Docket File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 8, 1994

Project No. 669

APPLICANT: Electric Power Research Institute (EPRI)

PROJECT: Advanced Light Water Reactor (ALWR) Utility Requirements Document

SUBJECT: SUMMARY OF MEETING BETWEEN THE NUCLEAR REGULATORY COMMISSION (NRC) STAFF AND EPRI HELD ON JANUARY 27, 1994, IN ROCKVILLE, MARYLAND, CONCERNING PHYSICALLY-BASED SOURCE TERM

A public meeting was held on January 27, 1994, at the NRC headquarters in Rockville, Maryland, between the staff, the ALWR vendors, and EPRI, to discuss the status of the staff's review of the new physically-based source term. A list of attendees and their affiliation is provided in Enclosure 1.

EPRI summarized a number of concerns with the draft Commission paper that was sent to the Advisory Committee on Reactor Safeguards (ACRS) on January 6, 1994, and with draft NUREG-1465. These concerns were addressed in greater detail by the individual vendors in their presentations that followed. The slides used by EPRI in their presentation are provided in Enclosure 2.

As indicated in Enclosure 3, GE Nuclear Energy (GE) would like to receive feedback from the NRC staff in several areas including:

- Simplified Boiling Water Reactor aerosol removal and aerosol removal calculations

- proposal that fission product release timing in draft NUREG-1465 be design-specific

- breakdown into soluble and insoluble inert radioactive aerosols

- organic iodide fraction - GE proposes using a value of 0.05 percent relative to the staff's value of 0.25 percent, appearing in the draft Commission paper

The slides used by Westinghouse in their presentation are provided in Enclosure 4. The following positions were proposed by Westinghouse for use in the AP600 design review:

- Westinghouse recommended that plants be required to provide adequate pH control for the containment sump

- Westinghouse would like NUREG-1465 to clearly state that the gap fraction for release to containment atmosphere is 5 percent

- Westinghouse recommended that the latest data on the release of low volitiles should be examined and incorporated in the source term

- Issue 1 from the draft Commission paper - NUREG-1465 should address gap release and early in-vessel release as defining design-basis accident (DBA) core analysis and eliminate reference to other severe accident release phases

- Issue 3 from the draft Commission paper - specification of source term basis

- Issue 5 from the draft Commission paper - Westinghouse retains credit for holdup of activity in auxiliary building for severe accident analysis but not DBA analysis.

- Issue 7 from the draft Commission paper - Westinghouse feels that the approach for calculating aerosol deposition in draft NUREG-1465 is excessively conservative.

- Issue 6 from the draft Commission paper - Westinghouse feels that it can justify at least a 60-minute interval for the release of activity from the core, relative to the 10-minute value proposed by the staff when the leak-before-break is credited.

The slides used by ABB-Combustion Engineering (ABB-CE) are included as Enclosure 5. The main points emphasized by ABB-CE include:

- Because the dominant fission product removal mechanism in the System 80+ design is containment spray, ABB-CE expressed concern that the Sandia report (draft NUREG/CR-5966) did not include any discussion of hydroscopicity and contained only limited discussion of mixing.

- The EQ for System 80+ consists of 2 levels depending on whether or not the system is required for long term cooling, post-depressurization.

    - Level 1 is based on 100-percent gap release, per draft NUREG-1465.

    - Level 2 is based on a combination of 100-percent gap release, plus early, in-vessel release, plus some additional conservatism.

At the end of the meeting, EPRI requested that the staff consider holding another meeting on source term this spring, prior to RES transmitting the final version of NUREG-1465 to the ACRS. The staff agreed to consider holding such a meeting, and stated that it would notify EPRI whether or not such a meeting would be useful.

Also at the end of the meeting, EPRI expressed a concern that the language in the draft Commission information paper seemed to indicate that the positions based on draft NUREG-1465 were final positions. Both the staff and industry recognize that discussions are continuing concerning implementation of the new source term in the individual applications for FDA/DC. The staff agreed to consider adding a clarification to the final Commission information paper indicating that, although the positions described in the information paper were current staff positions, the details of implementation would have to be resolved with the individual ALWR vendors during the course of each design review.

(Original signed by)

James H. Wilson, Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of Nuclear Reactor Regulation

Enclosures:
As stated

cc w/enclosures:
See next page

DISTRIBUTION w/enclosures:

Docket File	PDST R/F	JHWilson	PShea
PDR	DCrutchfield		

DISTRIBUTION w/o enclosures:

RBorchardt	RArchitzel	WTravers	TWambach
JMoore, 15B18	EJordan, MNBB3701	WDean, 17G21	MMalloy
JLee, 10D4	LSoffer, NLS324	CAder, NLS324	HWalker, 8D1
REmch, 9H15	ACRS (11)	TEssig, 10D4	JHayes, 10D4
KEccleston, 10D4	MSnodderly, 8H7	ADrozd, 8H7	EFox, 9H15

OFC	LA:PDST	PM:PDST JW	PRPB TWE	SC:PDST R
NAME	PShea PS	JHWilson:tz	TEssig	RArchitzel
DATE	02/7/94	02/7/94	02/7/94	02/9/94

OFFICIAL RECORD COPY: MTSM127.JHW

cc: Mr. R. P. McDonald, RP - ARC Bin 854
Southern Company Services Room 518
P.O. Box 2625
Birmingham, Alabama 35202

Mr. John Trotter
Nuclear Power Division
Electric Power Research Institute
Post Office Box 10412
Palo Alto, California 94303

Mr. Brian A. McIntyre, Manager
Advanced Plant Safety & Licensing
Westinghouse Electric Corporation
Energy Systems Business Unit
Post Office Box 355
Pittsburgh, Pennsylvania 15230

Mr. Joseph Quirk
GE Nuclear Energy
Mail Code 782
General Electric Company
175 Curtner Avenue
San Jose, California 95125

Mr. Stan Ritterbusch
Combustion Engineering
1000 Prospect Hill Road
Post Office Box 500
Windsor, Connecticut 06095

Mr. Sterling Franks
U.S. Department of Energy
NE-42
Washington, D.C. 20585

Mr. Steve Goldberg
Budget Examiner
725 17th Street, N.W.
Room 8002
Washington, D.C. 20503

LIST OF ATTENDEES AT MEETING WITH EPRI HELD IN
ROCKVILLE, MARYLAND ON JANUARY 27, 1994

Name	Affiliation
R. Borchardt	NRC
R. Architzel	NRC
J. H. Wilson	NRC
T. Wambach	NRC
J. N. Wilson	NRC
M. Malloy	NRC
T. Essig	NRC
J. Lee	NRC
J. Hayes	NRC
K. Eccleston	NRC
R. Emch	NRC
E. Fox	NRC
A. Drozd	NRC
H. Walker	NRC
M. Snodderly	NRC
L. Soffer	NRC
C. Ader	NRC
J. Mazetis	NRC
W. Pasedag	DOE
J. Trotter	EPRI
D. Leaver	EPRI
S. Ritterbusch	ABB/CE
W. Usry	GE
J. Grover	Westinghouse
J. Li	TENERA
S. Additon	TENERA
J. Metcalf	Stone & Webster
D. Teague	Winston & Strawn
K. Graney	Bechtel
R. Hobbins	RRH Consulting

Enclosure 1

Key Source Term Issues Requiring Further Discussion and Clarification

David E. Leaver
John Trotter

Presented to NRC

January 27, 1994

Enclosure 2

Key Source Term Issues

NRC Issue 1 - Selective Use of Draft NUREG 1465

NRC Issue 2 - Iodine Chemical Form

NRC Issue 5 - Secondary Building Holdup

NRC Issue 6 - Timing

NRC Issue 7 - Containment Natural Aerosol Removal

NRC Issue 9 - Containment Spray (ABB)

NRC Issue 12 - Failure of Heart Exchange Tubes in SBWR PCCS (GE)

Non-Fission Product Aerosol Quantity

Equipment Qualification and Equipment Survivability

Source Term Impact on Emergency Planning

NRC ISSUE 1 - SELECTIVE USE OF DRAFT NUREG 1465

0 Industry agrees with the use of gap and early in-vessel releases for DBA

0 The NUREG 1465 in-vessel and ex-vessel low volatile releases are much larger than ALWR proposed values based on experiment and the TMI-2 accident

0 The fact that revisions to low volatile release fractions would "not materially change the ongoing staff reviews" of ALWR designs is not a valid reason to utilize these large release fractions since:

(1) the rules for dose calculation are changing such that the low volatiles will have a greater impact on the dose

