PASSIVE ALWR SOURCE TERM

Prepared by
The Advanced Reactor Severe Accident Program
Source Term Group

D. E. Leaver, TENERA, L.P., Chairman

R. Denning, Battelle Columbus

R. Hobbins, EG&G Idaho

J. Metcalf, Stone & Webster

D. Osetek, Los Alamos Technical Associates

W. Pasedag, Department of Energy

R. Ritzman, EPRI

R. Sher, Consultant

February, 1991

Prepared in support of the Utility/EPRI Advanced
Light Water Reactor Program and for EG&G Idaho, Inc.
Under Subcontract No. C85-100740 and the US. Department of
Energy Under Contract No. DE-AC07-76ID01570

CONTENTS

Section			Page
1	**INTRODUCTION**		
	1.1	Background	1-1
	1.2	Objectives in Developing a Physically-Based Source Term	1-1
	1.3	Uses of the Source Term	1-3
	1.4	Standard Plant Design	1-4
	1.5	Role of ALWR Requirements Document	1-4
	1.6	Steps in Developing Physically-Based Source Term	1-4
	1.7	Margin in Physically-Based Source Term	1-8
	1.8	Organization of Report	1-9
2	**SUMMARY OF INTEGRATED SOURCE TERM FOR THE PASSIVE PWR AND PASSIVE BWR**		2-1
	2.1	Introduction	2-1
	2.2	Core Damage Event for Estimating Source Term	2-1
	2.3	Physically-Based Source Term for the Passive PWR	2-4
		2.3.1 Coolant and Gap Release	2-4
		2.3.2 Release Fractions to Primary Containment Atmosphere	2-6
		2.3.3 Chemical Form	2-12
		2.3.4 Containment Fission Product Behavior	2-12
		2.3.5 Secondary Structure Fission Product Holdup and Retention	2-14
	2.4	Physically-Based Source Term for the Passive BWR	2-15
		2.4.1 Coolant and Gap Release	2-15
		2.4.2 Release Fractions to Primary Containment Atmosphere	2-15
		2.4.3 Chemical Form	2-17
		2.4.4 Containment Fission Product Behavior	2-17
		2.4.5 Secondary Structure Fission Product Holdup and Retention	2-20
	2.5	Comparison of Physically-Based Source Term Containment Release with Existing Regulatory Guidance	2-21
	2.6	Format of the Physically-Based Term Expression	2-23
3	**COMPARISON OF ALWR SOURCE TERM WITH NUREG-1150 RESULTS**		3-1
	3.1	Comparison of Individual Aspects of Source Term	3-1
		3.1.1 Comparison of In-Vessel Release	3-2
		3.1.2 Comparison of RCS Retention	3-2
		3.1.3 Comparison of Revaporization Release	3-7

Section Page

3.1.4 Comparison of Ex-Vessel Release 3-7
3.1.5 Summary of Comparison 3-11

3.2 Comparison of Integrated Source Term 3-11

4 INDIVIDUAL ASPECTS OF THE PHYSICALLY-BASED SOURCE TERM 4-1

4.1 Coolant and Gap Activity Source Terms 4-1

4.1.1 Introduction 4-1
4.1.2 Coolant Activity Phenomena and Assumptions 4-1
4.1.3 Gap Activity Phenomena and Assumptions 4-9
4.1.4 Conclusions 4-18

4.2 Early In-Vessel Release Magnitude 4-19

4.2.1 Introduction 4-19
4.2.2 Important Phenomena and Assumptions 4-20
4.2.3 Results 4-20
4.2.4 Conclusions 4-24

4.3 Late In-Vessel Release Magnitude 4-26

4.3.1 Introduction 4-26
4.3.2 Important Phenomena and Assumptions 4-26
4.3.3 Results 4-27
4.3.4 Conclusions 4-27

4.4 RCS Retention 4-28

4.4.1 Introduction 4-28
4.4.2 Important Phenomena and Assumptions 4-29
4.4.3 Results for RCS Retention 4-37
4.4.4 Conclusions 4-37

4.5 Revaporization Release 4-39

4.5.1 Introduction 4-39
4.5.2 Important Phenomena and Assumptions 4-40
4.5.3 Results 4-42
4.5.4 Conclusions 4-43

4.6 Ex-Vessel Release Magnitude 4-43

4.6.1 Introduction 4-43
4.6.2 Important Phenomena and Assumptions 4-43
4.6.3 Results 4-45
4.6.4 Conclusions 4-45

4.7 Chemical Form 4-47

4.7.1 Introduction 4-47

Section			Page
	4.7.2	Important Phenomena and Assumptions	4-47
	4.7.3	Results	4-55
	4.7.4	Conclusions	4-56
4.8		Scrubbing in Suppression Pools	4-57
	4.8.1	Introduction	4-57
	4.8.2	Important Phenomena and Assumptions	4-58
	4.8.3	Results for Scrubbing DF	4-61
4.9		Primary Containment Aerosol Removal	4-64
	4.9.1	Introduction	4-64
	4.9.2	Important Phenomena and Assumptions	4-65
	4.9.3	Results	4-77
	4.9.4	Conclusions	4-77
4.10		Fission Product Holdup and Retention in Secondary Structures	4-85
	4.10.1	Introduction	4-85
	4.10.2	Important Phenomena and Assumptions	4-86
	4.10.3	Mixing Between the Secondary Building and the Environment	4-91
	4.10.4	Leakage through Steel Containment Shells	4-94
	4.10.5	BWR MSIV Leakage	4-97
	4.10.6	Results	4-98
	4.10.7	Conclusions	4-102
5		**REFERENCES**	5-1
APPENDIX 1		**CORE DAMAGE EVENT FOR PURPOSES OF QUANTIFYING THE PHYSICALLY-BASED SOURCE TERM**	A-1
A.1		Introduction	A-2
A.2		Evaluation of Functional Sequence Types Against Requirements	A-4
	A.2.1	ALWR Requirements	A-10
	A.2.2	Selection of Functional Sequence Type	A-17
A 3		Definition of Core Damage Event Characteristics	A-17
	A.3.1	PWR Thermal Hydraulics	A-19
	A.3.2	BWR Thermal Hydraulics	A-25
	A.3.3	Summary	A-32
APPENDIX 2		**SHOREHAM TECHNICAL SPECIFICATIONS PERTAINING TO REACTOR BUILDING INTEGRITY**	A-34

ABSTRACT

The purpose of this report is to provide technical support for the physically-based source term which is proposed as the licensing design basis fission product release from a major core accident for the Passive Advanced Light Water Reactor (ALWR) in Volume III, Section 5 of the ALWR Requirements Document. While TID 14844 and evolving, related regulatory guidance have served the industry well, a substantial body of new research motivated by the Three Mile Island (TMI) accident is maturing, and the ALWR Requirements Document provides an opportunity to incorporate this experience in an updated source term. This update will provide a more rational basis for Passive ALWR accident mitigation system designs, particularly where the designs afford opportunities for improvement and innovation.

Great attention has been paid to accident prevention in the ALWR Requirements Document which will reduce the likelihood of core damage by an order of magnitude or more compared to earlier LWR designs. Nonetheless, for defense-in-depth the Passive ALWR source term is based on evaluation of a core damage event. Selection of this core damage event and the associated quantification of the fission product release were done in a conservative, yet physically-based manner so as to provide significant margin to the expected releases, given an ALWR accident, while avoiding non-physical assumptions which could produce mitigation system designs not well-suited to the important accidents.

The physically-based source term presented in this report is intended for use in ALWR design basis analysis (i.e., offsite consequence analysis per the Standard Review Plan, Section 15.6.5, Appendix A), defining the radiological environment for plant systems and equipment (i.e., equipment qualification), and evaluating the offsite dose for emergency planning considerations. A summary of source term results follows.

SUMMARY OF SOURCE TERM RESULTS

The results for the physically-based source term are as follows:

- The event used to define the source term involves large scale core damage with core debris penetrating the reactor vessel lower head, but does not assume a large break LOCA initiated core melt which, on the basis of past PRAs and the ALWR design requirements and preliminary ALWR PRA studies, is not an important accident.

- The noble gas and iodine release magnitudes to containment are roughly equal to that from current regulatory guidance. Significant fractions of remaining elements (not included in current regulatory guidance) are also released to the containment atmosphere in the physically-based source term.

- Release timing for the physically-based source term occurs over a period of hours based on the time required to lose primary coolant inventory and the time required for core heatup and fission product release. This contrasts with the instantaneous release assumption in current regulatory guidance.

- Fission product chemical form is defined as primarily aerosol based on extensive analytical and experimental evidence for conditions corresponding to ALWR core damage sequences. This contrasts with regulatory guidance which includes very little fission product aerosol.

- Natural fission product (primarily aerosol) removal processes in containment are credited based on extensive analytical and experimental evidence. Such processes are not addressed in current regulatory guidance since the current regulatory source term has so little aerosol.

- Modest credit for fission product retention in secondary buildings is taken, again based on analytical and experimental evidence, and building design requirements. Such credit is addressed in current regulatory guidance for plants with active secondary building filtration systems, whereas the ALWR approach involves natural removal and holdup.

ACKNOWLEDGEMENT

The authors wish to acknowledge David Blanchard and Stephen Additon, both of TENERA, for their technical support and efforts in integrating this report.

ACRONYMS AND ABBREVIATIONS

ADS	Automatic Depressurization System
ALWR	Advanced Light Water Reactor
AMMD	Aerodynamic mass median diameter
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARI	Auxiliary Rod Injection
ARSAP	Advanced Reactor Severe Accident Program
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CRDM	Control Rod Drive Mechanism
DF	Decontamination Factor
DOE	Department of Energy
DPV	Depressurization Valve
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
GE	General Electric
HEPA	High Efficiency Particulate Air
IC	Isolation Condenser
IDCOR	Industry Degraded Core Rulemaking Program
INEL	Idaho National Engineering Laboratory
IRWST	In-containment Refueling Water Storage Tank
LDB	Licensing Design Basis
LOCA	Loss of Coolant Accident
LPIS	Low Pressure Injection System
LWR	Light Water Reactor
MSIV	Main Steam Isolation Valve
NDE	Non-destructive Examination
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PAG	Protective Action Guideline
PCCS	Passive Containment Cooling System
PDHR	Passive Decay Heat Removal
PRA	Probabilistic Risk Assessment
PWR	Pressurized Water Reactor

RBSVS	Reactor Building Standby Ventilation System
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SER	Safety Evaluation Report
SFD	Severe Fuel Damage
SRV	Safety Relief Valve
STCP	Source Term Code Package
STS	Standard Technical Specifications
TAF	Top of Active Fuel
TMI	Three Mile Island
USC	Utility Steering Committee
atm	atmosphere
cfm	cubic feet per minute
Ci	Curie
cm	centimeter
f.p.	fission product
ft	foot(feet)
gm or g	gram
in.	inch(es)
kg/m^3	kilogram per cubic meter
kw	kilowatt
kw/ft	kilowatt per foot
m	meters
max	maximum
min	minimum
MWd/tU	megawatt days per ton Uranium
MWt	megawatt thermal
psi	pounds per square inch
psia	pounds per square inch absolute
revap.	revaporization
scfd	standard cubic feet per day
scfh	standard cubic feet per hour
sec	second
$site_{80}$	80^{th} percentile site

$site_{ref}$	ALWR reference site
t	time
μCi	micro-curie
W/cm	watts per centimeter

1. INTRODUCTION

1.1 Background

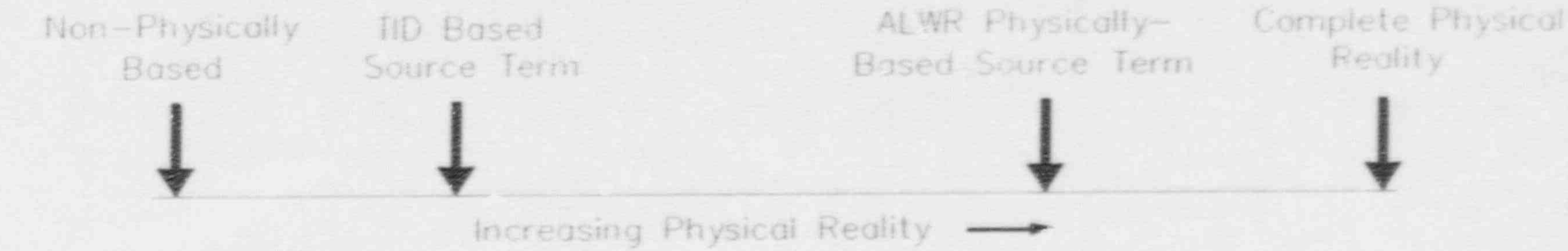
The purpose of this report is to provide technical support for the physically-based source term which is proposed for the Passive Advanced Light Water Reactor (ALWR). As specified in Volume III, Chapter 5, Section 2 of the ALWR Requirements Document,[1-1] a physically-based source term shall be used as the design basis for passive plant accident mitigation systems. Volume III, Chapter 5, Appendix B of the Requirements Document defines the physically-based source term, and this report provides supporting technical details. The report has been prepared by the Department of Energy (DOE) - sponsored Advanced Reactor Severe Accident Program (ARSAP) in support of the Utility/Electric Power Research Institute (EPRI) ALWR Program.

1.2 Objectives In Developing a Physically-Based Source Term

The Passive ALWR source term is based on evaluation of a core damage event which is defined for purposes of estimating the source term and which results in a conservative, yet physically-based source term for the important sequence types for a given standard plant design. Figure 1-1 provides a qualitative comparison of the physically-based source term with the current 10CFR Part 100 source term which is based on TID 14844.[1-2]

The ALWR Utility Steering Committee (USC) has two main objectives in developing a physically-based source term for the passive plant. The first is to factor in the source term experience of the nearly 30 years since TID-14844 was cited as a guideline document in 10CFR Part 100 and the 12 years since the Three Mile Island (TMI)-2 accident. The second objective is to provide a more rational basis for Passive ALWR accident mitigation system design.

With respect to the first objective, the industry has used TID 14844 as the basis for fission product release in the source term used for siting dose and other applications since the early 1960's. While TID 14844 and related regulatory guidance have served the industry well, resulting in a strong containment and associated engineered systems for accident mitigation, much



TID Based Source Term	ALWR Physically-Based Source Term
No coupling to specific sequences	Use functional sequence types as starting point; eliminate sequence types based on plant design requirements supplemented by PRA
Instantaneous release to containment	Physically-based sequence timing, release (e.g., many hours), and chemical form
Little coupling of containment thermal hydraulics and source term	Base aerosol removal and leakage on containment conditions for the sequences
Non-sequence specific source term applicable to all plants	Envelope source term for important sequence types for a given standardized plant design
Simple recipe in guidance documents	Recipe in guidance documents, but not quite as simple

Figure 1-1 Qualitative Comparison of TID Based and Physically-Based Source Terms

has been learned from the evaluation of the TMI-2 accident and subsequent severe accident research. The ALWR Requirements Document, with its emphasis on ALWR safety being based on the best available technical information, provides an ideal opportunity to incorporate this experience.

With respect to the second objective, use of a physically-based source term will lead to design features which enhance overall safety compared to that which would result from a non-mechanistic source term such as TID 14844. Steam condensation driven fission product deposition, main steam isolation valve (MSIV) leakage control, and secondary building fission product leakage control are examples of passive mitigation functions and systems for which a non-mechanistic source term could potentially produce non-optimal designs.

1.3 Uses of the Source Term

Design basis accident source terms are used in three ways in today's regulations:

1. as an input to licensing design basis analysis and to assess the effectiveness of accident mitigation functions and systems;

2. to define the radiological environment for certain plant equipment and systems; and

3. for siting evaluations as required by 10CFR100.[1-3]

Even if the rulemaking currently being contemplated by the Nuclear Regulatory Commission (NRC) staff[1-4] eliminates the use of the design basis source term for siting evaluations, the first two applications will remain. The source term is intended as a replacement for TID 14844 in deriving offsite consequences associated with the design basis LOCA required for Chapter 15 of the FSAR. Key passive plant equipment and systems affected by the radiological environment include the control room (i.e., habitability considerations) and equipment inside containment which must function during and after release of radioactivity.

The ALWR Program also is proposing that the physically-based source term be applied to evaluate dose for emergency planning considerations.

In addition to these uses other Chapter 15 analyses may also be affected by various aspects of the proposal source term. Examples include fuel handling accidents and transients and accidents which potential leaks are limited to coolant and gap activity release such as main steam line break and steam generator tube rupture. Further discussions ont he use of the proposed source term for these purposes would be useful in determining the benefits of pursuing these other applications.

1.4 Standard Plant Design

It is intended that each Passive ALWR design be licensed as a standard plant under 10CFR52.[1-5] As noted in Section 1.6 below, design requirements for the passive plants were considered in developing the physically-based source term. Thus there are differences in the source terms for the Pressurized Water Reactor (PWR) and the Boiling Water Reactor (BWR), although these differences are not major. The differences can be accommodated in the 10CFR52 design certification rulemaking for each standard plant.

1.5 Role of ALWR Requirements Document

As specified in Commission policy guidance, [1-6] the Passive ALWR Requirements Document is to be the lead ALWR document for NRC review of the ALWR. It is therefore expected that the Requirements Document and associated Safety Evaluation Report (SER) will be the primary vehicles for industry-NRC dialogue and ultimate agreement on the definition of the physically-based source term. In this manner, the important issues associated with the physically-based source term will be resolved well ahead of certification of standard plant designs.

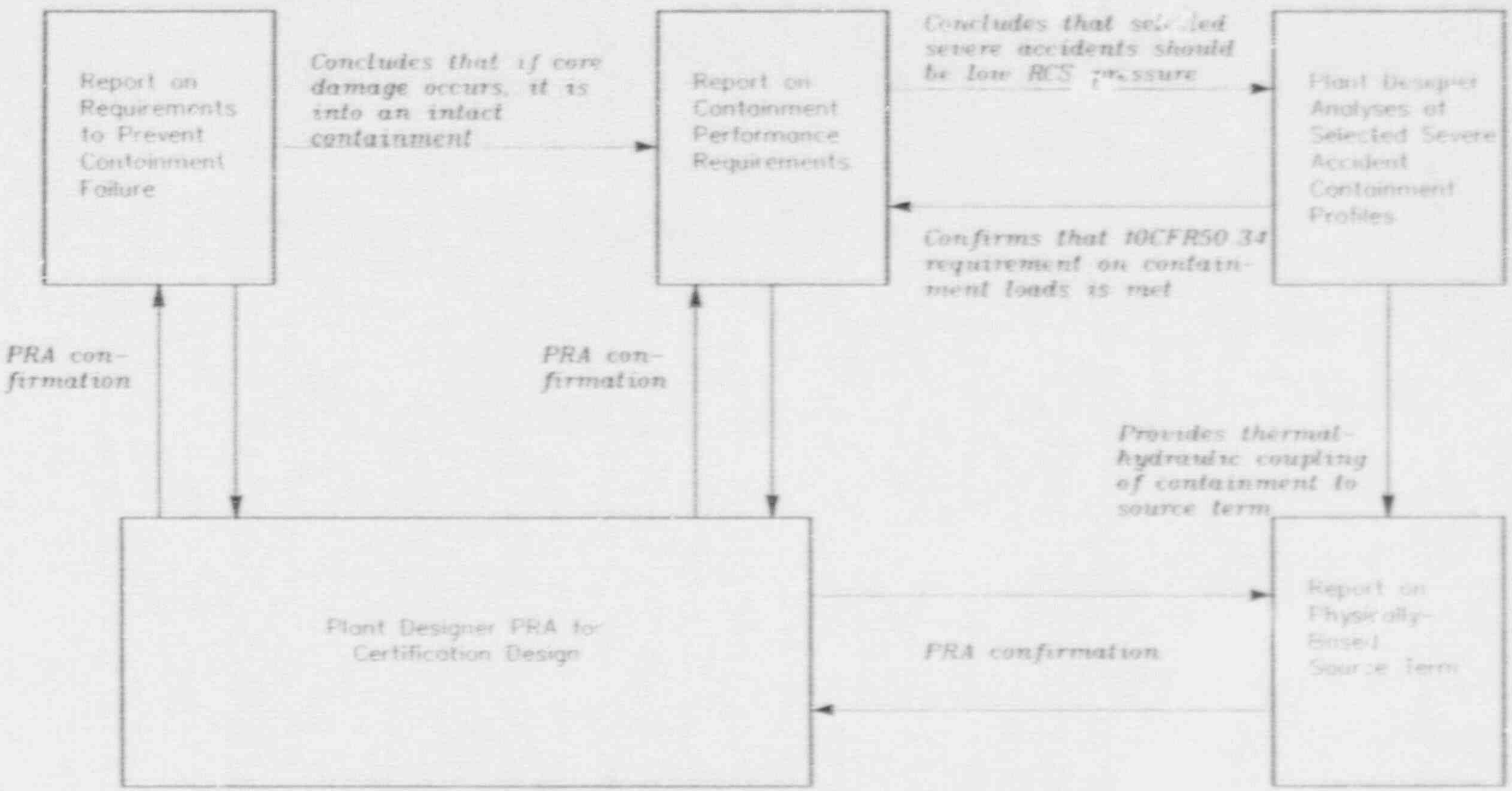
1.6 Steps in Developing Physically-Based Source Term

As noted in Section 1.2 above, the physically-based source term is based on evaluation of a core damage event which is conservatively defined for the purpose of estimating the source term. Passive ALWR design requirements make the likelihood of any core damage event extremely remote. Examples of such requirements are the greatly improved man-machine interface, passive safety systems which do not depend on ac power and other support systems, and the reduced need for operator action. Nonetheless, for defense-in-depth, Volume III of the Requirements Document specifies that containment performance under severe accident conditions be evaluated and that a core damage event be defined for use in developing a physically-based source term.

This core damage event and associated source term were defined so as to be consistent with the Passive ALWR design requirements. This is significant, since the requirements were developed to meet the ALWR safety policy that states that, even in the event of a severe accident, containment integrity should be maintained and the fission product release to the environment should be very low. Accordingly, the following steps were used to define the core damage event and associated source term. Several companion reports to this report are being prepared to address in more detail the technical basis for some of these steps. The purpose and interrelationships of these reports is illustrated in Figure 1-2.

1. Review the passive plant design requirements provided to eliminate containment failure sequences. Iterate as necessary on these requirements to assure that such sequences are effectively precluded through properly engineered means (e.g., multiple, independent, passive systems).

2. Define and implement a qualitative, engineering screening criterion, supplemented by a quantitative Probabilistic Risk Assessment (PRA) criterion, for defining and deciding upon functional sequence types most appropriate for containment performance evaluation and source term definition. This functional sequence selection process concludes that low Reactor Coolant System (RCS) pressure core damage sequences should be selected. Appendix 1 summarizes the results of the selection process.



SG01300B

Figure 1-2 Network of ALWR Reports and Analyses on Containment Performance and Source Term

3. Define the characteristics of the representative core damage event(s) to be used in quantifying containment performance. These characteristics are based on the functional sequence type(s) selected above, but are defined to produce a severe accident containment response that will envelope that from all accident sequences associated with the selected functional sequence type(s).

4. Perform analyses of the representative core damage event in order to assure that the 10CFR50.34(f) steam plus hydrogen requirement on containment loads is met (i.e., service level C for steel containments, unity factored load for concrete containments), and to provide thermal-hydraulic response of the RCS and the containment as a starting point for estimating the source term.

5. Define the characteristics of the representative core damage event to be used in quantifying the physically-based source term. These characteristics are based on the functional accident sequence type selected above, but are defined to produce a source term which will envelope that from accident sequences associated with the selected functional sequence type. Appendix 1 discusses these characteristics.

6. Quantify the physically-based source term from the representative core damage events so as to produce a robust envelope of the individual source terms from the accident sequences associated with the selected functional sequence type. The approach to quantification which provides this robust envelope includes the following:

   - core damage event characteristics which provide an enveloping source term (e.g., large scale core melting, vapor pathway from core to RCS);

- physically-based estimates of source term phenomena;

- avoiding significant credit for reductions in release (e.g., RCS retention of only 50%) except where the physical process is well understood and uncertainties are not large; and

- through a combination of design features and the quantification, assuring that the sensitivity of the environmental release to reasonable changes in individual aspects of the source term are not large (e.g., early cavity/lower drywell flood scrubs ex-vessel release even if it were to be non-negligible).

7. Require the Plant Designer to use the standardized plant PRA performed as part of design certification to check for any hidden vulnerabilities which could cause containment challenges exceeding those from the selected accident sequences and to confirm the physically-based source term specified in the Requirements Document.

1.7 Margin in Physically-Based Source Term

Consistent with the ALWR design philosophy described in Volume I of the ALWR Requirements Document, it is desired to have margin between the design basis source term and that which would occur, given a core damage event. However, it is also desired that the source term be based on a physical evaluation of a core damage event. The process described in Section 1.6 above is considered to provide appropriate margin while retaining the physical basis of the source term. Quantification of the individual aspects of the source term which is summarized in Section 2 and described in detail in Section 4 reflect this margin as well as incorporating a physical basis.

1.8 Organization of Report

This report is organized along the lines of the breakdown of the individual aspects of the source term. This breakdown is as follows:

- Definition of core damage event used to develop the physically-based source term
- Activity release from coolant and gap
- Early in-vessel release magnitude
- Late in-vessel release magnitude
- RCS retention of aerosols
- Revaporization release
- Ex-vessel release magnitude
- Chemical form
- Water pool scrubbing
- Primary containment aerosol removal
- Holdup and retention in secondary structures

The core damage event including thermal-hydraulic characteristics are defined in Appendix 1 and the various individual aspects of the source term are quantified in Section 4. The PWR and BWR integrated source terms are presented in Section 2. Section 3 presents a comparison of the ALWR physically-based source term with the NUREG-1150[1-7] expert elicitation on source term.

2. SUMMARY OF INTEGRATED SOURCE TERM FOR THE PASSIVE PWR AND PASSIVE BWR

2.1 Introduction

As specified in the ALWR Requirements Document,[2-1] a physically-based source term shall be used as the design basis for passive plant accident mitigation systems. Since each standard plant design will be different, the associated physically-based source term will be specific to that design. This report describes the bases for the source terms for the passive PWR and passive BWR designed to the ALWR requirements. This section provides a summary of each of the important aspects in deriving the source term and specifies the physically-based source term quantitatively. Detailed basis for each of these aspects is presented in Section 4.

The Passive ALWR source term is based on evaluation of a representative core damage event which results in a conservative, yet physically-based source term for the important sequence types. As noted in Section 1, a physically-based source term (vs. the current non-mechanistic approach embodied in the regulations) is considered to be necessary to provide a more rational basis for Passive ALWR mitigation system design and to incorporate the body of source term knowledge gained in the 30 years since TID 14844[2-2] was issued.

2.2 Core Damage Event for Estimating Source Term

The physically-based source term is developed from evaluation of a representative core damage event. Passive ALWR design requirements make the likelihood of any core damage event extremely remote. Examples of such requirements are the greatly improved man-machine interface, passive safety systems which do not depend on support systems, and the reduced need for operation action. Nonetheless, it is necessary to assume that a core damage event occurs in order to estimate a physically-based source term.

The core damage event defined for the purpose of estimating the source term is not intended to be a specific, PRA core damage sequence. Rather, it represents a more general, functional sequence with certain characteristics established to derive a conservative, yet physically-based source term. An event representative of a more general, functional sequence type is preferred over an individual PRA sequence since it does not depend upon the details of specific sequences or upon precise probabilistic quantification of such sequences.

Appendix I provides a summary of severe accident characteristics associated with sequences selected for the purpose of establishing the Passive ALWR source term. A complete set of functional sequence types for light water reactors was considered in identifying these characteristics. The functions considered to be important in establishing containment conditions associated with the source term are outlined in Chapter 5 of the ALWR Requirements Document and include the following:

- Reactivity Control
- Reactor Pressure Control
- Fuel/Debris Coolability
- Containment Pressure/Temperature Control
- Combustible Gas Control
- Containment Isolation
- Containment Bypass

Taking into account the ALWR design requirements for prevention and mitigation of severe accidents, selection of the functional sequence type(s) that should be considered in quantifying the physically-based source term was performed. Since multiple, independent means, at least one of which is passive, exist to perform each of the functions required to assure containment integrity in the Passive ALWR, the functional sequence type selected for the purpose of estimating the physically-based source term was a low RCS pressure core melt into an intact containment.

A representative core damage event was then defined based on the individual accident sequences associated with the selected functional sequence type. To provide margin in the physically-based source term, the characteristics of the core damage event have been established so as to give an enveloping, conservative estimate of the source term for these individual sequences. These characteristics are as follows:

- Rapid core damage progression to provide early fission product release and thus less time for radioactive decay and more time for leakage from primary containment.

- A vapor pathway in the RCS (i.e., from the core to the containment) to maximize fission product release to the containment atmosphere.

- A large scale core melt involving all or nearly all of the core.

- The potential for ex-vessel core damage progression.

For the passive PWR, a core damage event with these characteristics could be caused by a medium size loss of coolant accident (LOCA) with successful fourth stage depressurization but failure of the in-containment refueling water storage tank (IRWST) gravity drain. For the passive BWR, the core damage event could be caused by a liquid break below the core with successful RCS depressurization but failure of the gravity drain cooling system to inject.

The individual aspects of the physically-based source term are as follows:

- Activity release to containment from the coolant and gap;

- Early in-vessel release magnitude (from the fuel to the RCS prior to reactor vessel lower head penetration);

- Late in-vessel release magnitude (from the fuel remaining in the reactor vessel to the RCS after reactor vessel lower head penetration);

- RCS retention;

- Revaporization release;

- Ex-vessel release magnitude (from the ex-vessel debris);

- Chemical form;

- Scrubbing removal;

- Primary containment aerosol removal;

- Secondary building passive removal and holdup.

The core damage event timing, consistent with the above core damage event characteristics and relevant Passive ALWR design requirements, is defined in Table 2-1.

2.3 Physically-Based Source Term for the Passive PWR

2.3.1 Coolant and Gap Release

Coolant activity is being addressed since it is the earliest release into containment in the event of a LOCA. Potentially, the release of coolant activity could govern the containment isolation time. The radioactive element of concern is iodine. The evaluation focused on the passive PWR.

Current plant operating experience suggests that the peak plant equilibrium iodine concentration will not exceed about 0.1 µCi/g with the average plant being much lower. For these equilibrium levels, the data indicate that iodine spikes above 10 µCi/g are not credible. Given this experience together with the operating performance expected for the Passive ALWR, a reduction in the existing plant limits of 1.0 µCi/g and 60 µCi/g for equilibrium and spiking, respectively is warranted for the ALWR. A reduced equilibrium iodine limit of 0.3 µCi/g is proposed, together with a reduced spiking limit of 20 µCi/g with both limits applied to dose-equivalent I-131.

The release of this coolant activity at the start of a core damage accident has a negligible effect on source term compared with the fuel release initiated one hour into the event. Further, it is concluded that the initial coolant dose would be a small fraction of applicable dose limits and thus the doses would not control the containment isolation valve designs with either the existing or the proposed, reduced coolant activity limits.

TABLE 2-1. CORE DAMAGE EVENT TIMING

	Event	Time After Initiating Event	Relevant Requirements
1.	Core Uncovery	~1 hour	Large RCS Inventory, passive RCS heat removal which slows inventory loss, depressurized RCS, leak before break tending to limit size of RCS break, liquid break below core (BWR).
2.	Reactor Vessel Lower Head Penetration	~3 hours (BWR) ~5 hours (PWR)	Same as 1. Up to 75% of the reactor core material assumed to participate in the early stages of the melt progression.
3.	Ex-Vessel Debris Flooding	At Lower Head Penetration	Cavity/lower drywell flooded prior to or immediately upon lower head penetration.
4.	Ex-Vessel Release from Fuel Debris	At Lower Head Penetration	Limited due to debris cooling from flood; water pool also scrubs any release.
5.	Revaporization and Late In-Vessel Release	~3-24 hours (BWR) ~5-24 hours (PWR)	Remaining 25% of reactor core material relocates to cavity/lower drywell; assumed to begin immediately upon lower head penetration and to be complete by 24 hours; assumes a flooded cavity/lower drywell.

Gap activity was also addressed, focusing again on the passive PWR. Based on recent fuel performance experience and the conservative fuel design parameters for Passive ALWRs, a volatile fission gas release fraction of 3% is judged to provide adequate margin for use in the design basis source term for passive PWRs.

The Passive ALWR requirements specify that there shall be no fuel damage (and hence no gap release) for coolant breaks up to 6 inches in diameter. Hence, the physically-based source term assumes a gap release delayed until core uncovery. The gap release is treated together with the fuel release beginning one hour into the event.

2.3.2 Release Fractions to Primary Containment Atmosphere

Release fractions from the fuel to the RCS consist of early in-vessel releases (prior to reactor vessel lower head penetration) and late in-vessel release (subsequent to reactor vessel lower head penetration). Table 2-2 defines these release fractions. Detailed discussion of the bases for these release fractions are provided in Section 4.2 for early in-vessel release magnitude and Section 4.3 for late in-vessel releases. The release fractions are based on the most recent experimental fuel release data from the Severe Fuel Damage (SFD) Tests at the Power Burst Facility, the LOFT source term measurements, and the TMI-2 post accident examination. Releases to the reactor coolant system from the fuel are assumed to occur uniformly over time as early releases begin with the onset of core damage and late releases occur subsequent to vessel penetration.

These fuel release data were combined with experimental and analytical estimates of RCS retention for an intermediate size LOCA- initiated core melt (estimated to be 50% for iodine and 60% for other aerosols in the PWR as noted in Table 2-3) to define the early in-vessel fission product release to containment. These retention factors are a result of ALWR configuration which permits long residence time in the RCS following release from the fuel, turbulent deposition along potential release paths, such as the automatic depressurization system (ADS), and the presence of water and condensing steam in the RCS. No credit is taken for fission product aerosol flow through the depressurization lines to the IRWST since the fourth stage depressurization

TABLE 2-2. ALWR IN-VESSEL FUEL RELEASE ESTIMATES

	EARLY RELEASE (1)			LATE RELEASE (2)	
Element	Molten Fuel (3) (75%)	Remaining Fuel (4) (25%)	Total (6)	Remaining Fuel (5) (25%)	Total (6)
Nobles	1.0	0.25-0.30	0.80	1.0	0.20
I	0.9	0.25-0.30	0.75	0.9	0.15
Cs	0.9	0.25-0.30	0.75	0.9	0.15
Te			0.20	0.08	0.03
Sr			0.01		
Ba			0.01		
Ru			0.01		
La			0.0001		
Ce			0.0001		
Other			0.0001		

(1) Constant early release rate from ~1 hour to ~5 hours after accident (PWR), and from ~1 hour to ~3 hours (BWR).

(2) Constant late release rate from ~5 hours to 24 hours after accident (PWR), and from ~3 hours to 24 hours (BWR).

(3) Numbers are fraction of the original fission product inventory associated with the molten relocated fuel.

(4) Numbers are fraction of the original fission product inventory associated with the fuel remaining intact early, but melting and relocating late.

(5) Numbers are fraction of the fission product inventory associated with fuel remaining in- vessel (after early releases are complete).

(6) Numbers are fraction of the original total core fission product inventory.

TABLE 2-3. RCS RETENTION FRACTION FOR THE PWR

Aerosol	Retained Fraction (1)
I	0.5
All Other	0.6

FRACTION OF MATERIAL ORIGINALLY DEPOSITED IN RCS FOR THE PWR

Aerosol	Fraction Deposited
Iodine	0.38
Cesium	0.45
Tellurium	0.12
Sr, Ba	6E-3
Ru	6E-3
Other	6E-5

(1) Number is the fraction of material released from the fuel which is retained in the RCS.

location is currently in the loop compartment. Analytical and experimental bases for these retention factors are provided in Section 4.4.

The late in-vessel release to containment is the sum of the late in-vessel fuel release (Table 2-2) and the PWR revaporization release. The revaporization release is the product of the fuel release, the RCS retention, and the revaporization fraction.

The revaporization fraction is shown on Table 2-4 and is estimated for a wet cavity due to ALWR requirements for flooding the cavity during an accident. A discussion of the bases for the revaporization fraction is presented in Section 4.5. Revaporization is limited in the ALWR due to the presence of relatively low ambient conditions within the RCS due to available water and steam and the limited potential for buoyancy driven flow through the vessel following lower head penetration as a result of reactor cavity flooding to above the lower head.

The ex-vessel release magnitude is considered to be very small (i.e., release from the molten fuel debris which has penetrated the reactor vessel lower head and is located in the cavity/lower drywell). Release fractions from an uncooled, dry debris bed have been estimated in Table 2-5. These releases are based on a combination of ACE corium concrete tests and VANESA analyses as discussed in Section 4.6. However, ALWR requirements for cavity/lower drywell spreading area and water flooding provide rapid core debris quenching and thus will prevent significant core concrete interaction and ex-vessel release. For this reason, the ex-vessel contribution to fission products in the containment atmosphere in the 5-24 hour period is considered negligible. In addition, it has been noted that the passive PWR cavity has significantly more water present at the time of lower head penetration than needed simply to cool the debris. Even if some ex-vessel release from the debris occurs, there will be scrubbing of aerosols. The effects of pool scrubbing are presented in more detail in Section 4.8. Experimental and analytical studies indicate that for core debris water pool conditions (i.e., several meters deep, and either low steam fraction and subcooled, or high steam fraction and saturated) a pool decontamination factor of 10 or more is reasonable. Thus, the release to the containment atmosphere, should any release from ex-vessel debris occur, would be minimal. Total fission product

TABLE 2-4. FRACTION OF MATERIAL DEPOSITED IN RCS THAT REVAPORIZES OVER 5-24 HR. TIME FRAME FOR THE PWR

Aerosol	Wet Cavity	Dry Cavity[(1)]
Iodine	0.06	0.15
Cesium	0.055	0.10
Tellurium	0	0
Other	0	0

FRACTION OF ORIGINAL CORE MATERIAL THAT REVAPORIZES FOR THE PWR[(2)]

Aerosol	Wet Cavity	Dry Cavity[(1)]
Iodine	0.02	0.06
Cesium	0.03	0.04

(1) The dry cavity case is included for perspective only and does not apply to the Passive ALWR due to the early cavity/lower drywell flooding feature.

(2) Fractions are the product of the material originally deposited in RCS (Table 2-3) and the fraction of that which revaporizes from the table above.

TABLE 2-5. RELEASE ESTIMATES FROM EX-VESSEL FUEL DEBRIS FOR THE ALWR IN THE EVENT THAT DEBRIS BED IS UNCOOLED

Chemical Species	Fraction of Original Core Inventory Released From Debris
I, Cs	0.10
Te, Sb	0.35
Ru	0.01
Sr, Ba	0.002
Remainder	0.001

release fractions to containment as a function of time are presented in Table 2-6.

2.3.3 Chemical Form

The chemical form of radionuclide releases to the containment is developed in Section 4.7. The nobles are gaseous form. Iodine is 97% particulate, 2.85% elemental, and 0.15% organic. The remaining nuclides are particulate. This is based on recent experimental data including that from the SFD tests, LOFT, STEP tests, TMI-2 post accident examination, and the ACE tests as well as an extensive review of the potential chemical reactions in the RCS and containment. Also, ALWR requirements such as early flooding of the cavity and basic pH in the sump are considered.

2.3.4 Containment Fission Product Behavior

Explicit analysis of fission product behavior in the Passive ALWR containment atmosphere has been performed using NAUA and is presented in Section 4.9. Aerosol fission product removal from the containment atmosphere was calculated considering gravitational settling on horizontal surfaces and plateout, principally diffusiophoresis. The diffusiophoretic effect is significant because of the passive containment heat removal system which rapidly condenses steam generated from decay heat and from quenching of core debris. CsOH hygroscopicity was also considered in the treatment of particle growth. All removal processes were consistent with the thermal hydraulic conditions for an intermediate size LOCA with IRWST gravity drain failure, and were based on physical processes of aerosol mechanics which have been incorporated in calculational models and benchmarked against experimental data. The fission product release rate and the steaming rate from the core during boiloff and in-vessel core melt progression were assumed to be uniform. Parametric studies indicate that the total fission product leakage from primary containment is not particularly sensitive to this release rate assumption. The original core inventory was assumed to be that at the end of an equilibrium, 2 year operating cycle.

