February 20, 1992

Docket No. 50-605

APPLICANT: General Electric Company (GE)

PROJECT: Advanced Boiling Water Reactor (ABWR)

SUBJECT: SUMMARY OF MEETING HELD ON JANUARY 22, 1992

A public meeting was held between the Nuclear Regulatory Commission (NRC) staff and GE representatives at NRC headquarters in Rockville, Maryland, on January 22, 1992, from 9:00 a.m. to 4:00 p.m. The purpose of this meeting was for GE to present the status of their probabilistic risk assessment sensitivity and uncertainty analyses for the ABWR. Enclosure 1 is a list of the attendees, and Enclosure 2 is a copy of the handout presented by GE.

original signed by: Victor M. McCree
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Standardization Project Directorate
Division of Advanced Reactors
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Enclosures:
1. Attendees List
2. GE Handout

cc w/enclosures:
See next page

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Enclosure 1

GENERAL ELECTRIC'S (GE'S) OVERVIEW
OF THE ADVANCED BOILING WATER REACTOR

JANUARY 22, 1992

Name		Organization	Mail Stop
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R. Sherry	(GE contractor)	GKA	
J. Gabor	(GE contractor)	GKA	
G. Bagchi		NRC/NRR/ESGB	7-H-15
R. Nease		NRC/NRR/PDST	11-H-3
T. D'Angelo		NRC/NRR/SPLB	8-D-1
A. Drozd		NRC/NRR/SPLB	8-D-1
R. Palla		NRC/NRR/PRAB	10-E-4
B. Hardin		NRC/RES/ARB	NLS-169
J. Jo		Brookhaven National Laboratory	
R. Youngblood		Brookhaven National Laboratory	

GE Nuclear Energy

Advanced Boiling Water Reactor
PRA Backend Analyses

Presented to

NRC - NRR

Carol Buchholz
Jeff Gabor
Rick Sherry

January 22, 1992
White Flint Building

Enclosure 2

Items for discussion

- *Results of Literature Survey*
- *Planned Sensitivity Studies*
- *Details of DCH Analysis*
- *Core Concrete Interaction*
  - *Top events of DET*
  - *Determination of pedestal strength if sidewall erosion occurs*
- *BNL Report on ABWR*
- *Preliminary information on vacuum breakers, drywell head and rupture disk interactions*
- *ATWS issues from DSER*

Sources Reviewed for Possible ABWR CET Events

- *NUREG/CR 4551 Evaluation of Severe Accident Risks: Grand Gulf, Unit 1*
- *Kuosheng BWR/6 Mark III PRA*
- *NUREG/CR-4551 Evaluation of Severe Accident Risks: Peach Bottom*
- *NUREG-1150*
- *Generic Letter 88-20*
- *NUREG-1335*

NUREG/4551 Grand Gulf APET

Question Subject	No. of questions
• Plant damage state	16
• Structural capacities	4
• Systems availability	12
• AC/DC power	6
• Hydrogen issues	47
• Containment / drywell pressure	23
• CCI / pedestal failure	8
• Steam explosions	4
• Core damage progression / vessel breach	4
• Criticality	1

Other Questions Identified

Questions from other studies

- *Liner melt-through*

Questions specific to ABWR design

- *Flooder operation*
- *Rupture disk characteristics*

Sensitivity Study for ABWR

Information on key issues gathered from the following sources

- *NUREG - 1150*
- *NUREG - 1335*
- *Recommended Sensitivity Analyses for an Individual Plant Examination using MAAP 3.0B (EPRI)*
- *Advanced Light Water Reactor Requirements Document, Volume II, Chapter 1, Appendix A: Evolutionary Reactor Key Assumptions and Guidelines*
- *ARSAP/EPRI/NRC Technical Exchange Meetings*
- *Draft letter report on ABWR Uncertainty Analysis*

Sensitivity Issues to be Considered

- *In-Vessel*
  - *Hydrogen generation*
  - *Fission product release from core*
  - *CsI re-evaporation*
  - *Time of vessel failure*

- *Ex-Vessel*
  - *Debris entrainment and direct containment heating*
  - *Steam explosions*
  - *Debris to water heat transfer*
  - *Debris to curst heat transfer*
  - *Containment failure location*
  - *Containment failure area*

