SOURCE TERM ASSESSMENT,
CONTAINMENT ATMOSPHERE CONTROL SYSTEMS,
AND ACCIDENT CONSEQUENCES

Report to CSNI
by an OECD/NEA Group of Experts

April 1987
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CSNI Report 135 summarizes the results of the work performed by CSNI's Principal Working Group No. 4 on the Source Term and Environmental Consequences (PWG4) during the period extending from 1983 to 1986. It consists of five parts:

1. a Foreword and Executive Summary prepared by PWG4's Chairman;
2. a Report on the Technical Status of the Source Term (starting on page 11);
3. a Report on the Technical Status of Filtration and Containment Atmosphere Control Systems for Nuclear Reactors in the Event of a Severe Accident (starting on page 273);
4. a Report on the Technical Status of Reactor Accident Consequence Assessment (starting on page 291);
5. a list of members of PWG4 (starting on page 317).

Part 2, which constitutes the bulk of CSNI Report 135, is the Technical Annex to "Nuclear Reactor Accident Source Terms - Report by an NEA Group of Experts" published by the OECD (NEA) in March 1986. It was submitted to extensive peer review during the latter part of 1985 and during 1986. It should be stressed, however, that it does not contain information more recent than that summarized in "Nuclear Reactor Accident Source Terms", in order to maintain consistency between the Report and its Technical Annex. Results of work performed by PWG4 and its Groups of Experts during 1986 and 1987 will be published in other documents, still in preparation.

It should also be stressed that Part 3 does not yet include results and conclusions of post-Chernobyl studies. These will be included in new reports, which will become available some time in 1988.

CSNI Report 135 was endorsed for publication by PWG4 and CSNI at the end of 1986.
FOREWORD AND EXECUTIVE SUMMARY
(D.F. Torgerson, PWG4 Chairman)
Considerable effort has been expended in NEA member countries to improve the methodology pertaining to source terms and environmental consequences. Since there has been rapid development of these methodologies, PWG4 and its experts groups have been preparing state-of-the-art reports to ensure that current information is readily available. This document contains the latest information on some important topics relating to source terms, accident consequence assessment, and containment atmospheric control systems. It is intended that much of the information presented herein (particularly in the source term area) serve as an in-depth record that will prove useful to both technical experts and to readers who require more than a general knowledge of the issues. The following summary is an overview of the information included in this document for those readers who wish to become acquainted with the scope and nature of the current work.

**Accident Consequence Assessment**

Accident Consequence Assessments (ACA) have had wide application in several areas of nuclear safety, including risk assessments of nuclear installations, assessments of different design options, setting safety goals, getting research priorities, siting and emergency planning. The codes used for ACA calculations include atmospheric dispersion models, foodchain models, dose models, health effects models, and countermeasures models. The essential input data for the calculations include source terms, atmospheric data, population and agriculture data. The output from the assessments include information on air/ground concentrations, inhalation and ground doses, food contamination, numbers of late and early effects, interdicted land areas, and food restrictions. In general, there are two broad categories of offsite consequences that may result from accidental release that need to be evaluated in an accident consequence assessment: first, there are the health effects in the exposed population and its descendants; secondly, there is the economic (and social) impact of countermeasures that may be taken to reduce the exposure, and, thus, the health implications in the population.

The largest uncertainty in off-site consequences is due to the source term, but even in the case of fixed releases of radioactive material, the consequences will vary considerably with the conditions pertaining at the time - for example, with the prevailing meteorological conditions, the season, and the location and habits of the population. In addition, the ACA models themselves have uncertainties arising from an incomplete understanding of the phenomena and processes involved in the transport of released radioactive materials to man, and the health, environmental, and economic consequences that result. Thus, there is currently considerable work ongoing to further develop ACA tools in NEA member countries. Examples of this work are the US NRC sponsored MELCOR program, the CEGB/UKAEA CONDOR code, the Nordic Safety Program, and the Federal Republic of Germany UFOMOD code system.

The new development work is leading to a number of improvements. Some of the more important improvements are being made in the atmospheric dispersion and transport models (e.g., many models are now being modified to include a multi-puff model), in the capabilities for incorporating more detailed land-use characteristics (especially the differentiation of urban and rural areas), in emergency-response data and models, and in the models for radiological health effects and potential economic impacts.
Owing to the importance of pathway parameters for ACA evaluations, PWG4's Group of Experts on Accident Consequences has completed a survey of NEA member countries to collect pathway parameter values. The main objective was to summarize the available information and assess its adequacy for accident consequence modelling. The parameters evaluated were decontamination, radionuclide behaviour in urban areas, shielding, filtering effect of houses and deposition indoors, wet and dry deposition velocities, migration of radionuclides in soil, decontamination under winter conditions, and agricultural pathways. The value of the exercise has been considerable, since it brought together all available information in a systematic way. The results of the exercise are summarized in document SINDOC(86)38 - DRAFT (GRECA Survey on Pathway Parameter Evaluation). It is important to note, however, that the survey was completed before the Chernobyl accident; consequently, new projects are being initiated in ACA that could have significant impact. In particular, there are now actual data on a severe accident that can be addressed by ACA methodology.

The ACA area is currently rather dynamic, as a large number of projects are being carried out, or are near completion. The same statement applies to the source term, and since the source term has considerable impact on ACA, it is essential that ACA experts continue to closely review new developments in source term methodology.

Filtration and Containment Atmosphere Control Systems for Severe Accidents

Much of the recent interest in this area has focussed on filtered vented containment systems for the mitigation of releases during severe accidents. PWG4 has conducted a survey of member countries to determine the different types of systems that are, or will be, installed, and to examine how the systems would be used. Although the report takes into account some of the impact of the Chernobyl accident, the main part of the survey was completed before the accident. Therefore, it is possible that some national positions on filtered vented containments are still evolving. Accordingly, PWG4 will continue to monitor the situation.

Source Term

Due to the continuing rapid development of source term technology, PWG4 has put considerable effort into addressing the key source term issues. The report contains a comprehensive discussion of source term issues that has been prepared by CSNI's Special Task Force on Source Terms, as a Technical Annex supporting the conclusions in their state-of-the-art assessment of the new source term studies, "Nuclear Reactor Accident Source Terms", published by the OECD (NEA) in March 1986. It is intended that this document provide more detailed information on the various technical areas impacting on the source term. It has been peer-reviewed, and we feel it is a reasonable account of the state of knowledge at the time of the source term studies (late 1984/early 1985).
However, it was recognized that further in-depth information was required in some areas of source term technology. Accordingly, PWG4 and its Group of Experts on the Source Term (GRESt) identified some key issues that had not been completely addressed in previous studies, and GRESt undertook to carry out original appraisals to further characterize these issues. The detailed results of this work are given in CSNI Report 136, "Report on Selected Source Term Topics".

In general, it is felt that considerable progress has been made, and that source terms can be calculated on a more mechanistic basis. It is also apparent that source terms cover a very wide range of technologies and the various effects cannot be treated in isolation. In particular, there is a continuing need to couple thermalhydraulics and fission product behavior when assessing the transport of radioactivity during severe accidents. Another important undertaking is to assess the uncertainties associated with the new source term information.

The question of the sufficiency of current source term information has been thoroughly addressed in the report by the Special Task Force on Source Terms, and will not be repeated here. However, some of the remaining technical issues that have been identified in the Task Force's report have been further characterized by GRESt. It is emphasized that work is currently underway in NEA member countries that will likely have considerable impact on reducing the remaining uncertainties mentioned in the GRESt report.

There persist some uncertainties regarding the releases of some of the less-volatile species from fuel, such as Ba and Ru, and with the retention of Te due to reaction with the zircaloy cladding. The formation of SnTe (from the Sn in the cladding) may also have some effect on the subsequent transport of Te. Although it is expected that essentially all the iodine and cesium would be released from the fuel during a severe accident, there are currently some questions regarding the stability of CsI. In particular, there is strong experimental evidence that CsI reacts with boric acid (boron is present in the coolant of PWRs, and in the emergency coolant water of many LWRs; borates could also be produced by the decomposition of boron carbide in the control rods). The reaction forms volatile HI in the RCS that could alter the timing and magnitude of the iodine source term. However, the extent of CsI decomposition would be strongly accident-specific, and further quantification is required before the potential impact of this phenomenon can be better understood. The Cs source term, on the other hand, could be attenuated due to the formation of low-volatile cesium borates. Finally, there remain some questions concerning the chemistry of Ba, Sr, and Mo.

Revolatilization of fission products can occur by means of revaporization, resuspension, or re-entrainment. Resuspension refers to the re-introduction of previously deposited solid particles into an air stream, either from a surface or a pool. Re-entrainment describes the formation of liquid solution droplets in the gas phase above a liquid surface by mechanical disintegration of the surface film during bubbly flow, or by disintegration of the bulk liquid during churn turbulent flow. Obviously, these processes can have a strong influence on the timing and magnitude of the release of fission products into the gas phase. Accordingly, the GRESt review has carefully considered revolatilization phenomena, and some recommendations have been made
for future work. In general, it is felt that assessments of fission product leakage into the environment should take into account the new information describing re-entrainment and resuspension.

Pool scrubbing is an important mechanism for the attenuation of airborne radioactivity, but, at present, only limited credit is taken for fission product scrubbing by suppression pools in BWRs or by the quench tanks of PWRs in transient-type accidents. Computer codes have been developed, and extensive experimental work to validate the codes has been done.

The behavior of fission products within containment during a severe accident is a function of the physical conditions which prevail. Three important phenomena which may affect fission product behavior have been reviewed: hydrogen combustion, steam explosions, and pressurized melt ejection. All these phenomena are capable of depositing large amounts of energy into the containment atmosphere within a short time. While our current ability to predict the consequences of such events on the source term is limited, work is underway in those areas that are now recognized to be potentially important.

Hydrogen combustion can have both chemical and physical effects on fission products. With respect to chemical effects, iodine is the only fission product for which significant chemical effects are expected. For example, there is conclusive evidence that airborne organic iodides would be decomposed to yield I\textsubscript{2} or HI during a hydrogen deflagration. Also, the conversion of airborne CsI aerosols to I\textsubscript{2} and HI is possible if the aerosols are dried out by the rise in temperature due to the combustion. However, CsI would likely be airborne only under conditions where quantities of other aerosols would be present, and it is likely that rapid re-attachment of the gaseous iodine to these aerosols would occur. Concerning physical effects, turbulent inertial agglomeration of aerosols will be promoted by hydrogen combustion, but enhancement of deposition would be minimal. Although flame and shock turbulence has been shown to result in the resuspension of dry deposited aerosol particles, wet aerosol particles are unlikely to be resuspended.

There is considerable evidence that steam explosions in the reactor cavity would generate only a very small proportion (<1%) of the aerosols released to the containment atmosphere. Indeed, one of the main effects of steam explosions would be to suspend large quantities of water droplets into containment that would accelerate the removal of the pre-existing aerosols. The chemical effects of steam explosions on Ru and Te require further consideration.

Experiments on pressurized melt ejection phenomenology have demonstrated the potential for injecting large quantities of aerosols into the containment atmosphere. If the associated heating of the containment atmosphere could lead to containment failure, then a mechanism could exist for releasing considerable amounts of activity to the environment. However, recent core heat-up and degradation assessments of high-pressure accident sequences suggest that upper vessel or hot leg failure is likely before core slump, and, thus, the conditions for pressurized melt ejection may not occur.
The longer-term behavior of iodine in containment is important for containment management following a severe accident. Comprehensive computer models have been developed which account mechanistically for many of the important phenomena affecting iodine volatility in containment. Both analyses and experiments indicate that organic iodides would likely dominate the airborne iodine inventory on time scales of the order of a day or so following an accident. Owing to the complexity of the chemistry occurring in containment water (i.e., due to catalytic effects, radiation, impurities, and the variety of organic species present), it may be difficult to generalize long-term iodine behavior to all situations. However, one important result, that is becoming firmly established, is that iodine volatility can be effectively controlled under all relevant conditions if the pH is kept high.
PART 1

REPORT ON THE

TECHNICAL STATUS OF THE SOURCE TERM

(Last Revision September 1986)
This part of the report was typed at the Electric Power Research Institute (EPRI), Palo Alto, California, USA
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Section 1
TECHNICAL ANNEXES

1.0 INTRODUCTION

At its November 1984 meeting, the CSNI decided, following a proposal made initially by its Subcommittee on Licensing, to set up a Special Task Force on Source Terms, whose main task would be to intercompare the various source term studies published in late 1984/early 1985 with a view to:

(i) identifying areas where source term information appears to be sufficient;
(ii) identifying areas where current studies disagree, and why;
(iii) identifying generic source term issues that could be applied to most LWRs;
(iv) identifying barriers to applying source term information to different LWR plants.

The Task Force was also asked to draw the attention of the Subcommittee to areas where the new technical understanding resulting from the review of current source term studies may have direct and broad implications for nuclear safety and regulatory issues.

The following source term studies were considered by the Task Force:

- The Report of the American Nuclear Society Special Committee on Source Terms;
- The Report of the American Physical Society Study Group on Radionuclide Release from Severe Accidents at Nuclear Power Plants;
- The technical Summary Report of the Industry Degraded Core Rulemaking Program Study on Nuclear Power Plant Response to Severe Accidents;
- Work performed by and for the USNRC, including BMI-2104, "Radionuclide Release Under Specific LWR Accident Conditions";
- Studies performed by the Stone & Webster Engineering Corporation;
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- Studies performed by the New York Power Authority;
- Studies performed by the Electric Power Research Institute;
- Work performed in Canada, France, the Federal Republic of Germany, Italy, Japan, Sweden, and the United Kingdom.

Members of the Task Force wish to express their gratitude to the various organizations concerned, for their support throughout the study and for the large amount of documentation made available.

Finally, the phrase "Source Terms" has been used throughout the literature with a variety of meanings. In the past, for example, the amount of material released from the primary circuit to the containment and the fractional release of the core inventory to the environment have both been called the source term. In this Report the Task Force has used the definition adopted by the CSNI Senior Group of Experts on Severe Accidents. For completeness, the Senior Group has defined "the Source Term" as the quantity of radioactive material which might be released in a nuclear accident: its physical and chemical form and the other quantities needed to completely specify its dispersion in the environment (e.g., energy in the plume, height of release, duration of release, etc.). Of course, the probability of the various accident scenarios must be considered in parallel with the source term.

The documentation arising from the Task Force's activities is divided into three parts:

I Executive Summary
II Technical Summary
III Technical Annexes

The Executive Summary and Technical Summary were published in March 1986. The conclusions presented in these summaries were based on information discussed during Task Force meetings.

During these discussions, the Technical Annexes were used as one of the inputs to Parts I and II, but the final conclusions were based on a broader discussion of the various source term issues.

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Finally, we wish to stress that the Task Force completed its work in October 1985. Any information available after this date was not considered. Since source term technology is changing rapidly, it is highly likely that some new information that could have significant impact on the source term was not available for inclusion. Nevertheless, we believe the Technical Annexes are a useful introduction to the key issues affecting the source term.
Section 2

TABULAR PRESENTATION OF THE RESULTS OF RECENT SOURCE TERM STUDIES

2.1 INTRODUCTION TO THE TABLES

2.1.1 Purpose of the Tables

The purpose of these tables is to give a summary of the studies performed by different organizations and their main results for typical severe accident sequences leading to containment failure for both PWR and BWR plants.

Analysis of differences among plants with respect to specific design of ESF's was not in the scope of this task. The information presented here is to orient the reader with regard to the general range of source terms as they are now calculated. The reader is, however, cautioned to read the original documents for an exact description of the accident scenario.

2.1.2 Introduction and Clarification Notes

The summary tables presented in this section are related to severe accident sequences that could entail significant release of fission products to the environment.

Data reported in the tables are for typical severe accident sequences which have been evaluated in recent studies performed by different organizations using mechanistic models. If a PRA is available for the concerned plant, the tables report also PRA data for the above mentioned sequences.

The main source for the selection of plants and sequences was the ANS "Report of the Special Committee on Source Terms" (1).

In addition, other available recent work was considered. In some cases, despite original study results, recalculation showed that some considered sequences lead to no containment failure, and sometimes also that core-melting does not occur.
The investigated sequences for the considered plants have been arranged in two kinds of tables. These are:

"Plant oriented" tables (table 2.1 for PWRs and table 2.3 for BWRs) showing the following information:

- Plant name and its assessed total core-melt frequency (events/year)
- Sequence indicator
- Study name
- Data containing:
  - Release & core melt frequencies of the sequence (event/year)
  - Containment failure mode and its probability
  - Timing for core-melt start and containment failure
  - Fractions of environment releases of I, Cs, Te
- Notes

"Sequence oriented" tables (table 2.2 for PWRs and table 2.4 for BWRs) in which the order of columns (1) and (2) of the previous tables has been inverted and only the iodine release fraction is quoted.

The symbols used to identify the sequences in terms of initiating event, failed functions and containment failure modes are mainly derived from WASH-1400. When there is a deviation a specific note has been added to the tables.

Fission products released to the environment are simply indicated as I, Cs, and Te. Since various studies identify the releases in different ways we have used the following procedures:

- releases quoted as I-Br or CsI: I
- releases quoted as Cs-Rb or CsOH: Cs
- releases quoted as Te or TeSb: Te
Discrepancies, if any, among the different frequencies (core melt, release and containment failure mode frequencies) for a sequence within the same study are mainly due to the rounding off of some of numbers in the original study.

Finally, although the following tables present interesting information, it is important to state and understand the assumptions when source term data are used. For example, the cesium source terms for the Sequoyah TMLB* sequence may vary from \(4.5 \times 10^{-4}\) to 0.11. The higher value comes from RSSMAP and the lower value from the Battelle Columbus study. The differences are due to methodology differences and sequence differences. In general, the data generated in the later studies should be given the greater credance.

2.2 TABLE OF ACCIDENT SEQUENCES

<table>
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<th>KEY TO PWR ACCIDENT SEQUENCE SYMBOLS</th>
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<td>A - Intermediate to large LOCA.</td>
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<td>B - Failure of electric power to ESFs.</td>
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<tr>
<td>B' - Failure to recover either on-site or off-site electric power within about one to three hours following an initiating transient which is a loss of off-site AC power.</td>
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<td>C - Failure of the containment spray injection system.</td>
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<td>D - Failure of the emergency core cooling injection system.</td>
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<tr>
<td>F - Failure of the containment spray recirculation system.</td>
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<tr>
<td>G - Failure of the containment heat removal system.</td>
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<tr>
<td>H - Failure of the emergency core cooling recirculation system.</td>
</tr>
<tr>
<td>K - Failure of the reactor protection system.</td>
</tr>
<tr>
<td>L - Failure of the secondary system steam relief valves and the auxiliary feedwater system.</td>
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<tr>
<td>M - Failure of the secondary system steam relief valves and the power conversion system.</td>
</tr>
<tr>
<td>Q - Failure of the primary system safety relief valves to reclose after opening.</td>
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<tr>
<td>R - Massive rupture of the reactor vessel.</td>
</tr>
<tr>
<td>S^1 - A small LOCA with an equivalent diameter of about 2 to 6 inches.</td>
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<tr>
<td>S^2 - A small-small LOCA with an equivalent diameter of about 1/2 to 2 inches.</td>
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KEY TO PWR ACCIDENT SEQUENCE SYMBOLS

T - Transient event.

V - LPCI check valve failure.

α - Containment rupture due to a reactor vessel steam explosion.

β - Containment failure resulting from inadequate isolation of containment openings and penetrations.

γ - Containment failure due to hydrogen burning.

δ - Containment failure due to overpressure.

ε - Containment vessel melt-through.
KEY TO BWR ACCIDENT SEQUENCE SYMBOLS

A - Rupture of reactor coolant boundary with an equivalent diameter of greater than six inches.
B - Failure of electric power to ESFs.
C - Failure of the reactor protection system.
D - Failure of vapor suppression.
E - Failure of the emergency core cooling injection.
F - Failure of the emergency core cooling functionality.
G - Failure of containment isolation to limit leakage to less than 100 volume percent per day.
H - Failure of core spray recirculation system.
I - Failure of low pressure recirculation system.
J - Failure of high pressure service water system.
M - Failure of safety/relief valves to open.
N* - Failure to reset RPS.
P - Failure of safety/relief valves to reclose after opening.
Q - Failure of normal feedwater system to provide core makeup water.
S₁ - Small pipe break with an equivalent diameter of about 2 to 6 inches.
S₂ - Small pipe break with an equivalent diameter of about 1/2 to 2 inches.
T - Transient event.
U - Failure of HPCI to RCIC to provide core makeup water.
V - Failure of low pressure ECCS to provide core makeup water.
W - Failure to remove residual core heat.
α - Containment failure due to steam explosion in vessel.
β - Containment failure due to steam explosion in containment.
γ - Containment failure due to overpressure--release through reactor building.
γ' - Containment failure due to overpressure--release direct to atmosphere.
δ - Containment isolation failure in drywell.
ε - Containment isolation failure in wetwell.

*Added symbol
KEY TO BWR ACCIDENT SEQUENCE SYMBOLS

ξ - Containment leakage greater than 2400 volume percent per day.
η - Reactor building isolation failure.
θ - Standby gas treatment system failure.

2.3 SOURCE TERM RESULTS

See tables 2.1 through 2.4.

2.4 REFERENCES

2. IDCOR "Nuclear Power Plant Response to Severe Accidents", November 1984 by Technology for Energy Corp.
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Revised 3/11/86

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**INDIAN POINT**

- **PLANT**: PRA
- **EVENTS**: Multiple Events
- **ENVIRONMENT**: Multiple Releases
- **RELEASES**: Multiple Types
- **NOMENCLATURE**: Various Codes (EPR, ROC, ELA)

**TABLE 7.12** Indicates FI-4 as essentially equivalent to AS-1 and FI-4 to AS - 0

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**Notes:**
- The time for cont. fail. indicates the beginning and ending of meltthrough.
- "Assumed" denotes uncertainty in the data.

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Revised 3/11/86
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CALCULATIONS TERMINATED AT 12 h. NORMAL LOADS CONSIDERED (EQ. TO 0.05 SQ. INCH) |||
Table 2.3 PWR Source Term Data - Sequence Oriented (Continued)

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| 4AF | ZION | PRA | 9.8E-10 | 9.8E-6 | 2E-4 | 1.5 | 0.7 | ALFC plant damage state 0.8 ft, Zion PRA |
| INDOR | MC7 | negl. |

| 22B | ZION | PRA | 5.8E-6 | 5.8E-6 | ~1 | 10 | 0.7 | The sequence frequency refers to Basis-10 Exponential event in Zion PRA |
| INDOR | 6E-6 | 6E-6 | 3 | 32 | 3E-3 | probability assumed equal to 1.1, assumed containment breach of 0.7 ft |
| INDOR | 3E-3 | 8 | 3 | 1E-2 | Assumed Existing Opening of 0.8 ft |

| 22C | SUARY | WAH | 2E-4 | 1400 | 2E-4 | ~1 | 4.8 | 4 | 0.2 |
| EPRI | 5 | NS | MC7 | No risk significant sequence |

| 52CD | FRENCH | CEA | ERCO | MC7 | 4E-6 |
| CEA | 8 | 2E-2 | Assumed Existing Opening of 0.8 ft |

| 52D | ZION | PRA | 7.4E-10 | 7.4E-6 | 2E-4 | 2.5 | 0.7 | SIFC plant damage state 0.8 ft, Zion PRA |
| BNI | 5 | 2.5 | 2.5E-8 | |

| SEQUIMAP | BERMAP | 6.3E-6 | 6.3E-6 | ~1 | | | Releases are not quoted but |
| ZEODOR | ZEODOR | 5 | 1.6 | MC7 | |

| BUERY | WAH | 2E-8 | 6E-8 | 1E-2 | 0.2 |

RCV/k11/W30-3378c
Revised 3/11/86
### Table 2.3 PWR Source Term Data - Sequence Oriented (Continued)

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The three figures for the release are referred to three different paths.

- Direct release via larry safeguard buildings.
- Assumed containment breach of 2 ft².
- Break submerged by 3 ft of water from BVT.
- Assumed containment breach of 0.02 ft².
- Break submerged by water.
- Assumed containment breach of 0.2 ft².
Table 2.3 PWR Source Term Data - Sequence Oriented (Continued)

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Revised 3/11/86
Table 2.4 BWR Source Term Data - Sequence Oriented

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|          | GULF  | BMI   | —        | .46      | .9   | .3E-2   |     | With stuck dumbbell bypass 
Bypass flow about 10 times larger |
| TPOW     | GRAND | PRA   | 5.3E-6   | .28      | 20   | 20       | .57  |       | |
|          | GULF  | BMI   | —        | .28      | 20   | 20       | .57  |       | |
| TPOX     | GRAND | PRA   | 4.5E-7   | .28      | 20   | 20       | .57  |       | |

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Table 2.4 BWR Source Term Data - Sequence Oriented (Continued)

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Section 3

IN-CORE PHENOMENA

3.1 FISSION PRODUCT INVENTORY

The first information that is needed for the estimation of source term is a
detailed inventory of the radioactive isotopes in the core. A number of computer
programs, such as ORIGEN 2, FISPIN, are available to calculate fission product
inventories for various core operating histories (1,2).

Inventory predictions of ORIGEN 2 for example have been compared with the
inventories of actual fuel pins taken from operating reactors with reasonably well
characterized operating histories. The assumption is generally made that the
calculation for most of the fission-product inventories will be accurate if the
calculated most of the fission products buildup and depletion are correct. This
is because the inventory of most of the fission products is heavily dependent on
the fission yields which are relatively well known and do not vary substantially
from case to case. For this reason, validation efforts have concentrated on the
much more complex actinide region. Differences between ORIGEN 2 calculations and
experimental results concerning uranium, plutonium, americium, and curium are
typically within ±10% with the following exceptions: plutonium-238 is consist-
ently underpredicted by 4% to 12%, and, based on a small number of samples,
curium-245 is unpredicted by 20%, whereas plutonium-241 is overpredicted by 1% to
14% (3,4). Limited comparisons of fission product isotopics for krypton and xenon
shows relatively good agreement, with the exception of krypton-85, which is
overpredicted by ORIGEN 2 by about 30% (3,4). However, some experimental uncert-
ainties make the relevance of the krypton-85 results uncertain. The concentration
of major heat producing fission products has been indirectly verified by
comparison of ORIGEN 2 decay heat predictions. Comparison with the ANS decay heat
standard showed good agreement (±2%) for decay times between 10 sec and
30 years (4). Comparison with experimentally determined decay heat measurements
has typically agreed with ORIGEN 2 calculations to within ±5%, with the principal
uncertainty being the burnup of the spent fuel used in the experiments (5).
The FISSION computer program and its libraries have been validated against decay heat measurements, from which it can be concluded that calculation is within 8% of experiment for uranium fissioning over a wide range of cooling times (1-10^4 secs). The comparisons with fission product density data measured in Obrigheim reactor fuel, covering isotopes of Kr, Xe, Cs, Nd and Eu, showed differences within experimental error of 1σ = 7.5%, with the exception of xenon-131/134 where 2σ differences and cesium-134 where 4σ differences occurred. The comparisons with actinide density data, covering isotopes of U, Pu, Am and Cm, showed differences within experimental error, with the exception of plutonium-239 (3σ differences) and plutonium-241 (3σ differences) (6).

The prediction of the core inventory of radioisotopes is not associated with a great deal of uncertainty. Although not all computer programs predict the same numerical value for each radioisotope, it is the least uncertain portion of reactor accident analyses and is generally regarded to be sufficiently accurate for the source term determination.

3.2 FISSION PRODUCT/AEROSOL RELEASE FROM CORES

3.2.1 Technical Background

3.2.1.1 Chemical Forms and Interactions. The release of fission products from the core under accident conditions depends on their chemical and physical properties. The knowledge about the chemical states of fission products in the fuel pellets during normal operation can be summarized as follows:

- **Noble gases** (Kr-Xe) do not interact chemically with the fuel, structural materials, or other fission products.

- **Halogens** (I, Br): The most stable of the compounds, from a thermodynamic point of view, would be the alkali halides, mainly CsI.

- **Alkali metals** (Rb, Cs) in their elemental form are reactive and form compounds with most other elements present in the fuel. The most important species formed in these reactions are the alkali metal halides, compounds with uranates and also Cs2MoO4.

- **Chalcogens** (Te, Se) are expected to be present mainly in elemental form in LWR-fuel. The formation of tellurium-zirconium alloy on the clad inner wall has been observed experimentally.

- **Noble metals** (Ru, Mo, Rh, Pd, Tc) are strong alloy formers and serve to dissolve other metals. The chemical forms of the noble metals depend very much on the stoichiometric conditions in the fuel.

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- Alkaline earths (Sr, Ba): The simple oxides and the molybdates are regarded to be the most stable of the alkaline earth compounds in the fuel.

- Rare earths and refractory oxides (Ce, Sm, Pm, Pr, Nd, La, Y, Zr, Nd): The lanthanides and other refractory fission products are known to form rather stable oxides and would be expected to exist in solid solution in UO₂.

The knowledge about the chemical states of the remaining fission products in the fuel is rather limited.

The chemical status of the fission products after release from the fuel can be changed as the core heats up and finally melts in the course of an accident. The most important factors in this regard are (a) the composition of the steam/hydrogen mixture through the core, (b) reactions of fission products with the Zircaloy cladding material, (c) concentration of the fission products, and (d) perhaps reactions with control rod materials, especially B₄C in BWRs. The ratio of steam (an oxidant) to hydrogen (a reductant) sets the effective oxidation potential in the gas phase in the course of a meltdown accident which in turn can influence the chemical form of elements released and the degree of zirconium oxidation. A description of some of the different chemical reactions during core heatup and melting can be found, for example, in the ANS or RAMA-reports (7,8).

Chemical forms of material escaping from a degraded core can be predicted from thermodynamic calculations. These calculations are idealized and have the limitation of assuming equilibrium conditions, but they can at least indicate the dominant species released.

It is generally agreed that the dominant chemical species of iodine released from a degraded LWR core is CsI. When the cesium uranate in the fuel is exposed to steam upon release from the fuel, Cs₂OH would be formed and would be the principal cesium compound escaping from the core. The detailed knowledge of tellurium chemistry under the conditions expected in reactor accidents is limited. It is currently expected that released tellurium will mainly occur as elementary tellurium or hydrogen telluride. Under highly oxidizing conditions it was found experimentally that the chemical form of the tellurium, which was produced in the SASCHA-facility and had condensed onto aerosol particles was tellurium dioxide (9). There is some uncertainty about the chemical form of the released ruthenium related to the potential for oxidizing the ruthenium in the fuel, but it is believed that the potential for this during in-vessel melting should be quite

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small. A significant release of barium and strontium has been observed when there is little oxidation of the Zircaloy cladding. It is believed that the highly reactive zirconium is able to reduce BaO and SrO thereby converting them into the more volatile elemental form (10).

It is pointed out that the core is not at a uniform temperature and that releases on different nuclides are not isolated in time but may occur simultaneously in different regions of the core. This may potentially affect chemical speciation if the different nuclides are all airborne at the same time.

3.2.1.2 Escape of Materials from a Degraded Core. The different models and codes for estimating the release of fission products and structural materials from the core during heatup and melting are described in detail for example in the ANS or RAMA reports (7,8).

For volatile fission products such as Kr, Xe, I, and Cs, in accidents where core melting is involved, essentially all models agree that nearly all of these elements are emitted prior to core slumping. The uncertainty in estimating the release of these elements in core melting accidents is primarily associated with the uncertainty in evaluating the melt progression (temperature distribution, surface-to-volume ratios, exposure time). It should be recalled that the melting temperature can depend upon conditions. UO₂ melts at 3150°K while the Z, U, O eutectic melts at 2150°K.

In those cases, however, in which fuel overheating, but not melting, would take place, the choice of the escape model would be important because the differences in escape predictions between models tend to be greater at lower temperatures. These cases, however, should not have great risk significance. For the less volatile fission products such as Te, Se, Ba, Sr, Mo, Rh, Pd, Tc, and Ru, there are significant differences in the data from various sources. For Te, differences in release shown in different tests have been explained by a reaction of Te with metallic Zircaloy to form nonvolatile tellurides. The Te is then released on subsequent oxidation of the Zircaloy. The escape of Ba and Sr, however, shows opposite behaviour (10). In the case of little or no oxidation, the Zr-metal is able to reduce BaO and SrO by converting them into the more volatile elemental forms. In the case of high Zr oxidation, the Ba/Sr will stay in the less volatile oxide forms. This shows that the steam supply as well as the degree of Zircaloy oxidation must be considered as essential parameters when the fission product release is to be calculated for different types of core melt scenarios (12). The
potential for ruthenium oxidation (and enhanced release) does not appear to be high, although there is a lack of high pressure release data under oxidizing conditions.

The nonvolatile fission and activation products (such as Ca, La, etc.) may be nearly totally retained in the melt, at least until the time of vessel failure. The experimental release data for these elements are rather limited, so that the determination of escape fractions is relatively uncertain, but only at a very low level.

It should be noted that the uncertainty in estimating release of the less volatile and nonvolatile fission products depends to a great deal on the uncertainty in evaluating the temperature history and distribution in the core.

In addition to release of fission products from fuel, other core materials present in cladding, structural components, or control rods can be vaporized in an overheated core and contribute to the aerosol mass in the reactor coolant system and the containment. The volatilities of the most important nonfission products like Fe, Cd, Mn, Ag, Sn, Zr, and In, are known to a sufficient degree, but there is still uncertainty in the modeling of the behaviour of the sources of these materials in melting cores. For control rods, there are questions as to whether or not the silver will flow upon melting to cooler regions of the core, where it may pool or resolidify, or whether or not the control rod materials will be vaporized and escape in aerosol form as the temperatures increase. A relocation of these materials can cause a delay in time at which these materials may contribute to the aerosols in an accident sequence. This delay can significantly affect the behaviour of fission products in the RCS and in the containment. Recent experiments in the STEP program [13] carried out in the TREAT reactor have shown that the tin from the Zircaloy will also contribute to the released aerosol mass. The aerosol mass which is released from the melting core into the vessel for a PWR is estimated in different studies to be in the range of 400-5000 kg [10,11,15].

3.2.2 Areas Where Studies Agree and Disagree

3.2.2.1 Volatile Fission Products (Kr, Xe, I, Cs). There is a general consensus that in the case of a meltdown accident the gaseous and volatile fission products such as Kr, Xe, I, Cs are nearly totally released from those parts of the fuel which reach melting temperature. It is also agreed that the dominant chemical species of iodine released from a degraded core is CsI. Cesium is predominantly
emitted from the core as CsOH. For these volatile fission products, the knowledge of the release behaviour in the case of a meltdown accident is regarded to be sufficient.

In those cases, however, in which fuel overheating, but not melting, would take place, differences in escape predictions of different models are greater. Since the release of fission products in these sequences is believed to be lower, these sequences do not make such a large contribution to the risk and these discrepancies may not really be as important for source term studies.

3.2.2.2 Less Volatile Fission Products (Te, Ba, Sr, Ru, Mo). For the less volatile fission product, there are significant differences in the data from various sources. The chemical form and the release of fission products such as Te, Ba, Sr, Mo, and Ru are influenced by reactions with zircaloy and by the oxidation potential in the melt. The amount of Zr-oxidation and the ratio of steam to hydrogen which sets the effective oxidation potential are dependent on the accident sequences.

3.2.2.3 Nonvolatile Fission and Activation Products. The nonvolatile fission and activation products are believed to be nearly totally retained in the melt, through the time of vessel failure. The experimental release data for these elements are rather limited, so that the determination of escape fractions is relatively uncertain.

The low calculated releases of the "nonvolatile" fission and activation products are related to the maximum temperature calculated to be reached in the core melt process. If higher temperatures are possible, then significant in-vessel releases are conceivable.

3.2.2.4 Nonfission Products. In addition to fission products in the fuel, other core materials present in cladding structural components, or control rods can be vaporized in an overheated core and contribute to the aerosol mass in the reactor coolant system and the containment and thus will influence the behavior of fission products. The volatilities of the most important nonfission products like Sn, Fe, Mn, Cd, Ag, Zr and In are known to a sufficient degree, but there is still an uncertainty in the modeling of the behaviour of the control rods, especially silver and cadmium. Because the composition of the control rod material differs among reactors, the release of nonradioactive mass is subject to these differences. In addition, the formation and duration of a liquid pool at the
bottom of the vessel are also dependent on the plant design and on the accident sequence. For example, for reactors with no penetrations through the bottom of the vessel, the time of vessel melt-through would be expected to be delayed and the release from a molten pool at the bottom could be enhanced due to the resultant higher temperature and longer duration.

3.2.3 Conclusions

The in-vessel release of the noble gases and the volatile fission products (I, Cs) from a molten core is considered not to be associated with a great deal of uncertainty. There is also a general consensus about the predominant chemical forms of the respective volatile fission products escaping from the core but, of course, uncertainty in timing of release still exists which can impact the source term.

The in-vessel release of the less volatile fission products, however, is very much dependent on the accident sequences especially on the effective oxidation potential and on reactions with zirconium. Therefore it is not possible to derive a generalized release term from the core for these less volatile fission products without regarding the accident specific details.

The nonvolatile fission and activation products are believed to contribute a negligible amount to the in-vessel release term. The experimental release data for these elements are rather limited and this very low in-vessel release is associated with much uncertainty.

The in-vessel release of nonfission core material is dependent on both accident sequence and control rod composition. In addition, there is still uncertainty in the modeling of the behaviour of silver and cadmium. It is therefore difficult to generalize the release of these materials independent from both reactor type and accident sequence.

3.3 REFERENCES


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3.4 REACTOR COOLANT SYSTEM THERMAL-HYDRAULICS

3.4.1 Technical Background

3.4.1.1 Introduction. The objective of Section 3.4 is the assessment of RPV and RCS thermal-hydraulics and core degradation during sequences starting with intact fuel rods and proceeding to a molten core. In the initial part of such a sequence

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it is important to assess the effectiveness of the reactor protection and the emergency core cooling systems. The later part of the sequence with a degraded core influences the release of fission products. Therefore, the knowledge of thermal-hydraulics and core degradation is essential for source term release and transport modeling.

The study of thermal-hydraulics within the primary circuit of LWRs has been an important research area since the beginning of reactor safety. These studies, however, concentrated on core, component and system behavior with an undamaged core or a core with only a few rod failures, allowing some release of the gap inventory. Risk studies and the TMI-2 accident focussed attention on sequences with a highly degraded core. In addition, research programs were redirected towards the behavior of high system pressure (transients and small break loss-of-coolant accidents) due to their greater importance for risk evaluations.

The assessment of the effectiveness of emergency core cooling systems was, in the past, mainly based on conservative assumptions and conservative models. Only in the last decade has the tendency for realistic ("best-estimate") calculations grown; conservative calculations are still being performed for licensing purposes.

The regime of interest is the time when fission products may be released and transported; it includes the later times, during a postulated severe accident, when the release of the fission products may be due to revolatilization of the fission products deposited earlier on the RCS surfaces.

The thermal hydraulic parameters of interest, as a function of space and time, during the course of the accident are: (1) temperatures; (2) pressures; (3) gas compositions; (4) gas flow rates; (5) temperature gradients and (6) material compositions and locations. In addition, the physical state (or integrity) of the RCS and its insulation which determine, and are affected by, the history of the thermal conditions, play a crucial part.

Fission product release and transport in the RCS are affected by certain phenomena (see sections 3.2 and 3.7) whose propensity is determined by the prevalent thermal hydraulic conditions. These include e.g.,

- Zircaloy oxidation (affects core temperatures, fission products, and H₂ release);

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- Resuspension (affects transport after vessel melt-through);
- Chemical reactions of fission products with structures (affects revolatilization);
- Aerosolization and deposition (affects retention in RPV);
- Fission product revolatilization (affects retention in RCS).

In addition, it is important to know if any part of the RCS (in the path of the fission products) contains water, which will scrub the fission products released in the RCS.

3.4.1.2 What is Known. To better cope with in-vessel thermal-hydraulics and core degradation three regimes are introduced:

(1) Core damage stays within licensing limits (e.g., peak cladding temperatures below 1200°C)

(2) The core is highly damaged and partly molten. The coolability, however, can be restored.

(3) The melt progression continues and progresses beyond the vessel.

The boundary between categories (2) and (3) is not well defined; the CSNI-Senior Group of Experts on Severe Accidents is studying whether a defined definition can be found.

Category I. The efforts in the past were concentrated on the assessment of the effectiveness of the ECCS. Therefore, system behavior related to these sequences can be predicted with comparative success. Some special phenomena, e.g., three-dimensional reflood behavior or stratified flows however, are not well predicted.

There is hardly any fission product release during this regime of interest, however, what is important here is that the presence and location of water in the RCS be known accurately; when the thermal-hydraulic and geometric conditions make it possible. There is very little data on retention of water in the RCS during the early part of a postulated accident and the current codes have not been validated.

Category II. With increasing core heat-up, oxidation of zircaloy occurs, followed by liquefaction, freezing of fuel in lower parts of the core,
slumping of core and control rod material and formation of local blockages. Later core melting may progress further.

Volatile fission products are released during the early part of the second regime of interest. The fuel heat-up following Zircaloy oxidation, releases iodine (I), cesium (Cs), and tellurium (Te); which after interaction with the steam environment occur predominantly as CsI, CsOH and Te. Prediction of the thermal hydraulic conditions in the core and the RCS during the heat-up phase are perhaps the most detailed part of the codes MARCH (1), MAAP (2), MELCOR (3), SCDAP (4), and MELPROG (5). The effects of natural convection flows (6) on the core heat-up, as predicted by the CORMLT code (7), for the PWR accident scenarios, have recently been recognized and modifications have been made in the MAAP code to incorporate a simple model of natural convection. Experiments (6) have shown that the effects of the natural convection flows on the course of the accidents may be large. In particular, it appears likely (6) that during the core heat-up, temperatures reached in parts of the primary system may lead to a local failure, much before the vessel melt-through becomes likely. This will change the course of the relatively high probability accident TMLB and possibly that of S2D. This is a result which may lead to lower source terms since a vessel bottom melt-through at high pressure has been hypothesized to lead to containment atmosphere heating and containment failure. Furthermore, these high temperatures imply that the volatile fission products released during these accidents may not be able to deposit in the PWR RCS.

Much of the discussion in the above paragraph is based on very recent information and is subject to revision as new data and analyses become available. There is also no background currently available on the effects of natural convection flows in BWR accident scenarios. The later part of this category of core damage involves clad-fuel liquefaction, candling, blockage formation, core collapse, debris bed formation and melt progression to the point of attack on the core support plate. During this part of the accident, very high temperatures (UO₂ melting) may be reached in some parts of the core and a fraction of the relatively nonvolatile fission products (e.g., La, Sr, Ba) in the core may be released in the vessel. This may lead to a lower source term since fission products released within the RPV will not be released during the ex-vessel progression of the accident. Mechanistic, analytical treatments of this part of accident progression have been incorporated in the CORMLT code (7) and the MELPROG code (5); however, there
are no data available for benchmarking the codes, except that available from TMI-2 examinations.

The importance of this phase of the accident is related primarily to:

- release of La, Sr and Ba in the vessel;
- release of hydrogen;
- loadings on the core support plate; and
- configuration of the core melt and of the interaction of molten corium with the water in the lower head.

The early and later parts of this core damage category involve the interaction of the heating of the RCS structures, due to the deposited fission products, with the thermal hydraulics and aerosol transport. This has recently been recognized (8) and there are indications that revolatilization of the fission products from the RCS surfaces may occur much more readily than suspected earlier. Coupling of fission product heating, aerosol transport and thermal hydraulics has been incorporated in the MLT code (8) and the MAAP code, and some estimates have been made. However, detailed coupling between thermal hydraulics and aerosol transport will be achieved when the codes CORMLT, PSAAC and RAFT have been coupled, and the current estimates may be revised as new analyses and data become available. In this context, the amount of heat rejected to the containment atmosphere, through the insulation, which may be degrading with temperature rise, becomes very important in determining the amount of fission products revolatilized.

**Category III.** Depending on the sequence, progressive core damage may result in a melt pool and/or a debris bed. In case the accident is not terminated, part of the melt or debris will slump or drop into the lower plenum, depending on the failure mode of the lower core support structure.

Recovery of the accident may be possible during this phase if the capability to inject water into the RPV is restored.

A rapid melt/water interaction ("steam explosion") may occur if molten fuel is poured into water, resulting in fast energy transfer from fuel to water, which may threaten RPV integrity. The physical phenomena related to rapid melt/water interaction have been studied in different experimental facilities with different materials by different organizations and an extensive data
base has been collected. It is in the nature of these experiments that only small amounts of melt have been used. Substantial theoretical work has also been performed. Based on the experimental and theoretical work—taking into account engineering judgment in extrapolating beyond available experimental data—it has recently been agreed by an expert group (9) with broad consensus that steam explosions of sufficient energetics which could fail the containment have a very low probability of occurrence. Some experts even feel that this is physically impossible. It is, however, generally agreed that small rapid melt/water interactions may occur, which could be a factor in disposal of fuel or debris within the reactor coolant system.

The rapid generation of large amounts of steam that may occur due to melt-water reaction in the vessel lower head can be a source for the generation of hydrogen from the fuel clad that is still within the confines of the original core boundaries. Estimates of hydrogen generation rates are generally prone to uncertainties, since the geometry is undefined, blockages may restrict access of steam to Zircaloy clad and there may be alternate paths (e.g., downcomers) available for steam to flow out of the vessel.

A third regime of the in-vessel steam generation may disperse the molten corium in the RVP and part of the RCS. The deposition of the material in pipe bends, elbows, plena, etc., may pose thermal loads on the RCS components. The likelihood of large amounts of material displaced and of large thermal loads, however, seems remote.

3.4.2 Principal Areas of Agreement/Disagreement Between Various Studies

Recently, the NRC-sponsored studies, BMI-2104 and NUREG-0956 (10), and the IDCOR-sponsored study (12) have analyzed consequences for certain accident sequences in some reference plants. This has afforded comparisons (13) of methods used and results obtained for a broad range of application. In addition to these studies, the analyses performed by EPRI (14) for the Surry PWR and by NYP A (8.15), for the Indian Point 3 PWR and the Fitzpatrick BWR Mark I, have also obtained results for the source term for a number of risk-dominant sequences.

All of these studies had to calculate thermal hydraulic conditions in the RCS of the plants analyzed. The BMI-2104 study used the combination of MARCH and MERGE codes and the IDCOR study used the MAAP code. Results from these codes have been compared and the differences between them have been noted. In terms of the

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prediction of thermal hydraulic conditions, e.g., system temperatures, flows and Zircaloy oxidation, large differences have been observed for the hydrogen production magnitude. Both MARCH and MAAP codes employed once-through forced convection flow modeling in their analyses. The results of these codes do agree reasonably well for temperature histories within the core and for the volatile fission product generation rates. The MARCH and MAAP codes did not treat thermal hydraulics and fission product transport in a coupled manner; although recent modeling in MAAP has coupled thermal hydraulics with aerosol transport. The latter is represented through correlations.

The EPRI study on Surry plant consequences uses the MARCH 2.0 code for prediction of the in-vessel thermal hydraulic conditions, and therefore, for the in-vessel phase of the accident, has assessments similar to those reported in BMI-2104. The NYP A MCT code, however, feeds the thermal hydraulics, as calculated by the MARCH/MERGE codes, into the fission product deposition and transport calculated by the TRAP-MELT code. The fission product heat-up of the PWR primary system leads to relatively early release of fission products from the RCS to the containment. This result is different from that produced by the MAAP and MARCH codes, in which the PWR RCS retains a large fraction of fission products and revaporation occurs much later in time.

Resolution of the differences observed between various studies will have to wait for a detailed examination of the methods employed in each of these studies. Such a comparison may be more fruitful when all major phenomena (e.g., natural convection, chemical reactions, clad and fuel relocation blockage formation, revolatilization and the interactions between the phenomena (e.g., affect of fission product self-heating on the natural convection flow patterns)) occurring during the in-vessel progression of the accident are modeled in the codes. The process of improving the description of the accident progression has to include validation of the codes against data from tests conducted on ACRR, PBF, LFOT TREAT and MARVIKEN reactors and the TMI-2 accident.

3.4.3 Principal Areas of Sufficient/Insufficient Information

The information required to calculate the early part of a PWR core heat-up (which includes the effects of thermal convection flows) is currently being assembled. This includes the experiments performed at Westinghouse (16) the analyses performed at ANL with the COMMIIX (17) code and the CORMLT code, all of which are being sponsored by EPRI; and the analysis performed with the MELPROG code.
sponsored by the USNRC. Similar information about the early part of the heat-up of a BWR core is not yet at hand.

The release of volatile fission products can be predicted reasonably well since a significant part of the volatile release occurs during the early part of the core heat-up, when the core is relatively intact and the fuel and clad temperature distributions are reasonably well known. The disposition of the fission products in the RCS may not be as well-predicted currently.

The models for clad-fuel melting, core blockage formation, core slumping and melt progression during the second phase of the accident are based largely on some experiments performed in Germany and at the PBF facility. The German CORA (18) experiments and the ORNL 10 Kg core melt tests (18), coupled with the analytical results from the accident progression codes e.g., CORMLT, MELPROG and SCDAP should provide additional information. It is also very important to learn, as much as possible, about core melt progression from the TMI-2 examinations.

With respect to rapid melt/water-interactions, most experts agree that additional experiments will not substantially alter the assessment about the steam explosion threat to containment integrity being of very low probability and should not be a factor in assessment of the source term.

It is perhaps fruitful to reiterate the outputs required from the phenomenology and the various codes treating the RCS thermal hydraulics during the in-vessel progression of a severe accident. These are:

- Magnitude of hydrogen produced during core melt progression and released to containment;
- Magnitude and spatial distribution of the temperatures reached in the melt;
- Magnitude of the less volatile and nonvolatile fission products released in-vessel;
- The magnitude of the thermal and mass loading on the core support plate and the vessel;
- The magnitude (and the mode of interaction) of the corium melt with the water in the bottom head of the vessel and the consequent steam and hydrogen production rate;
- The mode of vessel failure;
The rate of the debris mass and energy exiting the reactor vessel into the containment and its temperature.

The uncertainties in prediction of some of the above parameters are larger than in the others, and more analysis and code validation is required before these uncertainties can be reduced.

3.4.4 Generic Applicability to LWR Plants

Most of the information base developed for the analyses of the in-vessel accident progression for PWR and BWR reference plants can be applied to other plants as well. Generally, the core designs are similar; the RCS geometries, the emergency cooling and reactor safeguard systems are not too different from plant to plant. It is perhaps important to focus the predictive methods to PWRs and BWRs separately, so that it becomes possible to represent the separate character of their core and system designs in greater detail.

3.4.5 Conclusions and Recommendations

Much progress has been made during the recent years in understanding and modeling in-vessel core degradation and thermal-hydraulics. In particular, the description of the early part of the in-vessel accident progression, during which a large part of the volatile fission products contained in the fuel may be released is reaching maturity and validation. It is expected, that the research and development currently in the pipeline, will provide the needed information to substantially increase the confidence in the predicted core and the RCS thermal hydraulic conditions which, in turn, will increase confidence in the predictions of the release from the core and the RCS of the fission product source term. In this context, it appears necessary to couple thermal hydraulics and fission product aerosol transport in the RCS of both the BWRs and PWRs. This is being performed and the results of analysis incorporating effects of coupling are being reported in the literature.

The prediction of the accident progression products (listed earlier) during core damage category 2 still requires further development of the methodology. It also needs data from some properly designed experiments and from TMI-2 examinations. The predictive methods have to be benchmarked against measured data to lend confidence in their results for prototypical accidents.
The third accident category 3, in which the vessel may melt-through needs development of methodology for its prediction. It is clear that information about the physical state of the core melt (mass, composition, temperature, pressure and driving gas volume) and about to breach in the vessel RPV is needed to assess the loads due to steam spill, direct heating, core concrete interaction on containment and the production of fission product aerosols from the core concrete interaction.

For both the second and third phases of degraded core accidents, one has to answer the question, "How complete must the information be?" Further analyses will be required before such a question can be answered rationally and a consensus reached.

3.6 REFERENCES


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3.5 RCS FISSION PRODUCT TRANSPORT AND CHEMISTRY

3.5.1 Technical Background

Fission products and other materials released from the core region during a severe accident will pass into the RCS. Their transport in and degree of escape from the RCS depend upon various interacting phenomena and system conditions. The several source term studies available at the present time (1-8) have addressed and/or incorporated these factors to a varying level of completeness. The general status of the technology is outlined below and then details are presented concerning the adequacy of the present methodology for supporting the RCS transport aspects of a severe accident source term definition process.