TABLE 2-6. PWR RELEASE FRACTIONS TO PRIMARY CONTAINMENT ATMOSPHERE[6]

Nuclide	0-1 hr. Coolant Activity	1-5 hr.[1,4] Early In-Vessel	5 hr.[2] Ex-Vessel	5-24 hr.[3] Late-In Vessel	Total
Nobles	See Note (5)	0.80	--	0.20	1.0
I		0.38	--	0.17	0.55
Cs		0.30	--	0.18	0.48
Te		0.08	--	0.03	0.11
Sr, Ba		0.004	--	--	0.004
Ru		0.004	--	--	0.004
Remainder		0.00004	--	--	0.00004

(1) Assumes in-vessel releases from Table 2-2 and RCS retention of a part of the early releases as given in Table 2-3.

(2) All nobles released either early or late in-vessel. Remaining fission products retained in quenched debris or scrubbed through overlying water pool in reactor cavity.

(3) Late in-vessel releases are sum of all late fuel releases (Table 2-2) and revaporization releases for wet cavity (Table 2-4).

(4) As noted in Section 4.1, the gap activity is included in the 1-5 hr. release.

(5) Coolant activity limits are quantified in Section 4.1. Although important for steam generator tube rupture and steamline break, coolant activity makes a negligible contribution to the source term from a core damage event and so is not included here.

(6) All numbers are fraction of original core fission product inventory.

Results of the analysis are presented in Figure 4-10. Aerosol leakage from containment over the first 24 hours of the accident totals 99 gm. Sensitivity studies were performed on various assumptions regarding the accident sequence progression. Most notably, the containment aerosol concentration is significantly reduced by condensation which occurs during the short quenching period following entry of the debris into the reactor cavity (or alternatively, by that associated with quenching the core debris in-vessel had the vessel not been penetrated). Aerosol leakage was shown to be relatively insensitive to large variations in the steam injection rate associated with this quench. Aerosol leakage rate as a function of time was used as input to the secondary building retention and holdup analysis.

2.3.5 Secondary Structure Fission Product Holdup and Retention

The following summarizes an evaluation of the potential for secondary structure holdup and retention presented in Section 4.10. Treatment of fission product aerosols leaking from containment to secondary building rooms and piping systems is similar to containment aerosols in that it is based on physical processes of aerosol mechanics which have been incorporated into benchmarked calculational models. Removal mechanisms, primarily gravitational settling, are considered consistent with the thermal hydraulics of the secondary building room into which the leakage occurs. All leakage pathways from the primary containment into the secondary building (i.e., from the primary containment atmosphere directly into a secondary building room, and from piping systems penetrating the primary containment with a potential leak location in the secondary building) must be included. Also, bypass pathways directly from the containment to the environment must be addressed.

Given a well-designed building and proper modeling of the holdup and retention characteristics, it is likely that an effective building decontamination factor of six or more can be demonstrated. Thus, the 99 gram integrated, 24 hour fission product release from containment calculated for the PWR in Section 4.9 would result in about a 15 gram release to the environment.

Preliminary dose calculations were performed considering the secondary structure holdup and retention. For containment leak rates at or slightly under 0.5%/day, the median 24 hour dose would not be expected to exceed the Protective Action Guidelines (PAGs) at 0.5 miles from the reactor.

2.4 Physically-Based Source Term for the Passive BWR

2.4.1 Coolant and Gap Release

The separate treatment of coolant and gap releases was not expected to be significant for the physically-based source term. To confirm this expectation, evaluations were performed that focused on the passive PWR.

While coolant activity in the passive BWR was not investigated in detail, improvements in BWR operating activity levels are expected to be comparable to those for PWRs (Technical Specification limits reduced by a factor of three). The release of this coolant activity at the start of a core damage accident has a negligible effect on source term compared with the fuel release initiated one hour into the event. Further, it is concluded that the initial coolant dose would be a small fraction of applicable dose limits and thus the doses would not control the containment isolation valve designs with either the existing coolant activity limits or any reduced limits that may be proposed.

The Passive ALWR requirements specify that there shall be no fuel damage (and hence no gap release) for coolant breaks up to 6 inches in diameter. Hence, the physically-based source term assumes a gap release delayed until core uncovery. The gap release is treated together with the fuel release beginning one hour into the event. Given this approach, the design basis accident source term for BWRs is also appropriate even though gap release was not specifically evaluated for the passive BWR.

2.4.2 Release Fractions to Primary Containment Atmosphere

The BWR release fractions from the fuel to the RCS are the same as for the PWR, defined in Table 2-2 and established in Section 4.2 (early releases) and 4.3 (late releases). The RCS retention is slightly higher (Table 2-7) than the PWR due to the large surface area of the steam separators and dryers. No credit is taken for the isolation condenser in removing early in-vessel

TABLE 2-7. RCS RETENTION FRACTION FOR THE BWR

Aerosol	Retained Fraction
I	0.6
All Other	0.7

FRACTION OF MATERIAL ORIGINALLY DEPOSITED IN RCS FOR THE BWR

Aerosol	Fraction Deposited
Iodine	0.45
Cesium	0.53
Tellurium	0.14
Sr, Ba	7E-3
Ru	7E-3
Other	7E-5

(1) Number is the fractio.ı of material released from the fuel which is retained in the RCS.

aerosol release. Thus, the BWR early in-vessel release to containment is obtained from the product of the fuel release and RCS retention.

The late in-vessel release to containment is the sum of the late in-vessel fuel release (Table 2-2) and the BWR revaporization release. The revaporization release is the product of fuel release, RCS retention, and revaporization fraction. The revaporization fractions are defined in Table 2-8 and discussed in Section 4.5. As discussed for the PWR, the relatively low ambient conditions resulting from water and condensing steam and the lack of buoyancy driven flow due to flooding the lower drywell and reactor vessel lower head assists in limiting the magnitude of revaporization.

With regard to ex-vessel releases, the PWR debris coolability and core debris water pool scrubbing discussion above and in Sections 4.6 and 4.7 are equally applicable to the BWR. Further, as noted below, a fraction of the aerosol which is suspended in the drywell will be scrubbed in the suppression pool due to drywell pressurization from steaming. Thus, the BWR ex-vessel contribution to the release fraction in the 3-24 hour period in Table 2-9 is also shown as negligible.

Table 2-9 presents the total fission product release fractions to the containment atmosphere.

2.4.3 Chemical Form

The chemical form for the BWR fission products is considered to be the same as for the PWR based on the Section 4.7 discussion.

2.4.4 Containment Fission Product Behavior

The treatment of BWR containment fission product behavior is similar to that of the PWR. The BWR also has scrubbing in the suppression pool. Like the PWR, an explicit analysis of BWR fission product behavior in the containment was performed (see Section 4.9).

TABLE 2-8. FRACTION OF MATERIAL DEPOSITED IN RCS THAT REVAPORIZES OVER 3-24 HR. TIME FRAME FOR THE BWR

Aerosol	
Iodine	0.10
Cesium	0.05
Tellurium	0
Others	0

FRACTION OF ORIGINAL CORE MATERIAL THAT REVAPORIZES FOR THE BWR(1)

Aerosol	
Iodine	0.05
Cesium	0.03

(1) Fractions are the product of the original material deposited (Table 2-7) and the fraction of that which revaporizes from the table above.

TABLE 2-9. BWR RELEASE FRACTIONS TO PRIMARY CONTAINMENT ATMOSPHERE[6]

Nuclide	0-1 hr. Coolant Activity	1-3 hr.[1,4] Early In-Vessel	3 hr.[2] Ex-Vessel	3-24 hr.[3] Late-In Vessel	Total
Nobles	See Note (5)	0.80	--	0.20	1.0
I		0.30	--	0.20	0.50
Cs		0.23	--	0.18	0.41
Te		0.06	--	0.03	0.09
Sr, Ba		0.003	--	--	0.003
Ru		0.003	--	-	0.003
Remainder		0.00003	--	--	0.00003

(1) Assumes in-vessel releases from Table 2-2 and in-vessel retention of a part of the early releases from Table 2-7.

(2) All nobles released either early or late in-vessel. Remaining fission products retained in quenched debris or scrubbed through overlying water pool in lower drywell.

(3) Late in-vessel releases are sum of all late fuel releases (Table 2-2) and revaporization releases (Table 2-8).

(4) As noted in Section 4.1, the gap activity is included in the 1-3 hr. release.

(5) Coolant activity limits are discussed in Section 4.1. Coolant activity makes a negligible contribution to the source term for core damage events and so is not included here.

(6) All numbers are fraction of original core fission product inventory.

The treatment of fission product removal in the suppression pool is based on experimental data for pool scrubbing and results of calculational models, together with the thermal hydraulic conditions in the drywell and wetwell. For aerosols forced through the pool, a high enough fraction is scrubbed that the residual aerosol leakage after pool scrubbing is negligible over a 24 hour period.

Flow of fission product aerosols through the drywell vents to the wetwell has been assumed to occur only at the time of reactor vessel lower head penetration. At ~3 hours, steaming from the ex-vessel debris is assumed to rapidly force a significant fraction of the drywell contents through the drywell vents to the suppression pool where scrubbing occurs.

No credit was taken for scrubbing of aerosol which flows through the isolation condenser vent to the suppression pool due to the shallow submergence of the sparger (~1 foot).

Results of the analysis are presented in Figure 4-9. Aerosol Leakage over the first 24 hours of the accident total 35 g. This is about one-third that presented for the PWR and is attributed to early suppression pool scrubbing during periods in which relatively significant steam addition to the drywell is occurring and the fact that MSIV leakage is not included. Sensitivity studies were performed on various assumptions regarding accident sequence progression. Like the PWR results, aerosol leakage was shown to be relatively insensitive to substantial variations in steaming rate during the quenching period following debris relocation to the lower drywell. Also, the fission product removal through operation of the isolation condenser is likely to be a significant contributor to aerosol reduction that is not credited in the analysis.

2.4.5 Secondary Structure Fission Product Holdup and Retention

Treatment of fission product aerosols leaking from the containment to the BWR reactor building and piping systems is similar to that of the PWR. Section 4.10 provides an evaluation of potential release paths through the BWR reactor building. There are three possible pathways: (a) from the primary containment atmosphere to the reactor building, (b) from the primary containment atmosphere through purge lines to the environment, and (c) through the MSIVs.

The MSIV fission product leakage flows into the main steam lines and ultimately to the main condenser. Fission product removal and holdup will be evaluated under these conditions once the results of the NRC review of the BWR Owners Group effort on existing plants in this regard are available. The quantification of the direct leakage through the purge lines should take into account features to minimize this leakage. Finally, the leakage into the reactor building should consider the building arrangement which requires the leakage to pass through several separate areas before reaching the environment.

Although the Section 4.10 feasibility assessment was performed for the passive PWR, the technical basis developed for quantifying retention and holdup is considered equally applicable to the passive BWR. Thus a factor of six or greater would be expected for the passive BWR reactor building DF.

2.5 Comparison of Physically-Based Source Term Containment Release with Existing Regulatory Guidance

Table 2-10 provides a comparison of the timing, magnitude, and chemical form of the release to the containment atmosphere for the Passive ALWR physically based source term vs. the existing regulatory source term.

The ALWR release is over a period of 24 hours (although most of the release occurs in the first several hours of the accident) vs. instantaneous release for the existing regulatory source term. The magnitude for nobles is identical. The iodine release is 50% for ALWR vs. 25% for existing regulatory, although the ALWR iodine release is spread out over a period of hours as noted above. The ALWR source term releases roughly half of the core inventory of cesium and lesser amounts of remaining elements, whereas the existing regulatory guidance has no other release to the containment atmosphere. The ALWR iodine chemical form is roughly a reversal of the elemental-particulate ratio for the existing regulatory source term. The

TABLE 2-10. COMPARISON OF RELEASE TO CONTAINMENT FOR PASSIVE ALWR SOURCE TERM AND EXISTING REGULATORY SOURCE TERM

	Passive PWR	Passive BWR	Existing Regulatory Source Term
Release Timing	Release over a 24 hr period beginning 1 hr after initiating event	Release over a 24 hr period beginning 1 hr after initiating event	Instantaneous release at time of initiating event
Release Magnitude to Containment Atmosphere			
• Nobles	100%	100%	100%
• Iodine	55%	50%	25% (1)
• Cesium	48%	41%	1% (to sump)
• Tellurium	11%	9%	1% (to sump)
• Ba, Sr, Ru	0.4%	0.3%	1% (to sump)
• Remainder	0.004%	0.003%	1% (to sump)
Chemical Form in Containment			
• Iodine	2.85% elemental 97% particulate 0.15% organic	2.85% elemental 97% particulate 0.15% organic	91% elemental 5% particulate 4% organic
• Cesium	100% particulate	100% particulate	Not Specified
• Tellurium and remaining Semi- and Low Volatiles	100% particulate	100% particulate	Not Specified

Notes: (1) The 25% figure is arrived at by the Regulatory Guide 1.3, 1.4 assumptions that 50% of the iodine inventory is released to the organic iodine fraction is correspondingly less for ALWR due to the low fraction of elemental iodine.

In general, the ALWR integrated release to containment is significantly higher than the existing regulatory source term. This difference is due in part to the ALWR objective that the source term be based on a physically-based evaluation of a core damage event (hence, the release over a period of hours, the release of cesium and other elements, and the particulate form of iodine). It is also due to the desire expressed above that the ALWR physically-based source term incorporate margin beyond the source term expected from an actual ALWR core damage event.

2.6 Format of the Physically-Based Source Term Expression

The sections above have described the fission product release and transport associated with the Passive ALWR physically-based source term. In fact, the source term may be expressed as a transient release to the environment for a given standard plant design (i.e., containment design, design leak rate, and secondary building design). This provides a simpler expression and may be preferable as a format for characterizing the source term for a given standardized plant design or for source term regulatory guidance.

3. COMPARISON OF PASSIVE ALWR SOURCE TERM WITH NUREG-1150 RESULTS

The purpose of this section is to provide a comparison of NUREG-1150[3-1] source term results with the ALWR physically-based source term. NUREG-1150 documents a PRA study of five U.S. commercial nuclear power plants. The second draft of the study was published in April, 1989 and represents an update, extension, and improvement upon the 1975 risk study, WASH-1400.[3-2] Thus, NUREG-1150 reflects current NRC staff and contractor thinking regarding source term.

Two types of comparisons are provided. The first is a comparison of individual aspects of the source term, e.g., ALWR in-vessel release vs. NUREG-1150 in-vessel release. The second is a comparison of the integrated source term as measured by the core fraction released from the containment.

3.1 Comparison Of Individual Aspects Of Source Term

NUREG-1150 develops source terms for individual accident sequences as is normally the case for a PRA. Probabilistic density functions were developed for the principal aspects of the source term by eliciting the judgments of experts in various relevant phenomena. It was intended that the uncertainty ranges represent the modeling uncertainty associated with the phenomena and not the variability that exists for different accident sequences. Thus, more than one density function was often obtained for a single aspect of the source term associated with different accident sequences.

As noted in Sections 1 and 2 above, the Passive ALWR source term was developed to provide a single, enveloping value for representative accident sequences using physically-based estimates of source term phenomena. Thus, the individual aspects of the ALWR source term are best compared with a central value of the closest corresponding NUREG-1150 distribution. The NUREG-1150 median was used for the comparison. In addition, the effect on the overall ALWR source term (i.e., release to the environment and the resulting offsite dose) of uncertainties reflected in the NUREG-1150 distributions is evaluated.

3.1.1 Comparison of In-Vessel Release

The ALWR in-vessel release fractions are compared with the NUREG-1150 values in Table 3-1 and with the NUREG-1150 distribution in Figure 3-1. The early and late periods of in-vessel release for the ALWR source term have been added. The specific NUREG-1150 case selected for comparison was the PWR low zirconium oxidation case. The only significant difference for other cases would be a greater tellurium release for high zirconium oxidation.

The agreement between the ALWR and NUREG-1150 releases is very close for all elemental groups. Although the overall NUREG-1150 uncertainty is quite large, the uncertainty in the direction of higher release for the more volatile elements that tend to dominate accident consequences is very small. The only exception is tellurium, although increases in tellurium release do not have a major effect on dose. For example, an increase in the tellurium release by a factor of 2 (i.e., to 0.44) increases whole body dose by only about 15%. Similarly, for the low volatile elements, increases in the in-vessel release do not have a significant effect on release to the environment and offsite dose. For example, increasing the in-vessel release fractions of Sr, Ba, Ru, La, and Ce as a group by a factor of 5 over the ALWR estimate would be expected to increase offsite acute whole body dose by only about 50%.

3.1.2 Comparison of RCS Retention

The ALWR and NUREG-1150 values for RCS retention are in close agreement as indicated in Table 3-2. Figure 3-2 illustrates a comparison between the ALWR values and the NUREG-1150 distributions for a typical PWR case. The BWR distribution is similar.

TABLE 3-1. COMPARISON OF ALWR IN-VESSEL RELEASE WITH NUREG-1150 MEDIAN

Element	ALWR(1)	NUREG-1150 Median(2)
Nobles	0.9	0.9
I	0.9	0.74
Cs	0.9	0.59
Te	0.22	0.15
Sr	0.01	0.0064
Ba	0.01	0.0086
Ru	0.001	0.0046
La	0.0001	0.0001
Ce	0.0001	0.00015
Other	0.0001	---

(1) Numbers represent the sum of the early and late in-vessel release fractions.

(2) PWR low Zr oxidation.

S-G03200-004
020491R91A

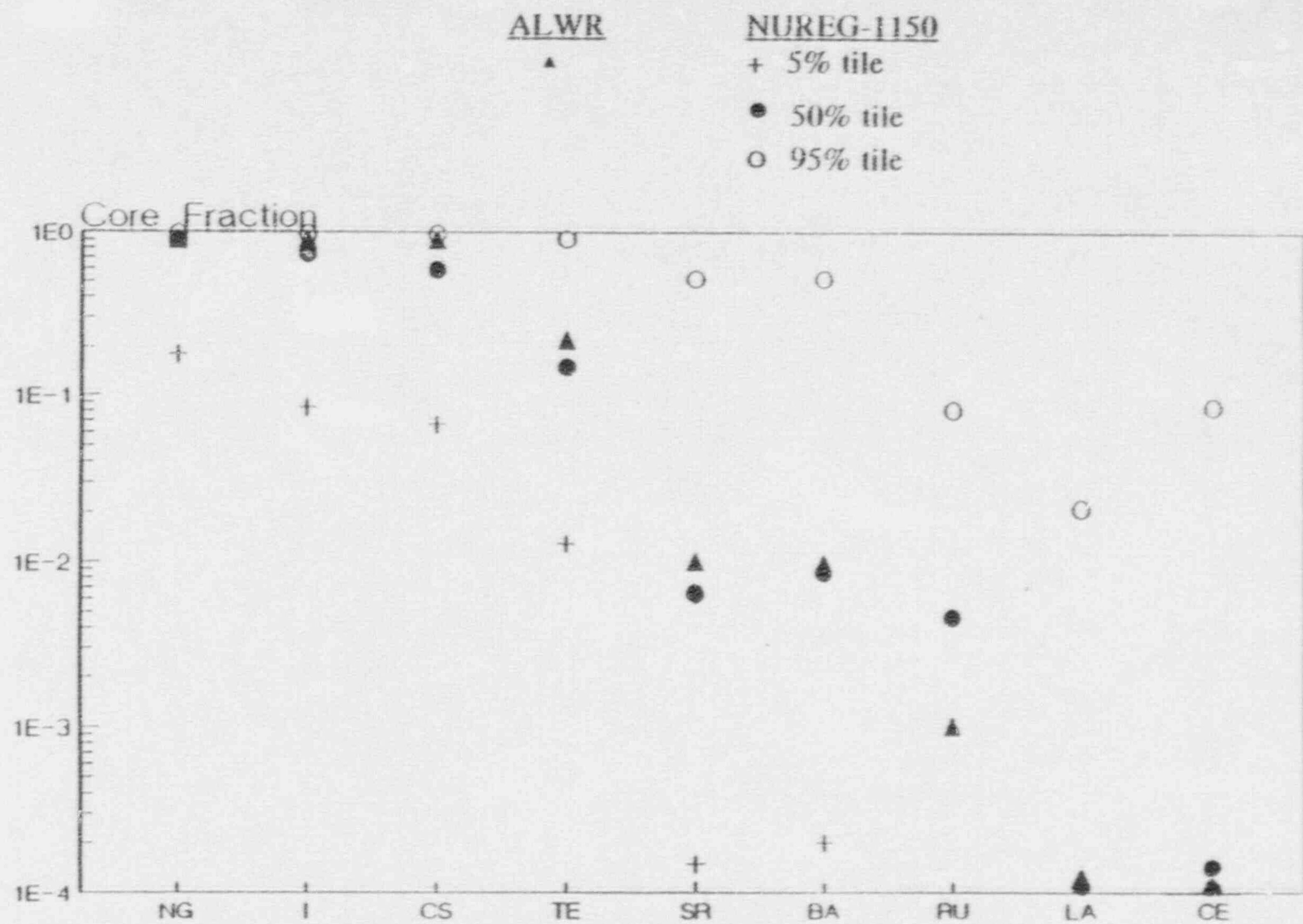


Figure 3-:   Comparison of ALWR In-Vessel Release with NUREG-1150 Distribution.

TABLE 3-2. COMPARISON OF ALWR RCS RETENTION WITH NUREG-1150 MEDIAN

Element	ALWR		NUREG-1150 Median	
	PWR	BWR	PWR	BWR
Nobles	0.0	0.0	0	0
I	0.5	0.6	0.48	0.59
Cs	0.6	0.7	0.60	0.70
All Others	0.6	0.7	0.67	0.74

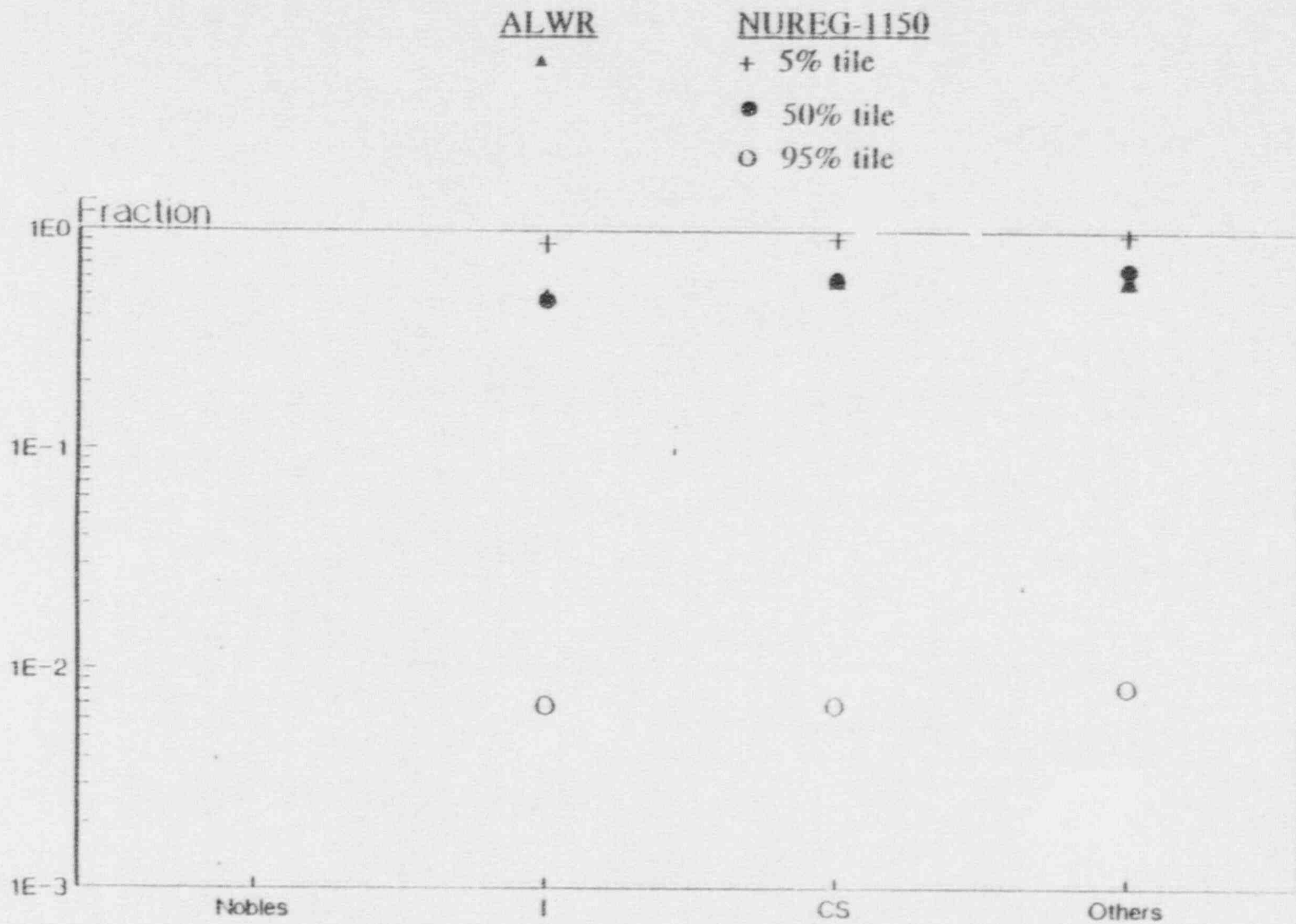


Figure 3-2. Comparison of ALWR RCS Retention with NUREG-1150 Distribution.

Although the NUREG-1150 uncertainty range is broad, lower RCS retention in the ALWR is not expected to have a significant effect on offsite dose since the fraction of aerosol retained is only about half. For example, reducing the RCS retention by a factor of two would increase the early (i.e., 1-5 hours for PWR) release to the containment atmosphere by about 50%. Since the early release constitutes about half of the total release from containment, the increase in offsite dose would be only about 25%.

3.1.3 Comparison of Revaporization Release

The ALWR and NUREG-1150 median values for revaporization following reactor vessel lower head penetration are in close agreement as indicated in Table 3-3. Figures 3-3 and 3-4 provide a comparison of the ALWR value with the NUREG-1150 distribution for the PWR and BWR, respectively.

Again, although the NUREG-1150 uncertainty range is broad, higher revaporization in the ALWR is not expected to have a significant effect on offsite dose, largely because the contribution of revaporization to total dose is small, i.e., on the order of a few percent. For example, a factor of 5 increase in revaporization would be expected to increase the late (i.e., 5-24 hours for PWR) release to the containment atmosphere by about 50%. This in turn would increase the offsite dose by about 25%.

3.1.4 Comparison of Ex-Vessel Release

As explained in Section 2 above, it is expected that the ex-vessel aerosol release from the core debris for the Passive ALWR will be negligible due to the requirement for early flooding of the cavity/lower drywell. Further, even if some release from the core debris were to occur, scrubbing from the overlying water pool would largely remove the aerosol and prevent its release to the containment atmosphere. Therefore, ex-vessel releases for the ALWR are not directly comparable to the NUREG-1150 values.

TABLE 3-3. COMPARISON OF ALWR REVAPORIZATION FRACTION WITH NUREG-1150 MEDIAN

Element	ALWR(1) PWR	BWR	NUREG-1150 Median(1) PWR	BWR
Iodine	0.06	0.10	0.06	0.11
Cesium	0.055	0.05	0.055	0.05
Tellurium	0	0	0	0

(1) Numbers represent the fraction of the mass deposited in the RCS which is revaporized by 24 hours.

S-G03200-004
020491R91A

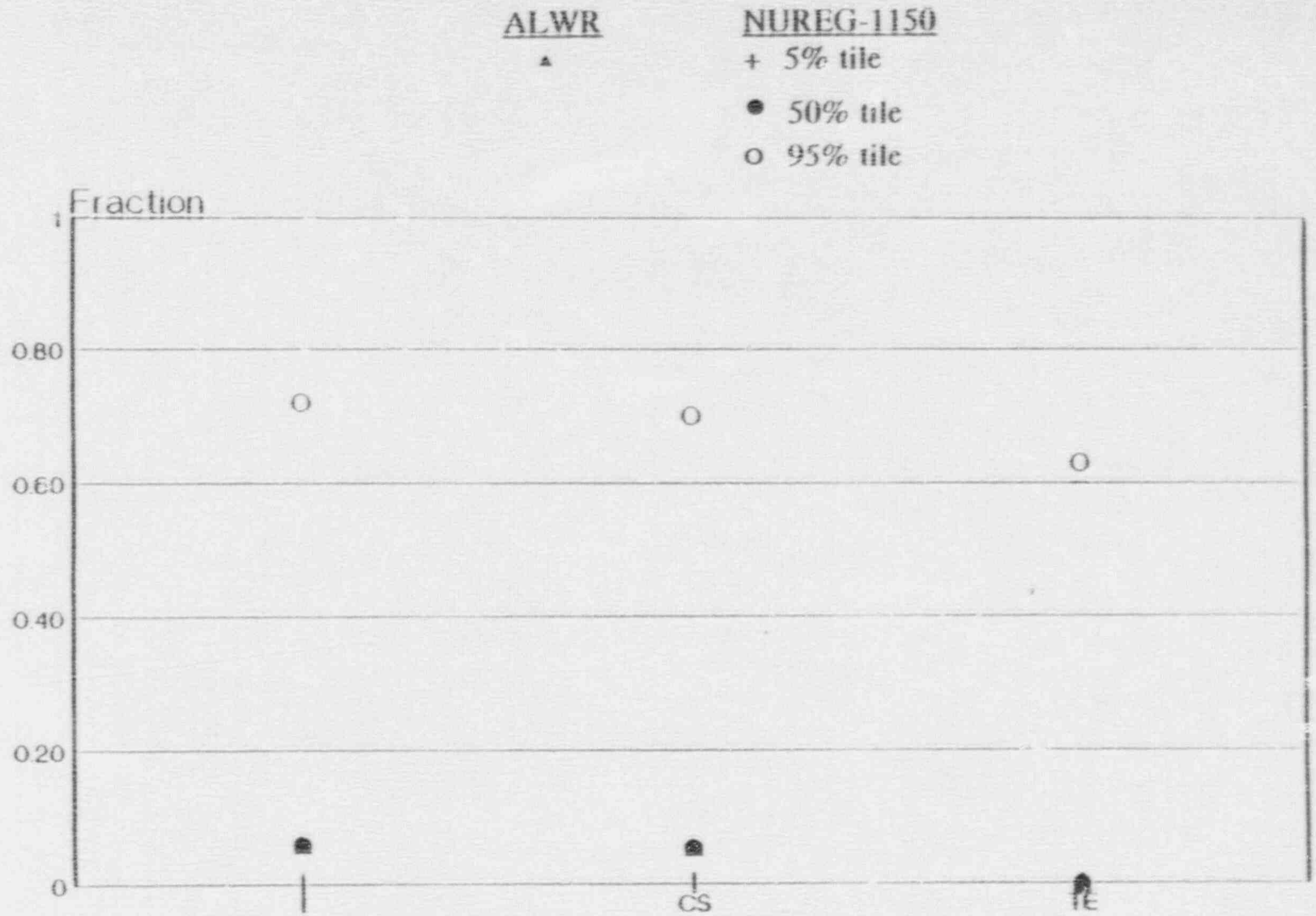


Figure 3-3. Comparison of ALWR Revaporization Release with NUREG-1150 Distribution for PWR.

S-G03200-004
020491R91A

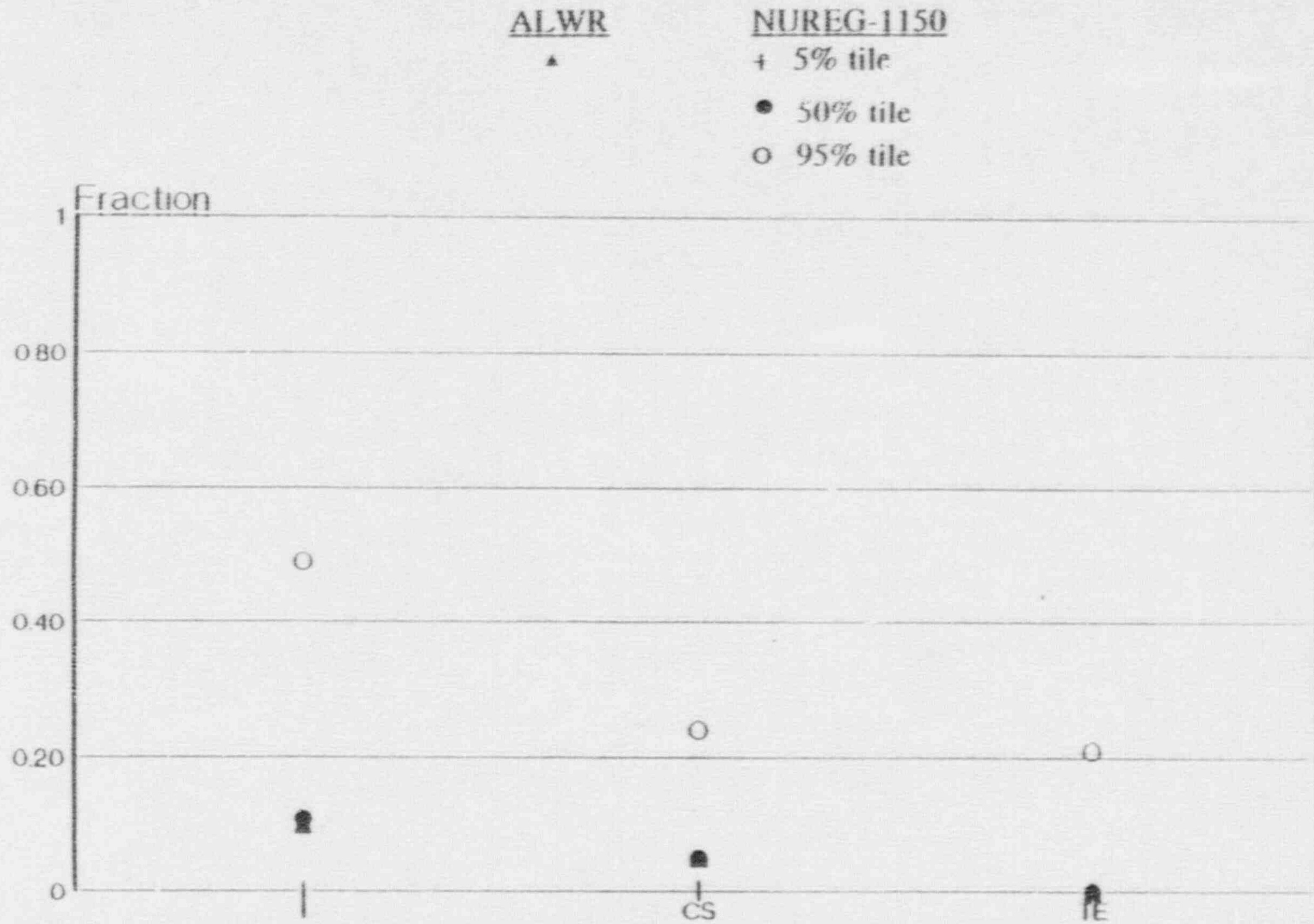


Figure 3-4. Comparison of ALWR Revaporization Release with NUREG-1150 Distribution for BWR.

Estimates have been made for a scenario in which there is a period of dry core concrete interaction and these estimates are compared with the NUREG-1150 median value for ex-vessel release with no cavity flooding in Table 3-4. The NUREG-1150 distributions are shown in Figure 3-5. It is noted that the addition to the containment atmosphere source term from the ex-vessel release is relatively minor since most of the volatiles have already been released, and since the additional low volatile release is small.

3.1.5 Summary of Comparison

Four of the main aspects of the source term have been compared for the ALWR physically-based source term and NUREG-1150 results. Other aspects, such as engineered safety feature effects and natural removal of aerosols in containment, are very dependent upon plant design features which are quite different for the Passive ALWR and the NUREG-1150 plants. Further, the fact that design requirements have been provided so as to essentially preclude containment failure in the Passive ALWR allows much greater opportunity for fission product removal inside containment than in the NUREG-1150 evaluations.

Based on the four comparisons which were made, the ALWR values are in reasonable agreement with the NUREG-1150 median values. Further, changes in the ALWR values to reflect the uncertainties existing in the NUREG-1150 distributions do not have a large effect on the integrated release to the environment.

3.2 Comparison of Integrated Source Term

To provide additional perspective, a comparison of the core fraction released to the environment for ALWR, WASH-1400, and NUREG-1150 has been made. For WASH-1400, a PWR-7 case is used, i.e., an intact primary containment with containment sprays operating. For NUREG-1150, a directly comparable case is not available, so the closest case was used, i.e., containment intact early, with late overpressure failure. Also included in the comparison are the releases for early containment failure cases from WASH-1400 and NUREG-1150.

TABLE 3-4. COMPARISON OF ALWR EX-VESSEL RELEASE (DRY CAVITY) WITH NUREG-1150 MEDIAN

Element	ALWR(1)	NUREG-1150 Median(2)
Nobles	1.0	1
I	1.0	1
Cs	0.9	1
Te	0.45	0.55
Sr	0.002	0.034
Ba	0.002	0.025
Ru	0.001	5.6(-9)
La	0.001	0.00071
Ce	0.001	0.00097

(1) Numbers represent ALWR estimates of ex-vessel releases (fraction of radioncludes in the debris as it exits the reactor vessel) if cavity were to be dry for ~30 minutes after reactor vessel lower head penetration.

(2) PWR, low Zr oxidation, no water in cavity.

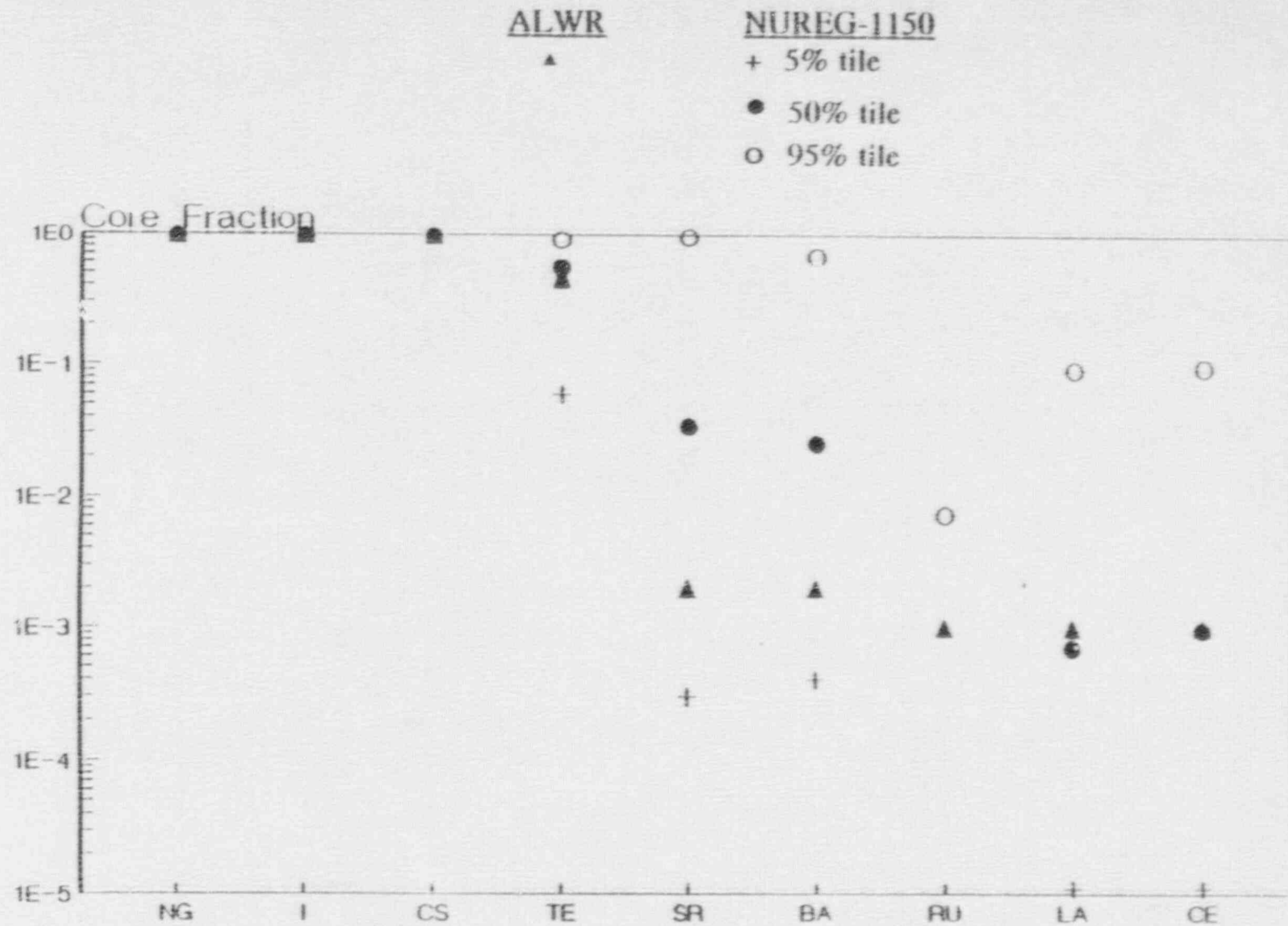


Figure 3-5. Comparison of ALWR Ex-Vessel Release (Dry Cavity) with NUREG-1150 Distribution.