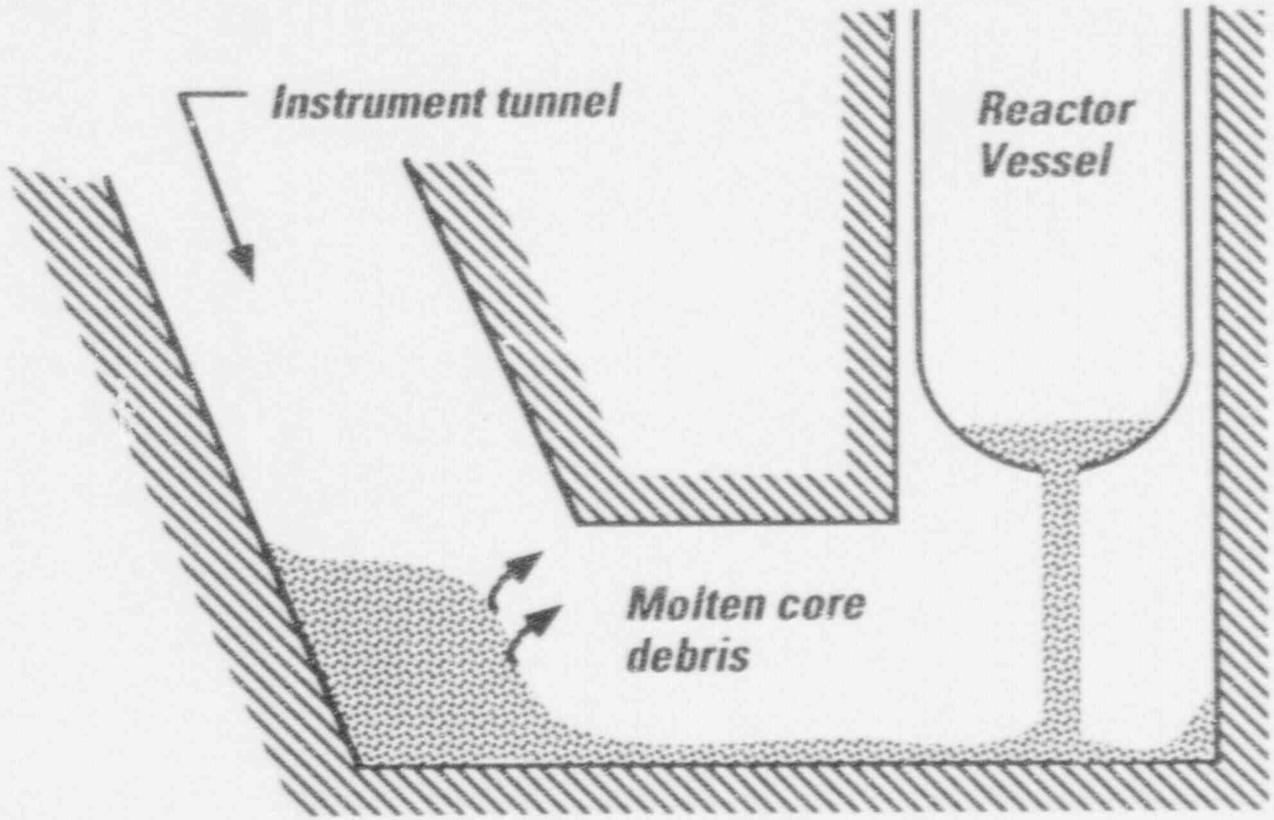
High Pressure Melt Ejection and Direct Containment Heating

- *Sensitivity step skipped and uncertainty analysis performed based on*
  - *Staff interest in topic*
  - *GE belief that this might be an important issue for ABWR*
- *Goal for today's discussion is staff concurrance on details of treatment*