(2) the best available technical information should be used

0 Evaluation of low volatile release data by Osetek (1992) supports ALWR proposed values

0 What are NRC's reservations about the ALWR proposed values of low volatile fission product releases?

Process for Containment and Source Term Evaluations

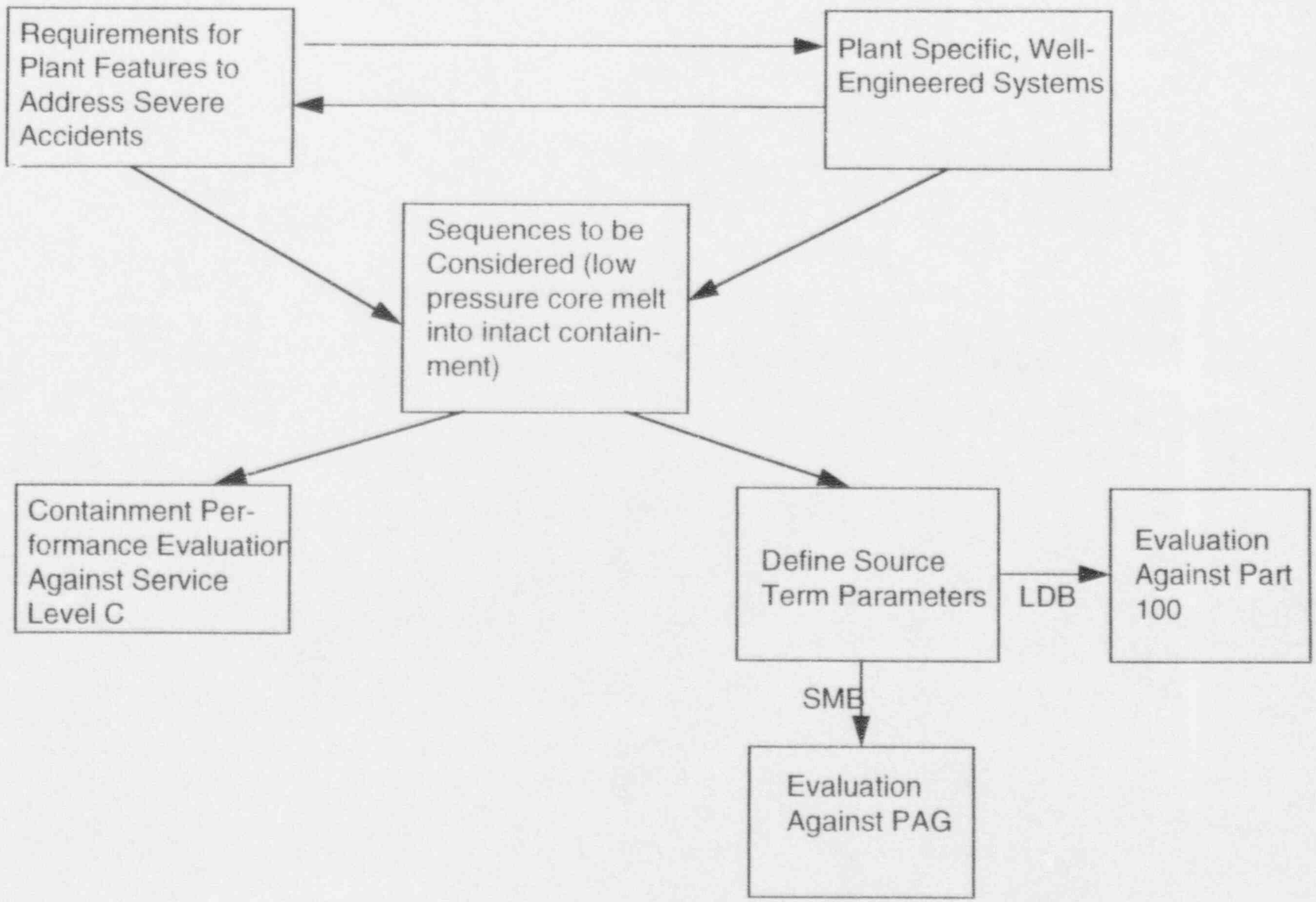


Table 2-1

Potential Severe Accident Containment Challenges

CHALLENGES/FAILURE MODES THAT ARE INDEPENDENT OF OR COINCIDENT WITH A SEVERE ACCIDENT

1. Containment Isolation
2. Interfacing System LOCA
3. Blowdown Forces
4. Pipe Whip and Jet Impingement
5. Steam Generator Tube Rupture (PWR)
6. Anticipated Transient Without Scram (ATWS)
7. Suppression Pool Bypass (BWR)
8. Reactor Pressure Vessel (RPV) Failure
9. Internal Vacuum
10. Internal (Plant) Missiles
11. Tornado and Tornado Missiles
12. Man-Made Site Proximity Hazards
13. Seismic

CHALLENGES/FAILURE MODES POTENTIALLY RESULTING FROM A SEVERE ACCIDENT

14. High Pressure Melt Ejection (HPME)
15. Hydrogen Detonation/Deflagration
16. In-vessel Debris-Water Interaction
17. Ex-vessel Debris-Water Interaction
18. Noncondensable Gas Generation During Core-Concrete Interaction
19. Containment Basemat Erosion or Reactor Pressure Vessel Support Degradation During Core-Concrete Interaction
20. Core Debris in Containment Sump
21. Core Debris Contact with Containment Shell Liner
22. Decay Heat Generation
23. Steam Generator Tube Rupture (SGTR) from Natural Circulation of Hot Gases (PWR)

Characteristics of Sequences for Evaluation of Containment Performance and Source Term

- Establish sequence characteristics based upon a **deterministic** perspective, i.e., confirm that well-engineered containment systems exist to mitigate challenges which could be an early threat to containment integrity
- Perform supplementary **probabilistic** evaluations - consider any sequences greater than approximately 10-7 per yr
- Based upon the deterministic and probabilistic perspectives, the sequence characteristics are:

  <u>Core Damage</u>

  - Rapid core damage progression over a time frame of several hours
  - Large scale core melt and associated gas and aerosol release
  - Steam out of phase with aerosol release
  - Consideration of early in-vessel release for LDB source term (and ex-vessel core damage for SMB)

  <u>Reactor Coolant System (RCS) Condition</u>

  - Limited aerosol plateout in the RCS
  - A vapor pathway exists in the RCS (i.e., from the core to the containment atmosphere)
  - RCS is depressurized to about 100 psig or less

Characteristics of Sequences
(continued)

<u>Containment Condition</u>

- Containment is isolated and otherwise intact at that the time of core damage (i.e., no containment bypass has occurred)
- Water exists in the reactor cavity/lower drywell prior to or immediately upon reactor vessel lower head penetration
- Containment systems are functioning as designed (heat removal, fission product removal, hydrogen control, pH control)
- Containment leaking at design basis leak rate (or at leak rate proportional to pressure)

<u>Secondary Building Condition</u>

- Containment leakage release into containment building
- Building volume mixing and exchange with environment is based upon plant design characteristics (e.g., safety envelope leakage)
- Building volume bypass pathways taken into account

URD Design Criteria in Support of Emergency Planning

- **Criterion 1 - Containment Performance**
  - Meet Utility Requirements Document provisions addressing comprehensive list of containment challenges
  - Containment loads from low pressure core melt sequences do not result in exceeding Service Level C for approximately 24 hours or longer
  - Beyond 24 hours, there shall be no uncontrolled release
- **Criterion 2 - Dose**
  - Dose from physically-based source term for median meteorology does not exceed 1 rem for approximately 24 hours at 0.5 miles from reactor

PRA Evaluation in Support of Design Criteria

- Meet 10^{-5} per year core damage frequency
- Meet 10^{-6}, 1 rem at site boundary
- Meet the quantitative health objectives of the NRC Safety Goal Policy with no credit for evacuation prior to 24 hours

Source Term Applicability to Design Basis vs. Emergency Planning

	Applicable Release	Applicable Removal	Applicable Limits
Design Basis (LDB)	Early In-Vessel	Safety-Related Systems and Structures	Part 100
Emergency Planning (SMB)	Early In-Vessel, Ex-Vessel, Late In-Vessel	Safety-Related and Non-Safety Related Systems and Structures	PAG

ALWR Mitigation Licensing Design Basis vs Safety Margin Basis

	Licensing Design Basis (LDB)[1]		Safety Margin Basis (SMB)[2]	
	Event	Applicable Limit	Event	Applicable Limit
Containment Load	LOCA	Service Level A	LOCA plus loads from severe accident phenomena (deterministic)	Service Level C
	LOCA plus Hydrogen (100% with Control System)	Service Level C	Containment loads from PRA sequences (>10-7/yr)	Service Level C
Source Term	Physically-based; early in-vessel release	Part 100	Physically-based; early in-vessel, ex-vessel, late in-vessel releases	Protective Action Guidelines (PAGs)
			Source term from PRA sequences (>10-7/yr)	PAGs

Notes:

1. LDB evaluation methodology uses conservative, established design methods and credit for safety grade systems (and selected nonsafety grade systems)
2. SMB evaluation methodology uses best estimate methods and credit for safety grade and nonsafety grade systems

SBWR

Offsite Dose Studies

Presented at

Source Term Meeting

Rockville, Maryland

January 27, 1994

Bill Usry

GE Nuclear Energy

San Jose, CA

(408) 925-3460

Enclosure 3

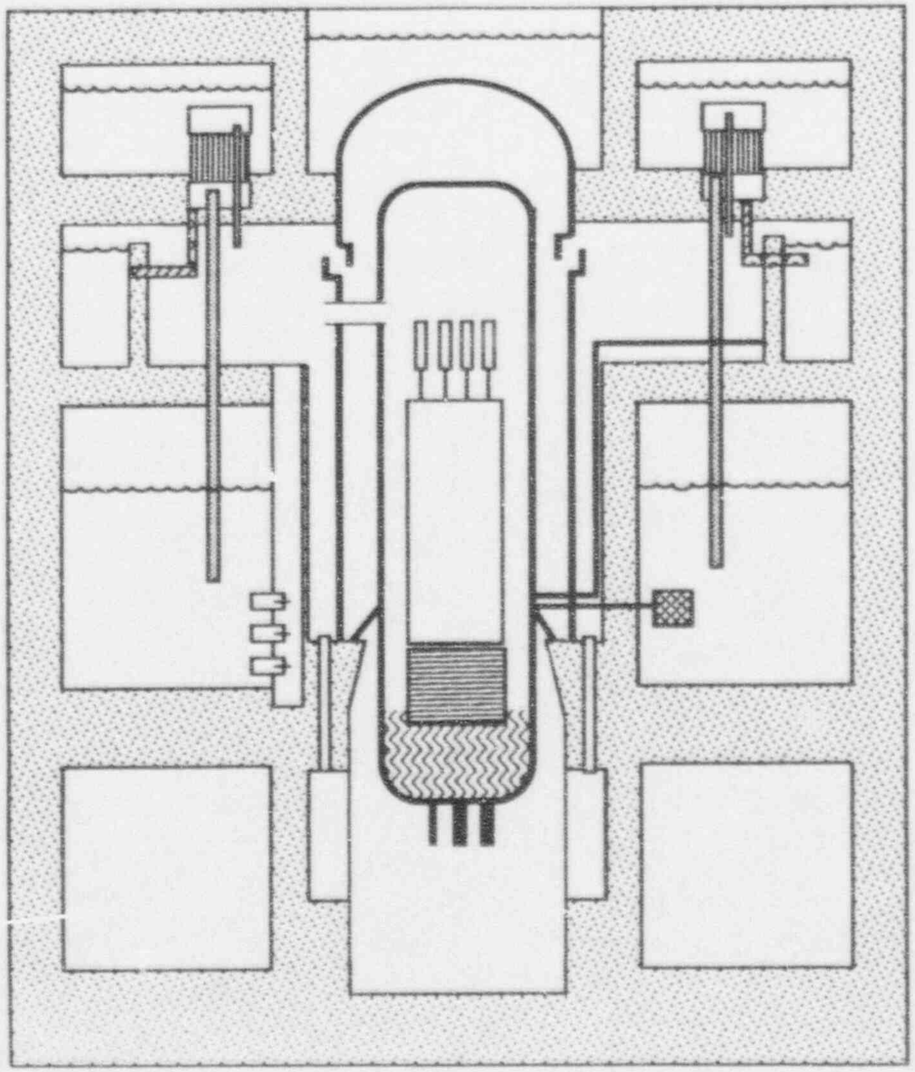
Overview

- Introduction
- SBWR aerosol removal
- SBWR aerosol removal calculations
- Fission-Product Realease Timing
- Nonradioactive aerosols
- Organic Iodide fraction
- Summary

Introduction

- GE appreciates the effort put into NUREG 1465
- GE has made two SSAR submittals
  - first used NUREG 1465
  - then used EPRI source term
- GE will make additional submittal this Spring
- Many organizations have contributed in the offsite dose studies
  - EPRI consultants: Containment aerosol removal
  - ECN: Holdup in Safety Envelope; containment aerosol removal
  - KEMA: Computational fluid dynamics study of Safety Envelope
  - UC Berkeley: PCCS removal analysis
- GE has a few disagreements with NUREG 1465
- Would like to get feedback from NRC

Aerosol Path



- *Generated in Vessel*
- *Released to drywell gas space*
  - *removed*
  - *enter PCCS*
  - *leaked from containment*
- *In PCCS*
  - *Condensate to GDCS*
  - *Uncondensed flow to WW*