S-G03200-004
020491R91A

Table 3-5 shows the comparison. The following points are relevant:

- The ALWR release is for an intact containment and the WASH-1400 release is associated with basemat melt-through which, in effect, is an intact containment insofar as release to the atmosphere is concerned. On the other hand, the NUREG-1150 release includes a late overpressure failure which has a non-trivial release of iodine.

- The ALWR release includes natural removal of aerosol inside containment (but no spray system). The ALWR releases compare reasonably closely with WASH-1400 except for iodine and cesium. The difference is in part due to the fact that the WASH-1400 releases include the effect of sprays. The ALWR secondary building effect noted in footnote (2) to Table 3-5 would make the actual I and Cs release to the environment comparable for ALWR and WASH-1400.

- The ALWR and NUREG-1150 releases compare reasonably closely, again except for the late iodine release in NUREG-1150.

- The early containment failure releases illustrate two points. First, the NUREG-1150 releases are significantly lower than WASH-1400, due for the most part to better understanding of source term phenomena (e.g., higher retention of aerosols within containment), and due to higher failure pressure ascribed to the containment in NUREG-1150.[3-3] Second, even with the large reduction in the NUREG-1150 release compared to WASH-1400, the early containment failure release is still significantly higher than the intact containment cases.

TABLE 3-5. COMPARISON OF INTEGRATED SOURCE TERM FOR ALWR, WASH-1400, AND NUREG-1150

	Intact Containment			Failed Containment	
	ALWR	WASH-1400	NUREG-1150	WASH-1400	NUREG-1150
	(1,2)	(1,4)	(1,3,4)	(1,4)	(1,4)
Nobles	3E-3	6E-3	--	0.9	--
I	2E-4	4E-5	2E-2	0.7	4E-2
Cs	2E-4	1E-5	<1E-5	0.5	1E-2
Te	4E-5	2E-5	<1E-5	0.3	5E-3
Sr	2E-6	1E-6	<1E-5	0.06	2E-4
Ru	2E-7	1E-6	<1E-5	0.02	1E-4
La	2E-8	1E-7	<1E-5	0.003	3E-5

(1) All numbers represent core fraction released from the containment.

(2) To obtain the actual release to the environment, the ALWR aerosol release fractions would need to be reduced by a factor of about six to account for secondary building holdup and removal.

(3) These NUREG-1150 releases are for an intact containment early in the accident with a late overpressure failure as noted in Reference 1-7, Volume 1, page 10-4.

(4) Median values were used for NUREG-1150. Only median data were available from WASH-1400.

4. INDIVIDUAL ASPECTS OF THE PHYSICALLY-BASED SOURCE TERM

4.1 Coolant and Gap Activity Source Terms

4.1.1 Introduction

In the event of a LOCA the first release to the containment is the activity circulating in the RCS. The element of primary concern (because of its biological effect) is iodine. Although the total quantity of iodine released during this phase of the accident, i.e., prior to any fuel failure, is small, the fact that this release occurs rapidly makes it potentially important for such issues as containment isolation time. For this reason, the release of coolant prior to the occurrence of fuel failure is addressed in this section.

It is also possible during a LOCA and certain transient conditions to release gap activity to the containment prior to more extensive fission product release from widespread core damage. For this reason, the release of gap activity is also addressed in this section.

Releases from the coolant and gap are most significant for design basis accidents that are terminated without more extensive fuel damage though such accidents are beyond the scope of this report. The contribution of distinct coolant and gap releases to the physically-based source term for an assumed core damage event is not expected to be important. Nevertheless, an evaluation was performed, focussed on the passive PWR, to confirm this expectation for an advanced, passive LWR.

4.1.2 Coolant Activity Phenomena and Assumptions

Nuclear plants are designed to operate with some cladding defects which permit the release of radioactive fission gasses to the reactor coolant system. For a given defect level, an equilibrium activity tends to be established in the reactor coolant system based on the rate of radioactive fission gas production and release and the operation of installed cleanup systems.

Coolant Iodine Spikes

Under certain conditions, the reactor coolant iodine concentration has been observed to increase rapidly (i.e., over a period of several hours) during normal operation. Although there is no consensus regarding the characterization of the specific mechanisms causing such "iodine spikes" in the RCS, observations of numerous occurrences in operating plants show that spikes can be correlated with changes in the condition of the fuel, such as thermal transients caused by rapid power changes (e.g., reactor scram), and depressurizations of the RCS.[4-1]

Therefore, it is not unreasonable to postulate that the reactor scram and subsequent depressurization of the RCS during a loss of coolant accident, or similar conditions in the RCS (e.g., steam generator tube rupture) could cause an iodine spike to occur. In addition to such "consequential" spikes, regulatory guidance requires the assumption that an accident occurs at the point in time that a previously initiated, unrelated iodine spike has reached a peak, i.e., a pre-existing spike.

Recent Operating Experience

A necessary condition for an iodine spike is the existence of one or more fuel rods with cladding defects or failures. The changes in power and RCS pressure cause an incremental release of activity from such rods. For a plant operating with significant fuel defects or failures, there is a potential for spikes when the RCS conditions exist for a release of iodine from the defected fuel rods. Conversely for a plant with essentially all fuel cladding intact, there is no potential for iodine spiking to occur, regardless of changes in reactor operating conditions. (Note that fuel failure as a result of the postulated accident is not considered an "iodine spike." Such failures are discussed as "gap releases" in Section 4.1.3). This relationship is borne out by the plant operating experience over the last several years.[4-2, 4-3] Steady improvement in fuel performance, as evidenced by low coolant iodine concentrations, have resulted in a significant reduction in the magnitude, as well as in the number of reportable iodine spikes.

Figure 4-1, for example, shows the fuel performance of Westinghouse-designed cores, as reported in Reference 4-3, for the period 1972 to 1988. This figure shows a steady improvement during the 1980's by the absence of entries in the higher coolant activity categories for the more recent years. For the years 1987-1988 less than 17% of these plants operated with coolant activity (I-131) above 0.01 μCi/g, and only a single plant exceeded 0.03 μCi/g.

Figure 4-2 shows historical trends in the number of assemblies with fuel defects normalized to installed capacity as reported in Reference 4-4. The BWR data shows a steady improvement to a range comparable to the experience with PWRs. The PWR data shows no trend in normalized number of defects implying that other factors such as defect size, defective assembly residence time, or cleanup system operation account for reduced equilibrium activity. EPRI and the industry are pursuing programs to achieve improved fuel performance in the 1990's (see Reference 4-4). INPO has also focussed attention on coolant activity (PWRs) and the related off-gas release activity (BWRs) by adopting these measures as indicators of both fuel and plant operating performance.

Table 4-1, also from Reference 4-3, shows the number of iodine spikes reported in the years 1980-1989. Note that while insignificant spikes occur frequently, the threshold for a reportable spike is the equilibrium concentration limit in the Technical Specifications as discussed further below. The total number of spikes reported dropped from a total of 36 observations at thirteen plants in 1982 to a total of 2 in 1987, and one each in 1988 and 1989. This dramatic reduction in the number of reportable iodine spikes is the result of improved fuel performance, (i.e., fewer number of fuel defects in operating plants) and of greater stability of operation (i.e., fewer large spike-inducing transients).

Technical Specifications and Modeling of Iodine Spikes

The course of an iodine spike, once initiated, progresses to a peak concentration followed by a more gradual return to a normal concentration. Increasing the cleanup flow rate is the only mitigating action available to

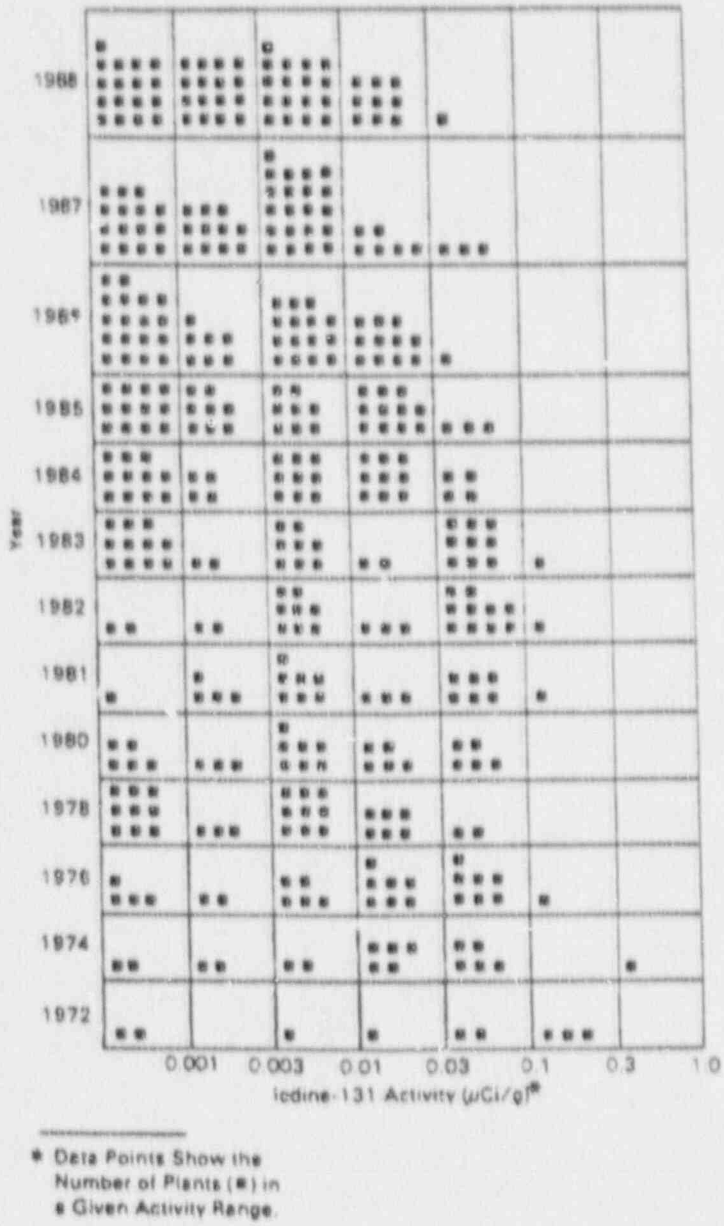


Figure 1-1. REACTOR COOLANT ACTIVITY DISTRIBUTIONS FOR WESTINGHOUSE-DESIGNED CORES (for all plant data to be on a common comparison activity basis, all plants with an increased coolant letdown rate have their coolant activity normalized to a single letdown rate; the iodine-131 uncorrected [i.e., uses normalized measured data with no adjustments for tramp uranium] values are for the end of each year [December basis]; and all data have been normalized to 100% power and the same cleanup rate).

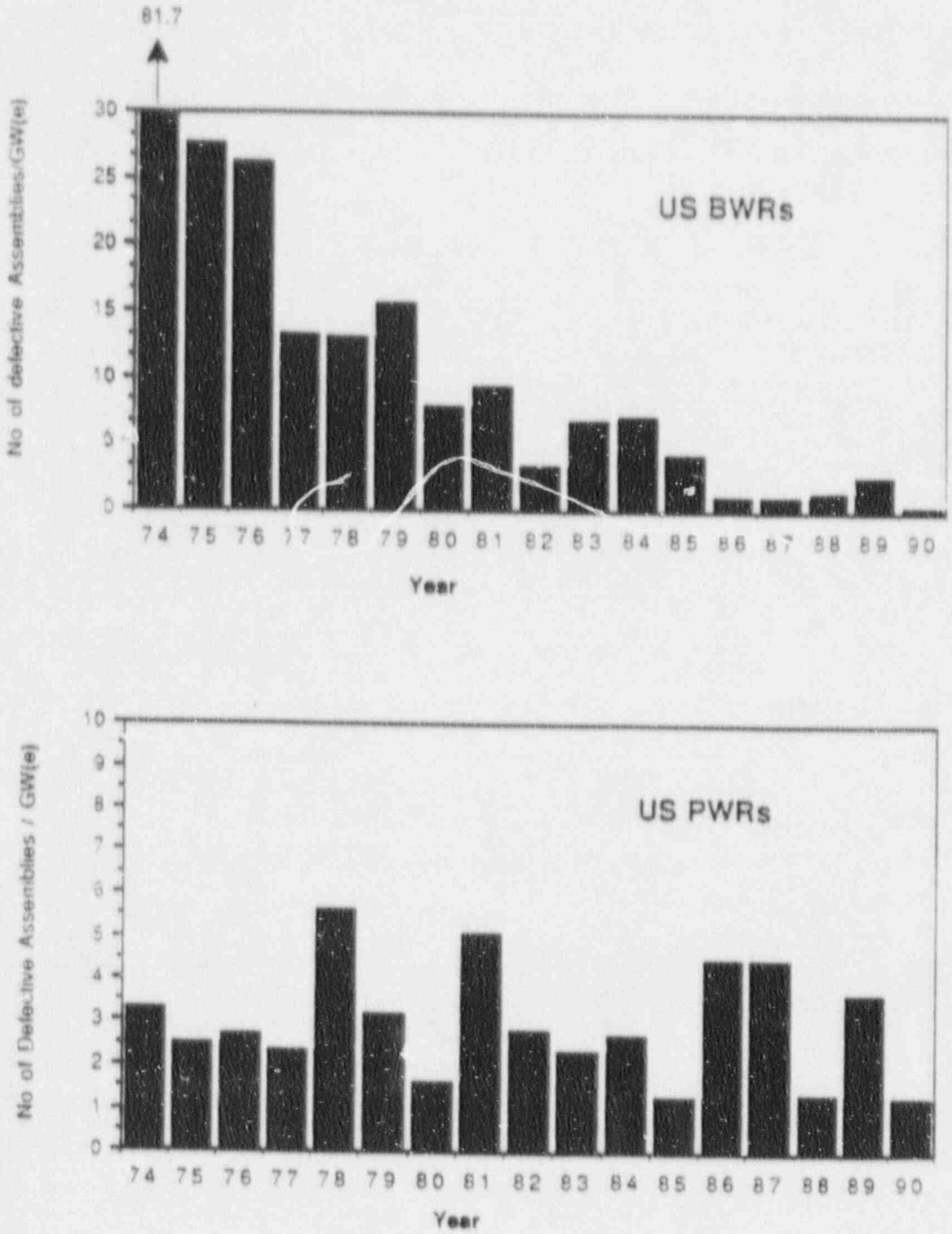


Figure 4-2. Normalized Fuel Failure Statistics (Number of Defective Assemblies per GW(e) Installed Capacity) for U.S. BWRs and PWRs.

TABLE 4-1. IODINE SPIKING OR RADIOACTIVE GAS RELEASE EVENTS*

Reactor	Reactor Type		Nuclear Steam System Supplier (NSSS)					Iodine Spiking or Radioactive Gas Release (No. of Events)									
	BWR	PWR	AC	B&W	C-E	GE	W	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989
Arkansas-1		•		•					X								
Arkansas-2		•			•			X	X								
Big Rock Point	•					•						X					
Brunswick-2	•					•		X	X	X							
Calvert Cliffs-1		•			•					X							
Calvert Cliffs-2		•			•						X(9)						
Catawba-1		•					•						X(1)				
Cook-1		•					•						X(2)				
Cook-2		•					•		X	X							
Crystal River-3		•		•				X	X					X(1)			
Davis Besse-1		•		•				X	X	X	X(5)						
Farley-1		•					•			X							
Ft. Calhoun-1		•			•							X(1)					
Ginna		•					•				X(1)						
Hatch-2	•					•				X							
La Crosse	•		•						X								
Limerick-1	•					•											X(1)
Millstone-2		•			•					X(3)	X(2)						
North Anna-1		•					•		X	X(4)	X(7)				X(1)		
Palisades		•			•				X	X(2)	X(1)						
Prairie Island-1		•					•			X							
Prairie Island-2		•					•				X(3)						
San Onofre-2														X(1)			
San Onofre-3		•			•						X(3)	X(10)	X(5)	X(1)			
St. Lucie-1		•			•			X	X	X(6)	X(2)						
Surry-1		•					•		X(7)	X(13)	X(3)	X(4)	X(3)	X(4)	X(1)	X(1)	
Surry-2		•					•			X							
Trojan		•					•		X								
Yankee Rowe		•					•		X					X(1)			
			No. of Reactors:					5	13	13	10	4	4	5	2	1	1

* Events of reportable magnitude, i.e. larger spiking events that caused coolant concentrations above the Technical Specification for equilibrium concentration.

the operators, but an increased cleanup flow rate can not prevent a peak concentration above the equilibrium limit for a larger spike. This fact is reflected in the two-tier structure of the Standard Technical Specifications (STS).[4-5] While the upper, or spiking, limit is set such that off-site doses due to coolant release accidents do not exceed established limits, the lower, or equilibrium level of the technical specifications is set such that a spike, once initiated, can exceed the equilibrium level and still remain within the spiking limit.

The STS establish an equilibrium concentration limit of 1 μCi/g (based on dose equivalent iodine-131) and a spiking limit of 60 μCi/g (spike duration less than 48 hours). The specific values incorporated into the STS were based on plant operating data available in the mid-1970's, (see Reference 4-1) and, therefore, do not reflect the improvement in fuel performance in recent years discussed above. During the subsequent 15 years of operating experience, no iodine spike exceeding, or even approaching the STS spiking limit has been reported. Further, as noted above, the number of reportable spikes has declined sharply and the average coolant concentrations have declined over the same period of time, as well.

From this operating experience, it is evident that coolant iodine concentrations in the 20 to 60 μCi/g range are not realizable, given typical initial coolant concentrations below the Technical Specification equilibrium level, even with an iodine spike. Although peak spiking concentrations above 10 μCi/g have been observed in a few instances, these peaks were all reached from equilibrium concentrations above 0.3 μCi/g (see Reference 4-1). For plants operating with equilibrium coolant concentrations more than an order of magnitude lower than that, as is typical for fuel loaded in the last several years, iodine spikes of that magnitude (i.e., above 10 μCi/g) are not credible. Therefore, a value of 10 μCi/gm would represent a reasonable envelope for iodine spike concentrations in such current plants.

<u>Application to Passive ALWRs</u>

A preliminary evaluation of potential coolant activity in a passive PWR has been performed which demonstrates that passive plant design features which affect expected coolant activity ensure less coolant activity in a passive plant for the same level of fuel defects.[4-6] The evaluation began with the maximum level of fuel defects that is being specified for passive plant shielding design, 0.25% of the fuel.[b] The passive plant evaluation considered the operating power level (55% of the four loop value), the letdown system design flowrate for coolant cleanup (133% of the four loop value), and the coolant inventory (64% of the four loop value). The coolant activity level that was calculated to correspond with 0.25% fuel defects was 0.4 to 0.5 μCi/g depending on the dose conversion factors used to determine I-131 dose equivalence. This value is well below the comparable value for a four loop plant (which would be above 0.9 μCi/g dose equivalent I-131 for a 0.25% fuel defect level).

A similar evaluation of the effect of increased passive plant fuel operating margins on fuel integrity over design life is not presently available. The increased operating margins should certainly be beneficial, however, assuming comparable fuel burnups. Moreover, the effects of extended burnups are not expected to erode this margin entirely, i.e., fuel integrity experience at least as good as current plants is expected for passive plant designs, considering improvements in the fuel assembly design, such as increased fission gas plenum volume.

Given Passive ALWR fuel performance that is at least as good as current plants and lower coolant activity for the same fuel defect rate, the equilibrium coolant activity in operating passive plants would be expected to be below the level of about 0.1 μCi/g that is at the high end for current plants. Similarly, passive plants would be expected to show improvement relative to the suggested spiking envelope value of 10μCi/g based on current plant data.

---

b. Note that traditionally, the Technical Specification limit on coolant activity has been more restrictive than the shielding design basis; for example, in current Westinghouse four loop plants, the shielding design is based on 1% fuel defect level which corresponds to about 3.7μ Ci/g, well above the 1.0μ Ci/g coolant activity Technical Specification.

4.1.3 Gap Activity Phenomena and Assumptions

Gap Release Modeling

For transients which cause fuel cladding failure, volatile fission products may be released from the fuel-cladding gap and plenum to the containment atmosphere. Only that fraction of the fission products which migrates from the fuel matrix to the gap and plenum regions during normal operation would be available for immediate release in the event of clad damage. The criteria in Appendix K to 10CFR50 govern the modeling of LOCA transients to predict cladding failure, while various Standard Review Plan sections and Regulatory Guides apply to the evaluation of gap release for other accidents. For conservative, regulatory calculations, the gap inventory is generally assumed to be 10% of the total volatile fission products present in the pin.

The total inventory of volatile fission products in the pin is a function of the rate of production during power operation, radioactive decay, and other applicable processes. The inventory reaches an equilibrium level within weeks for shorter lived isotopes, while it increases with fuel burnup for others.

The release of volatile fission products from the fuel matrix into the gap, expressed as a percentage of the total available inventory, is known to depend on diffusion and grain structure. The linear heat generation rate, the local temperatures, and the life-cycle transient history are significant operating parameters that affect the volatile fission product transport phenomena within the pin. Recent evidence indicates that the percentage of the total volatile fission product inventory which is released from the fuel matrix is only weakly burnup dependent (see experience in the following section). Cesium and iodine tend to collect as deposits on the cladding inner surface and their release following cladding breach will depend on temperature or dissolution by the coolant.

Analytical models have been developed for fission gas release and the American Nuclear Society has proposed a corresponding standard (ANSI/ANS 5.4-1982)[4-7] that addresses the abundance of many (but not all) volatile radioisotopes. Vendor fuel performance models are more current, but they focus on total gas production (not specific radioisotopes) to demonstrate conservatively that regulatory limits on peak pin gas pressure are satisfied.

Recent Operating Experience

Experimental data on volatile fission product release from the fuel to the gap and plenum regions have been reviewed. Iodine and cesium fractional releases are generally expected to be similar to those of the noble gases.[4-8] Recent measurements of noble gas gap inventory and iodine and cesium deposition on the inside surface of the cladding of spent fuel rods confirm that iodine and cesium releases from the fuel are no greater than the noble gas release.[4-9]

Figure 4-3 presents fission gas release data from 17 x 17 fuel irradiated in the Surry reactor.[4-10] The data are presented as a function of burnup, and an observed threshold in linear heat generation rate of 7 KW/ft (230 W/cm) is shown to separate the higher release data points from the lower release points. The release from fuel rods operated below 7 KW/ft is less than 2%. The data includes burnups up to 45,000 MWd/tU.

Fission gas release measurements on B&W 15 x 15 fuel irradiated in the Oconee-1 reactor to burnups approaching 50,000 MWd/tU are presented in Table 4-2 from Reference 4-11. The 16-rod average values are less than 2% over the burnup range from 30,940 to 49,570 MWd/tU. Individual rod release values range up to 3.8%, which may reflect operation early in life at linear heat generation rates up to 8 KW/ft and slightly higher operating temperatures due to the increased fuel pellet diameter relative to 17 x 17 fuel.

Fission gas release measurements from rods irradiated in Calvert Cliffs 1 to burnups of up to 54,000 MWd/tU do not exceed 2% as shown in Figure 4-4 from Reference 4-12.

Fission gas release measurements from the Zorita Research and Development Program are presented in Table 4-3.[4-13] Volatile fission product releases are less than 2% for pins whose linear heat ratings are consistently below 7 KW/ft. Burnups are up to 39,400 MWd/tU.

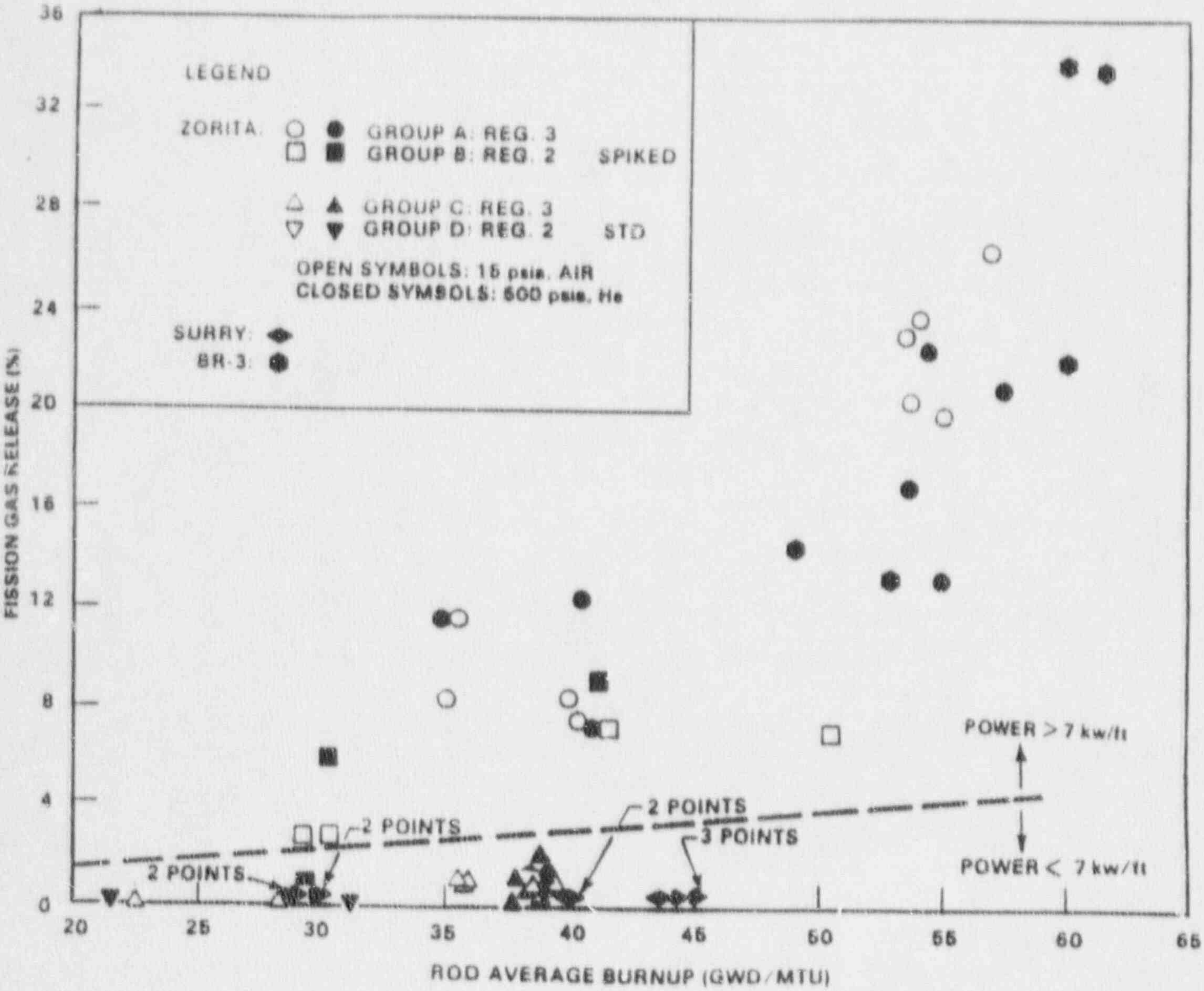


Figure 4-3. Surry Unit 2 fission gas data compared with data from other plants.

TABLE 4-2. EXTENDED BURNUP FISSION GAS RELEASE, OCONEE-1

Cycle	Rods	Release Avg. (%)	Release Range (%)	Burnup (MWd/tU)
3	6	0.6	0.1-2.4	31,940
4	16	1.5	0.5-3.4	39,180
5	16	1.6	0.7-3.8	49,570

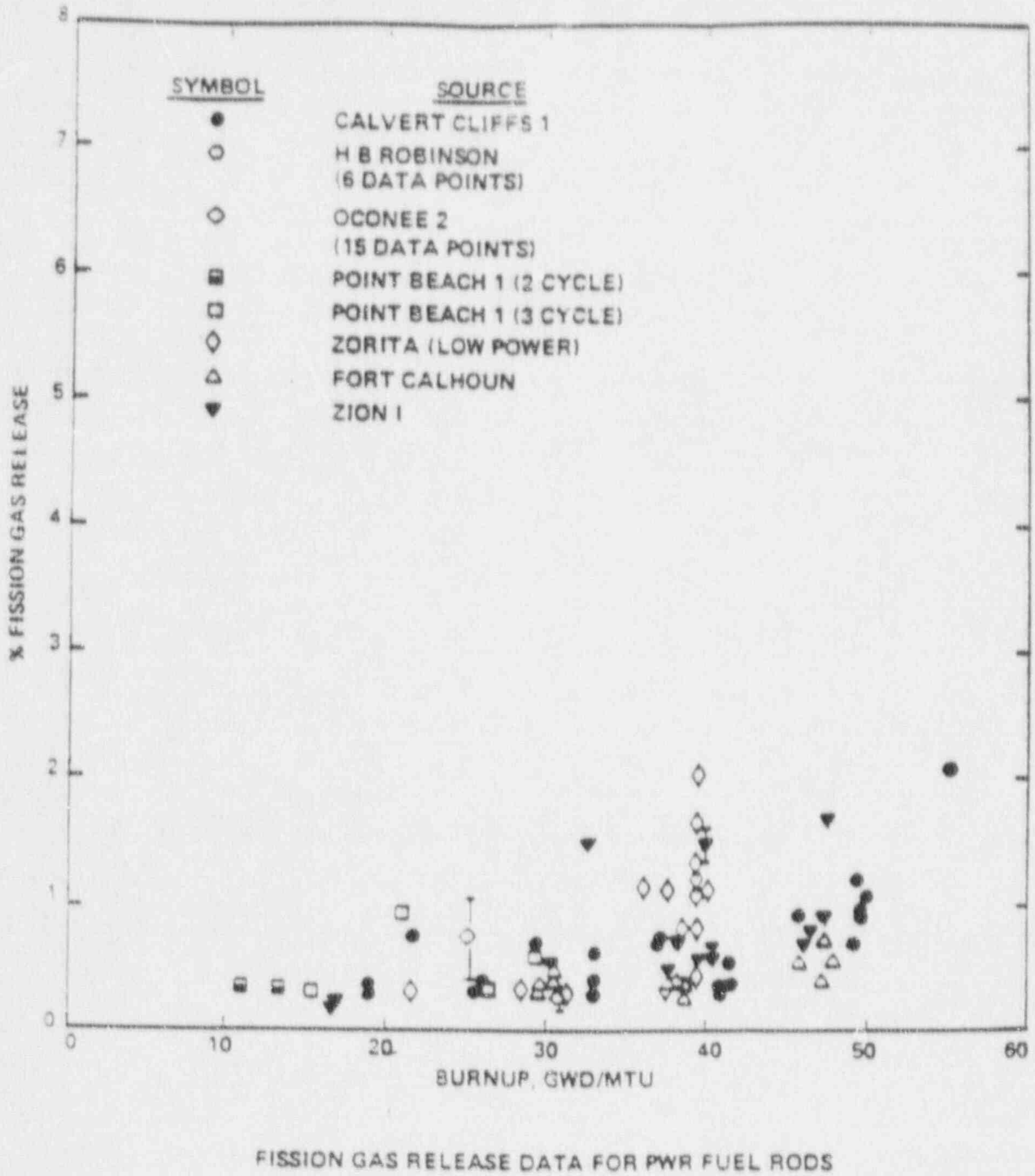


Figure 4-4. Fission gas release data from Calvert Cliffs 1 and Fort Calhoun fuel rods compared with data from other PWR fuel rods.

TABLE 4-3. FISSION GAS RELEASE DATA, ZORITA TEST RODS

Rod No.	Enrichment (w/o U-235)	Nominal Initial Pressure (psia)	Avg. Power Level (KW/ft) in Cycle[a] 1	2	3	Average End of Life Burnup MWd/tU	Percent Fission Gas Release Measured (Xe+Kr)
330	6.60	15	8.5	9.1	8.2	57,000	26.5
332	6.60	500	8.5	9.1	8.4	57,500	≥20.9[b]
334	6.00	15	7.4	9.3	8.3	53,600	23.0
344	5.81	500	8.4	8.2	7.7	53,800	16.9
379	5.81	15	8.8	8.3	7.4	55,100	19.9
383	5.81	15	8.6	8.4	7.4	54,800	20.4
384	6.00	15	7.4	9.4	8.2	54,100	23.9
385	5.81	500	8.6	8.4	7.4	55,000	17.2
386	6.00	500	7.4	9.4	8.2	54,400	22.6
326	5.81	15	6.6	9.1	--	35,100	8.3
328	5.81	500	6.5	9.0	--	34,900	11.7
329	5.81	15	8.8	8.0	--	40,300	7.6
331	5.81	500	8.8	8.0	--	40,800	7.3
333	5.81	15	8.6	8.2	--	40,000	8.5

a. At 510 MWt Reactor Power.

b. Lower limit; some gas lost during puncturing.

TABLE 4-3. FISSION GAS RELEASE DATA, ZORITA TEST RODS (Continued)

Rod No.	Enrichment (w/o U-235)	Nominal Initial Pressure (psia)	Avg. Power Level (KW/ft) in Cycle[a] 1	2	3	Average End of Life Burnup MWd/tU	Percent Fission Gas Release Measured (Xe+Kr)
335	5.81	500	8.6	8.2	--	40,500	12.4
345	6.31	15	6.5	9.8	--	35,700	11.6
313	3.6	500	5.4	6.4	6.5	39,100	1.3
314	3.6	500	5.2	6.5	6.6	38,600	0.7
316	3.6	500	5.4	6.4	6.5	39,100	0.8
317	3.6	500	5.2	6.4	6.6	39,000	1.1
318	3.6	500	5.2	6.3	6.6	38,900	2.0
363	3.6	500	5.4	6.4	6.5	39,500	0.9
364	3.6	500	5.2	6.5	6.5	38,800	0.4
365	3.6	500	5.2	6.5	6.6	39,100	1.7
368	3.6	500	5.2	6.4	6.1	37,700	0.2
369	3.6	15	5.4	6.4	6.0	35,600	1.2
370	3.6	15	4.0	6.6	7.0	35,800	1.1
371	3.6	15	4.0	6.6	7.0	35,700	1.1
387	3.6	15	5.9	6.0	6.1	39,400	1.0
388	3.6	15	5.6	6.0	6.0	38,500	0.9

a. At 510 MWt Reactor Power.

TABLE 4-3. FISSION GAS RELEASE DATA, ZORITA TEST RODS (Continued)

Rod No.	Enrichment (w/o U-235)	Nominal Initial Pressure (psia)	Avg. Power Level (KW/ft) in Cycle[a]			Average End of Life Burnup MWd/tU	Percent Fission Gas Release Measured (Xe+Kr)
			1	2	3		
307	3.60	15	5.9	6.4	--	28,400	0.2
321	3.60	15	4.0	6.6	--	22,600	0.1
230	4.32	15	9.2	7.3	5.3	50,600	7.1
293	4.08	15	8.8	7.2	5.3	49,300	3.3
294	4.08	500	8.8	7.3	5.3	49,300	2.5
281	4.53	500	8.9	7.7	--	41,100	9.1
284	4.32	15	9.1	7.6	--	41,400	7.3
280	4.32	15	9.1	--	--	30,600	2.9
282	4.32	500	9.1	--	--	30,400	5.9
283	4.08	15	9.0	--	--	29,400	2.8
285	4.08	500	9.0	--	--	29,600	1.0
292	2.40	500	7.0	--	--	21,600	0.15
266	2.9	500	6.8	6.2	--	31,200	0.2

a. At 510 MWt Reactor Power.

The NRC sponsored an assessment of the effects of using extended burnup fuel in current light water power reactors.[4-14] The effects of increasing peak burnup from 33,000 MWd/tU to 60,000 MWd/tU on expected fission gas release were specifically addressed, including ANS 5.4 calculations. The assessment concluded that 95-99% of extended burnup fuel would yield gas release in the range 1.5-2.5%. Peak pin gas releases were bounded at 10-12% by calculations assuming that overall gas generation resulted in end-of-life pin pressures at the allowable limits. The fuel design calculations which will determine actual peak pressures include regulatory conservations that are not appropriate for the physically-based source term. Thus, best estimate values would be lower and the core average gas release value (appropriate for a core damage event) would be expected to be below 2.5%.

Application to Passive ALWRs

The core designs for Passive ALWRs afford substantial margins in fuel performance when compared with current plants. The passive AP600, for example, is designed for an average fuel rod linear heat rating near 4 KW/ft using 17 x 17 fuel assemblies. This value is about 25% below the value for a current four loop plant (5.4 KW/ft). Thus, this passive PWR reactor would trip on overpower prior to reaching the operating condition of current plants. With a radial peaking factor no greater than 1.65, even the peak pin in the passive PWR will operate below 7 KW/ft.

Based on these fuel design parameters, the calculations and experience from current reactors are conservative if applied directly to the ALWR. ALWR specific calculations are not presently available but will be performed when fuel design parameters and fuel cycles are finalized. The 2.5% gas release value for most of a core load in Reference 4-14 is judged to envelope the core average release of radioisotopes to the gap and plenum regions of Passive APWR fuel pins for the physically-based source term.

4.1.4 Conclusions

Coolant Activity

Given the operating performance expected for the Passive ALWR, a reduction in the existing STS limits is warranted for these plants. Such a reduction reflects the expectation of lower dose rates due to coolant activity during normal operation and accident conditions. A reduced equilibrium concentration limit of 0.3 µCi/g, and a lower spiking limit of 20 µCi/g (both limits apply to dose equivalent I-131) are judged to provide adequate margin for Passive APWR designs. The proposal margins above the enveloping values derived in Section 4.1.2 are judged sufficient to preclude Technical Specification limitations on intended plant operation and to permit use of these values in conservative, licensing design basis analysis. While in the past the spiking limit was applied only to the pre-existing spike, it is appropriate to use the same limit for an accident initiated spike.

Such reduced limits reflect the observed improvements in current plant fuel performance and the expectation of even lower coolant activity levels in an ALWR. These reduced limits might also make predicted doses from accidents, such as steam line break and steam generator tube rupture, more realistic while retaining appropriate conservatism for Passive ALWR designs. The proposed changes would probably not have a significant impact on the design features affecting the timing of containment isolation. While the closure time for some isolation valves may be slowed to the range 30-60 seconds to improve their reliability, the coolant dose during this period would be a small fraction of applicable dose limits and thus the doses would not control the valve designs with either the existing or the proposed, reduced coolant activity limits.

The potential for coolant activity at the proposed Technical Specification level was assumed in developing the Passive ALWR design basis accident source term, as defined in this report. While the passive BWR was not specifically evaluated, based on the noted reduction in current BWR fuel defect rates and the industry-wide emphasis on fuel performance, comparable reductions in operating off-gas release and coolant activity would be expected.

Gap Activity

Based on recent fuel performance experience and the conservative fuel design parameters for Passive ALWRs, a volatile fission gas release fraction of 3% of radioisotopes is judged to provide adequate margin for use in the design basis source term for Passive APWRs. The proposal margin above the 2.5% enveloping value derived in Section 4.1.3 is judged appropriate for this purpose.