Basic Entrainment Phenomena

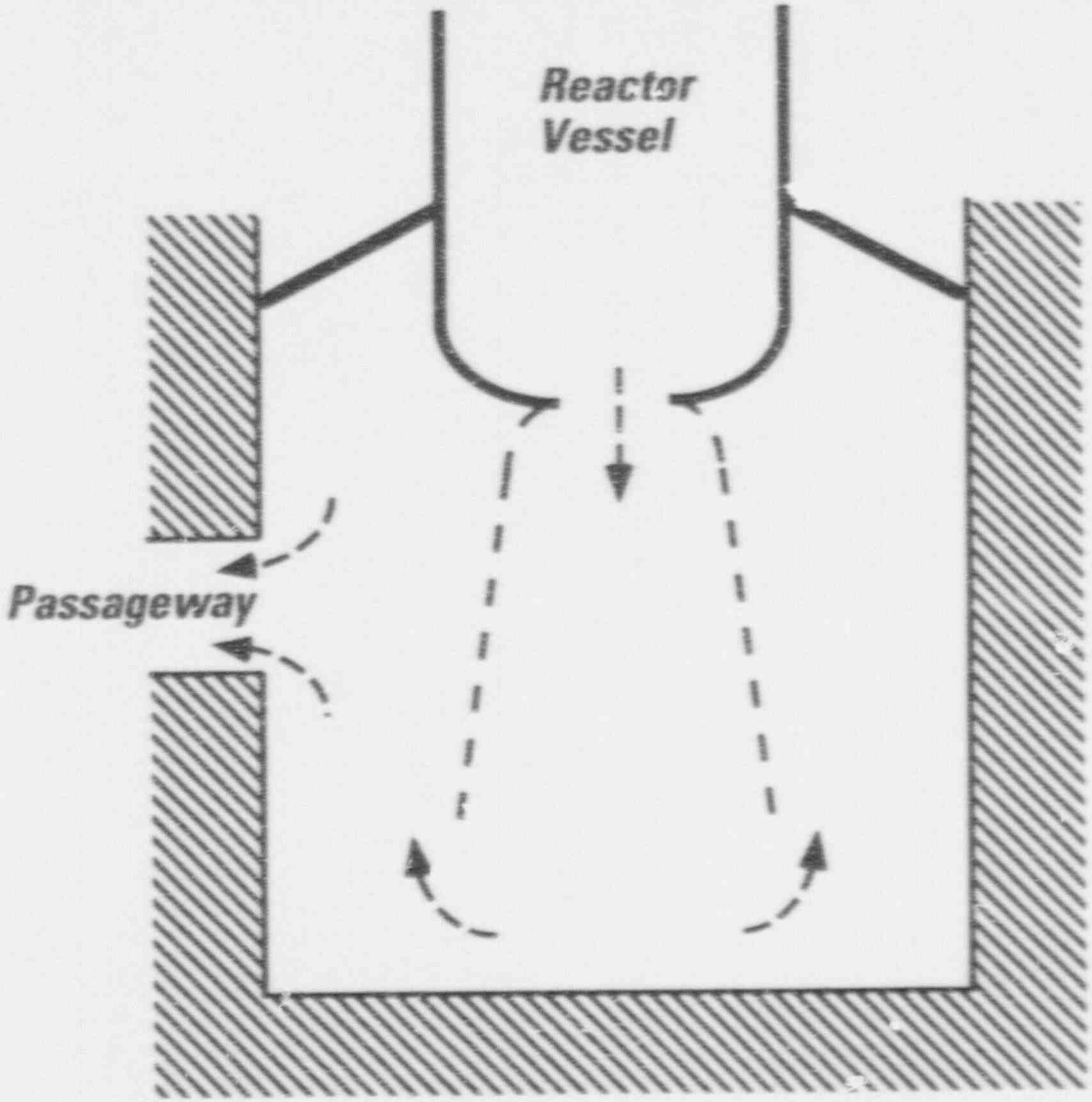
- *Entrainmnet occurs due to stability between the gas and liquid*
- *Two mechanisms may strip droplets from surface*
  - *Helmholtz instability*
  - *Taylor instability*
- *Debris which has been stripped from surface remains entrained until velocities fall below free field Taylor limit*