The analysis of fission product transport through and chemical interaction in RCS flow circuits during severe accidents has received much attention since the publication of the Reactor Safety Study (9) in 1975. Experimental work has also been done to measure critical parameters and to examine important phenomena. The
focus has been on the more volatile fission products (iodine, cesium, and tellurium) and core structural materials (silver, indium, cadmium control rod metals and some of the more volatile alloying elements in zircaloy and stainless steel). It became apparent that the behavior of source material during RCS transport is very dependent on the temperature and fluid flow information that was supplied for the accident scenario under study. This led to improvement in the analysis of RCS thermal-hydraulics—a process which is ongoing. The importance of flow path geometry led to development of separate models for PWR and BWR systems and to provision for analysis of alternate flow circuits in either design for different accident sequences. Separate theoretical assessments were performed to determine the likely chemical species for the source materials, and laboratory work was done to measure key property data such as vapor pressures and deposition rates.

The methodology that existed about two years ago formed the basis for conducting the RCS fission product behavior portions of most of the source term studies that are under consideration here. Some significant changes and alternate approaches have occurred during the completion of the studies. These aspects of the different studies are discussed below. The discussions are divided into four sections. The next section focuses on chemical species only. This is followed by a section on aerosol formation and transport, and then by a section on deposition processes. The last technical assessment section deals with special features that have more of a global impact with respect to RCS fission product behavior analysis.

3.5.2 Chemical Species and Reactions

The principal issues regarding fission product behavior in RCS flow circuits that pertain to the chemical species and the reactions they may undergo are discussed in this section. Gas phase chemistry factors as well as reactions with structural components of the RCS flow circuits are addressed.

3.5.2.1 Principal Areas of Agreement and Disagreement. Most studies (1-8) now presume that the major vapor species entering the RCS will be CsI, CsOH, and Te (if released from the core) in a steam-hydrogen (reducing environment) mixture of varying proportions. This conclusion can be supported by a combination of thermodynamic calculations and experimental observations (2,10). In addition, the studies basically assume that the above species do not change during their passage through the RCS except for the process of deposition/reaction at surfaces. Revolatilized species (in those studies which model the process) are treated as
being the same. There also seems to be a general acceptance that organic iodide formation along the RCS flow paths will be trivial. This position is likely a consequence of NUREG-0772 conclusions (11). In general, any other fission products released from the core region tend to be lumped in to the aerosol component and not treated as having a vapor state in the RCS.

The identity of the above vapor species is maintained in most but not all of the source term studies. Those analyses (1,5-8) which use a version of the TRAP-MELT code tend to follow the separate species, but the IDCOR analysis, which uses an empirical aerosol correlation, does not maintain identity during initial material transport along the RCS except in terms of the contribution of each fission product to the total aerosol mass and decay heat. This allows the IDCOR analysis to treat the revolatilization of deposited species (CsI, CsOH, Te) at later periods. The only study in which chemical reactions between the above vapor species and stainless steel surfaces in the RCS are specifically calculated is the BMI-2104 work. The BMI-2104 TRAP-MELT code uses simple physical condensation for CsI (no reactions), limited chemisorption* for CsOH vapor, and strong chemisorption for Te vapor. The APS review (2) seemed to accept the evidence for such chemisorption reactions but required more tests and confirmatory data. The other studies, including those which include treatment of revolatilization, simply model vapor deposition of CsI, CsOH, and Te as a physical condensation (evaporation in the case of revolatilization) process.

3.5.2.2 Principal Areas of Sufficient/Insufficient Information. The major forms of the volatile fission products that would enter the RCS are considered to be reasonably well defined as is the potential for argonic iodide formation. However, uncertainties exist regarding chemical interactions and their modeling in RCS fission product transport analyses. The issues of particular importance are listed below. In each case the importance of the issue to overall severe accident source term definition problem is estimated in terms of low, medium, or high impact.

1. Reactions between core materials (such as Sn and In) and gases (i.e., H₂) with tellurium are not treated. By removing hydrogen, these reactions could alter the Te transport and deposition character. This is judged to have a medium impact on source term definition efforts.

*Chemisorption as used here refers to sorption with chemical reaction between the vapor and the metal to form a compound having a lower volatility than the original vapor form.

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2. Uncertainty exists concerning the magnitudes and effects of reactions of fission product forms with control rod materials. One example is the possible reaction between CsI and boric acid (a coolant additive or B$_4$C hydrolysis product) to form more volatile iodine species. Other potential reactions include formation of cesium borate (CsOH reaction with boric acid) or formation of AgI (iodine species reactions with control rod alloy silver). This is estimated to have a medium impact on source term definition efforts.

3. The extent of formation of hydroxides of the alkaline earths and the impact on their transport and deposition behavior needs more complete evaluation. This is judged to have a medium impact on source term definition efforts.

4. There is a lack of data on the potential for species changes during revolatilization processes. A related issue is the possibility of Te-132 decay to I-132 causing delayed liberation of radiiodine from the RCS. However, both of these are judged to have a low level impact on overall source term definition efforts.

5. No analyses contain models for the calculation of fission product scrubbing or flushing by liquid water in the RCS. Some accident scenarios may involve transport through water (e.g., partially full pressurizer) or with water (e.g., delayed accumulator discharges). This also is estimated to have only a low impact on source term definition efforts.

6. The APS review (2) raised a concern about the effects of CsOH solubility in steam on fission product cesium transport. However, this is estimated to be a rather narrow issue which would have a low impact on overall source term definition efforts.

7. No analyses have included models for calculation of the effects of air entry into the RCS late in scenarios when revolatilization rates and species may be affected. This has the potential for a medium impact on source term definition efforts.

8. The effects of the high radiation field in the RCS during severe accidents on the chemical interactions that take place and on the stability of the reaction products have not been investigated sufficiently. This also has the potential for producing a medium impact on overall source term definition efforts.

3.5.2.3 Generic Applicability to LWR Plants. The several source term studies have performed RCS fission product transport calculations for various PWR and BWR plants and accident scenarios. There is nothing particularly unique with respect to chemical species or reactions among the collection of plant types. This is because materials and accident conditions are very similar. Consequently, the information is applicable to all LWR designs on an essentially equal basis.

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3.5.3 Aerosol Formation and Transport

The major factors that influence this aspect of RCS fission product transport are discussed below. The primary emphasis is on phenomena connected with the gas phase behavior of the aerosol component and not on the interactions of this component with the structures in the flow circuit.

3.5.3.1 Principal Areas of Agreement and Disagreement. There appear to be two rather fundamental areas of agreement among the various source term studies with respect to RCS aerosols. First, it is agreed that appreciable aerosol concentrations will form at or near the interface between the core and the RCS. Second, it is generally accepted that aerosol agglomeration processes will be quite active during transport such that significant particle growth will take place. The various studies tend to use the same or similar approaches to the aerosol formation problem. It is nonmechanistically assumed to occur, and a population of very small initial particles are simply specified by the analyst. This rationale is embedded in the origins of the TRAP-MELT code (1) and other related codes. However, in the IDCOR (4) approach, only the mass input rate of aerosol producing material is required.

The transport of aerosol material along the RCS flow circuit is uniformly assumed to follow the path of the steam-hydrogen mixture through the system. The studies assume the aerosols have no effect on the physical properties of the bulk fluid. All studies take agglomeration into account—some more directly than others. Most of the analyses use a version of TRAP-MELT (including RETAIN) which contains models for the more important agglomeration processes. Variations in results arise from representing the particle size range with a discrete binning model or a continuous (long-normal) function. The IDCOR method is the exception since it uses an empirical correlation which has the agglomeration influence embedded in the overall rate coefficient.

Condensation of vapors can also contribute to particle growth in TRAP-MELT and related calculations. It does not contribute much to size change. Particle formation and growth during revolatilization of deposited fission products is not treated in TRAP-MELT except possibly in the NYPA (7) and SWEC (8) analyses where again the analyst would specify the starting particle size. The IDCOR correlation can be applied to a revolatilization source, but only to derive a removal rate on the basis of a specified mass concentration of aerosol material.

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3.5.3.2 Principal Areas of Sufficient/Insufficient Information. The aerosol agglomeration processes that are calculated in TRAP-MELT and other similar codes are considered to be relatively well substantiated by a range of experimental programs that have been performed at various laboratories. This aspect of the technology is probably adequately defined. However, the process of aerosol formation is considered to be less well established. This and other areas where insufficient information may be important are outlined below. The estimated impact of the various uncertainties or limitations to the overall source term problem are indicated here as was done for the chemical species and reaction issues in section 3.5.2.

1. The need for incorporating primary particle formation in RCS aerosol behavior models is not fully assessed at the present time. Certain studies where this has been done (2,12) predict that under some conditions the subsequent growth of particles occurs not by agglomeration but by the process of heterogeneous condensation. This can affect growth rates and particle compositions. It is judged that uncertainty over the fundamentals of aerosol formation and growth may have a medium impact on overall source term definition efforts.

2. The problem of aerosol formation, transport, and growth during revolatilization processes has not been addressed satisfactorily. In particular, the particular sizes that would characterize the event are not known. This is judged to have a medium impact on overall source term definition efforts.

3. The IDCOR aerosol correlation needs to be evaluated and verified for application to RCS flow conditions. Since this represents only one analytical approach it will probably have a low impact on the overall source term definition effort.

4. The hygroscopic properties of fission product salts (CsI and CsOH for example) which would be part of the aerosol material, have not been addressed in any modeling to date. Similarly, hydroxide forms of core structural materials are usually not considered. In both cases, these could be important determinants of aerosol properties and growth rates at high stream partial pressures. Therefore, this is judged to have a medium impact on overall source term definition efforts.

5. The effect of natural convection and circulation in and between RCS regions on the aerosol transport and particle growth predictions has not been evaluated. A similar problem--aerosol effects on fluid properties--has also not been sufficiently addressed. Each involve the interaction between thermal-hydraulics and aerosol dynamics and as such are judged to produce a medium impact on the overall source term definition effort.

6. Current analyses do not incorporate models or other means to predict the nature of suspended aerosols that may be produced by

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entainment or resuspension events/processes. Such events/processes might include rapid RCS depressurization, small scale steam explosion events, or other causes of high velocity flows. This issue is estimated to have a medium impact on overall source term definition efforts since it can result in time dependent changes in the size distribution and composition of the aerosol component.

3.5.3.3 Generic Applicability to LWR Plants. The aerosol formation and transport aspects of RCS fission product transport analyses are equally applicable to all LWR designs. This includes the points of insufficient information outlined above.

3.5.4 Fission Product Deposition Processes

The deposition processes that are modeled in the RCS fission product transport codes/procedures are instrumental in defining the degree of retention that would occur in the RCS during severe accidents. The major factors which affect this portion of the RCS analysis are discussed below.

3.5.4.1 Principal Areas of Agreement and Disagreement. There is substantial agreement regarding the mechanisms by which fission product vapors and aerosols would deposit on RCS surfaces. The more important mechanisms are considered to include vapor condensation and vapor chemisorption (see section 3.5.2) and aerosol deposition by sedimentation, thermophoresis, turbulent motion, and Brownian diffusion. Other mechanisms that could apply to aerosols in an RCS circuit are impaction and diffusiophoresis (if steam condensation can occur).

It is generally accepted that the mathematical models which exist for these processes can be used to make reliable predictions in accident analyses. Most of the models are incorporated in the TRAP-MELT code or its derivatives. Since all except the IDCOR analyses use a version of TRAP-MELT, the effectiveness of deposition for the initial passage of source material through the RCS appears generally consistent between studies. The IDCOR results for the period also tend to agree. This is because sedimentation is the process which dominates the removal rate coefficient in the IDCOR empirical correlation, and inspection of TRAP-MELT results for similar accident sequences indicate a major influence of aerosol sedimentation on these deposition predictions as well.

At later times in accident sequences, however, the results of IDCOR (4) and other analyses (i.e., NYPRA (7) and SWEC (8), which model revolatilization by decay self-heating, can be quite different from the BMI-2104 TRAP-MELT results, which

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contain no decay heating effects. Lack of agreement in deposition (RCS retention) results between studies can also be caused by differences in the thermal-hydraulic conditions that are supplied to the fission product transport code(s) by the respective mass-transfer and fluid-flow analyses.

3.5.4.2 Principal Areas of Sufficient/Insufficient Information. At the present time there is probably sufficient recognition of the various mechanisms by which vapors and aerosols can undergo deposition in RCS flow circuits. However, a variety of areas of insufficient information can be identified. Those that are considered relatively important are listed below along with a judgement regarding their impact on the overall source term definition problem.

1. The modeling of revolatilization of deposited fission products due to decay self-heating is in an elementary stage but results indicate it will have major effects on long-term RCS retention predictions. Work is underway to produce improved analyses and data. This is judged to have a high impact on the overall source term definition effort, at least until its magnitude has been more completely quantified.

2. Little, if any, effort has been applied to modeling the state and character of source material deposits. Treating the deposits as solids when they should have liquid properties may introduce significant surface loading errors. However, this will probably have only a low impact on the overall source term definition effort.

3. No analyses contain any models for re-entrainment or resuspension of deposited materials during periods of high velocity bulk gas flows. Experimental data exist which show that liquid films and aerosol deposits can be modified in such circumstances (13,14). This is estimated to have a medium impact on overall source term definition efforts.

4. The current severe accident codes do not contain models for aerosol deposition by impaction or by diffusiophoresis (Stefan flow). Under certain conditions these may represent significant removal mechanisms and thus they should be included for completeness. However, it is estimated that these processes will have a low impact on the overall source term definition effort.

3.5.4.3 Generic Applicability to LWR Plants. The RCS vapor and aerosol deposition models or correlations used in current source term analyses are generally applicable to the various LWR designs. The rather wide application of the alternate methods in the several studies to date (a variety of BWR and PWR plants and accident sequences) illustrate this point quite effectively.
3.5.5 Special Features

There are several phenomenological issues or accident scenario analyses which involve the entire RCS fission product chemistry and transport methodology that were not addressed in the several source term studies of interest. These special features are discussed below and an assessment is provided regarding their respective impacts on current and future source term definition efforts.

3.5.5.1 Coupled Thermal-Hydraulics and Fission Product Transport. In a large part of the source term analyses performed to date, the thermal-hydraulic conditions for the RCS have been computed in advance of the fission product transport and deposition calculations. The thermal-hydraulic computations often assume uni-directional or single-pass flow through the RCS. This decoupling of heat-transfer and fluid-flow phenomena from the fission product vapor and aerosol transport phenomena prevents analysis of several important complicating effects and interactions. For example, the impact of natural convection and stratified flows on the transport, deposition, and aerosol formation/growth history of the source material cannot be followed properly (15). Also, the thermal feedback effects of the multicomponent moving decay heat source on the thermal-hydraulic responses of the RCS fluid and structures are lost. The methods that have been adopted to consider revolatilization effects represent a start towards such coupled analyses. However, an integrated capability for analyzing the early as well as the long-term mass and heat transport problem is required. This code may also incorporate aerosol formation and transport models, predict particle sizes, and allow for chemical species changes that will change deposition/revolatilization potentials. Such a capability, when available, is judged to have a high impact on the effort to define source terms for LWR severe accidents.

3.5.5.2 Analyses of Steam Generator Tube Rupture Scenarios. None of the source term studies has performed analyses of fission product transport and deposition for the primary RCS and the secondary RCS in the case of a large steam generator tube rupture accident. Tube failures would result in blowdown of the primary RCS to the secondary RCS with subsequent venting to the atmosphere, thus producing a containment bypass scenario. The consequence of this scenario may be estimated with current methods by extending thermal-hydraulic calculations and fission product transport calculations to include the secondary RCS flow circuit. Realistic modeling of the components and structures along the passage should be emphasized since appreciable attenuation of the atmospheric release would be expected. On this basis it is estimated that, for reactors where SGTR accidents...
contribute significantly to risk, this issue will probably have a medium to high impact in the overall source term definition problem.

3.5.5.3 Effect of Aerosol Deposits on RCS Flow. The aerosol mass loadings indicated by some of the source term analyses (hundreds of kilograms passing into the RCS in a severe accident) suggest the possibility that heavy deposits in small piping or at restrictions (pumps, valves or plates) in larger passages could interfere with the presumed fluid flow pattern and rates for the scenario. This is an issue which is not addressed in most analyses. Limited IDCOR calculations have used a published pipe plugging correlation (16) but the implications of partial or total blockages by material deposits in terms of altered scenario conditions or RCS flow circuit variations should be examined in more detail. On an overall basis, however, it will probably have only a low impact on the source term definition problem.

3.5.5.4 Analysis of Below Core LOCAs. An accident scenario which has not been evaluated in any of the source term studies is the case of a small or medium LOCA with the break in the RCS located at the bottom of the reactor vessel (i.e., instrument line penetrations or control rod drive penetrations). This creates an unusual flow condition for the RCS thermal-hydraulic and fission product transport analyses. It may also lead to low RCS fission product retention predictions. The scenario should be examined sufficiently to determine whether the inverted geometry indeed produces conditions and predictions that are outside the envelope of existing studies. This is more of a confirmatory issue than one of obvious consequence to source term definition and thus it will probably have only a low impact on the overall definition effort.

3.5.5.5 Timing of RCS Processes and Releases. This factor is an implicit component of several of the issues discussed in this and in earlier sections of the document. The timing of fission product release from the RCS is not particularly important if containment integrity is maintained throughout an accident, but it can be quite important in sequences where containment integrity is lost. In these latter cases, releases from the RCS that occur during or after containment failure may have an appreciable effect on the atmospheric source term because containment attenuation processes are usually less effective then. RCS processes which tend to produce delayed releases, such as revolatilization, resuspension, and slow chemical or radiolytic reaction, should have the greatest effect on release timing. The extent to which these processes are modeled or not modeled in accident analyses will influence the source term uncertainty for release timing.

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sensitive sequences. Since most of the processes which generate delayed releases from the RCS were judged to have a medium impact on overall source term definition efforts, the timing factor should also have a medium impact.

3.5.6 Conclusions and Recommendations

The RCS fission product retention results obtained from the several source term calculations reviewed in this work have been summarized in table 3.5.1. Considerable variation in values is apparent. The comparison illustrates the dependence of analytical predictions on (1) the accident scenario, (2) the plant type, and (3) the methodology used. On this basis, it appears that the chances are small for establishing generic values, even by plant type (PWR versus BWR).

In the previous sections of this discussion, it was pointed out that the definitions of primary chemical species for fission product iodine and cesium were reasonably well established. Although many aspects of aerosol behavior and modeling can be assumed to be relatively well established, the applicability of much of the current work to the conditions which are expected in RCS flow circuits during severe accidents requires more attention. The problems connected with revolatilization need to be investigated and appropriate changes made in RCS fission product transport analyses. This will require the continued development of a coupled thermal-hydraulic and fission product transport code which will be able to produce realistic heat and mass transport results, and accurately define the characteristics of the fission product vapors and aerosols.

The two issues just noted were identified as having a high impact on the overall source term definition problem. This is recognized in the R&D work that is underway and some results should be available by the end of this year. Complete resolution of the two issues probably will occur before the end of next year. Experimental and analytical work is also in progress on most of the issues that were given medium impact ratings. The expected time scale for their resolution in a manner that would be sufficient for source term definition purposes is estimated at two to three years. Limited work is underway or planned on the remaining (low impact) issues discussed in previous sections. Resolution of these issues is probably not critical for definition of severe accident source terms but it is expected to be effectively complete in the next two to three years also.
Table 3.5.1

RCS RETENTION SUMMARY
(FRACTION OF CORE INVENTORY HELD IN RCS)

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(a) Values obtained from tabulation in Reference (3).
(b) These analyses incorporated revolatilization processes.
3.5.7 References


Section 4
EX-VEssel PHENOMENA

4.1 CORE/CONCRETE INTERACTIONS AND OTHER EX-VEssel SOURCES

4.1.1 Introduction

A key event that appears in virtually all dominant severe accident sequences is the failure of the reactor vessel (RV) bottom head with the resultant transfer of molten core and structural materials into the containment cavity. The radioactive components that had not been released within the reactor coolant system (RCS) prior to vessel head failure will then be available for release directly into the containment. A number of processes have been identified by which both radioactive and inert materials could be made airborne directly into the containment or could be caused to change their form there so that they effectively become a type of "new" source. These so-called ex-vessel source which have been mentioned include the following:

- core/concrete interactions.
- pressurized ejection of core materials from the RV.
- energetic thermal interactions of the core materials with water in the cavity.
- fragmentation of water droplets due to blowdown ejection from the RCS.
- boiling or flashing of water pools releasing previously collected materials.
- resuspension by hydrodynamic forces after containment failure.
- aqueous or other chemistry effects that "revolatilize" some radionuclides.
- nucleation of water droplets from supersaturated steam.
- revaporization of fission products from aerosols as a result of flammable gas combustion.
- "direct" release from the RCS after RV failure if residual fuel is left in the vessel.

4-1
revaporization release from the RCS surfaces due to self heating by radioactive decay.

This section will discuss only the first six of these.

4.1.2. Core/Concrete Interactions

4.1.2.1 Technical Background. Experimental investigations (1) of melts interacting with concrete have shown that the concrete is aggressively attacked and that sometimes copious amounts of aerosol and fission product simulants are produced (some recent BETA experiments (2), that appear to be in exception to this may not be when extrapolated to accident scales). Some of the salient features of the process that are known from the observations include:

1. The attack on the concrete causes the release of H$_2$O and CO$_2$ coming from free water and thermally decomposed hydrates and carbonates within the concrete.

2. The gaseous products sparge up freely through the melt and it has been speculated that some portion bypasses the melt. The gases can chemically react with metallic components in the melt to produce H$_2$ and CO. In addition, the sparging process extracts fission products and other materials from the melt to be released at the surface as aerosols. The aerosols produced typically have a mass-mean diameter of about 1 µm with a standard deviation of about 2 (3).

3. Observations from X-ray images of melts interacting with concrete and from segmenting of frozen melts have indicated that the gases move through the melt as bubbles that are 1-2 cm in diameter (4,5).

4. The attack on the concrete occurs primarily by melting. The molten components (mostly oxides) of the concrete are incorporated into the melt itself.

5. Molten mixtures of oxides and metals are often observed to segregate into separate layers.

6. Measurements of the rates of release of the aerosols from the melt surface have shown that the release rate depends on the type of concrete. There is an exponential dependence on the melt temperature and a nearly linear dependence on the gas evolution rates.

In the process industries and arc welding industries, there are analogous situations of high-temperature melts being sparged with reactive gases. In these analogous situations, aerosols are also observed to be released (6-8). Studies of these processes and of core/concrete experiments have led to the hypothesis that

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there are two main responsible mechanisms. These are named "vaporization" and "mechanical" suspension.

**Vaporization.** Because of their volatilities, the various components in the melt vaporize into the gas as it sparges through. These vapors cool and condense into aerosols on exiting the melt.

**Mechanical.** Gas bubbles bursting at the melt surface cause fragmentation and ejection into the air of some quantity of the liquid. Also, rising bubbles with internal circulation could entrain liquid droplets from the bubble interface as a result of Kelvin-Helmholtz instabilities there.

Much more attention has been given to modeling of the first of these, vaporization release, because it is by far the dominant process until very late in the melt/concrete interaction period when the melt temperatures are too low to sustain much vaporization.

The driving force for release by vaporization is associated with the equilibrium partial pressure of the components undergoing release. As discussed by Powers (Ref. 9), the equilibrium partial pressures for components in an ideal mixture are linearly dependent on the component concentration in the melt,

\[ P_i(\text{eq}) = K \exp \left( -\frac{\Delta G}{RT} \right) \alpha_i \]

where

\[ P_i(\text{eq}) = \text{equilibrium partial pressure of component } i \text{ above an ideal mixture} \]

\[ K = \text{temperature dependence parameter that can also depend on compositions} \]

\[ \Delta G = \text{temperature dependent parameter} \]

\[ R = \text{gas constant} \]

\[ T = \text{temperature of the mixture, and} \]

\[ \alpha_i = \text{mole fraction of component } i. \]
Thus, the thermochemical driving force, again from Powers (9), would be expected to have the following characteristics:

- it is strongly dependent on temperature.
- it depends on the speciation of the vapors in the gas phase and, consequently, on the composition of gases sparging through the melt.
- as release progresses, the driving force for vaporization weakens if the lost components are not replaced in the melt.
- as concrete ablation progresses, the driving force for vaporization of components not coming from the concrete weakens (dilution effect).
- oxidation of metallic phases in the melt by gases evolved from concrete could increase the driving force for vaporization of unoxidized materials (opposite of dilution).

Just from these apparent characteristics alone, it is clear that prediction of the release by vaporization requires a very good model of the core/concrete interaction. Such a model must provide accurate prediction of:

- melt temperature.
- gas sparging rates.
- concrete ablation rates.
- melt composition.

It is also clear that the core/concrete interaction process can be viewed for understanding purposes as having two major phenomenological aspects. These are (1) the thermal and (2) the mass transport behaviors.

The Thermal Behavior involves heat generation within the melt and transfer to the surrounding concrete and cavity atmosphere (or a pool of water if present).

The Mass Transport Behavior that results as the evolved gases sparge through the melt.

Although these two aspects are phenomenologically coupled, severe accident analyses to date have generally treated them separately with the results of the thermal analysis providing input and boundary conditions for the aerosol generation analyses. It is by no means necessary, or even desirable, to uncouple those analyses.
4.1.2.1.1 Thermal Modeling. The Reactor Safety Study (WASH-1400) utilized a forerunner of the INTER subroutine that is now in MARCH for core/concrete interaction modeling. More recent (and believed to be better) models have been developed that include WECHSL (10), CORCON (11), and the DECOMP (12) subroutine used in the MAAP analysis package of the IDCOR program. The objectives of the present review should be adequately served by comparing the modeling that makes up CORCON and DECOMP. WECHSL and CORCON take similar approaches.

Overview Description of CORCON. A disconcerting aspect of reviewing any code or model that is under development is that the review runs the risk of being out-of-date even at the instant of its publication. The CORCON code is still under development at Sandia National Laboratories. The first released version, MOD-1 (13) with some improvements was used in the BMI-2104 studies. Recently CORCON-MOD-2 (11) has been released incorporating more changes and improvements beyond BMI-2104. Hence, in this review when results from BMI-2104, for example, are compared with IDCOR--the MOD-1 (with improvements) should be borne in mind. For state-of-the-art assessments, however, MOD-2 should be the reference. The overview presented here will focus on MOD-2 unless otherwise noted.

CORCON is a stand-alone computer code that models the physical and chemical processes which may occur during the thermal interactions between molten core debris and concrete although an arbitrary geometry operation is available. The concrete is assumed (input defined) to have formed a cavity of axisymmetric geometry. A mixture of molten oxide and metallic melt is "deposited" into the cavity. The mixture composition is specified as input but should consist primarily of molten fuel, UO₂, and oxidized iron and zirconium for the oxide constituents and steel and zirconium for the metal constituents.

Oxidic phases and metallic phases are assumed to segregate as layers in CORCON. The MOD-2 version also includes a coolant layer and is structured to recognize intermediate layers between oxidic and metallic layers that consist of a mixture of the two (these "mixture" layer models have not yet been activated in the code). The coolant layer was not included in MOD-1, but its fission product removal was included in the BMI-2104 study.

Therefore, CORCON-MOD-2 allows up to six layers: a "heavy" oxide layer; a heavy oxide--metallic mixed layer; a metallic layer; a light oxide--metallic
mixed layer; a light oxide layer; and a coolant layer. Almost invariably, the initial specification will be such that CORCON will initially align itself with a heavy oxide layer on the bottom and a metallic layer on the top (details of the accident sequence and plant will determine if there is a coolant layer present). As concrete decomposes due to the thermal attack, the oxidic decomposition products become incorporated into the melt. Those entering the metallic layer will be transferred to a developing top light oxide layer. Those entering the heavy oxide layer on the bottom are assumed to stay with that layer. This alignment of layers is a result of the assumption of volumetric additivity. There is evidence that such an assumption is wrong. The WECHSL model arbitrarily alters the density of UO₂ so that the metal is thermal dense phase. As these lighter concrete oxides dilute the heavy oxide layer, the mixture density decreases and eventually will become less than that of the metallic layer. At that instant, CORCON "inverts" the two layers—placing the metallic layer on the bottom and the "heavy" oxide layer on the top where it is instantly mixed with the light oxide layer that had been developing there.

Decomposition gases (H₂O and CO₂) from the concrete are assumed to enter the melt if generated at the bottom on surfaces with inclinations with the horizontal of less than 15° or to otherwise bypass the melt. The melt/concrete interface is, therefore, assumed to be made up of a thin flowing gaseous film. The gas flowing through the melt is calculated to chemically react with the metallic constituents generating internal heat and converting the H₂O and CO₂ to H₂ and CO respectively. The oxidation products of the reaction with the metallic layer are instantly moved into the oxidic layer. Gases flowing around the melt are not included in the chemical reaction in MOD-1 or BMI-2104 but are included in MOD-2.

Internal heat generation includes the chemical reactions (some are endothermic) and the decay heat. The decay heat power has been adjusted to account for those species that were lost during the in-vessel release period of the accident sequence, but is not adjusted as gas sparging continues to extract radionuclides (these latter losses may be relatively small compared to the decrease in decay heat via the normal radioactive decay).

The internally generated heat and the sensible heat are calculated to be transferred to the various interfaces by three different churn-turbulent-like heat transfer correlations: one for the bottom and top; another for internal...
transverse interfaces; and the third for the sides. These correlations are
experimentally based and have been developed from considerations of bubble
stirred liquids. They include the effects of the void fraction and bubbling
rates as manifested by the rate at which the gases pass through the melt.
Crusting is allowed and modeled in the MOD-2 code (but not in MOD-1/
BMI-2104). If crusts are calculated to form, heat transfer through them is
by conduction and they are treated as not inhibiting the passage of either
the decomposition gases or the concrete melt components.

The heat transfer across the gas film surrounding the melt is calculated as a
combination of convection and radiation. A different convection correlation
is used for the bottom (where the gas is in transverse flow) than for the
sides. A transition from laminar flow to turbulent flow is modeled in the
side films.

Heat transfer from the top surface without a pool of water present is also
additive radiation and convection. The convection coefficient is an
internally fixed constant. Radiation is modeled as between infinite parallel
plates and includes emissivity coefficients to adjust for gray body
conditions. With a coolant layer on top, the full pool boiling correlation
is used in addition to the above convection and radiation, but no adjustments
are made to the nucleate-to-film-boiling transition, ΔT, because of barbotage
effects of noncondensible gases.

Overview Description of DECOMP. Similarly to CORCON, DECOMP treats the
thermal interactions between the melt and the concrete in an axisymmetric
cavity. DECOMP also includes a prepool phase in which the jetting melt
impacts on the cavity surfaces. This phase is not found to have much source
term significance.

In contrast to CORCON, DECOMP models the melt as being a homogeneous mixture
of core, structure, and concrete materials rather than as being segregated
into layers of different compositions.

The concrete "melt" that gets incorporated into and mixes with the core melt
is modeled by assigning properties that approximately correct, but uses the
molecular weight of CaO for adjusting properties as it mixes with the melt.
The decomposition gases evolving from the bottom surface of the cavity pass
up through the melt where chemical reactions with the metal constituents are
modeled. Gases generated at the melt side walls are assumed to bypass the melt.

Internal heat generation comes from the gas/metal chemical reactions (some of which are endothermic) and from decay heat.

The heat transfer from the interior of the melt to the interface is modeled as being convective with a constant (input) value for the heat transfer coefficient. The coefficient is the same for each internal surface (bottom, sides, and top).

A dynamic crust is always assumed to exist at the interface between the melt and the concrete and at the top surface. This interface is, thus, modeled as a permeable crust of solidified core melt that allows concrete decomposition products (gas and liquid) to enter the melt. The heat transfer through all crusts is modeled as being conduction. A temperature distribution across the crust is represented in a quadratic equation and is assumed based on solution of the heat conduction equation with internal heat generation. The inner surface temperature is fixed at the melt temperature-of-fusion, and once concrete melting begins the outer surface temperature is fixed at the concrete "melting" temperature. The crust thicknesses are dynamically determined through a heat balance.

The resultant conductive heat transfer rate determines the concrete melting rate through a simple heat balance. At the crust top surface, the outer surface temperature is determined via a heat balance with the transfer off the top to the environment being additive convection and gray body radiation.

When water is added to the top, DECOMP calculates the heat transfer due to boiling. However, the full pool boiling curve is not modeled. Instead, the critical heat flux (CHF) value is used unless the flux that would result from film boiling exceeds the CHF value. The film boiling heat flux is then used up to a limited value called the "droplet-dispersal" limit. The heat flux that would entirely boil away the water at its feed rate is also fixed as an upper limit. The net effect of the DECOMP boiling model is to quickly quench the core debris.

4.1.2.1.2 Fission Product/Aerosol Release Models. The thermal modeling previously described produces the melt temperatures, the melt compositions,
the gas sparging rates, and the system geometry which serve as input to and boundary conditions for modeling of the resultant release of radionuclides and other aerosols. The release of nonradioactive aerosols was not considered in WASH-1400, but recent studies have underscored their great importance to the subsequent fission product transport in the containment as well as to equipment qualification. Modeling of the fission product/aerosol generation from core/concrete interactions was done in the BMI-2104 study by the use of the VANESA code (14). VANESA was originally intended to be used as a subroutine in CORCON—thus allowing integrated analyses. However, VANESA is presently a stand-alone code. For release calculations, the IDCOR program used a subroutine called CNAERO in the MAAP package (15).

Overview Description of VANESA. The VANESA code mechanistically calculates release of radionuclides and aerosols from the core melt as a result of both vaporization and bubble bursting. The removal of constituents from the melt by the sparging gases is modeled in a manner that is more-or-less standard for extraction processes. The driving force for the mass transport of a particular component is modeled as the difference between the components equilibrium partial pressure and its actual partial pressure in the bulk gas phase. The rate of vaporization is limited by three serial transport processes:

1. Mass transfer in the condensed phase to the bubble interface.
2. Conversion of the condensed species to a vapor species at the interfaces.

The overall mass transfer rate equation, therefore, is

$$\frac{1}{A} \frac{dN_i}{dt} = K (P_{\text{eq}} - P_i)$$

where

$$K = \frac{1}{k_k + \frac{1}{k_l} + \frac{1}{k_g}}$$

$k_k$ = mass transfer coefficient for the liquid phase
\[ k_1 = \text{an equivalent mass transfer coefficient for vaporization at the interface} \]

\[ k_g = \text{mass transfer coefficient for the gaseous phase} \]

Standard correlations are used for the above mass transfer coefficients. However, the appropriate driving forces for which the correlations were derived are the actual equivalent partial pressure differences (concentrations) at the ends of each transport path. Actual endpoint concentration (or equivalent partial pressures) are not determined in VANESA. As noted above, the driving force for each of the serial paths is treated as having the same value in VANESA. For most conditions, this should be an acceptable approximate treatment.

As noted previously, the equilibrium partial pressures are evaluated for most of the species by treating them as ideal solutions. The equilibrium partial pressures are, thus, exponential functions of the melt temperature and are proportional to the mole fraction of the component in the melt. The "back pressure," \( P_1 \), builds up as the bubbles pass through the melt thereby reducing the transport rate as \( P_1 \) approaches \( P_1 \) (eq). However, in VANESA the various constituents vaporized into the gas bubbles are allowed to chemically react in the gas phase with the decomposition gases as converted by their previous reactions with the metal phase of the melt. These gas phase chemical reactions are calculated via a chemical thermodynamic equilibrium approach (basically equivalent to minimization of Gibbs Free Energy Function). The reactions can convert the transporting species into different species within the gas phase effectively reducing the back pressure of the original species and thereby retarding its rate of approach to the equilibrium value. This has the effect of enhancing continued vaporization of the original species into the gas phase well beyond what might have been expected if these important gas-phase reactions had not been considered.

In VANESA, the approach is such that the melt is considered to be segregated into two layers (as contrasted with the several layers in CORCON)--a metallic layer on the bottom and an oxidic layer on the top. The decomposition gases on first entering the melt are immediately brought to chemical and thermal equilibrium with the metallic layer substantially converting the \( \text{H}_2\text{O} \) and \( \text{CO}_2 \) to \( \text{H}_2 \) and \( \text{CO} \). The gas is assumed to rise as 2-cm diameter spherical cap bubbles which have a particular rise velocity that is basically independent
of material properties. This fixes the transit time through each of the two layers and, along with the total gas-flow rate that had been input from the results of the thermal models, fixes the total surface area available for the mass transfer.

The various species in the melt are initially apportioned between the metallic layer and the oxidic layer by input based on oxygen potential considerations and empirical evidence. As time progresses, the composition evolves due to the effects of the addition of the decomposition concrete oxides, the chemical reactions of the metals with the sparge gases, and the loss of constituents via vaporization.

On reaching the melt top surface, the vapors are assumed to condense into aerosols of a size distribution that is calculated from an empirical correlation derived from SNL experiments (14). If there is an overlying pool of water, it would act much like a BWR suppression pool in removing aerosols. Such a suppression pool-like model is being added to VANESA. In the BMI-2104 analyses, this aspect was separately calculated. A separate calculation, however, was made in BMI-2104 to evaluate the aerosol removal due to an overlying pool. As the pool boils away to dryness, it is expected that the previously trapped material would either be released in the boiling process or would be reincorporated back into the melt for possible later release. This release or reincorporation into the melt was not included in the BMI-2104 calculations.

VANESA also includes the generation of aerosols by mechanical means as a result of the breaking of the bubbles as they pass through the top interface. The treatment of this is based on observations of isolated bubbles passing through water. Each gas bubble is assumed to produce 140 particles as it bursts with each particle being 1 μm in diameter and having the bulk oxide melt composition. Recently, the code has been revised to produce ~2000 particles to reflect later experimental data (see ref. 33).

Overview Description of CNAERO. The CNAERO model, like DECOMP, treats the melt as being a homogeneous mixture that is made up of the radionuclides and core components coming from the reactor vessel. The composition (mole fraction) is varied with time by the addition of CaO as a concrete ablation product surrogate, by the chemical reactions calculated in DECOMP, and by the vaporization loss of species.
Apparently some 34 different species are involved (in addition to the CaO). These are assumed to develop their equilibrium partial pressures in the gas phase at the temperature of the top surface of the upper crust. The equilibrium partial pressure values are developed in the code by assuming ideal mixture behavior. The CNAERO model does not include chemical vaporization as affected by possible reactions in the gas phase nor does it include any kinetic transfer processes in the melt or gas phases. Thus, with such an equilibrium model, only the total gas flow is needed. There is no need in CNAERO for actual bubble surface areas, rise velocities, or transit time.

The generation of aerosols by mechanical means is not included nor is there a model for determining the size distribution of the aerosols produced by the vaporization/condensation process.

The effects of any overlying pool on removal of aerosols are not included because the DECOMP model calculates that such a pool quickly "quenches" the melt and effectively inhibits production of gases and aerosols until the pool boils away.

4.1.2.2 Areas Where Studies Agree and Disagree. In general, it is possible to compare different codes and studies that purport to evaluate the same aspect of source term determination by looking for areas of agreement or disagreement that are of three basic types:

1. Model assumptions and corelations,
2. Physical property specifications for the system materials, and
3. Specification of the boundary conditions associated with specific plants and specific sequences

The core/concrete interaction ex-vessel source area "agreements-and-disagreements" among CORCON, DECOMP, VANESA, and CNAERO will be summarized here for the above three categories.

4.1.2.2.1 Model Assumptions and Correlations Areas of Agreement/Disagreement Between CORCON and DECOMP. The areas of agreement and disagreement between CORCON and DECOMP are summarized below in four different phenomenological categories:

1. Physical state of the melt,
2. Heat transfer processes,
3. Internal heat generation, and
4. Concrete ablation and gas liberation.

Assumptions on Layering Versus a Completely Homogeneous Melt. CORCON assumes the melt segregates into a series of oxidic and metallic layers, whereas DECOMP treats the entire melt as being one homogeneous mixture. This difference in assumption as to the physical state of the melt can have several implications with respect to fission product/aerosol release (which will be discussed later where VANESA and CHAERO are compared). With respect to the thermal calculations, this difference could manifest itself through the crusting behavior and related effects because the metallic layer in CORCON has a lower solidus temperature and a higher thermal diffusivity than does the oxidic layers. Thus, the tendency would be for CORCON to have a thinner crust for the metallic layer/concrete interface than for the oxidic layer and to exhibit higher heat transfer rates to the concrete in that area.

It has been observed in practically all core debris/concrete interaction tests that layering does occur when the mixture is steel and UO₂ (4,5,16). However, the technical basis for always assuming layering is not firm, especially in view of the complex chemistry that arises between the oxide and metal phases of core debris. In addition, the circulation influenced by the sparging gases may be sufficient to overcome any melt stratification.

Because the heat transfer models in DECOMP and CORCON are so different, it is difficult to assess the significance of the homogeneous melt assumption.

It is believed that the available information is presently insufficient to resolve this difference.

Formation of Crusts. The CORCON-MOD-1 as used with the modifications for BMI-2104 did not model crust formation. Instead, individual melt layers had viscosity correlations that were continuous with melt temperature so that the viscosity of the oxidic layer merely got very large as the melt approached what would normally be a solidification temperature.

In MOD-2, however, a crusting model was included that is conceptually similar to the modeling in DECOMP. DECOMP assumes that a crust is always present whereas CORCON calculates a crust only when boundary heat transfer dictates that one should form. With the exception of the differences in crust
thickness for the different layers in CORCON, the two models should give similar effects. The differences will be related to the assumptions on layering and, therefore, should be resolved when the issue of layering is resolved.

There is insufficient information to address questions regarding the stability and permeability of crusts and their potential ability to filter aerosols from bubbles passing through.

Experimental evidence at Brookhaven National Laboratory supports the contention that crusts remain porous to the passage of concrete decomposition gases as well as will even entirely solidified melts (A).

Melt/Concrete Interface Modeling. In CORCON, the interface between the melt and the concrete is viewed as consisting of a very thin flowing gaseous film made up of concrete decomposition gases. In contrast to this picture, DECOMP assumes that the interface is effectively a melt crust contacting molten concrete.

The potential effect of this difference would be that CORCON would have an extra path of resistance for the heat transfer from the melt to the concrete (made up of the gaseous film). However, the calculated thickness of this film is usually small so that the resistance to transverse heat flow is believed not to be of much significance.

There is presently insufficient information to justify the presence and/or stability of a continuous thin film around the melt.

Void Fraction Modeling. The gases bubbling up through the melt create a void fraction that establishes the boiled-up height of the pool (CORCON and DECOMP) as well as influencing the internal heat transfer coefficients (in CORCON only).

Both CORCON and DECOMP use the same model for void fraction involving the ratio of the bubble rise velocity and the superficial gas velocity (based on total gas flow). However, CORCON uses a bubble rise velocity correlation that is basically for an isolated bubble whereas DECOMP uses a correlation applicable to a swarm of bubbles which gives a rise rate that is 1.5 times the CORCON velocity.
The result is that CORCON will calculate a larger void fraction (although it has a limit of 0.42), a larger boiled-up height, larger heat transfer coefficients, and longer residence times for the bubbles than it would have had it used the bubble rise velocity correlation that is in DECOMP. The significance of this to the heat transfer is unknown but is believed to be small.

There is believed to be sufficient information already available in the literature to develop a criterion on the use of "swarm" versus a "single bubble" correlation.

Melt-to-Interface Heat Transfer. In CORCON, the sparging bubbles at high flow rates are seen to produce a churn-turbulent convective flow heat transfer condition. Based on literature data and simulant experiments \(^{(17)}\), separate correlations for the heat transfer coefficients are used in MOD-2 for: (1) the side walls and (2) the transverse interfaces (bottom, layer-to-layer, top). However MOD-1 used a correlation for (2) that gave heat transfer coefficients about an order of magnitude lower than those for MOD-2.

DECOMP uses a user-input value for the single internal coefficient for heat transfer from the melt to all internal surfaces (a value of 10,000 W/m\(^2\) · °K is suggested by the manual). For comparison, the coefficients calculated by MOD-1 and MOD-2 typically range from \(10^4\) to \(10^5\) W/m\(^2\) · °K except for viscous oxidic layers where MOD-2 predicts much higher values than MOD-1.

This difference between the treatments in CORCON and DECOMP can be significant because it can influence the apportioning of the heat flux between the concrete and the top surface into the cavity environment.

It is believed that there is sufficient information in the literature to support the use of a functional correlation rather than a simple constant input value, although the exact quantification of the correlation may still be in question.

Heat Transfer Across the Interface to the Concrete. The heat transfer from the melt to the concrete is assumed to take place across the melt crust, if present, and through a thin flowing gaseous film in CORCON-MOD-2 (but only across the film in MOD-1 which does not model crusting). The CORCON modeling includes radiation heat transfer as well as convective/conductive transfer.
In comparison, DECOMP only includes conduction through the crust as governed by the thickness and the differences between the melting temperatures of the "corium" and the concrete.

For even a relatively small crust thickness, the conduction heat transfer across the crust is likely to be the controlling resistance. Hence the modeling differences here between MOD-2 and DECOMP are not believed to be very significant (however, MOD-1 might give different results because it does not model crusting).

Although MOD-2 and DECOMP are quite similar with respect to this model area, there is still insufficient evidence to support the concept of a uniform stable crust around the melt as well as to determine an effective thermal conductivity for use under these conditions.

**Heat Transfer into the Depth of the Concrete.** Thermal conduction into the concrete is not modeled in CORCON. It is assumed that a steady temperature profile is developed in the concrete (detailed transient conduction calculations support the establishment of a quasi-steady state in about 1 minute). This is ample justification to use a simple heat balance for the ablation rate,

\[ q = \rho \Delta H \frac{\partial T}{\partial t} \]

where \( \Delta H \) must include the heat required to bring the concrete from ambient temperature to the ablation temperature.

Although a quasi-steady-state conduction analysis is made in DECOMP, it is solely for the purpose of calculating the gas evolution rates as distributed within the depth of the concrete and is not an influence in the actual concrete ablation rate. Therefore, the two models are in basic agreement in this respect.

All of the models treat concrete as a homogeneous material. They neglect the aggregate (rebar/cement heterogeneity). The mobility of water in the concrete is also neglected (water migrates both down the temperature gradient as a liquid and up it as a gas).
Heat Transfer to the Environment from the Top Surface Without an Overlying Pool of Water. Both CORCON and DECOMP calculate the heat transfer from the top surface of a dry debris bed as being an additive combination of convection and radiation. CORCON inputs a convective heat transfer coefficient of 10 W/cm² °K whereas DECOMP uses 50 W/cm² °K.

A very approximate estimate of the effective series heat transfer coefficients in CORCON and DECOMP was made by this writer as shown below (all h's are W/m² °K).

<table>
<thead>
<tr>
<th></th>
<th>h_{\text{internal}}</th>
<th>h_{\text{crust}}</th>
<th>h_{\text{radiation}}</th>
<th>h_{\text{convection}}</th>
<th>h_{r} + h_{c}</th>
</tr>
</thead>
<tbody>
<tr>
<td>CORCON</td>
<td>10^3−10^4</td>
<td>30−300</td>
<td>100−500</td>
<td>10</td>
<td>110−510</td>
</tr>
<tr>
<td>DECOMP</td>
<td>10^4</td>
<td>30−300</td>
<td>500−1500</td>
<td>50</td>
<td>550−1550</td>
</tr>
</tbody>
</table>

From this table, it is seen that the controlling resistance is likely to be the crust with perhaps some significant contribution from radiation. Hence, the modeling differences here are not likely to be very significant unless a cloud of aerosols (not included in the above table) blocks radiation to the degree that convection off the top surface controls.

This could be accounted for in CORCON but not in DECOMP. CORCON MOD-1 is likely to give higher heat transfer from the top surface because it does not include crusting. The resolution of differences here rests on the resolution of the crusting behavior question and the influence of an aerosol cloud on blocking radiation. There is insufficient information available at this time on these two issues.

Heat Transfer to a Pool of Water on Top of the Debris. Both CORCON MOD-2 and DECOMP model the thermal effects of an overlying pool of water. The modeling for the heat flux from the debris to the water for the two codes is compared in the figure below:
These two models basically predict quite different effects. The DECOMP model, with its relatively high and sustained "critical heat flux" value, quenches the debris and effectively stops the concrete attack and aerosol release until the water is all boiled away. The CORCON model, on the other hand, predicts that film boiling is quickly reached and sustained ($T_i - T_{sat}$ of 800-1200°K are usually calculated) producing only small effect on either the concrete attack or the release from the debris.

The CORCON model appears more consistent with known boiling behavior, and its predicted sustained film boiling has been observed in high temperature melt experiments (18).

Decay Heat Generation in the Melt. The CORCON model basically uses analytical "fits" over five different time intervals from zero to 20 days for the contributions of each of 27 different elements to the decay power versus time as determined for a reference equilibrium core. For a specific core, the initial inventory (before any fission product losses) is calculated to be proportional to the operating power relative to the reference core as calculated by ORIGEN. Retention factors are specified as input for each of the 27 elements in the correlation to account for losses in the RCS before the core/concrete interaction process begins. CORCON also accounts for the continued decrease in decay heat due to losses of alkali metals and halogens to the sparging gases.
The MAAP package determines the decay heat in two separate submodules of the code and inputs these into DECOMP as a function of time. The overall decay power (without fission product losses) is determined in the subroutine POWER which uses the ANSI/ANS standard curve that includes fission products and two actinides. To correct the decay-power for those fission products determined to have been lost prior to the start of core/concrete interactions, the contributions at initial shutdown due to five "groups" is determined in FPHEAT. MAAP specifies the fraction of each of these groups that are transported with the melt to participate in the core/concrete interaction. The fractional contribution to the total due to the lost fission products at time zero (initial scram) is subtracted out of the initial shutdown power and the so-adjusted decay power curve is then assumed to parallel the original ANSI-Standard curve. Additional losses due to sparging are not accounted for.

While in some agreement, there is still a significant difference between these two approaches. The relative contribution of the more volatile elements (that are generally lost prior to the start of core/concrete interactions) is not constant with time. They can contribute up to -40% of the power on initial shutdown but can decrease to -28% after one day. Hence, the decay curve with these elements lost should not be expected to "parallel" the ANSI-Standard curve. This could have some significance for release at long times when MAAP would be expected to significantly underpredict the decay heat power level.

This issue is easily resolvable with existing information by estimating the sensitivity of the overall source term to the longer-term releases.

Heat Generation from Chemical Reactions. The principal chemical reactions in CORCON and DECOMP involve oxidation of metals by concrete decomposition gases, H₂O and CO₂. A chemical thermodynamic equilibrium calculation is made in CORCON for 38 chemical species (11 elements). These species include relevant condensed species (the metals, their oxides, and reduced carbon), the principal gaseous species (water vapor, hydrogen, carbon dioxide, and carbon monoxide) and a variety of less important gases.

While CORCON treats simultaneous oxidations as it occurs, the result of the calculation is that the metals tend to be oxidized in a specific order: zirconium, chromium, iron, and nickel (not much nickel is likely to oxidize). Carbon that is produced during Zr oxidation by CO₂ is allowed to undergo a "coking" reaction that takes place generally after depletion of Zr.

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Heats of formation of all species are automatically included in the thermodynamics properties package.

DECOMP does not use a chemical thermodynamic equilibrium approach. Rather, it allows the reactions to proceed in five stages:

1. All of the zirconium reacts with all of the H\textsubscript{2}O and CO\textsubscript{2} simultaneously.

2. When all Zr is consumed, the reduced carbon from the reaction in stage 1 is reacted with the H\textsubscript{2}O and CO\textsubscript{2}.

3. After all the carbon has reacted, the gases then react together with Cr and Fe\textsubscript{0} to produce Fe\textsubscript{0}·Cr\textsubscript{2}O\textsubscript{3} and free-H\textsubscript{2} and CO. This stage proceeds until all of the Cr is consumed.

In the above three stages, the rate of reaction is governed strictly by the gas feed rate into the melt.

4. The fourth stage reacts the remaining iron with the H\textsubscript{2}O and CO\textsubscript{2}. However, this reaction is not treated as going to completion. Rather a quasi-equilibrium approach is used in which a fraction reacts to maintain a designated constant ratio of steam-to-hydrogen and CO\textsubscript{2}-to-CO.

5. The fifth stage begins when all of the iron is consumed. In this stage, it is assumed that Ni would form NiO and FeO would be further oxidized to Fe\textsubscript{3}O\textsubscript{4}. A quasi-equilibrium treatment similar to that of stage (4) is used here with the designated steam/hydrogen and the CO\textsubscript{2}/CO ratios set at high levels. These reactions are therefore not very significant.

It is believed that these two approaches will give significantly different results and that the CORCON treatment is the preferred.