SBWR Aerosol Removal

- Small containment yields high aerosol sedimentation
- High humidity enhances agglomoration
- PCCS tubes act like a filter
  - receive aerosols and steam rich gas mixture
  - aerosols removed by diffusiophoresis
  - removed aerosols drain down to GDCS pool
  - remaining aerosols are blown down to the suppression pool
- Most of aerosols are removed to a Nitrogen poor environment

SBWR Natural Aerosol Removal Calculations

- Calculations in SSAR currently use $\lambda = 0.6$/hr
- SBWR using NAUAHYGROS to calculate removal
  - goal is to calculate new removal coefficient
- MAAP-SBWR used to calculate the thermal hydraulic conditions and start time of gap release
- Sequence: low pressure, no injection, recovery before vessel failure
- Calculated removal coefficient
  - 0.0 - 2.0 hrs $\lambda = 1.83$
  - 2.0 - 3.4 hrs $\lambda = 0.57$
  - > 3.4 hrs $\lambda = 1.36$

SBWR ACCIDENT SEQUENCES CONSIDERED FOR CONTAINMENT AEROSOL CALCULATIONS

Sequence	Frequency (yr^{-1})	Time of Core Uncovery, hr	Onset of Core Damage, hr	Reflood time, hr
LPL (low pressure, loss of long term makeup)	$\sim 2 \times 10^{-8}$ (~8% of CDF)	7	8.3	12
LPE (low pressure, loss of short term makeup)	$\sim 8 \times 10^{-8}$ (~45% of CDF)	0.9	1.6	4.5
MPL (medium pressure, loss of long term makeup)	$\sim 6 \times 10^{-8}$ (~33% of CDF)	8	9.3	13.5
MPE (medium pressure, loss of short term makeup)	$\sim 5 \times 10^{-9}$ (~3% of CDF)	0.8	1.6	4.4
BDL (bottom drain line break, partial injection)	$\sim 1 \times 10^{-8}$ (~4% of CDF)	3	4.5	5.5
BDE (bottom drain line break. no injection)	$\sim 1 \times 10^{-9}$ (~0.5% of CDF)	0.6	1.4	3.1

July 21, 1993

The NRC June, 1992 draft source term report NUREG 1465 states that large quantities of nonradioactive or relatively low activity aerosols will be released into containment. Table 3.13 specifies fixed quantities for BWRs and PWRs in both In-Vessel and Ex-Vessel phases. For the SBWR the specified nonfission product aerosols in Table 3.13 are fourteen times the mass of the fission product aerosols specified in Table 3.11 of NUREG 1465 (gap release plus Early-In Vessel). Recent works suggest that this amount of nonradioactive aerosols may be over an order of magnitude too high.

Reference 1 estimates the composition and masses of aerosols for both BWR and PWR cores. Tables 5 and 6 list the estimated vaporized fractions for PWR and BWR cores, respectively. In both cases the structural aerosols account for approximately **10%** of the total aerosols. However, when control rods are considered for PWRs the nonfission product aerosols account for approximately 75% of the total aerosols. Tables 9 and 10 show the same information as Tables 5 and 6 with the inclusion of Boron. With Boron included the nonfission product aerosols (mostly B_2O_3) account for 77% of the total aerosols. However, the findings in the more recent work in Reference 2 show that the reaction kinetics between B_4C and stainless steel are rapid enough to preclude the reaction of B_4C and steam and thus eliminate the Boron aerosols. As a result, Table 6 (which shows that only 10% of the aerosols are nonradioactive) is more appropriate than Table 10 for specifying the masses and distribution of aerosols for BWRs.

Reference 3 further supports the notion of a smaller amount of nonfission product aerosols. The authors state in the summary and conclusions: "source term computer codes like CORSOR-M tend to overpredict the release of structural and control rod material relative to fission products because the models do not account for relocation of molten control, fuel and structural material during the degradation process which tends to reduce the aerosol source." The work was based on a PWR and they show that the nonfission product aerosols are 1 to 3 times the mass of the fission products. As indicated above, BWRs can be expected to have an even lower percentage of nonfission product aerosols because much of the PWR aerosols are the result of the Ag-In-Cd control rods which are not used in BWRs.

Based on the work in References 1, 2, and 3, GE feels that an appropriate ratio of nonfission product aerosols to fission product aerosols is 1:1 for the In-Vessel phase. This is still 10 times more than the amount of nonfission product aerosols found in Table 6 of Reference 1.

REFERENCES

1. Wichner, R.P. and Spence, R.D., 1984, "Quantity and Nature of LWR Aerosols Produced in the Pressure Vessel During Core Heatup Accidents - A Chemical Equilibrium Estimate," NUREG/CR-3181, ORNL/TM-8683.

2. Hobbins, R.R., Petti, D.A. and Osetek, D.J., 1991, "Review of Experimental Results on Light Water Reactor Core Melt Progression," *Nuclear Technology*, Vol. 95.

3. Petti, D.A., Hobbins, R.R. and Hagrman, D.L., 1993, "The Composition of Aerosols Generated During a Severe Reactor Accident: Experimental Results from the Power Burst Facility Sever Fuel Damage Test", Accepted for publication in *Nuclear Technology*.

Reference Accident Sequence

- Chose LFE because:
  - largest percentage of core damage frequency (CDF)
  - smallest time to core uncovery of sequences with significant CDF
- Time to core uncovery and T/H calculated by MAAP-SBWR
  - NUREG-1465 gap and early in-vessel volatile release fractions used
  - EPRI early in-vessel release fractions for low- and non-volatile fission products used
  - gap release begins at 0.9 hours (1 hour duration)
  - In-vessel release begins at 1.9 hours (1.5 hours duration)
  - reflood prior to vessel failure (at 4.5 hours)

Fission-Product Release Timing

- NUREG 1465 provides realistic estimates (source term, duration, etc.)
  - requiring a set release timing is inconsistent with the realistic estimate approach
- Release Timing (or time to uncovery) is a relatively mature phenomenon
- Should be plant specific
  - rewards good designs
  - Doesn't tie hands of future designers
- In Section 3.2, draft NUREG 1465 states:
  - "... the time to initial fuel rod failure is long for BWRs, even for large LOCAs,..."

Nonradioactive Aerosols

- Inert-to-fission product ratio = 1:1 in NAUAHYGROS analysis
- NUREG 1465 specifies fixed amount
  - works out to be 14:1 for SBWR
  - NUREG 1465 states that detailed analysis was not undertaken
- GE does not agree with NUREG 1465 on this matter as was stated in the July 30, 1993 letter to NRC
- July 30, 1993 letter:
  - presented references to more recent work
  - proposed inert-to-fission product ratio of 1:1 for BWRs
- GE would like NRC feedback on this issue
  - breakdown into soluble and insoluble inert aerosols?

Organic Iodide Fraction

- GE believes that an organic iodide fraction of 0.25% (5% of gaseous I_2 fraction) is too high for BWRs as detailed in July 30, 1993 letter.
- 4 BWR accident sequences were calculated in NUREG/CR 5732
  - average gaseous I_2 fraction was 0.02%
  - average organic iodide fraction was 0.0005%
- GE believes that the gaseous I_2 fraction can be conservatively set at 1% with a corresponding 0.05% organic Iodide value.
- GE would also like NRC feedback on this issue

July 9, 1993

The NRC June, 1992 draft source term report NUREG 1465, concluded that iodine entering ontainment from the RCS is composed of at least 95% CsI and no more than 5% I plus HI. At the May 18,19, 1993 source term meeting between the ALWR Program and the NRC it was noted that this 5% value is high, particularly for BWRs (see NRC meeting report of June 9, 1993). This letter is to provide further input on this matter.

The basis for the NUREG 1465 5% I/HI fraction is ORNL report NUREG/CR-5732 (Iodine Chemical Forms in LWR Severe Accidents). The executive summary states: "The gaseous I_2 fraction is considerably higher in PWRs than in BWRs because of the large water volumes in the latter, which both lower dose rate and retain greater quantities of dissolved I_2." Table 3.6 on page 26 of the same document shows the distribution of iodine species for various accident sequences at two BWRs and two PWRs for pH controlled above 7. The average gaseous I_2 fraction for the PWRs is 1.5% while it is only 0.016% for the BWRs, a difference of two orders of magnitude. On page 25 the authors state "Table 3.6 indicates a small production of volatiles for PWRs but virtually none for BWRs."

To further support the differences in I_2 production of PWRs versus BWRs Equation 35 on page 23 may be examined. Equation 35 is an expression of the fraction of iodine that is volatilized when organic iodide is ignored:

$$\text{Fraction volatilized} = [1 + V_L f(T)/V_g f(pH)]^{-1}$$

where V_g is the gas volume, V_L is the liquid volume and f denotes a function. Thus, the ratio of the fraction volatilized for PWR versus BWR can be approximated by:

$$(V_{g,PWR} V_{L,BWR})/(V_{g,BWR} V_{L,PWR})$$

The average V_L/V_g value for the four BWR cases (from Table 3.5 on page 24) is 0.33 and the average value for the three PWR cases is 0.015. Therefore, the ratio of the fraction volatilized for PWRs versus BWRs is approximately 0.33 / 0.015 = 22 which further supports the results from Table 3.6.

GE feels that based on the results of NUREG/CR-5732 and the fact that SBWR will have controlled pH levels, the fraction of iodine that appears as HI plus I for BWRs can be set at 1% and still achieve over an order of magnitude conservatism. We support the decision to make the fraction of organic I as 5% of the amount of I plus HI. This would specify that 0.05% of Iodine appear as organic Iodine. This is 50 times the maximum amount found on any BWR sequence in Table 3.6.

It is suggested that the final version of NUREG 1465 include a statement recognizing that a 1% HI/I value, and a corresponding 0.05% organic iodide value, is acceptable for BWRs as discussed above. This value can then serve as a conservative estimate for licensing the SBWR.

Summary

- SBWR has excellent natural aerosol removal capabilities
- GE is addressing the pertinent technical issues
- GE feels that NUREG 1465 is overall an excellent step forward from TID-14844
- GE has some disagreements with NUREG 1465
  - Fission product release timing
  - Organic Iodide fraction
  - Amount of nonfission product aerosols
- GE would like feedback from NRC

WESTINGHOUSE AP600 PLANT

AND THE REVISED NRC SOURCE TERM

Jim Grover
Containment & Radiological Analysis
Westinghouse Electric
January 27, 1994

Enclosure 4

Issues of Concern to Westinghouse

1. Statement made in draft SECY letter that non-safety charcoal adsorbers or a spray system would be required if a design does not provide adequate pH control post-LOCA

   Plants should be required to provide for adequate pH adjustment.

2. In NUREG-1465 it is stated that the gap release to the containment atmosphere is 5% of the core activity for iodines, cesiums, and noble gases. Although it is not directly stated in the NUREG, the implication is that the gap fraction is 5% and there is no credit for deposition of iodines or cesiums on RCS surfaces.