The designs of Passive ALWRs are required to preclude fuel damage (and hence gap release) for coolant breaks up to 6 inches in diameter. A medium LOCA was selected as the maximum credible break size for the physically-based source term. Hence, the design basis accident source term is based on gap release delayed until core uncovery. The gap release is treated together with the fuel release beginning one hour into the event. Given this approach, th design basis accident source term for passive BWRs is also appropriate even though differences in BWR fuel design may affect the expected gap release magnitude.

Other Design Basis Accident Applications

The reduced coolant activity for passive ALWRs should be applicable to the safety evaluation of specific design basis accidents such as steam generator tube rupture (PWR) and main steamline break (BWR). The peak gap activity and the chemical form of releases should be evaluated, given the large margins in Passive ALWR fuel operating conditions, to provide an appropriate degree of conservatism for other accidents such as fuel pool accidents.

4.2 Early In-Vessel Release Magnitude

4.2.1 Introduction

In this section, fission product releases from the fuel to the RCS are estimated and justifications for these releases are provided. The assumptions made on the extent of core melt progression for the representative core damage events described in Section 2.0 and Appendix 1 are identical for the passive plants of PWR and BWR design and the timing is similar. As a result, the early in-vessel release magnitude will be the same for both plants.

4.2.2 Important Phenomena and Assumptions

The core damage event defined for purposes of estimating the source term assumes that core melt progression (75% core melt) leading to reactor vessel lower head penetration takes place over a three-hour period (BWR) or five-hour period (PWR), beginning one hour after the initiating event. The rate at which fission products will be released from the fuel to the RCS during this period will tend to vary depending on the details of the core melt progression phenomena taking place, such as core heatup rate, location and extent of metallic melt relocation from molten control materials, extent of zircaloy oxidation upon cladding melting, rate and extent of candling and accumulation of liquefied fuel, extent of molten pool formation, ceramic crust thickness surrounding the molten pool, timing and location of crust thinning and failure, and duration of core melt relocation to the lower plenum and interaction with water. However, as a first approximation, the rate of fission product release is assumed to be constant over the period of core melt progression (3 hours BWR, or 5 hours PWR).

4.2.3 Results

Releases from fuel are proposed, and the technical bases are discussed, according to volatility groupings of fission products.

Noble Gases, Iodine and Cesium:

Analysis of fission product releases from the TMI-2 accident[4-15 thru 4-18] and from severe fuel damage experiments[4-19 thru 4-26] indicate that the releases of noble gases, iodine, and cesium are approximately equal and are closely related to the fraction of the fuel that becomes molten in the accident sequence. In the TMI-2 accident, about 45% of the core was molten and the releases of noble gases, iodine, and cesium were in the neighborhood of 55%.

Measurements of residual fission products in previously molten fuel indicate that up to ~10% of the original cesium inventory and somewhat less of iodine can be retained by the formation of chemical species that are stable at high temperatures and/or geometries having low surface-to-volume ratios (see References 4-16 and 4-27). On the basis of these results, releases of 90% of iodine and cesium from molten fuel are proposed. No residual fission gases were found in molten fuel debris from TMI-2 (see Reference 4-15), so 100% release of noble gas from molten fuel is proposed.

The early release of fission products from the 25% of the fuel which does not melt early should also be considered. The release of noble gases, iodine, and cesium increases with the extent of oxidation by steam of the unmelted UO_2 fuel during the heatup in an accident. In addition, fission product release may occur as a result of fuel pellet cracking during reflood. A release of 25-30% of noble gases, iodine, and cesium from unmelted fuel in a terminated accident appears to be a reasonable bound based on data from TMI-2 and the severe fuel damage tests conducted at the Power Burst Facility at Idaho National Engineering Laboratory (INEL).

Fission product releases from fuel in the TMI-2 accident and in the severe fuel damage tests are presented in Tables 4-4 and 4-5. These data support a release of about 80% for noble gases (100% from melted fuel and 25-30% from unmelted fuel) and 75% for iodine and cesium (90% from melted fuel and 25-30% from unmelted fuel) given an accident with about 75% fuel melting.

Tellurium

Considerable study has resulted in the understanding that tellurium is released from the fuel at about the same rate as noble gases, iodine, and cesium, but is largely retained by the surrounding metallic zircaloy cladding and is then released during oxidation of the cladding.[4-28,4-29] Tellurium has a chemical affinity for metallic zircaloy and most other metals.

Oxidation of the cladding has the effect of increasing the concentration (and therefore the chemical activity) of tellurium in the remaining metallic zircaloy, thereby increasing the partial pressure of tellurium. When the

TABLE 4-4. RELEASES FROM THE CORE IN THE TMI-2 ACCIDENT

Isotope	Fraction of Core Inventory Released
^{85}Kr	0.54
^{129}I	0.55
^{137}Cs	0.55
^{132}Te	0.06
^{90}Sr	0.001*
^{106}Ru	0.005
^{125}Sb	0.016
^{144}Ce	0.0001

* Leaching from damaged core after reflood increased Sr release to 0.0[illegible] two months after accident.

TABLE 4-5. FUEL RELEASE FRACTIONS FROM PBF SEVERE FUEL DAMAGE TESTS

Element/ Exp. Cond.	SFD-ST	SFD1-1	SFD1-3	SFD1-4
Kr, Xe	0.50	0.026-0.093	0.08-0.19	0.23-0.44
I	0.51	0.12	0.18	0.26
Cs	0.32	0.09	0.18	0.44-0.56
Te	0.40	0.01	0.01-0.09	0.03
Ba	0.011	0.006	0.004	0.008
Sr	0.00002		0.00024	0.0088
Sb			0.00019	0.0013
Ru	0.0003	0.0002	0.00003	0.00007
Ce	0.000002	0.00009	0.00008	0.00013
Actinides			<0.0001	<0.00001
% Zr Oxidized	75	26	22	32
% Fuel Melted	15	16	18	18

local oxidation of zircaloy is equivalent to less than about 90% active clad conversion to ZrO_2, the release rate of tellurium has been found to be 1/40 that of iodine and cesium, but equivalent to that of iodine and cesium when zircaloy oxidation exceeds 90%.

A value of 0.2 for in-vessel tellurium release from the fuel is suggested for use in the physically-based source term and will provide margin to the actual release expected based on the Table 4-4 and 4-5 data. This is based on the fact that a realistic evaluation of in-vessel clad oxidation for a core damage event is in the neighborhood of 30 to 60%. For example, at TMI-2 where clad oxidation was ~50%, the tellurium release was ~0.06.

Semi-Volatiles and Low Volatiles

The releases of strontium, barium, antimony, and ruthenium have been found to be quite low as demonstrated in Tables 4-4 and 4-5 and are bounded by a value of 1%. Barium and strontium exist as oxides within the UO_2 under accident conditions and have low volatilities (see Reference 4-25). Antimony and ruthenium are present as metals which are insoluble in the oxide fuel matrix and tend to separate from the fuel, concentrating with molten metallic debris (see Reference 4-27).

Cerium, lanthanum, and the actinides (uranium, plutonium, americium, curium) are oxides with very low volatilities which are dissolved in the fuel matrix and thus are released to a very small extent (<0.01%). (See Reference 4-25.)

Suggested Release Magnitudes

The proposed releases from fuel are listed in Table 4-6 along with the late in-vessel releases which are discussed in Section 4.3.

4.2.4 Conclusions

The proposed releases are a result of the assumptions of 75% core melt and 30-60% cladding oxidation and are based on experience gained in the analysis of core melt progression experiments and the TMI-2 accident. The

TABLE 4-6. ALWR IN-VESSEL FISSION PRODUCT RELEASE ESTIMATES [3]

Element	ALWR Early Release [1]	ALWR Late Release [2]
Noble Gas	0.80	0.20
I	0.75	0.15
Cs	0.75	0.15
Te	0.20	0.03
Sr	0.01	
Ba	0.01	
Ru	0.01	
La	0.0001	
Ce	0.0001	
Other	0.0001	

(1) Constant release rate from 1 hour to 3 (BWR) or 5 (PWR) hours after accident.

(2) Reflood of RPV to 1 meter above TAF reduces the in-vessel release to zero except for noble gases which are released in proportion to fuel cladding failure and gap activity.

(3) Ex-vessel releases need to be added to these to determine total release. All numbers are fractions of original core fission product inventory.

larger coolant volume in the RCS per unit power in the Passive ALWR relative to that of the Evolutionary ALWR and current light water reactors (LWRs) protracts coolant boiloff, and, therefore, extends the period of time over which in-vessel core melt progression takes place. Otherwise, the Passive ALWR cores should behave similarly to those of the Evolutionary ALWR and current LWRs, including in-vessel fission product releases from the fuel.

4.3 Late In-Vessel Release Magnitude

4.3.1 Introduction

In this section fission product releases from the fuel remaining in the core following vessel lower head penetration and the relocation of molten core debris into the cavity are discussed.

4.3.2 Important Phenomena and Assumptions

It is assumed that the design requirement to flood the reactor cavity by the time of reactor vessel meltthrough has been met and that flooding occurs to a height covering the opening in the vessel lower head. This means that an opening which can draw air into the reactor vessel does not exist and a steam environment remains within the reactor vessel. However, it is not assumed that the depth of the reflood will be sufficient to cool the fuel remaining in the core. Fuel remaining in the reactor vessel, primarily at the core periphery and near the bottom of the core, is heated by decay heat and loses heat to the reactor vessel walls and out the top and bottom of the core. It is not certain, without a detailed analysis, what the temperatures might be in this material. However, examination of similar material remaining in the TMI-2 core (see References 4-15 thru 4-18) revealed that in much of this material not only were cladding melting temperatures not reached, cladding oxidation was minimal, and cladding ballooning did not occur. Cladding temperatures were less than the transition from alpha zircaloy to two-phase alpha plus beta zircaloy (1105 K), and fission product releases were small to none. It is expected that the low power density in the Passive ALWR cores would tend to reduce temperatures in the fuel rods remaining at the periphery of the core (relative to TMI-2) following vessel meltthrough and melt relocation to the cavity. This supports the assumption that a large fraction of the fuel (~25%)

may remain relatively intact within the reactor vessel following early melt relocation and vessel penetration.

The TMI event was terminated by reflooding the vessel. In-vessel recovery or flooding of the containment to the top of the remaining fuel is also a likely possibility for the ALWR. However, as this is not an explicit utility requirement in the ALWR Requirements Document for severe accident conditions, scenarios can be postulated where the fuel remaining in vessel is only partially reflooded. For this reason, the fuel remaining in vessel is assumed to melt and relocate to the vessel lower head and reactor cavity/lower drywell over the remainder of the first 24 hours of the accident.

In the evaluation of early in-vessel releases from fuel it was assumed that volatile and noble gas fission product releases from unmelted fuel were 25-30%. Noble gases and volatile fission products in the fuel remaining in vessel are assumed to be released over the remainder of the 24 hour period in the same proportion as the release fractions assumed for the early part of the accident. This results in all of the noble gases and 90% of the Cs and I being released. Given relatively low temperatures, oxidation of the fuel cladding should be minimal and tellurium releases during this phase should be a small fraction of the volatile releases. A tellurium release of about 10% of that in the fuel remaining in the core is assumed. Releases of the less volatile fission products from the fuel remaining in the core should be negligibly small during this phase of the accident sequence.

4.3.3 Results

The assumptions and phenomena discussed above lead to a recommendation of a late in-vessel release (based on initial core inventory) of 20% each for noble gases, 15% each for iodine and cesium and 3% for tellurium in the period 3 to 24 hours (BWR) or 5 to 24 hours (PWR). These results are shown in Table 4-6 along with the recommendations for early in-vessel releases.

4.3.4 Conclusions

The recommended values for the late in-vessel volatile fission product release are bounding and may actually be somewhat lower than the values (25-

30%) assumed for the early release from the unmelted fuel because of the expectation, based on measurements of materials removed from the periphery of the TMI-2 core and the lower power density of Passive ALWR cores, that the fuel material remaining at the periphery of the core will be relatively cool.

4.4 RCS Retention

4.4.1 Introduction

Fission products released from the fuel during core damage events will be affected by physical and chemical processes during transport through the RCS to the break location. Depending upon the break location and the thermal-hydraulic conditions in the transport path, substantial quantities of fission products may be deposited in the RCS correspondingly reducing the source term to containment.

The NRC and the commercial nuclear industry have developed computer codes (e.g., TRAP-MELT[4-30] and MAAP[4-31]) which predict the extent of deposition in the RCS for various accident sequences and have undertaken experimental programs for the purpose of validating these calculational methods. Detailed analyses using these best-estimate computer models, supported by experimental evidence from in-pile and out-of-pile tests, indicate that iodine, cesium and less volatile radionuclides will condense on or interact with other structural materials released from the damaged core to generate aerosols.

Although some retention of fission product vapors would occur as the result of condensation on and chemical reaction with surfaces, the transport and deposition behavior of these fission-product bearing aerosols will control the quantities of radionuclides released from the RCS to containment. Important aerosol processes such as impaction, gravitational settling, thermophoresis, and diffusiophoresis will be very effective in forcing aerosol deposition under many accident conditions. In certain scenarios scrubbing of aerosols in water reservoirs and liquid streams will further reduce the transported materials.

4.4.2 Important Phenomena and Assumptions

The parameters known to be most important for effective aerosol deposition are residence time in the RCS, negative temperature gradients (cold spots), the presence of large quantities of condensing steam, and impaction losses generated by turbulent deposition at pipe bends. Thus, accident scenarios that lack these characteristics will minimize deposition and allow higher releases to containment. Only large pipe break scenarios that have a short pathway to the break will have short residence times and little or no steam condensation or impaction in the transport path. All other scenarios are expected to include strong deposition forces that limit aerosol transport.

Large break accident scenarios that lead to extensive core damage are ... unlikely ($<10^{-7}$/yr). Therefore, as described in Appendix 1, it is assumed that the RCS thermal-hydraulic conditions are those of the representative core damage events, i.e., an intermediate size LOCA. These events will be characterized by longer transport paths and greater impaction and turbulent deposition losses, as well as cold spots in the transport path to force steam condensation and diffusiophoretic and thermophoretic deposition. Estimates of the magnitude of these deposition effects have been made using best-estimate computer models supported by extensive in-pile and out-of-pile test data. Estimates have also been made based on the engineering judgement of experts familiar with the limitations of certain computer models. The following sections present these estimates and the technical bases used to develop them.

4.4.2.1 Experimental Evidence on RCS Retention. Experimental evidence of aerosol retention processes in the RCS is provided by the LACE[4-32,4-33] and Marviken[4-34] aerosol transport tests as well as by the SFD 1-4 test (see Reference 4-23) and the ... FP-2 test.[4-35] Table 4-7 summarizes the measured deposition results. Aerosol retention in the piping system of about 80% was measured in LACE tests LA3A and LA3C which had soluble/nonsoluble aerosol ratios on the order of that expected from core damage accidents. Tests LA3B had a lower retention, probably due to a very low soluble/nonsoluble aerosol ratio. The Marviken tests used prototypic core materials and found ~74% retention in the simulated RCS. These large retention fractions are representative of that expected when a piping system is included in the

TABLE 4-7. SUMMARY OF EXPERIMENT RETENTION FRACTIONS (% OF SOURCE)

Test	Species	DEPOSITION Close to Fuel Source	 Total Piping
LACE LA3A	CsOH/MnO=.21	26	77
LA3B	CsOH/MnO=.13	15	51
LA3C	CsOH/MnO=.61	46	83
LA1	CsOH/MnO=.43	--	99
Marviken	--	--	~74
SFD 1-4	Iodine	10	95
	Cesium	30	95
LOFT FP-2	Iodine	66	70
	Cesium	60	71

transport path, and deposition at bends due to particle impaction is a dominant removal mechanism. Retention fractions of the order of 25 to 50 percent were noted for the first few meters of piping in the LACE LA3A and 3C tests.

The SFD 1-4 test measured fission product deposition on surfaces downstream of the damaged fuel region. Large factions of iodine and cesium(up to 30%) were found to deposit close to the fuel, although small amounts of material (fine aerosols) were able to migrate long distances (~20m) before being deposited. Total system retention was 95%.

The LOFT FP-2 test simulated a LOCA without emergency coolant makeup in which fission products were transported from the RCS through a long low pressure injection system (LPIS) line. During the pre-reflood phase of the test 2-3% of the volatile fission products were released from the fuel. Approximately 2/3 of the released iodine and 1/2 of the cesium were deposited in the reactor vessel and hot leg pipe, and nearly 75% of this material was retained in combined RCS piping and the LPIS line. Because these experiments were performed with real fuel and control rod materials within a prototypic geometry, the fission product deposition behavior is expected to be representative of RCS deposition behavior in an actual plant.

Additional evidence of fission product retention during severe accidents is provided by the TMI-2 accident evaluation. Water pathways that existed throughout the duration of the accident retained nearly 100% of iodine, cesium and other aerosols generated during the accident. For accident scenarios in which a water pool is the pathway to containment (e.g., the IRWST in the AP600) little release of fission products to the containment other than noble gases would occur.

4.4.2.2 Analytical Results on RCS Retention. The experimental evidence is quite supportive of the argument that large fractions of iodine, cesium and less volatile radionuclides will deposit on system surfaces during transport through the RCS. However, the amount of RCS retention is dependent on the design details of the transport pathway and the thermal-hydraulics of the accident sequences. In support of NUREG-1150,[4-36] the NRC's TRAP-MELT code (one of the modules of the Source Term Code Package (STCP)) was used to estimate the amount of RCS retention that can be expected for a variety of accident sequences in modern, operating PWRs and BWRs.[4-37] The predicted retention factors for aerosols in the RCS range from approximately 15 percent to 85 percent. The lowest values are associated with large, hot-leg pipe break accidents in PWRs and low-to-intermediate pressure sequences in BWRs in which core uncovery occurs early (about one hour after shutdown). Three considerations must be factored into the evaluation of these computer code results relative to severe accidents in Passive ALWRs: low probability of large primary pipe breaks, limitations in the computer codes used to calculate these retention factors, and differences in Passive ALWR designs vs. the operating plants evaluated in NUREG-1150. These considerations, as discussed below, suggest that the low values of RCS retention are not applicable to the Passive ALWR.

Extensive experimentation and PRA analysis have shown that large RCS pipe break-initiated core damage sequences are very low in probability ($\leq 10^{-7}$ per year). Such sequences are reduced even further in likelihood by application of Leak Before Break technology. Extensive investigations of the fracture mechanics of piping provide confidence that a leak in primary system piping would precede a rupture, thus allowing the plant to be shut down and the RCS depressurized before a large break would develop. The NRC has recently issued an amendment to General Design Criterion 3 which acknowledges the need to address application of Leak Before Break to requirements other than dynamic effects of pipe rupture. This further reduction in likelihood of an already very low probability core damage sequence suggests that very large pipe breaks located close to the reactor vessel need not be part of the basis for determining RCS retention for Passive ALWR source term estimates.

The version of TRAP-MELT used in the Source Term Code Package is recognized to underpredict aerosol retention within the RCS because of unmodeled phenomena. In particular, this version does not model the effect of bends on particle deposition, a process that has been shown to be important in experiments. Figure 4-5 illustrates a post-test comparison of deposition measured in LACE Test LA3B versus predictions with versions of TRAP-MELT that do not contain models for predicting deposition in bends. Figure 4-6 shows the same test results compared with calculations of codes which do model bend deposition. The rapidly rising sections of the experimental curve represent

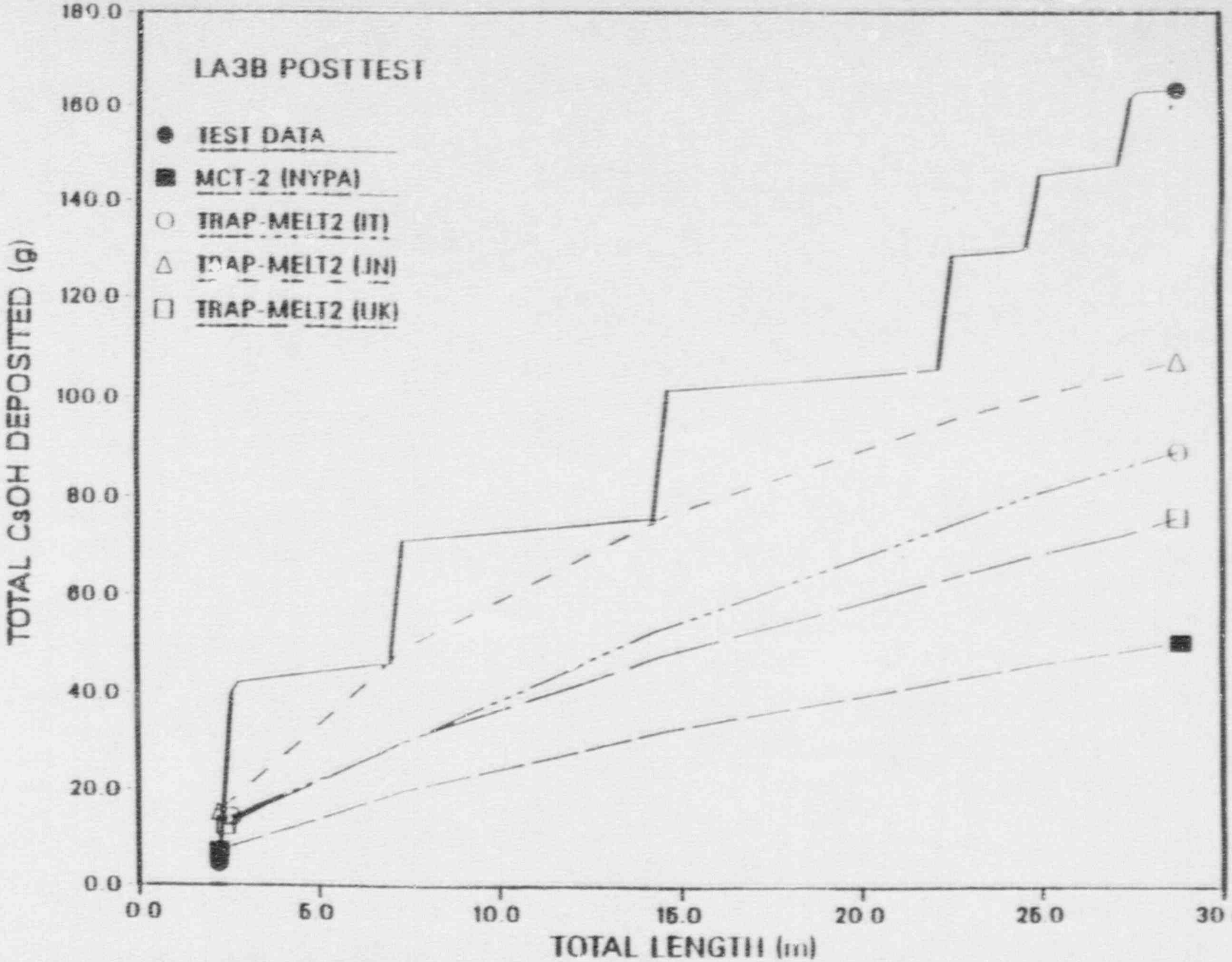


Figure 4-5. Post-Test Comparison of LA3B Deposition vs. Trap-Melt Predictions Without Effect of Bends.

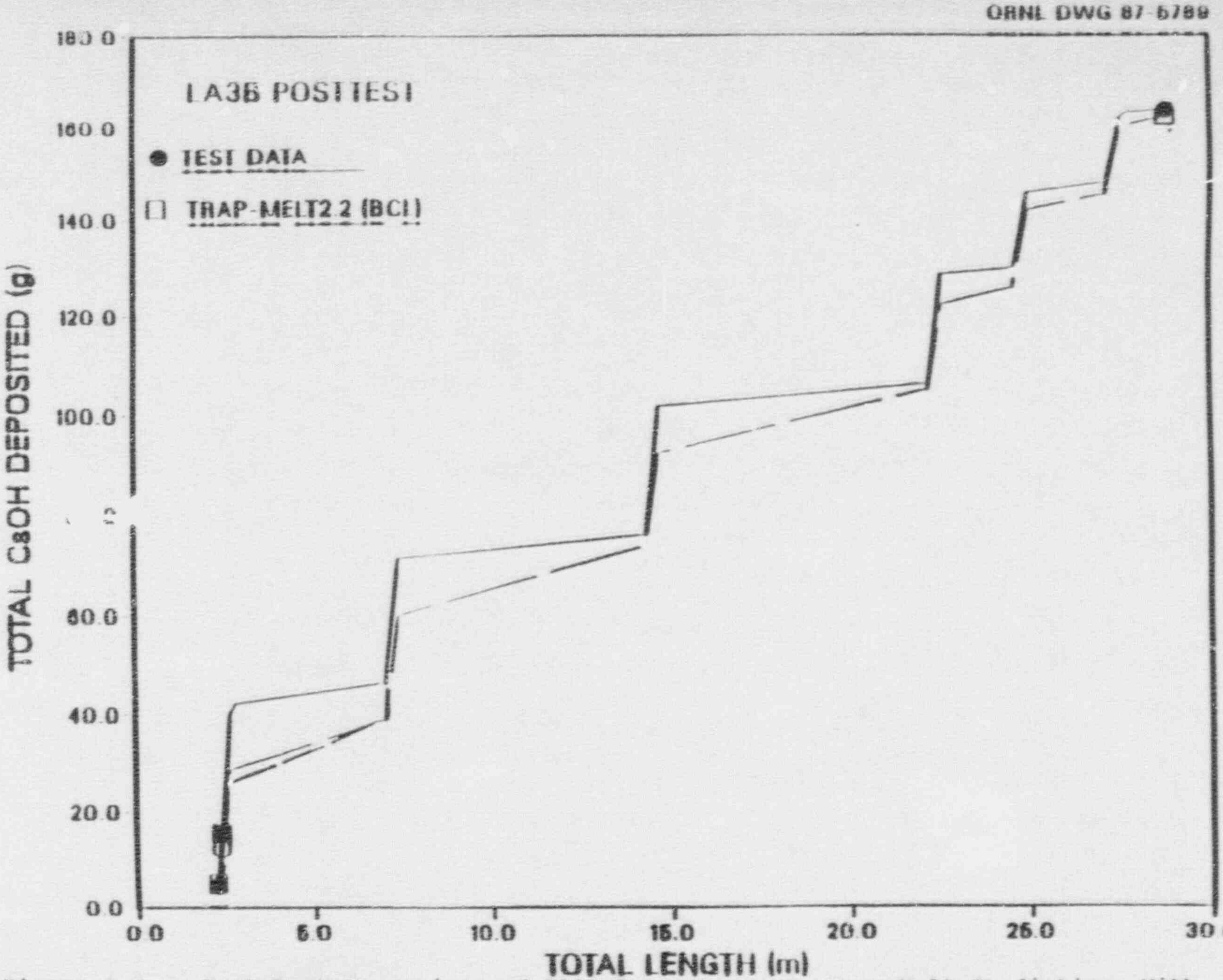


Figure 4-6. Post-Test Comparison of LA3B Deposition vs. Trap-Melt Predictions With Effect of Bends.

5-G03200-004
020791R91A

regions of high deposition at bends. The TRAP-MELT 2.2 code, which was quite successful in predicting deposition in the LACE LA3 test series, is an advanced version of the code which incorporates a turbulent deposition model for treating aerosol deposition at bends. On the basis of the Figure 4-5, Figure 4-6 comparison, it is evident that RCS retention estimated by codes like TRAP-MELT 2.C (used in support of NUREG-1150) will be underpredicted. Thus, the retention fractions used for Passive ALWRs should be greater than the values obtained with the STCP in support of NUREG-1150 for current plants.

An uncertainty analysis was performed as part of NUREG-1150 in which ranges were determined for uncertain parameters such as the RCS retention fraction by polling source term experts. In their evaluations the experts recognized the limitations of existing RCS deposition codes. Table 4-8 shows the median values obtained by evaluating the expert responses for different types of accident scenarios. The lowest values are again associated with low pressure accidents and accidents involving early core melt and revaporization, but these values are higher than the TRAP-MELT predictions and thus appear to have been corrected for the underprediction.

A final point regarding the applicability of the TRAP-MELT results and NUREG-1150 estimates is that both the PWR and BWR passive plant designs have automatic depressurization systems (ADS) which would be used to depressurize the RCS in the event of a core damage accident. The ADS for both the BWR and PWR include a path through SRVs to a water pool (suppression pool and IRWST, respectively), as well as a path directly to the containment atmosphere. For events requiring depressurization, most of the blowdown will occur through these large pools of water. Following depressurization, a portion of accident scenarios are postulated to occur as a result of incomplete depressurization. For these events continuous fission product releases to containment through a pool of water with substantial retention of aerosols will occur.

Other accident sequences may exist, however, in which there is complete depressurization but core damage occurs due to failure of active and passive reactor inventory makeup. As the ADS may be open to the containment atmosphere for this type of event, credit for pool scrubbing of fission products from the RCS should be conservatively ignored. Consideration of deposition and retention within the ADS blowdown path is appropriate.

TABLE 4-8. NUREG-1150 EXPERT ELICITATION MEDIAN RETENTION FACTORS

		PERCENT RETENTION		
Case	Conditions	Iodine	Cesium	Low Volatility Aerosols
PWR	Setpoint Pressure	91	96	97
PWR 2/3	High and Intermediate Pressure	59	71	76
PWR 4	Low Pressure	48	60	66
BWR 1	High Pressure, Early Melt	91	97	97
BWR 2	Low Pressure, Early Melt	59	70	74
BWR 3	High Pressure, Delayed Melt	72	75	92

In addition to the potential availability of pools to scrub releases as they occur from the RCS, Passive ALWRs also tend to have slightly larger RCS volume to power level ratios than the current generation of LWRs, which leads to delayed uncovery of the core and longer residence times during the period of release.

4.4.3 Results for RCS Retention

Table 4-9 summarizes the assessment of RCS retention factors to be used in the Passive ALWR source term. The basis for these retention factors is the experimental and analytical results noted above applied to the representative accidents for the passive plant.

4.4.4 Conclusions

For representative accident sequences, the RCS retention for CsI appears to be on the order of 70% for both BWRs and PWRs, based upon STCP and MAAP calculations. Experimental results from Marviken, SFD, LOFT, and LACE also support these high retention fractions. However, the RCS retention is a function of the accident sequence and the design of the RCS (number of pipe bends and lengths of pipe). Therefore, to provide margin, the 70% value is reduced to 50% and 60% for iodine in advanced PWRs and BWRs respectively. The retention factor used for all other aerosols is also reduced to 60% in PWRs, but because BWRs have a larger RCS surface area (e.g., dryers) the 70% value is retained.

For the representative accident sequences, the RCS retention will be the product of these RCS retention factors and the fuel release fractions.

These RCS retention values are considered to provide margin to the best estimate retention over a range of accident sequences based on the following:

- Experimental evidence indicating 70% or higher for aerosol retention in vapor pathway piping systems where the aerosol material and the controlling thermal-hydraulic conditions are similar to that of actual reactors.

TABLE 4-9. SOURCE TERM EXPERT GROUP RECOMMENDED RCS RETENTION FACTORS FOR SEVERE ACCIDENTS

FRACTION OF IN-VESSEL FUEL RELEASE RETAINED IN RCS

AEROSOL CHEMICAL SPECIES	PWR	BWR
I	0.5	0.6
All Other	0.6	0.7

- Experimental evidence and TMI-2 evidence indicating nearly complete aerosol retention in liquid pathways.

- The extremely low likelihood of a core damage accident initiated by large, close-to-vessel pipe breaks. Extensive investigation of the fracture mechanics of piping provides confidence that leaks in primary system piping would precede ruptures and would be detectable, allowing the plant to be shut down before a break could occur. This will significantly reduce the already very low frequencies of large LOCA initiated severe accidents obtained in PRAs.

- Extrapolation of analytical results and NUREG-1150 expert judgement to account for the extremely low likelihood of large size, close-to-vessel pipe breaks and for enhanced deposition by impaction.

- ALWR design features which would tend to increase aerosol retention beyond that expected for existing LWRs, e.g., internal refueling water storage tank, larger RCS volume.

4.5 Revaporization Release

4.5.1 Introduction

During the period of in-vessel melt progression prior to reactor vessel lower head penetration, a significant fraction of the fission products released from fuel will deposit on reactor coolant system surfaces either by aerosol or vapor deposition. Subsequent heating of these surfaces can lead to the revaporization of volatile fission products from surfaces, their redistribution to other surfaces further down the flow path and, for some fraction of the originally deposited radionuclides, release to the containment atmosphere. The revaporization and transport of radionuclides that occur prior to vessel failure is accounted for in the RCS retention factor discussed earlier. This is, in part, the reason for smaller retention factors for iodine than for the bulk of other aerosols. Thus the revaporization release term described in this section is only intended to represent the release that occurs after vessel lower head penetration. The duration of this release can be over a period of hours or days.

4.5.2 Important Phenomena and Assumptions

A number of aspects of fission product behavior, plant design, and the thermal-hydraulic characteristics of an accident sequence influence the potential for revaporization. The chemical form of the volatile radionuclides and possible chemical reactions with surfaces are key aspects of revaporization. As discussed in Section 4.7, the principal form of iodine is expected to be cesium iodide, and the balance of the cesium will primarily be in the form of cesium hydroxide. Even though the majority of the deposition on surfaces is likely to be as aerosols, substantial contact may be established with the underlying surface since a large fraction of the mass of fission product constituents is liquid at RCS temperatures. Based on experimental evidence, some of the cesium hydroxide will react with the steel surface, possibly to form a silicate. This cesium will not be subject to revaporization.

The amount of the deposited fission products that will revaporize depends on the temperature of the surface and the volatility of the chemical species. Figure 4-7[4-38] illustrates the equilibrium concentration of cesium iodide and cesium hydroxide (in kg/m^3) for the vapor above a pure liquid of that species as a function of temperature. If cesium hydroxide and cesium iodide are mixed according to their inventories (approximately ten times as much cesium as iodine), Raoult's law (i.e., the partial pressure of solvent vapor in equilibrium with a dilute solution is directly proportional to the mole fraction of solvent in the solution) indicates that the vapor pressure of the cesium iodide will be substantially reduced. In order for a significant fraction of cesium and iodine (i.e., kilogram quantities) to transport within the reactor coolant system, the surface temperature must exceed approximately 1000°K.

The ambient atmosphere within the RCS can also affect fission product revaporization. In an accident that involves an open flow path (two holes in the system), air can be drawn through the vessel, potentially oxidizing the

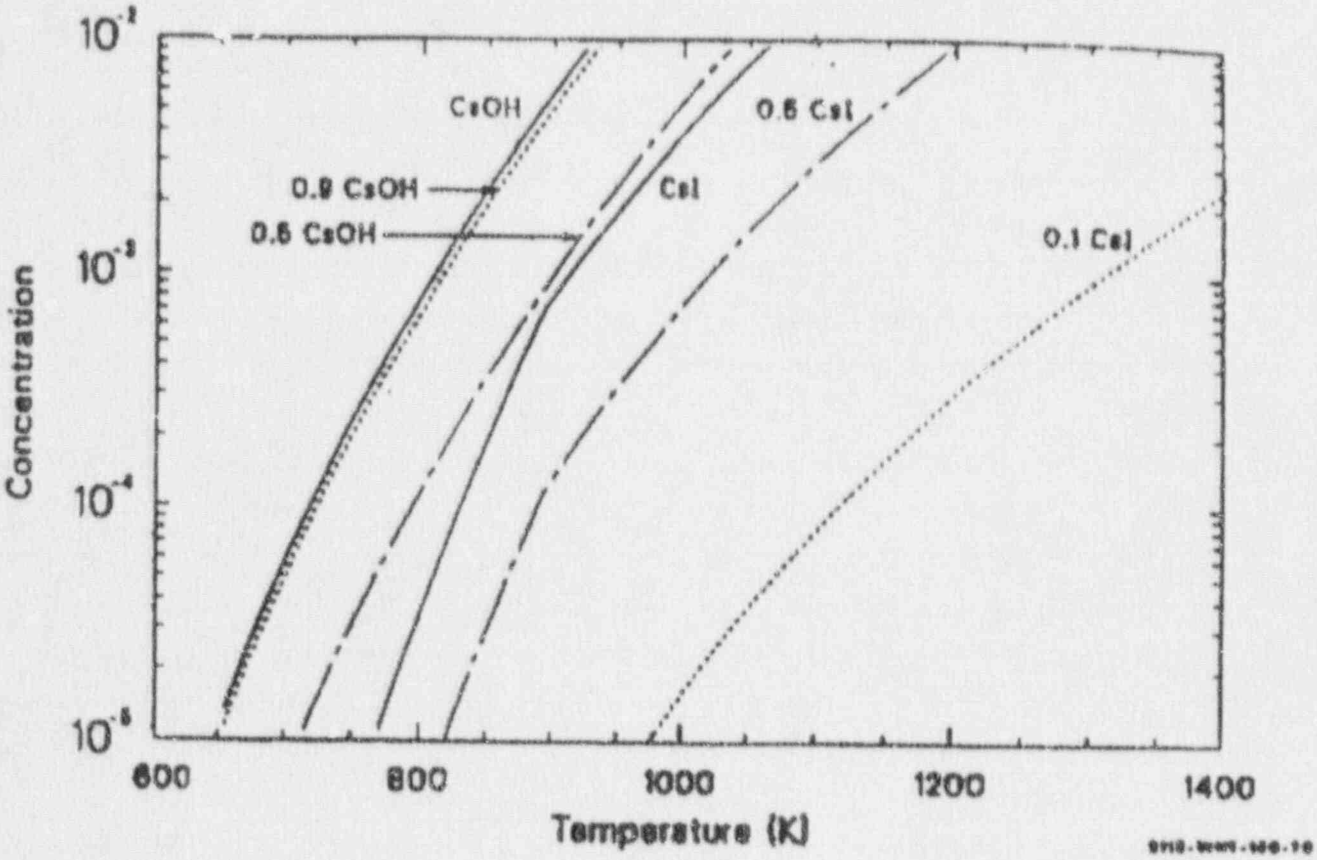


Figure 4-7. Calculated effect of mixing of cesium hydroxide and cesium iodide condensate on equilibrium vapor concentrations.

fission products if the temperature is sufficiently high and sweeping airborne vapors and aerosols into the containment.

The data base to support the prediction of revaporization is limited. The chemical forms of the key elements, iodine and cesium, are known as are the equations of state for the pure species. Data have been collected by Sallach and Elrick[4-39] on interactions of these species with steel surfaces but little information is available on subsequent revaporization.

The Source Term Code Package models revaporization prior to vessel failure but not following vessel failure. MAAP and MELCOR[4-40] have very simple revaporization models that tend to over-predict revaporization.

No specific computer analyses have been performed for the Passive ALWR to estimate the amount of revaporization that would be expected. In the Industry Degraded Core Rulemaking Program (IDCOR) analyses[4-41] of the Mark I BWR design, complete revaporization was predicted for deposited iodine and cesium over an extended time period. This was the result of very high estimated drywell temperatures in the absence of drywell sprays or other means of heat removal. In contrast, some features of the passive designs would tend to substantially limit or prevent revaporization. Because of the automatic depressurization features of the plants, the reactor would be at or near containment pressure. The reactor cavity/lower drywell would in all cases be flooded with water prior to or immediately upon reactor vessel lower head penetration, up to or above the bottom of the vessel depending on the combination of failures that may have led to core damage. The temperature of the containment atmosphere would be at or below the saturation temperature. It is likely that the decay heat would be convected from the RCS surfaces without achieving very high temperatures. In many scenarios water would eventually rise in the vessel, quenching the heated surfaces and removing the potential for further revaporization.

4.5.3 Results

It is necessary to rely on expert judgement at this time regarding the extent of revaporization for characteristic sequences in the ALWR. The revaporization fractions have been based on NUREG-1150 expert elicitations.

The values shown in Table 4-10 were taken from the median estimates for cases most similar to the passive plant conditions and are judged to provide margin to expected ALWR revaporization based on ALWR containment conditions. The revaporization of tellurium is estimated to be negligible.

4.5.4 Conclusions

The extent of revaporization that would occur in a severe accident depends on the details of the plant design and the accident scenario. Although methods for the analysis of revaporization are not well developed, the mechanisms are known well enough to provide assurance that under the conditions of a flooded cavity and low containment temperature that would exist in the passive designs, the potential for revaporization is small. Revaporization is thus likely to be a small contributor to containment fission product concentration as reflected in Table 4-10 values.