**Concrete Decomposition (or Melting) Temperature.** Melting ranges for some typical concretes have been determined experimentally (11) and are included in CORCON-MOD-2 for three "built-in" concrete varieties as shown below:

<table>
<thead>
<tr>
<th>Concrete Type</th>
<th>Solidus (°K)</th>
<th>Liquidus (°K)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Limestone/common sand</td>
<td>1420</td>
<td>1670</td>
</tr>
<tr>
<td>Basaltic</td>
<td>1350</td>
<td>1650</td>
</tr>
<tr>
<td>Generic S.E. U.S. (CRBR)</td>
<td>1690</td>
<td>1825</td>
</tr>
</tbody>
</table>

The user may also specify different values for these.
A single-valued melting temperature is specified as input to DECOMP. A typical value used in IDCOR for Basaltic concrete was -1370 K. The CORCON user must also specify an ablation temperature which is the temperature at which concrete constituents leave the concrete surface and are incorporated into the melt pool.

Hence, the two approaches are not sufficiently different to make much impact on source terms.

**Heat of Ablation of Concrete.** A heat of ablation built into CORCON represents: (1) the integrated enthalpy change from ambient temperature to the ablation temperature, (2) the energy required to vaporize both the free and the bound water and CO₂, and (3) the latent heats of fusion of the various melt components. A typical heat of ablation developed in this manner for, say, limestone/common sand is -720 cal/g. The ΔH varies from values of -550 up to 950 depending on the concrete.

By way of comparison, DECOMP uses an effective latent heat value of -430 cal/g which does not include Cp (Tₘₐₙ - Tₒ) because it is calculated separately.

Assuming Cp = 0.88 kJ/kg °C; Tₘₐₙ = 1100 °C; and Tₒ = 100 °C; then,

$$ΔH = 430 + \frac{(0.88)(1000)}{4.184} = 640 \text{ cal/g.}$$

The above values are close. However, it is expected that the release rate of fission products and aerosols will be approximately linearly dependent on this property value because ΔH determines ablation rate which determines gas sparging rate. Therefore, it is believed that it is important to have accurate values for this parameter. The information to develop reliable values is believed to be sufficiently available except for the difficulty in determining the actual concrete composition for specific plants.

**Concrete Composition.** One area of considerable difference between CORCON and DECOMP is the detail with which the concrete composition can be specified. CORCON allows very complete specification of the composition of any concrete in terms of the mass fractions of up to 13 components. The specified composition is used in CORCON to develop the physical and thermodynamic

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properties. DECOMP, on the other hand, specifies the thermochemical properties as input. When ablation occurs, the material added to the melt is considered to have the molecular weight of CaO.

From the thermal standpoint, this may not make much difference. But for later use in determining the melt composition as related to aerosol release determination it could be significant.

There is sufficient experience to have confidence in the more detailed treatment in CORCON and it is believed that particular plant concrete can be specified at the desired level of detail.

Gas Content of the Concrete. The gas content of the concrete consists of free water, chemically bound H₂O and chemically bound CO₂ from carbonates. Typical values used for a limestone concrete are compared below for CORCON and DECOMP.

<table>
<thead>
<tr>
<th>Component</th>
<th>Content % (by weight)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Free H₂O</td>
<td>CORCON: 2.7    DECOMP: 2.8</td>
</tr>
<tr>
<td>Bound H₂O</td>
<td>CORCON: 2.0    DECOMP: 2.8</td>
</tr>
<tr>
<td>Bound CO₂</td>
<td>CORCON: 21.2   DECOMP: 25.1</td>
</tr>
</tbody>
</table>

Free water content is a function of ambient humidity. Values roughly double those above will develop at 95-100% RH.

It is seen that the codes agree reasonably well for this concrete. Other concrete types can have much less bound CO₂ and, therefore, the code user must be careful to make the correct specification.

Bubble Formation and Rise Rate. In CORCON, a bubble size is needed because it enters into the internal transverse convective flow heat transfer coefficient correlation.

The bubble size in CORCON is selected as being the size of the Taylor instability cells. The rise rate is then determined from a correlation for a single spherical cap bubble. However, since the Taylor cell size is

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dependent only on physical properties, then the rise rate is also only a function of the melt physical properties.

For comparison, DECOMP uses a correlation due to Zuber and Findlay (19) for a swarm of bubbles in churn turbulent-flow. This correlation is basically identical to that of CORCON (for the same physical properties) except that it is 1.15 times that of CORCON.

There is little significance to this parameter in DECOMP (it is only used to determine the void fraction and, hence, the boiled up geometry). However, there are significant implications with respect to CORCON which also uses the void fraction in its heat transfer correlation (and with respect to VANESA release calculations discussed later). Since the DECOMP model is for a swarm of bubbles whereas the CORCON model is for a single rising bubble, then, intuitively, the DECOMP rise velocity correlation would appear to be more appropriate for use in CORCON.

There is sufficient information available in the bubble rise literature to be able to make a judgment as to whether or not to use a "swarm" correlation as opposed to a single bubble correlation and such a determination should be made and the appropriate correlation used.

Areas of Agreement/Disagreement Between VANESA and CNAERO. The results of the thermal calculations (CORCON and DECOMP) provide input and boundary conditions to the models for calculating the fission product/aerosol release and composition. For the BMI-2104 study, release calculations were provided by the VANESA code. For IDCOR, they were provided by the CNAERO module in the MAAP package. Important aspects of the modeling within these two codes are compared in this section.

Vaporization Release Treatment. The most important fundamental difference between the the VANESA and the CNAERO modeling lies in the treatment of the kinetics of mass transport due to sparging. Kinetics are not modeled at all in CNAERO. Instead, CNAERO assumes that each constituent within the melt reaches its equilibrium partial pressure at the temperatures of the top surface.

In contrast to this treatment, VANESA models the details of the kinetic processes as discussed in the Overview Description of VANESA, which includes:
1. mass transport in the condensed phase,
2. surface vaporization rates, and
3. transport within the bubble gas phase.

As with any modeling of rate processes, the difference in results obtained with an equilibrium treatment versus a kinetic treatment lies in how close an approach to equilibrium is actually obtained in the specific application. Therefore, the resolution of the importance of this difference can easily be obtained by applying VANESA under representative conditions and assessing the nearness to equilibrium. It is believed that evaluating the kinetic processes is important and is the preferred treatment in any event.

In both the kinetic and equilibrium treatments, values are needed for the various constituent equilibrium partial pressures. In both VANESA and CNAERO, a Raoult's Law assumption is made in which the equilibrium partial pressures are taken as the values for the pure substance reduced by their mole fractions in the melt. The implicit assumption here is that activity coefficients are basically unity (VANESA includes nonunity activity coefficients for a couple of constituents for which experimental data exist). There is presently insufficient data to establish actual activity coefficients for these materials under these conditions.

A most significant area of disagreement between VANESA and CNAERO is in the treatment of chemical reactions within the gas phase of the rising bubbles. No gas-phase chemistry is modeled in CNAERO. VANESA, on the other hand, performs a chemical thermodynamic equilibrium analysis (basically a minimization of Gibbs Free Energy) in which 129 different species are included. These species were selected as those that should form in a ternary component element-oxygen-hydrogen system. For example, the element La is assumed to be able to exist in the gas phase in the following possible forms:

La, LaO, La$_2$O$_3$, LaOH, and La(OH)$_2$.

In other words, in this analysis, all of the elements in the gas phase compete together for the oxygen and hydrogen but are not "allowed" to chemically react with each other. There are four exceptions to this because SnTe, SbTe, AgTe, CsI are also included as possible species.

It is very important to include these gas phase chemical reactions in the release modeling. The reason for this can be illustrated by the following example for barium. The assumed form for barium in the melt is BaO. It will transport through the liquid phase, evaporate at the interface, and transport through the gas phase according to the kinetic treatment in VANESA governed by the driving force determined by the difference, (P(eq)−P$_g$). However, the chemical reaction in the gas phase, presumably

\[ \text{BaO (s)} \rightarrow \text{BaO (g)} \xrightarrow{H_2O} \text{Ba(OH)$_2$ (g)}, \text{etc.} \]

will depress the BaO (g) partial pressure, P$_g$, in the gas phase. Hence, the chemical reaction acts as a sort of "chemical pump" that maintains the kinetic driving force at a higher level thereby significantly enhancing barium release compared to a treatment that neglects gas phase reactions and only assumes BaO reaches its equilibrium partial pressure in the gas phase.

Although the kinetics of the gas-phase reactions are not modeled, there is
sufficient reason to believe that they are very fast under these conditions as compared to the other rate processes. In addition, sufficient confidence can be placed in chemical equilibrium thermodynamics under these conditions to justify including this within the modeling of vaporization release.

While the above illustrates a major difference between VANESA and CNAERO, it also points out an artifact in the manner in which VANESA handles the specification and transport. As, for example, the Ba(OH)$_2$, etc. partial pressures build up in the gas phase due to the chemical reactions there, it should tend to "back-transport" into the melt by the overall kinetic model in VANESA. However, Ba(OH)$_2$ is not "recognized" as a melt-phase species, so that VANESA will treat the back transport within the melt as if the Ba(OH)$_2$ were converted back to BaO there. In reality, this is probably what would happen in this case, but not necessarily so for all other species. The equations are full reversible and the thermodynamic limits do not appear to ever be exceeded.

The application of the kinetic mass transport equations in VANESA requires determination of the total bubble surface area as well as the transit time through the melt layers as determined by the bubble rise velocity. Therefore, a specification of the bubble size is needed. In CORCON, the size of the bubbles sparging through the melt was assumed to be given by the Taylor cell size. A slight inconsistency occurs between CORCON and VANESA in that VANESA inputs a fixed, empirically determined, bubble diameter of 2 cm. The rise velocity associated with these bubbles is calculated in VANESA by

$$V_b = 0.72 \ (dg)^{1/2}$$

As mentioned previously, this correlation is for single rising bubbles. However, because of their hydrodynamic interactions, a swarm of bubbles will rise somewhat faster (~1-1/2 times the above correlation). Hence, VANESA could be using a transit time that is too long—giving more time for mass transport than would actually be available. It is believed that sufficient information exists to make a judgment as to whether or not a "swarm" correlation is required. There is some evidence, however, that existing swarm correlations overpredict data on mass transport.

Physical State of the Melt. As in DECOMP, the CNAERO model assumes the melt is a homogeneous mixture whereas VANESA assumes the melt is segregated into an oxidic layer and a metallic layer. For its gas-phase chemical thermodynamic equilibrium calculation, VANESA treats the concrete decomposition gases in the bubbles as if they had already reacted with the metallic phases of the melt. Therefore, by implication, in VANESA the metal layer is always on the bottom and the oxidic layer is always on the top. In this respect, then, there is some inconsistency with the thermal model in CORCON. In addition, the thermodynamic equilibrium/mass transport calculation is made in VANESA as if the melt layers were being independently sparged by the bubbles.

It is not known to what extent the layering versus homogeneous assumption affects the release calculations. One effect could be that individual components can be more concentrated in a given layer (for example La$_2$O$_3$ would exist only in the oxidic layer). The resulting higher mole fraction would give a higher equilibrium partial pressure (and hence, higher release rates) than if the melt were a homogeneous mixture, diluted by the metallic layer components. The completely homogeneous assumption require that steel, etc. dissolve in UO$_2$. For temperatures less than 3100K steel does not significantly dissolve in UO$_2$ and, therefore, the melt is to be "homogeneous." It must consists of a continuous phase with islands of a second condensed phase.
Melt Composition. For the mass transport calculations, VANESA includes the following species in the metallic phase:

- Ag  Sn  Fe  Zn
- Mo  Sb  Ni  Mn
- Ru  Te  Cr

and the oxidic phase includes:

- UO₂  Al₂O₃  BaO  CeO₂  Cr₂O₃
- ZrO  Na₂O  SrO  NpO₂  N₁O
- CaO  K₂O  CsI  PuO₂  MnO
- SiO  Cs₂O  La₂O₃  FeO  Nb₂O₅

For comparison, the CNAERO model results (in an IDCOR presentation) listed 34 different components as being released (these are not explicitly identified in the available CNAERO documentation). These were:

- Ru  Cu  Np  ZrO  Pm  Cr  Ce  Ni  Pr
- SrO  Sr  Te  Br  BaO  UO₂  Sn  PuO₂  Eu
- Fe  Y  Ag  Pd  I  Rb  Cs  Se
- Zr  Nd  FeO  La  Cd  Mo  In  Tc

It appears, therefore, that there is an important difference between VANESA and CNAERO in their choice of materials and chemical forms as assumed in the melt.

A combination of experimental data and consideration of chemical potentials is needed to resolve the differences.

**Effects of an Overlying Water Pool on Release.** In CORCON, an overlying pool of water is predicted to persist in a film boiling regime which does not quench the melt. Therefore, release of aerosols and fission products via gas sparging does not stop. However, the materials must pass through the overlying water pool before release to the containment. The removal of aerosols passing through the water pool may be treated with models similar to those for suppression pools. Limited versions of such models are being added to VANESA, but the calculations for the effects of such water pools were made...
separately in the BMI-2104 study. Application of suppression pool models to vigorously boiling overlying pools has not been validated experimentally. In addition, consideration is needed for the possible resuspension of previously removed material as the pool boils to dryness. Such resuspension was not included in the BMI-2104 calculations.

In DECOMP, an overlying pool of water is calculated to quench the melt and to stop concrete attack (and resultant release) so long as the water is there. Therefore, a treatment of the removal potential of such a water pool was considered unnecessary in CNAERO. A recent analysis of MCCI experiments supports the modeling approach taken in CORCON/VANGSA as long as steam explosives do not occur (ξ).

A question that needs resolutions is the initial particle size distribution impact to the water pool. VANESA uses and empirical model derived for circumstances without a pool present. To address this problem may require a fairly mechanistic model of particle formation and growth. In addition, the question of resuspension must be addressed. There is insufficient information in both of these areas at present.

Mechanical Release. As discussed in the overview, VANESA includes the generation of aerosols as a result of the bursting of the bubbles as they pass through the melt/atmosphere interface. Based on observation of isolated bubbles in water, each bubble produces 140 particles each 1 µm in diameter having the bulk oxide composition,

CNAERO does not include mechanical generation of aerosols.

This mechanism produces only a small quantity of airborne material and does not appear to be important until late in the sequence. If sensitivity studies show it to be an important contribution to source terms, then it can easily be included in CNAERO. Recent information suggests that the production of 140 particles per bubble is probably a lower limit. This value could be as high as 2000 particles/bubble. Hence, there is significant uncertainty in this area.

4.1.2.2.2 Physical Properties Specifications. In the calculation of the thermal and mass transport behavior of core/concrete interactions, a number of properties of the molten core/concrete mixture are required. Depending on
the detail of the modeling, different codes may require different sets of properties. We have found that there are considerable differences among various codes for physical properties specification with respect to:

- the number of components included.
- the chemical form.
- the level of detail of the specification.
- variations with temperature.
- the development of mixture properties from individual constituent specifications.

It is believed that sufficient information is available to specify most of the needed individual component physical properties at a sufficient level of detail. Insufficient information is available to validate the values for the mixture properties as determined in the codes. It is not clear to what extent differences in these property values will influence the core/concrete interaction material release rates.

4.1.2.2.3 Specification of the Boundary Conditions Associated with Specific Plants and Specific Sequences. Markedly different ex-vessel sources can be obtained from the core/concrete interaction models as a result of influences that are external to the code modeling itself as related to the definition of the conditions of application of the codes. Some of these external conditions depend on the results of calculations by other codes or models for other phases of accident sequences, some are related to the specification of the sequence itself, and others are related to interpretation of specific plant geometry or features.

Some such items that have been identified are listed below:

1. timing of the melt ejection into the cavity.
2. percent of core melted.
3. initial melt temperature.
4. initial composition of the melt.
5. cavity geometry.
6. dispersal of debris within the cavity and adjacent compartments.
7. concrete specification.
8. presence (or timing) of water addition to the debris.

The nature of how these can affect the core/concrete interaction were presented in the QUEST study (20) and are discussed below.

Timing of Melt Ejection. The timing of melt ejection into the cavity is generally a parameter that is determined by codes or code modules that address the in-vessel core melt progression phase of an accident sequence. Differences in timing that might be predicted by these can influence the core/concrete interaction portion of ex-vessel sources in basically two ways:

- the timing of melt ejection relative to RCS system pressure may determine the dispersal of the debris into adjacent compartments—thus influencing both the quantity and coolability of the debris in the cavity, and
- the timing can have an influence on the decay heat level as well as the composition and temperature of the ejected melt.

The QUEST study found that if one only considers the effect of timing as manifested in the decay heat level then there is not a severe effect on the ex-vessel source term. The results for aerosol generation rates and cumulative radioactivity release agree to within a factor of about 2 when the melt ejection is assumed to occur at 1 h after scram as compared to 10 hr after scram.

Percent of Core Melted. Traditionally, accident analyses have assumed that the entire core is ejected into the cavity. It is possible, however, that some unspecified portion of the core can remain with the vessel and not participate in the core/concrete interaction. The prediction of this quantity is a function for the core melt progression models and involves the issues associated with in-vessel thermal hydraulics. The resolution of how much of the core gets ejected may rest upon whether internal circulation and radiation heat transfer will cause uniform degradation of the core across its diameter.

DECOMP receives a time-dependent melt source whereas CORCON is now designed to handle a single melt drop at \( t = 0 \).
The percent of core participating in the core/concrete interaction has two basic influences. The total amount of mass of certain constituents that can be made airborne is reduced in proportion to the amount of core ex-vessel. In addition, the smaller the amount of core debris, the lower the debris temperature should be.

The QUEST study looked at the sensitivity of the core/concrete source to the percent of the core participating in the interaction by comparing a case of 100% core participation with a case of 50% core participation but with the same amount of steel. They found that there was a substantive effect on the ex-vessel source term primarily because of the effect on reducing melt temperature. The cumulative radioactivity release in the case of 50% of core involved was found to be one-fourth the 100% case and there was roughly a 50% decrease in gas release and concrete penetration distance.

Initial Melt Temperature. Determination of the initial melt temperature at the start of core/concrete interactions is also a function of the core-melt progression models. The QUEST study evaluated the effects on CORCON/VANESA results due to three different initial starting temperatures: 1807°C, 2173°C, and 2550°C. The primary thermal effects were found to be a shift in time of layer flip in CORCON—the higher temperatures giving earlier flip times. The melt temperatures remained higher for about the first 4 hr for initially higher melt temperatures but were lower thereafter in the oxidic layer (due to dilution via early concrete ablation) and approximately the same in the metallic layer.

The effects on fission product/aerosol release come from the sensitivity of vaporization release to melt temperature, to the gas evolution rates (which are tied to melt temperature through the heat transfer) and through the effects of dilution via the addition of ablated concrete.

The total span of radioactivity release for the three cases was only about a factor of 2. However, the percent release of strontium was increased by a factor of 2, and lanthanum release changed from insignificant values at initial temperatures of 1807 and 2173°C to the order of 2% release at 2550°C. The choice of initial debris temperature can influence release of refractory fission products at early times and can affect the overall releases at later times.
Initial Composition of the Melt. There is considerable uncertainty regarding the content of certain components within the initial melt—particularly the quantity of unreacted Zr, the quantity of molten steel, and the amounts of some of the relatively volatile fission products that could be affected by the quantity of unreacted zirconium present during the in-vessel release periods (e.g., Te, Ba, Sr, Ru).

It is apparent that relatively volatile fission products not released in-vessel will be released almost quantitatively during core/concrete interactions.

The effects of the amount of molten steel and unreacted zirconium, however, are not so apparent. The zirconium would be expected to represent an additional heat source through chemical reactions with the sparging gases—thereby tending to increase the melt temperature and enhancing the release rates. Steel, on the other hand, would tend to act more as a diluent and would tend to decrease the release rates.

The QUEST study using CORCON-MOD-1/VANESA found relatively significant effects of these two parameters when they were above and below certain "threshold" limits. However, much of the effects in this study could be attributable to a viscosity correlation on CORCON MOD-1 that has been improved in MOD-2. Later calculations with MOD-2 have shown roughly the same as MOD-1 sensitivities to the steel and zirconium contents.

Cavity Geometry (Floor Area). The selection of the value for cavity floor area to go into CORCON/VANESA is sometimes difficult to make. Interpretation of plant drawings and anticipation of the core debris behavior can lead to different conclusions. Two "limiting" cases have sometimes been considered:

1. The entire horizontal floor area in the reactor cavity and any attached tunnel or keyway.
2. Only the horizontal surface directly below the reactor vessel.

In the case of the QUEST study of the SURRY plant, these two choices lead to ~57 m² and 35 m² respectively.

The choice of floor area changes the surface to volume ratios for heat transfer, changes the relative heat flow resistance paths to the concrete.
versus to the cavity atmosphere, changes the depth through which bubbles sparge, and can perhaps bring additional structures into play.

Variation of the floor surface area at the two values above by QUEST had little effect on the source term and does not seem to be significant over that range.

More significant effects could result, however, if the cavity geometry and the details of the melt ejection are such as to strongly affect the quantity of debris participating in the core/concrete interactions as discussed below.

Dispersal of Debris within the Cavity and Adjacent Compartments. One area of significant difference between the IDCOR and the BMI-2104 studies lies in their assumption on the dispersal of debris on ejection out of the reactor vessel. The IDCOR study for Zion, for example, assumed that substantial amounts of the debris were dispersed through the instrument tunnel onto the floor in the lower compartment where it resided in a coolable geometry and did not participate in the core/concrete interaction. The BMI-2104 study allowed the full melt ejection mass to remain in the cavity area.

Resolution of this issue will require additional modeling as well as experimental validation to evaluate the dispersal of debris under such conditions.

Concrete Specifications. In the DECOMP modeling, specifications for different concrete types are not permitted. However, CORCON does allow specification of concrete type and composition. The QUEST study found little significant sensitivities to concrete type.

The QUEST study also indicated that it can be important to include species in the concrete that are less than about 1% by weight. In particular sodium, potassium, and iron contents of the CaO, Al₂O₃, and SiO₂. Late in the course of core/concrete interactions, vaporization of these species is the dominant source of aerosols and becomes the dominant feature of the late time processes that mitigate the accident source term.

Water Addition to the Core Debris. Some accident sequences involve the possible addition of water to the core debris in the cavity. Such water addition could act like a suppression pool to remove aerosols generated by
the core/concrete interaction or could quench the melt stopping generation of aerosols. In either event, the net effect would appear to be the absorption of some quantity of energy from the melt that might have gone into concrete attack and aerosol generation and a delay in the release until all of the water had been boiled away.

The sensitivity of core/concrete release to the above has not been evaluated—however, it could be anticipated that the effect would be similar to a combination of a lowered initial melt temperature and a later time of start of core/concrete interaction. Depending on the timing and amount of water addition, then, the influence could be somewhat significant. The Swiss tests at Sandia show DFs for 60 cm of water of 17-30.

4.1.2.3 Conclusions and Recommendations. Core/concrete interaction provides the driving forces for the later phases of severe accidents including loadings on containment and is a source of radioactive and nonradioactive materials. It can also be a direct cause of containment breach due to either basement penetration or melt-through of some local containment area that might contact core debris.

Aerosol production during core/concrete interaction is important to the estimation of radioactive inventories suspended in the containment atmosphere. There is considerable uncertainty in the calculation because of uncertainties in the models, in the inputs, and the boundary specifications.

The barriers to application to different plants include:
- the effects of cavity geometry related to melt surface-to-volume ratios and to the dispersal of debris.
- the concrete specification.
- items related to the RCS response that might affect
  - melt composition.
  - initial melt temperature.
  - decay heat level.
- sequence related items that might influence water addition to the debris.

There are substantial differences in how the various source term studies model and treat core/concrete interactions, and these differences give rise to substantial
effects on the predicted source terms. It is believed that there is insufficient information presently available to resolve the major differences.

4.1.3 Other Ex-Vessel Sources

Other ex-vessel sources to be discussed in this section include:

- pressurized ejection of core materials from the RV.
- energetic thermal interactions of the core materials with water in the cavity.
- fragmentation of water droplets due to blowdown ejection from the RCS.
- boiling or flashing of water pools.
- resuspension from containment surfaces due to hydrodynamic forces.

Of these, only the fragmentation of water droplets has been considered in any of the source term calculations available. The first two of the above were the subject of two speculative review papers prepared for GREST by D. A. Powers. The discussions that follow, related to these two items, were taken entirely from Powers’ papers, using his exact phraseology in many instances.

4.1.3.1 Pressurized Ejection Sources.

4.1.3.1.1 Technical Background. Previous studies of the progression of reactor accident sequences have indicated that relatively higher frequency accidents for PWRs involve core melting that takes place while the RCS is still pressurized to a level that may be near the normal operating pressure of the reactor and that meltdown under pressurized conditions is also an important component of the spectrum of BWR accidents. Under accident conditions where the core-melt is expected to penetrate the RV bottom head and be ejected into the cavity below while the RV is still pressurized, it is speculated that the ejected material could be dispersed into the cavity and surrounding containment volumes as small airborne particulates and, thereby, become a source of both heat and aerosols to the containment atmosphere. However, newer considerations using integrated fission product transport and thermal-hydraulic analysis including natural convection indicate that other regions of the RCS may fail prior to vessel melt-through, thus providing a path to relieve the high pressure prior to expulsion. Some newer models of core degradations have also indicated that core slumping may occur in a more
piecewise fashion, perhaps allowing vessel melt-through with only limited quantities of core melt available for ejection.

Nevertheless, high-pressure melt ejection could be an important source of aerosol and radionuclide release and for pressure loading of the containment through direct heating. The latter has been discussed in the technical annex, which is concerned with containment thermal hydraulics.

From a limited number of tests (21-25) conducted at Sandia National Laboratories, the following observations could be made about the ejection process, presumed to occur at high pressures.

1. Dispersing debris were observed to directly heat and pressurize the atmosphere to a certain extent. Pressure and temperature data indicate that the debris interacted thermally and probably chemically with the containment atmosphere.

2. Considerable aerosol generation was observed.

3. Tests conducted with nitrogen as the pressurizing gas showed at least a bimodal aerosol distribution with modes occurring at about 0.5 and 5 μm. The finest aerosol material consisted of agglomerates of finer particles (0.1-0.05 μm) and the second mode (∅ 5 μm) consisted of individual spherical particles.

4. In tests conducted with CO₂ as the pressurizing gas, the aerosol size distribution had no mode in the 5 μm range.

5. X-rays taken during the melt ejection process indicated a two-phase flow for the nitrogen pressurized tests but a single phase for the CO₂ pressurized test.

6. Recent large melt tests have shown melts to be dispersed to a height of about 40 m and a distance of about 80 m, in open surroundings.

7. Auxiliary experiments in which small (-0.25 mm) model droplets are suddenly exposed to air have shown that the particle is heated by chemical reaction up to some point at which the particle suddenly disintegrates into much finer particles (0.01-1 μm).

It should be pointed out here that a couple of tests conducted at Argonne National Laboratories (23,24) in confined geometry with corium melt ejected at high pressure in a 1/36th scale mock-up of Zion containment in which the direct heating of the containment atmosphere was rather small. However, a significant portion of direct heating is expected to come from oxidaton of
the metals. As the Argonne tests were conducted in an inert environment, this contribution was absent. Based on Sandia observations, a combination of mechanisms have been hypothesized as being influential in the aerosolization process:

1. **Pneumatic Atomization**: As the melt is ejected from the vessel, the pressurizing gas could "pierce" the melt and entrain the liquid as small droplets. Pilch (26) has analyzed this process and has shown, based on evaluation of maximum stable drop sizes, that at high pressures entrained melt would be expected to be reduced to fine particles.

2. **Jet Disintegration**: Even without the pressurizing gas penetrating the melt, the instabilities of the emerging liquid jet in relative flow with the cavity atmosphere can generate small droplets.

3. **Vaporization**: The finest, agglomerated aerosols observed in the tests are consistent with expectations of a process involving vaporization from the melt and subsequent condensation into aerosols.

4. **Gas Effervescence**: Powers has hypothesized that the difference in the observed behavior of the tests with N₂ pressurizing gas compared to CO₂ lies in the fact that N₂ is expected to be quite soluble in the test melts, whereas CO₂ should be effectively insoluble. For reactor conditions, Power (27) has presented analyses that show that H₂ and H₂O in the reactor vessel atmosphere would also be quite soluble in core melts. The analyses indicate that the volume of gas effervescing from a melt at 2800°K following depressurization from 170 atm to 1 atm would be about 1 cm³ gas/cm³ UO₂ melt and as much as 1360 cm³ gas/cm³ stainless steel melt. These volumetric expansions are large enough to cause substantial disruption of the melts. Effervescing gas from the melt could drastically increase the melt surface area and presumably the rate of vaporization of volatiles from the melt as well as mechanically shear the liquid into fine droplets.

5. **Chemical Reactions of Dispersed Droplets with the Atmosphere**: Debris expelled from the cavity will be hot and reactive toward the containment atmosphere. Steel, zirconium, and even UO₂ debris can react exothermically with either oxygen or steam. The resulting high temperature of the droplets could both enhance vaporization of volatile materials and cause further fragmentation of the droplets (perhaps due to internal boiling).

4.1.3.1.2 Agreements/Disagreements Among Various Source Term Studies: To date, no source term studies have incorporated high pressure melt ejection as an ex-vessel source of radionuclides and aerosols.
4.1.3.1.3 Areas Where Information is Sufficient/Insufficient: Models can be formulated to predict when the pressurizing gas will pierce the layer of melt being expelled from the vessel in terms of the molten pool depth, systems pressure, and breach size (26). Models for the droplet size exist. Criteria for entraining melt in gas exist. What is not available is a model of the rate of melt entrainment. Without such an entrainment rate model, the magnitude of aerosol generation rate by the pneumatic atomization mechanism cannot be assessed.

Once the melt has been expelled from the vessel, "blowdown" of the pressurizing gas could further entrain melt from the cavity surfaces. Analyses have shown that droplets so created would be much larger than those typically associated with aerosols. Models for such entrainment are available.

Neither comprehensive models nor definitive data on chemical reactions of debris droplets lofted from the cavity exist. A current hypothesis is that the rates of reaction are controlled by mass transport of oxidant to the debris surface. On the upward trajectory, this mass transport is controlled by entrainment of atmospheric gases into the debris cloud (a relatively inefficient process). Once the "cloud" has reached the trajectory peak and becomes more dissipated, more conventional mass transport processes control (i.e., mass transport between a droplet (or particle) in relative flow through a reactive gas). Experimental validation is lacking for the particular materials under the particular conditions of interest.

4.1.3.1.4 Conclusions. Heat transfer from dispersed droplets to containment pose interesting questions about the loads on containment and the possibility of containment failure nearly coincident with vessel failure.

The associated loads and aerosol production appear to require at least two conditions:

1. that the reactor coolant systems remain at significant pressure through the time of vessel lower head melt-through, and

2. that significant quantities of molten core debris are available for expulsion from the melt.

Characterization of the pressurized melt ejection source has only just begun. Models and data are needed, particularly in the following areas:
1. the magnitudes and characteristics of aerosol generation as
   a function of system pressure, melt temperature, and melt
   composition;

2. effects of reactor cavity and containment geometry on
   aerosol production;

3. radionuclide release caused by chemical reactions of
dispersed core debris with the containment atmosphere; and

4. whether in-vessel core degradation leads to the initial
   conditions necessary for high pressure melt ejection.

It is believed that the barriers to generic application are substantial. The
postulated models for prediction would require the details of the RCS failure
to establish depressurization conditions which would depend on reactor
type. In addition, the details of the reactor cavity and containment
connections will be required to evaluate the extent of dispersion.

4.1.3.2 Core Melt/Water Energetic Thermal Interactions.

4.1.3.2.1 Technical Background. The issue of steam explosions as a possible
ex-vessel source of fission products and aerosols gets folded into the
broader issues of the probability of occurrence of steam explosions, the
quantity of melt participating, and the resultant efficiency of energy
conversion. There have been speculations (e.g., WASH-1400) that an energetic
interaction occurring when the molten fuel encounters water in the reactor
cavity could loft finely divided debris into the containment atmosphere where
it could chemically react to release some fission products via vaporization.
The Reactor Safety Study made estimates of the release via this process using
data collected by Parker et al. at ORNL (28,29). In the ORNL experiments,
fragments of irradiated nuclear fuel were heated to temperatures of
500-1200°C for periods of up to 90 min in a stream of air during which time
the release fractions for a variety of fission products were measured. These
data were applied in the RSS to that portion of the core estimated to partici-
bate in a violent interaction with water. Near complete release of the
noble gases, halogens, alkali metals, ruthenium, and tellurium were
predicted. These predicted releases of the noble gases, halogens, and alkali
metals during steam explosions are not important because these are released
from the molten fuel anyway during severe accident sequences. By the time
melt can form, most of the volatile species would have already left the
fuel. Similarly, other phases of severe accidents (e.g., core/concrete
interactions) would result in nearly complete release of the tellurium.
The most important feature, then, of the RSS treatment of steam explosions is their predicted large release of ruthenium. Such high ruthenium releases could significantly influence the source term. However, the RSS predicted release of ruthenium (because they are based on Parker's data) appears to require the presence of air to oxidize the nonvolatile ruthenium to the much more volatile forms, RuO$_4$ and RuO$_3$.

In a containment environment, the kinetic processes of the oxidation must also be considered. The kinetics observed in the ORNL experiments may not be applicable to debris lofted into the atmosphere by a steam explosion. Not only would the nature of the fuel (porosity, surface area, accessibility to the metallic ruthenium inclusions, etc.) under fuel/coolant interaction (FCI) conditions be different from Parker's experiments, so would the conditions for mass transfer of the oxidant to the surfaces. Further, if other reacting materials such as Zr and steel are present, not all of the oxygen reaching the surfaces is available for reaction to form volatile oxides of ruthenium.

Detailed kinetic models for oxidation/vaporization release during speculated FCI events have not been developed. It is believed that consideration of the kinetic processes involved would result in much lower releases than the WASH-1400 estimates.

Another source of airborne material from FCI events might be the mechanical breakup of the lofted debris itself. Data from a variety of steam explosion tests have shown that the debris formed by steam explosions is very coarse. Mean sizes are typically greater than 100 μm. Seldom is more than 1% of the material by weight of sizes less than 1 μm. Thus, Powers concludes:

"Aerosolization of debris is not likely to be an important aspect...[and]...the explosion of even large masses of debris...may not drastically affect aerosol inventories since the ability of coarse particles to scavenge particles of aerosol dimensions is limited."

In addition, any water accompanying the dissipated debris is likely to have the beneficial effect of attenuating the inventory of suspended radioactivity.

Finally, it should be noted that within the reactor vessel, steam explosions may create high velocities and surface accelerations necessary to resuspend aerosol particles that have been deposited onto surfaces. Quantitative
evaluation of this process is not available because of the limited understanding of, and database on, resuspension phenomena.

4.1.3.2.2 Conclusions. The following are the conclusions as abstracted from Powers' paper:

1. Ruthenium releases as a result of steam explosions are likely to be overestimated in the Reactor Safety Study.

2. Formation of aerosols by mechanical breakup may not be a large source of particulate material which can remain suspended for significant times.

3. Water droplets produced by steam explosions would tend to have beneficial mitigating effects.

4. Resuspension via steam explosion may be important and needs evaluating.

There is insufficient data on the probability of steam explosions, the mass of melt participating, the efficiency of energy conversion, and the kinetics of ruthenium oxidation/vaporization release to evaluate this potential but unlikely ex-vessel source of aerosols.

4.1.3.3 Blowdown Fragmentation of Water Droplets.

4.1.3.3.1 Technical Background. Due to very similar processes discussed under section 4.1.3.1 (High Pressure Ejection of Molten Fuel), liquid water expelled from the RCS during the blowdown and refill/reflood periods of a severe accident can be fragmented into small droplets by the effects of the shearing action, jet instabilities, and water vapor production/expansion processes. The blowdown process then constitutes a potential source of liquid water droplet aerosols into the containment for some severe accident sequences. If some substantive portion of these water aerosols remain airborne at later times when core melt released fission products and aerosols are introduced into the containment either directly from the RCS for in-vessel releases or from core/concrete interactions and other ex-vessel sources, then they could participate in and possibly significantly affect the subsequent aerosol dynamic behavior and, hence, the eventual source term.

Warman (Stone and Webster Engineering Corp. (SWEC)) reported on an analysis of the effects of such water aerosols as presented in the ANS study (30). This is the only known source term analysis in which the initial presence of blowdown created droplets is included. The SWEC analysis was based on

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studies (31,32) that have shown that significant amounts of water are ejected
during the blowdown (-112,000 kg) and that substantial amounts (as much as
1/2) of that is made airborne as droplets. A bimodal distribution of the
airborne droplets is predicted with one mode at -1.0 μm number mean dramatic
(NMD) and the other at -10 μm NMD.

Because of the uncertainty in the actual quantity of water that is made
airborne by the blowdown process, SWEC conducted a parametric evaluation
varying the amount in the 1.0 μm mode from 25% of the total mass ejected down
to 0 (their study also included variation of the amount in the 10 μm mode,
but they found that this larger aerosol settled quickly and had little
influence on the source term).

Using a postulated preexisting hole of 1.0 ft² area in the containment (found
to be the optimum for maximum release), SWEC calculated the effects of the
water droplets on the release using the NAUA code. The RCS and core/concrete
sources were taken directly from BMI-2104, vol. 1 for the Surry AB and TMLB'
sequences. The timing (from the start of the accident sequence) and the
ex-vessel sources for the parametric study is summarized below.

<table>
<thead>
<tr>
<th>Sequence</th>
<th>Water droplet injection (sec)</th>
<th>RCS core/melt source (sec)</th>
<th>Core concrete source (sec)</th>
</tr>
</thead>
<tbody>
<tr>
<td>AB</td>
<td>0-60</td>
<td>1,680-3,600</td>
<td>5,000-</td>
</tr>
<tr>
<td>TMLB</td>
<td>8,000-10,900</td>
<td>10,900-12,000</td>
<td>16,560-</td>
</tr>
</tbody>
</table>

4.1.3.3.2. Conclusions. The following conclusions summarize the SWEC
findings:

1. The water droplets in the 10 μm size mode had little effect
on the release from containment.

2. The timing of introduction of core/concrete aerosols is too
late to be substantially affected by the water droplets.

3. If only a small fraction of the water injected into the
containment (e.g., 10% for AB and 15% for TMLB') is ini-
tially airborne (at the AMMD size of 1.0 μm and σ = 1.7) in
the containment atmosphere, the enhanced agglomeration and
gravitational settling of the aerosol particles from the
RCS results in significant retention of the fission
products. For example, the Cs release is reduced by a
factor of 2 over the case with no airborne water droplets.
It appears, then, that source term evaluations should include the creation of airborne water droplets due to the initial blowdown as a potentially significant influence. In some sequences (e.g., TMLB'), the path of ejection of the water is somewhat ill-defined (i.e., through the pressurizer, relief valve, and quench tank) so that it may be difficult to estimate the quantity and size distribution of the water droplets made airborne.

Additional analyses and data are needed to be able to specify the initial size and quantity of airborne droplets and the timing with respect to other ex-vessel sources. Given these specifications, the present containment models should be adequate to evaluate the effects on source terms.

4.1.3.4 Resuspension from Boiling or Flashing of Water Pools.

4.1.3.4.1 Technical Background. Fission products that may have settled out within the containment prior to containment failure are presumed to reside to a great extent in pools of water.

These fission products can provide sufficient heat to maintain the water basically in thermodynamic equilibrium with the containment atmosphere. The depressurization that would accompany containment failure then would tend to boil or flash the water pools. The boiling/flash process is speculated to be able to entrain the residual liquid and perhaps make it airborne as small droplets carrying their burden of dissolved and suspended fission products to become a type of ex-vessel source that occurs simultaneously with vessel failure.

This postulated ex-vessel source has not been included in any source term analysis to date. However, the QUEST study (20) made a preliminary evaluation of the potential effects of this mechanism for the TMLB' sequence for Surry. This study indicated that the blowdown of accumulator water would result in a total quantity of 183 tons divided between two pools—28.3 tons in the refueling canal with a surface area of ~8 m² and the rest in the basement with a surface area of 982 m².

The QUEST study developed a model based on the following concepts:

- The rate of depressurization is given by choked flow if \( P > 1.83 \) atm and by orifice flow if \( P < 1.83 \) atm.
The evaporation rate from the water pools is given by the amount needed to take away the excess heat to keep the remaining (water left after steam and entrained liquid is lost) mass of water saturated as the containment pressure decreases.

The entrainment of liquid by the evaporating steam is calculated using Rosen's correlation as reported by Ginsberg (33) that correlates the entrainment with the superficial velocity of the vapor leaving the pool surface.

The size of the entrained droplets was based on satisfying a critical Weber Number criterion (ratio of inertial forces to surface tension forces). The droplet diameter, $D_c$, based on the critical Weber Number was assumed to be the mass geometric mean of a log-normal distribution with $s = 2.3$. If the $D_c$ based on the critical Weber Number is greater than a size, $D_s$, that would be suspended by the superficial flow velocity, then $D_s$ is used as the mass geometric mean instead of $D_c$.

Using models based on these concepts, the QUEST study parametrically varied the containment failure pressure and the break area, for a 500 m² surface area pool to arrive at the results shown in the table below:

### Entrained Water and Droplet Size

<table>
<thead>
<tr>
<th>$P_0$ (atm)</th>
<th>$A_0$ (m)</th>
<th>Initial pool mass ($10^3$ kg)</th>
<th>Entrained water mass (kg)</th>
<th>DGM (droplet size) (μm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>3</td>
<td>1</td>
<td>211</td>
<td>0.17-14.4</td>
<td>3-30</td>
</tr>
<tr>
<td>3</td>
<td>10</td>
<td>211</td>
<td>2.6-14.4</td>
<td>30-300</td>
</tr>
<tr>
<td>3</td>
<td>100</td>
<td>211</td>
<td>$6 \times 10^3$-$2.4 \times 10^4$</td>
<td>300-3000</td>
</tr>
<tr>
<td>3</td>
<td>1000</td>
<td>211</td>
<td>$5.0 \times 10^4$-$2.1 \times 10^5$</td>
<td>60-600</td>
</tr>
<tr>
<td>5</td>
<td>1</td>
<td>157</td>
<td>0.12-16.2</td>
<td>2-20</td>
</tr>
<tr>
<td>5</td>
<td>10</td>
<td>157</td>
<td>1.5-16.2</td>
<td>20-200</td>
</tr>
<tr>
<td>5</td>
<td>100</td>
<td>157</td>
<td>$1.9 \times 10^3$-$7.2 \times 10^3$</td>
<td>200-2000</td>
</tr>
<tr>
<td>5</td>
<td>1000</td>
<td>157</td>
<td>$5.3 \times 10^4$-$1.07 \times 10^5$</td>
<td>135-1350</td>
</tr>
<tr>
<td>7</td>
<td>1</td>
<td>105</td>
<td>0.1-13.4</td>
<td>1.2-12</td>
</tr>
<tr>
<td>7</td>
<td>10</td>
<td>105</td>
<td>0.07-13.4</td>
<td>12-120</td>
</tr>
<tr>
<td>7</td>
<td>100</td>
<td>105</td>
<td>270-1070</td>
<td>120-1200</td>
</tr>
<tr>
<td>7</td>
<td>1000</td>
<td>105</td>
<td>$4.1 \times 10^4$-$7.8 \times 10^4$</td>
<td>600-6000</td>
</tr>
</tbody>
</table>

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The variations in initial pool mass result because a fixed amount of water was assumed in the containment to be apportioned between the pool and the atmosphere. The higher pressures require water vapor from the pools as the source of pressurization.

The resulting ranges in entrained water mass reflect uncertainties related to the entrainment correlation used arising from uncertainties in the transition between flow regimes.

The range in the droplet mean sizes reflects the change in superficial velocity with time as the pool water becomes depleted.

In addition to the entrainment breakup of water droplets, the QUEST study made an assessment of the possible breakup into smaller sizes due to flashing fragmentation of the drops themselves. Flashing fragmentation would occur if the droplet temperature (and hence, saturation pressure) does not drop as quickly as the containment pressure. A simplified lumped parameter analysis for either convection or conduction controlled vaporization was made. The results indicated that droplet fragmentation due to flashing would not occur except for very large particles and should not be important.

4.1.3.4.2 Conclusions. The major conclusions to be drawn from the QUEST study of the resuspension of water due to the containment depressurization is that this mechanism requires a large hole (10 to 100 m²) to be important. The diameter of the resuspended aerosols tends to be large so that they may not necessarily survive the transport to the containment opening.

4.1.3.5 Reususpension From Containment Surfaces Via Hydrodynamic Forces. Aerosols that are calculated to contact surfaces within containment are generally assumed to be permanently removed and are no longer available as sources of release to the environment. However, several mechanisms have been identified that are speculated to have the potential to resuspend such deposited material:

1. The depressurization of containment on failure at high pressure could give rise to local flow velocities past surface deposits that could resuspend particles due to shear and lift forces.

2. Depressurization of "deep" deposits of aerosols could lag the containment depressurization giving rise to a pressure differential from the depth of the deposit to the surface creating normal flow that could lift and reentrain the aerosols.

3. The depressurization could result in flashing of liquid pools that had become repositories for fission products and aerosols.

4. H₂ explosions could create shock fronts that interact with surface deposits to resuspend them. In addition, the shock front could fragment airborne particles into smaller sizes or the higher temperature could revaporize volatile species either from materials still airborne or deposited.

4.1.3.5.1 Technical Background. Item 4 has not been evaluated and its likelihood or importance is unknown. Item 3 was discussed under section 4.1.3.4.
Item 2 was evaluated within the IDCOR Program. For very large containment hole sizes of $-10 \, \text{m}^2$ (rapid depressurization), the resulting induced flow velocities from deposit beds 3 mm deep (greater than the maximum depth anticipated on containment horizontal surfaces) was at least an order of magnitude too low to levitate a packed bed of 10 μm diameter spheres. Therefore, this is not believed to be an important resuspension mechanism.

Item 1 was analyzed by the QUEST study (20). The containment depressurization gives rise to flow velocities that depend on the hole size. Lift and drag forces compete against adhesion forces for making particles deposited onto surfaces airborne. For wet deposits, surface tension adhesive forces are so strong that depressurization velocities are orders-of-magnitude too low for resuspension. Therefore, for deposited aerosols to be resuspended by this mechanism, the surface deposit must be dry. It is estimated that -25% of the internal containment surface consists of sheet steel (thin and with relatively low heat capacity per unit area). If it is assumed that these are dry surfaces, depressurization velocities for about a 15 m$^2$ hole size were estimated to be sufficiently high to overcome the estimated intermolecular adhesion forces for 10 μm particles. For 25 μm particles, the required hole size decreases to -8 m$^2$.

4.1.3.5.2 Conclusions. The major conclusion to be drawn from the QUEST and the IDCOR studies is that this postulated resuspension mechanism is not likely to be important. For any significant resuspension, the containment failure hole area must be very large.

However, the analyses were all made for large dry containment volumes. It is not clear whether or not smaller Mark I and Mark II BWR containments might have resultant depressurization flow velocities that are large enough to resuspend deposited material. The analyses need to be extended to these containment types.

There have been experiments that validate the models for resuspension due to the lag in depressurization of deposited beds (the IDCOR analysis). However, there has not been sufficient experimental validation of the QUEST analyses for shear and lift resuspension.
4.1.4 REFERENCES


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27. D. A. Powers. The Solubility of Gases in Reactor Core Melts, Sandia National Laboratories, Albuquerque, NM.


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Additional information can be found in the references listed below:


4.2 EX-VESSEL THERMAL HYDRAULICS

4.2.1 Technical Background

Containments are designed for loads resulting from loss-of-coolant accidents (LOCA). Most analyses of containment performance predict that they can cope with much higher loads than they are designed for, before containment integrity (leak-tightness) may become an issue.

Risk studies and the accident in TMI-2 have shown the importance of containment integrity for risk evaluations.

The thermal-hydraulic behavior within containments (pressure and temperature history, internal flows) can be divided into "short-term" and "long-term" containment behavior. "Short-term" usually characterizes the behavior within the first few hours after an accident or release of melt from the RPV. Thermal hydraulics within the containment is important for fission product retention.

Thermal hydraulics within a containment is very much affected by containment design, accident sequence and possible accident management actions.
The source term depends on the thermal-hydraulic conditions that prevailed in the containment till the time of release from the containment. The most important parameter is the time elapsed since the release of fission products from the primary system to containment failure, since it has been observed that the predominant physical form of fission products is aerosols, whose airborne concentration decreases greatly with time. The characteristic elapsed time of interest is several hours, since it has been found that generally in four or more hours airborne concentration may decrease by several orders of magnitude. Thus, thermal-hydraulic phenomena and events that may bring about an early (for example, <4 hours) failure of containment are of prime interest and have received the most attention. These include hydrogen combustion, direct heating, and steam spikes, each of which can result in high thermal and pressure loads on a containment shell in a short time span. In addition, for the BWR, the phenomena leading to early containment failure include (a) high heat deposition in the suppression pool and (b) high temperatures reached in the containment due to radiation from melt discharged into the drywell.

It must be recalled here that fission products are introduced into containment at several occasions during the accident progression, e.g., a fraction may enter containment during core heat-up, another during core/concrete interaction, and yet another when the reactor coolant system (RCS) becomes hot enough to revolatilize the deposited fission products. Thus, to ensure a low value of the source term, it is imperative to maintain containment integrity for several hours after each release of fission products to the containment. This holds true for both PWRs and BWRs. For BWRs, however, the suppression pool acts not only as an energy sink, but also as a scrubber for fission products. Thus, as long as the fission products are routed through the suppression pool as, for example, in containment venting scenarios, a large fraction of the fission products will be retained in the suppression pool. The scrubbing of fission products in BWR suppression pools is treated in another technical annex in this report.

Another scenario, for which containment thermal hydraulic conditions have to be predicted, is a preexisting opening in the containment. This scenario, although very unlikely, may not provide the time needed for depletion of the airborne source in the containment. It is important to know the spatial distribution in the multicompartmented containment of the fission product aerosols and the flow through the vent in order to estimate the release to the environment.
Containment bypass scenarios, such as the V sequence in a PWR and the pool bypass sequence in a BWR, have been identified as possibly leading to relatively large releases to the environment. These require prediction of thermal-hydraulic conditions in the auxiliary and reactor buildings, which may differ in size and layout from plant to plant. For the PWR V sequence, it may be necessary to determine flow patterns in the composite system of containment, reactor vessel and ECC piping in order to estimate the transfer of fission products from the containment building to the auxiliary building. The BWR Mark I and II pool bypass flows, transporting fission products to the reactor building, depend upon the drywell failure location, and the geometry of the reactor buildings surrounding the drywell. The calculations for thermal-hydraulic conditions have to consider plant-specific features, e.g., the standby gas treatment system (SGTS) in BWR buildings, water pools in PWR auxiliary buildings, spray systems, etc.

Resuspension of fission products is of some concern in the event of depressurization induced by sudden containment failure. The flow velocities, that may occur in the neighborhood of the failure location, may suspend some of the aerosols deposited on the walls, floors or other surfaces nearby. The flash of any accumulated water in the containments may also introduce fission product bearing droplets in the air space, which may be carried to the failure location.

A containment will eventually fail if no heat is removed from it. The longer-term flow fields that prevail in the multicompartmented containment are governed primarily by the heat and mass sources and sinks. In many scenarios, the containment cavity may contain a large part of the molten corium ejected out of the reactor pressure vessel (RPV), reacting with water and/or concrete and producing steam, hydrogen, CO and fission products. The flow patterns are also affected by occurrences of hydrogen and/or CO combustion. The rising pressure may lead to a sudden and large failure (possible for steel containments), or a leakage (probable for concrete containments), which may balance the mass and energy input from the longer-term corium/concrete interaction. Flow fields generated by these events will determine the release of the fission products from the containment to the environment.

Summarizing, the thermal-hydraulic phenomena relating to fission product behavior in the containment can be divided into two categories. The first category relates to early containment failure (which may lead to high releases to the environment) and the second category relates to later containment failure. The first category includes sudden events, e.g., corium jet ejection, steam spike, hydrogen...
combustion, direct containment heating etc., during which the thermal-hydraulic conditions may be difficult to determine precisely and only bounding estimates could be defended. The second category includes natural convection flows, bulk condensation (in presence of noncondensables), spray activation, water droplet formation, corium/concrete interactions, leakage flows, intercompartmental flows, stratification, etc.

4.2.2 Thermal-Hydraulic Phenomena Causing Early Containment Failure

In this section, a brief report will be given on thermal hydraulic phenomena which can cause early containment failure. If it can be shown that the conditional probability of early containment failure is very low ($10^{-2}$ to $10^{-4}$), the severe accident risk would be very small indeed.

4.2.2.1 Principal Areas of Agreement/Disagreement. There is a sufficient body of literature on the various thermal-hydraulic phenomena identified as possible causes of early containment failure. The IDCOR work was primarily directed towards establishing the very low probability of early containment failure. The study considered each phenomenon separately and established best-estimate pressure and temperature histories in risk-dominant scenarios for four reference plant containments. The study found that steam spike and hydrogen combustion (with installed igniter systems in Mark III and ice condenser containments) do not lead to early containment failure. The BWR Mark I and Mark II containments may be prone to early failure in ATWS scenarios due to the high heat deposition in the suppression pool. This depends on the power level, which in turn depends on the water level in the core during the ATWS scenario. Detailed analyses of ATWS scenarios is currently in progress at EPRI, Brookhaven National Laboratory, and General Electric Company.