   We would like to have a clear statement that the gap fraction is five percent.

3. The draft SECY letter states that the release of low volatiles is not a concern since they have little impact on immersion dose. However, if a new dose acceptance criterion is generated for "risk equivalent dose," the non-volatiles will have a major impact on caclculated doses.

   Latest data on the release of low volatiles should be examined and incorporated in the source term.

4. NRC Issue 1: With the selection of gap release and early in-vessel release as defining the DBA core releases, it seems that NUREG-1465 should be revised to eliminate reference to the other release phases that are associated with the severe accident.

5. NRC Issue 3: It seems that the specification of source term basis for EQ for design features needed for severe accident mitigation should not be in the main body of the NUREG. If in the NUREG, it should be as an appendix.

6. NRC Issue 5: Westinghouse retains credit for holdup of activity in the auxiliary building for severe accident analysis but not for design basis analysis.

7. NRC Issue 7: Westinghouse position is that the approach included in NUREG-1465 for aerosol deposition in the containment is excessively conservative.

8. NRC Issue 6: Timing for the release of activity from the core

BACKGROUND ON RELEASE TIMING ISSUE

Draft NUREG-1465 states that the gap release should be initiated at 10 to 25 seconds.

May 1993 Meeting with NRC Staff

> Westinghouse presented to staff that analysis shows that the initiation of gap release would not occur until over an hour into the LOCA.
>
> The statement was made by staff that it was believed that timing of release could be considered on a plant specific basis but that this position would have to be verified.
>
> It was strongly stated by staff that leak before break could not be used as an assumption in the LOCA dose analysis.

Now, the draft SECY letter on the source term states that the maximum delay time that can be considered is 10 minutes and that this delay is associated with leak before break.

BASIS FOR EARLY CORE FAILURE

The assumption that the initiation of gap release occurs at 10 - 25 seconds appears to be based on the assumption of large break LOCA with no reflood of the core.

LOCA PROVIDING GREATEST CORE DAMAGE FREQUENCY

Although the large break LOCA is being considered in draft NUREG-1465 as the initiating event for core damage, there is a much greater probability that core damage would result from some smaller LOCA.

	Frequency	Percentage
Large LOCA	1.6E-8	7.5
Vessel rupture	3.0E-8	14.1
All small & medium LOCAs	1.673E-7	78.4
Total	2.133E-7	100

The event sequence having the greatest probability for core damage is the safety injection line break (this is the rupture of one of the two IRWST gravity feed lines) with failure of the remaining feed line to deliver flow. The probability of this event sequence is 6.8E-8 events per year (32% of the total LOCA core damage frequency). This is somewhat below the EPRI defined frequency limit for consideration as a design basis event of 1.0E-7.

PROBABILITY OF EARLY CORE DAMAGE FOR A LARGE BREAK LOCA

With either of the two accumulators injecting the core would rapidly reflood, preventing fuel clad temperature from approaching 2200°F for more than an hour.

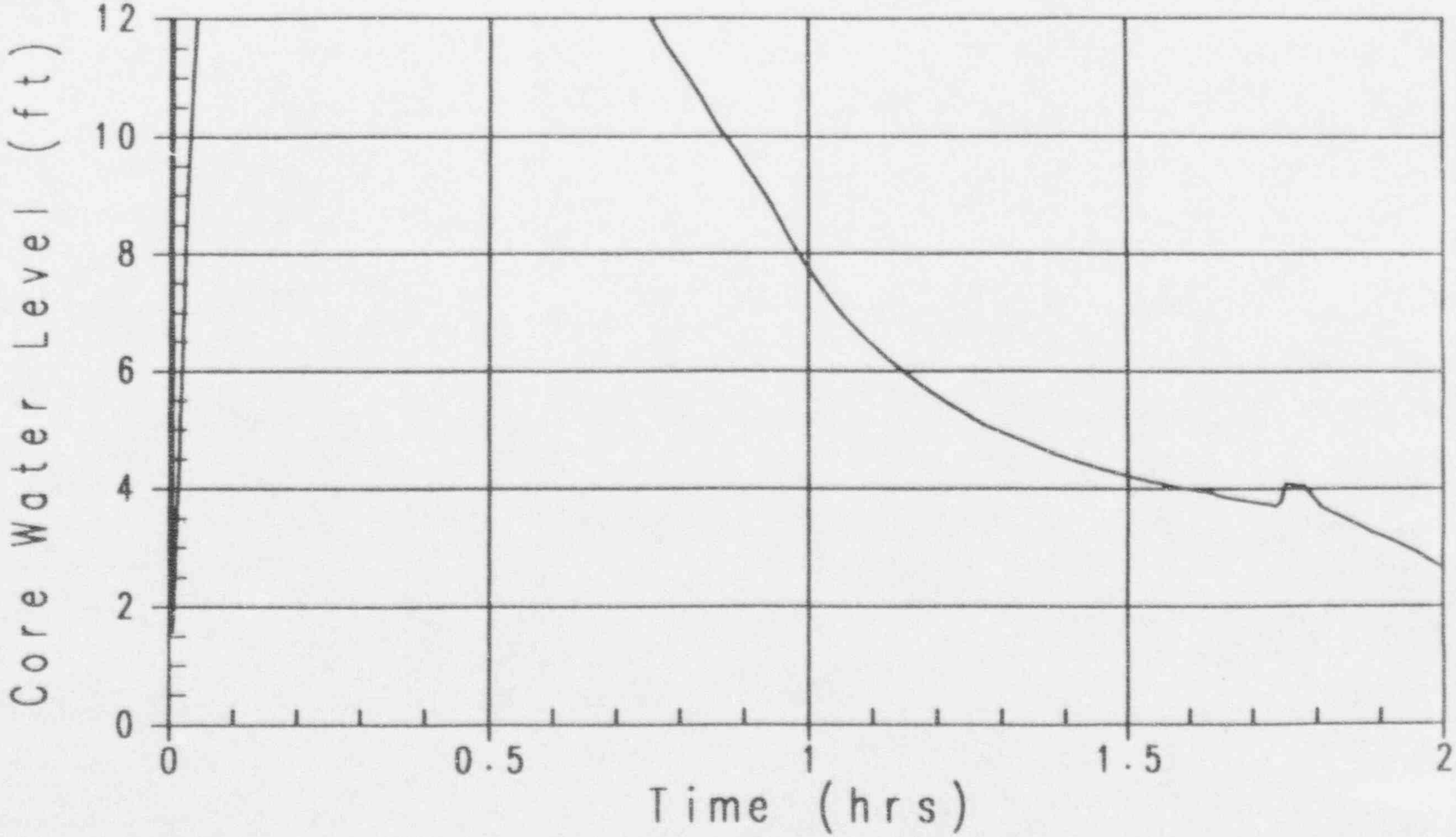
The only way to achieve core damage is to assume a sequence that involves multiple failures.

Even for a LOCA with both accumulators failing to inject (probability of 6.2E-9 events per year), core temperatures would not exceed 1700°F.

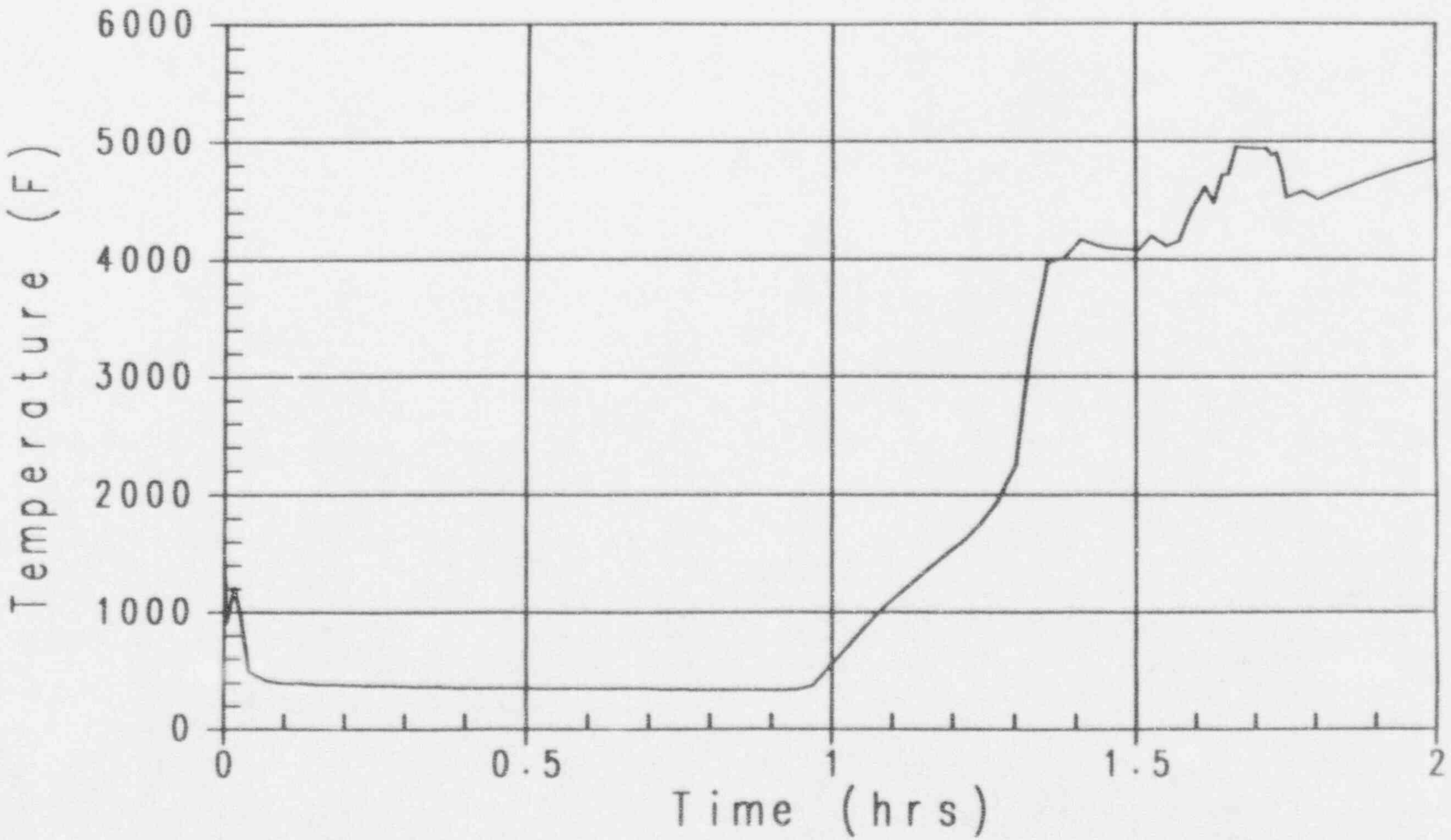
To achieve early core melt, it is necessary to fail both accumulators and one core makeup tank (probability of 4.9E-13 events per year).

The most probable sequence for a large break LOCA to result in core damage is the failure of the IRWST to inject due to common cause failure of the check valves in the gravity feed lines. Probability is 1.5E-8 events per year. This event has delayed core damage.