Entrainment due to Helmholtz instability



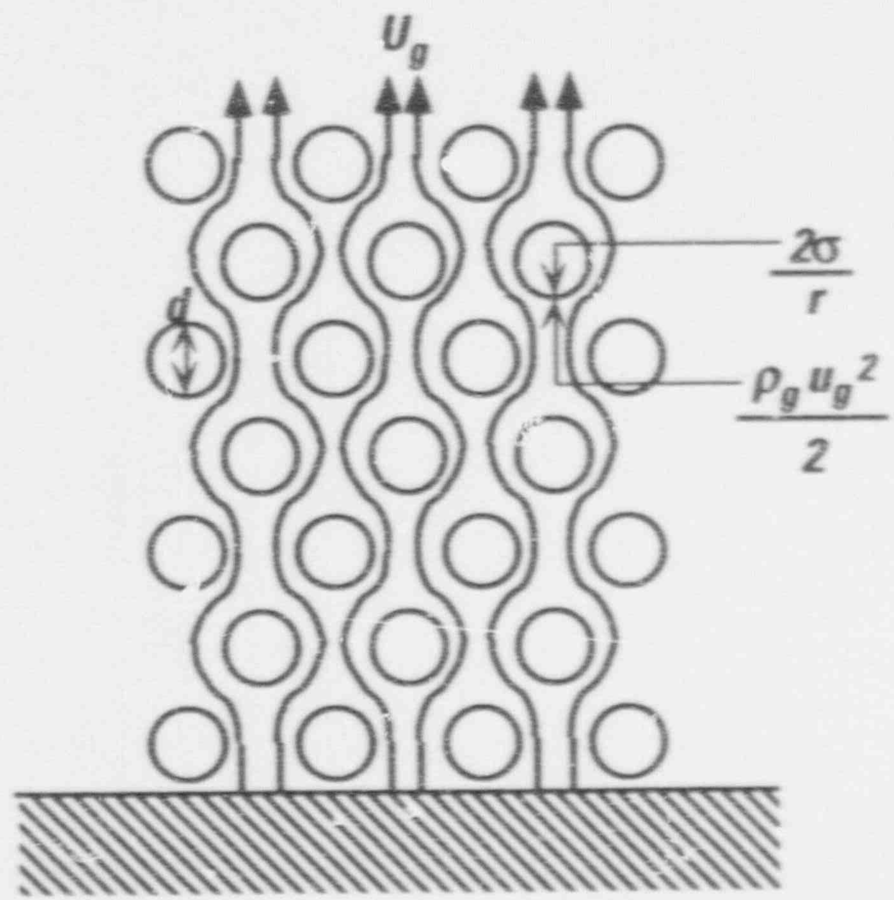
Mechanism proposed for PWRs

Entrainment due toTaylor instability