The USNRC instituted the Containment Loads Working Group (CLWG) and posed standard problems (based on certain risk-dominant scenarios) for the four containment types mentioned in the above paragraph. The results of their work confirmed the IDCOR findings for the Mark III, Mark I and PWR dry containments regarding steam spikes and hydrogen combustion. The ice condenser containment was identified as possibly vulnerable to pressures generated during possibly large-scale hydrogen deflagrations.

Direct-containment-heating was identified by Sandia Laboratories as a possible cause of early containment failure in a PWR. It may occur due to aerosolization.
and dispersal of material ejected at high pressure from the RPV into the containment. A difference of opinion exists between the NRC and EPRI (based on the work sponsored by each) on the estimated fraction of the energy transferred directly to the containment atmosphere. The EPRI-sponsored experiments and analyses at Argonne National Laboratory (ANL) estimated that 5% of the energy content of the corium debris could be transferred directly, while Sandia work suggest much higher values. There are several considerations which need further experimentation and analyses. These include the extent of the oxidation and size of the dispersed particles, hydrogen generation and recombination and the flow paths of aerosolized particles to the containment compartments. Further work is planned at both Sandia and ANL.

4.2.2.2 Principal Areas of Sufficient/Insufficient Information. So far, the early containment failure phenomena have been considered on a best-estimate basis (with some use of bounding assumptions, e.g., while core melt delivered to containment) by the IDCOR, EPRI and NRC-sponsored work for risk-dominant scenarios. Since the phenomena are complicated, improved or best-estimate treatments need more data from well-designed and pertinent experiments.

Most of the loadings calculated depend critically on the specific features of the containments considered, e.g., the Zion containment cavity and tunnel promotes dispersal of ejected material to containment compartments, while the Sequoyah containment cavity of similar design, but with much different dimensions for the tunnel may collect most of the ejected material at the lip in the tunnel. Similarly, the presence of a pedestal or a skirt around the drywell floor can change the geometry and disposition of the dispersed melt. Presently, there is sufficient information on thermal-hydraulic processes occurring in some of the containment designs, to lend confidence to the predictions made. Other designs need further study.

Direct containment heating models have shown large differences. The information base is not sufficient to resolve the differences.

4.2.2.3 Generic Applicability to LWR Plants. Some of the bounding models advanced for calculation of the pressur and temperature histories may be applicable to the various LWR plants. However, in general it is prudent to apply these models carefully, since differences between containment geometries are large. Therefore, thermal-hydraulic processes may differ considerably from one containment to another.
4.2.3 Thermal-Hydraulic Phenomena Relating to Late Containment Failure (Low Fission Product Release to Environment)

In this section, those thermal-hydraulic phenomena leading to late containment failure are addressed, which become operative during the many hours that a core debris may be interacting with water and concrete present in the containment. During this time, depending on the scenario, the engineered safeguard systems may be brought into play, e.g., hydrogen igniters, containment spray, fan coolers, etc., which will affect, not only the thermal-hydraulic conditions, but also the airborne fission product concentrations and chemistry directly.

4.2.3.1 Principal Areas of Agreement/Disagreement. Presently, only limited comparisons of the methods employed, and results obtained for the longer-term (several hours) thermal-hydraulic conditions, in PWR and BWR containments have been made. The summary results generally reported are the pressure and temperature in one or two regions of a containment. The analyses reported in BMI-2104 have employed very small numbers of regions in each type of containment; the IDCOR MAAP code analyses have employed a somewhat larger number of regions. In both analyses, the pressure and temperature histories are determined by a balance of mass and energy sources and sinks. The time history of mass and energy sources and sinks employed by the BMI-2104 study and IDCOR study are obviously different from each other. One outstanding difference is in the magnitude of radiated energy and gas mass flow rate emanating from the top surface of the corium melt (reacting with concrete) into the containment. The IDCOR results for Mark I drywell temperatures and pressure differ considerably from those predicted by the CLWG, who employed CORCON and a MARCH-type containment analysis code. This affects not only the failure time of a Mark I drywell but also the revolatilization rate of the fission products deposited earlier in the drywell, which causes their migration to the reactor building. The calculated magnitudes of the fission products deposited in a Mark I drywell and reactor building are, therefore, quite different in the two studies.

Differences in the hydrogen combustion energy-source are also evident in the various treatments, since (1) different magnitudes and rates of hydrogen are released in the scenarios and (2) somewhat different models for hydrogen combustion. The MAAP, in general, code obtains lower temperatures and pressures for hydrogen combustion in the containment atmospheres than those in the BMI-2104 analyses. There may also be differences in the treatment of the various energy sinks, e.g., sprays, fan coolers, bulk condensation and endothermic chemical
reactions. A systematic study of such differences between the two studies has not been completed.

The various studies have employed substantially different treatments with respect to representation of containment compartmentalization, or the flows between different containment compartments. The MAAP code has included the modeling for natural convection flows that may prevail between compartments. The BMI-2104 analyses have not employed such intercompartmental flow models. The MAAP analyses have, therefore, obtained different concentrations of steam, hydrogen, CO and fission products in containment regions than those obtained in BMI-2104 analyses. An outstanding difference, for example, is the hydrogen concentration in the dome of the ice condenser containment during the TMLB1 accident. Unlike the BMI-2104 analysis, the MAAP code results show that a large fraction of the upper dome hydrogen is convected down to the containment cavity, where it can burn. This difference affects, not only the estimates regarding the integrity of the ice condenser containments, but also the fission product distribution and deposition in the ice condenser containment. Similar differences may be found for other containments.

4.2.3.2 Principal Areas of Sufficient/Insufficient Information. The information base needs augmentation, since various investigators have had to employ different models for some of the key thermal-hydraulic phenomena. The major sources of energy and mass to the containment are from the corium/concrete interaction, and here there is a substantial lack of knowledge; the situation hopefully may be improved, with the growing database from BETA, SANDIA and ANL experiments.

The other major energy source, i.e., hydrogen combustion, has a larger experimental database. However, it appears that models employed in the named studies are not sophisticated enough to cover the various combustion modes that may occur, and influence the pressure, temperature and chemistry in the containment.

The sinks for bulk condensation such as cold structures, containment sprays, PWR fan coolers, BWR suppression pools, etc., are well-characterized for the conditions that might prevail following a large LOCA. However, degraded core accidents produce large quantities of hydrogen in containment that affect the condensation process, which is the main mechanism for reducing containment pressure-and-for-aerosol-deposition through diffusiophoresis. Similarly, the scrubbing effect of sprays and ice condensers on airborne fission products in the

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presence of large fractions of hydrogen and other noncondensables has not received much experimental attention. Models used to describe such processes in the presence of noncondensables may need to be reexamined.

4.2.3.3 Generic Applicability to LWR Plants. The phenomenological modeling of the time evolution of the energy and mass sources and sinks should be applicable to every LWR containment. However, there are such large configurational differences between various containment types, and between individual containments of the same type, that it is mandatory to be extremely careful in such generic applications. Compartmentalization also plays a large role in determining the distribution of mass, energy, gas concentrations and temperatures within the containment, since it determines flow paths, flow resistances, flow patterns and velocities. Modeling of this varies considerably from one containment design to another. It is expected, however, that the codes can be applied to most containment types if sufficient care is exercised.

4.2.4 Thermal Hydraulic Phenomena Relating to Containment Failure and to a PreExisting Opening (Possibly Large Fission Product Release to the Environment)

In this section, the thermal-hydraulic phenomena which determine the fission product release from a preexisting opening in the containment are discussed.

4.2.4.1 Principal Areas of Agreement/Disagreement. The one study that has systematically considered the release of fission products from a preexisting opening is that of Stone & Webster whose findings have been reported in the ANSI source term study. Additional studies that have considered preexisting openings in containments have been reported by Battelle Columbus. No systematic comparison of the results obtained in those studies has been performed.

4.2.4.2 Principal Areas of Sufficient/Insufficient Information. The calculations of thermal-hydraulic conditions in a containment with a leak path do not pose any problems for the formalism of mass and energy balance equations. The areas of insufficient information, as indicated in the last section for the mass and energy sources and sinks within the containment, during the long-term progression of the accident (there is no early failure of the containment in this case), apply here as well.

Another area of insufficient information relates to resuspension of deposited fission products due to the flow fields generated in the vicinity of the failure or a preexisting opening. It appears, that it would be difficult to resuspend...
fission products salts (CsOH, CsI, etc.) that were dissolved in water. However, more data may be needed with regards to resuspension following dryout of fission product deposits in the vicinity of the opening.

Containment failure due to excessive pressure and temperature loads can take the form of big openings (probably in free-standing steel containments) or leakage paths (probably in concrete containments and possibly in steel-lined concrete containments). Leakage flows, containing fission product aerosols, may experience attenuations, if the leak paths have large L/D ratios and convoluted pathways. There is a partial information base available; however, a systematic study of flow patterns and possible reduction of aerosol densities in accident scenarios has not been completed.

4.2.4.3 Generic Applicability to LWR Plants. The methodology in place is generally applicable to most LWR containment designs. The specific features of each containment, e.g., flow paths, flow resistances to the failure or opening location have to be considered individually.

4.2.5 Thermal-Hydraulics Phenomena Relating to Containment Bypass

In this section, the thermal-hydraulic phenomena that relate to the containment bypass scenarios are addressed. These include the PWR V sequence and the BWR suppression pool bypass scenarios.

4.2.5.1 Principal Areas of Agreement/Disagreement. A more systematic comparison of the methodology employed and the results obtained by various studies on the PWR V sequence needs to be made. The BMI-2104 and EPRI study evaluated the fission product release for Surry V sequence, while the IDCOR study considered a bypass sequence for the Zion Plant, whose timing and geometry are much different from that for the Surry plant.

The EPRI and BMI-2104 studies are quite similar in the methodologies employed for the thermal-hydraulic and fission product transport analyses. The auxiliary building is of relatively small volume, however, the water accumulation could scrub the released fission products. These studies considered some decontamination by the water present. The EPRI study also considered an opening in the auxiliary building, and used the measured values of decontamination factor from the LACE experiments.

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The suppression pool bypass scenario for Mark I BWRs has been treated by both the IDCOR and BMI studies. The treatments are relatively simple, and the flows are primarily based on (1) the leakage areas assumed, (2) the calculated pressure and temperature histories in the drywell, and (3) the operation or nonoperation of the standby gas treatment systems. Thus, there is substantial agreement in the approach taken by most investigators.

The IDCOR study has calculated the thermal-hydraulic conditions and the aerosol deposition and transport through BWR Mark I and II containment buildings and through the Zion auxiliary building. The BMI-2104 analyses have found very little fission product retention in the Mark I and Mark II reactor buildings. The IDCOR analyses, on the other hand, have calculated very significant retention of fission products in the Mark I and Mark II reactor buildings.

4.2.5.2 Principal Areas of Sufficient/Insufficient Information. The calculation of flows in containment or pool bypass scenarios do not need any additional methodology developments. However, for the PWR bypass flow rates, coupled with the transport of liquid-salt type fission products, which may have deposited on primary system and ECCS piping surfaces, it becomes necessary to consider the interaction of high velocity flows with liquid films and/or droplets. The entrainment of films has been investigated in fluid mechanics studies, and there is a considerable body of literature available; however, the specific cases of entrainment from films containing structural and fission product aerosol particles may need an experimental study.

The temperature and flow distributions in auxiliary and reactor buildings can be used for determining the rates of deposition of the fission product-bearing aerosols in such buildings. The development of transient thermalhydraulic conditions, however, depends upon the heat sources from the deposited fission products. Such evaluations have not been performed yet and the required input information, e.g., heat loss to buildings, has not been generated.

4.2.5.3 Generic Applicability to LWRs. The methodology required for evaluation of thermal-hydraulic distributions for containment bypass scenarios can be applied generally to all LWRs. However, the physical arrangements and local flow paths can be very different from one containment to another. Thus, generic application is possible but has to be done with care.
4.2.6 Conclusions and Recommendations

Thermal-hydraulic conditions in the containment and auxiliary buildings, for late containment failure scenarios, are perhaps of most interest, since the other scenarios (early failure, bypass, existing openings) are of relatively much lower probability. In this respect, perhaps the major source of energy and mass addition to the containment is the corium/concrete interaction. It is necessary to determine the split between the energy transferred to containment, versus that transferred to concrete, during the course of such interactions. Similarly, the major energy sink is water evaporation and its subsequent condensation on the walls, floors and other surfaces of the containment buildings.

Steam condensation can be affected by the copious amounts of noncondensables and aerosols generated during the core/concrete interactions. It is recommended that the effects of corium/concrete interactions on containment thermal-hydraulic conditions be investigated further, both experimentally and analytically.

The role of containment compartmentalization on thermal hydraulic distributions and on airborne aerosol distributions, as a function of time and space, within the containment has not been adequately examined. It is potentially very important with respect to deposition rates of aerosols. The natural convection flows, that might occur and homogenize the aerosol concentrations, should be modeled. Experimental information to benchmark the models should be obtained. It is also necessary to consider the relatively stable stratifications that might occur in a containment in some scenarios which would affect the distributions of aerosols in the containment. Modeling to treat such situations should be a part of any containment code, i.e., a flow equation with buoyancy-induced forces should be used.

Much progress has been made during the last few years. The research for global long-term containment behavior is nearly finished. A few physical phenomena, e.g., bulk condensation, internal flows, and energy splits in the corium/concrete interaction, have to be studied further.

For short-term containment behavior, the ongoing efforts to study melt dispersal and melt/atmosphere interactions should be continued.
4.3 FISSION PRODUCT/AEROSOL TRANSPORT IN CONTAINMENT AND OTHER BUILDINGS

4.3.1 Introduction

Section 4.4 is concerned with the methodologies for predicting radioactivity transport by natural processes in the containment and other buildings of an LWR during the first few hours or days after a severe core damage accident, up to and including containment failure, should this occur, and also with the results obtained by use of these methodologies. In pursuance of the terms of reference of the Task Force the aim has been to review the information which has become available in a number of recent studies, compare the different treatments, assess the completeness of the information, and consider its application to different types of LWR plant.

The studies which have been reviewed are principally of U.S. origin, or closely follow U.S. approaches to source term evaluation. They differ significantly both in purpose and content, and also in the computer codes which form the basis of the methodologies they refer. A brief summary of the reports, as they pertain to this section, follows:

4.3.1.1 USNRC Studies. The USNRC studies are calculations of radionuclide release under specific accident conditions, using the BMI suite of codes (NAUA-4 in the case of aerosol transport in buildings) (1). There are supporting studies of the status of validation of the codes (2), and of the uncertainties associated with the results (3,4).

4.3.1.2 Stone and Webster Studies. The Stone and Webster studies are essentially supplemented (1) by parametric studies of certain additional factors which influence the radioactive release (5).

4.3.1.3 The APS Studies (6). The APS report is the latest of the USNRC studies and its purpose is to review the adequacy of the technical base upon which the phenomenological models for radionuclide release from postulated severe accidents are constructed, the adequacy of the models themselves, and the correct use of the complex computer codes that incorporate these models in the analysis of accident sequences.

4.3.1.4 ANS Study (7). The ANS report both reviews and evaluates the state of knowledge of how to predict a source term deriving from a severe core damage accident and summarizes the results of specific calculations prepared by several organizations (including those in the other studies referred to in this report).

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4.3.1.5 IDCOR Studies (8). The objective of the IDCOR work is to develop a technically sound position on issues relating to severe accidents and to provide the basis for (U.S.) industry participation in NRC decision processes. Source term studies form part of the whole and both an evaluation of the state of knowledge and specific calculations are included. The latter are based on a Modular Accident Analysis Program (MAAP).

4.3.1.6 EPRI Studies (9). The objective of this particular EPRI effort is to update the WASH 1400 source term for SURRY, using current methods, and to show the effect this has on the calculated public risk. Aerosol transport in buildings is evaluated using the MATADOR code.

Taken together these studies cover a wide range of accident sequences in all the major classes of LWR, i.e., large, dry and ice condenser PWRs, and Marks I, II and III BWRs.

4.3.2 Technical Background

The period of a severe LWR accident which is of concern in this section usually begins with a discharge of coolant into the reactor containment from either a break in the RCS or the pressurizer quench tank. Generally speaking, the discharge will include both steam and a particulate water aerosol. It will be accompanied, or followed, by further discharges of noble gas fission products and aerosols of other fission products and core-derived materials. At later stages in the accident, additional aerosols may be generated by the discharge of the molten core through the RPV bottom head, when this occurs while the RCS is still at pressure, and by the subsequent interaction of the hot core materials with the concrete of the containment basement. In this latter case, the aerosols generated may also include constituents of concrete. In the longer term, there may be a continuing generation of small amounts of aerosol from the hot pool in the containment sump. At some point the containment building may suffer overpressure failure.

This general scenario will be different in the case of containment bypass accidents in that the initial discharge of coolant and core materials at the beginning of the accident will be into an auxiliary building.

This section is concerned only with the transport of the fuel and fission product aerosols through the containment and other buildings to the environment and
...factors which affect this transport. It is not concerned with the processes referred to by which the aerosols are generated. It includes the question of relatively short-term resuspension of radioactive materials which have deposited in the containment; however, it excludes the question of the longer-term chemical interactions within the containment leading to the generation of volatile fission product species. It also excludes the removal of radioactivity by ESFs in the containment and auxiliary buildings. The only significant change which the noble gas fission products will undergo in the containment and other reactor buildings is radioactive decay. With the exception of the effects of hydrogen combustion, which is dealt with elsewhere, short-term chemical interactions affecting the chemical form of the other fission products and fuel are probably of little importance. Hence, the principal phenomena with which this section has to be concerned are the following:

- **Within the building volumes**
  - steam/condensation evaporation on aerosols. This will occur as heat is lost to the building walls and other heat sinks, or possibly when the containment atmosphere is cooled by expansion on containment failure, and will alter both the particle size distribution and particle shapes of the aerosol.
  - aerosol agglomeration.
  - aerosol deposition on floors and other surfaces.
  - resuspension of deposited materials as a result of steam explosions, hydrogen combustion, or rapid depressurization of the containment.

- **Within leak paths in building walls**
  - aerosol agglomeration.
  - aerosol deposition.
  - aerosol resuspension

Superimposed on all of these phenomena is the question of radioactive decay and the possible self-heating of the aerosol which might result. This is thought to be a second-order effect but confirmatory calculations are needed.

In principle the aim should be to predict the time history of the composition, concentration and particle size distributions for the radioactive aerosols existing at appropriate points in the containment and other buildings, and those released to the environment.

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4.3.3 Areas Where Studies Agree and Disagree

4.3.3.1 Methodology: The calculation of aerosol transport in the containment entails the use of a range of physical models and numerical approximation schemes. In this section, these are reviewed as they appear in the various studies. Firstly, the modeling requirements are briefly described to provide a framework for the subsequent discussion. Then critiques of aerosol modeling, that is discussions of how aerosols should be modeled, are reviewed. This in turn breaks down into two headings: what processes need or need not be modeled, and how these processes should or should not be treated. Finally, the methods which are actually used in the calculations reported in the documents are considered.

4.3.3.1.1 Modeling Requirements.

Removal Processes. The main objective of containment aerosol modeling is to calculate how fast aerosol particles are removed from the containment atmosphere. This calculation is complicated by the occurrence of particle growth processes (here "growth" is taken to include negative growth, such as when water evaporates from particles). The removal rate as a function of radius shows a pronounced dip between 0.2 and 2 μm, which accounts for the stability of aerosols. Brownian diffusion causes rapid removal of particles smaller than this, but for dense aerosols, such as occur in in-reactor nuclear safety applications, this process is in competition with rapid Brownian agglomeration, which shifts the particles into a size range where Brownian removal is no longer effective, and as a result, this form of removal is usually of minor importance. Once particles get above this range gravitational settling begins to become rapid. In most containment applications, the removal rate is dominated by how fast particles can grow to a size at which settling becomes rapid.

This picture is modified by the phoretic removal processes, which operate when there are substantial heat or mass flows to the walls, and whose rates depend only weakly on particle size. In LWR containments, thermal gradients at the walls are usually small (because heat can be transported as latent heat with the condensing steam), so thermophoresis is generally of minor importance. Diffusiophoresis is a potentially important process for early removal, but for it to be so the episodes of significant steam condensation on the walls must coincide with times when the aerosol is present in the containment.
Growth Processes. In aerosols with mean radii between 0.1 and 1 μm and with mass concentrations greater than about 1 g/m³, agglomeration is an important mechanism of particle growth. There is, however, a dip in the agglomeration rate at approximately the same place as the dip in removal rates. Below the dip Brownian motion promotes agglomeration, while above gravitational and turbulent mechanisms become increasingly important. For a size distribution starting near its most stable position, it typically takes of the order of two hours or less for a significant number of particles to agglomerate to sizes where settling becomes rapid on the time scale of an hour.

When the thermodynamic conditions are such as to allow condensation of steam onto the particles, growth is much faster, because the additional mass is applied directly to the particles. However, like diffusiophoresis, this depends critically on the coincidence between aerosols and the right thermodynamic conditions. Generally, steam-based removal is fast but is very sensitive to the thermal-hydraulics, while the gravity-based processes are slower but more reliable, being largely indifferent to the thermal-hydraulics.

Complicating Factors. The early generations of aerosol codes shared the following simplifying assumptions:

- The containment volume is well mixed, that is to say, the number concentration distribution is the same everywhere except in the boundary layers next to surfaces.
- The aerosol components of interest (e.g., radionuclides) are well mixed over the size distribution, that is the amount in any size range is proportional to the total mass in that range.
- Once aerosol material is deposited it cannot become airborne again.

The first approximation has been removed for the situation where there are physically separate compartments connected by well-defined leak paths. Codes based on several well-mixed volumes with transport between them have been developed. These would be applicable where there is spatial variation of aerosol concentration within a single volume provided intracell flows were known. Codes have also been developed which follow the distribution of different components over the size range separately, so it should now be possible to investigate how important doing this is in the calculation of the radiological hazard of a release. The failure of the third assumption,
namely resuspension, is difficult to model, but it is important at least to scope the effect because it is the major reason to claiming substantial containment retention in late failure sequences.

Numerical Methods. Even when the physical models are agreed upon, it is still necessary to choose numerical methods for the approximate solution of the aerosol equations. Two methods are in general use: the method of moments which requires that the distribution be always lognormal, and methods based on dividing the size range up into finite intervals, over which the distribution is required to be constant or linear (discretization methods). The method of moments is simpler but can introduce large errors in certain circumstances.

Auxiliary Building and Leak Path Modeling. If the damage state of, and therefore leak path through, the auxiliary building in bypass sequences are known, then the aerosol transport there can be calculated using multi-compartment codes. This assumes that the flow through the building is slow enough to make the mixing time within each compartment shorter than the mean residence time. Aerosol retention in leak paths (e.g., capillaries, cracks, failed seals) requires different techniques. An empirical correlation due to Vaughan and Morewitz exists which predicts the amount of aerosol leaked before deposits plug the pathway, but this does not say whether the gas flow is impeded, or how the plug responds to a buildup of pressure. The main obstacle to predicting decontamination in leak paths is, however, the uncertainty in the nature of the leak paths in the first place. This is another example where the modeling of aerosols is ahead of the modeling of the boundary conditions of the aerosol problems.

4.3.3.1.2 Critical Review of Aerosol Modeling. The aim of this section was to compare what the studies under review say in the way of description and critical commentary on how aerosol processes should be modeled. However, of the documents considered here, only the APS study contains the level of detail needed to warrant the description "critical," and even there, in many cases, the treatment is only descriptive. The discussion in this section is, therefore, heavily weighted toward the APS document.

For the processes of removal and agglomeration, there is wide agreement between the various codes on both the individual models and on the assumption that the rates for the different processes can simply be added. The ANS and
APS studies appear to endorse this consensus (or at least in the latter case fail to criticize it). In its discussion of diffusiophoresis, however, the APS study mentions only the method based on steam concentration gradients, which needs an assumption about a boundary layer thickness, and fails to note the more recent development of the mass flux (10, 11, 12), which makes more direct contact with the output of the thermal-hydraulic codes. The study recognizes the importance of gravitational and turbulent agglomeration and notes that the models used still need individual validation. The call for small-scale separate effects experiments to provide such validation made in the recent CSNI (13) and CEC (14) reports is seconded in the APS report.

On the question of condensation and evaporation, the ANS and APS studies describe the model in the TRAP-MELT code. This is based on mass transfer alone. Although the APS study also mentions the possible role of latent heat, it does not go on to describe the Mason equation, which includes both heat and mass transfer effects, and which is used in one form or another in most containment aerosol codes for steam condensation and evaporation. Recent calculations have shown (16) that heat transfer effects play a major part in determining whether vapors condense onto aerosol particles or onto walls, and that therefore, the TRAP-MELT method is unreliable. Furthermore, as used for RCS applications TRAP-MELT calculates the condensation rate itself, whereas the basic information on steam condensation in containment comes from the thermal-hydraulic codes. And finally, the TRAP-MELT method does not take account of the addition of mass to the aerosol particle, which in the containment is precisely what is of interest.

The APS report identifies the well-mixed volume assumption as a possible weak point in the current aerosol modeling. It notes early DEMONA results wherein a considerable degree of stratification is reported. Given the appropriate input, it is simple in principle to include such effects in aerosol codes (although at the cost of increasing run times); the problem is mainly the thermal-hydraulic one of calculating intracell flows. Whether the containment sprays operate or not is likely to be one important factor amongst others here. The report also identifies multicomponent modeling and resuspension as areas in which further work is needed. The existence of the multicomponent code, MAEROS, is mentioned, but there is no discussion of the method underlying it (reviewed in the CEC report (14). Under the heading of numerical methods, the APS report states that "there appears to have been relatively little cross comparison between the various codes." It reviews
the results of blind calculations for the ABCOVE AB5 test; and notes there are orders of magnitude differences in predictions for airborne masses at late times, even between the same code with supposedly the same input run at different laboratories. The report concludes, "A careful, fully documented, intercomparison of the various codes is strongly warranted." For dry aerosol codes, such a study has recently been published by the CEC (14). Not only were the numerical methods compared in considerable detail, but the codes were run on a benchmark problem. Care was taken to ensure that both input data sets and physical models were the same. When this was done, very good agreement was achieved between discrete codes, even though there were still differences in numerical methods. The one log normal code in the study diverged from the rest, especially at the end of the source. The study recommended that log normal codes be abandoned. If the results of the log normal codes are removed from the AB5 comparison then the remaining results agree well both with each other and with the experiment.

The APS report does not comment on modeling of aerosols in auxiliary buildings or in leak paths. The general recommendations of the APS report for further aerosol work are good and in agreement with the recommendations of the CSNI and CEC reports. However in the discussions of some of the models the APS report is behind the current state of the art, for a better appreciation of which one should consult the two reports just mentioned and also the papers given to the recent CSNI aerosol experts meeting at Karlsruhe (15). A code comparison for LWR containment aerosols, organized by the CSNI, is currently underway.

4.3.3.1.3 Description of Models Used in Calculations. This section is intended as a comparison of the models used in the actual calculations, as described in the respective studies under review. To indicate the difficulty in making such a comparison, the words of the APS report are best repeated.

"The examination of the codes by the APS study group has been hampered by the poor quality of the documentation of their theoretical bases, of their validation against experiment, and of the codes themselves."

This point cannot be endorsed too strongly. Source term calculations must be accompanied by either a detailed mathematical description of the models and numerical methods used, or a reference to such a description. All of the reports are deficient in this respect. A verbal description of the processes
allegedly included in the codes followed by a statement of results do not suffice to place the results on a sound scientific footing.

This having been said, the models for agglomeration and for removal processes other than diffusiophoresis have, as noted in section 4.3.2.1.2, been around for a long time and are reasonably well-documented (for example in the CEC report (14)). In the main, containment aerosol codes agree on these "standard" models. The BMI and SWEC studies use the NAUA-4 code, which embodies these standard models, with the exception that turbulent agglomeration is omitted. The MATADOR code used in the EPRI report also appears to use the standard set of models. The IDCOR study uses MAAP, a code which replaces mechanistic modeling of agglomeration and removal with empirical correlations. MAAP is not described in the IDCOR technical summary report, and the correlations it adopts need to be published in sufficient detail to allow peer review.

The modeling of diffusiophoresis is a more recent addition to the codes, and there is some disagreement as to precisely what formula should be used. (For example, compare the three papers on the subject given to the Thermal Reactor Safety Conference in Cambridge, MA (10, 11, 12)). Therefore, the absence of details in any of the reports is particularly unfortunate. The BMI study uses the mass flux method, but details are not given, and the SWEC calculations are said to be "based on the volumetric removal rate of steam due to condensation." If the diffusiophoretic removal rate is simply identified with the volumetric condensation rate, then the rate will be overestimated by a factor of about 2, as is pointed out in the ANS' report. The EPRI calculations with MATADOR use the concentration gradient method for diffusiophoresis. The IDCOR containment calculations also include diffusiophoresis, but how is not mentioned in the technical summary report.

Details of how the calculations treat steam condensation onto particles are even scarcer. The BMI and SWEC studies use NAUA-4, which employs the Mason equation, but there is no indication given as to how the output of the thermal-hydraulic codes is converted to a form suitable for use in the Mason equation. The description of condensation in MATADOR is very similar to that of TRAP-MELT. How this code, even within the mass transfer framework, deals with the effects of condensation on the particle size distribution is not made clear.
The SNEC study also uses the NAUA-4 code to follow the water aerosol formed at the blowdown, which, if it remains until fission product aerosol release, can agglomerate with the fission product aerosol. However, it is not clear whether NAUA-4 models the tendency of these aerosols to evaporate when hot gases enter the containment from the primary system as the core heats up to its melting point.

None of the studies employ the modifications to aerosol modeling required to follow separately the distribution of different chemical components over the particle size distribution. NAUA-4 is claimed to be able to distinguish liquid water from other components (which is essential if evaporation is to be modeled), but nowhere is it described how this is done.

NAUA-4 is a discrete code, using the finite difference method. RETAIN uses the method of moments. These methods have been adequately documented elsewhere. The method of moments, as discussed in section 4.3.3.1.2, usually produces too rapid a drop in aerosol concentration at the end of sources. MATADOR is based on a hybrid method: the linear processes are calculated at each time step using a discretised distribution with a small number of nodes (six in the EPRI study), and then agglomeration is calculated by converting the discrete distribution into a log normal form and then solving the moments equations for agglomeration. The resulting log normal distribution is then rediscretized and is ready for the next time step. The number of nodes is very small; NAUA usually requires of the order of a hundred. It is difficult to see how, since the distribution is forced into log normal form at each time step, the method can avoid the drawbacks of a pure method of moments code. MATADOR requires careful checking against codes with more sophisticated numerical methods.

In summary, the modeling of fission product/aerosol transport in containment has been much improved since the first reactor safety study (17) and placed on a mechanistic basis rather than the empirical basis of the CORRAL code used at that time. The methods are also now applied to fission product retention in various auxiliary buildings as well as in the containment proper. However, in spite of its relative maturity compared with other aspects of severe accident analysis, containment aerosol modeling is still a rapidly developing subject. The methods used in the calculations reported in the studies are not based on the current state of the art, as described in the reviews quoted here (13,14,15). The major deficiencies in the model

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descriptions and the calculations die in the areas of steam condensation, multiespecies modeling and the coupling of aerosol and thermal-hydraulic calculations.

4.3.3.2. Results. This section comments on the source term results presented in the studies under review, and in other studies which have become available worldwide in recent years, and are tabulated for example in chapter 7 of the ANS report.

As noted above, in terms of the methodology used to evaluate fission product and aerosol transport in buildings, these results can be divided into three broad groups: those (BMI and SWEC) evaluated using the NAUA-4 code, those (IDCOR*) evaluated using the MAAP code, and those (EPRI) evaluated using the MATADOR code.

Within these broad groups of results, differences are clearly due simply to differences in the applied boundary conditions, such as the assumed layout and dimensions of the plant and the thermal-hydraulic conditions. The SWEC calculations represent an investigation of this kind into the effects of a number of factors relative to the containment additional to those more generally considered in fission product and aerosol transport analysis. The results show that for a realistic evaluation of the source terms, the following must be considered:

- the timing of any containment breach.
- the size of the breach.
- the extent to which the containment is compartmentalized.
- adjacent buildings and structures.

The most striking difference between results for nominally similar situations obtained using different fission product and aerosol transport codes occurs for the interfacing (V sequence) LOCA. The BMI evaluation of Surry predicts an environmental release fraction of $4.1 \times 10^{-1}$ for iodine, for example, whereas the IDCOR evaluation of Zion predicts $8 \times 10^{-5}$. (In both cases, no retention due to pool-scrubbing associated with a submerged break is assumed.) The corresponding effective decontamination factors in the auxiliary buildings into which the melt phase releases occur are respectively 1.2 and $1.25 \times 10^4$. The values for the

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*The results calculated by ELSAM in Sweden as part of the RAMA studies use a methodology based on that of IDCOR.

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other volatile fission products caesium and tellurium are similar. It is
difficult on the basis of the limited information given in the IDCOR technical
summary document to be certain to what extend this variation is due to differences
between the mechanistic modeling in NAUA-4 used by BMI and the simplified approach
in MAAP used by IDCOR, or is due to differences in the design of Surry and Zion
and, hence, in the detail of the accident sequences studied. However, it seems
certain that these latter differences are a major contributor.

In the Surry analysis by BMI, the break of a relatively large low-pressure ECCS
line is assumed, failing the ECCS and allowing a relatively rapid discharge of the
reactor coolant and RWST contents. The safeguards building into which the
discharge occurs is also relatively small, and the residence time of the fission
products and aerosols is correspondingly short. By contrast, the IDCOR analysis
of Zion assumes that the failure occurs at the RHR pump seals, the auxiliary feed
water system, the steam generator, and one train of the ECCS are assumed operable.
Hence, the blowdown is extended in time. The Zion auxiliary building is also
large and highly compartmented and is said to provide only a lengthy and devious
route for fission products and aerosols to reach the environment.

Whereas in the PWR V sequence accidents, it is the IDCOR study which predicts the
greater retention of fission products and aerosols in a building, for contained
accidents in which containment breach is delayed the reverse appears to be the
case for both PWRs and BWRs. Thus, in Table 7.17c and 7.18c of the ANS report the
effective decontamination factor provided by the containment for the volatile
fission products is of the order of 10 for the IDCOR cases, and generally in
excess of $10^2$ for the BMI and SWEC cases. Again, it is difficult to comment on
the basis of the limited information given in the IDCOR technical summary
document.

4.3.4 Application to Different Plants

This section considers how general is the application of the source term results,
and in particular those relevant to containment and auxiliary building retention,
which are reported in the studies under review.

This chapter is concerned only with changes which occur as the result of natural
physical and chemical processes which are, in this sense, universal in their
application. However, the different boundary conditions imposed by differing
plants and accident sequences produce a wide variety of results.

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At the time of the RSS (17), the calculational methods for source terms were relatively simple and conservative and two sets of source terms, each for a limited group of accident categories, were defined—one for PWRs and one for BWRs. In the period since, these source terms have tended to be regarded as of generic application. The improved source term calculational methodology which has been developed since the RSS is much more complex and this tends to militate against a similar approach for the future. The question as to what general conclusions can be drawn from the available results, and to what extent differences in plant type and design are a barrier to this, is thus an important one.

There is a need to develop a consistent argument by which the source terms for a relatively small number of accident sequences can be used to represent the hazard from a particular plant on a particular site with adequate precision. Such arguments would need to take account inter alia of the sensitivity of off-site consequences to changes in the source term (18). Only when these arguments have been developed will it be possible to judge, for example, whether accident category source terms can be defined which are independent of the site or the type and design of plant, or both.

In the meantime, a number of general conclusions of a qualitative kind can be drawn regarding fission product and aerosol transport in the containment and other buildings from the available results.

It would be anticipated on general physical principles that, where containment breach is delayed, the containment would provide large additional retention of the fission products and ensure small environmental releases. The results tabulated in the ANS report for late containment breach (>24 hours for PWRs, and after core degradation for BWRs) tend to support this view. As noted in section 3, irrespective of the class and particular design features of the reactor and its containment, and of the type of accident sequence, the effective decontamination factor provided by the containment is >10 for the IDCOR results and >10² elsewhere. As also noted in section 3, however, the current calculational methodologies do not include resuspension of materials previously deposited in the containment by rapid depressurization of the containment on breaching, or by hydrogen combustion or steam spiking occurring close in time to the containment breach. This needs further examination. Also, late containment breach in this context means late in relation to the cessation of aerosol source into the containment. When uncertainties which currently exist over the definition of these sources are resolved, it may be necessary to examine cases in which these sources are more protracted.
than has hitherto been assumed. Relevant considerations here are the re-evapor-
atization late in the accident of fission products initially retained in the RCS, the
aerosol source from core/concrete interaction and possibly long-term aerosol
generation from the hot pool in the containment sump. The APS study also refers
to an accident scenario in which there is only partial loss of the core at melt-
through, followed by intermittent degradation of the remaining fuel rods. An
examination of this kind will clearly need to look not only at the volatile
fission products but also at the refractory nuclides which may become important
when the release of the former is reduced. Only when these uncertainties can be
satisfactorily resolved will it be possible to conclude under what circumstances
delayed containment breach can be equated with low environmental release
fractions.

Reassessment and updating of both the BMI and IDCOR treatment of the PWR
V sequence accident may eradicate some of the existing large differences in the
results. However, any new evidence is unlikely to challenge a general conclusion
that in V sequence accidents some fraction of the fission products and aerosols
other than the noble gases will be retained in auxiliary buildings and other
structures outside the containment, but that the extent of this retention will be
very specific to the type of reactor and also to the details of the accident
sequence and hence of the engineering feature of a given design.

Clearly the importance of the two general conclusions arrived at in this section,
and their impact on the risk from a given reactor system, will depend on the role
of early and late containment failure and V sequence accidents in the risk profile
of the reactor.

4.3.5 Areas Where Information is Not Sufficient

The need for further information in the source term field can only be judged
against appropriate criteria for "closure" of the issue, and the need for further
information in a particular task area will depend on the contribution of the task
area both to reductions in the best estimate values for the releases and to the
overall uncertainties.

In the remainder of this section, current methodologies for calculating fission
product and aerosol transport in the containment and other buildings and the
results obtained are considered in these terms. It was concluded in section 4
that retention in the containment may well be a major contributor to reducing
releases of fission products other than the noble gases in accidents in which a breach of the containment is delayed. Hence, any associated uncertainties will be important in principle. Retention in auxiliary buildings and other structures outside the containment is more plant specific, but nevertheless may be important in particular cases.

4.3.5.1 Methodology. The following are believed to be the major areas of the calculational methodology where more information is needed. In some, but not all cases, they reflect the earlier discussion of where the studies under review are in disagreement.

1. Documentation of methods. A proper scientific peer review of source term calculations cannot be completed until the methods used are described explicitly. In this description the physical models and the numerical approximations need to be kept separate from one another.

2. Information on shape factors and effective densities. The Sandia National Laboratories QUEST study (4) identifies these as amongst the most sensitive parameters in the modeling of aerosol behavior. They need to be determined for the aerosols of interest to nuclear safety, prepared under realistic conditions.

3. Experimental test of gravitational and turbulent agglomeration. The models for these important processes need separate effects testing.

4. Hygroscopicity effects. Information on the water vapor pressures above the hygroscopic components of the aerosol is required, together with the modeling for including such effects in the codes.

5. Thermal-hydraulic calculations and experiments on intracell flows. Of particular interest here are departures from well-mixing, turbulence levels in containment and inertial deposition. The Sandia National Laboratories QUEST study (4) identifies turbulent energy dissipation as amongst the most sensitive parameters in the modeling of aerosol behavior. The operation of containment sprays could be important in this context.

6. Phenomena associated with rapid depressurization. Information, both experimental and theoretical, is needed on the flashing of pools and possible resuspension of material in aerosol form during sudden depressurization, and also on steam condensation onto aerosols under these conditions.

7. Resuspension as a result of hydrogen combustion or steam spikes. The potential importance of these processes needs to to be established.

8. The size distribution of water droplets formed during blowdown. It could be important to know how much of the fraction of
9. Retention in leak paths. More information is required about the processes affecting aerosols in leak paths, but more important, if we are to be able to claim retention in the paths, is the determination of the nature of the paths themselves.

10. Comparison of codes. In addition to the comparisons now being carried out on realistic accident cases, the codes should be subjected to simple comparison tests designed specifically to look at individual numerical methods.

11. Experimental test of the multicomponent modeling. These new features of aerosol codes need experimental testing.

12. Coupling between thermal-hydraulics and aerosol transport. None of the studies reviewed here use or even refer to the new generation of coupled thermal-hydraulic/aerosol codes. These codes can in the first instance be used to scope the importance of the couplings, once the appropriate models are included.

13. Integral thermal-hydraulic aerosol experiments. The new integrated codes need testing by integral aerosol experiments using realistic and well measured thermal-hydraulic conditions.

4.3.5.2 Applications. A number of areas were identified previously where further applications work is required to define under what circumstances delayed containment breach can be equated with low environmental release fractions. These include:

- analysis of the effects of resuspension.
- analysis of cases in which there is a more protracted release of fission products and aerosols into the containment.

Further calculations will also be necessary as part of a more general update of existing source term values when the uncertainties and inadequacies in the calculational methodology which are currently being addressed have been resolved. By then the existing NRC suite of codes, including NAYA-4, which has been used to produce many of the existing results will have been replaced by the new integrated codes MELPROG, CONTAIN and MELCOR.

4.3.6 Conclusions and Recommendations

1. Improved documentation of the available containment aerosol codes is required.

2. The retention of fission products and aerosols (other than the noble gases) in LWR containments and other buildings is likely to be a major contributor to reductions in source term release

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fractions in some severe accidents; the retention in auxiliary buildings in V sequence LOCA's in PWR's is likely to be very plant-specific.

3. In general terms, the calculation of aerosol behavior in these circumstances is one of the best-developed disciplines in the field of source term assessment and is well-supported by experiments. The principal need for the future is to improve the relevant thermal-hydraulic calculations and the coupling between these and the aerosol calculations.

The new integrated codes currently being developed will need to be tested against appropriate integral experiments.

4. Comparison between the studies under review indicates that there is essentially a consensus as to which are the important processes governing the aerosol behavior, and with certain exceptions, the appropriate mathematical models and input data representing these processes.

5. There are nevertheless significant remaining uncertainties associated with some of the currently accepted models and input data and further experiments are needed.

6. There are fundamental differences in the numerical methods adopted in the computer codes used by different groups of investigators in the various studies reviewed.

7. An NEA/CEC benchmark comparison of containment aerosol codes available worldwide is in progress and will throw light on the importance of the differences they exhibit. This may need to be supplemented by simple comparison tests designed specifically to look at individual numerical methods.

8. A number of processes have hitherto been omitted from all of these codes, including notably:
   
   - resuspension of previously deposited materials.
   - aerosol behavior in leak paths associated with building failures.

The first of these two effects will tend to increase the release fractions presently estimated and its potential importance needs to be established.

Leak path effects will tend to decrease the release fractions (as well as potentially modifying the particle size distribution of the aerosol), but to quantify the effect will require better definition of the leak paths, and this may prove to be impossible.

9. When the uncertainties and inadequacies in the methodology which are currently being addressed have been resolved, further applications work will be required to update existing source term values and provide estimates of uncertainties.
In particular, further calculations will be required to define under what circumstances delayed-containment breach can be equated with high containment retention of radioactivity.

10. For practical application of source term values, the "categorization" process adopted in the RSS needs to be reexamined and updated if necessary.

4.3.7 References

1. BMI-2104
2. ORNL TM 8842
3. NUREG/CR-3440 (SAND 83-1689)
4. SAND 84-0410
5. SWEC
6. APS
7. ANS
8. IDCOR
9. EPRI


14. EUR 9172 EN

15. KFK 3800, CSNI 95, February, 1985.


17. WASH-1400

4.4 FISSION PRODUCT BEHAVIOR IN WATER AND LONG-TERM EFFECTS

The issue of long-term release of airborne fission products during an accident is generic to all water-cooled reactors and all accident sequences. Long-term releases in the context of this paper do not include aerosols, which are the subject of section 4.3. With the exception of aerosols, which can be regenerated by physical phenomena late in an accident, delayed generation and release of airborne radionuclide-bearing species are associated with chemical reactions within containment. Since chemical reactions are involved, the chemical conditions within containment may have a profound effect on the quantity and rate of radionuclide release. This dependence on chemistry also makes it difficult to set a limit on the long-term contribution of the source term which is generic to all reactor types and accident sequences. Furthermore, the important phenomena which affect containment chemistry and the uncertainties associated with their current treatment are common to all water-cooled reactors.

The radionuclide which is expected to dominate the long-term contribution to the source term is iodine. Iodine is known to exist in several volatile chemical forms including I\(_2\) and organic iodides (e.g., \(\text{CH}_3\text{I}\)), and its release from reactors under both operating and accident conditions has been documented in the past (1). For licensing purposes, it is necessary to establish a limit for the maximum potential iodine release both as an aerosol, including iodine absorbed on a aerosols component and in molecular form.

In this section, we discuss the treatment of iodine chemistry in current analyses and the uncertainties identified by reviews of the methodology of source term calculations. Additional concerns arising from our current knowledge of chemistry in containment are also discussed.

A reading of the following material strongly suggests, in the case of an accident releasing iodine to the containment, that the aqueous solutions be made and kept alkaline and reducing to avoid many of the complications and unknowns described.

4.4.1 Severe Accident Analyses

The effects of containment chemistry and potential long-term release of radionuclides are not included in current integrated severe accident analyses. There is no consideration of airborne species other than noble gases and aerosols in the Stone and Webster studies (2,3). There is insufficient information in the available-IBCOR documentation to determine their treatment of long-term effects.
but from the summary document (4), their approach also does not appear to include other species. The EPRI-Sumry (5) analysis includes I₂ as an airborne species, but only based on an arbitrary assumption that 0.4% of the core inventory of iodine is released as I₂ from the reactor coolant system. This is used as an input into the MATADOR code for comparative purposes.

The BMI-2104 study (6) has included the generation of volatile iodides during a reactor accident on an empirical basis. The study assumes that 0.05% of the iodine, which is in containment at the time of containment failure, is airborne. After containment failure, additional airborne iodine is assumed to be released at the rate of $2 \times 10^{-7}$ of the containment iodine inventory per hour. Because of limited data, the airborne iodine is not divided into inorganic and organic iodide fractions. The study recognized that the database and assumptions used to obtain the above volatile iodine release fractions were uncertain, and suggested that a kinetic calculation based on a chemical model of iodine behavior may be required for an accurate description of the volatile iodine source term.

The NYP A Indian Point 3 source term assessment (7) includes I₂ as an airborne iodine species. It is not clear from the documentation how the source of I₂ to containment is determined. The MATADOR II code is used to model fission product behavior in containment. This code calculates the rate at which I₂ in contact with sprays will achieve a steady-state I₂ vapor pressure. An empirical partition coefficient is used to determine the concentration of airborne molecular iodine in contact with boric acid and caustic spray solutions. Long-term release is then controlled by this partition coefficient and containment atmosphere venting.

Omission of chemistry in integrated accident analyses is based on three assumptions:

- all iodine leaves the RCS as CsI,
- all CsI is removed from the atmosphere in containment as an aerosol, and
- all CsI in contact with water dissociates to form I⁻ which remains in solution indefinitely.

These assumptions are not valid in the sense of universal applicability to all fission product iodine. The airborne radiiodine measured at TMI-2 demonstrates this. Hence, there is a need to demonstrate, in a complete source term assessment, the degree to which the above assumptions are valid.

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The APS study (8) suggests that a release of 4% of the core inventory of iodine would equal the effective dose to the public from noble gases, and prompt fatalities might be expected. All severe accident analyses are deficient in demonstrating that long-term release of iodine from solution is less than this value. This requires that the analyses include some discussion of the water chemistry in containment. The iodine airborne at TMI-2 was only 0.003% of the core inventory (9). However, the water chemistry in the TMI-2 containment was nearly ideal, with the addition of hydrazine and sodium hydroxide forcing the containment water towards reducing and basic conditions. Even so, the APS study group estimates that up to 0.2% of the iodine could have been released if the containment had failed at the start of the accident.

Thermodynamically, it is simple to predict the fraction of iodine (originally present as I\textsuperscript{−}) in a pool of water which will escape to open air above the pool—100%. The question is how long will this take. None of the studies has attempted to demonstrate that the process will be slow enough to ensure sufficient decay of all isotopes (except I\textsuperscript{129}), prior to release, to warrant omission of long-term release from the source term. It is nonetheless true that the chemical reaction and mass transfer rates under accident conditions will limit releases to less than 100%.

To address the question of long-term iodine release, a number of separate computer models are being developed. These include the IMPAIR (1) code and the White shell chemical model (11). These codes address the problem of iodine conversion between different chemical forms and mass transfer, with the aim of predicting the release of radiiodine as a function of time and physical and chemical conditions within the containment building.

It is possible to object to the inclusion of delayed release of iodine in the source term for a degraded core accident. While important on the basis of industrial hygiene, a sufficiently delayed release will contribute little to the accident consequences in terms of short-term fatalities. Since this is the critical factor in emergency-evacuation requirements, it could be argued that delayed iodine release not be included in the source term. However, there is a problem in defining a cut-off time for short-term releases. One analysis (12) has shown that release of I\textsuperscript{132} from Te\textsuperscript{132} has the potential to dominate short-term doses in an accident with delayed containment failure. It is necessary to at least demonstrate that the delayed release of iodine is negligible in the short term before it can be omitted from source term analyses.
4.4.2 Issues in the Methodology of Source Term Calculations

The studies which have reviewed calculation of the source term have all recognized that there are deficiencies with respect to the treatment of iodine and in particular the chemistry of iodine in containment. These studies generally agree on the need for some consideration of chemistry in containment to completely determine long-term iodine releases. These releases are expected to be small, but should be defined more precisely before upper bound limits to releases can be set. The reviews have also identified several phenomena which specifically need to be addressed in order to predict long-term releases.

4.4.2.1 The APS Study. The summary of the APS report (8) states, "Phenomena that could generate aerosols or volatile iodine later in an accident sequence as a result of decay heating or chemical reactions may also be underestimated." It further notes that "the effects of possible chemical reactions [are] generally not included in the models used in current accident analyses." While the general merits of the three assumptions stated above are recognized, more work is recommended to document their validity. In conjunction with this, several phenomena were identified as important and inadequately assessed. These are the following:

1. While CsI is expected to be the dominant airborne iodine species in the pressure vessel, a significant fraction of the iodine may be HI in an H₂-rich atmosphere. No detailed consideration is given to the fate of this HI.

2. CsI may react with borate deposits in the reactor coolant system to release HI.

   \[ \text{CsI} = \text{HBO}_2 + \text{CsBO}_2(\text{solvated}) + \text{HI} \]

   This may occur later in an accident, and again the fate of the HI is not detailed (although most is expected to go rapidly into solution to form I⁻).

3. CsI may react with silica in Inconel to yield presumably I₂. The fate of I₂ formed in this manner is not considered.

4. Iodides in solution within containment may be oxidized to I₂ and released from solution. Factors which would affect the rate of release, including the solution pH, radiation field and oxidation potential, are not considered.

5. Radiolysis of air in the containment could lead to nitric acid formation, which would affect the pH of containment water:

   \[ \text{N}_2 + \text{O}_2 + \text{H}_2\text{O} \rightarrow \text{HNO}_3 \quad \text{G} = 1.5 - 2.7 \]

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The presence of H₂ will reduce nitric acid formation and promote ammonia formation:

\[ \text{N}_2 + \text{H}_2 \rightarrow \text{NH}_3 \quad G = 0.5. \]

The net yield of nitric acid under accident conditions is currently uncertain.

6. Radiolysis in solution may cause a buildup of hydrogen peroxide, which could lead to iodide oxidation:

\[ \text{H}_2\text{O}_2 + 2\text{I}^- + 2\text{H}^+ \rightarrow \text{I}_2 + 2\text{H}_2\text{O} \quad \text{pH} < 7. \]

7. Tellurium (particularly Te\textsuperscript{132}) decays to form iodine. No consideration is given to the fate of this iodine, particularly that which may be formed airborne from Te in an aerosol particle.

8. The solubility of CsI in steam is identified as an unknown, but is thought to be small and unimportant.

4.4.2.2 Reactor Accident Mitigation Analysis—RAMA. The RAMA study (13) has recognized that there may be several different sources of airborne radionuclides which are generated during an accident. The most important of these is the creation of volatile organic iodides. Based on their current knowledge of iodine chemistry, they have made a best estimate for airborne organic iodide formation.