LARGE LOCA WITH GRAVITY FEED FAILURE
CORE WATER LEVEL vs. TIME



LARGE LOCA WITH GRAVITY FEED FAILURE
CORE TEMPERATURE vs. TIME



CONCLUSIONS

The 10 minute upper limit for delaying the gap release should not be instituted.

NUREG-1465 should be revised to permit timing of the gap release phase to be based on plant specific analysis.

The use of a one hour delay in gap release for the AP600 should be accepted.

ABB-CE COMMENTS ON NRC ISSUE 9 - CONTAINMENT SPRAY

DOMINANT FISSION PRODUCT REMOVAL MECHANISM FOR SYSTEM 80+

ABB-CE APPROACH/ASSUMPTIONS FOR SPRAYS

- PARTICULATE λ CALCULATED WITH SWNAUA
- ELEMENTAL λ SET EQUAL TO PARTICULATE λ BASED ON SRP 6.5.2, REV 2 (I.E., $\lambda_S > \lambda_W$ USING $A_{PARTICULATE} > \lambda_P$)
- NO DF LIMIT FOR ELEMENTAL
- pH IN IRWST ASSUMES LINEAR $Na_3PO_4*12H_2O$ ADDITION OVER 2.5 HOURS WITH SENSITIVITY STUDY FOR 7.5 HOURS, pH $\geq$ 7.0
- NO ORGANIC IODINE REMOVAL
- MIXING USES EPRI EVOLUTIONARY PLANT SOURCE TERM REPORT
- CsOH HYGROSCOPICITY NEGLECTED DUE TO TIME CONSTRAINTS

Enclosure 5

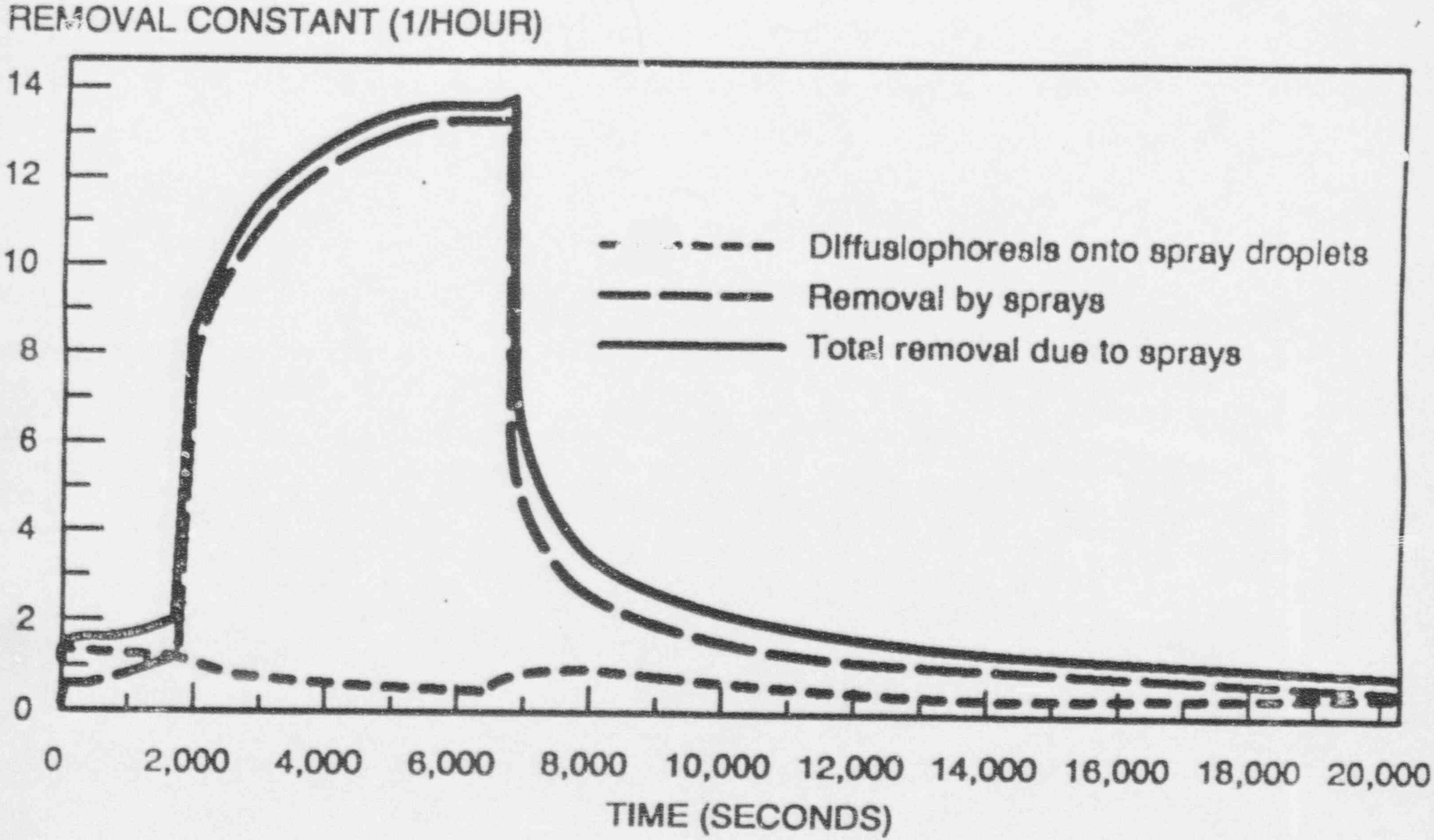
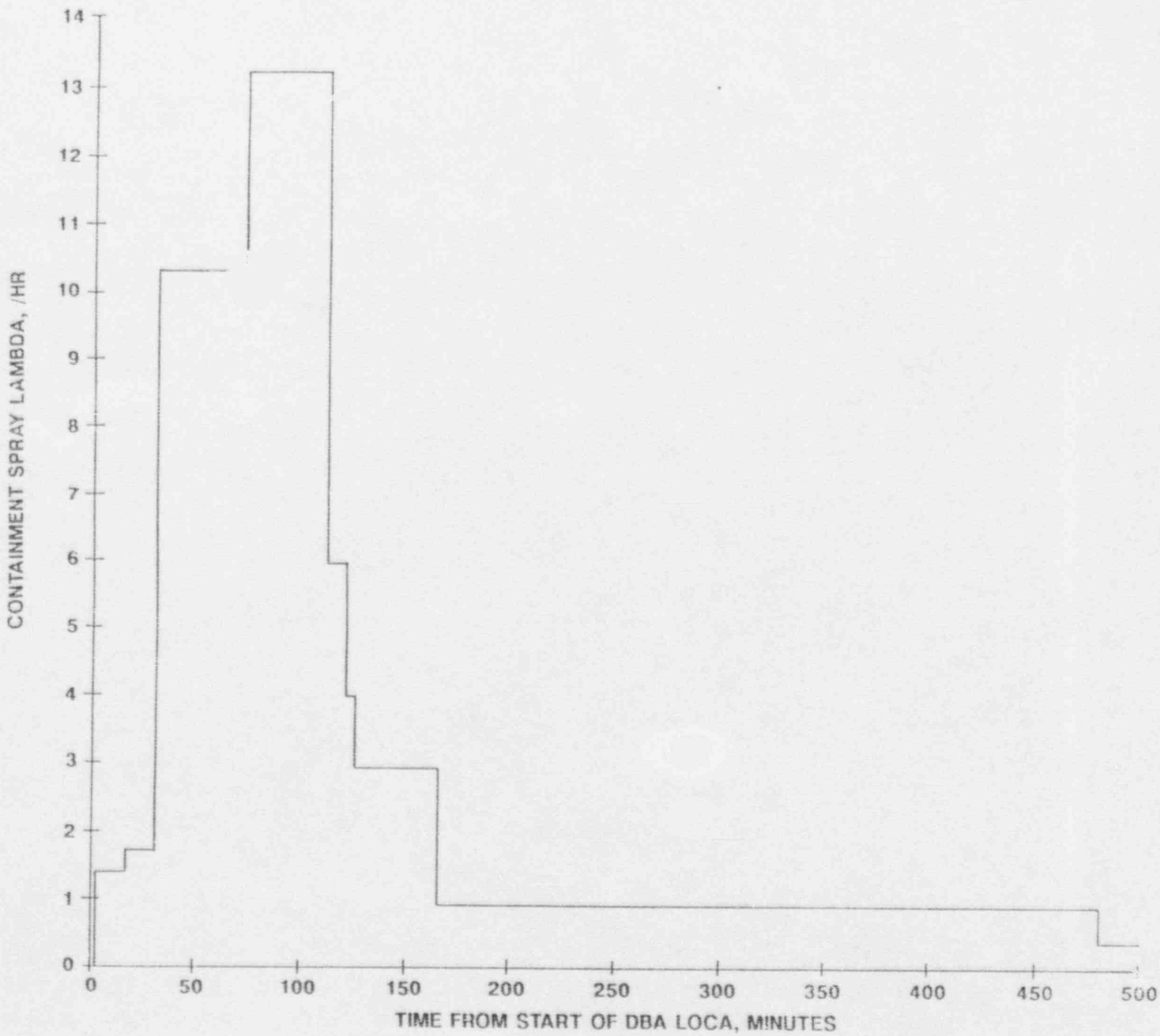


