUEST study is based on some of the insights and uncertainties identified in the present study.

Another activity which may have considerable benefit is sensitivity analysis, using accident analysis codes which exist or are under development. The key element in such studies is discovering ways in which the codes can be manipulated or modified to yield the ranges of propagated and output parameters that are needed to represent the currently estimated uncertainties adequately. A possible problem in such analyses is the large amounts of computer time which may be required. Properly performed, however, sensitivity analyses have the potential for providing needed quantitative perspective on the influences, feedbacks, and synergisms characterizing the relationships between accident phenomena and accident consequences.

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Understanding of severe accidents in light-water reactors is currently beset with uncertainty. Because the uncertainties that are present limit the capability to analyze the progression and possible consequences of such accidents, they restrict the technical basis for regulatory actions by the U. S. Nuclear Regulatory Commission (NRC) regarding severe accidents. It is thus necessary to attempt to identify the sources, and quantify the influence, of these uncertainties.

As a part of ongoing NRC severe-accident programs at Sandia National Laboratories, a working group was formed to pool relevant knowledge and experience in assessing the uncertainties attending present (1983) knowledge of severe accidents. This, the initial report of the Severe Accident Uncertainty Analysis (SAUNA) working group, has as its main goal the identification of a consolidated list of uncertainties that affect in-plant processes and systems. Many uncertainties have been identified. A set of so-called "key" uncertainties summarizes many of the identified uncertainties. Quantification of the influence of these uncertainties, a necessary second step, is not attempted in the present report, although attempts are made qualitatively to demonstrate the relevance of the identified uncertainties.

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