This is 0.2% of the inventory released from the core. This estimate can be compared with the estimates from WASH-1400 (14) of 0.4±1.1% for PWRs and 0.7±1.1% for BWRs and the estimate from NUREG-0772 (15) of 0.03%. This emphasis on organic iodine is in contrast to lack of emphasis in the APS study where organic iodine is not specifically discussed at all.

In conjunction with potential organic iodine formation, the RAMA study has identified several issues:

1. Heating of containment materials (such as insulation) during the early stages of accident may result in the release of volatile organic material, which could subsequently react to form organic iodides in either the gas or aqueous phase.

2. Pyrolysis of containment materials, via either decay and steam heating or hydrogen combustion, could produce organic acids which will reduce the pH of the water in containment.

3. Radiolysis is recognized as an important factor in determining iodine chemistry in containment, and may cause oxidation of I\textsuperscript{-} leading to organic iodide formation.
4. The presence of CsOH dissolved along with CsI will impose a high pH, which will reduce the effects of radiolysis.

5. The effects of the decay of radioactive tellurium to iodine on the volatility of iodine have been ignored.

The RAMA study in estimating organic iodides formation did not invoke the radiolysis-induced reaction of methane (from B4⁺C⁻ steam reaction) with iodine in solution.

4.4.2.3 The ANS Study. The authors of the ANS study (16) are more satisfied with the conclusion that iodine will be predominantly in the form of CsI initially and later I⁻ in solution. While they recognize that there may be some conversion of iodine to I₂ and organic iodine, they suggest that the fraction so converted will be small (based on NUREG-0772 (15)). This is despite the fact that they recognize that "the exact mechanisms for formation and decomposition of methyl iodide are not (as) well understood." Despite the confidence demonstrated in the assumption that all iodine will end up in solution as I⁻, they have identified three areas where more work is desirable.

1. Surface/fission product chemistry—Reaction of CsI and particularly CsOH with surface materials in the reactor coolant system may cause a depletion in airborne cesium and a shift in the gas-phase iodine specification to more HI and I₂. This may have an impact on delayed releases from the RCS.

2. Boron/fission product chemistry—Reaction of boron from control rods may deplete the Cs inventory and shift iodine from CsI to HI and possibly other volatile iodine species. This possibility has not been sufficiently considered.

3. Tellurium Chemistry—While this report does not regard the behavior of the Tl³², from Te³², as important in determining the iodine source term, they have recognized that the chemistry of tellurium is not yet adequately described.

4.4.3 Current Status of Iodine Chemistry

As indicated above, there are a number of areas where the chemistry in containment is imperfectly understood, and therefore, source term calculations do not adequately treat iodine chemistry. This has been recognized by the existence of programs in several laboratories around the world to study aspects of iodine chemistry of relevance to nuclear reactor accidents. Briefly, in the following, we have delineated the areas of iodine chemistry where our knowledge is sufficiently complete, for accident analysis purposes, and have identified those areas where further work is required. A more comprehensive review of iodine
4.4.3.1 Thermodynamics. The thermodynamic database for aqueous iodine species at 25°C is in good shape (17). Experiments at higher temperatures are less complete, but current extrapolations provide an adequate description of the chemistry up to 100°C. The situation is worst at high oxidation potentials and high temperatures. Above 100°C the data are inadequate.

4.4.3.2 Kinetics. Data on iodine reaction kinetics in solution are patchy. There is very good information on a limited number of reactions between inorganic iodine species and selected reagents (such as N₂H₄ or H₂O₂). Rate constants for reactions leading from I⁻ to IO₃⁻ under some conditions have been established (18). However, there is a large uncertainty in the rate of I⁻ oxidation to I₂. This reaction is crucial to the formation of organic iodides and long-term release of airborne iodine. Unfortunately, the oxidation reaction is catalyzed by many agents and hence the rate is difficult to predict. This is a key problem requiring further work.

4.4.3.3 Radiolysis. Radiolysis of water leads to the production of both oxidizing and reducing agents. There is uncertainty in the impact that this will have on the oxidation of I⁻ in solution (19). This is further complicated by the fact that radiolysis will affect other species in solution (e.g., N₂H₄ and metal ions), which will have further impact on the rate of oxidation. More work is required in this area and studies are currently in progress to reduce the uncertainties. There is also evidence from work in progress that radiolysis of water containing aerosol material and, in particular, silver, may lead to trapping of iodine in an involatile form (20).

4.4.3.4 Organic Iodides. The dominant form of airborne iodine in containment after aerosol depletion is expected to be organic iodine. This is an ill-defined term, embracing all volatile organic iodides, but is generally associated with methyl iodide (CH₃I). While current theories suggest that the organic iodides are formed in water in the containment (21), the exact mechanism is not certain. Other possibilities include hot-atom reactions, surface reactions and combustion-driven reactions. The rates of formation and the volatilities of the airborne iodides are not known. If the dominant mechanism is via reaction in the aqueous phase, the rate of organic iodide formation will be controlled by the rate of I⁻ oxidation. Hence, the chemistry of organic iodides may be of secondary
importance in assessing a limit for the rate of long-term evolution of airborne iodine.

4.4.3.5 Solution Chemistry. The chemistry of the water in containment will control the distribution of inorganic iodine species. Iodide will remain the dominant species only if the solution can be maintained at a basic pH (17). Some factors potentially affecting pH, such as H₂ combustion and HNO₃ formation via radiolysis, have been neglected. The latter has the potential, in principal, to lower the pH to less than 4 and cause all of the aqueous I⁻ to be converted to I₂ and be evolved from solution.

While this may be unlikely for the entire containment, it may be important for CsI aerosols which have formed airborne water droplets.

While radiolysis of air may contribute to acid formation, this may be counter-balanced by release of aerosol Na₂O and K₂O from the concrete during the core/concrete interaction. These species will dissolve in water as basic alkali hydroxides. However, the core/concrete models also currently neglect the release of halides and sulphates from the concrete, which would lead to acid formation. Furthermore, unreduced CO₂ liberated during the core/concrete interaction will buffer the containment water pH. There are many conflicting contributors to the containment water chemistry, and prediction of the pH during an accident is an unresolved problem.

4.4.3.6 Hydrogen Combustion. A hydrogen combustion may produce reactive organic material which may promote organic iodide formation and may also produce organic acids which could affect water pH. It will convert any airborne organic iodide into HI or I₂. Hydrogen combustion has been demonstrated to cause the oxidation of airborne CsI to I₂ (22). The yield of I₂ versus HI depends on the completeness of the combustion and the partial pressure of H₂ remaining in the atmosphere. Consumption of a large fraction of the H₂ in the containment may lead to increased HNO₃ production and more oxidizing conditions in the containment water. There is currently no quantitative estimate for the impact of hydrogen combustion on airborne iodine.

4.4.3.7 Borate Reaction. There is evidence that CsI may react with borates, both in the solid and gas phases, at high temperatures to release I₂ (23,24). This is a particular problem for long-term release in reactor accidents which proceed to dryness. It could also result in a slow release of I₂ from the reactor coolant.
4.4.3.8 Tellurium Behavior. The behavior of \( ^{132}\text{I} \) produced as a daughter of \( ^{132}\text{Te} \) is inadequately treated. After 24 hr, the total activity of \( ^{132}\text{I} \) is approximately equal to that of \( ^{131}\text{I} \). Since \( ^{132}\text{I} \) is produced later in an accident and possibly airborne on aerosol particles, it may encounter conditions which favor \( \text{I}_2 \) or organic iodide formation. A fraction will be created under conditions where the iodine will not immediately transfer into solution and will be available for release in the event of later containment failure. The potential impact of delayed \( ^{132}\text{I} \) release has been addressed in the Sandia QUEST study (12). Their work suggests that, in accidents with late containment failure, \( ^{132}\text{I} \) may dominate the short-term dose release. However, the short half-life of \( ^{132}\text{I} \) limits its contribution to long-term doses via inhalation or ingestion.

4.4.3.9 Surface Reactions. Molecular iodine, and to a lesser extent organic iodides, have been shown to react rapidly with painted surfaces (particularly rapidly with epoxy-based paints). This rapid gettering action of airborne iodides may be credited if any accident scenario predicts substantial release of iodine from solution. The existing database and codes are sufficient for this purpose.

4.4.4 Other Fission Products

The only fission products besides the noble gases and halogens which have the potential to exist as airborne molecular species are the chalcogens (tellurium), technetium and ruthenium. Tellurium may potentially exist as volatile \( \text{H}_2\text{Te} \) or as volatile organic tellurides. However, it is difficult to postulate conditions in the primary system which would lead to creation and release of substantial quantities of these species (25). The tellurium will be released to containment as stable, solid oxides or alloys with metal elements. High temperatures during a hydrogen combustion may volatilize some tellurium from surface or aerosol particles, but reaction to form \( \text{H}_2\text{Te} \) or organic tellurides is unlikely. Irradiation of tellurium in solution in the presence of organic material may lead to production of organic tellurides.

Ruthenium may potentially be volatile as \( \text{RuO}_4 \) or \( \text{RuO}_3 \). However, in the environment of a containment with free \( \text{H}_2 \) present, these species will not be formed and become airborne. Technetium may potentially be volatile as \( \text{Te}_2\text{O}_7 \), but also will not be formed.

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Long-term release of all radionuclides except iodine is generally neglected by all analyses and is well justified.

4.4.5 Summary

The current treatment of long-term release of iodine in reactor accident analyses is consistent between the various studies under review. The assumption is made that 100% of the iodine in a containment will remain as I\textsuperscript{-} in solution once removed from the air as an aerosol particle. The basic chemistry in a containment supports the argument that the majority of the iodine will indeed be contained as aqueous I\textsuperscript{-}. However, the accident analyses do not demonstrate the degree to which this will be true, although any potential problems will clearly be minimized if the reactor operator makes the solutions in the containment alkaline and/or reducing.

The long-term release of iodine in the event of an accident is a question of chemical kinetics and mass transfer rates, not equilibrium thermodynamics. This will probably be a slow process, but this needs to be demonstrated over an appropriate range of conditions. The most important outstanding questions are:

- the rate of I\textsuperscript{-} oxidation in solution within containment,
- the pH and redox potential of water within containment,
- the effect of radiolysis on I\textsuperscript{-} oxidation, and
- the mechanism and rate of organic iodine generation and evolution.

Additional information on these effects is being developed.

4.4.6 References


7. Source Term Safety Assessment of Radionuclide Releases Under Severe Accident Conditions at Indian Point 3 Nuclear Power Plant, New York Power Authority (NYPA), 1984.


4.5 RADIONUCLIDE REMOVAL BY WATER POOLS

4.5.1 Technical Background

4.5.1.1 Introduction. A number of the pathways currently identified for what are believed to be the risk significant severe LWR accident sequences involve pathway segments through water pools. Specifically, transport through the BWR pressure suppression pool and the PWR quench tank represent examples of segments of the in-plant pathways for many risk-significant sequences in these plants. In order to analyze the fission product transport for these sequences, it is necessary to describe the effect of these water pools on the steam/noncondensable gas mixtures and associated entrained radionuclides. The amount of aerosol collected in the water pool is dependent on the gas-liquid hydrodynamics of the particular system. Historically, it has been assumed that the major effect of subcooled water pools on the entrained aerosols is to attenuate the airborne radionuclide inventory. This attenuation is usually expressed quantitatively as the "decontamination factor (DF)" which is defined as the ratio of the mass injected to the mass which escapes from the surface of the pool. The RSS assumed a DF of 100 for subcooled water pools and 1 for steam saturated pools. A recent review by Potter of General Electric of the pertinent information available through 1981 cited experimental test results obtained under a variety of test conditions to suggest that the water pool scrubbing process might be significantly more effective for attenuating aerosols than was assumed in the RSS. However, the available database was sparse at that time and suggested that the observed pool scrubbing DFs were sensitive to
the test conditions. Recent analytical work has shown that the conditions under which steam and noncondensable gas mixtures with associated entrained fission product aerosols are injected into water pools change significantly during the course of a severe accident. It follows that best estimates of the role of pool scrubbing in accident source term mitigation require close coupling, in time, of the accident thermal-hydraulic and aerosol/fission product transport through the pool analysis and pool scrubbing DF evaluation. Recognition of this has led to the development of several codes which are designed to analyze pool scrubbing DFs in severe core damage accident analysis. The validation of these methods rely in part on experiments conducted by Battelle Columbus Laboratories for EPRI and by General Electric.

4.5.1.2 What is Known?

Experimental Results. An overview of single orifice scrubbing results are presented in reference (2).

The conclusions that can be drawn from analyzing the trends in the experimental data are that for the range of experimental conditions examined, the pool scrubbing DF is large if the steam mass fraction of the carrier gas is large and/or the aerosol AMMD is greater than about 1 μm. It appears, on the other hand, that small DFs occur only when the steam mass fraction of the carrier gas is less than about 0.7 and the aerosol particle size is in the 0.4 μm AMMD range. These conclusions apply to single particle sizes and do not address the effect of particle size distribution variation during an accident and the accident DF. In the case of hot pool data, the most significant observation that can be made is that the experimental DFs are higher than expected based on early model calculations.

Modeling of Pool Scrubbing--EPRI-Sponsored SUPRA Code. The EPRI analytical program is focused through the development of a computer code SUPRA (Suppression Pool Retention Analysis) which currently can describe the scrubbing of fission products in water pools from multiple orifice injectors such as heaters or quenchers.

The water pool is divided into four zones, each characterized by different heat and mass transfer. The zones are sequential in nature. The three pool zones are: the injection zone, the bubble rise zone, and the surface zone. In addition, a pool compartment zone is available.
The injection zone is characterized by transfer that can occur during the formation and release of the gas globules, bubbles, slugs, or jets.

The bubble rise zone is characterized by transfer that begins just after the bubbles, globules or jet are detached from the orifice and have broken up into small bubbles that rise in the pool as a swarm. No interaction between bubbles is assumed to take place, and a monodisperse initial bubble size distribution is considered.

The surface zone is characterized by transfer that can occur at the pool surface as a result of desorption of dissolved gases or liquid entrainment by the portion of the injected gas stream that penetrates the pool surface and agitates the surface.

The above-pool compartment zone is characterized by transfer that can occur between the gas phase above the pool and the surface of the pool, as well as the surrounding walls. Thermal nonequilibrium between the gas phase and the surrounding walls can lead to steam condensation and appreciable removal of aerosols, as can sedimentation, given significant residence time of aerosols above the pool.

With this four-zone conceptual basis, time-dependent calculation of the DF is made for a generic aerosol species and vapor species of elemental iodine, cesium iodide and methyl iodide. The pool is treated as a simultaneous heat and mass exchanger. Time-dependent conservation of mass, energy and species equations are formulated for the liquid side and provide the time history of the pool conditions. Spatial conservation equations for the injected gas stream provide the temperature and composition of the gas phase as it travels through the pool. The analyses are, therefore, composed of the following parts:

- Overall pool material and energy balances provide the time-dependence of bulk pool conditions and thus the integral decontamination factors.
- The surface zone models provide the heat transfer, evaporation rate, gas desorption rates and liquid entrainment rate at the pool surface.
- The injection and rise zone models provide the quasi-steady spatial conditions of the gas phase. The gas phase, which is treated as a single stream simultaneous heat and mass exchanger during its passage in the pool, provides the differential decontamination factors.

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The compartment models provide the temporal conditions of the gas phase above the pool. The temperature, mass species concentration and aerosol particle concentration provide the removal rates of radionuclide species above the pool, and hence the system decontamination factors.

The calculations carried out and comparisons against the experimental data indicated the following:

1. For subcooled pools the agreement between data and predictions is well within the uncertainties of the experimental measurements. The deviation between the predicted and measured scrubbed mass is between 0 and 60%. The standard deviation in the experimental scrubbed mass is between 15 and 50%. For steam-noncondensable mixtures the agreement between data and theory is good (within 25%). The bulk of the data indicated scrubbing efficiencies between 29% and 83%, and the corresponding predicted values were between 25% and 72%.

2. The results showed that higher DFs are obtained for higher steam mass fractions, and removal efficiencies in excess of 99% are attainable.

3. High values of DF can be obtained for deep hot pools due to condensation near the orifice. The pool is subcooled at the injection depth due to the hydrostatic pressure.

4. The calculations showed that the DF is strongly dependent on the particle size and steam mass fraction. Minimum DFs are predicted at a particle radius of 0.1 μm for cesium iodide aerosols. Inertial deposition is effective for large particles and diffusive mechanisms are effective for small particles. Convective deposition dominates in condensing situations.

5. Condensation in the compartment above the pool can result in high system DF. For hot pools, the system DF was about 10 times the pool DF for selected calculated conditions.

6. Variation in the injected particle size distribution results in variation in the predicted DF.

**NRC Sponsored SPARC Code.** The Suppression Pool Aerosol Removal Code (SPARC) is being developed, under NRC sponsorship, to model the removal of aerosol particles in rising bubbles and swarms. This code models the effects of steam condensation, soluble particle growth in a humid atmosphere, gravitational settling, inertial deposition, and diffusional deposition. It also models the evaporation of steam into the bubbles, which tends to retard particle removal in hot pools. Early versions of the code did not model the bubble hydrodynamics, the user has to specify the initial size and shape of the bubbles and the bubble swarm rise velocity was calculated in the code.
Some of the assumptions inherent in the early prototype version of the SPARC computer code include the following:

- Bubbles are spheres or oblate ellipsoids having a circulation velocity unimpeded by interfacial contaminants.
- Aerosol particles are spheres.
- The degree of particle growth depends on the fraction of the particle mass that is soluble; thus the soluble fraction must be specified.
- An upper limit of $10^{+5}$ is placed on the DF calculated for any particle size.
- The user provides the correct bubble shape (either spherical or oblate ellipsoid or defined axis ratio).
- The user provides an accurate spherical bubble size at the entrance depth.
- A single bubble size and shape adequately describes the bubble population.

The current version of the code includes:

- improved correlations for representing the hydrodynamics of bubble swarms, specifically empirical correlations developed from information presented in ref. (2) for the determination of bubble size, shape, and swarm rise velocity;
- condensational particle growth models for supersaturated as well as unsaturated environments;
- subroutines to estimate the extent of elemental iodine and organic iodine removal by suppression pools; and
- expressions to estimate the extent of particle retention as a result of bubble coalescence and redispersion during swarm rise and from inertial forces associated with forming bubbles from single orifice entrance geometries.

General Electric Model. General Electric has developed models for the scrubbing process associated with quencher, downcomer and horizontal vent injection configurations into BWR pressure suppression pool. These models are based on analysis of the pertinent hydrodynamic phenomena (i.e., bubble formation at the injector and subsequent breakup as it rises in the suppression pool) together with an analysis of the processes that affect the scrubbing of aerosols entrained in the gas in these bubbles. These models are General Electric proprietary.
4.5.1.3 Agreement/Disagreement Between Studies. General Electric has employed very large DFs in their risk assessment. The RSS used DFs = 1 for saturated pools and larger values for subcooled pools. BMI-2104 employed the SPARC code to calculate DFs based on TRAP-MELT calculated aerosol loadings and particle size distributions. IDCOR employed constant DFs of 600 for orifice and 400 for horizontal vent injected aerosols. Current EPRI-sponsored large-scale scrubbing data shows some completed comparisons with SUPRA code are encouraging. Comparisons between the SPARC code and EPRI data are underway. Differences between predictions with SUPRA and SPARC codes for the DFs in BWR accident scenarios cannot as yet be determined.

Large-scale data acquisition is complete for subcooled pools and saturated pools with single orifice injection. Data acquisition has been initiated on subcooled pools with horizontal vent injection and tests are planned on downcomer injection tests.

4.5.1.4 Conclusions and Recommendations. Pool scrubbing is a key element in BWR source term estimation. A key part in determining realistic values of DF is using the large amount of experimental data currently in hand as well as that which is being planned for validation of the computer code models.

There is reasonable assurance of data completeness for subcooled pools. Some saturated pool experiments are completed and bounded estimates will be made. The modeling with the SUPRA code of orifice geometry is complete and shows encouraging comparisons with the data obtained. Modeling of other geometries is underway.

Recommendations are to compare assumptions and models in SPARC versus SUPRA code and to validate both codes versus the measured data for orifices, horizontal vents, and downcomers as well as to combine validated codes with a validated aerosol transport code to obtain realistic estimates of scrubbing DFs in accident scenarios.

4.5.2 References


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4.6 ENGINEERED SAFETY FEATURES

4.6.1 Introduction

Water-cooled water reactors are equipped with engineered safety features (ESFs) to prevent and/or mitigate a reactor accident. These ESFs include emergency core cooling systems (ECCS), containment, pressure suppression systems such as containment sprays, ice condensers and suppression pools, and emergency air cleaning systems. Figures 4.6.1 and 4.6.2 schematically show those ESFs for PWR and BWR, respectively.

Among the ESFs, the ECCS is designed to prevent core degradation in the case of a loss of coolant accident (LOCA) by injecting additional water into the reactor core and maintaining its coolable geometry. This implies that if ECCS functions and no multiple failure exists, the loss of coolant accident would be mitigated and the source terms would be low. This is the design basis of ECCS. Therefore, in source term evaluations of severe accidents, ECCS is assumed to fail in most of risk dominant sequences.

Containment is the final physical barrier for fission product release. It is designed and constructed to withstand pressures and temperatures which are anticipated in the design-basis LOCA. However, containment integrity may, in theory, be lost in severe accidents. As a result, radioactive materials might be released outside of the containment. Containment integrity is discussed later in this document.

The pressure suppression systems are designed to reduce containment pressure and temperature loadings by condensing steam in the containment. PWRs and BWRs are equipped with containment sprays. All BWRs are equipped with a suppression pool. Some PWRs have icebeds in the containment to reduce pressure during an accident and consequently the design pressure of the containment is also reduced. Pressure suppression systems are also capable of removing fission products during an accident. This section discusses the pressure suppression systems other than suppression pools which are discussed separately in another section.

Emergency air cleaning systems consist of filter systems to reduce fission product releases to the environment. Fan coolers are expected to remove heat generated in the containment. In addition to the emergency air cleaning systems within the containments, a standby gas treatment system (SGTS) is provided as a safety system in BWR reactor buildings for removal of fission products.
Some reactors are equipped with a filtered-vented containment for reducing fission product release specifically in severe accident conditions. This concept is also discussed in this section along with other ESFs in terms of reducing fission product release to the environment.

### 4.6.2 Containment Sprays

#### 4.6.2.1 Technical Background

The containment spray system is provided to suppress pressure in the containment by condensing steam released from the primary system to the containment in the event of a loss of coolant accident. The spray system is also used to remove fission products from the containment atmosphere to prevent release of radioactive materials to the environment in the event of containment failure. Steam condensation is the key mechanism of pressure suppression by the spray. Therefore, parameters affecting the steam condensation rate should be taken into account in the thermohydraulic analysis.

For fission product removal, the mechanism is different depending on the physical state of fission products, namely gaseous or particulate. It is common practice in many countries for the siting evaluation safety analysis to assume that iodine is in the form of elemental iodine. Therefore, gaseous iodine must be absorbed by water spray droplets and the condensate on the containment wall. The following must be considered in the analysis along with important thermohydraulic parameters.

- partition coefficient.
- iodine concentration in liquid phase and gas phase.
- composition of environment gas.
- spray additive and pH value.

Recent source term assessments have shown that the chemical form of inorganic iodine is likely to be mostly CsI which would exist as aerosol in a containment. An injection of spray is equivalent to an addition of larger-sized particles to a containment, and the following natural retention mechanisms must be considered for particulate removal by spray droplets.

- interception.
- impaction.
- Brownian diffusion.
- diffusiophoresis.

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4.6.2.2 What is known. A fairly large amount of experimental data is available for iodine removal by containment sprays especially for elemental iodine and organic iodide (12). Table 4.6.1 summarizes test conditions of large-scale containment spray experiments conducted by several organizations, and table 4.6.2 shows results of the tests for fresh spray and recirculation spray modes.

These experiments showed that containment spray is very effective for removal of inorganic iodine. For this case, the reduction rate depends on the pH of spray water and initial iodine concentration in gas phase. The spray is more effective in the case of high pH and low initial concentration of iodine. From JAERI's BWR simulation tests, the half-life of initial iodine reduction rate was found to be less than 10 min and the partition coefficient was larger than 100. For the PWR simulation, the half-life of initial iodine reduction rate was approximately 37 s and the partition coefficient was 4300.

For organic iodide removal, the containment spray was not as effective as it was for iodine. The CSE results showed that 2-hr dose reduction factor (DRF) was 1.5 for a large PWR containment. From the NSPP test results, it was found that methyl iodide was more effectively removed by the spray when the spray flow rate was higher and the spray droplet was smaller.

It was found from the CSE experiments that containment spray was very effective in removing aerosol particles as shown in figure 4.6.3 (7).

4.6.2.3 Areas Where Studies Agree and Disagree. Among the reports presented at the first source term task force meeting, the ANS, APS and IDCOR reports covered ESFs. The ANS report describes ESFs and sprays in the most detail. The APS report does not cover containment sprays in detail since electric power is necessary to activate containment spray and by assumption it is lost during severe accidents such as TMLB (3). However, the APS report mentioned that estimation of fission product removal by containment spray can be made with relatively high confidence because of the existence of a large amount of experimental data (2). The IDCOR report also acknowledged the effectiveness of containment sprays in a very short statement by saying that if sprays are available, consequences will be much less significant (8).

Other reports did not mention specifically containment spray effectiveness, mainly because it is agreed implicitly that containment spray is effective and a lot of information is available for iodine removal, especially for elemental iodine and organic iodide.
4.6.2.4 Areas of Information Sufficiency or Insufficiency. Sufficient information exists on containment spray effectiveness for elemental iodine and methyl iodide. However, experimental data on removal of high-density aerosols by containment sprays is not sufficient.

Vulnerability of containment spray components should be evaluated for long-time cooling and also for activation of the containment spray system during a severe accident. There is a possibility of pump suction being prevented by debris and other foreign materials during the recovery process and the containment spray may not be available even though external power is restored. There is also a possibility of generating a large amount of steam and the dispersal of debris if spray or ECCS is initiated later in the transient and water is sprayed onto the debris bed at the bottom of a containment.

4.6.2.5 Application to Different Plants. Basically, the physics should be applicable to all types of plants as long as the spray system is used and the spray parameters are specified.

4.6.2.6 Conclusion and Recommendations. The effectiveness of containment spray to remove fission products is fairly well understood for elemental iodine and methyl iodide for representative accident conditions used for the siting evaluation. Less data are available for high density aerosols. This situation represents a more realistic severe accident condition. Studies agree that aerosol removal by containment spray is very effective even if the iodine concentration is not high enough to represent severe accident conditions (2).

Further research may be necessary in the area of vulnerability of the spray components and effect of the late spray initiation on debris dispersal.

A possible implication on current regulations would be a consideration of physical form of iodine for spray calculations since the effectiveness of spray removal has been evaluated by assuming elemental iodine rather than particulates.

4.6.3 Ice Condenser

4.6.3.1 Technical Background. Ice condensers are used in certain types of PWRs in order to reduce pressure by condensing steam. A typical configuration is shown in Figures 4.6.4 and 4.6.5 (2). The benefit of having ice condensers is that the design pressure of the containment can be lowered as compared with a large dry containment. It is also expected that fission products can be trapped in the ice.
condenser by the large amount of melting ice as well as by the large structural surface area in the ice compartment.

As in the case of spray removal of fission products in a containment the mechanisms of fission product removal are different depending on the physical form of fission products. For gaseous fission products such as elemental iodine, adsorption and absorption are the primary mechanisms of removing fission products by icebeds. Water produced from the melting of ice and the condensation of steam will absorb gaseous fission products. Absorption onto the large area of the baskets may be effective in the ice compartment even after all of the ice melts. Parameters affecting absorption depend on the thermohydraulic conditions in icebeds. Thermohydraulic phenomena occurring in icebeds are complex and detailed analysis may be difficult. For example, the rate of ice melting depends on gas flow rate, temperature of the gas, heat transfer between ice and flowing gas, direction of flow, local flow obstructions, changing shape of ice while melting, and other factors. The absorption of gaseous fission products will be predictable if the partition coefficient, water chemistry, concentration of gaseous fission product in gas and liquid, water-gas interface geometries, phase velocity, and other physical states are known in adequate detail.

Important mechanisms of particle removal are gravity settling, flow-induced removal such as impaction, interception and fluid turbulence and diffusiophoresis and thermophoresis due to temperature gradients and steam condensation. Therefore, estimation of flow patterns and stea, condensation rates on the wall and ice itself is important.

4.6.3.2 What is Known. The theoretical treatment of fission products removal in icebeds has been developed in the ICEDF code (13). Particle removal mechanisms considered in the code were sedimentation, Brownian diffusion, inertial impaction, particle interception, thermophoresis, diffusionphoresis, and turbulent deposition. The code assumes well mixed nodes along the flow path.

The sensitivity calculations showed that removal efficiency depends on size distribution of the aerosols. The minimum removal efficiency occurred at a particle diameter of 0.4 μm as shown in table 4.6.3 (13). Particles smaller than 0.01 μm were efficiently removed by diffusion, and particles larger than 2 μm were effectively removed by sedimentation and inertial impaction.
The model predictions were in reasonable agreement with experimental data on the removal of elemental iodine as shown in figure 4.6.6. However, little data are available for particle removal and flow pattern of gas flow.

4.6.3.3 Areas Where Studies Agree and Disagree. The APS report recognizes the lack of experiment data and the uncertainty associated with availability of ice and estimation of ice area on which iodine is deposited. The ANS report also describes the size dependence of aerosol removal and the uncertainty of ice surface area. Both reports are essentially based on the same reference. The IDCOR report emphasizes that the structural area in the ice condensers is fairly large and must be considered even after ice melts.

Agreement exists among the reports that ice condensers will remove fission products if ice is still there. Quantitative estimates of fission product removal in the ice condensers are uncertain because of the lack of definition of the extent of mixing and the extent of surface area available for settling.

4.6.3.4 Areas Where Information is Sufficient/Not Sufficient. Experimental data on aerosol removal in the ice condenser are scarce although information is available for elemental iodine removal. Theoretical models have been developed and important physical process related to aerosol removal are incorporated in the modeling.

Thermohydraulic data for the ice condenser are also insufficient and flow patterns in the icebeds is not well predicted at present.

4.6.3.5 Application to Different Plants. All PWRs with ice condensers are designed by the Westinghouse Electric Corporation and the configuration is nearly the same so that the result should be applicable to all of the ice condenser-type PWRs.

4.6.3.6 Conclusions and Recommendations. The effectiveness of ice condensers on fission product removal depends both on size distribution of particles and also to the extent ice is available during the accident. In addition, thermohydraulic conditions such as flow patterns and steam condensation rates are important to better predict the scrubbing of fission products in the ice condenser.

Due to the lack of experiment data on aerosol removal in the ice condensers, predicted values of fission product removal may be conservative estimates.
However quantitative verification of the computer codes require an experiment representing the reactor situation.

4.6.4 Filter System

4.6.4.1 Technical Background. The filter system of the ESF system forms a part of an emergency air cleaning system; its purpose is to remove fission products from containment air. Filter systems consist of demister, electric heater, prefilter, HEPA filter, adsorber and HEPA filter as shown in Figure 4.6.7 [11]. Normally two such trains are provided for redundancy.

In the filter train, the demister is to protect prefilters, HEPA filters and adsorbers from water droplets and is made of wave plates for larger droplets and of wire mesh for smaller particles. Prefilters are installed to collect coarse particles to extend life of HEPA filters. HEPA filters are used to remove the rest of particles with high efficiency. Adsorbers are made of charcoal and iodine removal occurs on these charcoal adsorber beds.

Those components are designed for thermal-hydraulic conditions corresponding to the design basis accident and therefore if those conditions are exceeded, efficiency of the filter systems might be lower or the filter system might deteriorate. For severe accidents, the filter system efficiency must be evaluated in terms of the following items.

- thermal-hydraulic conditions (pressure, temperature),
- chemical forms,
- radiation effect,
- high humidity,
- loading capacity.

4.6.4.2 What is Known. The USNRC Regulatory Guide 1.52 requires an air cleanup system in the design of light water reactors. The postulated design basis accident conditions are shown in table 4.6.4. Therefore, the filter system should be intact under the conditions specified in the table.

However, in the case when the design basis conditions are exceeded as anticipated in severe accidents, the capability of the filter systems is not certain. Therefore experimental data about the behavior of the components of the filter systems are needed to establish the limits of fission product confinement at the
extreme conditions. The state of knowledge is summarized in tables 4.6.4 and 4.6.5 respectively for the HEPA filter and iodine absorbent (11).

4.6.4.3 Area Where Studies Agree and Disagree. The APS report gives credit to the filter system for fission product removal and as the ANS report. However, the ANS report describes the possible inefficiency of the filter if severe accident conditions persist for a prolonged time and the filter loses its integrity due to high-density fission product aerosol.

4.6.4.4 Areas Where Information is Sufficient/Not Sufficient. Filter performance at the extreme conditions beyond the design limit may need further investigation. Especially information on the filter performance under the following circumstances is not sufficient.

- high temperature and high humidity.
- radiation doses.
- mechanical loads due to high differential pressure.
- combination of the above items.

4.6.4.5 Application to Different Plants. All nuclear reactors have filter systems to remove fission products from exhaust gas from the plant. If the design criteria of the filter systems and the postulated accident conditions are the same, conclusions should be applicable to all plants.

4.6.4.6 Conclusions and Recommendations. Since the existing filter system is designed for the postulated design basis accident there exists some uncertainty for the filter system efficiency at high temperature and high humidity which will exceed the design limit for the severe accident. Therefore, experimental data are needed to confirm filter efficiency under those extreme conditions to quantify the bounding capability of the filter system.

4.6.5 Fan Coolers

4.6.5.1 Technical Background. Many reactors are provided with fan coolers consisting of a forced containment air circulating system. The heat is removed by the heat exchanged. Water is circulated by a pump in the heat exchanger and the removed heat is ejected outside the containment.
Fan coolers may be effective in removing fission products since cold surfaces exist in the heat exchanger causing steam to condense on the cold surfaces. Fission products will be absorbed by the water. If fission products are in the form of aerosols, they will be deposited in the heat exchanger.

However, fan coolers require electricity to operate. If it is operable, an appropriate credit should be taken in severe accident analysis.

4.6.5.2 What is Known. The ANS report describes containment fan coolers as heat removal and fission product removal systems. It is noted in the ANS report that no credit for fission product removal is taken in severe accident analysis.

4.6.5.3 Areas Where Information is Sufficient/Not Sufficient. Integrity of pumps for circulating water to the heat exchangers must be evaluated. The vulnerability of the fan cooler components should also be investigated.

4.6.5.4 Application to Different Plants. The information should be applicable to all plants.

4.6.5.5 Conclusions and Recommendations. Fan coolers are used to remove excess heat from the containment. Fan coolers may also remove fission products. An appropriate consideration in severe accident analysis is necessary.

4.6.6 Filtered-Vented Containment

4.6.6.1 Technical Background. In order to relieve excessive pressure and to reduce radioactive material release to the environment the concept of a filtered-vented containment has been developed for the Barseback BWR in Sweden and to the French PWRs.

The filtered-vented containment is unique to these plants. The Swedish filtered-vented containment is schematically shown in figure 4.6.8. The filtered-vent system consists of a cylindrical vessel of 10,000 m$^3$ volume, 20 m diameter and 40 m height. The vessel is a cylindrical reinforced concrete structure built above ground. The vessel is filled with 1-inch size crushed quartzite rock gravel.

The gravel bed filter acts as a heat sink for condensation of the steam from the containment and it removes aerosols and iodine from effluent gas from the containment after the rupture disk is broken at 650 KPa.

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The removal mechanisms of aerosols by the gravel bed are similar to those of ice condensers, that is, sedimentation, Brownian diffusion, impaction, interception, diffusiophoresis and thermophoresis. Thermohydraulic conditions must be specified in order to reliably predict fission product removal by the gravel bed. There is a difference between the gravel beds and the ice beds in that the configuration of the gravel beds stays the same while that of the ice beds changes due to melting of ice.

The efficiency of aerosol removal by the gravel bed depends on the grain size, bed path, gas velocity, residence time and steam fraction of the gas flow. Large-scale experiments have been conducted to determine the removal efficiency of aerosols and iodine. Analytical models have also been developed to characterize a full scale gravel bed.

The French filtering system is based on a moderately expensive sandbed filter placed between the containment and the stack, the gases being monitored before release.

Specifications for the design of the venting system are as follows:

- maximum gas flow rates: 3.5 kg/s.
- typical gas composition: air (33%), CO₂ (33%), CO (5%) system (28%);
- gas density: 4 kg/m³;
- inlet gas temperature: 140°C;
- inlet pressure (before decompression in the pipes): 0.5 MPa;
- pressure drop through the filter: 0.01 kPa;
- pressure at filter outlet: nearly atmospheric;
- maximum airborne aerosol concentration: 0.1 g/m³ total amount: 5 kg;
- aerosol mean diameter: 1 micron; and

The filtering system--about 40 m² in surface area and 0.8 m of sand height--is not expected to constitute a heat sink large enough for the bulk of the steam passing through to be condensed.
Further information on this subject can be found in sections 6.2.1 and 6.3.1.

4.6.6.2 What is Known. Experiments were conducted to obtain thermohydraulic data in terms of heat removal and gas flow in the gravel bed. They showed that a gravel bed with a volume of 4,000 to 10,000 m³ filled with a gravel size of 25 to 35 mm can condense all expected flows from the containment postulated in the safety analysis of the Barseback reactor. The condensate did not close the gravel bed and most of the condensate flows downward.

In order to determine amount of fission products removed by the gravel bed, experiments were conducted with aerosol particles and iodine. The experiment was performed at 100°C with condensing steam. Results showed that elemental iodine was removed by physical absorption to the gravel and trapped chemically in the bed through chemisorption.

The sandbed filter to be used in French PWRs has been specified according to the results of the "PITEAS-Filtration" experimental study, in which the best combination of filtration efficiency/pressure drop parameters was established.

4.6.6.3 Areas Where Studies Agree and Disagree. The filtered vented containment is adopted only in Sweden and France, and no other detailed study is available. Therefore, no comparison is possible.

4.6.6.4 Areas Where Information is Sufficient/Not Sufficient. Sufficient data seem to exist for the filter efficiency of the gravel bed of the Swedish reactor and of the sandbed of the French PWRs. Since the filter system is plant-specific, new data would be needed if a different type of filter is selected.

4.6.6.5 Application to Different Plants. The effectiveness of the filtered, vented containment depends on plant and/or the design of the filter of the system. Therefore, evaluation must be plant-specific.

4.6.6.6 Conclusion and Recommendation. The filtered-vented containment is employed in the Swedish Barseback reactor and the French PWRs to mitigate severe accident by relieving the containment pressure. The filtered vent consists of a gravel bed or of a sandbed in which fission products are removed. Choice, size and configuration of materials, gas composition and temperature are important parameters for designing the filter. Several experiments were conducted to
determine the design criteria. It was found that the gravel bed and the sandbed are efficient in reducing fission product release to the environment.

4.6.7 Summary

The following ESFs were discussed.

- Containment sprays
- Ice condensers
- Filter systems including SGTS
- Fan coolers

In addition to the ESFs, the filtered-vented containment was also included in the discussion. The following comments can be made on the impact of ESFs on source term estimate.

- ESFs are, in general, effective in reducing fission products.
- ESFs will be important when operator action is considered. In that case, vulnerability of ESFs must be evaluated.
- A filtered-vented containment seems effective in scrubbing fission products.

4.6.8 References


2. Report of the Special Committee on Source Terms, American Nuclear Society, pp. 4.2-20, 4.4-11 to 4.4-21, 5-11 to 5-12, 1984.


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4.7 THE EFFECTS OF HYDROGEN COMBUSTION ON THE SOURCE TERM

4.7.1 Introduction

The release from reactor containments to the outside atmosphere or to adjacent volumes through an opening can be estimated by the standard orifice discharge equations. Thus:

\[ \dot{m} = f(P, P_a, P_c, T, C_R, A, \gamma, P_L) \]  \hspace{1cm} (1)

where P and T are the pressure and temperature near the opening, \( C_R \) is the concentration of radioactive species near the opening, \( P_L \) is the flow loss, \( P_c \) and \( \gamma \) are the critical pressure and specific heat ratio for the containment atmosphere, \( P_a \) is the pressure outside or in the adjacent volume and A is the area of the opening.

Combustion of hydrogen mixtures within the containment will increase the pressure and temperature, and it may also change the concentration of the airborne radioactivity by physical and chemical processes (resuspension of aerosols, oxidation of iodides, etc.). As seen from eq. 1, these effects could directly influence the source term.

4.7.2 Excess Pressure and Temperature Caused by Hydrogen Combustion

4.7.2.1 Key Factors. The potential increase of the pressure in containment atmospheres due to \( H_2 \) combustion depends on the following factors:

A. Amount of hydrogen which could be burnt since this limits the total energy release.
B. Rate of burning which determines the time available for pressure relief and heat losses.

The amount of hydrogen burnt can depend on time of ignition, the mixing time for this, and the amount and distribution of the hydrogen concentration within the mixture. Thus, hydrogen and steam mixing in air and ignition and flammability limits are some of the key factors which affect the amount of hydrogen that could be burnt for a specified time of ignition.

The rate of pressure increase due to burning competes with the rate of relieving the pressure by venting to other lower-pressure parts of the containment and the rate for reducing the temperature by heat loss to the surrounding walls and other surfaces.

The rate of burning depends on the initial thermodynamic state (pressure, temperature and composition of the mixture) and the prevailing mechanism for flame acceleration. For laminar burning of initially quiescent flammable mixtures, the burning rate is described by the burning velocity which is a well-defined physicochemical property dependent on the thermodynamic state of the mixture. Such laminar burning is not likely in containments where there is turbulence generated by coolers and fans. This turbulence results in enhanced transport of mass and heat and increased flame area which increase the burning rate. Traditionally, the burning rate under such turbulent conditions is expressed as a ratio to the laminar burning rate. Turbulence parameters such as intensity and scale of turbulence are considered to influence this ratio.

Other mechanisms which have been observed to increase the burning rate are obstacles in the path of the flame, inhomogeneity of hydrogen concentration, instabilities caused by local burning accompanied by venting to other parts of the enclosure, and certain types of confinement which lead to pressure piling. Thus, laminar burning velocity and the flame acceleration mechanisms are the key factors influencing rate of burning for a given flammable mixture.

The following sections review the status of current knowledge of the key factors identified above as influencing the amount of hydrogen burned and the rate of burning.

4.7.2.1.1 Hydrogen-Mixing and Distribution. Transient gas mixing is not as well studied as mixing between liquids and between solids. However, simu-
lated tests for LWR ice condenser plants have been carried out in a scale-
model facility using conditions based on similarity parameters (1). The
results indicate that the maximum concentration differences between various
locations are of the order of 3 vol%. Tests have also been carried out at
Battelle Frankfurt in a simulated German reactor containment (2). Also codes
(RALOG, HECTR, HMS, CAP) have been developed which predict these tests well
except in the cases where a temperature inversion exists (2). This appears
to be an area where models may need to be improved.

4.7.2.1.2 Flammability and Ignition Limits. The flammability limits define
the range of composition for quiescent mixtures in which a flame, once
initiated, will sustain and propagate. Considerable work has been carried
out by the U.S. Bureau of Mines (3) on the flammability limits of hydrogen
mixtures and the effects of diluents on these limits. More recently, a
systematic study of the effects of diluents and temperature on flammability
limits has been completed for upward, downward and horizontal
propagation (5). Thus, flammability limits of hydrogen mixtures can be
adequately estimated for containment conditions.

A major discrepancy between the different sets of flammability-limit data for
hydrogen-air-steam mixtures is in the concentrations of steam required for
inerting. A value of 55% steam by volume is often quoted in literature.
However, in recent tests, mixtures with 63% steam successfully sustained a
flame for upward propagation at 100°C (see figure 4.7.1). Also, igniter
tests in a 17-l vessel with turbulence (7) indicated that 59% steam mixtures
can be easily ignited. It thus appears that, for safety analyses, the
inerting steam concentration should be revised upward to at least 60%.

Ignition limits have been studied for different hot-surface type igniters
currently considered for hydrogen control by deliberate early ignition during
severe accidents (7,8). Igniters were found to be effective for near-limit
mixtures even under condensing-steam conditions. Thus, the igniters can
initiate a flame in mixtures with near-limit concentrations for which flame
speeds and peak adiabatic pressures are relatively low under quiescent
conditions.

4.7.2.1.3 Laminar Burning Velocity. The laminar burning velocity determines
the rate at which energy is released by combustion during laminar burning.
Until 1983, experimental data for the laminar burning velocity of H₂-air-
steam mixtures were scant (9,10). In 1983, experimental values as well as a
correlation for the laminar burning velocity of H₂-air steam mixtures at
ambient pressures were reported (11) for temperatures up to 200°C and steam
concentrations up to 12% (see figure 4.7.2). More recently, the database has
been extended to steam concentrations up to 50%. Also, the effects of other
diluents such as CO₂ and nitrogen have been determined. Thus, the laminar
burning velocity database appears to be complete for H₂-air steam mixtures.
Data for CO containing mixtures do not appear to be this extensive.

4.7.2.1.4 Flame Acceleration Mechanisms.

Turbulence. The effect of turbulence on burning rates has been
quantitatively determined by a number of researchers (12-17). Attempts have
also been made to correlate the "turbulent burning velocity" data with
turbulence parameters to arrive at "universal" correlations. Success in this
area has been limited (19,20). Recent attempts to predict the experimentally
observed effects of turbulence on the combustion time and peak pressure for
H₂-air mixtures (15) indicate the need for a better quantification of the
effect of turbulence and laminar burning velocity on the turbulent burning
velocity.

Methods are now under development to estimate upper bounds for turbulent
burning velocity from first principles (21,22). These can help in estimating
upper limits for turbulent burning rates.

Obstacles Although it has been known for a long time that obstacles in the
flame path produce turbulence which accelerates the flame, systematic tests
have been carried out only recently (23,24). It is now known that continuous
flame acceleration can occur if obstacles are located at regular distances.
The maximum flame speed appears to have an upper limit corresponding to the
sonic velocity. Some large-scale experiments have also been carried out with
gratings as obstacles (25). In these cases, although the obstacles increased
the burning rate, they also reduced the peak pressure by absorbing heat from
the burnt gases. Thus, the overall effect of obstacles depends on obstacle
size and geometry as well as the geometry of the system.

Venting Since some of the containments have interconnected compartments,
burning in one compartment will be accompanied by venting to the second com-
part ment. Large-scale experiments have shown that such venting can lead to
large flame accelerations and large-amplitude pressure oscillations with
momentary peaks exceeding adiabatic combustion pressures (26). Recently, a
number of tests have been carried out in an intermediate-scale facility to
investigate the effect of vent size on the overpressure caused by the
combustion of H₂-air-steam and H₂-air-CO₂ mixtures (27). Large-amplitude
pressure oscillations result for H₂-air-mixtures with hydrogen concentration
greater than 20 vol%. The addition of diluents suppresses these oscil-
lations. These test results cannot be directly applied to all containments
since vent sizes are different. Appropriate tests in relevant geometry are
necessary to determine the potential overpressures.

**Concentration Gradient.** Most of the analyses use steady-state burning rates
to predict combustion pressure transients. However, when concentration
gradients exist, propagation from a fast-burning mixture to a slow-burning
mixture will result in a transient burning rate in the second mixtures which
will be larger than the steady-state burning rate. The extent of this
accelerated burning has not been studied. A few tests have been carried out
in an experimental assembly consisting of a sphere connected to a pipe
through a burst disk (25). The compositions of the flammable hydrogen-air
mixtures initially contained in the sphere and the pipe were different in
these tests. However, the results were not amenable to full interpretation
because of the effect of turbulence generated by the venting of the sphere to
the pipe.

**Deflagration to Detonation Transition by Pressure Piling.** It has been shown
that under certain confinement conditions pressure waves can pile up
resulting in a shock wave which can enable the deflagration to transit to
detonation (28). This transition to detonation has not been fully
understood.

**Detonation Limits.** These are concentration ranges outside of which a
detonation cannot be sustained. Recent experiments at Sandia appear to
suggest that these ranges are wider than initially assumed (29). In recent
years, detonation sensitivity of a mixture has been well correlated with the
dimension of the cell size in the pattern produced on smoked aluminum foil
placed in tubes in which detonation tests have been carried out. These cell
dimensions have also been correlated with various geometries through which a
detonation can be transmitted (30). These results are applicable to reactor
containments in determining whether a local detonation initiated at some

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locations can be transmitted to other locations. Tests are now being conducted at Sandia to determine the effect of temperature and diluent steam on detonation limits (30).

Multiple Acceleration Mechanisms. Some test results are available on the combined effect of more than one acceleration mechanism: e.g., venting plus turbulence (25). Concentration gradients, turbulence and venting are flame acceleration mechanism which may exist simultaneously in some of the accident scenarios. Combined effects, depending on the mixture composition, can increase flame speeds enough to cause momentary pressures which are above adiabatic combustion pressures.

Applicability to Containment. Of the various key factors listed and discussed above, hydrogen-gas-mixing, effects of venting and obstacles on flame acceleration and effect of confinement on pressure piling and transitions to detonation are geometry-specific. Hence, laboratory data are not directly applicable unless "scale models" are used. For some of these phenomena, complete scale-down may be impossible because of contradictory requirements of different similarity laws.

Available information on other aspects such as flammability and ignition limits, laminar burning velocity, the effects of turbulence on burning velocity and effects of concentration gradients are generic and can be applied to any containment provided the conditions such as mixture composition and turbulence parameters are known.

4.7.2.2 Combustion Models. A large number of codes have been developed to predict pressure and temperature transients due to hydrogen burns (31,32). These use a variety of combustion models ranging from completely phenomenological to sophisticated, conservation-equation based models. However, their assessment with experimental data has not been extensive especially for turbulent combustion, flame acceleration by obstacles and venting and burning in multicompartment geometry. Although detonation can be well-modeled, deflagration-to-detonation transition has been elusive so far.

4.7.2.3 Equipment Survivability. Deflagration is considered too rapid to transfer enough energy to essential equipment and cause damage. However, under certain situations, a standing diffusion flame can occur. If such flames are likely in the vicinity of equipment essential for post-LOCA management, it is
necessary to assess if there is potential for damage which might affect the source term.

Laminar and turbulent diffusion flames have been well researched. However, limited data are available for H₂-air-steam diffusion flames (33). The models have recently been extended (34) to predict heat flux and flame length of these diffusion flames. These predictions need to be assessed with data which are presently being generated.

4.7.2.4 Conclusions. For a complete understanding of the behavior of hydrogen-air-steam mixture in containments, further work appears to be required in the following areas:

1. Improvement and assessment of hydrogen mixing models for conditions during which temperature inversions may exist.
2. Improvement of the knowledge of the effects of turbulence on burning rates.
3. Effects of venting on burning rates and pressure transients.
4. Effects of concentration gradient on the rate of burning.
5. Pressure transients when multiple flame acceleration mechanisms are present.

Of these areas, (3) and (5) are geometry dependent whereas information in areas (2) and (4) are directly applicable to any general containment.

For assessing safety of reactor containments under accident conditions, the above information is not critical as long as containment integrity is assured based on a conservative estimate of peak pressures. However, the estimate of releases through preexisting openings depends on the pressure transients due to combustion as influenced by the existing flame acceleration mechanisms. The effect of the uncertainties in the understanding in the above areas on the expected performance of the various containments and on the source term are discussed later.
4.7.3 Effect of Combustion on Airborne Radioactivity

Hydrogen combustion is expected to have its greatest chemical impact on the behavior of airborne iodine. Experiments at Sandia (35) have demonstrated that a significant fraction of dry CsI aerosol can react with intermediate radicals produced during combustion to release molecular iodine (I₂). There is not yet sufficient data to predict the amount of I₂ (and possibly HI) which could form during combustion in containment. The residual cesium will condense on surfaces as an oxide or hydroxide. Hydrogen combustion is also expected to rapidly convert any airborne organic iodides to I₂ and HI. Combustion is not expected to create any other volatile species from other fission products. It may promote oxidation of some metallic fission products on surfaces or aerosol particles to involatile oxide forms.

The physical effects of combustion will influence aerosol behavior in containment. If the combustion is rapid, shock waves may develop which will lead to some deagglomeration of aerosol particles via high-energy collisions. However, the passage of the shock front will also lead to increased turbulence which will enhance agglomeration. The latter is expected to dominate the former and hence lead to increased aerosol agglomeration.

The particles deposited on surfaces may be resuspended by the impact or passage of a shock wave (combustion shock front) and by thermohydraulic effects associated with local high gas velocities. Large particles (30 μm) have the greatest probability for resuspension but will quickly resettle. The resuspension of smaller particles is different due to the strength of adhesion forces and will be considerably hampered by the likely presence of surface films of water, in a humid environment, which will also increase adhesion.