FIGURE 1

SYSTEM 80^{+} SPRAY REMOVAL



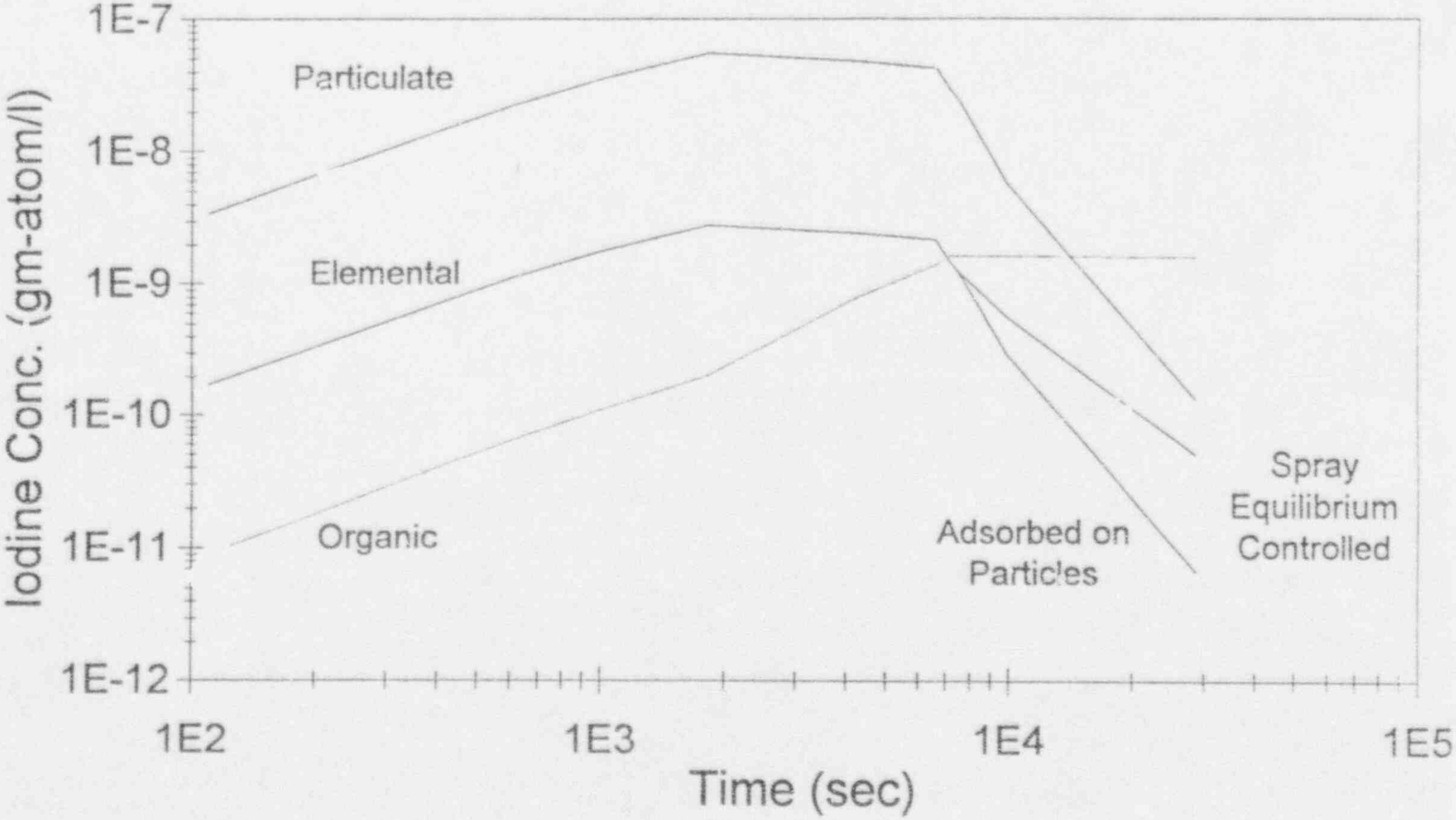
SYSTEM 80+™

DBA LOCA SPRAY LAMBDA

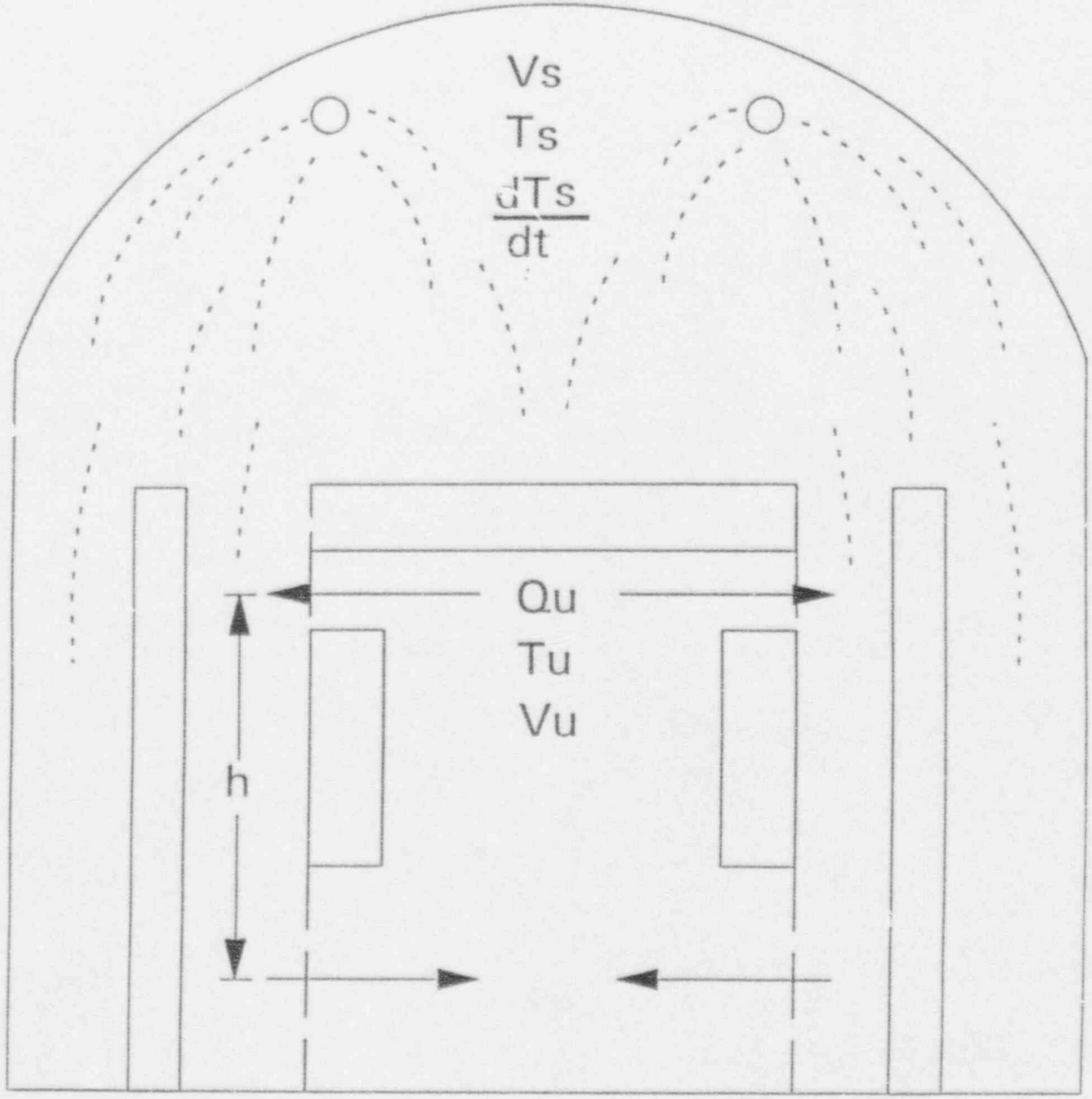
Figure 6.5-5

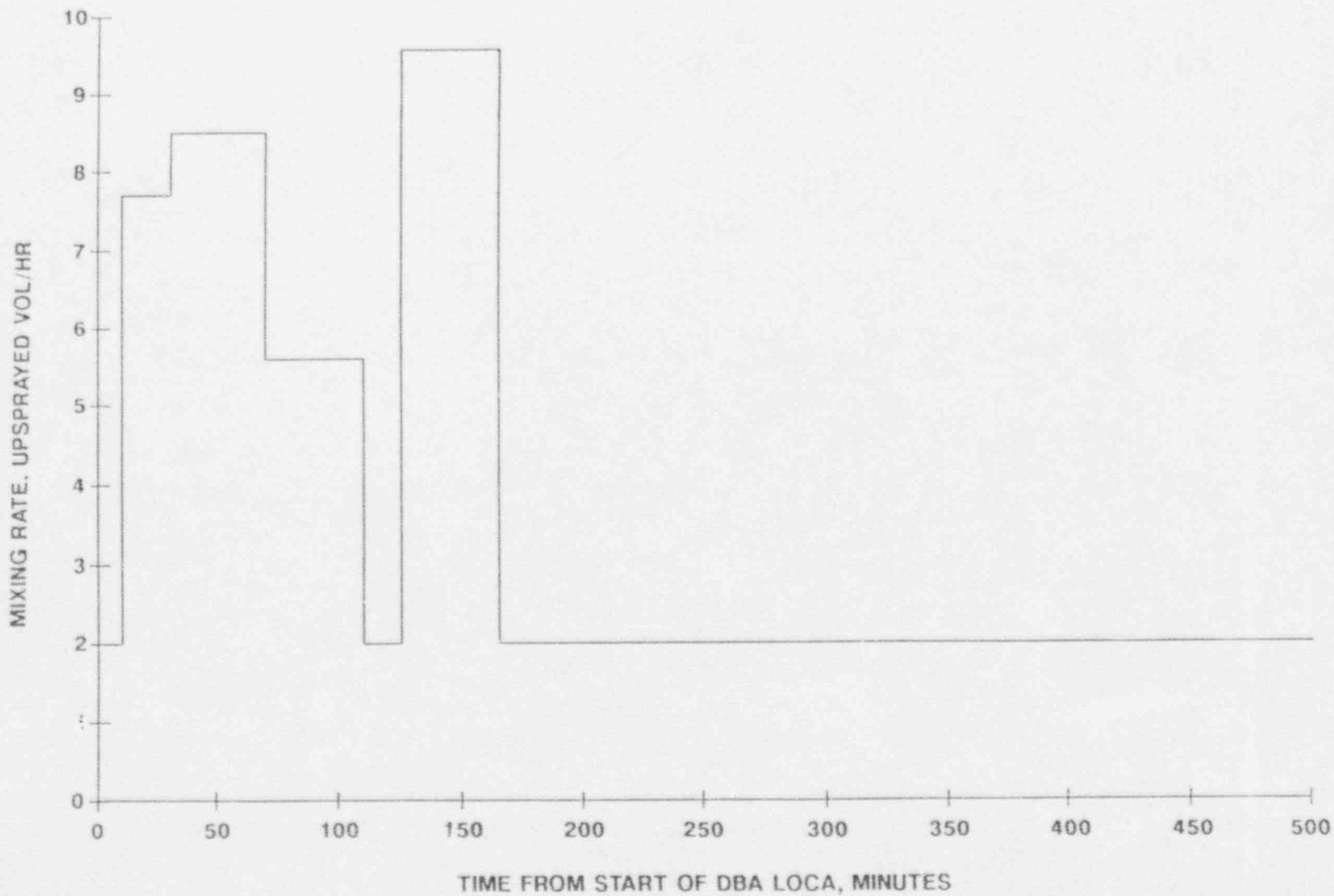
Amendment R
July 30, 1993

Airborne Iodine Concentrations



CONTAINMENT SPRAY MIXING





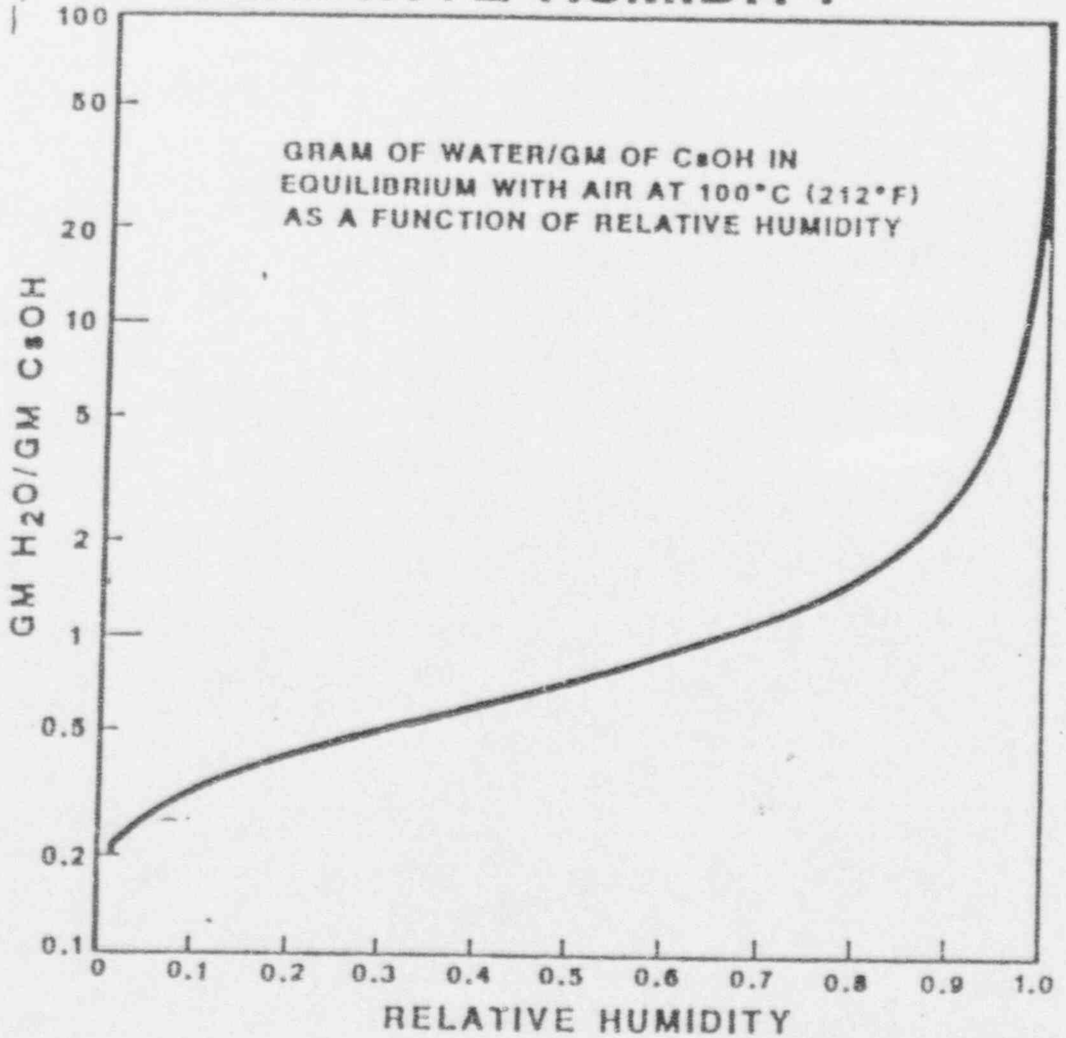
Amendment R
July 30, 1993

SYSTEM 80+™