4.6 Ex-Vessel Release Magnitude

4.6.1 Introduction

In this section fission product releases from core debris exiting the reactor vessel lower head into the reactor cavity/lower drywell are estimated.

4.6.2 Important Phenomena and Assumptions

Fission product release from core debris which has penetrated the reactor vessel lower head and is located in the cavity/lower drywell will be significant only if the debris is allowed to be dry with no cooling or scrubbing by an overlying pool of water. This would permit the potential for core concrete interaction and the continued generation of additional aerosols. The ALWR requirement for early cavity/lower drywell flooding will provide rapid debris quenching thus minimizing core concrete interaction and ex-vessel release. Further, even if some ex-vessel fission product release were to occur, the cavity/lower drywell would have significant water overlying the core debris such that there will be scrubbing of aerosols.

TABLE 4-10. REVAPORIZATION RELEASE
(Fraction of Elemental Group Initially Deposited)

	Iodine	Cesium
PWR	0.06	0.055
BWR	0.10	0.05

4.6.3 Results

Because of the ability of the Passive ALWR to provide ample coolant to the reactor cavity/ lower drywell prior to or immediately upon vessel penetration, little or no fission product release to the containment atmosphere is expected from ex-vessel debris. All of the noble gases from the molten material are assumed to have been released as a part of the in-vessel core melt progression. The remaining volatiles and low volatiles will be retained within the cooled debris or even if some release were to occur, largely removed by pool scrubbing.

If ex-vessel releases were to be estimated for a given scenario, some period of time for uncooled core concrete interaction would need to be assumed. Table 4-11 provides such an estimate under the assumption of approximately 30 minutes delay in providing coolant to the debris. All of the remaining I and Cs are assumed to be released. The amount of tellurium released is based upon a review of worst case VANESA calculations which indicate that over a 30 minute period about 35% of the tellurium is the maximum that can be released and corresponds to the time when zircaloy oxidation may be occurring.[4-42] The remainder of the radionuclide releases were estimated from the results of the ACE corium concrete tests.[4-43] The results of these tests were extrapolated to envelope the releases which might be expected from either basaltic or limestone concrete.

4.6.4 Conclusions

Little or no fission product release to the containment atmosphere from ex-vessel debris is expected due to the ALWR design requirement for early flooding of the cavity/lower drywell. Even if a short period of core concrete interaction were to occur, only a limited fission product addition to the containment atmosphere would be expected over and above that already being assumed for the in-vessel releases.

TABLE 4-11. RELEASE ESTIMATES FROM EX-VESSEL UNCOOLED FUEL DEBRIS

Chemical Species	Fraction Released from Debris Inventory[1,2]
I, Cs	1.0
Te, Sb	0.35
Ru	0.01
Sr, Ba	0.002
Remainder	0.001

1. In the event that debris bed is uncooled for a period approximately 30 minutes in duration.

2. Numbers are fractions of the fission product inventory contained in the core debris as it exits the vessel lower head.

4.7 Chemical Form

4.7.1 Introduction

The chemical forms for important fission products in the physically-based source term may be defined by carefully considering the chemical environment which the fission products experience after being released from the fuel. This environment is determined by conditions existing in both the RCS and the containment region during representative accident sequences. Early (few hours) as well as long term (days) conditions and phenomena must be considered in defining the chemical forms, which have a strong influence on the fission product transport and deposition behavior, and thus on the fission product inventory in the containment atmosphere that is available for leakage.

4.7.2 Important Phenomena and Assumptions

At the high temperatures characteristic of core damage accidents the fission products are usually assumed to escape from the fuel as atomic or simple molecular species and enter the steam-hydrogen mixture flowing up through the core. As this mixture moves downstream and cools thermodynamic analyses generally have been successful in predicting the stable end products. Except for the noble gases, the end products tend to consist of various condensed compounds including salts, hydroxides, oxides, and intermetallics which would be in aerosol form at the expected RCS exit conditions. The aerosol character of this source material should remain essentially unchanged during its airborne lifetime in the containment atmosphere.

In such cases, the considerable body of aerosol data and models which have been developed over the last decade or more can be applied to reliably estimate the time dependent behavior of the source material during an accident. This is true even though changes in chemical species may occur, provided the new species are also condensed compounds (and hence aerosols) at prevailing conditions.

In cases where chemical changes can produce species which are gaseous at prevailing conditions, the modeling will have to reflect the unique transport and deposition properties of these forms. This situation is particularly relevant for radio iodine because processes are known which can generate several volatile forms of iodine during severe accidents. The purpose of this section is to describe the important phenomena and to evaluate their effect on the overall chemical composition of the iodine source material, first in the RCS and then in containment.

4.7.2.1 RCS Conditions and Reactions. There is significant analytical and experimental evidence that CsI will be the dominant chemical form of iodine in a core damage event. As the steam-hydrogen gas mixture and entrained fission products move from the damaged core to cooler regions of the RCS, thermodynamic analyses predict that CsI and CsOH will be the stable end products.[4-44,4-45]

The results of several experimental programs are in agreement with the above predictions. In the STEP tests,[4-46] fission product iodine was frequently found to be collocated with fission product cesium on deposition coupons and aerosol collection samples. In addition deposit morphology was consistent with the presence of CsOH which would have been a liquid droplet aerosol at test conditions. The investigators concluded that CsI was the principal iodine-containing species in the tests, and they also concluded that flow blockages in two of the tests probably had been caused by accumulation of viscous CsOH plus structural component aerosol material at constrictions in the downstream flow systems.

In the SFD tests,[4-47] the deposition patterns of Cs and I fission products were very similar and it was concluded that the overall behavior of iodine in these tests was consistent with that predicted for CsI, but inconsistent with the assumption that the iodine was elemental or hydrogen iodide. CsOH was also identified as the dominant cesium form.

In the LOFT FP-2 test the deposition pattern of fission product iodine indicated that it existed as an aerosol rather than a gas in the upper plenum (see Reference 4-35). Analysis of the test results indicates that AgI was probably the dominant chemical form of iodine in that particular experiment (i.e., low burnup fuel, low pressure RCS, and Ag-In-Cd control rod failure in the upper core region prior to fission product release). No evidence was found for volatile forms of iodine.

In a series of out-of-pile fission product release experiments with high burnup fuel at Oak Ridge National Laboratory (ORNL),[4-48] the investigators concluded from analysis of thermal gradient tube deposition profiles that CsI and CsOH were the dominant downstream iodine and cesium species for conditions which simulated LWR core damage conditions. Finally, measurement of the iodine speciation in the containment sump water from the TMI-2 accident followed by an analysis of how the species could have been produced concluded that fission product iodine entered the water primarily as iodide and not as elemental iodine (see Reference 4-18).

Notwithstanding the above evidence for CsI, there are two potential reactions within the RCS which can potentially convert CsI into volatile hydrogen iodide (HI) and need to be explicitly addressed. The first is thermal hydrolysis by steam which can be expressed as

$$CsI\ (g) + H_2O\ (g) = CsOH\ (g) + HI(g). \qquad (1)$$

This gas phase reaction causes measurable conversions only at higher temperatures (>1000K) where CsI begins to have significant vapor pressure. At these higher temperatures experimental work has also shown that the reaction can be permanently shifted to the right in a stainless steel flow system because the CsOH tends to be retained at reaction sites in the oxide corrosion film which is present on interior metal surfaces.[4-49,4-50] Even with this shift, however, the amount of HI formed by reaction (1) should be small in the reactor accident case. This is because of the excess CsOH which is expected to accompany CsI during its early transport through the RCS. (On a molar basis, the fission yield of cesium is about ten times that of iodine.) The CsOH will tend to shift the reaction equilibrium to the left and also occupy reaction sites on the metal surfaces.

The second reaction which can produce HI within the RCS is the reaction between CsI and boric acid. The condensed phase reaction is most simply expressed as

$$CsI(s) + H_3BO_3(l,s) = CsBO_2(s) + H_2O(s)\ HI(s). \qquad (2)$$

Although this condensed phase reaction should not occur to a significant extent under core damage conditions, appreciably more reaction can occur when one or both of the materials is in vapor form.[4-51] The potential for realizing such conditions depends upon accident scenario details, relative times and rates of release from the core region, and specific RCS transport and deposition patterns. Since CsOH readily reacts with boric acid, its presence tends to inhibit the CsI reaction to the extent that it sequesters the boric acid vapor. The LOFT FP-2 experiment (see Reference 4-35) supports the limited CsI-boric acid reaction since this experiment was borated and little, if any, evidence of $CsBO_2$ was found during pre-reflood. The boric acid reaction is, of course, much less relevant to those reactor designs which do not employ boric acid as a coolant additive.

Hydrogen iodide, while gaseous, is chemically quite reactive and highly soluble in water. Thus it would tend to react with bare metallic surfaces in the RCS (i.e., structures or aerosols) to form metal iodides. These would be condensed species at exit conditions. Its water solubility means HI would be very hygroscopic so it would also tend to form embryonic fog droplets (e.g., molecular clusters) in cooler parts of the RCS. However, at temperatures above about 600°K hydrogen iodide can undergo dissociation to form hydrogen and iodine as follows[4-52]

$$2HI(g) = H_2(g) + I_2(g). \qquad (3)$$

This change in iodine species should be minimized in a core damage accident because of the hydrogen already present in the RCS gas mixture; that is, the equilibrium expressed in the above equation would be shifted to the left.

For core damage accidents that result in reactor vessel lower head penetration it is conceivable that natural convection forces could be strong enough to induce circulation of portions of the containment atmosphere through the RCS. This would expose any resident cesium iodide to a more oxidizing environment and thus introduce the possibility of generating some molecular iodine. While the Passive ALWR design is such that the reactor cavity/lower drywell will be flooded above the bottom of the reactor vessel lower head very quickly after lower head penetration such that significant circulation through the RCS will not occur, the potential for molecular iodine release from this mechanism has been considered to assess the margin for this aspect of iodine chemistry. Thermodynamic analysis using the SOLGASMIX computer code[4-53] indicates that temperatures greater than about 1000°K would be needed and the principal oxidized iodine species would be atomic iodine (I) at these temperatures rather than molecular iodine. The analysis also shows that the presence of excess cesium would reduce the amount of CsI decomposition. Furthermore, as the reaction mixture would flow to cooler locations of the RCS, cesium iodide would reform if the I or Cs had not been removed by reactions such as with structural surfaces. Any I that might react with a surface would be removed from the gas phase and thus cease to be a volatile form. Loss of Cs from the mixture would be minimized by resident cesium that would be expected to be deposited throughout the RCS flow circuit. On this basis, air (oxygen) entry into the RCS would probably not cause escape of significant amounts of volatile iodine.

4.7.2.2 Containment Conditions and Reactions. The thermal-hydraulic conditions in containment during a core damage accident can be characterized in general as consisting of a steam saturated atmosphere at moderate pressures accompanied by condensate wetted surfaces and various standing water pools or reservoirs. The large, passive heat transfer surfaces and the large amounts of water inside containment assure these conditions for the passive plant. Steam condensation on the cooler, heat transfer surfaces will result in diffusiophoresis which drives suspended aerosol particles to the condensed water film on these surfaces. The abundance of moisture will also tend to increase the size of suspended aerosols and hence their sedimentation rates. For non-hygroscopic materials the aerosol particles will serve as nuclei for steam condensation provided the atmosphere is supersaturated (since there is no solubility to drive absorption). For hygroscopic materials water absorption will readily occur, even in sub-saturated atmospheres, to form droplets composed of concentrated electrolyte solutions.

The latter case applies to CsI, HI, CsOH, and any other chemical compound that is highly water soluble. The resulting aerosol droplets will participate in all ongoing aerosol removal processes. This will effectively lead to a steady buildup of fission product iodine, cesium, and other materials in the various containment water reservoirs and to a lesser extent on containment structural and equipment surfaces. In the water reservoirs soluble materials will be dissociated into non-volatile ionic species while insoluble materials will settle out as a precipitate sludge. Surface deposited material will tend to remain fixed because of physical adhesion and/or chemisorption phenomena. The generation of volatile fission product species under these circumstances requires consideration of further chemical reactions that may take place in the aqueous phase, on surfaces, or even in the gas phase (i.e., while fission product material is airborne).

As noted earlier, the consideration of volatile fission product species essentially reduces to an evaluation of iodine behavior. Other fission products have little or no tendency for forming volatile species at containment conditions and the radiological significance of possible exceptions to this generality are minor when compared against iodine. In the case of iodine the volatile species of concern historically are elemental iodine and organic iodides (principally methyl iodide). The remainder of this section will attempt to provide perspective on the various processes which can generate these species in containment during the course of a severe core damage accident.

The iodine specie formed when either CsI or HI first dissolve in water is the iodide ion (I-). However, even if a substantial fraction of the core iodine inventory should dissolve in available containment water reservoirs the resulting aqueous concentration would be quite low; typically in the range of 10^{-5} molar. At such concentrations a variety of reactions with other substances in the water (i.e., dissolved gases or other minor impurities) can occur which will produce additional iodine species. The relationship between important aqueous iodine species at low concentration can be illustrated through use of the following equation[4-54]

$$3I_2 + 3H_2O = 5I^- + IO_3^- + 6H^+. \quad (4)$$

This expresses a global equilibrium situation which really involves numerous intermediate reactions and species that will have different rates and lifetimes depending on the specific thermochemical conditions. However, it also illustrates the observed fact that solution pH has an important influence on iodine speciation. High H^+ concentrations (low pH) tend to shift the equilibrium to the left (i.e., higher relative molecular iodine concentrations) while low H^+ concentrations (high pH) tend to increase the relative concentrations of the ionic species iodide and iodate.

A very important process that can affect iodine speciation in containment water reservoirs is radiolysis. At the radiation levels that would be expected in core damage accidents (tens of kilogray per hour) radiolysis generates appreciable aqueous concentrations of oxidizing entities such as hydroxyl free radicals, hydrogen peroxide, etc.. These can readily oxidize I^- to I_2 and further through HIO to IO_3^-. The steady-state concentrations of the different iodine species depend upon ambient conditions, particularly pH.[4-55,4-56,4-57] In general low pH conditions favor formation of I_2 while high pH tends to stabilize I^-. The production of I_2 in solution has two significant consequences. First, its limited solubility in water will cause some of it to volatilize (partition) into the overlying gas space where it can become available for leakage from containment. Second, I_2 in solution as well as I_2 in the gas phase can participate in reactions with a variety of organic materials (i.e., paints, oils, cable insulation, volatile solvents, methane, etc.) to generate low molecular weight organic iodides such as methyl iodide (CH_3 I).[4-58,4-59] Since these species are only slightly soluble in water and have relatively long airborne lifetimes in containment, they constitute another potential leakage form.

The importance of minimizing radiolytic I_2 formation by controlling pH is clear. Thus the ALWR design requirements specify that the pH of containment water pools is to be maintained in an alkaline state for the accident duration. This may require addition of a relatively strong alkaline buffer to protect against long term acid sources such as nitric acid formation from radiolysis of moist air[4-60] or perhaps carboxylic acid generation from radiolytic decomposition of oxygenated organic substances (paints, solvents, etc.) that might be in containment. Sodium borate or any other similar alkaline salt could be used for this purpose.

In addition to radiolysis of aqueous iodide, which tends to produce a steady-state concentration of I_2 in the system, there are several other potential generators of I_2 that must be considered which are more transient in nature. These include possible oxidation of suspended CsI during hydrogen combustion events,[4-61] I_2 formation during evaporation to dryness of shallow water puddles (see Reference 4-56), radiolysis of acidic droplets containing HI which may have been released from the RCS,[4-62] and possible oxidation of iodide species that could be evolved should corium-concrete interactions occur in the reactor cavity/lower drywell following penetration of the reactor vessel lower head. Limited experimental data on the relevant phenomena combined with uncertainties in accident progression make precise quantification difficult, but I_2 yields from each of these processes are expected to be small.

With regard to the first process, any CsI that has dissolved in a water pool would not be affected by a hydrogen deflagration. In order to produce a significant effect, a large energetic hydrogen deflagration would have to occur early in an accident when most of the CsI aerosol is still suspended in the containment atmosphere. At early times steam partial pressures tend to be sufficiently high to preclude global combustion. Furthermore, recent experimental work shows that relatively low steam concentrations will act to protect airborne CsI from oxidation even if a hydrogen deflagration should occur.[4-63]

Concerning the second process, experimental work has indicated that appreciable volatile iodine is produced when iodide solutions evaporate to dryness in a high radiation field. Alkaline conditions reduce but do not eliminate liberation of volatile iodine which is presumably I_2. However, the Passive ALWR plant has such a large inventory of water in the containment that only an insignificant fraction would be expected to experience evaporation to dryness. It follows that I_2 generation from this process should also be insignificant.

Radiolysis of suspended aerosol droplets containing HI could in principle generate I_2 if they were to remain acidic. However, interactions between these droplets and the alkaline aerosol droplets which are expected to be in greater abundance should minimize the I_2 yields from this process. In addition, a recent scoping experiment with a concentrated HI solution indicated very low rates of radiolytic I_2 formation.[4-64] Thus, this process should not be a significant source of containment I_2.

Finally, any fission product iodine that may remain in the core debris that would be discharged from the reactor vessel at lower head failure would be readily released if the reactor cavity remains dry such that the high temperature core-concrete interaction could proceed unchecked. The vaporized iodine species would encounter oxidizing conditions upon entry into the containment atmosphere which would favor I_2 formation. However, Passive ALWR design requirements will result in cavity flooding before or coincident with core debris entry. Under these circumstances, even if any iodine should escape the core debris it would be expected to dissolve in the overlying water and not experience any significant contact with the containment atmosphere. Thus I_2 formation by this process would not occur to any significant extent.

4.7.3 Results

The qualitative description and assessment of phenomena given above must be translated into quantitative fission product chemical form compositions. The noble gases will, of course, exit from the RCS and exist in containment entirely as chemically inert gases during the course of a reactor accident. With respect to other fission products, all except iodine can be assumed to exit the RCS and transport in containment entirely as mixed aerosols during an accident. Specifications for iodine are as follows.

Almost all the iodine leaving the RCS during a core damage accident should be in particulate (aerosol) form. The dominant chemical species will be CsI but some other metallic iodides could be present as well. The thermal hydrolysis of CsI and reaction with boric acid (if present) could produce some HI which might escape from the RCS. The expected yield is relatively uncertain but, on the basis of the discussion above, we estimate only a few percent of the iodine would leave the RCS as HI. In containment, the HI would effectively behave as an aerosol due to (a) its hygroscopic nature, and/or (b) its sorption by the accompanying/existing suspended aerosol. Finally, the RCS is expected to discharge a negligible amount of I_2 ($<0.1\%$) to containment during either the early or later phases of an accident.

Within containment, the four processes discussed earlier could each produce a small amount of I_2. Even with control of aqueous phase pH as assumed, radiolysis is judged to represent the most important and persistent source. The steady-state I_2 level reached in the containment atmosphere will depend upon the net effect of many competing rate processes and upon plant specific design parameters. However, it is considered that the pH control specified in the Requirements Document will be adequate to assure that the airborne I_2 will not exceed 1.5% of the core iodine inventory as a result of aqueous radiolysis reactions. The other three potential sources of airborne I_2 are together judged capable of generating 1.5% as well so the peak I_2 level in the containment atmosphere would be about 3.0%.

It is generally recognized that thermal and/or radiolytic reactions between I_2 and a wide range of organic substances which may be present in containment vessels are responsible for the appearance of organic iodides in these systems (see References 4-58, 4-59). Measured yields depend on a variety of parameters which include I_2 concentration, temperature, radiation dose, type of organic, and geometry effects among others. No satisfactory mechanistic model of organic iodide generation in reactor accidents has been developed as yet although research on the problem has been going on for many years. However, an empirical procedure was devised some time ago which tends to overpredict steady-state organic iodide levels in containments.[4-65] Application of the procedure, which relates percent organic iodide formation to airborne iodine concentration, to ALWR conditions would indicate conversion of roughly 5% of the airborne I_2 into organic species. This amount of organic iodide would thus correspond to about 0.15% of the core iodine inventory (i.e., 5% of 3% is 0.15%) and the amount of I_2 would then be reduced to 2.85%. The remainder of the iodine inventory, 97% of the core inventory, would exist as non-volatile species. These would be either particulate aerosols or deposited material (attached to surfaces or trapped in water).

4.7.4 Conclusions

The discussion above has attempted to summarize the important factors involved with determining the fission product chemical forms in water reactor core damage accidents and to make use of the available data on important phenomena to develop quantitative estimates for Passive ALWR physically-based source term applications. As indicated, it is difficult to precisely quantify some of the phenomena effects but in most cases this difficulty probably has a minor impact on accident source term predictions. For example, fission product cesium is expected to exist in a number of different chemical species depending upon the details of an accident, the specific point in an accident, and perhaps the particular reactor design. However, the general transport behavior of cesium within the system and its leakage characteristics will not be sensitive to these details because the different species have very similar physical and chemical properties.

In the case of iodine, which can form dissimilar volatile species, the abundance of water in the Passive ALWR combined with aqueous pH control is of considerable help in simplifying the chemical issues requiring quantification. The former eliminates significant core-concrete interaction and the need to assess accompanying volatile iodine generation and the latter tends to stabilize non-volatile aqueous iodine species and limit partitioning into the gas phase. Quantification of organic iodide formation is the most uncertain aspect of the evaluation but the procedure used should overpredict such yields, providing margin in the organic iodine fraction. The levels of organic iodide measured in the TMI-2 containment atmosphere following that accident (about 0.03%) would indicate that this is true.

4.8 Scrubbing In Water Pools

4.8.1 Introduction

The attenuation of radionuclides in water pools is usually expressed as a "decontamination factor" (DF), which is defined as the ratio of the quantity injected into the pool divided by the quantity which escapes the surface of the pool. Although it has been generally accepted that large pools of water can be very effective in scrubbing contaminants from a gas stream passing through them (for example, in iodine scrubbing in the spent fuel pool), it is recognized that the effectiveness varies significantly with a number of parameters. The Reactor Safety Study (WASH-1400) assumed a DF of 100 for subcooled suppression pools, and 1.0 for steam saturated pools. Since that time, detailed models for the analysis of aerosol removal during gas transport through the suppression pool have been developed by the NRC (i.e., the SPARC code[4-66]) and EPRI (the SUPRA code[4-67]) in the U.S., as well as by several foreign countries (e.g., BUSCA[4-68]).

4.8.2 Important Phenomena and Assumptions

A reasonably complete data base exists to permit the assessment and verification of these pool scrubbing models. Recent experimental work began with small scale experiments performed by General Electric to demonstrate the effectiveness of aerosol scrubbing in suppression pools.[4-69] Small scale "single orifice" and large scale experiments for a number of different injection configurations were performed at Battelle Columbus Laboratory.[4-70] Phase A of the ACE Program also included pool scrubbing tests[4-71] in an intermediate scale. In addition, there is on-going experimental work in Japan,[4-72] and Italy.[4-73,4-74] At the time of this writing, the results of these on-going programs are not yet available.

The results of analytical models, confirmed by the available experimental results, indicate that suppression pool scrubbing of aerosols depends on parameters associated with:

- the carrier fluid (steam/non-condensible gas ratio, temperature, mass flow rate);
- the entrained aerosol characteristics (size, material, density, solubility, aerodynamic characteristics);
- the injection configuration (submergence depth, orifice size and orientation, number of orifices in proximity); and
- the water pool (subcooling, geometry, impurities).

Of these, the aerosol size is the most sensitive parameter. The observed DF, for example, varies over several orders of magnitude for aerosol sizes of interest in the region between 0.5 and 1.0 microns (radii of 0.25 to 0.5 microns, see Figure 4-8).

A second important parameter shown in Figure 4-8 is the condensible/non-condensible fraction of the carrier fluid. Large steam mass fractions result in large decontamination factors, while the minimum DF is calculated for dry (hydrogen or air) gas flows. In contrast, the experiments showed the effect

EFFECT OF PARTICLE SIZE AND STEAM FRACTION ON DF

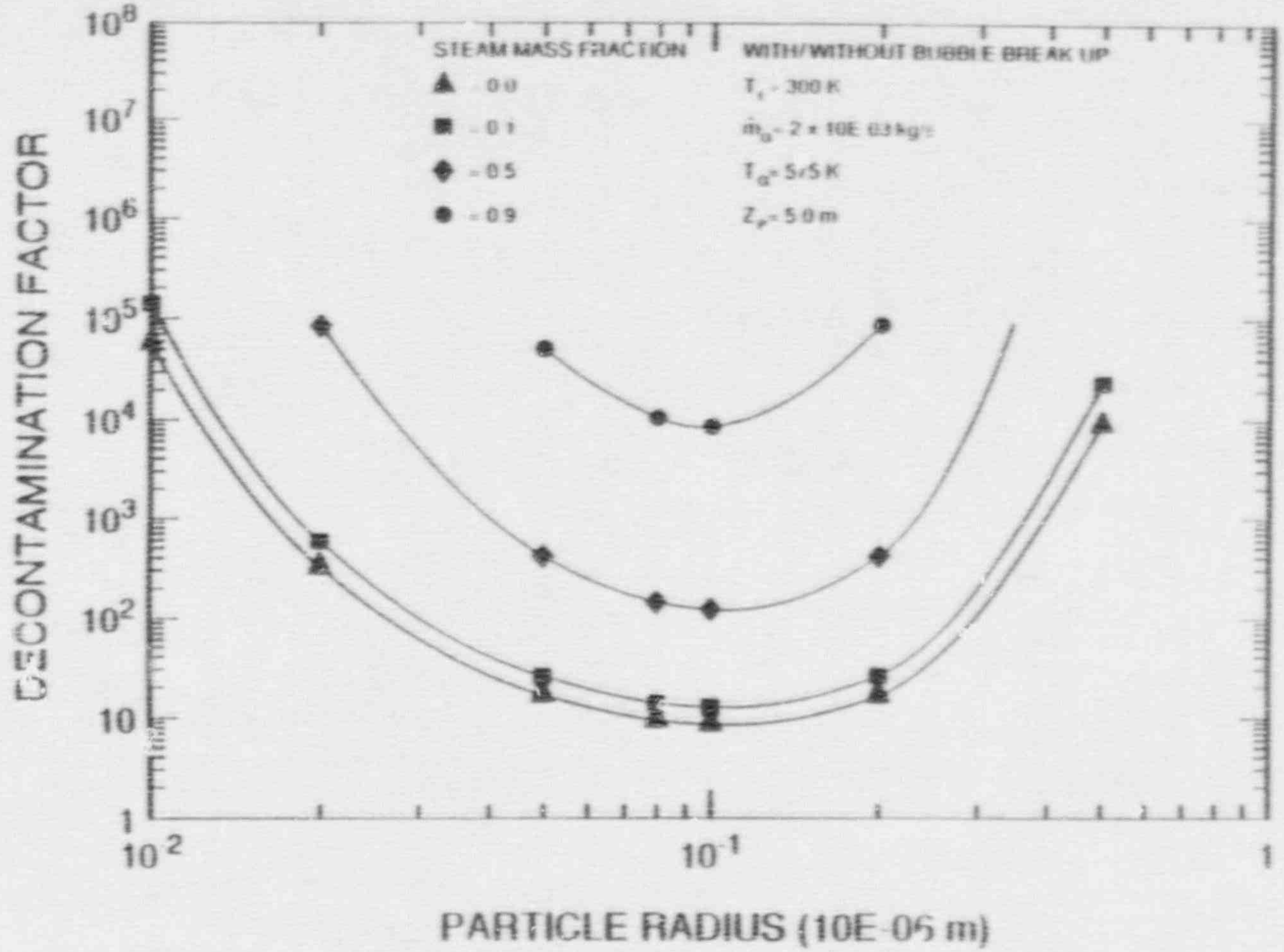


Figure 4-8. Suppression Pool DFs Calculated by the SUPRA Code.

of pool saturation to be much less than anticipated, as a result of additional removal mechanisms (such as diffusiophoresis) associated with high steam fractions above the surface of the pool (see Reference 4-70).

The injection configuration can have a significant effect on scrubbing in the pool entrance region. In contrast to the bubble rise region, where bubble dynamics and aerosol phenomena are well characterized by the models, in the entrance region the breakup of the gas stream entering the pool is more difficult to model. As a result, conservative models have neglected the scrubbing in this region of bubble formation, breakup and interaction. This entrance region is much more pronounced for large horizontal vent configurations than for multi-port quencher injection configurations. However, pool scrubbing experiments (see References 4-69,4-70) have shown that scrubbing at the injection site can be significant, and should be included in suppression pool scrubbing analyses. An analytical model for scrubbing at pool injection sites,[4-75] developed under EPRI sponsorship, also concluded that scrubbing at the injection site can be appreciable.

Although this model requires validation, it indicates that decontamination factors between 2 and 5 (depending on the steam fraction) could be expected for particles of 0.3 micron size. Battelle Northwest Laboratory has attempted to model the entrance effects, and has concluded that entrance effects would not extend beyond ten diameters of the vent pipe for horizontal vent injection configurations (i.e., Mark III-type suppression pools).[4-76]

For any given set of the important parameters identified above, the existing models permit a reasonably accurate determination of the corresponding pool DF. With high steam content carrier gas, an anticipated aged particle size (e.g., 1-5 microns), and sufficient pool depth to minimize the effect of the entrance region, pool decontamination factors well above 1000 are calculated and have been observed experimentally.

During any specific accident sequence several important parameters may change significantly. In particular, fission products may be carried to the drywell and into the suppression pool in a hydrogen-rich gas mixture. However, low steaming rates during such periods would also reduce the total gas flow rate resulting in a slower transport, and hence additional aging, of the fission product aerosols prior to injection into the suppression pool. Therefore, the reduction in scrubbing efficiency resulting from the higher non-condensible fraction is likely to be balanced by an increase in efficiency resulting from increases in the aerosol size distribution. Suppression pool models incorporated into integral severe accident codes will produce time varying suppression pool decontamination factors which quantitatively account for such changes in the important parameters.

4.8.3 Results for Scrubbing DF

With regard to the timing and flow paths through water pools in the passive ALWR there are four locations where scrubbing of fission products in water pools needs to be considered:

1. the discharge of the reactor coolant through the safety relief valves into the IRWST or suppression pool, via the quenchers;

2. flow from the drywell to the suppression pool through the horizontal vents;

3. the discharge of non-condensible gases from the isolation condenser vent line into the suppression pool; and

4. the bubbling of core-concrete interaction gases through the overlying water pool in the reactor cavity or lower drywell.

A discussion of each of these pathways follows:

Safety Relief Valves (PWR and BWR)

Any aerosols carried from the reactor vessel to the IRWST or suppression pool via the quenchers will experience conditions resulting in maximum pool scrubbing effectiveness as a result of the high steam fraction of the carrier fluid, injection into the pool through quenchers (which results in small bubble formation), deep submergence, and subcooled pool temperatures. Decontamination factors expected on the basis of code calculations as well as experimental observations for these conditions are on the order of 1000. Any aerosols carried to the suppression pool via this pathway, therefore, are a negligible contribution to the source term.

Horizonal Vents (BWR)

Effective decontamination (although at a somewhat lower DF) can also be expected for fission products other than noble gases and organic iodide swept into the suppression pool during periods of high pressure in the drywell. The DFs are expected to be somewhat lower as a result of the higher content of non-condensible gases and an injection configuration less optimum than the quencher system. Conversely, the aerosols carried by this flow stream are characterized by a significantly larger particle size as a result of aging (i.e., growth in aerodynamic size characteristics resulting from agglomeration) and the absorption of water vapor by hygroscopic materials such as cesium hydroxide. DFs on the order of 100 are estimated for this flow path.

Isolation Condenser (BWR)

A third pathway for fission products to enter the suppression pool and wetwell gas space is the non-condensible gas vent line from the isolation condenser. This line has a shallow submergence in the suppression pool, and the carrier gas for aerosols transported this way has a low steam fraction. As a result, the DF estimated for this pathway is very small. Some scrubbing is expected as a result of the inclusion of very large particle sizes caused by the long pathway from the reactor vessel through the isolation condenser at the top of the reactor building, and back again to the suppression pool. As a first estimate, however, no pool scrubbing credit has been assumed for this pathway.

Core-Concrete Interaction (BWR and PWR)

In addition to scrubbing in the suppression pool, the passive plant arrangement affords scrubbing of fission products released from any core-concrete interaction. While the ALWR design requirement of flooding prior to or immediately upon reactor vessel lower head penetration is expected to result in rapid quenching of ex-vessel debris and therefore little, if any, core concrete interaction and ex-vessel fission product release, the scrubbing effect of the water pool overlying the core debris has been evaluated to assess the margin for this aspect of the source term.

The reactor cavity or lower drywell is flooded with sufficient water to cover any debris on the cavity floor with approximately 10 or more feet of water. The positive factors affecting aerosol scrubbing by this water pool are:

1. water depth of 10 or more ft;
2. small bubbles created when core-concrete interaction gases reach the debris-water interface;
3. a bimodal particle size distribution consisting of very small particles resulting from nucleation of fission product vapors in the bubbles and large particles created by mechanical forces at the debris surface; and
4. initially subcooled water.

On the negative side, the gases reaching the surface of the core debris may have a high non-condensible content, and the pool water will eventually approach saturation conditions.

Recent experiments conducted by the ACE Consortium (see Reference 4-71) provide a useful baseline for the scrubbing of aerosols from a core-concrete interaction. These tests (Test No. AA1 and AA3) were conducted with very low steam fractions (0.012-0.013), submergence by 1.38 and 2.62 meters of water and temperatures between 26 and 86 °C. The aerosols were generated by plasma torch and included Cs, I, and Mn. The decontamination factors obtained varied between 11 and 160 for the 1.38 meter submergence, and 75 to 330 for the 2.62 meter depth. The lower values of DF were associated with the Mn aerosol, which is insoluble in water, while the higher DFs were measured for the highly soluble Cs and I. These experimental results are entirely consistent with the expectation that the scrubbing effect of the overlying deep water pool will reduce fission product releases from a core-concrete interaction by at least an order of magnitude.

4.9 Primary Containment Aerosol Removal

4.9.1 Introduction

Previous sections of this report have dealt with the release of fission products from the fuel and their transport through the RCS to the containment. The dominant physical form of the fission products (other than noble gases and a small fraction of the iodine) upon release to the containment is expected to be particulate. The size distribution of this particulate will be in the aerosol range (i.e., less than 100 μm in diameter) with an estimated concentration in the containment less than several g/m^3.

The principal means of removing the suspended fission products from the containment atmosphere traditionally include use of containment spray and pool scrubbing. As a part of plant simplification, the Passive ALWR design requirements do not specify a containment spray system. Other means of fission product removal therefore become important in determining the overall concentration of aerosols in the containment.

Scrubbing in the IRWST or wetwell remains effective for scenarios in which releases from the RCS (BWR and PWR) or drywell (BWR) are directed to these pools. The effectiveness of pool scrubbing is presented in the preceding section (Section 4.8) of this report.

In this section, natural mechanisms for depleting the containment atmosphere of particulate are evaluated and an estimate of the aerosol concentration and potential leakage as a function of time is derived. These natural processes include sedimentation and diffusion of aerosols.

4.9.2 Important Phenomena and Assumptions

In this section a brief definition of important physical processes in the removal of aerosols is provided followed by a description of the evaluation performed on the PWR and BWR containments to determine overall aerosol concentrations resulting from these effects.

4.9.2.1 Phenomena Associated with Natural Depletion. The following is a brief discussion of the physical processes of aerosol mechanics which could be taken into account in establishing the source term. These processes provide a basis for crediting the "natural" depletion of fission product material in the containment atmosphere.

4.9.2.1.1 Agglomeration--Agglomeration is the process by which the size distribution of airborne particulate tends to shift with time to larger sizes until an equilibrium condition is reached. It is not a separate removal process, but affects several removal processes: sedimentation, pool scrubbing, and spray removal. There are three agglomeration mechanisms that are generally treated, which include:

1. Brownian - the random movement of particles and the resultant collisions
2. Gravitational - the relative movement of particles of different size under the influence of gravity
3. Turbulent - the result of localized mixing with an effect of relative movement similar to gravitational

In containment, Brownian agglomeration is important for submicron particles while gravitational agglomeration is important for particles larger than one micron. Turbulent agglomeration is generally unimportant in containment.

4.9.2.1.2 Sedimentation--Sedimentation is deposition due to the effects of gravity on the particles, with accumulation generally on horizontal surfaces. In "stirred" systems, sedimentation still occurs, because if the system is closed, there is always a net downward movement of the particles. If the system is turbulent, both agglomeration and deposition will be enhanced.

4.9.2.1.3 Hygroscopicity--Hygroscopicity is the term used to characterize the affinity of a substance for water. Substances that can maintain large quantities of water in solution are termed "hygroscopic." As noted earlier in this report, the dominant chemical form of fission product cesium released to the containment in the course of a severe accident would be CsOH (i.e., cesium hydroxide), and CsOH is one of the most hygroscopic materials known. If in particulate form and in the aerosol size range, and it is exposed to atmospheres near saturation (saturation ratios greater than about 0.95), it can absorb factors of ten to one hundred times its mass in water.

Hygroscopicity can be credited in aerosol removal if it can be demonstrated that the containment atmosphere is maintained near saturation. The effect of hygroscopicity is to increase the rate of particle growth and the sedimentation.

4.9.2.1.4 Diffusiophoresis--As steam condenses on a surface, aerosol particles will migrate with the flux of water vapor moving to the surface and be deposited. This deposition process is referred to as diffusiophoresis, or sometimes Stefan flow. The importance of diffusiophoresis depends on the amount of condensation occurring in the accident sequence. If the surfaces in the containment are not cooled, the structures will tend to saturate thermally, steam condensation on the walls will slow, and the amount of diffusiophoretic deposition will decrease with time. Diffusiophoresis is a well-established phenomenon that is modeled in mechanistic computer codes of aerosol behavior such as CONTAIN[4-77] and NAUA,[4-78] as modified for incorporation into the NRC Source Term Code Package. Although not typically found to be the dominant deposition mechanism in severe accident analyses, diffusiophoresis can be an important contributor; it is not sensitive to particle size, and, as a result, can be effective in the removal of an otherwise persistent airborne concentration of small aerosols. Another diffusive mechanism for aerosol removal is Brownian diffusion. It is usually of minor importance compared with sedimentation, except for small (submicron) particles.

4.9.2.2 Primary Containment Aerosol Removal. For purposes of defining the physically-based source term, primary containment aerosol removal is modeled using the EPRI version of the NAUA code. The modeling assumptions in EPRI's version of NAUA are first reviewed, then specific features of the calculations for the passive PWR and BWR are described.

4.9.2.2.1 General Assumptions in EPRI's NAUA--The following are general assumptions:

1. The containment is a single well-mixed compartment whose volume and surface areas for aerosol sedimentation, Brownian diffusion, and Stefan flow diffusiophoresis are user-specified.

2. The containment atmosphere temperature, wall temperature, steam injection rate, and leakage rate are user-specified, all as functions of time.

3. Steam condensation on aerosol particles and on the containment wall are modeled. (The term "wall" might include other heat sink surfaces, but these are assumed to be at the same temperature as the "wall.") Steam condensation on the particles is modeled using the Mason equation. Condensation on the wall is currently modeled using a combination of the Nusselt and kinetic theory approaches.

4. Up to 50 species of particles can be handled. Any specie can be assumed to be hygroscopic. Injection of each aerosol specie into the containment can be continuous, at a user-specified rate as a function of time, or can be a "puff" at user-specified times, or any combination of these.

5. The source size distribution of each aerosol specie is assumed to be log-normal, and is characterized by an average particle radius and geometric standard deviation for that specie. These parameters are input as functions of time. (Note that only the source size distribution is assumed to be log-normal; the suspended aerosol is not necessarily log-normal and has whatever distribution evolves in the calculation). NAUA is a "sectional" code; the suspended aerosol size distribution is apportioned into size "bins." The number of bins and their upper and lower limit radii are user-specified. (Calculations reported herein use 30 bins, r_{min} = .0025 μm, r_{max} = 50 μm).