The direct heating of aerosol particles or surface-deposited material by the high-temperature combustion front may cause local evaporation of fission products. This could cause a decrease in the size of aerosol particles already airborne and could lead to formation of new small aerosol particles via nucleation and condensation of the evaporated material. This effect is expected to be small owing to the relatively high velocity of the combustion front, and hence, the short contact time for heat transfer.

Overall, hydrogen combustion is not expected to lead to substantial resuspension of aerosol material or changes is the source term. However, experiments in this
area are very limited, and theories are inadequate to fully describe some of the complex phenomena which will occur. If hydrogen combustion results in containment failure, the resulting depressurization may lead to evaporation, sump-pool boiling and resuspension and consequent increase in airborne activity within the containment. The effect on the release would then depend on the depressurization time scale. The ongoing LACE program should provide the required information on thermohydraulics processes. The interpretation of the data with combined containment-aerosol codes should help estimate the effect on the source term.

4.7.4 An Assessment of the Reviews by Various Study Groups

The background information used in the reviews of the various study groups (IDCOR, ANS, APS, Stone & Webster, NRC and others) is generally valid. In this section, we point out the limitations of some of the assumptions in the analyses reported in the reviews and some discrepancies between the reviews in interpretation.

The Stone & Webster Engineering Company, in support of the ANS review, estimated the influence of various parameters on fission product retention during accidents. However, the calculated releases do not appear to include the effects of hydrogen combustion on the pressure in the containment and on the concentration of airborne radioactivity. Also, the releases appear to be based on representing the containment as a single well-mixed node. Finer nodalization may be required to determine release rates through preexisting openings during combustion.

The ANS review considers high concentrations of hydrogen to be unlikely in the later core/concrete interaction stage of the accident because the hot core debris near the hydrogen source is considered to be an effective igniter. However, near the source hydrogen may not be flammable depending on the environment. Thus, flammable compositions may be attained at locations quite removed from the hot core debris. If so, high concentrations cannot be ruled out.

Further, among the factors listed as influencing combustion overpressure, turbulence has been neglected in the summary document of the ANS group. This could be significant especially for concentrations close to the flammability limits where incomplete combustion is usually assumed. Turbulence significantly increases the extent of combustion as well as the burning rate of these mixtures.

The ANS study also equates the release through openings which are preexisting to release through openings which would result during early breach. The initial
release through preexisting openings are subsonic, whereas releases through openings resulting from early breach are likely to be sonic. Thus release rates are likely to be different.

The APS review considers that when hydrogen is present, steam is present in large quantities to inert the mixture. However, the calculated results which indicate such inerting appear to be based on a single-node representation of the containment. A finer nodalization may indicate that flammable pockets can form. Further, the 55% steam concentration quoted for inerting is not universally accepted. Mixtures containing up to 60% steam have been found to be flammable (6). The presence of turbulence could lead to nearly complete combustion of these mixture (7).

The APS review considers the minimum combustible gas concentration to be 10% by volume whereas the measured lower flammability limit for H₂-mixtures is in the range of 4-9% by volume. Thus, combustion cannot be ruled out for some parts of the accident sequence considered in the review. The APS study also considers that maximum pressure in containment is determined by overall energy changes and not the transient. This may not be true for fast combustion. Under conditions leading to fast combustion, pressure waves produced could move back and forth, merging, reflecting and causing local pressure in excess of adiabatic pressures (36). For instance, the review quotes a time of 1/10 sec for complete combustion in containments. Assuming a minimum containment diameter of 100 ft, this corresponds to a flame travelling at the speed of sound in room-temperature air. Under such conditions, the combustion time scale and pressure equilibration time scales overlap indicating the possibility of local momentary pressure in excess of adiabatic combustion pressures.

The IDCOR study considers the mixing processes to be sufficiently rapid compared to the time scale of the accident phenomena. However, ignition can occur prior to complete mixing. Thus, what is of concern is the growth of the volume of the flammable region, the concentration gradients within the flammable pockets and the presence of other flame acceleration mechanisms such as turbulence, obstacles and intercompartment venting. The study concludes that deflagration data for mixtures slightly within flammability limits result in low pressures and flame velocities. The forced convection due to fans and sprays, which the study concluded would increase mixing, has associated turbulence which would increase burning rate and the extent of combustion of these mixtures.
The NRC study and the EPRI study are some of the reviews which consider the hydrogen combustion effects. The computer code (MARCH) used for calculating pressures and temperatures caused by hydrogen combustion accounts for laminar as well as turbulent combustion. However, the formula used to determine flame acceleration appears to be based on one set of experiments. Its applicability to containment conditions does not appear to have been well demonstrated. Also, flame acceleration due to venting to adjacent compartments is not considered.

The remaining study group reviews (Italian, French, and the Swedish) do not appear to include the effects of hydrogen combustion in their analysis.

A few of the review groups have considered the effects of hydrogen combustion on the concentration of airborne radioactivity. The APS review includes the effect of combustion on the oxidation of aerosols and iodides to volatile species. The ANS study refers to oxidation of organic iodides. Other reviews do not appear to consider this effect.

The effects of slow hydrogen burns such as diffusion flames on essential equipment survival is considered only by the IDCOR group.

To conclude, many of the study groups have neglected the effect of hydrogen combustion on the source term. Several of them consider mixing to be rapid compared to the accident time scale which results in low hydrogen concentrations. However, during accident scenarios it is not possible to rule out nonuniform concentrations at least for a short time with potential for high flame speeds and overpressures. The effects of turbulence on the burning rate has been explicitly included only in a few reviews (IDCOR, NRC and EPRI). A few of the reviews mention the effects of combustion on the concentration of airborne radioactivity, although these effects do not appear to be included in the calculated releases.

4.7.5 Containment Integrity With Regard to Hydrogen Combustion and Source Term

4.7.5.1 Containment Integrity

4.7.5.1.1 PWR Dry Containments. These have a large volume (-3 x 10⁶ ft³) and a relatively high design pressure (-45 psig). The predicted failure pressure is two to three times the design pressure. Even a global H₂-air deflagration (with no flame acceleration) in the containment is thus unlikely to cause failure since the concentration of hydrogen required is greater than 25% (see figure 4.7.3) by volume which is much larger than likely concen-
trations. The expected presence of steam further confirms that early failure due to hydrogen deflagration is unlikely in these containments.

Transition to detonation is also unlikely because of the expected low hydrogen concentrations, the presence of steam, and the relatively open volume. Thus, it can be concluded that hydrogen combustion is unlikely to cause early containment failure.

4.7.5.1.2 BWR Mark I and II Containments. These containments are inerted with nitrogen in the United States thus precluding hydrogen combustion. Such inerting is not practiced in some of the European BWRs. These containments have a high design pressure (about the same as or higher than for dry PWRs). However, since the containment volumes are small, and the steam is condensed in the suppression pool, flammable mixtures of hydrogen have to be expected early in an accident.

4.7.5.1.3 BWR Mark III and Ice Condenser Containments. Ice condenser containments have a relatively low design pressure (15 psig) smaller volume than the dry PWRs. The amount of steam will likely be reduced in this type of containment by condensation, and thus the hydrogen mixtures will potentially lead to faster flames and higher peak pressure. Hence, this type of containment is equipped with deliberate igniters. Large-scale tests indicate (16,17,35) that the hydrogen release may then be burned continuously resulting in diffusion flames with very low excess pressures.

4.7.5.2 Source Term. Hydrogen combustion directly affects the source term when releases through preexisting openings are estimated. For this situation, releases depend directly on the pressure transient and thus the combustion rate as determined by the key factors including the various flame acceleration mechanisms. Thus, an increased understanding of these flame acceleration mechanisms is essential unless releases based on conservative pressure estimates are acceptable.

The diffusion flames expected in the Mark III and Ice condenser Containments could act as long-term heat sources at high temperatures which may be of concern for the survivability of safety-related equipment required for accident control and management. The tests to be conducted in the quarter-scale Mark III containment, now being constructed for determining the expected temperature and heat flux, should help quantify the expected effect on the source term.
4.7.6 **Recommendations**

The recommendations in the order of decreasing priority are:

1. Source term estimate for preexisting openings should include the effects of combustion on the release. In this regard, the ignition time should be considered as a parameter since it will influence the amount of hydrogen burned. Also, the sensitivity of calculated releases to node size and number of nodes should be determined under combustion conditions. The combustion pressure transients for calculating releases should be conservatively estimated taking into account the influence of flame acceleration mechanisms such as turbulence, venting, concentration gradients and confinement.

2. Source term estimates for preexisting openings should be updated by including the effects of hydrogen combustion on airborne radioactivity. It should be determined from models based on tests carried out to determine the extent of vaporization and oxidation of aerosols, oxidation of iodides and resuspension of aerosols from surface deposits.

3. The models for hydrogen mixing should be assessed more extensively with relevant experimental data especially for large temperature gradients and temperature inversion conditions.

4. The diffusion flame test results in the quarter-scale BWR Mark III containment should be used to ascertain the survivability of safety-related equipment.

5. The models for turbulent combustion and multicompartent burning should be assessed more extensively with relevant experimental data.

4.7.7 **References**


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<table>
<thead>
<tr>
<th>Item</th>
<th>NSPP</th>
<th>CSE</th>
<th>JAERI Model C.V.</th>
</tr>
</thead>
<tbody>
<tr>
<td>1. Test vessel</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Material</td>
<td>Stainless Steel</td>
<td>Carbon steel (painted)</td>
<td>Stainless Steel</td>
</tr>
<tr>
<td>Size</td>
<td>3.0 m (dia.) x 4.6 m H</td>
<td>7.6 m (dia.) x 20 m H</td>
<td>7 m (dia.) x 20 m H</td>
</tr>
<tr>
<td>Volume</td>
<td>38 m$^3$</td>
<td>750 m$^3$</td>
<td>708 m$^3$</td>
</tr>
<tr>
<td>2. Volume of spray tank</td>
<td>0.38 m$^3$</td>
<td>7.6 m$^3$</td>
<td>80 m$^3$</td>
</tr>
<tr>
<td>3. Type of spray nozzle</td>
<td>1713</td>
<td>3/4 TC3.3/8 A20, 3/8 A50</td>
<td>1EX-554L-1C</td>
</tr>
<tr>
<td>Number of nozzles</td>
<td>1</td>
<td>3 or 12</td>
<td>6</td>
</tr>
<tr>
<td>4. Heating of test vessel</td>
<td>Steam injection</td>
<td>Steam injection</td>
<td>Steam injection</td>
</tr>
<tr>
<td>5. Iodine injection</td>
<td>$^{12}$I or $^{131}$I (swept by carrier air)</td>
<td>$^{137}$Cs$^{*}$ + UO$_2$ mixed aerosol (swept by steam)</td>
<td>$^{12}$ or $^{131}$I (swept by N$_2$ gas)</td>
</tr>
</tbody>
</table>

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### Table 4.6.1
Iodine Removal—Comparison of Test Conditions (Continued)

<table>
<thead>
<tr>
<th>Item</th>
<th>MSPP</th>
<th>CSE</th>
<th>JAERI Model C.V.</th>
</tr>
</thead>
<tbody>
<tr>
<td>6. Sampling of iodine</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Gas phase</td>
<td>Maypack sampler (2ch)</td>
<td>Maypack sampler (14ch)</td>
<td>Maypack sampler (4ch)</td>
</tr>
<tr>
<td>Liquid phase</td>
<td>Water sampler</td>
<td>Water sampler</td>
<td>Water sampler (+Water concentration vessel)</td>
</tr>
<tr>
<td>7. Iodine measurement method</td>
<td>Radioactivity measurement</td>
<td>Radioactivity measurement</td>
<td>X-ray fluorescence analysis or radioactivity measurement</td>
</tr>
<tr>
<td>8. Number of tests</td>
<td>17</td>
<td>7</td>
<td>2</td>
</tr>
</tbody>
</table>
### Table 4.6.1
Iodine Removal—Comparison of Test Conditions (Continued)

<table>
<thead>
<tr>
<th>Item</th>
<th>NSPP</th>
<th>CSE</th>
<th>JAERI Model C.V.</th>
</tr>
</thead>
<tbody>
<tr>
<td>8. Free gas phase volume: V</td>
<td>38 m³</td>
<td>594 m³</td>
<td>708 m³</td>
</tr>
<tr>
<td>10. Spray nozzle height: H</td>
<td>4.5 m</td>
<td>11.7 m (average)</td>
<td>15 m</td>
</tr>
<tr>
<td>11. Spray flow rate: F</td>
<td>3.4 m³/hr</td>
<td>2.9 to 32 m³/hr</td>
<td>21 m³/hr</td>
</tr>
<tr>
<td>12. Spray flux: PH/V</td>
<td>0.40 m/hr</td>
<td>0.057 to 0.59 m/hr</td>
<td>0.44 m/hr</td>
</tr>
<tr>
<td>13. Initial gas phase temperature</td>
<td>130°C</td>
<td>120°C</td>
<td>125°C</td>
</tr>
<tr>
<td>14. Temperature of spray</td>
<td>Room temperature</td>
<td>32°C or room temperature</td>
<td>70°C</td>
</tr>
</tbody>
</table>
Table 4.6.1
Iodine Removal—Comparison of Test Conditions (Continued)

<table>
<thead>
<tr>
<th>Item</th>
<th>NSPP</th>
<th>CSE</th>
<th>JAERI Model C.V.</th>
</tr>
</thead>
<tbody>
<tr>
<td>15. Initial pressure</td>
<td>4.2 ksf/cm² (Air at 1ata + saturated steam)</td>
<td>same as left</td>
<td>same as left</td>
</tr>
<tr>
<td>16. Spray period</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>• Fresh spray</td>
<td>6 min for 5 tests</td>
<td>3~60 min for 6 tests</td>
<td>50 min for 2 tests</td>
</tr>
<tr>
<td>• Recirculation spray</td>
<td>0 tests</td>
<td>0~60 min for 6 tests</td>
<td>25 hr for 2 tests</td>
</tr>
<tr>
<td></td>
<td></td>
<td>6 min for 12 tests</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>40 min for 1 test</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>20 hr test</td>
<td></td>
</tr>
<tr>
<td>17. Quality of spray water</td>
<td>Borax : boric acid (B=3000ppm) +NaOH, PH=7.2~9.4 or water</td>
<td>Borax : boric acid (B=525~3000ppm) +NaOH, PH=9.5 or boric acid (B=3000ppm, PH=5.0)</td>
<td>Borax : boric acid (B=2450ppm)+NaOH PH=9.5 or boric acid (B=2500ppm)</td>
</tr>
<tr>
<td>18. Initial iodine concentration in gas</td>
<td>100 mg/m³</td>
<td>160 mg/m³</td>
<td>41 mg/m³</td>
</tr>
<tr>
<td>phase</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
Table 4.6.2
Test Results (Removal of Inorganic Iodine by Borax Spray)

Fresh spray mode

<table>
<thead>
<tr>
<th>Item</th>
<th>NSPP</th>
<th>CSE</th>
<th>JAERI Model C.V.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Partition coefficient*</td>
<td>&gt;100,000</td>
<td>&gt;10,000</td>
<td>(not measured)</td>
</tr>
<tr>
<td>Ratio of iodine reduction rate ((\lambda_s(E)/\lambda_s(C)))</td>
<td>2</td>
<td>1</td>
<td>0.66 (half life of iodine reduction = 37 sec)</td>
</tr>
<tr>
<td>General tendency</td>
<td></td>
<td>iodine reduction rate is very high until the iodine concentration in gas phase reduces 1/100 of initial state.</td>
<td>same as left</td>
</tr>
</tbody>
</table>

\* partition coefficient is defined as the ratio of iodine concentration in liquid phase to that in gas phase.
Table 4.6.2
Test Results (Removal of Inorganic Iodine by Borax Spray) (Continued)

Recirculation spray mode

<table>
<thead>
<tr>
<th>Item</th>
<th>NSPP</th>
<th>CSE</th>
<th>JAERI Model C.V.</th>
</tr>
</thead>
<tbody>
<tr>
<td>Partition coefficient $^*$</td>
<td>&gt;10,000</td>
<td>&gt;10,000</td>
<td>4,300</td>
</tr>
<tr>
<td>Ratio of iodine reduction rate ($\lambda s(E)/\lambda s(C)$)</td>
<td>2</td>
<td>1</td>
<td></td>
</tr>
<tr>
<td>General tendency</td>
<td>Overall dose reduction factor is smaller in recirculation spray mode than in fresh spray mode.</td>
<td>Iodine concentration reduces to 0.1% of initial value after 1 day of recirculation spray.</td>
<td>Iodine concentration reduces to 0.5% of initial value after 1 day of recirculation spray.</td>
</tr>
</tbody>
</table>

$^*$ partition coefficient is defined as the ratio of iodine concentration in liquid phase to that in gas phase.
Table 4.6.3
Predicted Decontamination Factors (DFs) for Particles in an Ice Compartment(a)

<table>
<thead>
<tr>
<th>Aerodynamic Particle Diameter, μm</th>
<th>Case A</th>
<th>Predicted DF</th>
<th>Case B</th>
<th>Predicted DF</th>
<th>Case C</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>εF = D(D)</td>
<td>X_i = X_0 = D(C)</td>
<td>εF = D.3</td>
<td>X_i = X_0 = D</td>
<td>εF = D.3</td>
</tr>
<tr>
<td>0.001</td>
<td>8.9</td>
<td>202.3</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>0.01</td>
<td>1.4</td>
<td>10.4</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>0.10</td>
<td>1.0</td>
<td>1.5</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>0.40</td>
<td>1.0</td>
<td>1.3</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>1.0</td>
<td>1.0</td>
<td>1.8</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>2.0</td>
<td>1.1</td>
<td>3.7</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>4.0</td>
<td>1.1</td>
<td>11.2</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>8.0</td>
<td>1.4</td>
<td>40.7</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>10.0</td>
<td>1.6</td>
<td>62.6</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>20.0</td>
<td>6.1</td>
<td>248.1</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>50.0</td>
<td>24.2</td>
<td>1528.7</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>100.0</td>
<td>55.8</td>
<td>6063.2</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

(a) gas flow rate = 18.9 m^3/s, T = 293 K; P = 1 atm; u = 1.8 x 10^-4 poise.
(b) ε = fraction of bed filled with ice; F = fraction of ice surface available for sedimentation.
(c) X = mole fraction of gas that is steam; i refers to inlet, o refers to outlet.
Table 4.6.4
Typical Accident Conditions for ESF Atmosphere Clean-Up Systems
in Light-Water Cooled Nuclear Power Plants (1)

<table>
<thead>
<tr>
<th>Environmental Condition</th>
<th>Atmosphere Clean-up System</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Primary</td>
</tr>
<tr>
<td></td>
<td></td>
</tr>
<tr>
<td>Pressure surge</td>
<td></td>
</tr>
<tr>
<td>Maximum pressure</td>
<td>Result of initial blowdown</td>
</tr>
<tr>
<td>Maximum temperature of influent</td>
<td>60 psig (410 kPa)</td>
</tr>
<tr>
<td>Relative humidity of influent</td>
<td>280°F (138°C)</td>
</tr>
<tr>
<td>Average radiation level</td>
<td></td>
</tr>
<tr>
<td>for airborne radioactive materials</td>
<td>10^6 rads/hr</td>
</tr>
<tr>
<td>for iodine buildup on adsorber</td>
<td>10^9 rads</td>
</tr>
<tr>
<td>Average airborne iodine concentration</td>
<td></td>
</tr>
<tr>
<td>for elemental iodine</td>
<td>100 mg/m^3</td>
</tr>
<tr>
<td>for methyl iodide and particulate iodine</td>
<td>10 mg/m^3</td>
</tr>
</tbody>
</table>

Table 4.6.5
Current Knowledge of HEPA Filters

<table>
<thead>
<tr>
<th>Operational Parameters</th>
<th>Extent of Knowledge*</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Normal</td>
</tr>
<tr>
<td>Pressure Differential</td>
<td>VC</td>
</tr>
<tr>
<td>Vibration</td>
<td>G</td>
</tr>
<tr>
<td>High Humidity</td>
<td>F</td>
</tr>
<tr>
<td>Free Water</td>
<td>P</td>
</tr>
<tr>
<td>Chemicals</td>
<td>G</td>
</tr>
<tr>
<td>Radiation</td>
<td>G</td>
</tr>
<tr>
<td>Temperature</td>
<td>G</td>
</tr>
</tbody>
</table>

* VC = Very Good.
  C = Good
  F = Fair
  P = Poor
  NE = Non-Existent.

Little is known on any combination of extreme values of these parameters.

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Table 4.6.6
Current Knowledge Regarding Carbon-Based Iodine Adsorbers

<table>
<thead>
<tr>
<th>Operational Parameters</th>
<th>Extent of Knowledge*</th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>New</td>
<td>Aged</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Normal</td>
<td>Accident</td>
</tr>
<tr>
<td>Pressure Differential</td>
<td>VG</td>
<td>C</td>
<td>C</td>
</tr>
<tr>
<td>Vibration</td>
<td>VG (USA)</td>
<td>C</td>
<td>NE</td>
</tr>
<tr>
<td></td>
<td>P (Others)</td>
<td>NE</td>
<td>NE</td>
</tr>
<tr>
<td>High Humidity/Free Water</td>
<td>VG</td>
<td>VG</td>
<td>F</td>
</tr>
<tr>
<td>Chemicals (Poisoning)</td>
<td>P</td>
<td>F</td>
<td>P</td>
</tr>
<tr>
<td>Radiation</td>
<td>VG</td>
<td>P</td>
<td>P</td>
</tr>
<tr>
<td>Temperature</td>
<td>VG</td>
<td>P</td>
<td>P</td>
</tr>
<tr>
<td>Loading Capacity</td>
<td>VG</td>
<td>F</td>
<td>P</td>
</tr>
</tbody>
</table>

* VG = Very Good
  C = Good
  F = Fair
  P = Poor
  NE = Non-Existant.

Little is known about combinations of extreme values of these parameters.

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CSI - Containment Spray Injection
LPIS - Low Pressure Injection System
HPIS - High Pressure Injection System
CSR - Containment Spray Recirculation
RUST - Refueling Water Storage Tank
CSXX - Containment Spray Heat Exchanger
RPV - Reactor Pressure Vessel
ACC - Accumulator
PGRV - Power Operated Relief Valve
P - Pump
CRD - Control Rod Drives

Figure 4.6.1. PWR - ESFs (Ref. 2)
Figure 4.6.2. BWR - ESFs (Ref. 2)
Figure 4.6.3. Cesium Removal by Sprays in Containment Systems Experiment (Ref. 2)
Figure 4.6.4. Sectional Elevation View Showing the Ice Compartment and Containment Region Boundaries (Ref. 13)

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Figure 4.6.5. Isometric View of Ice Basket Arrangement (Ref. 13)
Figure 4.6.6. Comparison of Predictions With Measurements of I₂ Removal Reported by Malinowski (Ref. 10)
Figure 4.6.7. Air Cleanup Unit (Ref. 11)

Figure 4.6.8. Schematic Drawing—Filtered Venting of Reactor Containment
Figure 4.7.1. The Flammability Limits for Hydrogen-Air-Steam Mixtures at 100°C and 200°C
Figure 4.7.2. Laminar Burning Velocities of Hydrogen-Air-Steam Mixtures at 200°C
Section 5
OPERATOR ACTIONS

5.1 INTRODUCTION

The behaviour of various individuals involved may strongly influence the start and the course of an accident.

First of all, the control room operators may commit diagnostic or manual errors or, conversely, correct an abnormal plant situation (operator error and recovery).

In case of malfunction of safety systems once an accident has been initiated, plant personnel and external support staff may succeed, if enough time is available, in restoring system functionality (repair).

On the basis of the conclusion drawn from recent source term studies and from operating experience, it seems that recovery and repair actions by the operating crew and by external backup personnel have not received an adequate recognition in the analysis of accident sequences in all countries.

The IDCOR project, in particular, has demonstrated that various repair actions are possible and highly effective for all of the dominant sequences.

Operating plant personnel action, with support from the outside in case of emergencies, is a realistic expectation. Both good judgment and experience indicate that, given time, operating and maintenance teams are a powerful and decisive safety factor.

The above statement is especially true if emergency procedures and tools are available at the plant together with state-of-the-art training on Severe Accident Mitigation.
5.2 SURVEY OF NATIONAL POSITIONS ON THE ROLE OF PLANT EMERGENCY ACTIONS IN
THE CASE OF A SEVERE ACCIDENT

5.2.1. The Various Kinds of Operating Procedures

All countries consider operating procedures based on original DBAs. Nevertheless, efforts are currently being made to supplement such procedures to account for intrinsic improvements in the plant safety. Some procedures even address the case of failures of safeguard systems.

All countries also envisage the possibility of new kinds of procedures addressing both design basis and beyond design events. These procedures now being developed are not based on recognition of pre-identified accident sequences, but rather rely on the analysis of a set of symptoms in the plant and on the identification of available means to bring the plant back to a safe state.

In many countries this new approach to plant operator actions is merely at a study level. Finland, France and the United States have implemented such symptom-based procedures to varying extents.

5.2.2 Failure Modes

All countries consider the possibility of failures of safeguard systems. More specifically, provisions are made to cope with situations such as the ATWS, the loss of heat sink, the losses of the main and auxiliary feedwaters, of the S.G.s, and the total loss of power. Some countries add to the previous list the failure of the high pressure injection system, of the low pressure injection system—whether in the direct or in the recirculation mode of operation—with or without the simultaneous failure of the containment heat sink.

5.2.3. Remedial Actions

5.2.3.1 Technical Means. All countries agree to make the best possible use of all existing means related to the plant design, whether they are classified as safety devices or not.

All countries also agree on some limited design improvements such as some instrumentation developments or better control room ergonomics. The latter includes the use of a safety panel with more or less sophisticated piloting aids.

Some countries are considering the addition of supplementary preventive means such as: an improved protection of sensitive systems against external impacts (FRG),
the protection against sabotage, the use of mobile external support means (FRG, France), and some specific backup systems. The United States intends to judge the advisability of such additions on a cost-benefit basis. It is generally agreed that these supplementary means do not have to comply with the stringent rules edicted for design basis situations.

5.2.3.2 **Staff Organization.** All countries intend to have, or already have, a specific organization to respond in the event of a severe accident.

In some countries, the shift supervisor can only rely on on-call personnel. In other countries, the shift supervisor has to pilot the plant according to event-related procedures, whereas a shift engineer analyzes the resulting plant behavior; such an engineer will make the decision to shift to procedures based on a set of symptoms if the previous event-related procedures do not work as expected. When the situation seems to call for it, the head of the plant puts in place an on-site technical backup group, made up of people from the plant, supported by external experts.

In any case, the head of the plant has the full responsibility for making the final decisions as regards on-site accident management, he usually stays in a command room, connected with all the administrative authorities.

5.2.4 **Core Melt Accidents**

For those situations where preventive actions have failed—when the reactor core is highly degraded or already melting—most of the countries envisage mitigation measures, so as to limit external releases of radioactivity.

For the implementation of such measures, the line personnel has to be educated on the severe accident phenomenology and on the expected effects of specific emergency actions on the course of the accident. This is currently done in Germany, in France and in the United States.

In addition, some specific operating procedures are under study (FRG, USA) or already have been approved (France) to control external releases. An example of the latter is the French US procedure which relies on the operation of a sandbed filter system to limit and monitor the release of radioactivity when the containment is vented.
5.2.5 Procedure Validation

In all countries, projected operating procedures or methods in the case of accidents are carefully examined and checked. In most countries they have to be approved by the safety authorities.

Simulators are already playing and will play a large role in the validation of all the technical and human facets of plant operation for conditions that do not involve significant core degradation. All countries are interested in advanced simulators able to represent primary system behavior for all accident cases. Such simulators will be soon in operation in the FRG, Finland, France and the USA.

All countries also conduct exercises involving reactor accident simulations. Some are limited to plant salvage and release limitation considerations while others involve all intervening parties, including the civilian authorities.

5.3 OPERATOR ACTION RATIONALE

The deterministic approach used for the design of NPPs usually identifies the following four classes of operating conditions:

I. Normal operation and transients;
II. Incidents of moderate frequency;
III. Improbable accidents, such as small LOCAs (primary system);
IV. Hypothetical accidents, as large LOCAs (primary system).

Associated with these defined conditions are estimated annual probability of occurrence and associated allowed radiological consequences.

For example, French designers use the following indicative table:

<table>
<thead>
<tr>
<th>Class</th>
<th>Annual Frequency F</th>
<th>Radiological Consequences</th>
</tr>
</thead>
<tbody>
<tr>
<td>I and II</td>
<td>$10^{-2} &lt; F$</td>
<td>authorized releases</td>
</tr>
<tr>
<td>III</td>
<td>$10^{-4} &lt; F &lt; 10^{-2}$</td>
<td>5 mGy whole-body dose</td>
</tr>
<tr>
<td>IV</td>
<td>$10^{-6} &lt; F &lt; 10^{-4}$</td>
<td>0.15 Gy whole-body dose</td>
</tr>
</tbody>
</table>

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For these events the plant design is expected to ensure a proper achievement of the following basic safety requirements:

- reactor scram, cooling down and maintenance in a safe state;
- radioactive matter confinement.

Sets of operating procedures have been established for the above classes of accidents taken into account in the reactor design.

Beyond the aforementioned four classes of operating conditions lies to so-called beyond-design domain.

The question is whether the remote probability events of that realm—which nevertheless can result in severe consequences—are to be investigated or not, and, if so, in what way should they be considered.

Some elements of an answer can be provided once a safety objective has been set. The evaluation process still remains difficult to put into practice due to significant shortcomings in our current knowledge.

In various countries, a probabilistic approach has been used in the beyond-design domain so as to identify possible "weak points" in the design for which provision could easily be made by means of minor modifications and/or adequate staff actions.

In the United States, certain industry groups, such as IDCOR, are presently performing significant work on possible additional technical guidance for degraded core accident management. Besides, the current Emergency Procedure Guidelines (EPGs)—as required by NUREG-0737, supplement 1—are not limited to coping with the traditional design basis events (LOCAs, SGTR, etc.) but also consider multiple failures to various degrees. In particular, the focus in all cases is on maintaining or restoring certain key symptoms (or critical safety functions) to their preferred states.

In France, significant consideration is given to the following beyond-design situations:

- Loss of redundant systems;
- Unanticipated events: how to prevent severe core degradation;
Severe core degradation: how to control the releases of radioactivity and make them compatible with the Off-Site Emergency Plan and with acceptable long term consequences.

The relevant studies resulted in the definition of some emergency procedures (H and U procedures), some of them implying the addition of new, but inexpensive, safety features to the current design.

5.4 OVERVIEW OF THE U.S. EMERGENCY GUIDELINES

5.4.1 General

Emergency Operating Procedure Generic Technical Guidelines have been developed by Owners' Groups representing the four Nuclear Steam Supply System (NSSS) vendor types for use in developing plant-specific Emergency Operating Procedures (EOPs). Each Owners' Group, Babcock and Wilcox (B&W), Combustion Engineering (CE), General Electric (GE), and Westinghouse (W), have taken a different approach to preparing a program for EOPs, but the guidelines are all based upon a similar concept.

When parameters monitored by control room operators approach degraded conditions, the guidelines provide the operator with guidance on how to verify the adequacy of critical safety functions by using parameter measurements and how to restore and maintain these functions when they are degraded.

5.4.2 Combustion Engineering

The CE Owners' Group approach to Emergency Operating Procedure Generic Technical Guidelines divides emergency situations into two types. The first type is general events (e.g., LOCA, SGTR) that are recognizable by correlated symptom sets and can be readily and accurately diagnosed. Since these types of events have been well analyzed and understood, it is possible to write emergency procedure guidelines to optimize the recovery. For these events, Optimal Recovery Guidelines (ORGs) are provided to strategically address symptom sets. The second type is emergencies not easily recognized by a symptom set. This could be due to errors in symptom assessment by the operator, multiple and simultaneous failures in the plant, unanalyzed events, or instrumentation failures. A Functional Recovery Guideline (FRG) is provided for these situations. In case of a misdiagnosed event and subsequent selection of the wrong ORG, each ORG provides for diagnostic confirmation and for critical safety function checks. Thus, if symptoms are not responding to treatments, or if the core is not being adequately cooled, the ORG

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is exited and the FRG is implemented. The FRG provides measures to restore the critical safety functions without event diagnosis.

5.4.3 Babcock & Wilcox

The Babcock & Wilcox Owners’ Group Abnormal Transient Operating Guidelines (ATOG) approach uses heat transfer symptoms as the source of information for operator action to assure adequate core cooling. The three basic heat transfer symptoms are lack of subcooling margin, overheating and overcooling (SGTR is handled as a special case). These three heat transfer symptoms are subgroups of control functions which are reactor core coolant inventory and pressure, and steam generator inventory and pressure. If control of one of these four functions is lost, primary-to-secondary heat transfer will be affected, thereby disrupting one of the three heat transfer symptoms. Therefore, when a heat transfer symptom appears, one or more of these functions are not being properly controlled. Regaining control of the four control functions will restore controlled core cooling. The basic philosophy relies on restoration of stable controlled safety functions rather than immediately diagnosing the cause of the problem (event).

5.4.4 General Electric

General Electric Owners’ Group BWR Emergency Operating Procedure Technical Guidelines are the most symptom-oriented set of vendor guidelines. These guidelines are divided into four major areas: Reactor Pressure Vessel (RPV) Control, Primary Containment Control, Secondary Containment Control, and Radioactivity Release Control.

Operations using the RPV Control Guidelines maintain adequate core cooling, shut down the reactor, and reduce the reactor coolant system temperature to cold shutdown conditions. The Primary Containment Control Guideline protects equipment in primary containment. The Secondary Containment Control Guideline maintains primary containment integrity and protects equipment (such as control rod drive and emergency core cooling systems) in the secondary containment, limits radioactive releases to the secondary containment, and either maintains secondary containment integrity or limits radioactive releases from the secondary containment. The Radioactivity Release Control Guideline limits release of radioactivity into areas outside the primary and secondary containments. Entry into the guidelines may be caused by an abnormal event that may degrade into an emergency. Therefore, entry into a guideline is not necessarily indicative of an emergency that already exists.
5.4.5 Westinghouse

The Westinghouse Owners' Group (WOG) approach is based on (1) Optimal Recovery Guidelines (ORGs) and (2) Function Restoration Guidelines (FRGs). If diagnosis of the event can be readily made, the operator uses the appropriate ORGs to return the plant to normal conditions. During the recovery from a known event, the operator monitors all critical safety functions, using Critical Safety Function Status Trees to assure that all safety functions remain within acceptable bounds. The coordinated use of the guidelines provides a means of continuously monitoring the plant critical safety functions (through use of the Status Trees), permits optimal plant recovery (through use of the ORGs) and directs systematic operator response to conditions outside the areas covered by the ORGs (through use of the FRGs).

5.4.6 Status

NRC staff review of the proposed emergency guidelines for the four Owners' Groups has been completed. The staff SER for BWRs was completed during the first quarter of FY83. The SERs for the three major PWR Owners' Groups were completed during the third and fourth quarters of FY83. A summary of SER findings for each of the four vendors is as follows:

1. General Electric - Acceptable for implementation. Open issues have been identified for longer term resolution.

2. Combustion Engineering - Acceptable for implementation only after being updated with reactor vessel level instrumentation. Also, open issues have been identified for longer term resolution.

3. Westinghouse - Acceptable for implementation. Open issues have been identified for longer term resolution.

4. Babcock & Wilcox - Acceptable for implementation only after being updated with additional ATWS and natural circulation cooldown guidance. Also, open issues have been identified for longer term resolution.
5.5 OVERVIEW OF THE PHILOSOPHY IN JAPAN ON REACTOR OPERA TION MANAGEMENT

The following philosophy is adopted for the management of reactor operation which is specific to Japan.

In order to clarify the responsibility of management of reactor operation and to maintain technically qualified personnel for the reactor operation, the government gives a qualifying examination to each chief nuclear engineer who is to have an advisory and supervisory role over the utility management to maintain reactor safety. In addition to the aforementioned examination for chief engineers, a certificate has been issued since the TMI-2 accident for qualified shift supervisors who are in charge of operators in the control room. The utility is required to have a licensed chief nuclear engineer for each reactor site (of multiple units) and a qualified shift supervisor for each shift crew of reactor operation.

Regular inspection and maintenance have been made with maximum care by implementing the concept of preventive maintenance in order to maintain the high standard of safe and reliable operation of nuclear power plants in Japan. Annual inspection which includes disassembly is required on the important components so as to ensure safe operation as well as a continuous supply of electricity. Voluntary inspection of various components and equipment by the utility is a common practice in Japan for early diagnosis of component malfunction, earlier repair of the failed component and confirmation of normal function of each component.

Education and training of operators and maintenance workers are being performed by the owner of the reactor with the use of a training simulator, an engineering mockup and other aids. The fact that permanent employment is common practice in Japanese society is believed to be one of the contributors to maintaining the high quality of the operators and workers.

In order to further enhance safe operation of a reactor, several developments are being made. One example is the development of the operator supporting system in which a newly designed CRT is utilized for easier recognition and judgment of the plant conditions by operators. Another example is the development of a three-dimensional nondestructive examination technique for achieving higher accuracy.

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5.6 OPERATOR ACTION IN FRANCE

5.6.1 Introduction

The origin of the work which ultimately resulted in the establishment of emergency operating procedures for severe accident management can be placed in 1980. After discussions between the safety authorities and the utility Electricité de France, the need was recognized to consider "beyond design basis" or "at the limit of design basis" accidents in connection with the implementation of feasible off-site emergency plans.

5.6.2 Safety Objective for the Protection of the General Public

In France the following safety objective is prescribed by the safety authorities for the protection of the population:

"The design of a nuclear unit comprising a PWR should be such, that the overall probability that the said unit can induce unacceptable consequences to the public will not exceed $10^{-6}$ per year."

Such a probability level is to be understood rather as an order of magnitude than as a strict limitation.

"Unacceptable consequences" implies that efficient steps have to be taken for the protection of the general public. These include short-term actions intended to safeguard human health and life, such as confinement and/or evacuation, which are part of the Off-Site Emergency Plan. Longer term effects, i.e., health and property damage, must also be considered. Presently, in France, there are no regulatory limits to the absorbed dose requiring public confinement or evacuation. Medical experts estimate that actions must be determined on a case-by-case basis referring to an individual whole-body dose for evacuation ranging from 0.05 to 0.5 Gy.

To warrant a good chance of success in Off-Site Emergency Plan implementation, this must be based on reasonably feasible actions such as:

- A time span of 12-24 hours for evacuation, which implies that no major reactor containment loss of tightness will occur during that time;
- The emergency operations must be limited to a reasonable area downwind of the reactor site. In France, evacuation is commonly envisaged within a 5-km radius and confinement is not usually considered farther than 10 km from the site; other
choices are possible, depending, in particular, on population
density distribution;

- The previous protective actions must be sufficient for the few
days following the actual release of the bulk of the
radioactivity, that is, no late major protective action should
be taken to supplement the earlier ones.

Such constraints in relation with the Off-Site Emergency Plan feasibility, as well
as limitations of individual whole-body exposures, result in an upper bound for
the amount of radioactivity to be released to the environment (source term).

In France, according to the proposed size of the emergency zones and the above
tacit whole-body dose restrictions, this upper limit for the source term is known
as S3 for standard weather conditions on most of the French sites. The S3 limit
corresponds to a delayed release (- one day after the onset of the accident) of a
few thousandths of the initial core inventory of I and Cs, and of 100% of the
noble gases.

5.6.3 Corollary Design Safety Objective for Core-Melt Prevention

As a corollary to the above risk objective, a secondary objective was set up by
the French safety authorities as regards the rules for taking into account or not,
in the design of the plant, the potential threat presented by certain "groups of
events" as they can be identified in the event trees; this objective is stated as
follows:

"When a probabilistic approach is to be used to assess whether a group of events
should be allowed for in the design of a unit, it should be assumed that this
group of events must be allowed for if the probability that it may lead to
unacceptable consequences exceeds 10^{-7} per year."

As similarly stated for the risk objective for the population protection, the
"10^{-7} per year" level is to be considered as an order of magnitude rather than as
a strict threshold.

"Unacceptable consequences "currently refers here to extensive core damage
resulting in reactor vessel melt-through. No assumption is made on containment
efficiency in the application of this objective.

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As a consequence of this corollary design safety objective for core-melt prevention, it appears necessary to assess the consequences of the total loss of those systems which are important for the plant safety.

A list of reference accident situations has been established for the reactor designers. As mentioned in S3, the accidents are classified by categories, depending upon their estimated probabilities. The engineered safety features, which are intended to avoid unacceptable consequences both in the short-term (automatic action) and in the longer term ("A" operating procedures in France)**—should the worst of these situations occur—are designed using, in particular, the single failure criterion.

**Example of French "A" operating procedures:

- A.1.1.: small break LOCA in primary circuit
- A.1.2.: large break LOCA in primary circuit
- A.2.1.: steam pipe break outside the containment,
- A.2.2.: steam pipe break inside the containment (primary circuit cooling maintained),
- A.2.3.: feed water pipe break inside the containment (with an increase in primary circuit temperature),
- A.3.: steam generator tube rupture
- A.8.: rapid primary pressure drop caused by pressurizer LOCA.

However, a probabilistic analysis made in France of the reliability of these systems has pointed out the necessity of taking some limited complementary steps, if the above design safety objective is to be attained. These are:

- either an increase in the redundancy and/or diversification of some systems concerned;
- or the addition of complementary safety features intended to allow for the implementation of the so-called "H" operating procedures. These procedures are aimed at preventing a major core degradation by stabilizing the situation at a safe level during a period of time sufficient for the recovery of the failed function.

5.6.4 The "H" Operating Procedures for Core-Melt Prevention

Although the "H" procedures are design specific, the approach for establishing them is somewhat general. The application to the French designs is given below.

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Other possible implementations may exist or be projected, but could not be reported here due to the lack of information.

The "H" operating procedures, which result from the comparison of reliability studies with a stringent safety objective, are at the limit of the design which means they presently do not comply strictly with normal design rules.

Due to this fact and also to uncertainties linked with human factors, the probability of failure of a particular H procedure cannot be assessed at less than $10^{-2}$ per reactor-year. This means that the implementation of an adopted H procedure allows the corollary design safety objective of $10^{-7}$ per year to be satisfied in the case of a series of failures that would otherwise result in an extensive core damage with a probability of occurrence possibly as high as $10^{-5}$ per reactor-year.

Up to now five H procedures have been defined in France for the following categories of events:

- Loss of external heat sink ($H_1$);
- Total loss of main and auxiliary feedwater to the steam generators ($H_2$);
- Total loss of off-site and on-site power supplies ($H_3$);
- Long-term failure of a vital safety function in case of a LOCA ($H_4$);
- Flood exceeding the millennial flood level, for some river sites ($H_5$).

As an example, the response to a loss of feedwater in the steam generators ($H_2$ procedure) consists in a voluntary opening of the three pressurizer relief valves as soon as the steam generators are dry. The safety injection is automatically actuated. The containment spray is started either automatically when the containment pressure reaches a given upper limit, or manually according to the evolution of the containment temperatures.

If a feedwater system is recovered, the relief valves are closed and the safety injection stopped. If the feedwater remains unavailable, the decrease of primary system temperature and pressure makes it possible, after a time interval which depends on the decay heat, to use the shutdown cooling system (RPA), to stop the safety injection and to close the relief valves.

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Finally the reactor is put in the cold shutdown state. The containment spray system is deactivated as soon as the containment pressure decreases below a given pressure.

5.6.5 The Ul Ultimate Operating Scheme for Core-Melt Prevention

Procedure Ul must allow the best use of all available means, once procedures based on event diagnosis prove faulty to ensure optimum cooling of the Nuclear Steam Supply Steam (NSSS) and to prevent, limit, or retard severe radiological consequences.

The following steps are necessary to achieve this objective:

- Definition of criteria that make it obvious that procedures based on events cannot be applied or are no longer effective;
- Implementation of human resources and equipment for the purpose of monitoring and detecting the crossing of setpoints.
- Specification of actions to be carried out for the specific situation.

Once a significant incident has occurred, such as an emergency shutdown, it appears beneficial to carry out a duplicate simultaneous continual diagnosis outside the operating team following a logical process which is not related to the "event oriented" procedure currently going on:

- The shift Safety Engineer is called to the control room to provide human redundancy and an independent diagnosis; the operators would in fact have some difficulty in modifying their original judgment because they are engrossed by their diagnosis and absorbed by the work of operation;
- The shift Safety Engineer's diagnosis is based on the analysis of the NSSS cooling state, the availability of the safeguard system, and by monitoring the sub criticality of the core and radioactivity of the containment.

This results in a three-stage procedure (figure 5-1):

- Post-incident supervision that allows the Safety Engineer to check that the system is following the initially diagnosed sequence; to ask the operator to carry out limited complementary action or some work of restoring the system; and (as needed) to give orders to abandon the procedure and apply Ul.
- Procedure Ul itself which lays down actions appropriate to the status of the NSSS and of the safety systems;
OPERATOR

Event

Diagnosis

Reactor scram or $A_{sat} < 20^\circ C$

Call safety eng

Actions:

Event oriented procedures

SAFETY ENGINEER

Permanent post accidental survey

Confirmation or additional actions

Decision of applying $U_1$

Criteria not over reached

Safe state

End

$U_1$(Ultimate state procedure)

$P_{st}$

Actions

Systems

Figure 5-1. Post Accidental Survey and $U_1$
- Permanent ultimate supervision that the Safety Engineer carries on after applying U1 and which allows continuous monitoring of the system status.

Following the decision by the Safety Engineer to apply U1, the operator takes all the action specified for each of the NSSS states which are defined by functional and physical criteria.

For example, if no steam generator is available, the operator opens the pressurizer relief valves in order to lower the primary system pressure, to increase injection flow rate, and to attempt cooling by "Feed and Bleed."

Moreover, actions for restoring the safety functions are specified in U1 with the purpose of:

- Restoring system availability by local repair,
- Restoring the functions by any available means, calling in other systems or modifying the circuit configuration.

As soon as the decision is made to apply U1, the Safety Engineer abandons post-incident supervision (SPI) and undertakes permanent ultimate supervision (SPU).

The purpose of SPU is to supply uninterrupted redundancy of diagnosis of the NSSS state in order to confirm to the operator the action he has to carry out.

To facilitate implementation of procedures SPI-U1-SPU, computer aids are being developed. These aids are accessible via the safety panel and the Safety Engineer is provided with his own console to carry out the work.

A preliminary version of procedure U1 is currently enforced on all French 900 MW(e) PWRs.

5.6.6 The "U" Emergency Procedures for Consequence Mitigation of Class-9 Accidents

If the U1 operating scheme fails, it is clear that the containment, which is designed to withstand the consequences of specific design basis accidents, possibly will not be able to support all severe conditions resulting from a core melt-down. This is the reason why a subsidiary safety objective has been defined in France, which can be summarized as follows:
"In the case of a core-melt, the containment should constitute an ultimate line of defense which would reduce, with a reasonable probability, the radioactive releases in the environment at a level compatible with a feasible Off-Site Emergency Plan."

In the treatment of very severe accidents, we have to rely on the knowledge of the physical phenomena which govern the propagation of damage in core, and on the analysis of accidental sequences leading to containment failure.

This is a field where intense research is being carried out around the world, and where more work is still needed before we can reach final conclusions. However the French approach is founded on the following assumptions:

- A steam explosion, energetic enough to break the French large dry containment and lead to a very high radioactive release within a few hours after reactor shutdown, is very unlikely indeed if not completely impossible;

- Hydrogen explosions will not be able to cause significant impairment to the tightness of the containments used in French PWRs; this point results from the analyses, the French have made on the hydrogen risk.

- Containment isolation failures, which may result from leakage at penetrations or failure of isolation systems, must be taken into account. These potential leak paths are to be examined with care, as they presumably constitute major ways for the radioactivity to be released, although some retention is to be expected along these routes (e.g., in the auxiliary buildings). A specific procedure "U_2" has been proposed in order to detect, localize and repair any containment isolation failure. The final version of this U_2 procedure should be enforced in a couple of months on the French 900 MWe reactors. The containment bypass mode due to interfacing systems is associated with this mode of leakage as are potential leak paths through faulty steam generator tubes.

- The basemat melt-through is the least improbable failure mode of the containment, but it is also the one which leads to the lowest radioactive release to the environment. It therefore satisfies the above subsidiary safety objective, except in some 1300 MWe plants with double containment and a drainage system in the basemat, which could create some pathways for gaseous and volatile fission products towards the atmosphere. To prevent this unwanted phenomenon, the U_4 procedure was proposed by the French utility and is still under examination by the safety authorities.

- Finally, depending on the assumptions made about the rate of erosion of the concrete and the corresponding gas production, one cannot exclude a slow but steady increase of pressure inside the containment, and, after some time, a loss of
integrity. The corresponding release of fission products is currently difficult to assess. As an extra precaution, it was decided in France to install on all French PWRs a sand-bed filter which would make possible some controlled and filtered venting of the containment. The target filtration efficiency for aerosols is a factor of 10.

The $U_5$ procedure, which will make use of this filter-vent system, aims at the following objectives:

- to avoid uncontrolled loss of containment tightness;
- to reduce by a significant factor the release of radioactive materials in aerosol form;
- to conduct the filtered gas to the stack and use accident monitoring equipment to measure the amount of radioactivity released;
- to mitigate the impact to the environment of a containment isolation failure—which could not be controlled by $U_2$ procedure—by creating a preferential leak path through the sand-bed filter;
- to avoid overpressurizing the containment in the event a source of water is restored for cooling the corium on the basemat. The containment pressure rise can be accommodated by venting through the filter.

Specifications for the design of the venting system are as follows:

- maximum gas flow rate: 3.5 kg/sec;
- typical gas composition: air (33%), CO$_2$ (33%), CO (5%), steam (29%);
- gas density: 4 kg/m$^3$;
- inlet gas temperature: 140°C;
- inlet pressure (before decompression in the pipes): 0.5 MPa;
- pressure drop through the filter: 0.01 MPa;
- pressure at filter outlet: nearly atmospheric;
- maximum airborne aerosol concentration: 0.1 g/m$^3$; total amount: 5 kg
- aerosol mean diameter: 1 um;
- filtration efficiency: 10 (objective).
The filtering system—about 40 m² in surface area and .8 m of sand height—is not expected to constitute a heat sink large enough for the bulk of the steam passing through to be condensed.

5.7 CONCLUSIONS

Operating procedures aimed at core-melt prevention and consequence mitigation, should a severe accident occur, are currently at various stages of development in many countries. Their elaboration is design specific, and largely depends on the national policies as regards the maximum admissible risk to which the general public could be exposed for remote probability reactor accidents.

In the United States, Emergency Operating Procedure Generic Technical Guidelines for use in developing plant-specific Emergency Operating Procedures (EOPs) have been developed by Owners' Groups representing the four Nuclear Steam Supply System vendor types. Each Owners' Group, Babcock and Wilcox, Combustion Engineering, General Electric, and Westinghouse, has taken a different approach to preparing a program for EOPs, but the guidelines are all based upon a similar concept. When parameters monitored by control room operators approach degraded conditions, the guidelines provide guidance to the operator on how to verify the adequacy of critical safety functions by using parameter measurements and how to restore and maintain these functions when they are degraded.

In France, a reference source term (S3) has been identified which does not result in any significant damage to the health of the general public, insofar as the Off-site Emergency Planning is implemented. The corresponding I and Cs releases are about a few thousandths of their inventory in the core.

Due to the current uncertainties related to the assessment of best-estimate source terms for beyond-design accidents, provisions were made to comply with the above release limitation for I and Cs. This first resulted in the elaboration of the H and U₁ operating procedures, aimed at core-melt prevention, the former being based on the classical event-tree/fault-tree analysis, and the latter on the real-time reactor cooling state identification.

Should these preventive procedures fail, further ultimate procedures for consequence mitigation are currently examined by the French safety authorities.

The principle of the vented/filtered containment has been approved in France and will shortly be implemented on all plants. Such an inexpensive safety feature...
(sand-bed filter), which can reduce I and Cs releases by a factor of ten, is believed needed to ensure a proper implementation of the Off-Site Emergency Plan and to reduce the property damage for releases below the S3 level.

Besides other existing filtered vented containment systems, such as the FILTRA plant at Barseback (Sweden) or the Ontario Hydro CANDU containment system, other filtration devices are under examination from a technical standpoint in some countries (e.g., stainless steel fiber filters in the F.R.G.), but information is lacking as regards their eventual effective use on NPPs.
Section 6

RELATIONSHIP OF CONTAINMENT TYPE TO SOURCE TERM

6.1 INTRODUCTION

The basic function of the containment in a nuclear power plant is to reduce the probability that fission products will be released to the environment if they should escape from the primary coolant system of the reactor. This function may be accomplished in many ways, but in general, the practice is to surround the reactor and much of its directly connected equipment with a mechanical barrier of steel or concrete or both. The piping leading from the reactor directly to other equipment outside of this barrier is equipped with isolation valves which can be closed to prevent release of fission products through that piping.