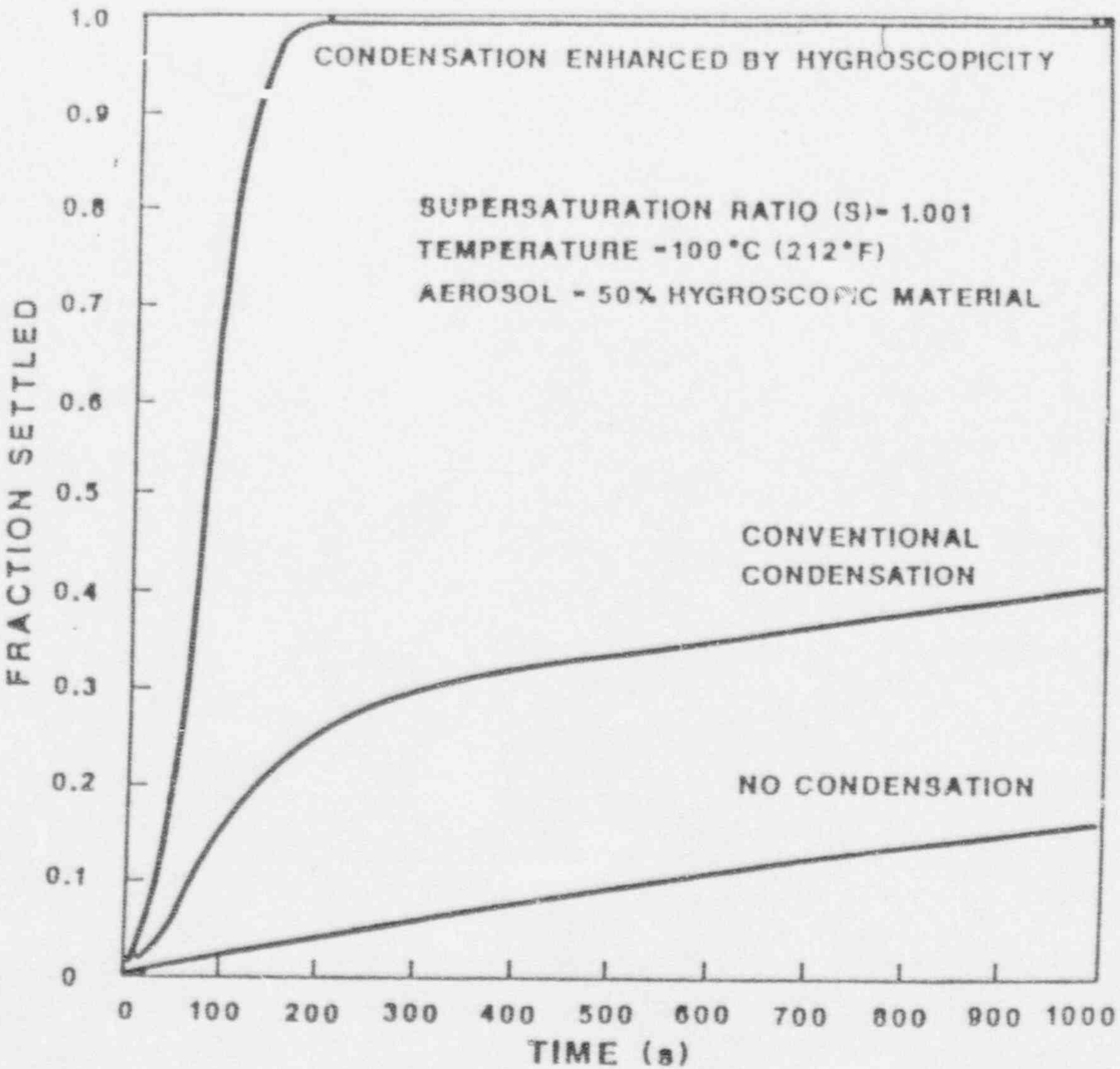
DBA LOCA CONTAINMENT MIXING RATE

Figure 6.5-4

GRAM OF H_2O/GRAM OF CsOH VS RELATIVE HUMIDITY



FRACTION SETTLED VS TIME



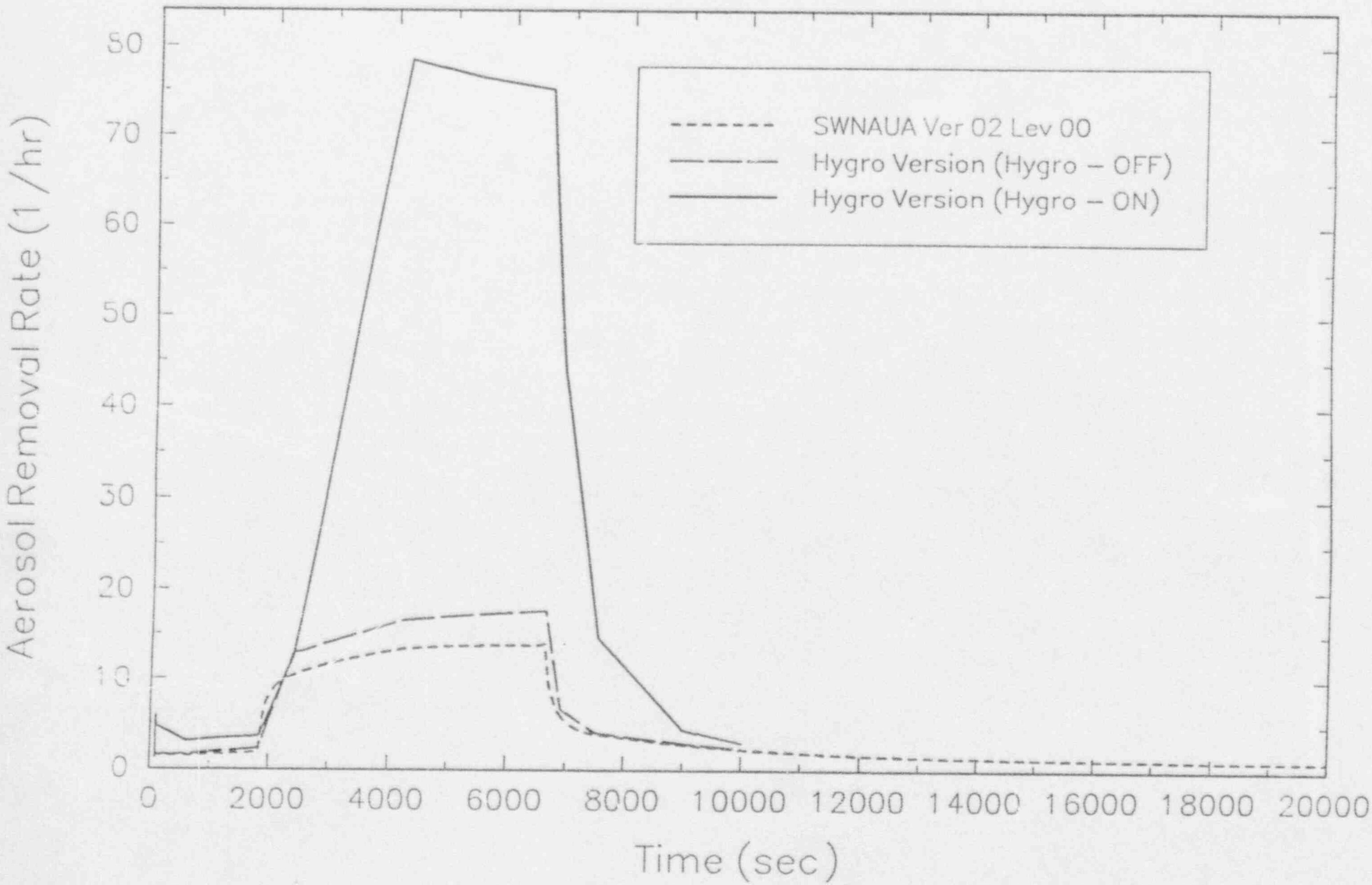

SWNAUA Ver 02 Lev 00
Hygro Version (Hygro – OFF)
Hygro Version (Hygro – ON)
Aerosol Removal Rate (1 /hr)
Time (sec)
0
10
20
30
40
50
60
70
80
0
2000
4000
6000
8000
10000
12000
14000
16000
18000
20000