6. The aerosol processes tracked are:

   - Brownian and gravitational agglomeration
   - Brownian diffusion
   - Stefan flow diffusiophoresis
   - Sedimentation
   - Leakage

7. The processes are affected by the following thermal-hydraulic phenomena, which are calculated at each time step:

   - Steam condensation on the wall
   - Steam condensation on the particles

   A steam mass balance is calculated in each time step, which permits determination of the relative humidity. This in turn supplies the driving force for both condensation processes (particle and wall).

8. The following outputs are provided (among others, all as functions of time):

- suspended mass of each aerosol specie, and of water condensed on the particles;
- integrated sedimented mass of each specie;
- integrated diffused mass of each specie;
- integrated leaked mass of each specie;
- number and mass size distribution, both total and of each specie (mass only), of the suspended aerosol;
- aerodynamic mass median diameter (AMMD) of the total size distribution;
- geometric standard deviation of the total size distribution; and
- various thermal-hydraulic parameters (relative humidity, atmosphere and wall temperatures, etc.).

It should be noted that EPRI's version of NAUA has been well validated against the LACE-2, -4, and -6 experiments.

4.9.2.2.2 Passive PWR Calculations - Input Assumptions--Calculations for the passive PWR were done for the Westinghouse AP600 design as an illustrative example. The containment is modeled as a cylinder of height 43.1 m and radius 18.3 m. The volume is 45,300 m^3. The surface area for sedimentation (floor area) is 1050 m^2, for Brownian diffusion 7040 m^2, and for wall steam condensation 4940 m^2. (Condensation and diffusiophoresis are assumed to take place only on the vertical wall surface, while Brownian diffusion occurs over the entire surface area of the cylinder, including the floor and ceiling).

The accident sequence timing and thermal-hydraulic inputs correspond to a sequence initiated by an intermediate LOCA. In the base case, core uncovery is assumed to begin at one hour after accident initiation, with vessel lower head penetration at five hours. Note that hereafter, because of convenience in the NAUA calculation, t=0 is the start of core uncovery, and all times will be referred to this.

Consistent with the Passive ALWR design requirements, flooding of the cavity is assumed to occur either before or just at vessel failure, so the core debris is quenched. As a result, no molten core-concrete interaction is assumed to take place. Over the first 30-minute period following vessel penetration, only the heat addition from debris quenching is assumed with no additional heat generation from zirconium oxidation considered as a result of rapid quenching of the debris. After this period, further steam is produced from decay heat. Aerosol release following vessel penetration occurs from late in-vessel release and revaporization. No aerosol release from the core debris and concrete is assumed because of the quenching assumption and because the flooding water will scrub any released aerosol.

Table 4-12 shows the assumed steam injection rates to the containment.

Containment atmosphere temperatures and wall temperatures were adapted from MAAP calculations for a similar scenario for the AP600.[4-79] They are shown in Table 4-13. Leak rate was assumed to be approximately the Requirements Document proposed limit of 0.5%/day.

Nineteen aerosol species were considered, including CsOH, CsI, Te, TeO_2, BaO, SrO, CeO_2, La_2O_3, Ru, Sb and various structural and concrete materials (while it is considered that debris quenching will occur in the passive plant design, the concrete materials were included to allow core-concrete interaction sensitivity studies). For all species at all source injection times the assumed log-normal distribution parameters were r(average) = 0.1 μm, geometric standard deviation = 2. CsOH and CsI were assumed to be hygroscopic; all others were non-hygroscopic. (As it turned out, the relative humidity in this problem was always low enough that steam condensation on the particles was not important).

TABLE 4-12. PASSIVE PWR SEQUENCE TIMING

Time (hours)[(1)]	Accident Status	Steam Injection Rate, (lb/hr)
0 - 4	Boiloff of reactor inventory from TAF to vessel penetration	1.5×10^4
4 - 4.5	Debris quenching in reactor cavity	Sensible heat: $5\text{-}7.5 \times 10^4$[(2)] Decay heat: $3\text{-}4.5 \times 10^4$
4.5 - 23	Long term relocation of remaining fuel to reactor cavity and steam generation due to decay heat.	Decay heat: $4.5\text{-}6 \times 10^4$

(1) t=0 is start of core uncovery. Vessel penetration is t = 4 hrs.
(2) $5 - 7.5 \times 10^4$ means that over the given time interval the steam injection rate was linearly ramped from 5×10^4 to 7.5×10^4 lb/hr.

TABLE 4-13. PASSIVE PWR CONTAINMENT TEMPERATURE

t(sec)	T_{atm} (°C)	T_{wall} (°C)
0 (core uncovery)	89.4	66.0
3600	89.4	66.0
5040	104.3	66.6
9000	87.8	58.1
10800	90.0	60.0
14400 (vessel penetration)	90.0	60.0
16200	106.3	82.2
18000	95.1	77.1
25200	90.4	72.9
32400	87.8	71.1
39600	86.0	70.1
50400	84.4	68.8
82800	84.4	68.8

The following summarizes fractions of core inventory that are released to the containment atmosphere. The basis for these fractions was provided in Sections 4.1 through 4.8 of this report. Explicit derivation of the figures below is summarized in Section 2.

Fraction of Core Inventory
Released to Containment

	0-4 hours	4-23 hours
I	.38	.17
Cs	.30	.18
Te	.08	.02
Sr, Ba	.004	--
Ru	.0004	--
Remainder	.00004	--

For iodine, the initial total core inventory is estimated to be approximately 18,000 gm. As determined in section 4.7, 97% of the iodine is assumed to be released as CsI; thus over the 4-hour core degradation period 0.97 x 0.38 x 1.8 x 10^4g of I are released, or 0.46 g/sec of iodine. This corresponds to 0.93 g/sec of CsI from 0 to 4 hours.

From 4 to 23 hours, 0.17 x 1.8 x 10^4g = 0.045 g/sec iodine or 0.092 g/sec CsI arises from late in-vessel release. In addition, revaporization leads to 0.06 x 0.38 x 1.8 x 10^4 = 0.006 g/sec iodine or 0.012 g/sec CsI. Thus the total release rate for CsI from 4 to 23 hours is 0.092 + 0.012 = 0.104 g/sec. (In the actual calculation the rate used was 0.101 g/sec, based on a slightly different value of the iodine revaporization rate.)

The release rates for CsOH and the other fission product species were obtained in a similar fashion. The release rates for the structural materials were estimates based on observed releases in fuel degradation experiments. Results are presented in Section 4.9.3.

4.9.2.2.3 Passive BWR Calculations - Input Assumptions--The General Electric SBWR design is used as the example for the passive BWR. Preliminary MAAP calculations have been performed by GE to support assumptions regarding severe accident containment conditions.[4-80] The accident scenario is a small LOCA in the lower head of the reactor vessel.

As mentioned previously, the NAUA input requires specification of the containment volume, height, radius, and surface areas (including internal structure surfaces where appropriate) for sedimentation, Brownian diffusion and diffusiophoresis. For the SBWR, these were taken respectively as 5324 m^3 (188000 ft^3), 6.1 m, 16.7 m, 875 m^2, 2566 m^2, and 815 m^2. The height and volume are approximately those of the drywell, which is an annular cylinder. The radius is that of a right circular cylinder of the same volume and height. To account for the annular geometry with its inner wall, the vertical wall area (815 m^2) and total surface area (2566 m^2) are those of an annular cylinder whose inner and outer radii are those of the drywell.

ALWR requirements establish that a means to flood the lower drywell prior to or immediately on vessel failure shall be provided. As a result, rapid quenching of the ex-vessel debris and little or no core concrete interaction is assumed to occur. Steam injection rates to the containment were approximately that presented in Table 4-14. Containment atmospheric and wall temperatures were adapted from the preliminary MAAP calculations performed for a bottom drainline break for the SBWR. These temperatures are shown in Table 4-15. Leakage from the containment was assumed to be the Requirements Document specified limit of 0.5%/day. Only drywell leakage was considered in this evaluation. MSIV leakage is discussed in Section 4.10.

A significant feature for the passive BWR is the sweepout of fission products from the drywell and their attenuation in the wetwell during the quenching period following reactor vessel lower head penetration. In the NAUA calculation these phenomena were simulated by introducing an artificial leakage whose rate was chosen to remove the fission products at the same rate as the sweepout-scrubbing process. The "true" leaked mass (that contributes to the source term) during this time interval was determined from the calculated suspended mass concentration and the design leak rate of 0.5%/day.

TABLE 4-14. PASSIVE BWR SEQUENCE TIMING

Time (hours)[1]	Accident Status	Steam Injection Rate, (lb/hr)[2]
0 - 2	Boiloff of reactor inventory from TAF to vessel penetration	--
2 - 2.5	Debris quenching in lower drywell	1.1E5
2.5 - 10	Long term relocation of remaining fuel to lower drywell and steam generation due to decay heat.	3.5E4
10 - 23		2.5E4

(1) t=0 is start of core uncovery. Vessel penetration is t = 2 hrs.
(2) Steam injection rates have been reduced by fraction entering IC.

TABLE 4-15. PASSIVE BWR CONTAINMENT TEMPERATURE

Time (sec)	Containment Temp. (C)	Wall Temp. (C)
0 (core uncovery)	127	77
2300	129	109
7300 (vessel penetration)	154	129
12600	157	157
17300	167	174
22300	176	186
27300	189	198
32300	201	208
37300	212	218
42300	224	228
47300	233	237
52300	244	245
57300	250	252
62300	260	260
67300	268	269
72300	277	277
77300	283	285
82300	288	294
87300	296	302

Fractions of core inventory that are released to containment as a function of time are presented below. These values were developed in Sections 4.1 through 4.8. Derivation of these fractions are presented in Section 2. Again, time zero is placed at the time of core uncovery for convenience in deriving NAUA results.

	0-2 hours	2-23 hours
I	.30	.20
Cs	.23	.18
Te	.06	.02
Sr, Ba	.003	--
Ru	.003	--
Remainder	.00003	--

4.9.3 Results

Table 4-16 presents the 24 hour integrated leaked masses for the passive PWR analysis. Figure 4-9 contains the results of the containment aerosol concentration as a function of time. The 24-hour integrated leaked masses for the passive BWR are shown in Table 4-17. Figure 4-10 shows the suspended aerosol mass as a function of time. In Figure 4-10, the sharp drop in the suspended mass at ~12,000 seconds is due to hygroscopicity effects, in contrast to the PWR case in which they were of no importance, as mentioned earlier. In the BWR calculation, the relative humidity is close to saturation at this time.

4.9.4 Conclusions

Several variations of the base cases were calculated in order to obtain some idea of the sensitivity of the integrated leaked mass to various assumptions made in the analysis. The PWR sensitivity analysis results are summarized in Table 4-18.

In the first variation, an additional contribution to the steaming rate assuming continued Zr oxidation during the 4-4.5 hr period following vessel failure was added. This approximately doubles the total steam injection during that period.

TABLE 4-16. PASSIVE PWR AEROSOL MASS LEAKAGE

	24-Hour Integrated Leaked Mass (gm)
CsOH	46.8
CsI	8.8
Te + TeO_2	1.8
Total	99.0

TABLE 4-17. PASSIVE BWR AEROSOL MASS LEAKAGE

	24-Hour Integrated Leaked Mass (gm)
CsOH	19.2
CsI	3.8
Te + TeO_2	0.6
Total	34.5

TABLE 4-18. PASSIVE ALWR (PWR) AEROSOL REMOVAL SENSITIVITY ANALYSIS TO INPUT ASSUMPTIONS

	24-Hour Integrated Leaked Mass (gm)			
	CsOH	CsI	Te	Total
Base case	46.8	8.8	1.8	99.0
Var. 1 (ex-vessel Zr oxidation)	42.4	8.0	1.6	88.0
Var. 2 (early f.p. release)	49.4	9.4	1.8	105.6
Var. 3 (late f.p. release)	33.8	6.2	1.2	66.8
Var. 4 (no revap.)	29.0	5.8	1.4	72.4
Var. 5 (accelerated revap.)	44.2	8.2	1.6	90.2

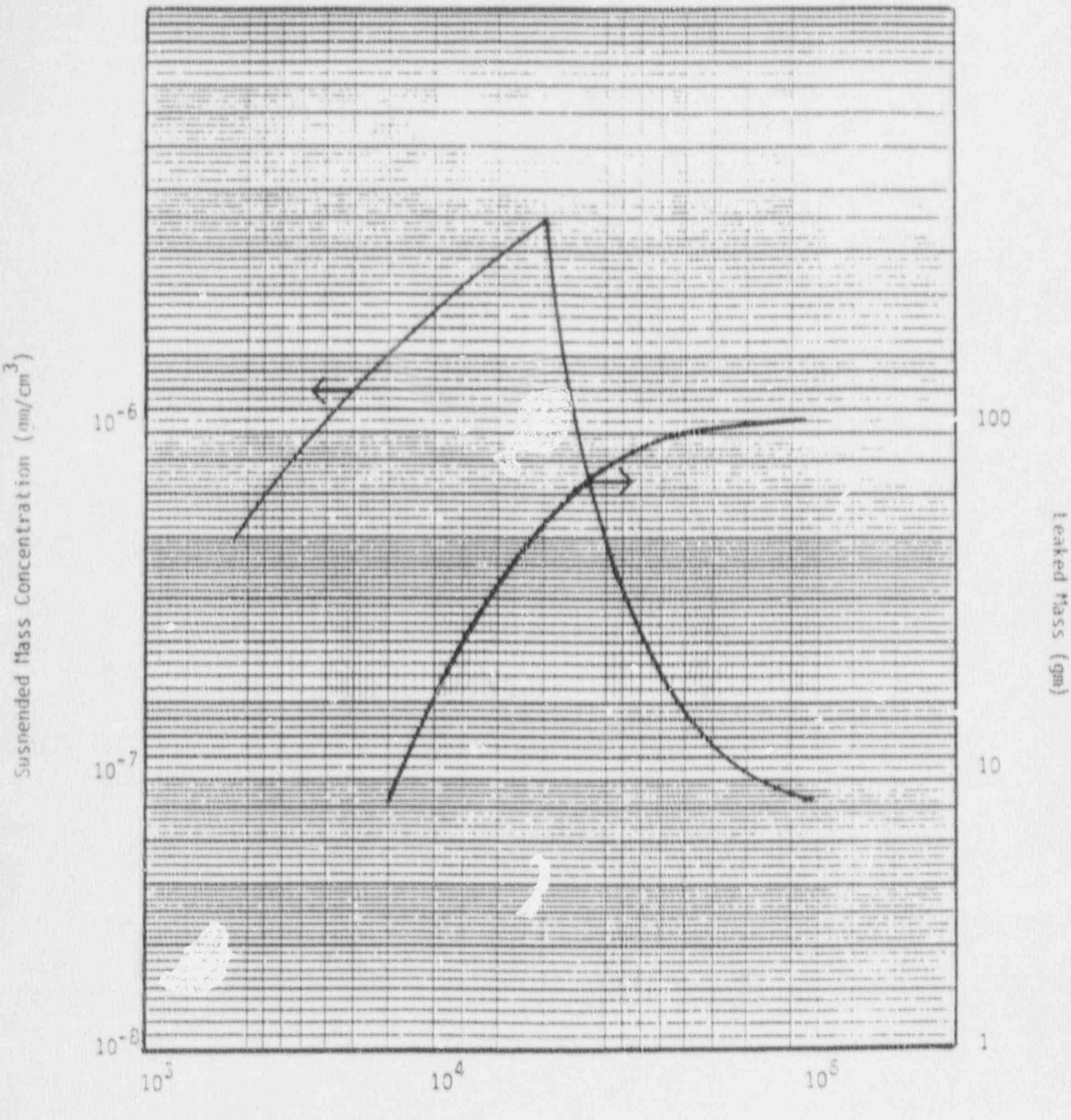


Figure 4-9. Containment Aerosol vs. Time for PWR Medium LOCA, Base Case.

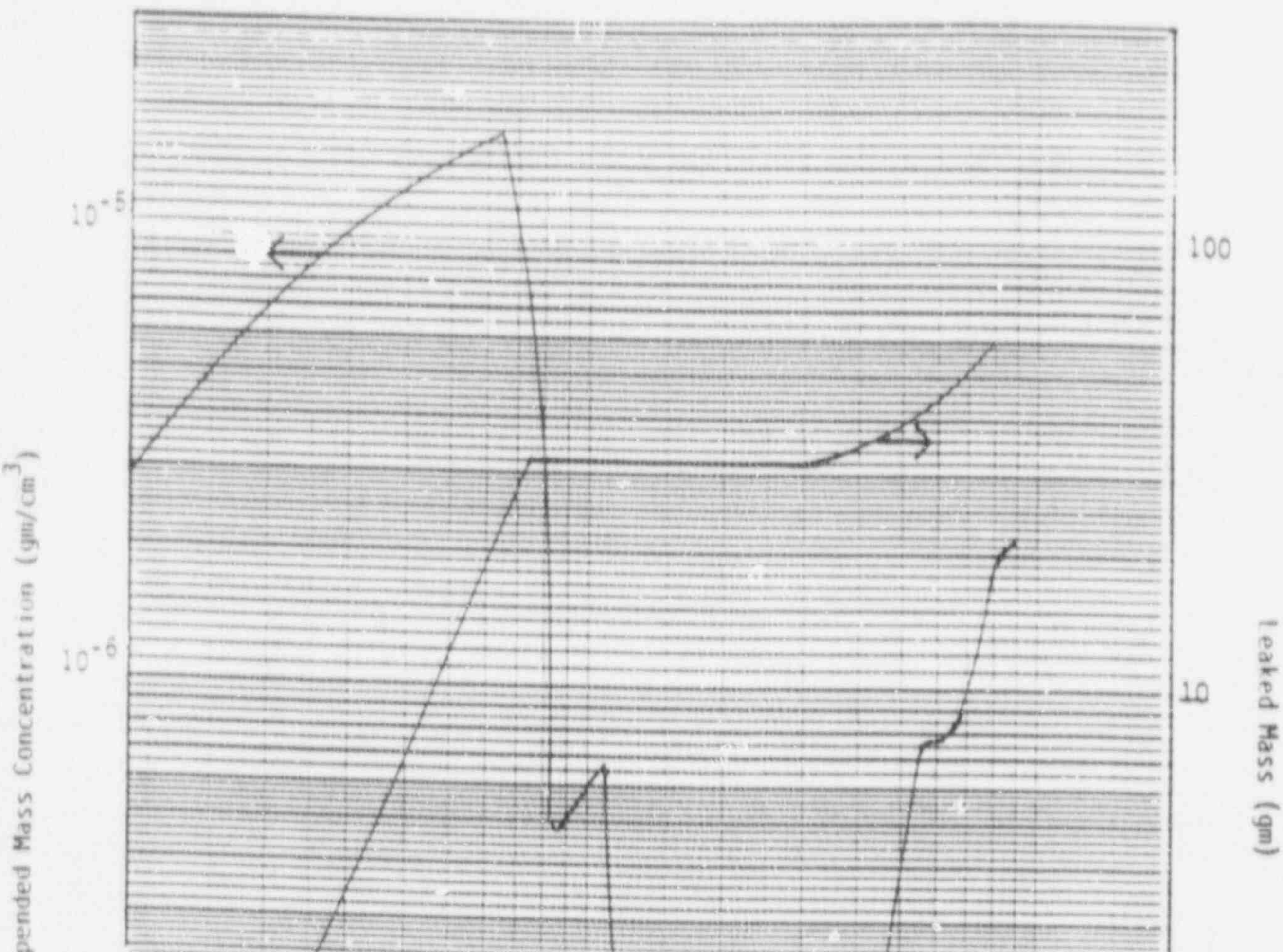


Figure 4-10. SBWR - Suspended Mass Concentration.

In Variation 2 (early fission product release), the in-vessel fission product release during the core degradation period (0-4 hr) was assumed to take place only over the first hour, at a rate four times the base case rate. The additional steam addition assumed for Variation 1 was also included in the analysis.

In Variation 3 (late fission product release), the in-vessel fission product release was assumed to take place over the period 3-4 hr, at a rate four times the base case rate. Again, extra ex-vessel steam addition was also assumed.

Variation 4 assumed zero late in-vessel and revaporization releases from 4 to 23 hours. The additional steam addition from Variation 1 was also included.

Variation 5 assumed that late in-vessel and revaporization releases occurred from 4 to 13.5 hours, at twice the rate of the base case. Additional ex-vessel steam generation was included for this case, as well.

Results of Variation 1, which addresses the sensitivity of the leaked mass to the steaming rate at the time of reactor vessel lower head penetration (if, for example, some of the ex-vessel core debris energy were to go into core-concrete interaction vs. steaming to the containment), indicate that increasing steaming rate by a factor of 2 decreases the leaked mass by only about 10%. Variation 2 (early fission product release) gave the highest leaked masses, while Variation 4 (no late in-vessel release or revaporization) gave the lowest. Overall, the variations illustrate the robust nature of the source term, since the changes in leaked mass are relatively minor.

A set of sensitivity studies was also performed using the BWR data and is summarized in Table 4-19. As noted in Section 4.9.2.2.3 above, the BWR leakage does not include the MSIV leak path. MSIV leakage is discussed in Section 4.10.

TABLE 4-19. PASSIVE ALWR (BWR) AEROSOL REMOVAL SENSITIVITY ANALYSIS TO INPUT ASSUMPTIONS

	24-Hour Integrated Leaked Mass (gm)			
	CsOH	CsI	Te	Total
Base case	19.2	3.8	0.6	34.5
Var. 1 (reduce ex-vessel steam generation)	19.5	3.9	0.6	35.2
Var. 2 (IC retention)	15.0	3.0	0.5	28.9

The first variation assumes a steam injection rate half that of the base case during the quench period following vessel failure. This sensitivity study simulates a smaller degree of quenching of the debris generating steam to sweep out fission products to the suppression pool.

The second sensitivity simulates the potential benefits of the containment isolation condenser in removing fission products from the containment atmosphere during the late release period. This was performed by assuming fission product retention in the isolation condenser was proportional to the steam flow through the IC (~40% of the steam flow rate to containment per MAAP).

As with the PWR sensitivity studies, overall fission product leakage appears to be insensitive to large changes in the steaming rate associated with the quenching of debris following vessel penetration. Also, the aerosol removal through isolation condenser operation is a potentially significant contributor to reduction in fission product leakage that has not been considered as a part of the analysis at this time.

A final point regarding the above primary containment aerosol removal evaluation is the existence of reasonable margin between the integrated leaked mass defined for the physically-based source term and the expected integrated leaked mass given an ALWR accident. This margin results from the following:

- The core damage event selected to develop the source term envelopes the release expected from the important sequences. For example, the integrated release for a small LOCA in the PWR (the highest probability sequence) was evaluated to be about 65g compared to 99g for the selected core damage event.

- The aerosol phenomena involved in the containment aerosol removal evaluation are well understood and the models have been experimentally validated, thus resulting in relatively small uncertainties.

- The containment aerosol removal evaluation is very robust in that the integrated leaked mass result is relatively insensitive to changes in input assumptions such as steaming rate and timing of aerosol source injection.

- The EPRI NAUA single compartment, simple geometric shape assumptions provide margin from the standpoint that more complicated, multi-compartment geometries would tend to promote aerosol removal due to additional surface area and irregular flow paths.

- As noted above for the BWR, no credit has been taken for aerosol/removal in the isolation condenser.

4.10 Fission Product Holdup and Retention in Secondary Structures

4.10.1 Introduction

Previous sections of this report have dealt with the release of fission products from the fuel, the transport of fission products through the RCS and release to the containment atmosphere, the expected chemical forms in the containment atmosphere, and the release from the containment. This section deals with the release paths from the containment and the potential for holdup (delayed release to the environment) and permanent retention of fission products in structures contiguous to or surrounding the containment ("secondary structures").

Since holdup and retention in secondary structures is the last modeling step for the environmental source term, this section also includes a discussion of the final environmental source term and the associated offsite dose calculations. This discussion is for the PWR, but the results are considered generally applicable to the BWR. In addition, the BWR MSIV leakage holdup and retention in the main steam lines and condenser are discussed.

4.10.2 Important Phenomena and Assumptions

Description

The following phenomena and related assumptions are key with respect to the quantification of holdup and retention in the secondary plant structures:

- The previous section treats the aerosol component of the fission product release from the containment. This release includes all radionuclides (and "inert" or non-radioactive aerosols) except for the noble gases and the small fraction of the radioiodine considered to be in gaseous form (discussed in Section 4.7). To determine the contribution from the release of the gaseous component (nobles and iodine) from the containment, a rate established by the concentration in the containment atmosphere and the containment design leak rate is assumed. Derivation of the transient concentration of nobles and gaseous iodine in the containment atmosphere is based on the following:

    - Noble gases and organic iodine are not removed.

    - For the passive PWRs, elemental iodine is removed according to the model presented in Standard Review Plan 6.5.2 which accounts for diffusiophoretic deposition. In accordance with that model, a typical elemental iodine removal coefficient (λ) is 1.7 hr^{-1}, with a maximum decontamination factor (DF) of 200. (Elemental iodine concentration derived in this manner is used and input to the Secondary building analysis of Section 4.10.6.)

    - For the passive BWRs, elemental iodine would be treated in the same way as particulate iodine and should be subjected to suppression pool scrubbing with the same DF. (Refer to Section 4.8 on water pool scrubbing).

- As with the containment discussed in the previous section, the NAUA code (or some other equivalent treatment of aerosols) may be used to calculate the aerosol removal coefficients in secondary structures as a function of time. However, unlike the containment case, hygroscopicity and diffusiophoresis (not expected to be significant) are not credited in secondary structures. Moreover, the particle size distribution from the containment release is changed to remove all condensed water to provide margin in accounting for dry particles; the condensed water is removed mainly because of the low expected humidity within the secondary structure.

- The removal of gaseous fission products is not included in the secondary building retention calculations; only holdup is considered.

- The secondary structure internal partitioning and flowpaths may be taken into account in calculating the holdup and retention, with individual internal, well-mixed control volumes being defined. These control volumes can be defined as "source volumes" (one or more) where containment leakage is introduced into the building as well as "sink volumes" (one or more) with a given exchange rate between the source and sink volumes and between the sink volume and the environment.

- A relatively large exchange rate between the environment and the secondary structure (as well as between control volumes within the secondary structure) should be used to provide margin in assessing offsite doses in order to reduce the need for secondary structure room leaktightness. (Note that large exchange rates may not be conservative for assessing onsite accident management exposures, and accident management needs must also be recognized in the design of the secondary structures). This exchange rate value should take into account the following:

- Seismic design (non-seismic structures are not credited)

- Containment leakage into interior rooms

- Sealed penetrations in rooms with potential containment leakage

- Gaskets in doorways

- Trip for forced ventilation on high radiation signal

- Spring-loaded, bubble-tight dampers in ventilation ducts

- Floor drain provisions

- Demonstration of leakage performance by analysis, pre-operational testing, and/or maintaining negative pressure in the secondary building by operation of the normal ventilation system.

A conservative estimate of an exchange rate which can be used in evaluating the holdup/retention characteristics of a secondary structure would be 200%/day. This assumed value is supported by the discussion presented in Section 4.10.3 below.

- Leakage assumed to bypass the secondary structure is limited to that which leaks through the containment isolation valves in any line penetrating the containment and terminating outside the secondary structure or potentially leaking within the secondary structure in a room which has a door or other penetration (e.g., for ventilation) which communicates directly with the environment. Important corollary assumptions include:

- Negligible leakage through flanged containment penetrations (such as the equipment hatch) which exist outside the secondary structures

- Negligible leakage through the steel containment shell (for the passive PWR, as discussed further in Section 4.10.4 below).

A special case of bypass leakage is that which passes through the steamlines for the passive BWR. The treatment of this leakage is discussed in Section 4.10.5. For all other valve leakage bypassing the secondary structures the following assumptions may be made:

- Valve leakage may be assumed to be at the ASME/ANSI OM-10[4-81] limits of 7.5 SCFD/inch of valve diameter

- Valve leakage leading to bypass may be integrated and then subtracted from the design leak rate - the difference being assumed to enter the secondary structure.

- For certain of the lines bypassing the secondary structure, HEPA filters (with a particulate efficiency of 0.99) may be installed to limit the bypass leakage dose.

Effect on Magnitude of Environmental Source Term

In order to estimate the overall effect of the assumptions listed above on the calculation of secondary building holdup/retention, a preliminary study was performed on AP600 for an example of a passive PWR auxiliary building. This study involved the following:

- The aerosol source term from the containment (both radioactive and inert) is taken from the passive PWR analysis presented in Section 4.9. The total mass leaked in 24 hours was calculated to be approximately 99 grams.

- A review of all containment penetrations was made to determine their points of termination outside containment. All flanged joints and valves were included as potential leak points. Using the definition of secondary building bypass given above, the "bypass" leakage was determined to be about 14 Standard cubic feet per hour (SCFH), or about 4% of the assumed constant leak rate of 0.5%/day (333.3 SCFH).

- The auxiliary building has a total volume of approximately 3.5 x 10^6 ft^3. Excluding the volume which includes containment leak points <u>and</u> which could communicate directly with the environment (about 10% of the total), the remainder of the building is divided into two control volumes; one containing the rest of the containment leak points (about 10% of the remaining volume), and the other mixing with the environment at a rate of 200%/day. The mixing rate between the two volumes is also assumed to be 200%/day (of the smaller volume).

- Aerosol removal was calculated with NAUA as described above. The aerosol removal coefficient (λ) was observed to vary (during the 24 hour period of interest) from about 0.15 hr^{-1} near the start of release to a value about half that at the end of the calculation.

The overall effect of these assumptions on the aerosol release to the environment was observed in this study to be as follows:

- Of the 99 grams released from the primary containment, about 4 grams bypassed the secondary structure. About half of this amount could be filtered prior to release by placing HEPA filters on the shutdown purge supply and exhaust lines.

- Of the remaining 95 grams, the release to the environment during the first 24 hours after the start of release was about 12 grams, or a building DF of about 8 (including both holdup and deposition). For comparison, the DF of the primary containment over the same time interval is about 300 times greater.

It would appear, therefore, that the effect of treating secondary building holdup/retention using the assumptions listed above is to reduce the aerosol release from the containment to the environment by about a factor of six including bypass), with the building providing a "non-bypass" DF of the order of 10.

4.10.3 Mixing Between the Secondary Building and the Environment

In the above assessment of holdup and retention characteristics for a secondary structure, an upper bound mixing value between the building and the environment of 200%/day is utilized. The purpose of this sub-section is to provide additional support for this mixing value.

For seismically-designed structures with gasketed doorways, the potential for exchange between the building and the environment is limited. Such structures are already in existence today at licensed nuclear power plants, and are being used to provide additional mitigative capability for the postulated design basis accidents.

An example of such a structure is the reactor building of the Shoreham Nuclear Power Station. It is typical of reactor buildings used at operating Mark I and Mark II containment BWRs. The major difference between the Mark I/II reactor buildings and the "secondary structures" being discussed in the context of the Passive ALWR is that in the case of the current plant reactor buildings, an active, safety-related, filtered exhaust system is being used to maintain the building at a slightly subatmospheric pressure under accident conditions. Such a system is not planned for use on the passive plants.

The operating plants rely on an active system in order to meet 10CFR100 offsite dose limits for the source term specified in current regulatory guidance. The building "DF" (the ratio of what enters the building to what leaves it) is equal to:

$$1 / (1 - e)$$

where "e" represents the efficiency for the filter used in the active system exhaust. No degree of bypass is acceptable and radionuclide retention from natural removal mechanisms is not addressed in the regulations, leading to the need for subatmospheric building pressure and a filtered exhaust. (It should be noted here that the effective reactor building DF obtained with filtration in currently operating BWR reactor buildings is of the order of 20 - 100, corresponding to the 95 to 99 efficiency of the filters being used. This can be compared with the PWR auxiliary building assessment presented here which establishes an effective DF of about 6.) An important factor in the determination of the need for active filtration in operating BWR reactor buildings (as opposed to dependence on natural depletion mechanisms within the building) is the present requirement that the fission product release from the containment be considered primarily gaseous.

Given that the fission product release from the primary containment is now understood to be principally particulate, it is now possible to demonstrate an effective DF for the Passive Plant secondary structure that is somewhat lower than for the operating BWR reactor buildings, but still significant. Moreover, adequate DFs can be shown without the need for subatmospheric pressures and active filtration. However, the buildings must still be "tight", and much can be learned about the degree of "tightness" achievable from the BWR reactor building experience.

Appendix 2 of this report includes the three technical specifications pertaining to the Shoreham reactor building integrity, 3/4.6.5.1, 3/4.6.5.2 and 3/4.6.5.3. The first defines "integrity" and provides the limiting conditions for operation in terms of the negative pressure that must be maintained by the normal ventilation system, the monitoring of building access-ways and penetrations, and the performance requirements of the safety-related standby ventilation system. The second establishes the operability requirements for the reactor building isolation valves; the third the operability requirements for the reactor building standby ventilation system

(RBSVS). Of these, the first is the one that bears most directly on the question of what can be achieved in terms of building leaktightness.

The RBSVS, with a nominal single-train exhaust flowrate of 1160 cfm, must be capable of achieving (within 120 seconds of the loss of normal ventilation) and maintaining a reactor building negative pressure of 0.25 inches of vacuum water gauge. This pressure is defined such that for any condition of low-to-moderate windspeeds (where χ/Q values and, therefore, offsite doses could be expected to be high), the pressure difference between the outside of the reactor building (even with consideration given to the pressure distribution created by the wind) and the inside of the building can be assured to be negative at all points. In other words, the design negative-pressure operating point of the RBSVS exhaust fans must be such that it will ensure a reactor building negative pressure lower than the most negative pressure created by the wind at or below the windspeeds of concern.

In practice, this means that the forced exhaust flowrate from the building needed to create this very negative pressure would greatly exceed the flow through the building if only the "natural" wind-induced pressure distribution (positive to negative) existed around the structure. Therefore, the design flowrate of a single RBSVS train can be used to characterize the maximum through-flow that would be expected without an active system. In the case of the Shoreham reactor building, this value (as noted above and in Appendix 2) is 1160 cfm. This value was demonstrated in the pre-operational testing of Shoreham.

The Shoreham reactor building is relatively "open" inside, with an internal free volume of the order of 1.5×10^6 ft^3. With a "through-flow" of 1160 cfm, the exchange rate between the building and the atmosphere would be approximately 100%/day.

The Shoreham reactor building is mostly above grade. It includes a truck-access lock as well as several doors and ventilation penetrations. While it is mostly concrete, the upper (refueling floor) area includes gasketed, metal siding and a metal, truss-beamed roof. Given the size, function, and design of this structure, it would appear to be a reasonable analog for secondary structures associated with the passive plants.

Therefore, it would seem that a mixing rate twice that specified and confirmed for the Shoreham reactor building (i.e., 2 x 100%/day or 200%/day for the passive plants) is a reasonable maximum value to be assumed in the evaluation of passive plant secondary structure holdup and retention.

4.10.4 Leakage through Steel Containment Shells

An important issue regarding the effectiveness of secondary structures in mitigating radionuclide releases to the environment is the degree to which primary containment leakage can bypass the secondary structure. In Section 4.10.2 a method to quantify the bypass leakage is described, but it assumes that essentially all bypass leakage is through containment penetrations, not through the body of the containment itself. In the case of the passive PWR the containment structure is a 1.5" thick steel shell. This sub-section provides the basis for assuming that leakage through the steel shell will contribute only a negligible amount to the bypass leakage already considered.

Construction Inspection and Testing

The design leak rate will be 0.5%/day or less, and is referred to as L_a. In Section 3.10.2 it was shown that bypass leakage can be controlled to 4-5% L_a as long as the leakage is not through the steel shell. For purposes of discussion, "negligible" can be viewed as leakage on the order of 20-25% of the bypass leakage already included in the analysis, or approximately 1% L_a. In absolute terms "negligible" bypass leakage would be 0.005%/day or less.

At a design leak rate of 0.5%/day the absolute leak rate would be approximately 0.1 cfs. One percent of that value would be 0.001 cfs (approximately 1.7 cubic inches per second) and would correspond to a leakage area on the order of 0.0005 in^2. While this would seem to be a very small number it is well within the capability of non-destructive examinations to detect (e.g., a 0.010 inch crack nearly 1/16 of an inch long), and would certainly be expected to be found during the construction inspection and testing program.

A number of containment construction inspection programs have used leak chase channel testing of containment pressure boundary welds. These leak chase channels have been placed after completion of non-destructive examination (NDE) of the welds as a means of testing for leakage locally. Occasionally, leakage has been observed during this localized, leak chase channel testing; inevitably when this has been the case the problem has been found in the leak chase channel welds or the testing apparatus, not the containment pressure boundary welds.

In the absence of leak chase channel testing vacuum box testing has been employed to verify leak tightness of containment pressure boundary welds. It is unlikely that a weld defect leaking in the range of one cubic inch per second would pass the required "bubble-tightness" requirement. Additionally, on free-standing steel containments like that of the passive PWR, soap-bubble test of all welds during the structural acceptance test can be performed. Such a test would provide even greater assurance that the passive PWR steel shell is leak-tight.

Review of testing experience with a number of containments yields important insights applicable to the Passive ALWR.[4-82] The bulk of the containments reviewed were of steel-lined, reinforced concrete construction where the liner thickness has been considerably less (of the order of 3/8") than the steel shell thickness contemplated for the passive PWR containment structure. There are two important aspects of that difference:

- The passive PWR shell welds will involve many more passes with intermediate NDE of the weld surface than would be the case for a concrete containment steel liner where surface examinations (dye penetrant and/or magnetic particle) are generally confined to the final cover passes. The likelihood of a continuous weld defect constituting a leak path through the full shell thickness of the passive PWR would be expected to be correspondingly smaller.

- As a stamped ASME MC steel pressure vessel, the passive PWR steel shell would require 100% volumetric examination (radiography) of all pressure boundary welds. Concrete containment steel liners require only a minimum of 2% radiography by length of weld.

Given the above, it is viewed as extremely unlikely that non-negligible leakage through the steel containment shell would occur.

Potential for Plugging of Through-Shell Leakpaths

As noted in Section 4.7 above, the vast majority of the activity suspended in the containment atmosphere will be in aerosol form, and aerosols do not readily pass through pathways that are long relative to hydraulic radius (i.e., "capillaries") and which would typically be described as tortuous, such as a very small weld defect in a 1.5" thick shell. For this reason, even if a non-negligible leak path were to exist, it is expected that plugging would occur. Another effect which should be noted is that the inside of the steel shell will be acting as a condenser, and any capillaries which might exist would tend to fill with water under containment pressure, preventing or greatly limiting gas leakage.

The question of aerosol plugging can be addressed using the Vaughn/Morewitz plugging model that was incorporated into the MAAP code as described in Section 13 of Reference 4-83. The Vaughn/Morewitz plugging model is based on experimental observations that small capillaries passing aerosols with concentrations of 2 g/m^3 and above will become plugged when the total aerosol having entered the passage exceeds a mass defined by the expression Kd^3, where K is an experimentally-determined constant equal to 30 $g/cm^3 \pm 20$ g/cm^3 and d is the capillary diameter in cm. For the representative accident sequence, the aerosol concentration in the containment atmosphere is close to or exceeding the threshold of the range tested (1 to 2 g/m^3 or slightly higher) during the period that the bulk of the 99 grams of aerosol is released from the primary containment (refer to Figure 4-9). In the worst case, if all of the leakage (i.e., the full containment design leakrate of 0.5%/day or a hole of area 0.05 in^2 and 0.25" in diameter) were concentrated in a single leakpath through the steel shell (the most conservative representation), only about 13 grams (or 13%) of the 99 grams released would be needed to plug the leak. (This value was obtained using the upper bound value for K, 50 gm/cm^3). For an opening corresponding to 1% of this total area only 0.013 grams or 0.013% of the release would be needed to plug the leak. For this reason, sustained leakage through small and tortuous leak paths such as those that might result from weld defects in the containment structure is viewed as virtually impossible. Sustained leakage is likely to occur only through leak paths which could properly be characterized as "orifices."

Conclusions

From the material presented above it is evident that the amount of leakage through the ~1.5 inch steel containment shell would be expected to be negligible; i.e., less than 1% of L_a. Even in the extremely unlikely event that gas flowpaths greater than 1% L_a did exist, it is expected that they would be sealed by the flow of condensed water or, if dry, by the passage of aerosol from the containment atmosphere. Therefore, leakage through the steel containment shell is not included in the secondary building holdup and retention assessment presented in Section 4.10.2.