The volume and strength of the containment must be such as to withstand the release of the primary coolant. The smaller the volume or pressure of the vapor or gas to be accommodated, the less costly the containment barrier can be. For this reason, the concept of pressure suppression, i.e., condensation of escaping steam by passing it into relatively cool water or over ice, has been applied. Particularly, in those cases where the primary cooling system volume, and hence that of the released steam, are relatively large, as in boiling water reactors (BWRs), pressure suppression has been used extensively. Pressure suppression has been used with pressurized water reactors (PWRs) also, but is less usual. In any classification of containment types, the presence or absence of pressure suppression is an important determinant.

The types of containment will be discussed in section 2 of this paper. In subsequent sections, certain auxiliary features which might be found in almost any of the containment types will be discussed, and, finally, the relationship of containment type or features to the source term will be discussed.

6.2 CONTAINMENT CLASSIFICATION AND DESCRIPTIONS

In this section, containment types will be defined, and examples of each type will be cited. It must be realized that the choice of containment type and features for a particular reactor is influenced by many factors, both tangible and
intangible. Therefore, it is not surprising that hybrid containment designs are plentiful. Containments may be roughly classified as shown in table 6.1.

Table 6.1

CONTAINMENT TYPES

<table>
<thead>
<tr>
<th>PWR and PHWR*</th>
<th>A</th>
<th>Atmospheric Without Pressure Suppression System</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>B</td>
<td>Subatmospheric System</td>
</tr>
<tr>
<td></td>
<td>C</td>
<td>Ice Condenser System</td>
</tr>
<tr>
<td></td>
<td>D</td>
<td>Confinement System</td>
</tr>
<tr>
<td>BWR</td>
<td>E</td>
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</tr>
<tr>
<td></td>
<td>F</td>
<td>Pressure Suppression, Mark II</td>
</tr>
<tr>
<td></td>
<td>G</td>
<td>Pressure Suppression, Mark III</td>
</tr>
<tr>
<td></td>
<td>H</td>
<td>Pressure Suppression, European Style</td>
</tr>
</tbody>
</table>

Containment types used in the Soviet Union are not considered in this paper.

6.2.1 Type A, Atmospheric Without Pressure Suppression System

Type A, basically an air-filled chamber at atmospheric pressure containing the reactor, main coolant pumps, pressurizer, steam generator and other equipment is the most common type of PWR containment (figure 6.1). Because of its large size, typically about 70,000 m³ (2,470,000 ft³) gross volume, and the fact that it has no pressure suppression feature, it is usually referred to as a "large dry" containment. It is typically designed for a pressure of about 0.43 MPa (47 psig). In the United States, the containment structure usually is a single posttensioned reinforced concrete cylinder with a curved dome. As a precaution against leakage through the concrete, there is a relatively thin, nonstructural steel liner covering the interior surface of the concrete and secured to it by embedded anchors. The concrete is several feet in thickness for strength and for radiation shielding in case of an accident. In order to assist in reducing the pressure after an accident, the containment usually has two redundant overhead spray systems. The floor and the necessary shielding inside the containment are reinforced concrete which is also used for various equipment supports. The heat capacity of the walls and containment internal structures can be a significant factor in reducing the rise of pressure following a loss-of-coolant.

*PHWR = Pressurized Heavy Water Reactor
accident (LOCA), particularly after that rate of rise has slowed down from its initial fast rate.

Containments for the 900 MWe series of French PWRs are basically similar to the U.S. design just described, being based originally on Westinghouse designs. The French 1300 MWe reactors (figure 6.2) have an inner load bearing domed cylinder (design pressure 0.41 MPa, 45 psig) made of reinforced posttensioned concrete, surrounded by a second concrete shell of smaller pressure capability with an annular space between the two shells. Neither shell has a liner. Any leakage through the inner shell is captured in the slightly subatmospheric pressure annular space and may be disposed of, through a filter, to the atmosphere. Containment spray is provided. Recent information is that provisions will be added to at least some French PWRs for venting the containment if, during an accident, its pressure should rise too high. The vented gases would be passed through a sand or gravel filter in a manner generally similar to that of the Swedish Filtra project.

The standard 1000 MWe West German PWR containment is also of the large dry type. The pressure retaining structure is a large spherical steel shell which contains the reactor, its shield, the steam generators, pressurizer, etc. This spherical shell is contained in a concrete cylinder having a hemispherical dome (figure 6.3). The design pressure for the sphere is about 530 kPa (62 psig). Internal containment spray is not provided in this design.

Some U.S. PWRs resemble the German design in that they have a free-standing steel shell inside a concrete shield building. However, the U.S. containments of this type are cylindrical rather than spherical.

Containment of the 600 MWe series of CANDU PWRs are also similar to the U.S. design described above. A quick-acting high flow rate dousing system is provided which serves to suppress the pressure increase. An internal plastic coating serves to reduce leakage while allowing concrete cracking for pressure relief will beyond design conditions.

6.2.2 Type B, Subatmospheric Containment

There are two types of subatmospheric containments in use. One type of subatmospheric containment is basically a minor variation on the large dry type. In this variation the containment is normally kept at about two-thirds of atmospheric
pressure. This permits reduction of the containment volume by about one-third, with a corresponding cost saving. Containment design pressure is 0.42 MPa (46 psig). The Surrey (U.S.) plant exemplifies this design.

The other principal type of subatmospheric containment is that associated with the Canadian multiple CANDU reactor stations (figure 6.4). The eight-unit Canadian Pickering station is the prototype of this design. In this general design, there are several slightly subatmospheric reactor buildings with pressure operated relief valves. In the Pickering design, the individual reactor buildings are normally isolated from the pressure relief duct by bulkheads containing pressure relief panels. In the other version, exemplified by the Bruce station, the reactors, although in separate buildings, are located in the same containment volume connected by the duct used by common refueling machines which also acts as a pressure relief duct. In either case, the sudden rise in pressure, which would immediately follow a LOCA, causes operation of the pressure relief valves, admitting the pressurized D₂O vapor/air mixture to the vacuum building. The containments normally operate between 100.8 kPa(a) and 97.8 kPa(a). The vacuum buildings are normally maintained between 7 and 10 kPa(a) and contain an elevated dousing tank which is self-actuated by a rise in vacuum building pressure. The dousing water condenses the steam from the LOCA. Igniters are provided within the containment for burning hydrogen as soon as its concentration reaches combustible levels. The positive design pressures for CANDU negative pressure containments range from 142.7 to 197.8 kPa(a). The containments are also designed for negative pressures resulting from spurious activation of the vacuum buildings. Negative design pressures of 51.3 kPa(a) are typical.

6.2.3 Type C, Ice Condenser Containments

The ice condenser containment is basically like a large dry containment except that the entire primary system is enclosed in such a way that any coolant escaping from it is channeled upwardly through an annular ring of steel baskets containing ice at a temperature well below its melting point. The escaping steam is condensed by the ice, which melts, and the residual noncondensable gases (air and hydrogen, if any) emerge into the upper part of the containment (figure 6.5). The condensation of the steam by the ice is intended to prevent the generation of high pressures in the upper part of the containment which, therefore, is designed for the relatively low pressure of 184 kPa (12 psig). Due to the condensation of the steam the containment volume also is smaller than in a large dry containment, having a free volume of 33,800 m³ (1,192,000 ft³). The pressure-retaining member
is a free-standing steel cylinder 35 meters (115 feet) in diameter, with a hemispherical head. Surrounding the steel vessel and separated from it by several feet is a reinforced concrete building which serves to protect the steel vessel externally. Containment sprays are provided to assist in reducing the pressure in the upper compartment.

In the event of a core-damaging accident, hydrogen could be produced and released, through the ice condensers, to the upper compartment. Combustion of a large accumulation of hydrogen could produce a pressure pulse which might damage the steel containment. Accordingly, electrical igniters are strategically placed throughout the containment to intentionally burn the hydrogen before it accumulates sufficiently to cause a damaging pressure pulse.

In addition to condensing the steam, the ice causes deposition of some of the fission products. The size of this decontamination factor is under study at present.

The Sequoyah reactors (U.S.) exemplify this configuration.

6.2.4 Type D, Confinement System

The N-reactor, which produces both nuclear products and steam for the United States Department of Energy at Richland, Washington, utilizes what is termed a "confinement," rather than a "containment" system. The distinction is that in a confinement system, the steam from the rupture of a large pipe (a LOCA) emerges into the "confiner," a large structure designed to withstand about 125 kPa (5 psig) and about 85,000 m³ (3,000,000 ft³) in volume. The confiner is equipped with a number of normally closed vents which open automatically upon a pressure signal, and reclose automatically when the pressure declines, i.e., after the initial out-rush of reactor coolant in a large LOCA has spent itself. Any further discharge from the pipe break is then forced to exit through filters which tend to remove any fission products except the noble gases. In a containment system, all of the escaping steam and fission products are held in a closed space (the containment structure) until disposed of by operator action.

The confinement system is based on the premise that the initial effluent in a large LOCA will be essentially free of fission products and, therefore, may be released to the environment. Only later, after fuel heat-up and cladding damage have occurred, will fission products appear in the effluent, and by that time all escaping flow will be directed through the filters.
In a small LOCA, the coniner vents never open and all discharged steam and fission products either condense in the coniner or are discharged through the filters.

The coniner is copiously equipped with water sprays which aid in steam condensation and in fission product wash-out.

The coniner contains the reactor, steam generators, pressurizer and primary loop pumps, so it is capable of dealing with a LOCA anywhere in the primary system.

6.2.5 Type E, Pressure Suppression, Mark I

The earliest well-established type of BWR containment is the General Electric Mark I. As shown in figure 6.7, this type comprises a drywell, shaped like a light bulb, small end-up, and a wetwell, a toroidal chamber surrounding the wetwell and connected to it by eight large vents. In the event of a LOCA, escaping steam, gases, and fission products flow through these vents from the drywell. The vents terminate beneath the surface of the water in the wetwell where the steam is condensed and where fission products are removed, by their being bubbled through the water, occurs.

The drywell contains the reactor, recirculation pumps, and drywell cooling equipment for normal operation. Feedwater and steam piping enter or leave the containment through pipes which contain remotely operated isolation valves, thus preventing escape of reactor fluids or fission products from the containment when it is desired to isolate the containment. The drywell is normally filled with nitrogen, thus obviating the danger of a postaccident hydrogen explosion. The containment design pressure is typically about 487 kPa (56 psig). The free volume in the drywell is about 4260m$^3$ (150,000 ft$^3$). The water volume in the wetwell is about 3000m$^3$ (106,000 ft$^3$) and the free space in the wetwell is about 4260m$^3$ (150,000 ft$^3$) all for a plant of about 820 MWe capacity.

The primary containment is surrounded by the reactor building, which, although not designed to withstand pressures higher than those due to normal environmental and industrial loads, provides a space within which aerosol removal processes such as settling can occur. The decontamination factor achievable in this space can be significant.
In the event of a LOCA, steam, nitrogen and hydrogen, if any, are discharged beneath the surface of the suppression pool through the large drywell vents. In addition, the main steam lines are provided with safety/relief valves (S/RV) which can be opened either automatically or by the operator in order to discharge steam through the S/RV lines to the suppression pool. The S/RV lines are not shown on figure 6.7; they are smaller than the main drywell vents and, depending on plant size, there may be up to a dozen or more of them.

In some accident sequences, it is calculated that the containment pressure could rise sufficiently to cause the containment to develop gross leaks or perhaps to burst (the structural failure pressure for this type of containment is estimated to be more than 1.7 times its design pressure) (3). The U.S. Nuclear Regulatory Commission has given its approval, in principle, to the operator's venting the wetwell if the pressure becomes high enough to threaten containment integrity, but procedures and provisions for doing so are not yet available in most plants of this type.

The drywell and wetwell are both made of welded steel plate. The drywell is almost completely surrounded by a cast concrete structure, from which it is separated in most places by a gap of about 5 cm (2 inches). The gap is filled with a compressible foam-like material. The wetwell typically sits in a large concrete-walled room. Thus, the drywell is partly free-standing, and the wetwell is entirely free-standing. The lower spherical portion of the drywell shell is covered with a layer of concrete to provide a level floor upon which to work. The drywell also contains numerous grills, ladders, electrical conduits, fans, coolers, etc.

The pressure suppression pool was originally intended merely to condense the steam. The fact that it also removes fission products is now recognized as a very significant extra benefit. The decontamination factor in suppression pools is under intensive study, particularly by EPRI (see section 6.3.14).

6.2.6 Type F, Pressure Suppression, Mark II

This type of containment, a successor to the Mark I type, is similar to it in principle. The Mark II wetwell is a pool which lies directly under the drywell (figure 6.8). Escaping steam, gases and fission products will flow through the downcomers and emerge under the surface of the pool. Both the wetwell and the drywell are inerted with nitrogen. The containment design pressure is typically
about 412 kPa (45 psig). The drywell free volume is about 6300 m³ (222,000 ft³), the wetwell free volume is about 4700 m³ (166,000 ft³), and the water volume in the wetwell is about 3100 m³ (109,000 ft³). The wetwell is made of reinforced concrete with a steel liner. The drywell is, like the Mark I, essentially a free-standing steel structure, surrounded in most areas by a reinforced concrete shield at a distance of about 5 cm (2 inches). The foregoing Mark II figures apply to a plant of about 1100 MWe capacity (4).

The fact that in some Mark II designs the suppression pool extends into the area directly under the reactor creates the possibility that, if molten core debris should penetrate through the pedestal region floor, the debris would fall directly into the suppression pool where it could be quenched. The conditions necessary for quenching are the subject of disagreement among workers in this field at present. Additionally, in some of these versions of the Mark II containment, there are downcomers located directly under the reactor. Molten debris could then flow into these downcomers, probably melting them and falling into the pool and being subject to quenching them. In other versions of the Mark II design, the area directly under the reactor is dry, so that debris falling into this area would not be quenched.

Like the Mark I design, the Mark II has safety/relief valves which discharge beneath the surface of the suppression pool.

6.2.7 Type G, Pressure Suppression, Mark III

This type of containment is the successor to the Mark II type. While similar to Mark I and Mark II in its use of the pressure suppression principle, it embodies some significant differences. As indicated in figure 6.9, the primary containment is a free-standing steel shell, completely surrounded at a distance of a few feet by a reinforced concrete shield building. The steel shell has a design pressure of 205 kPa (15 psig). The reactor is enclosed within the drywell, a cylindrical reinforced concrete structure. The cylindrical portion of the drywell is immersed to a depth of several feet in the suppression pool which, therefore, surrounds it up to that depth. Most of the suppression pool forms an uncovered annulus outside of the lower portion of the drywell. The rest of the suppression pool is inside of the drywell, forming an annulus about 60 cm (2 feet) thick around the inner surface of the lower part of the drywell. The inner wall (the "weir wall") of this annulus extends slightly above the normal height of the pool water. Below water level, the drywell wall is traversed by many large openings at three levels. These openings connect the two parts of the pool.
In the event of a LOCA, the pressure due to the discharged steam, gases and fission products depresses the water surface behind the weir wall, exposing the first level of holes through the drywell wall and permitting the gases to flow through them and into the external portion of the pool. Steam condensation and fission product decontamination thus occur. If the surge of steam is sufficiently large, the level of the water behind the weir wall is further depressed sufficiently to expose the second or third row of holes in the drywell wall, thus permitting more steam and gases to escape into the main portion of the pool.

The Mark III drywell and wetwell are completely inside of the containment. Unlike the Mark I and Mark II containments, the Mark III is not inerted. Instead, compressors are provided which take air from the wetwell and force it through valves into the drywell so that it depresses the water level behind the weir wall and bubbles up through the outer pool. Thus, the drywell atmosphere is, during an accident, continually diluted with wetwell air, and the probability of reaching the lower combustible limit of hydrogen is reduced. In order to assure that hydrogen does not accumulate in the wetwell space, that space is provided with igniters which will cause combustion of hydrogen if its concentration rises to the combustible range.

A supplementary source of water is provided by an additional pool which is located at the refueling level. Water from this pool may be used to replace water lost by evaporation or by being spilled in the drywell.

For a plant of about 1200 MWe capacity the following data apply (reference 5):

- Drywell free volume: 7,800 m³ (275,000 ft³)
- Suppression chamber (wetwell) free volume: 32,300 m³ (1,140,000 ft³)
- Water in lower pool: 3,680 m³ (129,600 ft³)
- Water in upper pool: 970 m³ (34,200 ft³)
- Depth to first row of drywell vent holes: 2.3 m (7.5 ft)

6.2.8 Type H. Pressure Suppression, European Style

All of the BWR pressure suppression containment types so far described are of U.S. origin. Pressure suppression containments from European (Swedish or German) manufacturers are based on the same principles, but are geometrically different. Figure 6.10 shows the Swedish BWR 75 containment. The structure is made of reinforced, partly prestressed concrete and has a steel liner. The design
pressure is typically 540 kPa (64 psig). Earlier German BWR containments were spherical steel shells which contained the reactor, suppression pool, recirculating pumps, etc. Later German designs comprise cylindrical prestressed concrete structures with a steel liner. A typical design pressure is 520 kPa (61 psig). Both the Swedish and German designs have considerable resemblance to the General Electric Mark II containment (6).

It is reported that the Swedish BWR containments are inerted and that the German ones are not.

6.3 Containment Auxiliaries and Features

A number of containment features mentioned only briefly above or not at all may be important for source term considerations.

6.3.1 Filtration

It has frequently been proposed to vent the containment through a filter, and both Sweden and France are proceeding to do so on a limited basis. The Swedish Filtra project was the first commercial filtration concept in use. The Filtra system has one vent pipe from the wetwell vapor space of each of the two Barsebäck reactors. Each pipe contains a rupture disk and an isolation valve of about 60 cm (2 ft) diameter, and also contains a flow restricting orifice of about 100 cm² (0.11 ft²) area. These pipes lead to a gravel bed contained in a cylindrical concrete vessel about 20 meters in diameter and 40 meters high (66 x 131 ft). The volume of the gravel bed is about 10,000 m³ (353,000 ft³), and it is calculated to be capable of condensing and collecting about 500 m³ (18,000 ft³) of condensate. The rupture disks are supposed to give away at a pressure of 650 kPa (94 psig) which is about 1.5 times the containment design pressures. Flow through the gravel bed is in the downward direction. The effluent is vented through a stack.

The French system is basically similar, but instead of gravel is proposed to use sand. The sand bed would be considerably thinner than the Swedish gravel bed. The gas velocity through the sand would be about 10 cm/s. Laboratory experiments demonstrate that a decontamination factor of more than 10 can be expected. The pressure drop through the filter would be about 100 mbars. The sand would be contained in a steel vessel. The filter bed would consist of two layers of sand, of 0.7 and 1.5 mm size.
CANDU negative pressure containment systems also contain filtered venting systems consisting of series high-efficiency particulate and activated charcoal filters. The filtered venting system is used after post-LOCA repressurization of the containment to filter and monitor releases from the station and maintain the containment pressure slightly subatmospheric to control containment out-leakage.

Filters of these type will have little effect on the noble gases. Their effect on the other types of fission products will probably not be large if the gases discharged through the filter will already have passed through the suppression pool (see figure 6.10), where considerable decontamination will have occurred. This will be typical of BWRs, but by preventing drywell structural failure, it is made sure that all the escaping gases go through the suppression pool. This is a major advantage of venting.

6.3.2 Containment Sprays

Containment sprays are included in U.S. and French containments. They accelerate steam condensation, tend to wash out suspended fission product aerosols or vapors, and provide excellent mixing of the containment atmosphere. This last effect is beneficial because it tends to prevent accumulation of hydrogen pockets. The sprays used in PWRs usually contain sodium hydroxide. Some also contain sodium thiosulfate. These materials are intended to enhance the removal of iodine by the sprays, on the assumption that iodine is present as I₂, HI or some other volatile iodine compound. BWR containment sprays are usually just water.

6.3.3 Calcareous versus Siliceous Concrete Aggregate

Most U.S. reactor concrete contains calcareous (limestone or dolomite) aggregate. Most reactors in other countries have used siliceous or basaltic aggregate. In the event that hot core debris should find its way onto the concrete floor of a reactor building, it has been found that the calcareous aggregate would yield more gases (primarily CO₂) than the siliceous or basaltic, but the debris would tend to penetrate the latter types more rapidly. Thus, the choice of aggregate type could depend on whether it is calculated that initial containment failure would be by overpressurization or basemat penetration. However, in many cases, the choice of aggregate has been based on what type was readily available, rather than on its behavior under a pool of molten curium.

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6.3.4 Location of High Pressure/Low Pressure System Interfaces

The "interfacing system LOCA" is one of the standard accidents in source term studies. In this scenario, full reactor pressure is accidentally permitted to be applied to some of the low pressure piping or equipment which is used to supply water to the reactor after it has been depressurized. If the high pressure/low pressure interface is outside of the containment, the interfacing system LOCA (also called the "V" sequence) can provide a path for direct release of fission products from a damaged core to the environment. In some plants, the sensitive high pressure/low pressure interface is located inside the containment, thus reducing the probability of this accident by putting the site of the break inside of the containment.

6.3.5 Design Pressure versus Failure Pressure

There is a substantial margin between design pressure and failure pressure, usually a factor of about two to four. The failure pressure must, however, be determined for each containment individually since design practices vary (7). In this context, failure pressure has been defined for commence to mean 1% strain. Catastrophic failure would be expected slightly above this pressure. Safety analysts prefer to define failure in terms of leakages which, however, comes at a pressure which is hard to quantify.

6.3.6 Leakage Through Containment Penetrations

Containments are required to meet certain maximum leakage requirements. Typically, it is required that the leakage at the pressure of the design basis accident (DBA) not exceed 0.25% per day of the mass of the containment atmosphere, in U.S. reactors. Similar but numerically different requirements apply in other countries. At higher pressures, such as those which could occur in a severe accident, there is no mandated leakage rate which must not be exceeded, but a recent study (3) indicates that leakage at nonmetallic seals under some conditions may become so large that the structural failure pressure of the containment could never be reached. Thus, the leakage would become sufficient to relieve the pressure buildup due to decay heating of the atmosphere and to the generation of CO₂, CO, H₂ and CH₄ from reactions of the hot core debris with the concrete floor of the containment. This type of behavior is, however, quite design dependent. Leakage estimates are difficult to make, so credit for containment seal leakage is usually not taken when estimating the buildup of containment pressure after an accident. While this procedure is conservative with respect to predicting how soon the failure pressure will be reached, it disregards the environmental effects.

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of leakage before the failure pressure is reached, which is nonconservative. Hence, detailed analysis is necessary to determine the net effect of high pressure leakage on the source term.

Temperature is important in ascertaining the behavior of elastomeric seals, and some accident sequences involve relatively high containment temperature. However, incomplete experimental evidence indicates that elastomeric seals continue to function at temperatures well above their laboratory-measured softening temperatures. Data on elastomeric seal materials appear in (3).

6.3.7 Leakage Through Left-Open Piping or Ducts

Containments typically have purge piping, which may be of the order of 60 cm (2 ft) in diameter, and other openings which may be accidentally open or fail to close at the time of an accident. Such events have been analyzed by the IDCOR project (8). Paradoxically, the fission product release in a large LOCA with ECC failure may be less if the containment is impaired (i.e., a valve is left open) than if no valve had been left open. This is because the initial surge of steam contains little fission product activity and is released quickly through the open duct. This leaves less steam to carry out fission products later when they emerge from the primary system. In accidents other than a large LOCA, additional analyses would be necessary to completely evaluate the effect of an impaired containment, which can be of considerable significance.

Because of the large number of penetrations through containment walls, the large size of some of the penetrations, and the fact that some such penetrations are subject to being open through both equipment failure and human error, containment penetrations can be a significant contributor to frequency of release of primary system contents to lower pressure systems or, ultimately, to the environment. Moreover, some such losses of containment integrity can be part of a chain of causation of core-damaging accidents (see section 6.3.4 of this paper where the V sequence is described).

6.3.8 Steam Generator Tube Failure

In PWRs the steam generator tubes form a part of the primary coolant system pressure boundary. Rupture of these tubes is infrequent (fewer than a half dozen failures have occurred in U.S. PWRs), but accident analysts recognize the possibility that such rupture might be caused by a reactor pressure surge during a core-damaging accident. Such an event could lead to an environmental release of
fission products via the secondary (steam) loop which then becomes part of the containment and its relief valves. Some fission product deposition would undoubtedly occur in the steam generator or in the secondary loop, thus reducing the fission product release to the environment. The size of the decontamination factor thus achieved is not well established, but probably is less than a factor of 10 (9).

It has been suggested that fission products deposited in the steam generator tubes could, through local heating, be the cause of tube rupture. This suggestion is controversial, and so far lacks a satisfactory answer.

6.3.9 Time of Containment Failure

Current belief is that the key fission products would exist in the containment as aerosols. Experimental data indicate that the mass concentration of an aerosol will decrease by an order of magnitude in the first hour or two. Thus, if containment failure can be postponed for a few hours after the emergence of the fission products into the containment, significant reduction in the source term will ensue. The IDCOR finding that containment failure will usually, in fact, be delayed until well after fission product emergence is an important factor in showing that some of the source term estimates of the Reactor Safety Study (10) are erroneously high.

6.3.10 Containment Failure Mode

The way in which an overpressurized containment fails is important since it determines whether the release to the environment will be a sudden puff release, expelling most of the airborne fission products in the containment over a short time, or a gradual release through cracks or seams which open gradually as the internal pressure builds up.

In the case of concrete containments, there is considerable experimental evidence (7,11,12,16) that cracks will develop gradually in the concrete itself. There is evidence also that tears will develop in the steel liner, if any. These tears may connect up with cracks in the concrete, thus providing paths for the escape of gases from the containment. The ability of artificial cracks to transport gases has been measured, and rough calculations indicate that the cracks would be of sufficient size to halt the rise of containment pressure long before catastrophic failure of the structure had occurred (16), provided there is adequate connection between the liner tears and the concrete cracks.
It has been suggested that as the containment atmosphere flows through the cracks in the concrete, any suspended aerosol particles will tend to be filtered out, thus providing additional decontamination (20). Experimental data on similar systems show that such filtration can occur, but quantifying its effectiveness is difficult; it, therefore, remains an unevaluated conservatism in most analyses.

If the liner is less ductile than the structural rebars or tendons which are the main load carrying steel members of a concrete containment, it would be expected that liner tearing would precede rebar or tendons failure, and leak-before-break would result. If the liner is more ductile than the load-carrying steel the rebars or tendons would be expected to fail first unless there are local regions in which liner strain consist to fail first. In many cases, the liner is more ductile than the load carrying steel, so, without a detailed analysis, one cannot predict whether the containment would fail in the leak-before-break mode or in the catastrophic mode.

The failure modes of free-standing steel structures, such as those of the Mark III, ice condenser, German large dry and some U.S. large dry containments, as well as some portions of the Mark I and Mark II containments, have also been studied. Four experiments at the Sandia National Laboratory (13,14) in which steel vessels, representing steel containments in various degrees, were pressurized with nitrogen until failure occurred, led to leak-before-break in one case and to catastrophic failure in three cases. It thus appears that in steel containments leak-before-break cannot be relied upon, although it may occur.

The effect of a catastrophic failure of a steel containment on the source term has not been quantitatively evaluated so far. Analyses by the IDCOR program indicate that pressures sufficient to cause rupture of a steel ice condenser containment would occur only after many hours from the beginning of the accident. During that time interval natural aerosol removal processes would produce a large decontamination of the containment atmosphere, thus reducing the fission product release. The delay in reaching failure pressure would also provide time for corrective action by the operator, thus reducing the probability that the failure pressure would actually be reached.

In certain BWR sequences, e.g., the failure to scram or ATWS sequence, the situation is different in that it is calculated that a pressure sufficient to cause containment failure could be reached in less than 1.5 hours and before fuel damage if corrective action were not taken. For this reason, owners of such
plants in the United States have subscribed in principle to a procedure in which the wetwell is vented to the environs through a filter system (the standby gas treatment system). The gases released in this way experience a high degree of decontamination (except for the noble gases) in their passage through the suppression pool, and failure of the steel containment is averted. The capability of controlled venting presently exists in only a few plants, if any.

6.3.11 External Threats to Containment Integrity

In some countries, e.g., Germany, France and Switzerland, there are design requirements relating to external threats. For example, in Germany and Switzerland, it is required that the containment be capable of withstanding the crash of a military airplane such as the F-104. In others, such as the United States, reliance is placed on locating power plants away from paths of aircraft take off or landing. While massive containment, such as those in Germany and Switzerland, might make failure during a nuclear accident less probable than otherwise, actual analyses of the effect on the size of the source term have not been reported. The mechanism by which such an accident would significantly increase the size of the source term is not apparent.

6.3.12 Auxiliary Buildings

All light water reactors have lines through which water can be supplied to the reactor for emergency cooling, and usually these lines originate outside the containment. These lines, themselves, therefore, would constitute potential paths for the escape of fission products directly from the reactor to the environment were it not for the fact that they characteristically terminate either in the reactor building or in a separate auxiliary building. The reactor building and the auxiliary building thus can exert a substantial decontaminating and delaying effect on fission product release to the environment. In effect, the auxiliary buildings become part of the containment system in some accident sequences. Further study in this area is needed.

The passage of the escaping steam, gases and fission products through the high pressure piping itself can result in some decontamination, or at least in aerosol agglomeration into larger particles which are more likely to be removed in the various decontamination processes in the auxiliary building. These agglomeration processes have been observed in the Marviken and LACE experiments.
The fate of the fission products once they reach the auxiliary building is a function of the design of that building. Characteristically, the postulated accident comprises the accidental opening or failure of valves in high pressure lines connected to the reactor, thus permitting the release of the reactor contents at high pressure into low-pressure piping, pumps or valves connected thereto. While it might be possible for bursting of the low-pressure piping or valves to occur, it is estimated that the usual failure point would be in pumping or valve seals. The pumps and valves are frequently located in a pit where water from the break would accumulate, eventually submerging the break itself. Any exiting steam or gases would then experience some decontamination by bubbling through this water.

Fission products which escape this type of decontamination would emerge into the auxiliary building. Such buildings vary widely in size, tightness and geometry, so the decontamination factor experienced by the escaping fission products must be determined for each plant, but substantial decontamination factors are possible.

As earlier noted, this accident sequence is known as the V sequence, or the "interfacing system LOCA."

6.3.13 Inert Atmosphere

With few exceptions, the primary containments of the Mark I and Mark II type are inerterd, i.e., the air initially in them is largely replaced with nitrogen, so that the residual oxygen content is about 5% or less by volume. This procedure was originally instituted as a means of preventing combustion of any hydrogen which might be released in the containment as a result of radiolysis of water or zirconium/water reaction. As a result, failure of these primary containments due to hydrogen combustion overpressure is obviated. Such an event could occur only after failure of the primary containment due to steam or hydrogen overpressure and then only in the secondary containment (reactor buildings) of these types of plants.

6.3.14. Decontamination Due to Suppression Pool

As noted earlier, the suppression pool was intended originally just to condense steam, thereby making it possible to use a smaller primary containment in the General Electric BWRs. Subsequently, it was found that considerable fission product decontamination also occurs in the suppression pool.
It is found that the decontamination factor is highly sensitive to the size of the aerosol particles, a minimum decontamination factor occurring at a particle diameter of about 0.2 \( \mu \)m. Particles larger or smaller than this are removed much more efficiently. It has, therefore, been necessary to estimate the distribution of sizes of the aerosol particles which would enter the pool in a severe accident. Final results of this work are not yet available, but it is clear that substantial decontamination of the exiting gases will occur in the suppression pool.

The fission product aerosol decontamination factor attained in the suppression pool is also a function of water temperature. At room temperatures, relatively large (much more than 1000) decontamination factors have been measured. At boiling temperature, considerably smaller decontamination factors are measured. The decontamination factor is also a function of pipe size and submergence depth. In general, larger pipes and smaller submergence depths lead to lower decontamination factors. The relationships among these parameters are not yet completely worked out, but it is clear that, overall, the suppression pool is a major decontaminating agency. For engineering purposes, and in lieu of detailed and precise information, the IDCOR program originally assigned decontamination factors of 600 to the drywell vents and 1000 to the safety relief valve discharge lines. Since then, a mechanistic method of calculating decontamination factors has been developed for future use (18). The overall decontamination factors thus calculated will be lower than those originally assumed, but not enough lower to make a significant difference in the calculated source term.

6.4 INFLUENCE OF CONTAINMENT TYPE ON SOURCE TERM

As noted earlier, the basic function of containment is to reduce the probability or amount of fission product or other radioactive nuclide release to the environment if fission products should escape from the primary coolant system in an accident. In view of the fact that analyses of hypothetical reactor accidents (15,17,18) show a spectrum of fission product release fractions and probabilities, it is reasonable to ask whether the type of containment is an important factor in determination of the character of the fission product release.

The important characteristics of an accidental fission product release are:

- Its composition. Which species make up the bulk of the biologically significant release?
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- Its amount. What fractions of the relevant fission products are released?
- Its energy. Are the fission products carried out with highly heated gases or high pressure steam to a considerable altitude, or are they released gradually at building level?
- Its timing. Does the release come without warning, or is there an appreciable interval between the beginning of the accident and the release of fission products to the environs?
- Its probability. How likely is the release?
- Its final resting place. Is it all discharged to the outdoor atmosphere or does it remain in-plant or on-site so as to complicate recovery or cleanup operations?

The influence of containment type on each of these characteristics will be discussed briefly. In this discussion, only above-ground containment failures will be considered, since, aside from the effect of concrete type, as already discussed in section 6.3.3, there is little difference among containment types in regard to underground fission product releases.

**Composition**

The fission products released to the atmosphere are conventionally assumed to contain all of the noble gases plus calculated fractions of the iodine, cesium, tellurium and sometimes other fission products such as lanthanum in small but significant quantities. The nongaseous fission products are believed to be predominantly in aerosol form. The composition of the aerosols is determined by their source and method of formation. That is, they may be formed from vapors freshly out of the primary loop, from vapors regenerated from fission products deposited in the primary-loop, from vapors produced during the core/concrete interaction or from vapors formed during a direct containment heating event. The compositions of the aerosol particles arising from these four sources may vary according to their respective sources, but, once formed, are not likely to be changed by any of the containment-determined decontamination processes. That is, to a first approximation, containment type would be expected to have little effect on source term composition.

**Amount**

An important factor in estimating the overall fractional release of fission products to the environs is determining what fraction of the fission products released from the primary cooling system are actually subjected to the

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decontamination processes inherent in the containment. Once out of the primary cooling system, the fission products may escape directly from the containment, as in the V sequences, or they may remain in the containment and undergo decontamination by sedimentation, spray removal, suppression pool scrubbing, etc. Or, some part of that portion which is initially discharged into the containment may bypass some of the decontamination agencies, e.g., there could be a by-passing of the suppression pool in a Mark III containment, or an escape from a large dry containment through an open valve or other path to the environs. Or it could be that the sequence under investigation required an assumption that the containment fail at a given pressure, or that it be initially impaired. Thus, while it would be possible to estimate a decontamination factor (or factors) inherent in a particular type of intact containment system, containment rank would vary with the type of accident analyzed.

Energy

The energy with which fission products are released to the atmosphere determines the height to which they will rise and thereby affects the radiation levels which will be experienced on the ground at a given distance from the accident. The energy may be in the form of high pressure steam or of hot, buoyant gases. In either case, gross failure of the containment must occur before plume energy can become important. No advantage of one type of containment over another is apparent in this respect, since the containment would make no contribution to the plume energy.

Timing

Given that the containment functions as normally expected, there would be a delay of a few hours or more before fission product release from the containment would occur. The reason for the delay depends on the particular accident sequence assumed and on the type of containment.

For example, a large LOCA could occur without warning but the development of sufficient pressure to fail the containment would take several hours, regardless of containment type (15).

An interfacing systems accident would open the primary system to the auxiliary building (not designed as a containment structure), but it would require hours to reach core uncovering due to continued operation of the ECC systems. The same is true of PWR systems.

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An ATWS (failure to scram) accident would produce early containment failure in a Mark I or Mark II type BWR containment, but core uncovering would not occur until later, and fission product release from the containment would not be expected until three or four hours after the beginning of the accident.

Thus, for various reasons, it is expected that there would be a period of a few to many hours between accident initiation and fission product release, regardless of containment type.

**Probability**

The probability of a fission product release to the environment is dominated by the probability that an accident will occur and the probability that the containment would not perform as expected. The probability of occurrence of an accident is determined by factors other than containment type. The probability that the containment will not perform as expected depends on what is expected. Existing analyses (15) show that it would be advantageous if venting of the wet well in Mark I and Mark II containments could be counted on with a reasonable probability. At this writing, it is not clear that the physical ability to vent the wet well, without damage to some plant equipment, exists. Also, a source of containment spray water, independent of the usual power sources would be advantageous. These matters are the subject of on-going consideration in both industry and regulatory quarters. Thus, in these respects it appears that given the occurrence of an accident, the probability of release of fission products to the environment may depend on containment type; specifically, the Mark I and Mark II containments as presently configured may have a higher than average probability, but the question is not yet resolved.

**Final Resting Place of Fission Products**

The final resting places of those fission products remaining in the plant after a severe accident can be important in considering recovery or cleanup operations. If the nonreleased fission products are in shielded locations, postaccident activities will be facilitated. In the case of all modern PWR containments and the Mark III containment, there is a heavy shield wall surrounding or integral with the reactor building. In the case of Mark I containments, there is shielding around the suppression pool, but if containment failure should occur in a way to permit fission products to enter the reactor building (the secondary containment) there would be little shielding of them. There are plant-to-plant differences in auxiliary buildings also which would influence the final location and shielding of
fission products, but these are not associated with particular containment designs. The chance that an operator may mistakenly export liquid wastes containing fission products from the containment to the auxiliary building has to be considered as part of the problem in this case.

Conclusion

A definitive quantitative analysis of the ways and degrees in which containment type influences the source term would be a complex task requiring not only containment analyses but accident analyses and consideration of differing plant designs and operating procedures as well. Such an analysis is clearly beyond the scope of the present work.

From the foregoing considerations, it could appear that some types of containments appear inherently likely to be associated with source terms which, in one way or another, are more easily dealt with than those from other containment types. But this could be misleading because those containments are associated with different types of reactors and therefore must withstand different accidents. A meaningful comparison would require putting each type of reactor in each type of containment, and evaluating their performance, failure probability and, of course, cost. So far as type is concerned, any of the enumerated types of containment can, if competently designed and built, yield acceptable source terms. The differences in performance among existing containments of different types are due mainly to the fact that they are now being tested against accidents which are more severe or of different kinds than those for which they were designed. For example, the trend toward venting of containments is a direct result of having now to consider accidents more severe, and in some cases more probable, than a decade ago. If one were starting a new at the present time to select a type of containment in which to house a particular type of reactor, the type selected might not be the same as the type in which existing reactors of the subject type are housed. But, by definition, the type selected would yield an acceptable source term; the choice of type would be made on some other basis such as the cost of achieving an acceptable source term.

6.5 REFERENCES


4. As reported on LaSalle County Station in Nine Mile Point Nuclear Station Unit 2 Final Safety Analysis Report, 1985.


Part 2

REPORT ON
THE TECHNICAL STATUS OF
FILTRATION AND CONTAINMENT ATMOSPHERE
CONTROL SYSTEMS FOR NUCLEAR REACTORS
IN THE EVENT OF A SEVERE ACCIDENT

(Last Revision October 1986)
1. Foreword

Air cleaning and Containment Atmosphere Control is achieved by means of Air Cleaning Systems (ACSs) which include Engineered Safety Features (ESFs) that may provide one or more of the following functions:

- maintenance of a negative differential pressure between the containment and the free atmosphere to minimize uncontrolled leakage to the environment;

- reduction of the hydrogen concentration in the containment (purge or recirculation / recombine system);

- cooling of the essential components located inside the containment - recirculating systems are generally used for this purpose and may incorporate filtration equipment, which serves to reduce contamination inside the containment and to reduce any losses of radioactive materials in leakage to the environment;

- minimizing the radioactive content of deliberate discharges to the environment: a standby (emergency) gas treatment system (SBGTS) is usually provided for this purpose.

The potential performance of ACSs in accident situations was reviewed in a report published in 1984 and prepared by the CSNI Group of Experts on Air Cleaning and Containment Atmospheric Control Systems (GENAC). The report considered the problems which are outstanding together with their significance and concluded that:

- the conventional Design Basis Accident (DBA) concept may not have been applied in the past in a manner appropriate to considering the performance of ACSs in accident conditions. In particular, the DBAs may not have represented the "worst conditions" from the point of view of the ACSs,

- the work on source term definition for severe accidents had not yet advanced sufficiently to define the severe challenges that may be anticipated and which may lead to the failure of the ACSs,

- in any case there was a lack of data on the performance of ACS components in the kind of conditions which might be encountered in an accident, this being a natural consequence of the lack of suitable test facilities. Even so it seemed likely that development of new, more robust equipment would be necessary.
The Group produced a whole serie of recommendations designed to respond to these inadequacies and emphasized the need for ACSs to be examined not only at the filtration component level but also in terms of the response of the entire system including instruments, ducts, fans etc... In this way consideration will be given not only to externally generated influences, humidity, temperature, radiation and dust-loading, but also to influences generated within the system such as increased pressure drop due to dust loading of filters and consequent changes in fan performance. While progress has been recorded in a number of areas since GENAC's report was prepared further work is certainly required.

Progress has already been achieved in particular areas; for example in the development of:

a) improved conventional HEPA filters by the use of material better suited to higher temperatures and by an increase in structural strength;

b) packed fibre filters made from media and casing materials capable of resisting high thermal and mechanical loads, e.g. metal cases with metal or ceramic fibres fabricated into mats;

c) deep-bed filters with gravel and sand in suitably robust casings.

Equally, new filter test facilities to simulate accident conditions continue to be commissioned and test results are becoming increasingly available. However, the adequacy of these facilities and the validity of results obtained will require to be further analysed in the light of progress in LWR accident source term studies which is the subject of the present report.

Moreover, the implications of these same studies will also require to be considered from a much wider point of view in order to assess the performance of the many other components of an ACS and of the system as a whole in an accident situation. Consider for example the potential effects of a high aerosol concentration on typical containment fan-coolers, these might suffer from:

- blockage of protective screens, where fitted;
- deposition on the finned heat exchanger tubes;
- deposition in fan bearings and motor housings.

All or any of these effects would at least degrade the performance not only of the coolers but also of other equipment whose continued operation depends on the coolers functioning correctly.

Just as such coolers are only one example of the many component parts of air cleaning systems potentially at risk, aerosol loading of the containment atmosphere is only one example of the way in which such risks can arise. Exposure to elevated levels of moisture, temperature, pressure and radiation, singly or in combination, could introduce many additional failure pathways. The detailed implications of the source term review will require to be investigated for each and all of these hazards to components of ACSs taken individually and as entire systems.
2. Questionnaire on Filtered Vented Containment Systems (FVCS)

A survey was conducted among members of PWG4 on systems especially designed for mitigating the consequences of severe accidents with core melt in pressurised water reactors (PWRs) by depressurising the containment and filtering releases to the environment.

The questionnaire had been designed so as to draw up a list of the different systems installed or envisaged in PWRs, evaluate the technical design basis and find out why they had been established and how they were to be used. Since only one country, France, stated that it was envisaging the introduction of such systems at PWRs, the survey was extended to include other types of thermal neutron and water cooled reactors in the comparison, such as boiling water reactors (BWRs) and Canadian CANDU reactors as they were likely to use systems with a similar purpose.

3. Overview of countries' current positions

Fourteen nuclear bodies replied to the questionnaire distributed to PWG4 members, so that a fairly comprehensive overview of country positions could be drawn up regarding the principle of introducing special filtered vented containment systems in reactors in order to mitigate the consequences of severe accidents.

FRANCE was the only country which clearly stated that it was considering the introduction of such system in PWRs. This is the outcome of a general study request by the French safety authorities concerning the introduction at PWRs of ultimate measures aimed at preventing or mitigating the consequences of severe accidents with core melt, although the probability of such accidents is estimated to be very low (less than 10^{-6} per reactor per year). As for the mitigation of consequences, the objective is to reduce releases for all physically conceivable severe accidents to a level compatible with the feasibility of Specific Emergency Plans (source term referred to as S3 in the French safety analysis). The ultimate measure U5 which covers the basic filtration system is not the only ultimate measure planned for reducing the consequences. There are two others: U2 (detection and treatment of abnormal loss of containment integrity faults) and U4 (minimization of releases to the atmosphere in the event of meltthrough).

The other countries with major installed nuclear capacity including PWRs (United States, Federal Republic of Germany, Japan) have not at the moment such a clear position. In short, the arguments on which the current positions of the safety authorities of these three countries are based are as follows:

- UNITED STATES : NRC (Nuclear Regulatory Commission) bases its position on an economic criterion, namely that previous United States studies on FVCS indicate an unfavourable cost/benefit ratio for PWR's. During the past six months to one year, evaluations have been underway on the feasibility of venting BWR Mark I containment during a severe accident. As part of the NUREG 1150 Risk Rebaselining Study, the NRC staff is completing a preliminary study on venting which will be included in that report. The BWR Mark I owners group is also exploring this issue.
- FEDERAL REPUBLIC OF GERMANY: According to the general objective to restrict the core meltdown accident consequences as long as possible to the reactor containment, it is suggested, to provide a device for controlled depressurization in order to avoid late overpressure containment failure. In the actual case of the NPP BROKDORF the authorities have required the installation of such a device in a sense of a pre-condition to be fulfilled. This venting device also will comprise a filter system in order to minimize the radiological source term. A general solution for already existing LWRs is being in discussion.

- JAPAN: JAPAN has conducted an analysis dividing accidents beyond the design basis into two groups, depending on their probability of occurrence. It estimates that for accidents of probability above 10^{-6}, referred to as SEA (Site Evaluation Accidents), there is no risk of loss of containment integrity and hence no need to introduce FVCS. As for accidents of probability under 10^{-6}, the risk of which is taken as residual risk, Japanese regulatory authority, at present, do not require any specific countermeasures such as FVCS.

If the survey is extended to other types of reactor which are likely to raise similar problems in the event of severe accidents with core melt, i.e. in our opinion thermal neutron and water cooled reactors, three systems may more or less react as FVCS: the Swedish system FILTRA, the Canadian EFAD system and BWR pressure suppression pools.

The Swedish system FILTRA can be regarded as an FVCS proper. The system was fully installed in 1985 at the Barsebäck 2 X 580 MWe BWR plant (type Mk II) as a result of the decision taken by the Swedish Government following discussions on TM12 by a government committee, this measure constituting a prerequisite for the plant's operating licence as from 1st September 1986.

Meanwhile, the Swedish Government launched a broader study programme concerning the ten or so other Swedish nuclear plants, the final objective being that any measure decided should be operational by end 1988. The programme is divided into two parts: the RAMA (Reactor Accident Mitigation Analysis) project aimed at providing a generic technical basis for the evaluation and design of the measures and completed early 1985, and projects carried out by the utilities in terms of plant specific studies and implementations of the safety measures developed. So far the Swedish Government has not taken any decisions concerning plants other than Barsebäck.

The vacuum building and the emergency filtered air discharge system EFAD installed at CANU plants owned by one of the Canadian utilities (Ontario-Hydro) may not be regarded as an FVCS proper. The chief purpose of this system is to prevent uncontrolled leakages of radioactives products out of the containment following a LOCA by providing a preferential controlled and filtered leakage path. Optimum operation is possible only under atmospheric conditions (temperature, pressure, gaseous composition) less than or as severe as those corresponding to design basis accidents and provided certain power supplies are maintained in permanence. Nevertheless, although not designed for this purpose, the Canadian system would undoubtedly be efficient to some extent in the event of an accident beyond the design basis.

BWR pressure suppression pools are not true FVCS either. Originally, their purpose was to maintain internal pressure at an acceptable level in the event of steam leaking into the containment, by bubbling steam in a
pool to condense it. However such a pool obviously provides some potential for retaining any radioactive products carried by the steam, but as far as is known it has not been optimised in this respect (e.g. a fairly high water level would be required in order to trap aerosols and chemical additives would increase the capacity of the water to retain molecular iodine).

In conclusion the main grounds for installing the last two systems mentioned above (CANDU vacuum building, BWR pressure suppression pool) are clearly not derived from a request by the safety authorities of the countries concerned relating to the handling of severe accidents with core melt.

4. DESCRIPTION AND TECHNICAL BASIS OF THE SYSTEMS

4.1 French basis filtration system

The FVCS currently studied by France is a small sandbed filter; it is not designed to trap steam through condensation. The main specification is an efficiency coefficient (defined as the ratio of the mass entering the filter to the mass leaving it) of more than 10 for aerosols.

The reference characteristics of the fluid to be filtered which are taken into account in the filter design have been established on the basis of studies carried out in 1979 by the Institut de Protection et de Sûreté Nucléaire (IPSN), Commissariat à l'Energie Atomique (CEA), using two scenarios AB and TMLB (A = LOCA, B = total loss of electrical sources, T = transient, M = loss of feedwater supply to steam generators, L = loss of emergency feedwater supply to steam generators). They are as follows: maximum mass flow = 3.5Kg/s, steam content = 29 per cent, temperature = 140°C, total aerosol content = 5 kg, maximum aerosol concentration = 0.1g/m², diameter = 1 um.

Although Electricité de France (EDF) has not yet finalized the design of the system to be introduced at its PWRs, a 40m² and 80cm high filtration bed is likely to be fitted in a leaktight metal tank isolated by valves and linked to the internal containment atmosphere by already existing pipework leading to the stack. The pressure of the gas must be reduced to about 0.1 MPa prior to entry into the filter, the maximum pressure drop in the filtration bed being 0.01 MPa.

IPSN is conducting a research and development programme on sandbed filtration. The PITEAS programme is divided into two parts, the first now being completed and the second in progress:

- The first part consists of small scale laboratory tests aimed at identifying and characterising the filtration medium.

- The second part includes tests carried out in a vessel of a few cubic metres, aimed at determining the system's effectiveness and suitability under more representative conditions, particularly scale.
Following the experimental results obtained in the first part, a log-normal particle size sand (mean diameter 0.6 mm and standard deviation 2) meeting efficiency and pressure drop specifications was selected as the filtering material for the basic filtration system to be installed in reactors. Furthermore, although conducted on a small scale and for short filtration times, the experiments did not indicate any risk of filter clogging by aerosols or the condensed steam or appreciable letting out of trapped aerosols under foreseen conditions of use.

4.2 Swedish system FILTRA

The FILTRA system installed in 1985 is applied at the two BWR units of the BARSEBACK power plant. Its design is based on the following major principles:

- If used during a severe accident with core melt the containment and the filtration system must hold back 99.9 per cent of the total radioactive product content of the core except for rare gases;

- Passive operation during the first 24 hours of the accident must be guaranteed.

It is worth noting that with this system, Swedish engineers not only aim to mitigate the short-term consequences of accidents but also the long-term consequences, i.e. contamination of the ground by long-lived radioactive products.

The accident sequences used as reference for designing the FILTRA system are as follows:

- Reference N°1: transient with simultaneous prolonged loss of all power supplies;

- Reference N°2: break with simultaneous failure of the pressure suppression function and prolonged loss of all power supplies.

The first case is representative of the group of sequences where core melt precedes containment overpressure. The second case is representative of the group of sequences where containment overpressure precedes core melt. The total probability of occurrence of each group has been estimated at under $10^{-7}$ per reactor per year. For the depressurisation and filtration functions of the FILTRA system, reference N°2 is the more binding one.

The FILTRA system is illustrated in Figure 1. It consists of two systems: one for depressurising the containment and another for filtering the gas released.

The depressurisation system for each reactor includes the following:

- Main piping of diameter 0.6m connected:
Upstream, to the wetwell through a blowout disk calibrated at 0.65 MPa (or 0.15 MPa above the design pressure of the containment, but calculations have shown that up to this limit the containment shows a mainly elastic behaviour) followed by two isolating valves in series and blocked in the open position during normal operation;

Downstream, to the single pipe leading into the filtration system;

- Two other 0.15 m diameter pipes connected:
  - Upstream, to the drywell through two isolating valves in series blocked in the closed position during normal operation (these are used for manually depressurising the containment even if filled with a large quantity of water or prior to the rupture of the blowout disk calibrated at 0.65 MPa);
  - Downstream, to the above-mentioned 0.6 m diameter pipework before entry into the filtration system.

In order to prevent the accumulation of condensation water, the inlet pipework into the filtration system is linked to a drainage system including tanks large enough to store all the water condensed during the first 24 hours of the accident.

After passing through the filtration system, the gas is released via the stack of one of the units. The exit pipework system includes an isolating valve blocked in the open position during normal operation, which can completely isolate the filtration system from the environment after use.