ABB-CE COMMENTS ON NRC ISSUE 9 - CONTAINMENT SPRAY

- CONTINUED -

SUMMARY OF NUREG/CR-5966 (SIMPLIFIED SPRAY MODEL)

- AS USUAL, HIGH QUALITY WORK
- AS USUAL, DON'T LIKE EVERYTHING
- COMMENT PERIOD PRIOR TO ISSUANCE (AS WITH SCRUBBING REPORT) WOULD HAVE BEEN DESIRABLE
- LIMITED DISCUSSION OF MIXING
- NO TREATMENT OF HYGROSCOPICITY

MEAN VALUE FOR 0.01 $CM^3/S\text{-}CM^2$, 2000-3000 CM FALL DISTANCE, STEADY-STATE (FRACTION REMAINING 0.9) $\approx$ 10 HR^{-1}

SWEC REGARDS ABSENCE OF HYGROSCOPICITY IN MODEL AS SIGNIFICANT CONSERVATISM

ABB-CE COMMENTS ON NRC ISSUE 9 - CONTAINMENT SPRAY

- CONTINUED -

HYGROSCOPICITY OF IN-VESSEL RELEASE BEING INCLUDED IN BWR POOL SCRUBBING MODEL - SPRAY MODEL COULD BE SUPPLEMENTED

SUPPLEMENT COULD ALSO EXPAND MIXING DISCUSSION

RADIATION EQ/SURVIVABILITY FOR SYSTEM 80+

ABB-CE HAS DEFINED THE 10CFR100 DBA AS A LOW-PRESSURE CORE MELT REPRESENTING 94% OF CORE MELTS IN PRA

SDS RDVs PROVIDE FOR RAPID DEPRESSURIZATION IN THE EVENT OF CORE DAMAGE BACKED UP BY HYDROGEN VENTING FROM RV HEAD

INTEGRITY OF CONTAINMENT AND OPERABILITY OF CONTAINMENT SYSTEMS MUST BE ENSURED FOR 10CFR100 DBA, INCLUDING THE CAPABILITY OF REMOVING DECAY HEAT FROM IN-VESSEL CORE DEBRIS IF DEBRIS OTHERWISE COOLABLE

TWO SAFETY-RELATED SYSTEMS ARE NOT REQUIRED FOR REMOVAL OF DECAY HEAT WITH REACTOR AT LOW PRESSURE:

- EFW
- SIS

RADIATION EQ/SURVIVABILITY FOR SYSTEM 80+

- CONTINUED -

QUALIFICATION OF THESE SYSTEMS FOR 10CFR100 DBA SOURCE TERM PER 10CFR50.49/10CFR50.35(f) IS NOT REQUIRED EXCEPT FOR TRANSITION PERIOD CONSERVATIVELY ASSUMED TO BE 72 HOURS

VITAL ACCESS ANALYSES WOULD CONSIDER 10CFR100 DBA SOURCE TERM IN SIS LINES

QUALIFICATION BASIS FOR EFW/SIS WILL BE 100 DAYS AT 100% GAP RELEASE PLUS MARGIN

DEFINITIONS:

- LEVEL 1 - 100% GAP RELEASE AS DEFINED BY DRAFT NUREG-1465
- LEVEL 2 - 100% GAP RELEASE PLUS EARLY IN-VESSEL AS DEFINED BY DRAFT NUREG-1465

RADIATION EQ/SURVIVABILITY FOR SYSTEM 80+

- CONTINUED -

LEVEL 1 IMPROVED TO INCLUDE INSTANTANEOUS 20% GAP RELEASE (WILL COVER LEVEL 2, AS WELL)

LEVEL 1 WILL BE APPLIED WITH SUFFICIENT MARGIN TO ENSURE 100 DAYS @ LEVEL 1 $\geq$ THREE DAYS @ LEVEL 2

APPROXIMATE DURATION EQUIVALENCY IN DAYS FOR GAMMA

(GAMMA/BETA) ← (SUMP AIRBORNE)

Time at Level 1	Time at Level 2	Time for Severe Accident
	180/180	30/180
	100/100	10/100
180/180	4/4	1/4
100/100	3/3	0.8/3
30/30	1/1	0.5/1
-/3	-/0.3	-/0.3

SOURCE TERM IMPACT ON EMERGENCY PLANNING - ABB-CE POSITION

ABB-CE SUPPORTS SIMPLIFIED EMERGENCY PLANNING FOR ALWRs AS PROPOSED BY EPRI

THE KEY ELEMENTS ARE:

- MAINTENANCE OF CONTAINMENT INTEGRITY FOR MOST CORE MELTS
- DEFINITION OF A BOUNDING SOURCE TERM FOR CORE MELT WITH CONTAINMENT INTEGRITY
- DEMONSTRATION THAT FOR MOST WEATHER CONDITIONS, THE PAGs WILL NOT BE EXCEEDED AT THE SITE BOUNDARY USING ALWR METEOROLOGY

FOR SYSTEM 80+ MOST CORE MELTS DO NOT FAIL CONTAINMENT OR RESULT IN LOSS OF CONTAINMENT SYSTEMS

SOURCE TERM IMPACT ON EMERGENCY PLANNING - ABB-CE POSITION

- CONTINUED -

PAG ANALYSIS IDENTICAL TO 10CFR100 DBA EXCEPT:

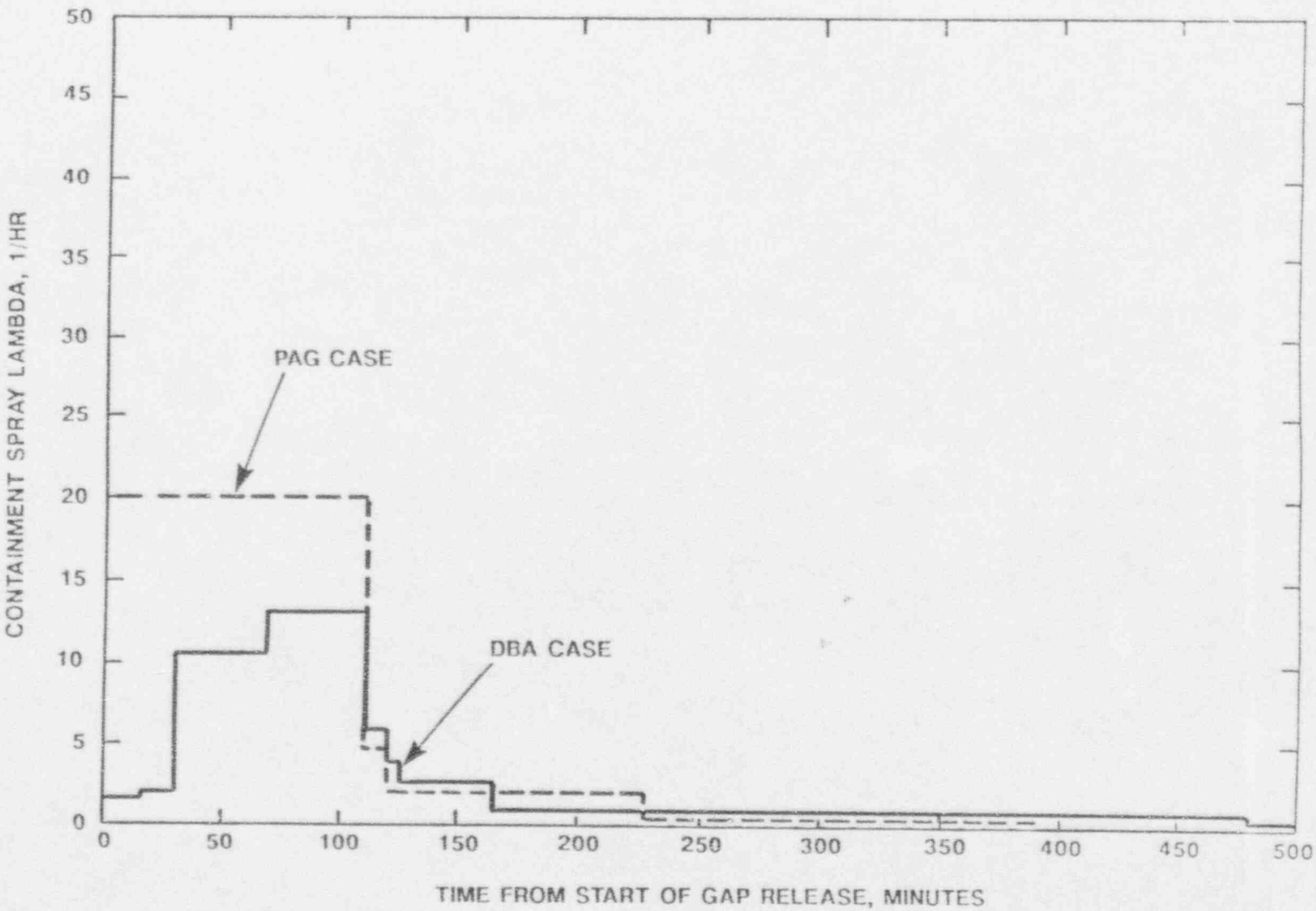
- SOURCE TERM INCLUDES DRAFT NUREG-1465 EX-VESSEL (WITH DF OF 10 FOR CAVITY FLOOD) AND LATE IN-VESSEL
- SOME CREDIT FOR CsOH HYGROSCOPICITY
- DOSE CALCULATION PERFORMED WITH MACCS (MEDIAN DOSE REPORTED PER URD)

RESULTS:

- 0.33 REM CEDE
- 2.7 REM THYROID

PAGs ARE ONE REM CEDE, FIVE REM THYROID

	PAG	DBA
• Noble gas:	100%	100%
• Iodine:	50%	40%
• Cesium	40%	30%
• Tellurium	20%	15%
• Strontium	4.2%	3%
• Barium	5%	4%
• Ruthenium	0.84%	0.8%
• Cerium	1.2%	1%
• Lanthanum	0.35%	0.2%



System 80+™

LOCA SPRAY LAMBDAS

Figure 15.6.5-1

Amendment R
July 30, 1993

SOURCE TERM IMPACT ON EMERGENCY PLANNING - ABB-CE POSITION

- CONTINUED -

GIVEN THE ABOVE, IT IS EVIDENT THAT EMERGENCY RESPONSE ISSUES FOR SYSTEM 80+ WOULD HAVE LESS IMPACT ON RISK THAN THE ALREADY SMALL IMPACT EVIDENT IN NUREG-1150

ABB-CE IS OF THE OPINION THAT EMERGENCY PREPAREDNESS FOR SYSTEM 80+ SHOULD BE VIEWED IN THE CONTEXT OF PREPAREDNESS FOR OTHER LARGE INDUSTRIAL AND TRANSPORTATION EMERGENCIES

- GENERAL PREPAREDNESS OFFSITE

- SPECIAL PREPAREDNESS ONSITE