4.10.5 BWR MSIV Leakage

The discussion in sections 4.10.2, 4.10.3, and 4.10.4 apply to secondary structures and bypass pathways. A special case for bypass pathways, due primarily to its large size, is the MSIV leakage in a BWR.

As specified in the Passive ALWR requirements, it is intended that main steam line and main condenser holdup and retention be utilized to reduce the environmental source term from MSIV leakage, and thus, active MSIV leakage control systems would not be necessary. A methodology for crediting fission product aerosol retention and holdup (and gaseous fission product holdup) in the steam lines and main condenser has been developed for operating BWRs. The BWR Owners Group has submitted this analysis to NRC for review. Once the review of this analysis is complete, the passive BWR evaluation will be completed and included in the physically-based source term. This passive BWR evaluation will factor in the methods and results from the BWR Owners Group analysis, and will determine the environmental source term and associated contribution to offsite dose which is expected from MSIV leakage.

4.10.6 Results

Presented above are the basic assumptions underlying a technically sound approach to quantifying the effect of secondary structures on the release of radionuclides to the environment (i.e., the effect on the environmental source term for a representative accident). This subsection describes the application of that approach to the determination of offsite doses associated with the physically-based source term. Two points must be covered; first, the dose calculation methodology, and second, the actual dose calculation.

Dose Calculation Methodology

The dose calculation methodology used for a particular application must be consistent with the requirements of that application. For example, dose calculations done to demonstrate compliance with 10CFR100 must conform with the requirements stated in that rule and should also be performed in accordance with regulatory guidance. In the case of 10CFR100, for example, plume dispersion (as expressed by the instantaneous values of χ/Q) should be calculated in accordance with Regulatory Guides 1.3[4-84] or 1.4[4-85] (BWR and PWR, respectively) or Regulatory Guide 1.145[4-86], as applicable, depending on the availability of site-specific meteorological data. The dose calculations presented below were performed with an assessment of the potential for emergency planning simplification in mind; accordingly, the method described below is consistent with the dose calculation approach used to establish the current emergency planning requirements. Dose calculations for 10CFR100 purposes will be performed by the Plant Designer as part of design certification.

The basis for the current emergency planning requirements is presented in NUREG-0396.[4-87] NUREG 0396 includes calculations of the potential for exceeding specific dose levels (as a function of distance from the plant) for a range of source terms using representative site meteorological data. It was observed in both NUREG 0654[4-88] and NUREG 0396 that the probability of exceeding the protective action guidelines (PAGs) at a distance of 10 miles was less than 50% given a core melt accident (i.e., that the PAGs would not be exceeded for "most" core melts). The fact that the median core melt accident dose was less than the PAGs was identified in NUREG 0654 as one of a number of justifications for the ten mile radius emergency planning zone (EPZ).

The dose calculations supporting NUREG-0396 were performed using the CRAC computer code which was later developed into CRAC2; it is the CRAC2 methodology which has been used in the dose calculations reported below. Dose calculations used to evaluate the potential for emergency planning simplification should be performed using CRAC2 or similar "severe accident" methodologies (e.g., MACCS) which include the following:

- meteorological sampling instead of a fixed χ/Q over a specified time interval,
- a complete inventory of radionuclides, and
- deposition and exposure to ground contamination.

For consistency with the NUREG 0396/NUREG 0654 basis for emergency planning, the reported doses for comparison with the PAGs are median values.

Important assumptions for the offsite dose calculations are as follows.

- A ten-hour duration release (maximum possible for CRAC2) starting at five hours into the event should be used. A comparison of the normalized release profile for this assumed release to the normalized aerosol release profile from the primary containment (from Figure 4-9, appropriate for the release that bypasses the secondary structure) and that for the release from the secondary structure (auxiliary building) mentioned in Section 4.10.2 is presented in Figure 4-11. Shown also on Figure 4-11 is a weighted-average aerosol release profile which recognizes that 3.9 grams of aerosol is calculated to bypass the auxiliary building while 11.6 grams is calculated to pass through the auxiliary building prior to release. It is this weighted-average release profile that should be compared to the ten-hour CRAC2 release profile to justify the use of the 10-hour profile in the dose calculations.

Normalized Aerosol Source Term Release Profiles

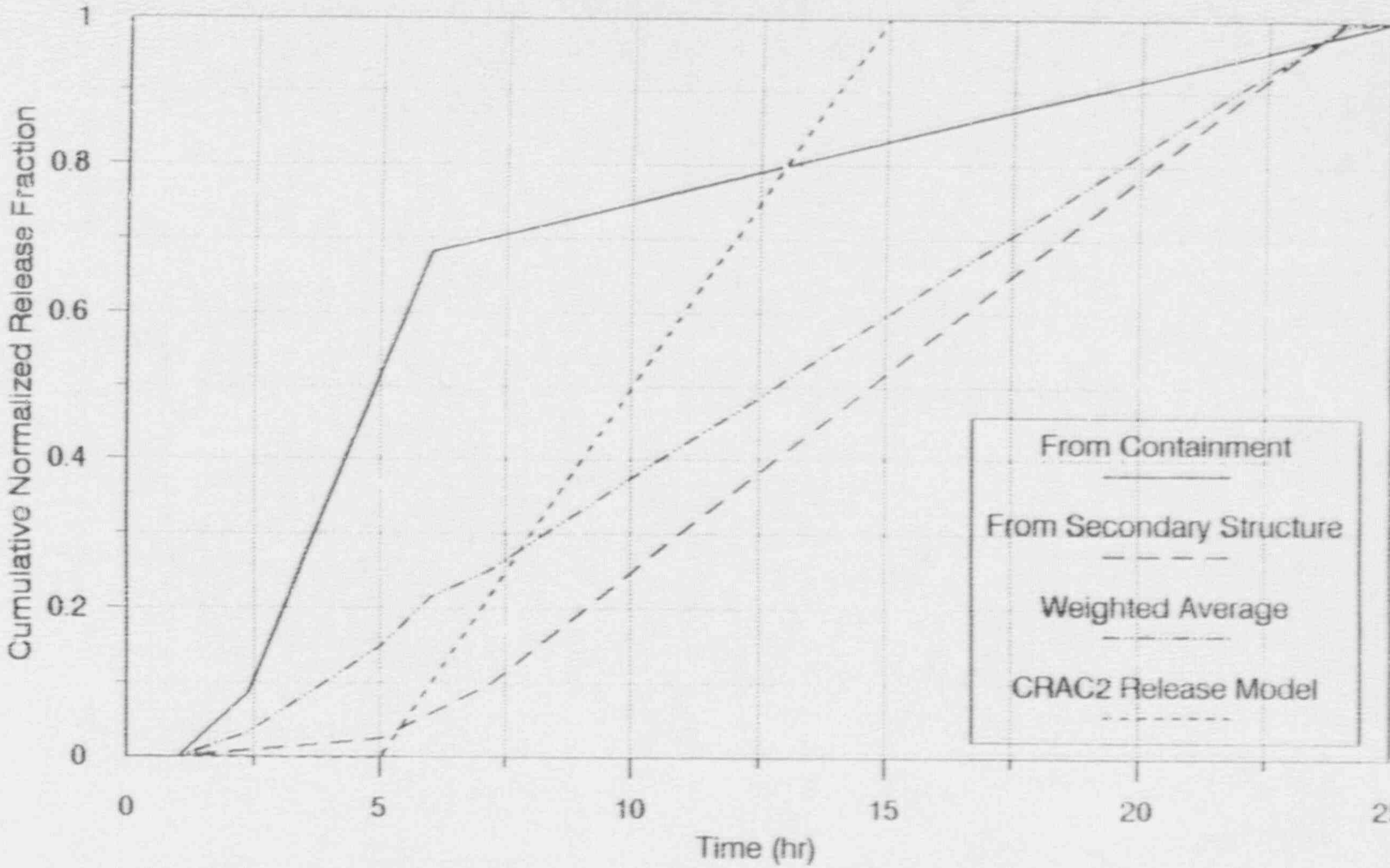


Figure 4-11

- Ground level release (tending to maximize the dose at short distances).

- One cm/sec dry deposition velocity (from NUREG-0396 supporting calculations).

- Breathing rate, ground and cloud shielding factors: same as NUREG-0396 "normal activity", i.e., $2.66 \times 10^{-4} m^3/sec$, 0.33, and 0.75, respectively.

- Doses are calculated assuming a 24-hour exposure to ground contamination.

- Doses are calculated assuming a lifetime dose commitment due to inhalation.

Dose Calculation Results

The following results are preliminary for the passive PWR, calculated using containment leakage of approximately 0.5%/day and the auxiliary building treatment described above.

The preliminary environmental source term (expressed as a percentage of core inventory released during the first 24 hours, beginning one hour after the initiating event) is approximately as follows for the important radionuclides:

- Xe, Kr - 0.1%
- I, Cs - 0.003%
- Te - 0.001%
- Ba, Sr - 0.00003%

For this source term, the whole body dose calculated using the method and assumptions described above is well under the PAGs (i.e., < 1 Rem) at a distance of 1/2 mile from the plant. Accordingly, whole body doses will not be discussed further.

For the thyroid dose two sets of preliminary calculations have been performed; one using meteorology for the ALWR reference site (referred to as "$Site_{Ref}$") and one using meteorology for a site which has been characterized as an "80th percentile" site. The "80th percentile" site (referred to as "$Site_{80}$") was determined by ranking the Regulatory Guide 1.145 "two-hour" χ/Qs for 44 U.S. sites. It is intended that these preliminary calculations will be repeated as the secondary building analyses are improved and updated.

Assuming all leakage were to bypass the auxiliary building, the median thyroid dose at a distance of 1/2 mile for the two sites would be as follows (noting that for such a calculation with no auxiliary building delay, the 10-hour CRAC2 puff is assumed to begin one hour into the event, at the time of the start of the fission product release from the damaged core):

- $Site_{Ref}$ - 35.2 Rem
- $Site_{80}$ - 44.6 Rem

For the case corresponding to the source term given above (0.003% iodine) which includes auxiliary building holdup and retention, the median thyroid dose at a distance of 1/2 mile for the two sites would be as follows:

- $Site_{Ref}$ - 5.3 Rem
- $Site_{80}$ - 6.6 Rem

These data suggest that for containment leak rates at slightly less than 0.5%/day, at most sites the median doses would be below the thyroid PAGs, even for a conservative treatment of secondary structure holdup and retention.

4.10.7 Conclusions

From what has been presented above the following is concluded:

- A technically sound basis for quantifying the holdup and retention characteristics of secondary structures is available. Design-specific calculations need to be performed by the Plant Designer to quantify the actual holdup and retention characteristics for a given standardized design. Secondary structures should be laid out to effect holdup and retention and to prevent the spread of containment leakage (to the extent it occurs) throughout the building.

- By including detailed modeling of holdup and retention characteristics of secondary structures, it is likely that a building decontamination factor of at least 10 can be demonstrated. If careful attention is paid to minimizing building bypass, it is likely that the "effective" DF (that which includes the effects of bypass) should not be less than about 70% of the building DF.

- Special leak rate testing for secondary structures used for retention and holdup is not required to demonstrate adequate leak tightness. As is the case for BWR reactor buildings already in operation, the negative pressure maintained by the normal ventilation system can provide assurance that the required secondary structure leakage limits are being met. This assurance can be increased by periodic (e.g., monthly) inspection of hatches, doorways, and other important isolation features.

- Holdup and retention of radionuclides in secondary structures should be sufficient to assume that, given the maintenance of containment integrity and leak rates at or slightly under 0.5%/day, the median dose will not exceed the PAGs at a distance of 1/2 mile from the plant for a period of 24 hours after the start of the accident.

5. REFERENCES

Section 1.

1-1. Advanced Light Water Reactor Utility Requirements Document, ALWR Passive Plant, Volume III, prepared by Electric Power Research Institute on behalf of ALWR Utility Steering Committee, Rev. 0, August 1990.

1-2. U.S. Atomic Energy Commission, Calculation of Distance Factors for Power and Test Reactor Sites, TID 14844, March 1962.

1-3. Code of Federal Regulations, "Reactor Site Criteria," Title 10, Part 100, January 1, 1988.

1-4. NRC Staff Briefing of Commissioners, October 15, 1990.

1-5. Code of Federal Regulations, "Early Site Permits; Standard Design Certificates; and Combined Licenses for Nuclear Power Plants," Title 10, Part 52, April 18, 1989.

1-6. Staff Requirements Memorandum, June 22, 1990, in response to SECY-90-146.

1-7. U.S. Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150, June 1989.

Section 2.

2-1. Advanced Light Water Reactor Utility Requirements Document, ALWR Passive Plant, Volume III, prepared by Electric Power Research Institute on behalf of ALWR Utilities Steering Committee, Rev. 0, August 1990.

2-2. U.S. Atomic Energy Commission, Calculation of Distance Factors for Power and Test Reactor Sites, TID 14844, March 1962.

Section 3.

3-1. U.S. Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150, June 1989.

3-2. U.S. Nuclear Regulatory Commission, Reactor Safety Study--An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400 (NUREG/75-014), October 1975.

3-3. U.S. Nuclear Regulatory Commission, Special Committee Review of the Nuclear Regulatory Commission's Severe Accident Risks Report (NUREG-1150), NUREG-1420, August 1990.

Section 4.

4-1. W.F. Pasedag, "Iodine Spiking in BWR and PWR Coolant Systems," ANS Proceedings of the Topical Meeting on Thermal Reactor Safety, CONF-770708, 1977.

4-2. W. J. Bailey and S. Wu, Fuel Performance Annual Report, NUREG/CR-3950, Volumes 1 through 5, U.S. Nuclear Regulatory Commission, 1984 through 1989.

4-3. W. J. Bailey and S. Wu, Fuel Performance Annual Report for 1989, NUREG/CR-3950, Vol. 6, U.S. Nuclear Regulatory Commission, 1990.

4-4. R. L. Yang, O. Ozer, and H.H. Klepfer, "Fuel Performance Evaluation for EPRI Program Planning," Light Water Reactor Fuel Performance Topical Meeting, Avignon, France, April 21-24, 1991, to be published.

4-5. U.S. Nuclear Regulatory Commission, Standard Technical Specification for Babcock and Wilcox Pressurized Water Reactors, NUREG-0103, Rev. 4, 1980.

4-6. J. Grover, private communication, Westinghouse, January 1991.

4-7. ANSI/ANS 5.4-1982, "Method for Calculating the Fractional Release of Volatile Fission Products from Oxide Fuel," approved November 10, 1982.

4-8. R.R. Hobbins, D. J. Osetek, and D. L. Hagrman, "In-Vessel Release of Radionuclides and Generation of Aerosols," Proceedings Symposium on Source Term Evaluation for Accident Conditions, Columbus, Ohio, October 28-November 1, 1985, p. 45, International Atomic Energy Agency, Vienna, Austria, 1986.

4-9. T.K. Campbell, R. J. Guenther, and E. D. Jenson, "Volatile Fission Product Distributions in LWR Spent Fuel Rods," Ceramic Transactions, 9, Nuclear Waste Management III, G. B. Mellinger, editor, p. 409, The American Ceramic Society, Westerville, Ohio, 1990.

4-10. J.A. Kuszyk, Hot Cell Examination of Surry Three- and Four-Cycle 17x17 Demonstration Fuel, Vol. 1, Principal Results and Evaluation, WCAP-10514, Vol. 1, DOE/ET 34014-14, Westinghouse Electric Corporation, June 1984.

4-11. G.M. Bain, W. A. McInteer, and T. P. Papazoglou, "Release and Migration of Fission Products in High Burnup Fuel," Proceedings of the American Nuclear Society Topical Meeting on Light Water Reactor Fuel Performance, Orlando, Florida, April 21-24, 1985, Vol. 2, p. 4-1.

4-12. S.R. Pati and A. M. Garde, "Fission Gas Release from PWR Fuel Rods at Extended Burnups," Proceedings of the American Nuclear Society Topical Meeting on Light Water Reactor Fuel Performance, Orlando, Florida, April 21-24, 1985, Vol. 2, p. 4-19.

4-13. M.G. Balfour, E. Roberts, E. DeQuidt, and P. Blanc, Zorita Research and Development Program: Vol. 1, Final Report, WCAP-10180, Vol. 1, Westinghouse Electric Corporation, September 1982.

4-14. D.A. Baker, W.J. Bailey, C.E. Beyer, F.C. Bold, and J.J. Tawil, Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors, NUREG/CR-5009, U.S. Nuclear Regulatory Commission, February 1988.

4-15. D.W. Akers, E.L. Tolman, P. Kuan, and D.W. Golden, "Three Mile Island Unit 2 Fission Product Inventory Estimates," Nuclear Technology, 87, 205, 1989.

4-16. D.W. Akers, and R.K. McCardell, "Fission Product Partitioning in Core Materials," Nuclear Technology, 87, 264, 1989.

4-17. D.A. Petti, J.P. Adams, J.L. Anderson, and R.R. Hobbins, "Analysis of Fission Product Release Behavior from the Three Mile Island Unit 2 Core," Nuclear Technology, 87, 243, August 1989.

4-18. R.R. Hobbins, A.W. Cronenberg, S. Langer, D.E. Owen, and D.W. Akers, "Insights on Severe Accident Chemistry from TMI-2," Proceedings of the American Chemical Society Symposium on Chemical Phenomena Associated with Radioactivity Released During Severe Nuclear Plant Accidents, Anaheim, California, 1986, Rept. NUREG/CO-0078, June 1987, pp. 4-1 to 4-20.

4-19. R.R. Hobbins, D.J. Osetek, D.A. Petti, and D.L. Hagrman, "Fission Product Release as a Function of Chemistry and Fuel Morphology," International Seminar on Fission Product Transport Processes in Reactor Accidents, International Center for Heat and Mass Transfer, Dubrovnik, Yugoslavia, May 22-26, 1989.

4-20. A.D. Knipe, S.A. Ploger, and D.J. Osetek, PBF Severe Fuel Damage Scoping Test -Test Results Report, EG&G Idaho Rept. NUREG/CR-4683, EG&G-2413, March 1986.

4-21. Z.R. Martinson, D.A. Petti, and B.A. Cook, PBF Severe Fuel Damage Test 1-1 Test Results Report, EG&G Idaho Rept. NUREG/CR-4684, EGG-2463, Volume 1, October 1986.

4-22. Z.R. Martinson, M. Gasparini, R.R. Hobbins, D.A. Petti, C.M. Allison, J.K. Hohorst, D.L. Hagrman, and K. Vinjamuri, PBF Severe Fuel Damage Test 1-3 Test Results Report, EG&G Idaho Rept. NUREG/CR-5354, EGG-2565, January 1990.

4-23. D.A. Petti, Z.R. Martinson, R.R. Hobbins, C.M. Allison, E.R. Carlson, D.L. Hagrman, T.C. Cheng, J.K. Hartwell, K. Vinjamuri, and L.J. Siefken, PBF Severe Fuel Damage Test 1-4 Test Results Report, EG&G Idaho Rept., NUREG/CR-5163, EGG-2542, February 1989.

4-24. D.J. Osetek, "Results of the Four PBF Severe Fuel Damage Tests," Transactions of the Fifteenth Water Reactor Safety Information Meeting, Gaithersburg, Maryland, Rept. NUREG/CP-0090, 1987, pp. 20-15 to 20-16.

4-25. R.R. Hobbins, D.J. Osetek, D.A. Petti, and D.L. Hagrman, "The Influence of Chemistry on Severe Accident Phenomena in Integral Tests," Proceedings of the Second Symposium on Nuclear Reactor Severe Accident Chemistry, American Chemical Society, Toronto, Canada, June 5-11, 1988.

4-26. R.R. Hobbins, D.J. Osetek, D.A. Petti and D.L. Hagrman, "The Influence of Core Degradation Phenomena on In-Vessel Fission Product Behavior During Severe Accidents," Proceedings of the International ANS/ENS Conference on Thermal Reactor Safety, Avignon, France, October 2-7, 1988.

4-27. R.R. Hobbins, M.L. Russell, C.S. Olsen, and R.K. McCardell, "Molten Material Behavior in the Three Mile Island Unit 2 Accident," Nuclear Technology, 87, December 1989.

4-28. R.A. Lorenz, E.C. Beahm, and R. P. Wichner, "Review of Tellurium Release Rates from LWR Fuel Elements under Accident Conditions," Proceedings of the International Meeting on Light Water Reactor Severe Accident Evaluation, ANS publication number 700085, Vol. 1, 1983, pp. 4.4-1 to 4.4-9.

4-29. E.C. Beahm, Tellurium Behavior in Containment Under Light Water Accident Conditions, NUREG/CR-4338, February 1986.

4-30. H. Jordan and M.R. Kuhlman, TRAP-MELT2 User's Manual, NUREG/CR-4205 May 1985.

4-31. Fauske and Associates, Inc., MAAP Modular Accident Analysis Program User's Manual, Vols. I and II, IDCOR Technical Report 16.2-3, February 1987.

4-32. F.J. Rahn, J. Collen and A.L. Wright, "Aerosol Behavior Experiments on Light Water Reactor Primary Systems," Nuclear Technology, 81, May 1988.

4-33. D.R. Dickson et al., Aerosol Behavior in LWR Containment Bypass Piping - Results of LACE Test LA3, LACE-TR-011, July 1987.

4-34. "The Marviken Aerosol Transport Tests," Fifth Series, Joint Reactor Safety Experiments in the Marviken Power Station, Sweden, Results from Test 1, MX5-59, February 1984.

4-35. M.L. Carbonneau et al., Experiment Analysis and Summary Report for OECD LOFT Project Fission Product Experiment LP-FP-2, OECD LOFT-T-3806, June 1989.

4-36. U.S. Nuclear Regulatory Commission, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150, June 1989.

4-37. R.S. Denning et al., Radionuclide Release Calculations for Selected Severe Accident Scenarios, Battelle Columbus Laboratories, NUREG/CR-4624, BMI-2139, July 1986.

4-38. I.K. Hohorst et al., SCDAP/RELAPS/Mod 2 Code Manual, Volume 4: MATPRO-A Library of Materials Properties for Light-Water-Reactor Accident Analysis, NUREG/CR-5273, Vol. 4, February 1990.

4-39. R.M. Elrick and R.A. Sallach, "Fission Product Chemistry in the Primary System," Proceedings of the International Meeting on Light Water Reactor Severe Accident Evaluation, ANS publication number 700085, Vol. I, 1983, pp. 4.6-1 to 4.6-5.

4-40. R.M. Summers et al., "MELCOR In-Vessel Modeling," Proceedings of the Fifteenth Water Reactor Safety Information Meeting, Gaithersburg, Maryland, NUREG/CP-0091, February 1988.

4-41. The Industry Degraded Core Rulemaking Program, Peach Bottom Atomic Power Station - Integrated Containment Analysis, IDCOR Technical Report 23.1PB, March 1985.

4-42. D.A. Powers, J.E. Brockmann, and A.W. Shiver, VANESA: A Mechanistic Model of Radionuclide Release and Aerosol Generation During Core Debris Interactions with Concrete, NUREG/CR-4308, July 1986.

4-43. J. Fink (Argonne National Laboratory), ACE Technical Advisory Committee Meeting, Palo Alto, CA, October 1990.

4-44. U.S. Nuclear Regulatory Commission, Technical Bases for Estimating Fission Product Behavior During LWR Accidents, NUREG-0772, June 1981.

4-45. D. Cubicciotti and B.R. Sehgal, "Vapor Transport of Fission Products in Postulated Severe Light Water Reactor Accidents," Nuclear Technology, 65, 1984, pp. 266-291.

4-46. L. Baker, Jr., J.K. Fink, R. Simms, B.J. Schlenger, and J.E. Herceg, Source Term Experiments Project (STEP): A Summary, EPRI report NP-5753M, March 1988.

4-47. D.J. Osetek and R.R. Sherry, "Analysis of Fission Product Transport Behavior During Severe Fuel Damage Experiments," Proceedings of the Symposium on Chemical Phenomena Associated with Radioactivity Releases During Severe Nuclear Plant Accidents, NUREG/CP-0078, June 1987, p. 4-85.

4-48. J.L. Collins, M.F. Osborne, R.A. Lorenz, and A.P. Malinauskis, "Fission Product Iodine and Cesium Release Behavior Under Light Water Reactor Accident Conditions," Nuclear Technology, 81, 1988, pp. 78-94.

4-49. D.J. Wren, R.K. Rondeau, and M.D. Pillow, Studies on the Radiation Stability of Cesium Iodide, EPRI report NP-6344, April 1989.

4-50. R.M. Elrick, R.M. Merrill, and A.L. Ouellette, The Thermal Instability of Cesium Iodide, NUREG/CR-5155 (SAND 88-1187), August 1989.

4-51. B.R. Bowsher, D.M. Bruce, and S. Dickinson, "The Interaction of Cesium Iodide with Boric Acid in the Temperature Range 100 to 1000°C," Specialists Meeting on Iodine Chemistry in Reactor Safety, AERE Harwell, U.K., September, 1985.

4-52. K.J. Laidler, Chemical Kinetics, McGraw-Hill, New York, 1950, p. 51.

4-53. T.M. Bessman, SOLGASMIX-PV, A Computer Program to Calculate Equilibrium Relationships in Complex Chemical Systems, ORNL/TM-5775, Oak Ridge National Laboratory, 1977.

4-54. P.N. Clough and H.C. Starkie, "A Review of the Aqueous Chemistry and Partitioning of Inorganic Iodine Under LWR Severe Accident Conditions," European Applied Research Reports, 6, no. 4, 1985, pp. 631-776.

4-55. P.W. Marshall, J.B. Lutz, and J.L. Kelly, "Gamma Radiation Effects on Time-Dependent Iodine Partitioning," Nuclear Technology, 71, 1987, pp. 400-407.

4-56. E.C. Beahm, W.E. Shockley, C.F. Weber, S.J. Wisby, and Y.M. Wang, Chemistry and Transport of Iodine in Containment, NUREG/CR-4697, September 1986, p. 15, p. 23.

4-57. M. Lucas, "Radiolysis of Cesium Iodide Solutions at Conditions Prevailing in a Pressurized Water Reactor Severe Accident," Nuclear Technology, 82, 1988, pp. 157-161.

4-58. E.C. Beahm, Y.M. Wang, S.J. Wisby, and W.E. Shockley, "Organic Iodine Formation During Severe Accidents in Light Water Nuclear Reactors," Nuclear Technology, 78, 1987, pp. 34-42.

4-59. J.N. Lutz and J.L. Kelly, "The Effects of Organic Impurities on the Partitioning of Iodine," Nuclear Technology, 80, 1988, pp. 431-442.

4-60. J.K. Linacre and W.R. Marsh, The Radiation Chemistry of Heterogeneous and Homogeneous Nitrogen and Water Systems, British report AERE-R-10027, June 1981.

4-61. L.S. Nelson et al., The Behavior of Reactor Core Simulant Aerosols During Hydrogen/Air Combustion," Sandia National Laboratory report SAND85-1817C, 1985.

4-62. E.C. Beahm, W.E. Shockley, and C.F. Weber, "Chemistry and Transport of Iodine in Containment," Proceedings of an International Symposium on Source Term Evaluation for Accident Conditions, IAEA, Vienna, 1986.

4-63. W.C.H. Kupferschmidt et al., The Effects of Steam on the Oxidation of Cesium Iodide Aerosol During Hydrogen Combustion, ACE draft report ACE-TR-B8, October 1990.

4-64. E.C. Beahm, private communication, Oak Ridge National Laboratory, October 1990.

4-65. A.K. Postma and R.W. Zavadowski, Review of Organic Iodide Formation Under Accident Conditions in Water-Cooled Reactors, Battelle Report BNWL-B-213, October 1972.

4-66. P.C. Owczarski, R.I. Schreck, and W.K. Winegardner, SPARC5: A Code for Calculating Aerosol Particle Capture in Suppression Pools, NUREG/CR-4285, 1985.

4-67. A.T. Wassel, A.F. Mills, and D.C. Bugby, "Analysis of Radionuclide Scrubbing in Water Pools," Nuclear Engineering & Design, 90, 1985, pp. 87-104.

4-68. S.A. Ramsdale, BUSCA-Jun 90 Reference Manual, SRD-R542, 1990.

4-69. General Electric Co., "Suppression Pool Scrubbing Tests," GESSAR II, Attachment A to Appendix 15D.

4-70. M.R. Kuhlman et al., "Scrubbing of Fission Product Aerosols in LWR Water Pools under Severe Accident Conditions," Proceedings of the IAEA Symposium on Source Term Evaluation for Accident Conditions, IAEA, Vienna, Austria, March, 1986.

4-71. J.D. McCormack, D.R. Dickinson, and R.T. Alleman, "Experimental Results of ACE Vent Filtration, Pool Scrubber Tests AA1-AA4 and DOP1-DOP5," ACE-TR-A1, Advanced Containment Experiments Consortium (Coordinated by EPRI, Palo Alto, CA), 1989.

4-72. K. Hashimoto, private communication, JAERI, 1990.

4-73. R. Passalaqua, SPARTA Project: SPARTA Test Matrix review and SPARC Code Prediction Calculations, Technical Report ITS4B 87063, ENEA-NEEC, Italy, December 1987.

4-74. M. Furrer et al., Risultati del primo test con immissione di aerolsol e delle prove termoidrauliche condotte sull'impianto SPARTA, Technical Report 2TS4B 90002, ENEA-NDAP, Italy, May 1990.

4-75. A.T. Wassel et al., "Aerosol Scrubbing at Water Pool Injection Sites," ASME National Heat Transfer Conference, August 1987.

4-76. P.C. Owczarski, private communication, Battelle Northwest Laboratory, May 1990.

4-77. K.K. Murata et al., Users Manual for CONTAIN 1.1 a Computer Code for Severe Accident Containment Analysis, NUREG/CR-5026, SAND 87-2309, 1987.

4-78. H. Bunz, M. Koyro, and W. Schock, NAUA-Mod 4: A Code for Calculating Aerosol Behavior in LWR Core Melt Accidents, KfK-3554, Kernforschungszentrum Nuclear Research Center, Karlsruhe, Germany, August 1983.

4-79. Westinghouse AP600 Plant Description Report, DE-AC03-86SF16038, January 1989.

4-80. General Electric Presentation to ARSAP on SBWR Response to Severe Accidents, December 1990.

4-81. ASME/ANSI OM-10, Appendix C, "Inservice Testing of Values," Section 4.2.2.3E(2), 1990.

4-82. P. Ward, Internal Correspondence: P. Ward to J. Metcalf, Stone and Webster Engineering Corporation, November 5, 1990.

4-83. The Industry Degraded Core Rulemaking Program, Technical Support for Issue Resolution, IDCOR Technical Report 85.2, July, 1985.

4-84. U.S. Nuclear Regulatory Commission, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Regulatory Guide 1.3, June 1974.

4-85. U.S. Nuclear Regulatory Commission, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Regulatory Guide 1.4, June 1974.

4-86. U.S. Nuclear Regulatory Commission, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, February 1983.

4-87. U.S. Nuclear Regulatory Commission, Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants, NUREG-0396, December 1978.

4-88. U.S. Nuclear Regulatory Commission, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, NUREG-0654, November 1980.

APPENDIX 1

CORE DAMAGE EVENT FOR PURPOSES OF QUANTIFYING THE PHYSICALLY-BASED SOURCE TERM

APPENDIX I CORE DAMAGE EVENT FOR PURPOSES OF QUANTIFYING THE PHYSICALLY-BASED SOURCE TERM

A.1 Introduction

As noted in Section 1 to this report, the physically-based source term is based on evaluation of representative core damage events which are defined conservatively for the purpose of quantifying the source term. The purpose of this Appendix is to define the characteristics of the representative core damage events and establish the technical basis for these characteristics given plant design features established in the Passive ALWR Requirements Document.

The process used to define the core damage events is conceptually illustrated in Figure A-1. It begins with the identification of all potential functional sequence types which are postulated to be applicable to light water reactor designs including those which could fail containment due to the effects of the damaged core or which could bypass containment for reasons independent of core damage. Passive ALWR design requirements have been developed to address both types of containment challenges.[A-1] Factoring in these design requirements, this complete set of functional sequence types can be screened to determine which are most appropriate for selection in the evaluation of severe accident containment performance and in defining the physically-based source term. Representative accident sequences associated with selected functional sequence types are then identified and used as the basis for defining the characteristics of the core damage event. It is important that the characteristics of the representative sequences be defined so as to result in a source term which envelopes that from accident sequences associated with the selected functional sequence types.

Using the process outlined in Figure A-1, the representative sequence selected for the purpose of establishing the Passive ALWR source term is a core melt at low RCS pressure into an intact containment where adequate debris cooling and decay heat removal occur. This Appendix summarizes the plant design features and operating characteristics contained in the Utility ALWR Requirements Document and credited in demonstrating the applicability of this representative core damage event in defining the source term.

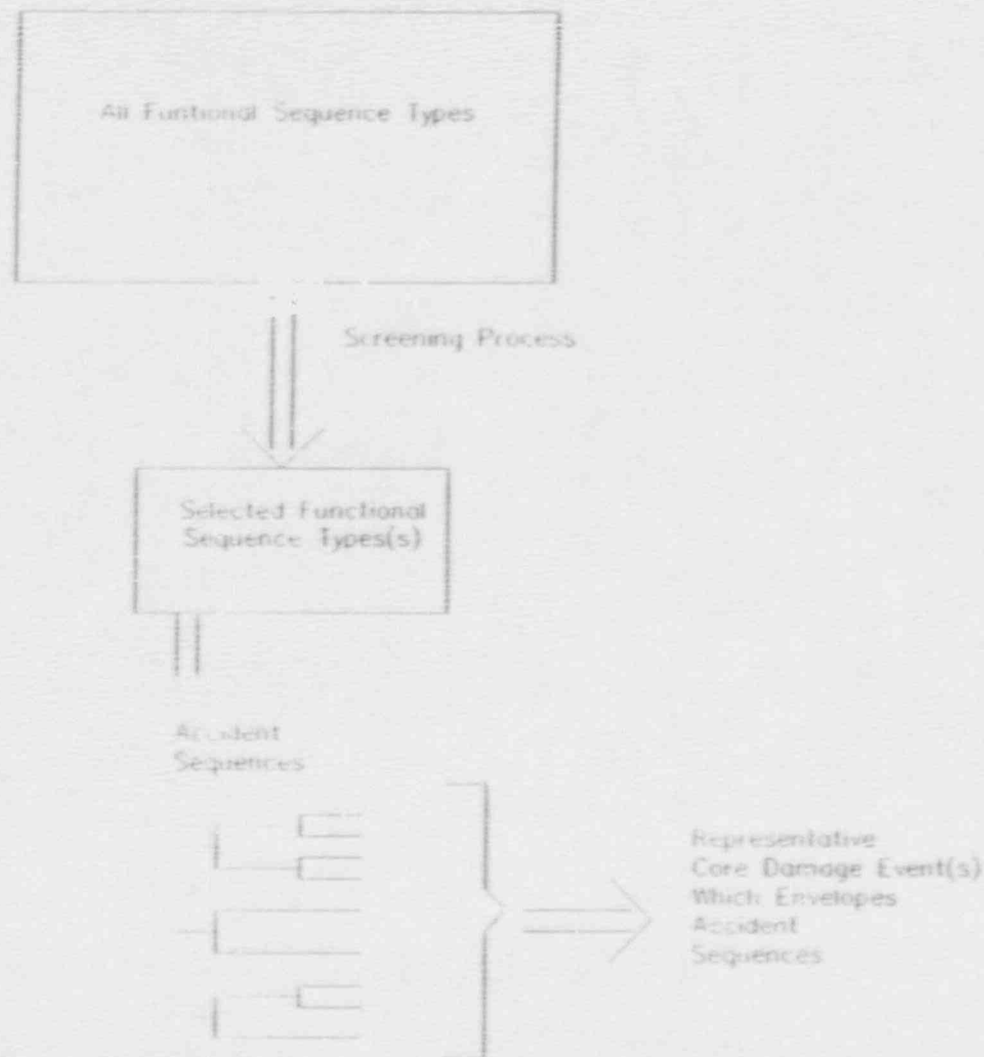


Figure A-1. Process for Defining Core Damage Event for Purposes of Quantifying Physically-Based Source Term.

It should be noted that the actual Passive ALWR designs may go beyond the ALWR requirements, perhaps reducing low pressure core melt scenarios to probabilistically insignificant levels. However, even if screening of functional accident sequence types for specific designs were to eliminate all sequence types based on additional design features, ALWR requirements still establish low pressure core damage events as an accident sequence which, as a minimum, must be addressed by the Plant Designer in evaluating severe accident containment performance and source term.

A.2 Evaluation of Functional Sequence Types Against Requirements

Table A-1 provides a list of nearly more than twenty potential challenges to a light water reactor containment which potentially threaten its integrity. This list of containment challenges encompasses initiating events which by definition result in bypass of containment, random system and equipment failures which may compromise containment independent of the reasons for core damage and potential dependent failures that could be caused by phenomena resulting from a severe accident. The list is broken into two groups. The first 12 challenges are those which might precede or occur for reasons other than the effect of the core damage event. The remaining challenges are associated with severe accident phenomena and would occur as a result of the effects of the core damage events. Listed with the challenges are Passive ALWR requirements that are directed at limiting the potential for the challenge, accommodating the challenge should it occur, or both. A number of Passive ALWR design requirements exceed capabilities for the current generation of light water reactors and are noted with an asterisk in the table.

The various challenges presented in Table A-1 can be grouped under a number of functions important in assuring the integrity of the containment during severe accident events. These functions, presented in Table A-2, are those established in Chapter 5 of the ALWR Requirements Document as being important to the prevention and mitigation of severe accident challenges. The following discussion provides a summary of specific utility imposed ALWR design requirements which accomplish each of these functions thereby assuring integrity of containment under postulated accident conditions.

TABLE A-1. POTENTIAL CONTAINMENT FAILURE MODES THAT ARE INDEPENDENT OF OR PRECEDE CORE DAMAGE

Challenge	Timing	Design	Limit Potential for Challenge	Accommodate Challenges
Containment Isolation	Short Term	PWR/BWR	*Minimal fluid line penetrations Configuration per standards *Valve Configuration Position Ind/Locks *Closed System integrity during severe accident *Fail safe or DC powered isolation valves	*Passive Residual Heat Removal (RHR) fail safe
Interfacing System LOCA	Short Term	PWR/BWR	Minimal Interfaces between the Reactor Coolant System (RCS) and low pressure system interlocks Double Isolation Optimize Testing	Pressure Relief *Pressure Rating
Steam Jet/Pipe whip	Short Term	PWR/BWR	Design and ISI in accordance with ASME BPV Code *Leak Before Break	Missile Protection Separation
Multiple Steam Generator Tube Rupture	Short Term	PWR	Improved water chemistry Materials *Design to prevent erosion, corrosion, cracking, sludge collection	*Passive RHR *Automatic Depressurization System (ADS)
ATWS	Short Term	BWR	Diverse Reactor Protection System (RPS) *Diverse Control Rod Drive Mechanisms (CRDMs)	Standby Liquid Control (SLC) *Checkerboard layout
ATWS	Short Term	PWR	*Diverse RPS	Borated Safety Injection (SI) *Negative moderator temperature coefficient
Suppression Pool Bypass	Short Term	BWR	Vacuum Breakers: Evaluate Loads, Position Indication, Minimal leakage *No high energy lines in wetwell airspace	ADS

TABLE A-1. (Continued) POTENTIAL CONTAINMENT FAILURE MODES THAT ARE INDEPENDENT OF OR PRECEDE CORE DAMAGE

Challenge	Timing	Design	Limit Potential for Challenge	Accommodate Challenges
Catastrophic RPv Failure	Short Term	PWR/BWR	Ductile Materials *No welds in beltline region Relief Capacity	
Excessive Vacuum	Long Term	BWR		Vacuum Breakers
		PWR		Design for external pressure loads
Turbine Missiles	Short Term	PWR/BWR	Turbine overspeed protection Turbine disk integrity	Separation of Systems Turbine orientation or Missile protection
Tornado and Tornado Missiles	Short Term	PWR/BWR	Conformance with ANSI 2.12 and ANSI 51.5	*Passive systems located within containment
Seismic	Short Term	PWR/BWR		SSE at .3g Evaluation at >SSE

* Passive Plant design features which exceed requirements for current LWRs.