For the filtration system, Swedish engineers selected a gravel bed as the filtration medium. The gravel stones have an average diameter of 2.5cm and are contained in a concrete tank of volume $10^4$m$^3$, diameter 20 m, height 40 m and wall thickness 1 m. The tank has a leaktight metal inner lining. The gas inlet pipe rises axially inside the tank and is topped by a distributor spraying the gas evenly over the upper surface of the gravel bed. The latter acts not only as a filter but also as a heat sink condensing the steam: 500 m$^3$ of water may be stored in this way.

During normal plant operation, the depressurisation and filtration systems are filled with nitrogen to prevent any hydrogen combustion during the initial phase of the accident and also any development of organic matter in the gravel bed. Finally, the filtration system tank is designed so as to resist any hydrogen fire during a subsequent stage in the accident when oxygen might enter it.

Under the FILTRA project, an experimental verification and analytical programme have been implemented in order to evaluate the behaviour of the filtration system in terms of heat, fluid mechanics, retention of radioactive products and hydrogen combustion. Tests at the KARLSHAMN centre have shown that the gravel bed can condense all the steam likely to be produced during the first 24 hours of an accident sequence without
risk of clogging and without any prohibitive increase in pressure drop. Aerosol and iodine vapour (molecular and organic iodine) retention has been studied at STUDSVIK by means of laboratory or intermediate scale experiments using sand and gravel. The application of the models developed by means of these experiments to the accident sequence N°2 mentioned at the beginning of Section 3.2 and of the calculation codes concerning the behaviour of radioactive products in the containment has led to the following prediction for releases out of a plant fitted with the system, expressed as a fraction of the core inventory: molecular iodine 0, organic iodine less than $10^{-2}$, aerosols less than $10^{-4}$. Studies conducted by the Swedish Defence Research Institute at Stockholm on combustion modes for a mixture of air and hydrogen in a gravel bed have shown that the gravel has a considerable mitigating effect on combustion velocities and pressure surges.

The FILTRA was developed and implemented at the BARSEBACK plant without extensive investigations of possible alternatives in order to provide strengthened environmental protection as soon as possible, considering the proximity of this plant to urban areas.

According to the government guidelines filtered containment venting systems are required also for the other nuclear power plants in Sweden. However, alternative, smaller and less costly aerosol retention systems than the FILTRA gravel bed are being considered still, at unchanged level of required protection of the environment.

4.3 Vacuum building and emergency filtered air discharge and containment system in CANDU plants operated by Ontario-Hydro

A filtered vented containment system has been introduced at all CANDU plants operated by Ontario-Hydro. It is designed for protecting the entire plant containment from the short-term effects of any pressure transient following a LOCA and for maintaining containment depressurisation in the medium and long terms, with controlled filtered releases to the atmosphere.

It consists of two systems: a vacuum building and an emergency filtered air discharge and containment system (EFAD).

A typical concrete vacuum building (specifically the one at DARLINGTON) is shown on Figures 2 and 3. It is linked to the general plant containment covering not just the reactors but also a fuel loading/unloading tunnel linking the four units. It has an internal volume of $10^3$ m$^3$ equivalent to the plant containment internal volume. Under normal operation:

- The vacuum building is isolated from the plant containment by self-actuating valves in the closed position;

- Slight underpressure is maintained in the plant containment;

- A "vacuum" is maintained in the vacuum building (pressure $7.10^3$ Pa).

In the event of a LOCA in one of the reactors, the release of steam into the plant containment automatically closes the containment isolating valves, the valves linking the plant containment to the vacuum building
open and spraying in the vacuum building cools and condenses the steam. In the event of a full clean break in the heavy water inlet pipe to the reactor, this system enables the pressure in the plant containment to be brought below atmospheric pressure within a few dozen seconds and kept there for several days or hours depending upon the containment impairments assumed in the analyses.

The emergency filtered air discharge and containment system EFAD is used in a post-accident situation to maintain depressurisation in the plant containment. It includes two identical filtration banks isolated by protection screens. It may take in air either from the general plant containment (from the pipe leading to the vacuum building), the vacuum building itself or from a vacuum pump outlet of the vacuum building. The air filtered by the system may be directed to the plant stack or partly or totally recycled. Each filtration bank consists of:

- A droplet separator to prevent downstream filters from being damaged;

- An air heater to reduce relative humidity from 100 to 70 per cent without increasing the temperature too much, thus preventing any loss of efficiency in iodine filtration by the downstream activated carbon filter and also not subjecting downstream filters to a temperature which might damage them;

- A prefilter to retain large particles;

- A HEPA filter (High Efficiency Particulate Air Filter) to retain particles with a minimum efficiency of 99.97 per cent for particles of diameter less than 0.3 µm;

- An activated carbon filter for retaining iodine in the form of vapour;

- A second HEPA filter;

- A fan.

Whereas the operation of the vacuum building is initially passive, the operation of the emergency filtered air discharge and containment system presupposes non-automatic action and the availability of power sources. The system has been designed in the context of design basis accidents and its suitability for handling accidents beyond the design basis has not been evaluated.

4.4 BWR pressure suppression pool

There are various models of BWR all including a pressure suppression pool designed in order to depressurise the wetwell in the event of a LOCA thus enabling condensation of steam by bubbling through water and simultaneously filtering some of the gases. In the survey, the most detailed reply to the questionnaire was supplied by Switzerland. Consequently, the design of the type Mark 1 BWR operating since 1972 at MUHLEBERG (Switzerland) is described below.

Figure N°4 is a cross-section of the reactor showing the two pressure suppression systems included in the secondary containment of the reactor building:
- The first consists of a circular section metal ring containing 2 120 m³ of water and directly linked to the wetwell;

- The second consists of a polygonal section concrete ring-shaped bunker located in the peripheral area of the reactor building and lined with a synthetic resin to make it leaktight. It contains 1 000 m³ of water and is linked upstream by pipes fitted with valves to the internal space of the secondary containment and also to the first pressure suppression system, releasing gases downstream to the plant stack. The second system can therefore depressurise the secondary containment of the reactor or the first system when the water in the latter reaches boiling point; at the same time it filters some of the gases.

The information supplied gives no indication as to whether these systems operate automatically or passively.

As far as filtration efficiency is concerned, it should be noted that:

- the gas bubbling height in the pool is very low (less than 1 metre according to Figure N° 4);

- Apparently Switzerland is not conducting any work on evaluating the retention of radioactive products in the pool water.

5. Procedures and criteria for the application of the systems

No guide seems to have been drawn up on the use of the Canadian system and the BWR pressure suppression pools in the event of an accident beyond the design basis.

Since the design of the French basis filtration system has not yet been finalized, the application procedure (US) is not yet defined. However, French engineers are considering controlled, in order words non-automatic, application.

Sweden is obviously in the lead, since the FILTRA system has been introduced and is designed for use in the event of an accident beyond the design basis. It is automatically triggered by the rupture of a blowout disk calibrated at 0.65 MPa (see Section 3.2). The application procedures, which are currently being drawn up, also include the following manual actuation criteria:

- When the containment pressure reaches 0.45 MPa and continues to rise beyond that threshold;

- When the pool temperature rises above 95°C;

- Through simultaneous high pressure and high activity signals in the containment;

- Through a high containment water level signal.

It seems that filtered gas releases into the environment are controlled, in other words non-automatic.
6. Advantages and disadvantages of the systems

The advantages of such systems are obvious since they prevent irreversible loss of containment and provide a preferential leakage route to the environment which is both controlled and filtered.

It remains to be settled whether they might increase the risk of releasing radioactivity into the environment and whether their cost is prohibitive.

It should be noted that the cost aspect is not considered in the same way or given the same importance everywhere. The United States base their current position on an unfavourable cost/benefit analysis (see Section 3). Sweden has not taken the same stance since it has installed the FILTRA device at one of its power plants at a cost of US S 20 million, in line with the estimate made by the United States for such systems. This does not mean that Sweden would also disregard this aspect in the case of other power plants. France does not take the cost/benefit aspect into consideration in the principles on which its safety analyses are based whether in the nuclear or non-nuclear fields. In the case of accidents beyond the design basis in nuclear power plants, the French safety objective is as follows: the probability of a severe accident with unacceptable consequences for the public must be less than 10^{-6} per reactor per year ("unacceptable" means any measure that would restrict the public such as confinement or evacuation). However France has obviously taken cost into account in its initial FVCS selection since it has decided to study a basic filter "of reasonable size".

The second aspect, namely the impact of FVCS on risk, is obviously very closely governed by FVCS design. Here again, since the only true FVCS in existence is the Swedish system FILTRA, assessment may only be based on the Swedish study alone. It has been shown that unscheduled bursting of the blowout disk during an accident whether within or beyond the design basis could not have a significant effect on other safety functions and that in any case releases to the environment remained low since they would be filtered by the gravel bed. Nevertheless it seems that this analysis remains fairly deterministic and has not yet progressed enough to allow a true comparison between the probabilistic risk analysis without the FILTRA system and that with the FILTRA system, taking into account the possible failures of the different components of the system.
FIGURE 3
VACUUM BUILDING CROSS SECTION
FIGURE 4 : BWR UNIT AT MUHLEBERG (SWITZERLAND)

3 : First pressure suppression system
4 : Second pressure suppression system
Part 3

REPORT ON
THE TECHNICAL STATUS OF
ACCIDENT CONSEQUENCE ASSESSMENT

(Last Revision March 1987)

Note:

This Report is based on information available at the
beginning of 1986. Some aspects of this work will need
to be revised in due course in the light of the results
of Chernobyl studies.

This part of the report was typed at the Institute for
Energy Technology (IFE), Kjeller, Norway
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3. ACCIDENT CONSEQUENCE ASSESSMENT

3.1 Introduction

It is important to note that this chapter describes the status pre-Chernobyl.

The risk to the general public from the operation of a nuclear installation arises from two distinct sources, namely radiation exposure from the discharge of radioactive effluents during normal operation and that from potential accidents which may result in the release of radioactive material to the environment. There is a marked difference both in the nature of these risks and the manner in which they need to be assessed. The exposure from the discharge of effluents is an unavoidable consequence of the operation of the installation, as designed and constructed, although this exposure (and thus risk) is maintained at very low levels through the general requirements to keep discharges "as low as reasonably achievable". In many cases the discharges from nuclear installations are "quasi-continuous" and, with knowledge of the surrounding environment and habits of the population, reliable estimates can be made of both the exposure of individuals and the population as a whole. Detailed methods for evaluating such doses (and risks) have been developed and documented, e.g. (Ref. C079).

A different approach must, however, be used to assess the risk from potential accidents. Despite the elaborate precautions taken in the design, construction and operation of nuclear installations, there will always remain the possibility, however small, of accidents which may lead to the release of radioactive material to the environment. The amounts released may range from the trivial up to a significant fraction of the activity inventory of the facility under consideration, although the most recent research results on reactor accident source terms make the possibility of major releases very remote. In contrast to the discharges of effluents in normal operation, which occur in a controlled manner and more or less continuously throughout the lifetime of an installation, the occurrence of accidents can only be predicted on a probabilistic basis. Similarly, while the exposures (and thus risk) from effluent discharges can be reliably estimated, those from postulated accidents can only be determined probabilistically.

The techniques of probabilistic risk assessment (PRA) have been used to evaluate the risk from potential accidents e.g. (Refs. KE82, MI76, NU75). PRA can be conveniently categorised into two broad disciplines, plant safety analysis and accident consequence assessment (ACA), and the interfaces between them are illustrated in (Fig. 1).

From the plant safety analysis, predictions emerge of the frequencies with which particular accidents may occur and the characteristics of the radioactive material released in each case.

The characteristics of the released material are frequently referred to as the "source terms" and include the amount of each radionuclide released, the physical and chemical forms of the released material, the duration and heat content of the release etc. The accident "source terms" and their frequencies of occurrence are input to the accident consequence assessment (ACA) which, in turn, predicts the consequences in terms of health and economic impacts and the frequencies with which they might occur.
3.2 The essential elements of an accident consequence assessment

Compared to a plant safety analysis an accident consequence assessment appears inhomogenous, and it may be difficult to see the connection between the various parts of the assessment. The reason is that required information input to the assessment spans a very wide range of separate and unconnected disciplines; from meteorology to properties of building materials, from data on the health effects of radiation to data on population, land use and agricultural practices.

The essential elements of an accident consequence assessment (ACA) are shown schematically in (Fig. 2). For a given release source term and weather condition, an atmospheric dispersion model is used to calculate a matrix of air and ground concentrations for each nuclide released. These are converted by a dose model into inhalation doses and doses from external exposure. The ground concentrations are also input to a model for calculation of dose from inhalation of resuspended radioactive materials, and a foodchain model which computes matrices of contamination levels for the specified foods. The various dose and food contamination level matrices are then used, in conjunction with population and agricultural census data, to calculate the following typical "end-points":

- number of late and early effects, using a health effects model and countermeasures model.
- interdicted land areas and food restrictions using a countermeasures model.
For a given release source term the environmental consequences will be quite different for different weather situations. The calculations are therefore repeated for a series of different possible weather conditions sampled from a meteorological data base. The results of these calculations are a collection of consequence magnitudes (one for each possible weather condition). Since the probability of each weather

![Flowchart Diagram](image)

**Fig. 2** Schematic representation of an accident consequence calculation
condition can be determined from weather statistics, the probability of occurrence of each consequence magnitude is known. It is customary to present the results as complementary cumulative distribution functions (CCDF); as a presentation of the accident risk. The calculation procedure described in this paragraph will produce a conditional CCDF, since it is assumed that a specific accident has taken place.

If it is desired to assess the accident risk of a specific plant, it is necessary to take all types of accidents that may lead to releases to the environment into consideration. Using plant safety analysis methods, a range of possible source terms with connected annual frequencies of occurrence are determined. For each source term a CCDF can be assessed, as described in the previous paragraph. These can then be coupled, using the annual frequencies of occurrence, to form one CCDF. Thus, the production of the overall CCDF can involve a very large number of calculations of environmental consequences under different conditions. It is, therefore, important to keep the number of different source terms to a minimum; i.e. assure that they are appreciably different (result in appreciably different consequences), and that efficient computational techniques are employed in the accident consequence assessments.

There are two broad categories of offsite consequences which may result from an accidental release and which need to be evaluated in an accident consequence assessment. First, there are the health effects in the exposed population and its descendants, and second, there is the economic (and social) impact of countermeasures which may be taken to reduce the exposure and thus the health implications in the population.

Experience to date has shown that accidental releases to the aquatic environment make only a small contribution to the overall risk from nuclear installations. For this reason, an ACA is generally concerned with assessment of the consequences of atmospheric releases. Likewise, it has been shown that the risk via deposition of atmospherically released materials onto water bodies is normally considerably smaller than when deposition takes place on land.

The analysis methods in this area have been fairly stable for some years, but the need for certain specific improvements is recognized. Improvements in some recent ACAs have included more detailed treatment of certain meteorological and topographical effects, and enhanced models for the mitigation of radiation exposure (e.g. evacuation and sheltering). And a particular need has been felt for more realistic treatment in the conditions in urban areas. In particular, needs for improving specific aspects may result from the ongoing activity in the source term area.

3.3 Historical background

The first accident consequence assessment (ACA) model in the newer "generation" was CRAC, which was developed in 1975 primarily for the U.S. Reactor Safety Study (Ref. NU75). Since then CRAC, and several modified versions of CRAC, has been used extensively in a number of countries. The first application of CRAC outside of the U.S. was in a risk study carried out for the Norwegian Government Commission on Nuclear Power in 1977 (Refs. N078 and TV79).
Completion of further ACA models followed rapidly (development had in many cases been initiated some years before publication of NU75). The Finnish model ARANO was the first model to be completed after CRAC. It was first employed in a risk-benefit analysis (Ref. MI76 and SA76). Then followed several new models, and among them the German model UFOMOD, developed primarily for the German Risk Study (Ref. DE79). Since 1975 modelling capabilities have been improved, model and parameter evaluation studies have been performed. The existing models have also been applied to provide guidance in such areas as emergency planning and reactor sitting.

Much of the recent development of methods and data has been carried out within the framework of the 1980-1984 CEC (Commission of the European Communities) Radiation Protection Research Program and within the Nordic Safety Program (participants Denmark, Finland, Norway and Sweden).

Within the CEC program a two-year plan was drawn up in 1982, referred to as MARIA (Methods for Assessing the Radiological Impact of Accidents) to review and develop the probabilistic consequence assessment methods in use in the countries of the European Community. The principal contractors for this work were NRPB (National Radiological Protection Board, United Kingdom) and KFK (Kernforschungszentrum Karlsruhe, Federal Republic of Germany) who were jointly responsible for the following areas of investigation:

- atmospheric dispersion including topographical aspects, mesoscale dispersion, wet deposition and meteorological sampling;
- external exposure from ground deposition;
- transfer through foodchains;
- uncertainty analysis.

A number of supportive study contracts on particular aspects were placed with other organisations within the European Community. Moreover, within the CEC radiation protection program certain other research contracts dealt with related topics.

The work within the Nordic Safety Program was carried out in the years 1981 to 1985. It was administrated by the Nordic Liaison Committee for Atomic Energy, and partially funded by the Nordic Council of Ministers. A new joint Nordic Safety Program will be carried out in 1985-1989, also sponsored by the Nordic Liaison Committee for Atomic Energy/Nordic Council of Ministers.

Some of the aspects investigated in the ACA-related work in the Nordic Safety Program are:

- Compilation of a data bank of Nordic fall-out data.
- The shielding and filtering effects of buildings.
- Deposition in urban areas.
- Natural decontamination of roofs.
- Run-off and decontamination under winter conditions.

Some important current development work:

- Development of a new ACA model, at Sandia National Laboratory, as part of the MELCOR program, expected to be completed in 1986.
- Development of a new ACA model (CONDOR), jointly by CEGEB and SRD, expected to be completed in 1987.
- Continuation of the MARIA program, sponsored by CEC, planned through 1988.
- Improvement of ACA data and models, within the Nordic Safety Program, planned till summer 1989.
- Improvement of ACA data and models, as part of the German Risk Study, Phase B.

Short descriptions of these programs are given in section 3.8.2.

3.4 Computer programs for accident consequence assessment

A number of probabilistic accident consequence assessment codes have been developed in different countries. A list of codes is shown in (Table 1). This information has been taken from (Ref. IN84).

<table>
<thead>
<tr>
<th>NAME OF CODE AND REFERENCE</th>
<th>ORGANIZATIONS PARTICIPATING IN SOME OR ALL BENCHMARK PROBLEMS</th>
<th>COUNTRY</th>
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<tr>
<td>IFDM/DOSDIM (KR78)</td>
<td>SCK/CN Automotive Environment Service</td>
<td>Belgium</td>
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<tr>
<td>CONSEQU (MA80)</td>
<td>Atmospheric Energy of Canada Limited</td>
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</tr>
<tr>
<td></td>
<td>(SN80) Atomic Energy of Canada Limited</td>
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</tr>
<tr>
<td>PLUCON2 (TH80)</td>
<td>RISØ National Laboratory</td>
<td>Denmark</td>
</tr>
<tr>
<td>ARANO (NO79)</td>
<td>Technical Research Center</td>
<td>Finland</td>
</tr>
<tr>
<td>ALICE (MA81)</td>
<td>Commissariat a l'Energie Atomique</td>
<td>France</td>
</tr>
<tr>
<td>UFOMOD (DE81)</td>
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<td>FRG</td>
</tr>
<tr>
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<tr>
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<td>CRAC (NU75)</td>
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<td>Norway</td>
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<tr>
<td>UNIDOSE (KA79)</td>
<td>Studsvik Energiteknik AB</td>
<td>Sweden</td>
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<tr>
<td>CHRONEX/SPAD (ZU78)</td>
<td>Federal Office of Energy</td>
<td>Switzerland</td>
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<td>TIRION (KA76)</td>
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<tr>
<td>CRACUK</td>
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</tr>
<tr>
<td>NECTAR (MA82)</td>
<td>Central Electricity Generating Board</td>
<td>UK</td>
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<tr>
<td>MARC (CL81)</td>
<td>National Radiological Protection Board</td>
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<tr>
<td>BADAC1 (BE78)</td>
<td>National Audubon Society</td>
<td>USA</td>
</tr>
<tr>
<td>CRAC (NU75)</td>
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<td>USA</td>
</tr>
<tr>
<td>CRAC2 (RI83)</td>
<td>&quot;</td>
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</tr>
<tr>
<td>CRACIT (CO81)</td>
<td>Pickard, Lowe and Garrick, Inc.</td>
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<tr>
<td>NUCHAC (KA81)</td>
<td>Science Applications Inc.</td>
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</tr>
</tbody>
</table>

Table 1 List of accident consequence assessment codes

* Added to the list.
Broadly these codes all have a similar structure, but differ in detail at the submodel level. There are a number of different reasons for these differences. The more important of these reasons are:

- There has been development over the years in the ACA area, and a code will reflect the conditions at the time at which it was developed (modified by the extent to which it incorporates the more recent developments).
- The design of a code (including the sophistication of the models used) is influenced by the intended application of the code.
- Submodels in different codes may be developed on the basis of the same basic data, but may differ in the exact interpretation of these data (these differences in interpretation are rarely of a fundamental nature and are largely a reflection of the uncertainty in the original data on which the models are based).

The ACA codes mentioned above have found wide application in several areas of nuclear safety including risk assessments of nuclear installations, assessing the merits of different design options, safety goals, siting and emergency planning. In these applications the existing codes have proved to be broadly adequate. However, in the context of their increasing use in a decision-making framework there is a need for further code development with the main objectives of gaining a better understanding of the uncertainties involved and reducing these where appropriate, and enhancing the range of applicability of the existing codes.

3.5 Limitations and uncertainties

The largest uncertainty in off-site consequences is due to the variation of the accident source, but even in the case of fixed releases of radioactive material, the consequences will vary considerably with the conditions pertaining at the time, in particular with the prevailing meteorological conditions, the season, the location and habits of the population etc. It is, for example apparent that, for a nuclear installation located on the coast, the consequences of an accident would be very different in the following two situations: when the released material is dispersed inland over an urban area, and when the released material is blown out to sea. For any given release, therefore, there will be a spectrum of possible consequences, each having different probabilities of occurrence determined by the environmental characteristics of the release location and its surroundings. In assessing the risk from accidents, therefore, both the release and the consequences which might result from it must be treated in a probabilistic framework, in contrast to the approach adopted when assessing the risk from effluent discharges during normal operation.

Besides carrying through the uncertainties related to the previous steps of a PRA (the plant safety analysis) the off-site accident consequence assessment has its own limitations, and these stem principally from two sources: uncertainty in the models and uncertainty in the data required by the models.
Modelling uncertainty arises from:

- An incomplete understanding of the phenomena and processes involved in the transport of released radioactive materials to man, and the health, environmental and economic consequences that result.
- Simplifications made in the modelling process to reduce computational costs, complexity and requirements for input data.

Data uncertainties arise from:

- Problems associated with the quality, availability and appropriateness of the data.
- Statistical variability.

In addition to uncertainty in the models and data, the weather conditions during and following a release can have a very large impact on predicted consequences. The variability of meteorological conditions is usually addressed by treating weather as a stochastic parameter.

Validation of a complete ACA code is not possible although at the sub-model level this is possible to varying extents (e.g., atmospheric dispersion, transfer of nuclides through foodchains). In these circumstances code intercomparisons have an important role in attaining some measure of quality assurance in the techniques used. Intercomparisons at various levels of detail have been undertaken both internationally and bi-laterally (Refs. TH78, TH79, IN84 and HE83). While differences have emerged between the predictions of the various codes their magnitudes have in general been consistent with the inherent uncertainties in the predictions. In some cases the measure of agreement is greater than might have been anticipated and this is often due to the adoption of the same modelling approach in the respective codes in areas associated with significant uncertainty. This further reinforces the need for uncertainty estimates in accident consequence predictions if undue reliability is not to be attached to the results.

Quite recently two projects have been carried out to gain increased knowledge of the uncertainties in offsite consequence assessments; one in Germany (Ref. HO85) and one in the United States (Ref. AL85). One more uncertainty assessment is scheduled for completion in late 1986. This assessment is performed in Sweden as part of the joint Nordic Safety Program.

A full assessment of the uncertainties in consequence analysis has not been performed. A number of studies have presented estimated uncertainties in the calculated distributions of health and economic consequences based on subjective considerations or on a limited number of parametric sensitivity studies. In parametric sensitivity studies, consequences are calculated for a range of plausible values of a single key variable or model while other variables are held fixed, usually as some nominal value. Uncertainty estimates based on parametric studies are conditional on the values selected for all other (non-varied) parameters. In addition, the effects of possible synergisms between variables are ignored because only a single variable at a time is varied.
While most previous efforts to estimate uncertainties in offsite consequences have been based on subjective judgement or on parametric sensitivity studies, the two current efforts are developing techniques for performing uncertainty analyses that permit the simultaneous variation of the variables under study. The use of "multivariate" input allows the effects of possible synergisms between variables to be included in the analysis. Under the auspices of the Commission of the European Communities, a major effort is underway jointly at Kernforschungszentrum Karlsruhe (KfK) and the National Radiological Protection Board (NRPB) to develop and apply uncertainty analysis techniques for consequences models. Two multivariate techniques have been compared, statistical tolerance confidence limits (STL) and response surface replacement. To date, the two techniques have been applied to study uncertainty in the atmospheric dispersion model of MARC (JO82) and UFOMOD (BA84) and also the food-chain submodel of MARC. The current conclusions of the study, described by Hofer and Krzykacz (HO85), were that for the submodels examined, the STL approach provided at relatively little cost estimates of uncertainty in output parameters as well a determination of the important parameters. However, the relatively large number of required computer runs (59) may not be affordable when a full consequence model is examined.

At Sandia National Laboratories in the U.S., the MELCOR project is identifying and compiling techniques suitable for performing uncertainty and sensitivity analyses on computer codes used for analysis of severe accidents. As part of MELCOR, Iman and Helton (IM85) compared three techniques for performing uncertainty and sensitivity analyses with complex computer models. The three techniques were: Latin hypercube sampling (a stratified Monte Carlo technique), response surface replacement, and differential analysis. Overall, the authors judged the Latin hypercube sampling (LHS) method to be well suited for use with complex computer codes such as those being developed for MELCOR. Their observations with respect to the advantages of the LHS technique were: 1) Uses input from any multivariate structure and can be modified to incorporate correlations between variables, 2) the entire range of each variable is utilized, which is important if there are thresholds or discontinuities in output (e.g., thresholds for early health effects), 3) directly produces estimates of output distribution functions, expected values and variances, 4) permits a variety of sensitivity analyses techniques (e.g., stepwise regression, partial correlation, scatter plots), 5) does not require extensive modification of the model under analysis, and 6) is designed to make efficient use of the number of computer runs required. A demonstration of the use of the LHS technique on the reactor accident consequence model has recently been performed (AL85).

3.6 Potentially important pathway parameters

At the first meeting of GRECA (Group of Experts on Accident Consequences) it was decided to carry out a pathway parameter survey/evaluation. It was argued that considerable effort had been spent in recent years on improving accident consequence modelling techniques, but that there were still large uncertainties associated with many of the pathway parameters.
The purpose of the conducted survey has been to collect the parameter values (and associated data) used in the various countries represented, with their range of uncertainty, evaluate these uncertainties, and determine the sensitivity of consequence assessments to the uncertainties. The survey will be published separately.

The parameters included in the survey are:

- Influence of the source term characteristics on offsite consequences and priorities for future research.
- Estimation of accident consequence uncertainties and evaluation of sensitivities.
- Decontamination (including effectiveness and cost)
- Radionuclide behavior in urban areas (including run-off and rainstorm events).
- Shielding factors (particularly, but not exclusively, in urban areas).
- Filtering effects of houses and deposition indoors.
- Wet and dry deposition (velocities).
- Migration of radionuclides in soil.
- Deposition and decontamination under winter conditions.
- Agricultural pathways (deposition on crops and root uptake, not including animal products data).
- Meteorological sampling techniques.
- Sensitivity of dose pattern to grid size for population.

The importance of many of these parameters to offsite consequences has been examined in the large number of sensitivity studies that have been performed throughout the world. For severe accidents, uncertainties in early health effects are quite large and have generally been found to be driven by uncertainty in the source term and in the assumed emergency protective measures. Uncertainties in long-term health effects (latent cancers) and offsite costs are generally smaller than for early health effects. A good discussion of uncertain parameters can be found in the U.S.N.R.C.'s PRA Procedures Guide (NU83).

The recent uncertainty analyses performed at KfK and NRPB (H085) found the uncertainty in predicted areas of ground contamination and in predicted levels of ground contamination at a particular distance to be driven by the uncertainty in dispersion parameters and in wet and dry deposition parameters. However, these studies have not yet examined uncertainties in health or economic consequences. The study at Sandia (AL85) did determine uncertainty in consequences, but only examined a limited number of parameters. The study at Sandia was primarily aimed at demonstrating the use of uncertainty and sensitivity analysis techniques on a consequence model. Full uncertainty analyses are planned for the near future by groups at KfK, NRPB and Sandia.

3.7 Applications of accident consequence assessment

Since the publication of WASH-1400, ACA has been used as an investigative tool in an increasing number of fields. Risk assessment results for different plants and a variety of environmental conditions have been compared with one another to show the influence of changed safety concepts, parameter values or different situations. In the following, the most important areas where ACAs have been applied are briefly described.
In general, probabilistic risk assessment (PRA) models have been developed to evaluate the aggregate risk of potential accidents at many reactors and sites, such as in the Reactor Safety Study and the German Risk Study. However, PRA models are currently being applied in many countries (particularly the United States) to examine the risk posed by reactors at specific sites and to provide guidance for planning and decision making. Besides use in risk evaluation, other areas of important application include evaluation of alternative design features, emergency planning and response, reactor-siting recommendations, and development of acceptable risk criteria.

**ACA in safety evaluation**

The ACA model CRAC was developed primarily for the U.S. Reactor Safety Study (Ref. UN75), to assess the societal health risks resulting from the concurrent operation of light water reactors at different sites. After the publication of the Reactor Safety Study, the need for site-specific risk studies arose in the USA. A number of such studies have been performed within the USA regulatory and licensing procedures (Refs. CO82, PA82, NU83a).

Another example of the use of consequence modelling for specific sites is the relatively recent series of environmental statements for several reactor sites (see Refs. NU81, NU82, NU84 as examples). These environmental statements contain analyses of the potential consequences of core-melt accidents and have been produced by NRC to fulfill its interpretation of the requirements of the National Environmental Policy Act (NEPA). Ref. NU82 contains an assessment (using the dispersion and deposition models and meteorological sampling scheme of CRAC) of consequences of atmospheric fallout of radionuclides on Great Lakes from the Fermi-2 reactor accidents. Ref. NU84 contains a very detailed assessment (also using CRAC) of public risks from the Limerick reactors near Philadelphia including the consideration of severe earthquakes.

ACA is playing an increasing role within Europe particularly in the context of the introduction of new, novel plants. Within the UK it has been used at the Public Inquiry into the proposed PWR reactor at Sizewell (Ref. KE82), the first introduction of commercial PWR technology into the UK. In Germany, as a result of a comparative probabilistic safety analysis of the proposed reactor and an existing PWR, the German Federal Parliament rejected a petition to stop the construction of a new fast breeder reactor, SNR-300.

**ACA in plant design development**

PRA techniques may be used to evaluate the risk reduction potential of possible plant modifications. In this context, the evaluation of the benefits and costs of proposed new safety features designed to reduce the probability or consequences of severe accidents is an important issue. The ultimate goal of this use of PRA is to be able to quantify the safety benefit of specific mitigation features.
In USA design alternatives for light water reactors have been investigated in the following areas:

- the siting of the reactor (e.g. underground)
- alternative containment designs (e.g. increased volume, filtered venting, double wall)
- the inclusion of core retention devices (e.g. core catcher)
- alternative or additional systems for heat removal from the core and containment.

ACA has been shown to be useful in quantifying the risk reduction and the cost-effectiveness of safety options.

**ACA in emergency planning**

ACA may be used to judge the likely effectiveness of actions taken to reduce the consequence of an accident. This effectiveness will depend upon the type of the nuclear facility under consideration, site characteristics such as the population distribution and traffic network, the conditions at the time of the accident (e.g. weather) and the characteristics of the accident. But ACA may also be used to analyze emergency planning in more generic terms. Analyses, whether generic or site specific, may be used to develop "best strategies" either for general application or for specific kinds of accidents or weather conditions. In addition, they allow estimates to be made of the facilities which would be needed to deal with accidents, and allow plans to be made for such facilities to be prepared should an accident ever occur. These may be operational such as providing sufficient monitoring stations to give an adequate description of the situation in a suitable timescale, or logistical such as the provision of an appropriate supply of stable iodine tablets for distribution.

At least two studies, (Refs. AL78 and AL80), have made use of the Reactor Safety Study source terms and consequence model CRAC to provide guidance in the area of emergency planning and response.

One study, (Ref. AL78), examined the relative merits of evacuation and sheltering followed by population relocation as protective measures for severe core-melt accidents, the distances to which (or areas within which) they might be needed, and the time available for their implementation. Based in part on this analysis, NRC has required implementation of emergency planning zones for the plume exposure pathway, with an approximate 10-mile radius, for all operating plants in the United States, (Ref. CO78). If source terms are significantly reduced, this study will probably be re-performed.

Another study, (Ref. AL80), was performed to provide guidance to policymakers concerning (1) the effectiveness of potassium iodide (KI) as a thyroid blocking agent in potential reactor accident situations, (2) the distance to which (or area within which) it should be distributed, and (3) its relative effectiveness compared with other available protective measures. Again, the analysis was performed using the Reactor Safety Study consequence model CRAC. The study concluded that KI did not appear to be a cost effective protective measure.
ACA in siting

There are many aspects that play a part in determining choice of siting. Some of them are radiological. One of the inputs to the radiological aspects of the siting of a nuclear facility is the possible impact of an accident at the site. ACA may be used to generate "siting criteria" which provide some numerical guidelines aimed at limiting the impact of a potential accident. The development of such criteria requires a detailed analysis of

- the consequences from possible plant accidents,
- the population distributions of the sites considered in the risk assessments,
- the economic impacts of possible accidents.

Because only a limited number of consequences of an accident are dependent upon the characteristics of the immediate vicinity of the site it may be possible to reduce the number of consequences considered in an analysis. For example, since in densely populated regions like Europe, the collective risk of latent health effects is mainly caused by low radiation doses to a large number of persons living far from the site, siting has negligible influence on this kind of consequence. In contrast to this, the number of early health effects which may occur after an accidental release of radioactive material is strongly dependent on the population distribution in the vicinity of the site. In addition, the site-specific traffic network may influence the effectiveness of emergency actions and hence the consequences of the accident.

Comparative site evaluations have been performed e.g. in Finland (Ref. SA79), and for the Norwegian Government Commission on Nuclear Power (Refs. N078 or TV79).

Following the accident at TMI, the U.S. NRC requested Sandia National Laboratories to perform a study to develop technical guidance to support the formulation of new regulations for siting nuclear power reactors in the United States, (Ref. AL82) Guidance was requested regarding (1) criteria for population density and distribution surrounding future sites and (2) standoff distances of plants from offsite hazards. Generic and site-specific consequence calculations were performed using CRAC2 to evaluate the dependence of consequences on source term, population distribution, meteorology, and emergency response. The results of these analyses can be used to determine many of the impacts of alternative siting criteria and to provide guidance in evaluating trade-offs among criteria. In addition, the broad range of data and analyses contained in the study can be useful for a wider community of users interested in evaluating the consequences of reactor accidents. The study concluded that estimates of offsite consequences depend strongly on site population distribution but are relatively insensitive to site meteorology (It has, however, been argued that this conclusion may not be valid in Europe). Predicted numbers of early fatalities are very sensitive to source-term magnitude and the timing of emergency response. A decision by the NRC about new siting policy has been deferred until after the source term issue is resolved.
ACA and safety goals

The development of a set of quantitative rules governing decisions on the safety of light water reactors has intensified greatly in the USA over the past five years. Development work with a similar basic purpose has high priority in all countries with nuclear programs, but the approach differs from one country to another, and the concept of "safety goals" has mainly been adopted by the USA. The process of defining such "safety goals" is not solely a scientific or objective decision and various formulations of risk criteria have been discussed. At present no broad consensus exists on any of the proposed safety goal concepts. Safety goals may be based on different criteria, derived from

- plant damage
- radioactive release
- radiation dose
- individual risk
- societal risk
- cost-benefit consideration

or combinations of these quantities. Most attention has been paid to the limitation of mortality risks for individuals and society as a whole. In the USA provisional numerical guidelines are under discussion, which limit early and late fatality risks from a nuclear power plant to a small fraction of the risk resulting from all other causes. In addition a cost-benefit guideline is proposed for use in decisions on safety improvements which would reduce individual and societal risks at levels of risk below those specified in the health risk guidelines. Further safety goal criteria which are consistent with the complete presentation of ACA results in the form of CCFDs i.e. limiting curves on frequency/consequence or frequency/dose diagrams, have been proposed, which can be directly compared with the results of existing ACAs. However, any mathematical formulation cannot disregard the fact that the limits themselves are more or less arbitrary.

ACA in determining research priorities

There are a number of different possible approaches to the task of allocating priorities to research and development (R & D) programs under the constraint of limited resources, whilst attempting to maximise the benefit in terms of improved safety. A possible approach is cost-effectiveness investigations. These investigations can be performed at a qualitative level or by applying quantitative methods.

One of the ultimate goals of improved safety is to reduce potential accident consequences in the environment. The reduction in environmental consequences is then a measure of the effectiveness of the additional cost, in providing improved safety. Accordingly ACA methods must be utilized in the process of choosing strategies for improving the safety.

Another, perhaps equally important, R & D objective is to reduce the uncertainties in estimated accident consequences. In this case it is, however, more difficult to determine how to measure the effectiveness of the R & D funds allocated to a specific R & D program. But the ACA methods can themselves be used to help set research priorities in the ACA area. Sensitivity and/or uncertainty analyses of ACA data and models can provide information needed to set such priorities. Further discussion of this application of ACA is found in section 3.8.1.
PRA Procedures Guide

A joint NRC-industry effort, under the aegis of the Institute of Electrical and Electronics Engineers (IEEE) and the American Nuclear Society (ANS), has prepared a procedures guide for the systematic application of probabilistic and reliability analysis to nuclear power plants (Ref. H182). The objective of the guide is to document the current state of the art of all major subject areas in a PRA. For each of these areas, including consequence analysis, the guide delineates acceptable analytical techniques, acceptable assumptions and modeling approximations, acceptable methods for treating uncertainties, and acceptable standards for documentation and quality control. The chapter in the guide that deals with environmental transport and consequence analysis discussed the information that is required as input to a consequence analysis, the type of output that might result and how it should be interpreted, the potential pitfalls associated with the use of existing codes, and the uncertainties inherent in the data and modeling. The chapter is an excellent compendium of information, references, and discussion of important areas in the field of consequence analysis.

3.8 Potential for improvements

It is recognized that, although ACA methods have proved very useful over the last ten years, improvements are needed in many areas. It is also recognized that, since ACA methods were developed primarily for evaluation of the consequences of large reactor accidents, further development may be needed if it is desired to use the methods to evaluate accidents with significantly different source characteristics. Here the close connection between the source term work and the ACA development work is evident.

Efforts to improve accident consequence analysis methods and data are under way in a number of countries. Some of the larger of these projects are described in the following.

3.8.1 Source term characteristics and accident consequence assessment

The two distinct stages of a probabilistic risk analysis (PRA), namely plant safety analysis and accident consequence assessment, involve very different technical disciplines. As a result there may be interface problems. One of CSNI's purposes in coupling the two areas in Principal Working Group 4 is to resolve some of these problems.

A major step in this direction was the preparation of the Source Term Report by GRECA (Ref. GR84). The most important source term characteristics in determining offsite consequences are given in the following list (Ref. GR84):

- Release fractions.
- Duration of release.
- Warning time.
- Rate of heat release (for some accident sequences).
- Particle-size distribution (in connection with contamination, property damage).
- Moisture content of release (ongoing investigations may show this characteristics to be important).

The impact of the source term characteristics is large, not only on consequence assessment results, but on the priorities of future work in this field. The present priorities are based upon qualitative or quantitative sensitivity studies or parameter variation analyses; but are valid only for the present source term status. Drastically different source terms may have large implications. A previously unimportant pathway may become important, other parameters may dominate the uncertainty, a nuclide for which data are deficient may become important. Areas of ACA methodology that have been given low priority, because uncertainties in these areas had little impact upon the ACA results, may become important. Better data and models may be needed, where they previously had little impact upon the results.

It is, therefore, imperative to obtain an early indication, if source term work indicates that source term status will change, so that ACA research priorities can be shifted accordingly. Quite recently a large report on source term status was prepared by a Special Task Force on Source Term, set up by CNSN. It has not yet been determined explicitly what influence this work may have upon ACA research priorities.

3.8.2 Current development work

Work in several areas of ACA development are carried out as part of the NRC sponsored MELCOR program (SP83). Specifically, improvements are being made in the atmospheric dispersion and transport model; these include developing a multi-puff model which will permit the analysis of site-specific terrain and plume trajectories and provide an improved treatment of long-duration releases and precipitation modelling. Other improvements in modelling capabilities will include the incorporation of more detailed land-use characteristics, especially the differentiation of urban and rural areas, and a reevaluation of the available emergency-response data, which will provide improved estimates of the risk of early health effects. Improved models for radiological health effects and potential economic impacts are also being developed.

In addition, a key objective of the MELCOR offsite-consequence modelling effort is to develop tools that can provide estimates of the uncertainties in the predicted consequences. Although uncertainties are likely to remain quite large, a thorough examination of their origin and magnitude will provide both a firmer basis for applying offsite-consequence analysis and a better understanding of its limitations. Finally an NRC sponsored program is currently underway to assess the potential impact on offsite consequences of localized "rain-out" from a moist plume.

In the United Kingdom a new accident consequence assessment code called CONDOR is currently under development jointly by CEGB and UKAEA (Safety and Reliability Directorate). The code will incorporate significant novel features. For example, it will include both puff and plume models for atmospheric dispersion. It will also account for doses from deposition over sea and for a range of different dose-
effect relationships. CONDOR has been structured in a very general manner so that specific modules can be updated or replaced without disturbing the over-all architecture. The modular structure also easily permits uncertainty analyses to be carried out, this being one of the intended applications of CONDOR.

Several projects in the ACA area are included in the new four year period, starting 1985, of the Nordic Safety Program. They are concerned with the following project areas:

- Extension and utilization of the Nordic fallout data bank.
- Various improvements in ACA modeling, including use of multi-puff models.
- Mitigating actions and economic consequences.
- Conditions in urban areas.
- Winter conditions.
- Resuspension.
- Shielding.
- Uncertainty analysis of ACA.

A number of other projects financed nationally are also carried out in the Nordic countries.

In the Federal Republic of Germany the UFOMOD code system is presently under revision. The new version of UFOMOD was expected to be completed in spring 1986.

There will be considerable modifications to submodels and improvement of data sets in the following areas:

In atmospheric dispersion and deposition:

- Application of trajectory/puff type model to assess all kinds of early effects in distances up to 50-100 km resulting from short-term releases (≤ 1 d).
- Implementation of the MESOS code (Imperial College, UK) for dispersion calculations in the large distance range (> 50 km up to the borders of Europe) to assess the late effects.
- Development and application of a simple dispersion model for long-term releases (> 1 d up to some weeks).
- The particle size distribution will be taken into consideration (and taken into account in the dose and health effects calculations via the inhalation pathway).
- The cyclic sampling scheme will be replaced by a stratified sampling scheme.
- Improvement of the meteorological data base (synoptic recordings).

Behaviour of radionuclides in the environment:

- Inclusion of deposition and post-deposition behaviour of radioactive material in urban areas (run-off and wash-off processes).
- Dynamic modelling of the transfer of radionuclides through foodchains (results of ECOSYS code of GSF for 8 food products).
- Improved description of the time dependent migration of radionuclides into soil.
Revision of the countermeasures modelling:

- Timing and range of emergency actions with respect to new source terms; intervention levels of relocation, decontamination and interdiction of food.

Dose assessment:

- Shielding factors of various building types according to newest shielding calculations for German houses.
- Completely new data sets of dose conversion factors for external (ground, cloud) and internal (inhalation, ingestion) radiation, prepared by GSF for an extended list of nuclides (> 54).
- Consumption habits of the German population for 8 food products.

Health effects assessments:

- Revised dose-risk relationships for mortality and morbidity (early and late health effects).
- Implementation of models to assess genetic consequences.

Presentation of results:

- Frequency distributions of activity concentrations, radiation doses, health effects, areas and number of persons affected by countermeasures.
- Time and distance dependent presentations of the above mentioned consequences (especially the time dependent occurrence of the late health effects and the time dependence of interdicted areas).
- Presentation of fatalities in the form of loss of life expectancy.

3.9 Conclusions, recommendations

Once more it is appropriate to remind the reader that this chapter describes the situation in the accident consequence assessment area before the accident in Chernobyl took place. The widespread contamination following that accident has had and will have a strong impact upon work in the area. After May 1986 much more attention is focused on long-range atmospheric transportation models than previously. Also priorities of experimental work has also been shifted. The contamination presents a unique opportunity for studying the behaviour of radioactive materials in the environment, and it is important to collect the information while this possibility is here.

The techniques of probabilistic risk assessment (PRA) have found extensive use for evaluating the risk from potential accidents in nuclear installations. PRA can conveniently be categorised into two broad disciplines, plant safety analysis and accident consequence assessment (ACA). Plant safety analysis is concerned with describing the behavior of the plant under accident conditions, and the end result of the analysis is the so-called source term. The source term describes a release to the environment, and contains information on the amount of different radionuclides released, the release duration, release height etc. The source term is input to the ACA, and the end
results of the assessment are all the potential types of health and economic consequences in the environment, coupled with probabilities of occurrence. These results are often presented in integral form as "risk curves".

The essential elements of an ACA are many and span a broad range of disciplines. One gets some understanding of the width of this range from the fact that an ACA contains elements from disciplines as varied as e.g. meteorology, medicine and agricultural practices.

A large number of ACA codes have been developed in different countries and have been used in an increasing number of fields. Although ACA methods have proved very useful over the last ten years, improvements are, however, needed in many areas.

The largest uncertainty in off-site consequences is due to the variation of the accident source, but even in the case of fixed releases of radioactive material, the consequences will vary considerably with the conditions pertaining at the time, in particular with the prevailing meteorological conditions, the season, the location and habits of the population etc. It is, for example apparent that, for a nuclear installation located on the coast, the consequences of an accident would be very different in the following two situations: when the released material is dispersed inland over an urban area, and when the released material is blown out to sea. For any given release, therefore, there will be a spectrum of possible consequences, each having different probabilities of occurrence determined by the environmental characteristics of the release location and its surroundings. In assessing the risk from accidents, therefore, both the release and the consequences which might result from it must be treated in a probabilistic framework, in contrast to the approach adopted when assessing the risk from effluent discharges during normal operation.

Besides carrying through the uncertainties related to the previous steps of a PRA (the plant safety analysis) the off-site accident consequence assessment has its own limitations, and these stem principally from two sources: uncertainty in the models and uncertainty in the data required by the models. Large efforts are underway many places in the world for determining the uncertainty in ACA under varying conditions. The results obtained will also be of value in helping to set priorities for future research.

Priorities for future research are, however, also to a large extent determined by the situation in the source term area. It is, therefore, imperative to obtain an early indication, if source term work indicates that source term status will change, so that ACA research priorities can be shifted accordingly. A description of the coupling between the ACA area and the source term area (seen from the point of view of ACA) is found in the Source Term Report, published in 1984 by the expert group GRECA, under Pwg4. Quite recently a large report on source term status was prepared by a Special Task Force on Source Terms, set up by CSNI. It has not yet been determined what influence this work may have upon ACA research priorities.

The status of a number of aspects of ACA have recently been investigated in the Pathway Parameter Survey/Evaluation, performed within GRECA. This survey has been of great value as a help for choosing the best data and methods, as well as a guide for future work, though in a less quantitative manner than what can be obtained from an uncertainty analysis.
Development work in the ACA field is at present very active. Several new models are under development, and several older models undergo extensive modifications. Data are also continuously improved. Some of the more important improvements are being made in the atmospheric dispersion and transport models (many models are now modified to include a multi-puff model), in the capabilities for incorporating more detailed land-use characteristics (especially the differentiation of urban and rural areas), in emergency-response data and models, and in the models for radiological health effects and potential economic impacts.

The largest development efforts are carried out in the USA as part of the MELCOR program, in the United Kingdom, in Germany, in the Nordic countries (Denmark, Finland, Norway and Sweden) as part of the Nordic Safety Program, and in several European countries as part of CEC's MARIA program.

The situation in the ACA area is at present rather dynamic, as a large number of projects are being carried out, or are near completion. This is also true in the source term area, and the impact upon ACA from the source term area may also be considerable.

3.10 References


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In 1948, the United States offered Marshall Plan aid to Europe, provided the war-torn European countries worked together for their own recovery. This they did in the Organisation for European Economic Co-operation (OEEC).

In 1960, Europe's fortunes had been restored; her standard of living was higher than ever before. On both sides of the Atlantic the interdependence of the industrialised countries of the Western World was now widely recognised. Canada and the United States joined the European countries of the OEEC to create a new organisation, the Organisation for Economic Co-operation and Development. The Convention establishing the OECD was signed in Paris on 14th December 1960.

Pursuant to article 1 of the Convention, which came into force on 30th September 1961, the Organisation for Economic Co-operation and Development shall promote policies designed:

- to achieve the highest sustainable economic growth and employment and a rising standard of living in Member countries, while maintaining financial stability, and this to contribute to the development of the world economy;

- to contribute to sound economic expansion in Member as well as non-member countries in the process of economic development; and

- to contribute to the expansion of world trade on a multilateral, non-discriminatory basis in accordance with international obligations.

The original Signatories of the Convention were Austria, Belgium, Canada, Denmark, France, the Federal Republic of Germany, Greece, Iceland, Ireland, Italy, Luxembourg, the Netherlands, Norway, Portugal, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The following countries acceded subsequently to the Convention (the dates are those on which the instruments of accession were deposited): Japan (28th April 1964), Finland (28th January 1969), Australia (7th June 1971) and New Zealand (29th May 1973).

The OECD Nuclear Energy Agency (NEA) was established on 20th April 1972, replacing OECD's European Nuclear Energy Agency (ENEA, established on 20th December 1957) on the adhesion of Japan as a full member.

NEA now groups all the European Member countries of OECD and Australia, Canada, Japan and the United States. The Commission of the European Communities takes part in the work of the Agency.

The primary objectives of NEA are to promote co-operation between its Member governments on the safety and regulatory aspects of nuclear development, and on assessing the future role of nuclear energy as a contributor to economic progress.

This is achieved by:

- encouraging harmonisation of governments' regulatory policies and practices in the nuclear field, with particular reference to the safety of nuclear installations, protection of man against ionising radiation and preservation of the environment, radioactive waste management, and nuclear third party liability and insurance;

- keeping under review the technical and economic characteristics of nuclear power growth and of the nuclear fuel cycle, and assessing demand and supply for the different phases of the nuclear fuel cycle and the potential future contribution of nuclear power to overall energy demand;

- developing exchanges of scientific and technical information on nuclear energy, particularly through participation in common services;

- setting up international research and development programmes and undertakings jointly organised and operated by OECD countries.

In these and related tasks, NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has concluded a Co-operation Agreement, as well as with other international organisations in the nuclear field.
The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers who have responsibilities for nuclear safety research and nuclear licensing. The Committee was set up in 1973 to develop and co-ordinate the Nuclear Energy Agency's work in nuclear safety matters, replacing the former Committee on Reactor Safety Technology (CREST) with its more limited scope.

The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries. This is done in a number of ways. Full use is made of the traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences. Some of these arrangements are of immediate benefit to Member countries, for example by improving the data base available to national regulatory authorities and to the scientific community at large. Other questions may be taken up by the Committee itself with the aim of achieving an international consensus wherever possible. The traditional approach to co-operation is reinforced by the creation of co-operative (international) research projects, such as PISC and LOFT, and by the organisation of international standard problem exercises, for testing the performance of computer codes, test methods, etc. used in safety assessments. These exercises are now being conducted in most sectors of the nuclear safety programme.

The greater part of the CSNI co-operative programme is concerned with safety technology for water reactors. The principal areas covered are operating experience and the human factor, reactor system response during abnormal transients and accidents, various aspects of primary circuit integrity, the phenomenology of radioactive releases in reactor accidents, containment performance, risk assessment, and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

The Sub-Committee on Licensing, consisting of the CSNI Delegates who have responsibilities for the licensing of nuclear installations, examines a variety of nuclear regulatory problems and provides a forum for the review of regulatory questions, the aim being to develop consensus positions in specific areas.