TABLE A-1. POTENTIAL CONTAINMENT FAILURE MODES RESULTING FROM CORE DAMAGE

Challenge	Timing	Design	Limit Potential for Challenge	Accommodate Challenges
High Pressure Melt Ejection (HPME)	Short Term	BWR	Diverse Depressurization Systems *Passive RHR	Suppression Pool Inerted containment (no combustion heat addition)
		PWR	Diverse Depressurization System	*Cavity Configuration to entrain core debris
Hydrogen Generation to Detonatable Limits	Short Term	BWR	Inerted	
		PWR	Limit H_2 generation *Depressurization Systems *Cavity Flooding *Containment Size	
Hydrogen Deflagration	Short Term	BWR	Inerted	
		PWR		*Containment Size
In-Vessel Steam Explosion		BWR/PWR	Phenomena limited in probability	RPV Capability
Ex-vessel Steam Explosion	Short Term	BWR/PWR	Phenomena limited in probability	*Rugged lower drywell reactor cavity
Core Concrete Interaction NonCondensible Gas Generation Base Mat Penetration Vessel Support	Long Term	BWR	*Coolable lower drywell geometry *Lower Drywell Flooding *Water addition from sources external to containment	Containment Size
		PWR	*Coolable reactor cavity geometry *Overflow from Incontainment Refueling Water Storage Tank (IRWST) *Water addition from sources external to containment	Containment Size
Sump line Failure	Short Term	BWR/PWR	Sump Configuration *Reactor Cavity/Lower Drywell Flooding	

TABLE A.1 (Continued) POTENTIAL CONTAINMENT FAILURE MODES RESULTING FROM CORE DAMAGE

Challenge	Timing	Design	Limit Potential for Challenge	Accommodate Challenges
Liner Melt Through	Short Term	BWR/PWR	*Liner protected by concrete *Lower Drywell Flooding	
Decay Heat Generation	Long Term	BWR	Main Condenser *Reactor Water Cleanup System *Passive RHR	
		PWR	Steam Generators/Main Feedwater (MFW)/Backup Feedwater Reactor Shutdown Cooling *Passive Containment Cooling	*Passive Heat Removal through containment shell without PCCS tank
Tube Rupture from Hot Gases	Short Term	PWR	*Steam Generators/MFW/Backup Feedwater *Depressurization System	

* Passive Plant design features which exceed requirements for current LWRs.

TABLE A-2. LIGHT WATER REACTOR FUNCTIONS

Reactivity Control
Reactor Pressure Control
Fuel/Debris Cooling
Containment Pressure/Temperature Control
Combustible Gas Control
Containment Isolation
Containment Bypass

A.2.1 ALWR Requirements

Reactivity Control

Failure to scram accident sequences are assumed to result in high pressure challenges to the RCS and the containment. Therefore, the ALWR Requirements Document includes provisions to assure that such failures are prevented and that there are effective means to mitigate the failure to scram in the unlikely event it occurs.

- In the BWR, design features to prevent ATWS include the RPS and hydraulic control rod drives. In addition, an Auxiliary Rod Injection (ARI) System is provided to assure diversity in the electrical portion of the RPS. Beyond existing regulatory requirements, the Passive BWR also provides independent means of rod insertion in the form of electrically driven motor drives. Still further diversity is provided in the reactivity control function through the operation of SLC. Many of these functions (such as hydraulic drive insertion and SLC injection) are passive in nature requiring no continuously operating components and limited dependence on support systems.

- In the PWR, multiple means of rod insertion are provided in the form of gravity insertion into the PWR, which provide an alternative to the motor drives. The PWR also has the potential for riding out an ATWS through implementation of a negative moderator coefficient throughout the entire operating cycle and use of steam generators and backup feedwater to remove the power being generated in the core. Means of shutting down the reactor independent of control rods are provided by Chemical Volume and Control System (CVCS) and if necessary borated passive Safety Injection. Like the BWR, rod insertion and boron injection through safety injection are passive, having little or no dependence on support systems.

Given the above design features to preclude and accommodate ATWS, a number of which exceed current LWR capabilities and are passive in nature, containment challenges from ATWS are considered to be effectively eliminated.

Reactor Pressure Control

RCS pressure control impacts severe accident conditions within the containment in that core damage at high primary system pressure can potentially lead to high pressure melt ejection should an event progress to the point that reactor vessel lower head penetration occurs. The potential impact of high pressure core melt on containment heatup and pressurization has been substantially addressed in the ALWR requirements.

- Both PWR and BWR designs require safety grade RCS automatic depressurization systems (ADS) in the passive plant. These systems are automatically initiated and designed for reliability, having multiple trains sufficiently independent so as not to be vulnerable to common cause. The ADS is effectively independent of passive plant support systems except dc power. Furthermore, concern regarding the dc power dependence is obviated by the fail-safe capability of the passive residual heat removal (RHR) system to provide the capability to remove decay heat and thus gradually depressurize the primary system. Passive RHR requires no other support systems for operation. Also, the probability of a LOCA which requires ADS together with a complete loss of dc power is very remote. Thus, based on the depressurization system design, the driving force necessary for high pressure melt ejection would not be present in a Passive ALWR in the event of a severe accident.

- Mitigation of the effects on containment of ejection of core debris from the RCS is also provided in Passive PWRs by cavity design features which preclude transfer of significant core debris to areas of containment outside the reactor cavity. For BWRs a number of design features (including the reactor lower internals structure and penetrations design, the inerted containment,and the suppression pool) also act to mitigate the impact of melt ejection.

Given these passive plant preventive and mitigative features, it is concluded that high pressure core melt ejection effects have been precluded as a significant challenge to the integrity of the containment.

Fuel/Debris Cooling

In the Passive ALWR, a number of means are provided to assure adequate reactor inventory control. In the BWR, these include a motor driven feedwater system, a control rod drive system capable of making up for more than decay heat losses, a passive RHR system to lim[illegible]vent reactor inventory loss, and a gravity injection system capabl[illegible] the reactor upon depressurization. PWR requirements include steam generator inventory makeup from both motor driven feedwater and backup feedwater to prevent inventory loss during transients, a passive RHR system which also prevents or limits reactor inventory losses, a low pressure injection capability from the shutdown cooling system, and gravity injection from the IRWST once the RCS is depressurized. In the unlikely event that all of these systems fail to maintain reactor inventory and core damage progresses to the point of lower head penetration, still other provisions are made for cooling core debris in the reactor cavity or lower drywell. These design features, listed below, preclude potential long term severe accident containment challenges which could result from core concrete interaction such as non-condensible gas generation, containment overtemperature, basemat penetration, and loss of reactor vessel supporting structures.

- A number of ALWR design requirements are directed at assuring a coolable geometry for the core debris should it penetrate the reactor vessel lower head. These include a large reactor cavity/lower drywell spreading area, equal to or greater than 0.02 m^2/MWt. This provides a relatively thin debris bed depth which can be quenched and remain coolable given an overlying pool of water.

- Passive means to flood the lower drywell/reactor cavity prior to or immediately upon vessel penetration are required for the ALWR design. The means of flooding are required to be independent of the causes for core damage and the amount of water supplied must be capable of quenching the debris and removing decay heat.

Given these features to provide cooling of core debris, at least one of which is passive with no support system dependencies, core concrete interaction related containment challenges are considered to be effectively eliminated.

Containment Pressure Control

Given Passive ALWR capabilities to preclude containment challenge from non-condensible gas generation, the principal means of long term containment pressurization comes from decay heat generation. Passive plant requirements provide significant redundancy and diversity in the ability to remove decay heat during transients and design basis accidents as well as during severe accident events, thereby assuring that pressurization of containment from steaming caused by decay heat generation does not significantly contribute to the risk of containment failure in the passive plant. The BWR and PWR passive heat removal system capabilities are adequate for both an intact core in the reactor vessel and a damaged core in the reactor vessel or relocated to the cavity/lower drywell.

It should also be noted that traditional active systems are available for decay heat removal (and thus containment pressure control). These systems include the main condenser (BWR) and steam generators (PWR), and additional active systems in the event these systems are unavailable.

- For the passive BWR, passive RHR in the form of isolation condensers are automatic, single failure proof, independent of ac power, fail safe on loss of control power, and sized to have a capacity sufficient to remove decay heat for 72 hours without makeup to replenish the water stored as a heat sink for the system. This passive heat removal (PDHR) affords a diverse, assured backup to the normal non-safety shutdown cooling system which is provided by the reactor cleanup system and can be placed in service at any reactor pressure. In addition, the fuel pool cleanup and cooling system provides capability to remove heat from the suppression pool.

- The PWR passive containment cooling system, like that for the BWR, is an automatic single failure proof, fail safe system that is independent of normal decay heat removal systems and capable of limiting containment pressurization for 3 days. While diversity to passive containment cooling is provided by systems such as the steam generators and reactor shutdown cooling, additional assurance of the passive function is provided by requirements affording the capability to supply fire water flow to the containment shell or even heat transfer through the shell directly to the atmosphere outside containment.

These requirements provide substantial assurance that pressurization of containment from steaming caused by decay heat generation does not significantly contribute to the risk of containment failure in the passive plant. The capabilities of decay heat removal by the BWR and PWR passive heat removal systems is adequate regardless of the location of the core i.e., whether it is being cooled normally in the reactor or has relocated to the reactor cavity/lower drywell as a result of a severe accident.

Combustible Gas Control

Protection of containment from challenges associated with hydrogen is provided by design requirements which limit the amount of hydrogen generated and which accommodate the loads which could result from hydrogen. These requirements are as follows:

- Size containment so as to not exceed 13% dry concentration of hydrogen assuming the equivalent of 75% active clad oxidation, or inert the containment.

- Meet Service Level C (steel containments) or Unity Factored Load (concrete containments) for the maximum steam plus hydrogen load from 75% equivalent active clad oxidation.

- Provide redundant, diverse RCS depressurization and early reactor cavity/lower drywell flooding in order to minimize the amount of hydrogen generated in the event of an ex-vessel accident.

- Either demonstrate that local accumulations of detonable hydrogen concentrations will not occur or that these detonations can be accommodated.

The above requirements, while still evolving for the Passive ALWR, will assure that containment challenges from hydrogen are effectively eliminated.

Containment Isolation Requirements Summary

Given both regulatory based requirements as well as those imposed by the industry, the passive plant containment isolation system has been significantly improved over current plant designs and is considered to be extremely reliable.

- Fewer penetrations exist in the Passive Plant containment than in current plants; many of these are normally closed during power operation limiting the likelihood that they would have to actuate to accomplish isolation.

- Specific ALWR utility requirements ensure that each penetration is protected from single failures which would result in containment bypass.

- All automatic valves are fail safe or dc powered; this effectively eliminates dependencies on support systems and makes the containment isolation system essentially independent of failures which initiate the event or otherwise might contribute to reducing the reliability or loss of core cooling capabilities.

- Remotely operated valve position is indicated in the control room permitting frequent verification of their position by the operator. Effective administrative controls are required to provide assurance that local manual containment isolation valves are locked in position with access to the valve and valve configuration such that it is easy to verify their position.

- A means of periodic, gross leakage check, with the plant on-line is required.

Based on these requirements, it is concluded that containment isolation failure sequences are effectively precluded and will not be a significant contributor to loss of containment integrity.

Containment Bypass

Passive ALWR requirements exist which are explicitly directed at both the prevention and mitigation of containment bypass due to interfacing systems LOCA.

- Because of the design and location (within containment) of Passive ALWR safety systems, few high pressure/low pressure interfaces exist in the passive plant.

- Low pressure systems have been designed to withstand pressures and temperatures which would be associated with exposure to RCS conditions.

- Interlocks are required to prevent inadvertent opening of the high and low pressure boundary.

- Pressure relief capability is provided in low pressure systems to mitigate an interfacing LOCA event should it occur.

Given these requirements, which prevent and accommodate interfacing LOCA events and which go well beyond the requirements in existing LWRs, the threat to containment from interfacing LOCA is essentially eliminated.

A.2.2 Selection of Functional Sequence Type

Based on requirements addressing the complete list of containment challenges in Table A-2, and briefly discussed in Section A.2.1, it is concluded that if a core damage event occurs in the Passive ALWR, the reactor would be shutdown and substantially depressurized. Should the accident progress ex-vessel, debris cooling would be provided independent of the means for core damage. Containment would be isolated (also independent of the reasons for core damage) and combustible gas concentrations would be limited or the containment inerted. Decay heat removal would be capable of limiting containment pressurization for days without makeup or other intervening operator action.

The characteristics of this type of event is one in which core damage at low RCS pressure occurs into an intact containment.

A.3 Definition of Core Damage Event Characteristics

As noted in the previous section, the functional sequence type selected to define the core damage event for the purpose of estimating the physically-based source term for the Passive ALWR is a low RCS pressure core melt, progressing to vessel lower head failure, with an intact containment. A LOCA is a representative initiator.

To provide margin in the physically-based source term, the characteristics of the core damage event have been selected so as to give an enveloping estimate of the radioactive release to containment and the associated off-site dose. These characteristics are as follows:

- rapid core damage progression to provide early fission product release and thus less time for radioactive decay and more time for leakage from the primary containment;

- a vapor pathway in the RCS (i.e., from the core to the containment) to maximize fission product release to the containment atmosphere;

- a large scale core melt involving all or nearly all of the core; and

- the potential for ex-vessel core damage progression.

For the passive PWR a core damage event with these characteristics could be caused by a small or intermediate size LOCA with successful fourth stage depressurization but failure of IRWST gravity drain. For the passive BWR, the core damage event could be caused by a steamline break or a liquid break below the core with successful RCS depressurization but failure of the gravity drain cooling system to inject.

The selected core damage event is further defined in Table A-3, which illustrates the core damage event timing for both BWRs and PWRs. The related thermal hydraulics for each reactor type are discussed in more detail below. In-vessel characteristics are presented first followed by containment response. The emphasis is on sequence characteristics important to developing the source term.

A.3.1 PWR Thermal Hydraulics

Response of the passive PWR is presented in this section and is based on preliminary analysis performed by Westinghouse for the AP-600.[A-2]

PWR In-vessel Thermal Hydraulics

As noted above, the representative sequence type for the Passive ALWR begins with a loss of coolant accident. A medium break (6 inches) in the hot leg of the PWR is simulated as the initiating event. A medium size break is selected as it results in a rapid loss of reactor inventory. This leads to a

TABLE A-3. CORE DAMAGE EVENT TIMING

Event	Time After Initiating Event	Relevant Requirements
1. Core Uncovery	~1 hour	Large RCS Inventory, passive RCS heat removal which slows inventory loss, depressurized RCS, leak before break tending to limit size of RCS break, liquid break below core (BWR)
2. Reactor Vessel Lower Head Penetration	~3 hours (BWR) ~5 hours (PWR)	Same as 1.
3. Ex-Vessel Debris Flooding	At Lower Head Penetration	Cavity/lower drywell flooded prior to or immediately upon lower head penetration
4. Ex-Vessel Release from Fuel Debris	At Lower Head Penetration	Limited due to debris cooling from flood; water pool also scrubs any release
5. Revaporization and Late In-Vessel Release	~3-24 hours (BWR) ~5-24 hours (PWR)	Assumed to begin immediately upon lower head penetration and to be complete by 24 hours; assumes a wet cavity/lower drywell

relatively early core uncovery and release of fission products from the fuel, as compared to small break LOCAs. Larger LOCAs are effectively precluded through the implementation of Leak Before Break piping design, material and detection features.

The medium break also provides for a vapor pathway directly to the containment atmosphere. Smaller breaks in the reactor coolant piping would permit significant releases from the fuel to be directed through safety valves to the IRWST. The assumption of a LOCA as large as 6" effectively maximizes the releases from the reactor coolant system to the containment, as a result.

Table A-4 provides a tabulation of important events within the RCS as a function of time for the medium LOCA.

At time zero the pipe break is assumed to occur. The drop in reactor pressure and rise in containment pressure cause a safety injection signal resulting in the opening of the valves from the core makeup tanks to the reactor coolant system. Within several minutes the reactor coolant system pressure drops to less than several hundred psig allowing the accumulators to inject.

As core makeup tank level falls, trains of the automatic depressurization system are actuated. The first three stages are directed to the IRWST to limit the containment pressure rise and provide scrubbing of releases from the reactor coolant system. The last stage is directed to the loop compartment and thus discharges to the containment atmosphere.

To simulate the low pressure core damage event, passive means of inventory makeup to the vessel from the IRWST are assumed to fail. In addition, it is assumed that active safety injection is not initiated throughout the event. Operation of any of these systems during the accident will begin providing coolant injection to the vessel permitting recovery of core cooling and terminating the event within the vessel.

TABLE A-4. SUMMARY OF EVENTS FOR PASSIVE PWR MEDIUM LOCA

Event	Time
LOCA	0 sec
Reactor Scram	3 sec
RCS Depressurization (<200 psig)	<5 min
Accumulator Depletion	5 min
Core Uncovery	1 hr
Vessel Failure	5 hr

Following core makeup tank and accumulator injection, reactor inventory begins to drop due to boiloff and completion of depressurization to a pressure near or slightly above containment pressure. Given the capacity of accumulators and/or core makeup tanks, the top of the core does not uncover until more than an hour into the event. Inventory depletion rate at this point is relatively slow as passive RHR operation returns condensate to the vessel limiting the losses due to boiloff. Steam cooling of the upper part of the fuel rods limits the extent of heatup and cladding oxidation. As non-condensibles begin to collect in the passive RHR heat exchanger, however, heat transfer and condensation of steam becomes less effective, core uncovery and heatup become more significant and the zirc-water reaction rate increases. More than four hours into the event, fuel melting begins, molten fuel flowing into lower regions of the core where the remainder of the reactor coolant causes it to solidify. Eventually, however, inventory depletion occurs to the point where molten fuel relocates to the core support plate. Approximately five hours into the event the core support plate fails allowing molten debris to contact the lower head. This eventually leads to penetration of the vessel and relocation of core debris into a submerged reactor cavity.

Total hydrogen generation in the period of time up to vessel penetration is dependent on the rate of fuel uncovery and assumptions regarding fuel blockage during melt progression. At most, 30% to 60% oxidation will occur in-vessel for a relatively slowly evolving event (slower than the medium sized LOCA postulated in this scenario) depending on the degree to which channel blockage limits steam flow and cladding oxidation.

For the purpose of deriving the source term, it is assumed that approximately 75% of the original reactor core has participated in the melt at the time of vessel penetration. The remaining 25% of the fuel remains at the core periphery and lower water covered regions within the vessel. The lower power density of the Passive ALWR core, the potential for radiation to cooler components such as the vessel wall and steam cooling from submerged fuel and core debris permit this fuel to remain intact within the vessel. However, it is also assumed that the remaining 25% of the fuel continues to heat up, eventually melting and relocating to the reactor cavity, conservatively providing a basis for late in-vessel releases. These releases occur subsequent to early removal mechanisms for fission products in the containment associated with condensation of steam from the blowdown and debris quench in the reactor cavity.

Conditions within the reactor coolant system following vessel penetration are follows:

- RCS at approximately the same pressure as the containment due to the initial primary system pipe rupture, ADS operation and penetration of the lower head by core debris.

- Lower head submerged due to condensation of reactor coolant inventory and initial ECCS flow from the core makeup tanks and accumulators.

- Steam filled RCS due to submergence of vessel, debris in lower head and possibly the lower portions of fuel assemblies.

These conditions are important in that they result in in-vessel conditions near saturation, with limited driving head from the vessel to the containment providing long transport times for fission product release. Only limited buoyancy driven flow exists from containment through the vessel to sweep out fission products or promote further oxidation of remaining zircaloy within the vessel.

PWR Ex-vessel Thermal Hydraulics

Containment pressure as a function of time for the medium LOCA just described is presented in Figure A-2.

Mass and energy release from the initial blowdown of the RCS cause a pressure spike to approximately 35 psia. This pressure increase result in actuation of the passive containment cooling system (PCCS) resulting in water flow down the outside of the containment shell. This provides a heat sink for steam condensation on the containment walls and a reduction in containment pressure. Active system backup to the PCCS is provided by the ability to initiate fire protection system flow to the top of the containment shell. In

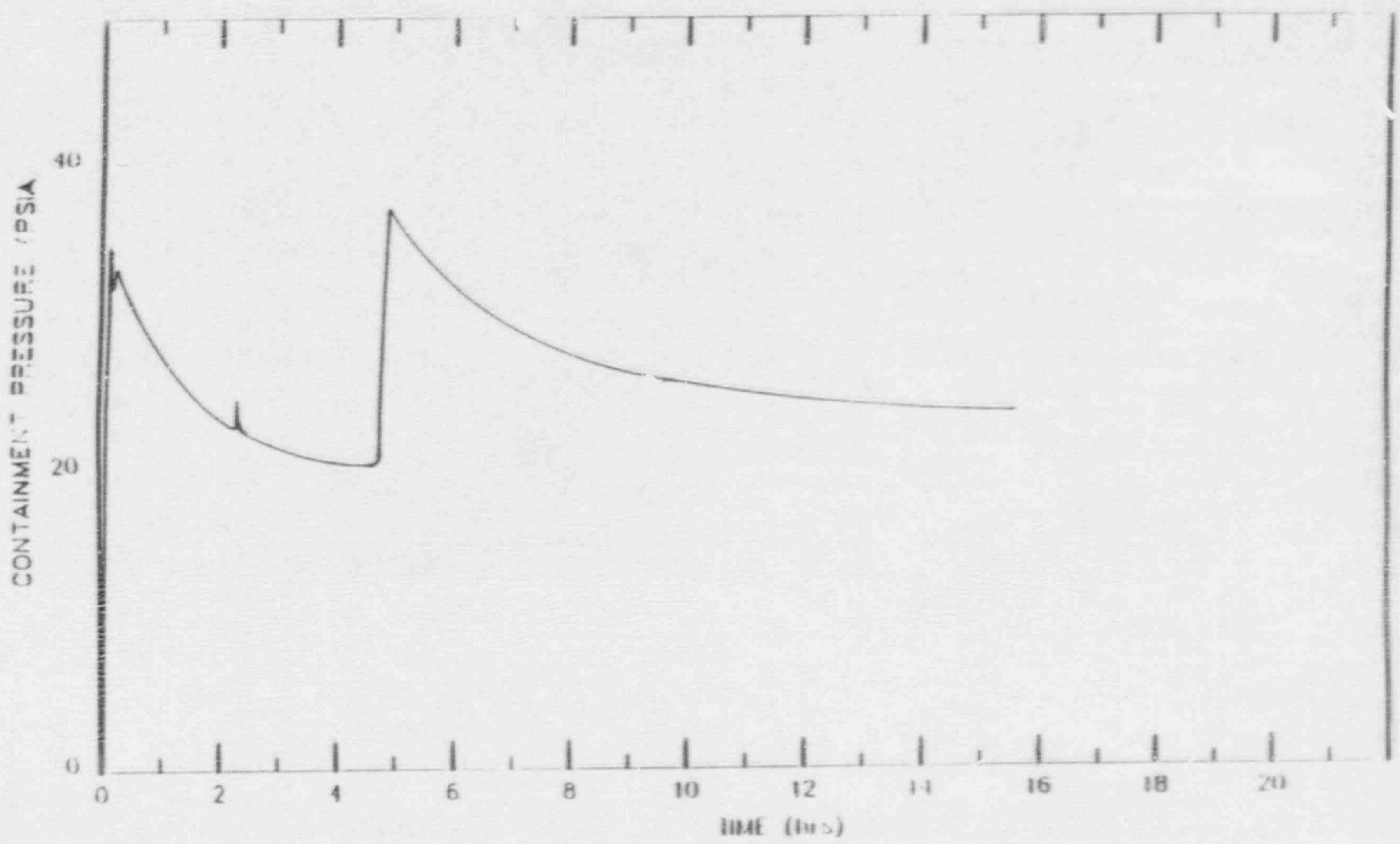


Figure A-2. Postulated Accident Profile for the Passive PWR.

adequate to prevent containment failure even without water flow on the containment shell.

Following core makeup tank and accumulator injection, energy release to containment slows. PCCS operation and other heat sinks slowly reduce containment pressure as the reactor depressurization is completed.

Zirc water reaction in the vessel and heat addition from the oxidation begins at approximately two hours. Slumping of the core debris into the lower vessel and vessel penetration occur approximately 5 hours into the event causing a pressure rise to 35 psia resulting from quenching the debris in the lower vessel and reactor cavity.

Early in the event, two phase flow from the pipe rupture in the loop compartment caused the reactor cavity to begin collecting water. In addition, as condensation of steam occurred on the containment walls and heat sinks, flow of condensate is channeled back to the IRWST causing it to overflow to the refueling cavity. The refueling cavity in turn overflows to the loop compartment and reactor cavity. This IRWST overflow contributes to the filling of the reactor cavity as reactor inventory is depleted within the vessel. The volume of the normal inventory in the vessel plus the core makeup tanks and accumulators permit collection of water in the reactor cavity to several feet above the bottom of the reactor vessel even before slumping of debris to the lower head.

At the time that vessel penetration occurs, the reactor cavity is submerged and the containment atmosphere is saturated. Quenching of debris occurs in the flooded reactor cavity as it leaves the vessel. This minimizes the potential for molten core concrete interaction and scrubs any further fission product releases. Steam from debris quenching and decay heat is condensed through passive heat removal on the containment walls and returned to the reactor cavity through the IRWST to provide long term cooling.

A.3.2 BWR Thermal Hydraulics

Response of the passive BWR is based on preliminary analysis performed by General Electric for the SBWR.[A-3]

BWR In-Vessel Thermal Hydraulics

The representative sequence for the source term quantification begins with a loss of coolant accident. A failure of the two inch drain line in the lower head of the reactor vessel is postulated as the initiating event. This break location is selected as it results in a fairly rapid loss of reactor inventory. This leads to a relatively early core uncovery and release of fission products from the fuel, as compared to small, or even medium size LOCAs at higher elevations in the RCS. As noted in A.3.1 above, large LOCAs are assumed to be effectively precluded through the implementation of Leak Before Break piping design, material, and detection features.

Because the break location is low in the containment and likely to be submerged as the accident progresses, the vapor pathway directly to the containment atmosphere is provided by successful operation of the depressurization valves (DPVs). Small breaks in the reactor coolant piping without DPV initiation would permit significant releases from the fuel to be directed through safety valves to the suppression pool. The assumption of DPV actuation effectively maximizes the releases from the reactor coolant system to the containment, as a result.

A list of significant events within the RCS as a function of time is presented in Table A-5.

At time zero, the pipe break is assumed to occur. Within seconds, reactor trip occurs as a result of reactor low level and drywell high pressure signals. Within tens of seconds of the pipe rupture, MSIV closure on low steam line pressure terminates main steam line flow from the reactor, and low reactor level initiates operation of the isolation condensers (IC). Isolation condenser operation begins depressurization of the reactor and returns condensate to the vessel, thereby limiting the rate of inventory loss from the primary system.

Approximately ten minutes into the event, inventory depletion to the reactor low level permissive for ADS occurs. Depressurization of the reactor to the suppression pool through SRVs and to the upper drywell through DPVs completes primary system blowdown to near containment pressure.

For the purpose of this discussion, passive means of inventory makeup to the vessel from the suppression pool is assumed to fail. In addition, it is assumed that active safety injection is not operable throughout the event. Initiation of any of these systems during the remainder of the accident will begin providing coolant injection to the vessel permitting recovery of core cooling and terminating the event within the vessel.

Following reactor depressurization, reactor level is still significantly above the top of the fuel. More than an additional half hour of boiloff and drainage through the lower head drain line is necessary before the top of the fuel is reached. Steam flow past the upper portion of the fuel rods provides cooling as reactor level continues to drop, the rate of depletion being limited by low reactor pressure near that of the containment and the continued return of condensate to the reactor vessel through steam condensation in the isolation condensers.

Fuel heatup commences more than an hour into the event, with relocation of the hotter regions of the core beginning at 1-1/2 to 2 hours. Slumping of core material into the water in lower regions of the vessel occurs, quenching the debris and generating steam. Relocation of core debris to the core plate occurs on the order of three hours into the event, at which time failure of lower head penetrations is assumed. At that point, core material would begin flowing into the lower drywell.

Total hydrogen generation in the period of time up to vessel penetration is dependent on the rate of fuel uncovery and assumptions regarding fuel blockage during melt progression. A maximum of 30% to 60% oxidation occurs in-vessel for a relatively slowly evolving event (slower than the medium-sized LOCA postulated in this scenario) depending on the degree to which channel blockage limits steam flow and cladding oxidation.

TABLE A-5. SUMMARY OF EVENTS FOR THE PASSIVE BWR LOWER HEAD LOCA

Event	Time
LOCA	0 sec
Reactor Scram	3 sec
Reactor Low Level (ADS)	<10 min
Core Uncovery	~45 min
Vessel Failure	~2.6 hr

Up to this point, approximately 75% of the original reactor core participates in the melt. The remaining 25% of the fuel is located at the core periphery and lower water covered regions within the vessel. The lower power density of the Passive ALWR core, the potential for radiation to cooler components such as the vessel wall and steam cooling from submerged fuel and core debris permit this fuel to remain intact within the vessel. However, for the purpose of deriving the magnitude of the source term for the Passive BWR, the remaining 25% of the fuel is assumed to continue to heat up, eventually melting and relocating to the lower drywell. This assumption conservatively provides a basis for late in-vessel releases. These releases occur subsequent to early removal mechanisms for fission products in the containment associated with condensation of steam from the blowdown and debris quench in the lower drywell.

Conditions within the reactor coolant system following vessel penetration are assumed to be as follows:

- RCS at approximately the same pressure as the containment due to the initial primary system pipe rupture, ADS operation and penetration of the lower head by core debris.

- Lower head submerged due to condensation of reactor coolant inventory and initial ECCS flow from the suppression pool to the lower drywell.

- Steam filled RCS due to submergence of vessel, debris in lower head and possibly the lower portions of fuel assemblies.

These conditions are important in that they result in in-vessel conditions near saturation, with limited driving head from the vessel to the containment providing long transport times for fission product release, and no thermally driven flow through the vessel to sweep out fission products or promote further oxidation of remaining zircaloy within the vessel.

BWR Ex-Vessel Thermal Hydraulics

A plot of containment pressure as a function of time for the BWR lower head LOCA is provided in Figure A-3. Mass and energy release from the initial blowdown of the RCS cause a pressure spike to approximately 30 psia. Steam entering containment is quenched in the suppression pool and initiates flow and condensation in the containment isolation condensers. This provides a heat sink that limits the extent of the containment pressure rise early in the event. Active system backup to passive containment cooling is provided by fuel pool cooling alignment to the suppression pool and operation of the reactor isolation condensers and the reactor cleanup system.

Following depressurization of the reactor, energy release to containment slows as reactor boiloff and draining occur.

Zirc-water reaction in the vessel, and heat addition from the oxidation, begins 1-1/2 to 2 hours. Slumping of the core debris into the lower vessel and vessel penetration occur approximately 3 hours into the event, causing a pressure rise to 45 psia, resulting from quenching the debris in the lower vessel and drywell.

Early in the event, water flow from the pipe rupture in the lower drywell caused the drywell to begin collecting water. At the time the vessel penetration is assumed to occur, flow from the suppression pool is initiated submerging the vessel. Quenching of debris occurs in a flooded reactor cavity. This minimizes the potential for molten core concrete interaction and scrubs any further fission product releases. Steam from debris quenching and decay heat removal is condensed through passive heat removal in the isolation condensers and returns to the drywell through the gravity drain tank and vessel to provide long term cooling. The drywell is principally saturated steam at this time with nearly all the non-condensibles being location in the suppression chamber airspace.

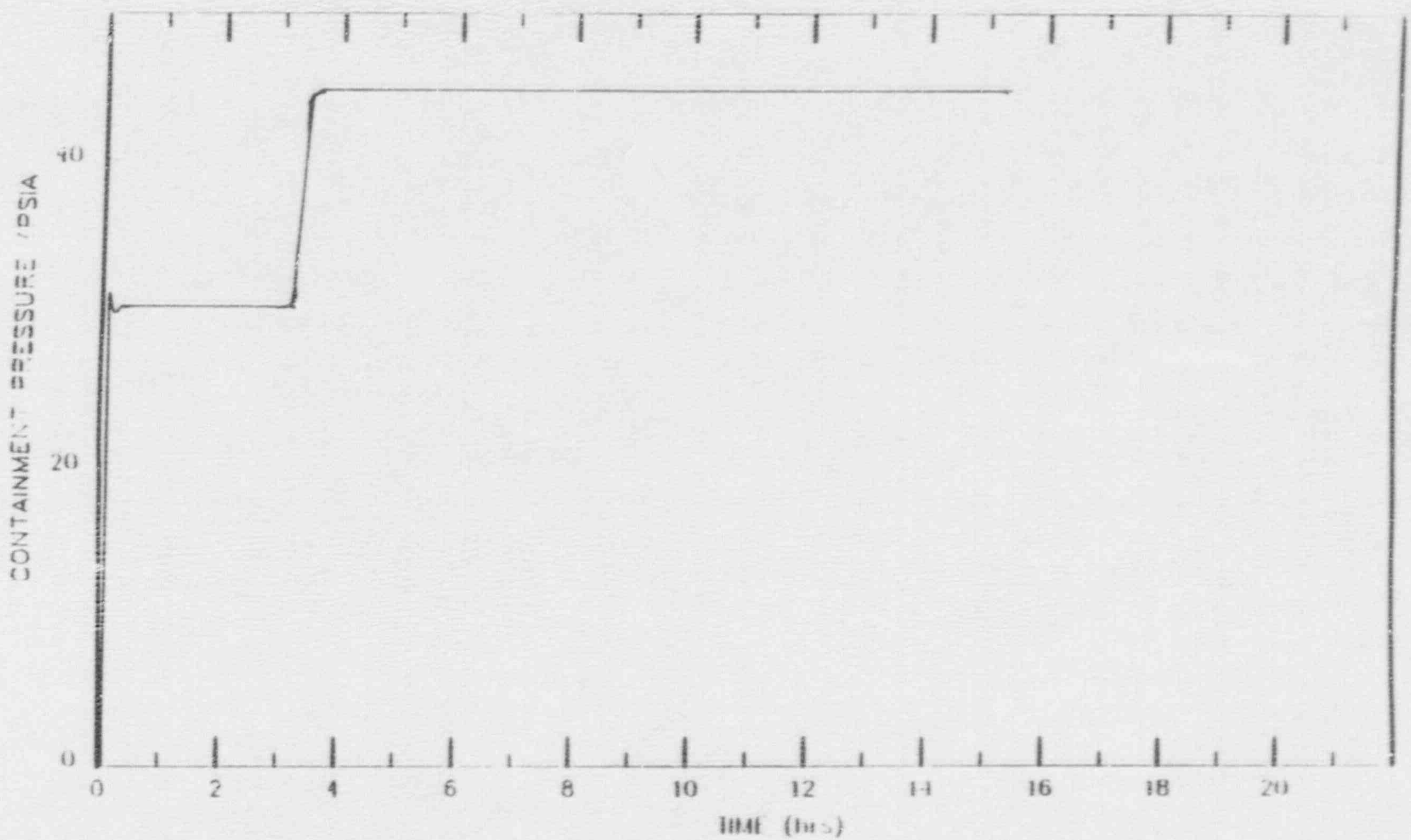


Figure A-3. Postulated Accident Profile for the Passive BWR.

A.3.3 Summary

A representative event has been defined that incorporates generalized severe accident melt progressions for Passive ALWRs. This event definition is based on the representative core melt scenarios for an advanced plant design. The event definition includes sufficient conservatism to envelope the most likely scenarios while avoiding a greater degree of conservatism that would lead to an unrealistic basis for engineered safety feature design and evaluation. Both the unique plant design features and the applicable severe accident uncertainties are considered in determining what are the more credible core melt scenarios and in defining the generalized enveloping event. The Passive ALWR source term event also reflects the conclusion that the expected source terms from the most likely core melt scenarios will be releases into an intact containment.

Appendix 1 References

A-1. Advanced Light Water Reactor Utility Requirements Document, ALWR Passive Plant, Volume III, prepared by Electric Power Research Institute on behalf of ALWR Utility Steering Committee, Rev. 0, August 1990.

A-2. Westinghouse AP600 Plant Description Report, DE-AC03-86SF16038, January 1989.

A-3. General Electric Presentation to ARSAP on SBWR Response to Severe Accidents, December 1990.

APPENDIX 2

SHOREHAM TECHNICAL SPECIFICATIONS
PERTAINING TO REACTOR BUILDING INTEGRITY

CONTAINMENT SYSTEMS

3/4.6.5 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2, or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the pressure within the secondary containment is less than or equal to 1.0 inch of vacuum water gauge.

b. Verifying at least once per 31 days that:

1. All secondary containment equipment hatches are closed and sealed.

2. At least one door in each access to the secondary containment is closed except for routine entry and exit.

3. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position.

c. At least once per 18 months:

1. Verifying that one reactor building standby ventilation system will maintain the secondary containment to greater than or equal to 0.25 inch of vacuum water gauge in less than or equal to 120 seconds, and

2. Operating one reactor building standby ventilation system for one hour and maintaining greater than or equal to 0.25 inch of vacuum water gauge in the secondary containment at an exhaust flow rate not exceeding 1160 cfm.

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

CONTAINMENT SYSTEMS

REACTOR BUILDING AUTOMATIC ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.2 The reactor building ventilation system automatic isolation valves shown in Table 3.6.5.2-1 shall be OPERABLE with isolation times less than or equal to the times shown in Table 3.6.5.2-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *.

ACTION:

With one or more of the reactor building ventilation system automatic isolation valves shown in Table 3.6.5.2-1 inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 8 hours either:

a. Restore the inoperable valves(s) to OPERABLE status, or

b. Isolate each affected penetration by use of at least one deactivated valve secured in the isolation position, or

c. Isolate each affected penetration by use of at least one closed manual valve or blind flange.

Otherwise, in OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, in OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each reactor building ventilation system automatic isolation valve shown in Table 3.6.5.2-1 shall be demonstrated OPERABLE:

a. Prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

b. During COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each isolation valve actuates to its isolation position.

c. By verifying the isolation time to be within its limit when tested pursuant to Specification 4.0.5.

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

TABLE 3.6.5.2-1

REACTOR BUILDING VENTILATION SYSTEM AUTOMATIC ISOLATION VALVES

VALVE FUNCTION	MAXIMUM ISOLATION TIME (Seconds)
1. Reactor Building Normal Ventilation Supply Valve 1T46*AOV35A	10
2. Reactor Building Normal Ventilation Supply Valve 1T46*AOV35B	10
3. Reactor Building Normal Ventilation Exhaust Valve 1T46*AOV37A	10
4. Reactor Building Normal Ventilation Exhaust Valve 1T46*AOV37B	10

CONTAINMENT SYSTEMS

REACTOR BUILDING STANDBY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.5.3 Two independent reactor building standby ventilation systems (RBSVS) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and *.

ACTION:

a. With one RBSVS inoperable, restore the inoperable system to OPERABLE status within 7 days, or:

   1. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. In OPERATION CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

b. With both RBSVS inoperable in OPERATIONAL CONDITION *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATIONS or operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3. are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.5.3 Each RBSVS shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters OPERABLE.

*When irradiated fuel is being handled in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire, or chemical release in any ventilation zone communicating with the system by:

   1. Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 1% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, at a system exhaust flow rate of 1160 cfm ± 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%; and

   3. Verifying a system exhaust flow rate of 1160 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978, for a methyl iodide penetration of less than 1%.

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 4.2 inches water gauge while operating the filter train at a system exhaust flow rate of 1160 cfm ± 10% in the single train operating mode.

   2. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      a. Manual initiation from the control room, and

      b. Simulated automatic initiation signal.

   3. Verifying that the heaters dissipate 5.7 ± 1 kW when tested in accordance with ANSI N510-1975.

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 while operating the system at a system exhaust flow rate of 1160 cfm ± 10% in the single train operating mode.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the inplace penetration and bypass leakage testing acceptance criteria of less than 1% in accordance with ANSI N510-1975 for a halogenated hydrocarbon refrigerant test gas while operating the system at a system exhaust flow rate of 1160 cfm ± 10% in the single train operating mode.