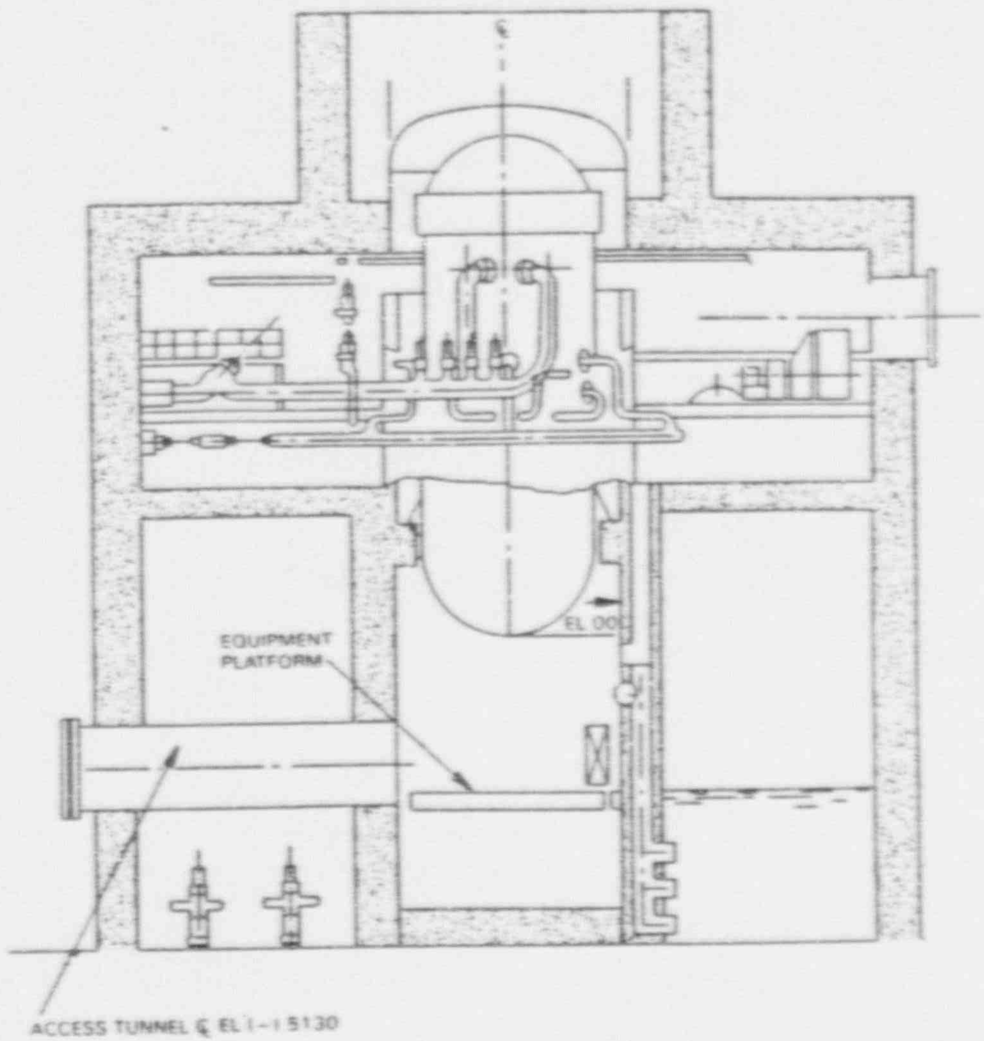


Probable mechanism for BWRs

Taylor Instability

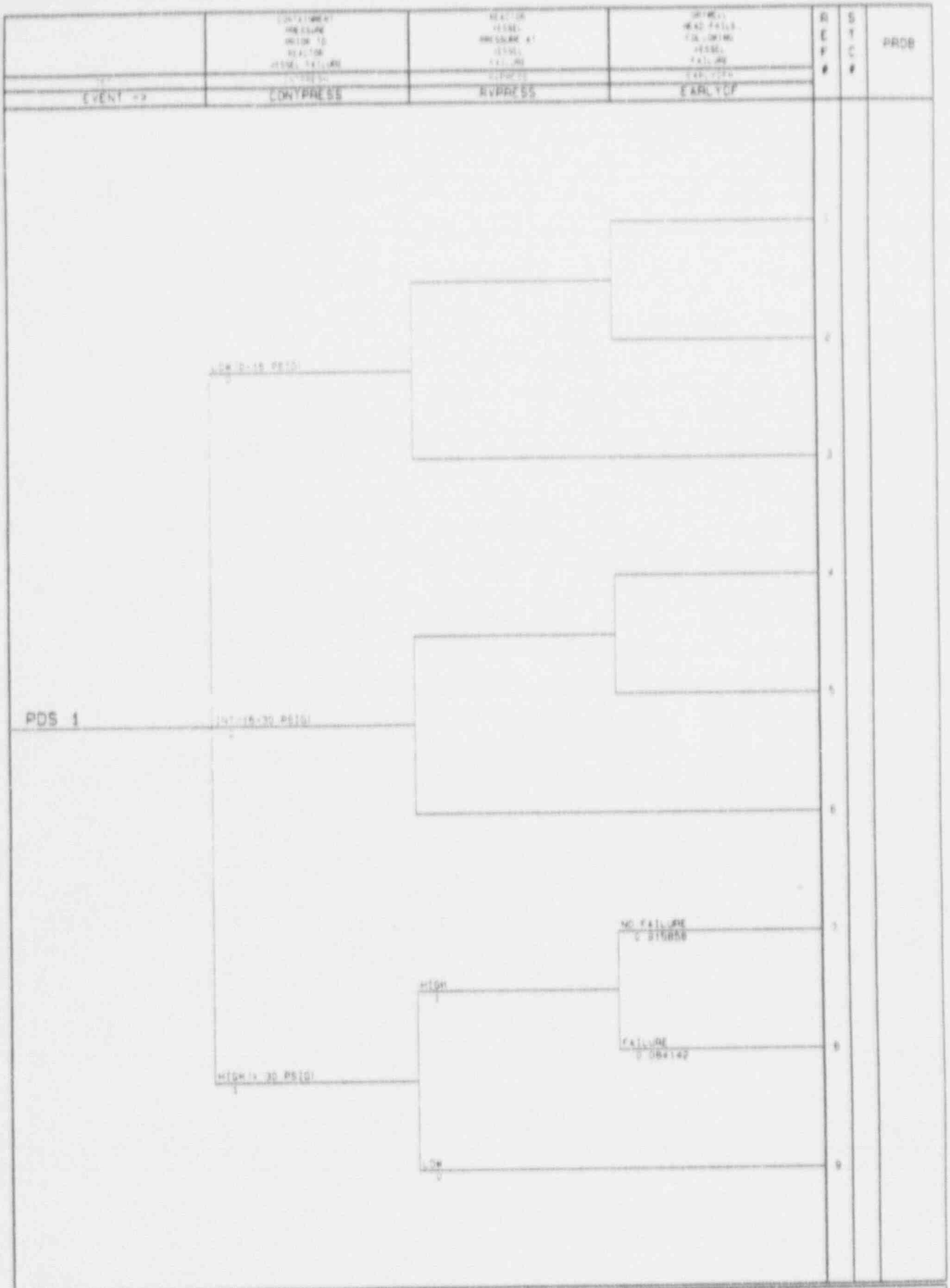


ABWR Containment Configuration



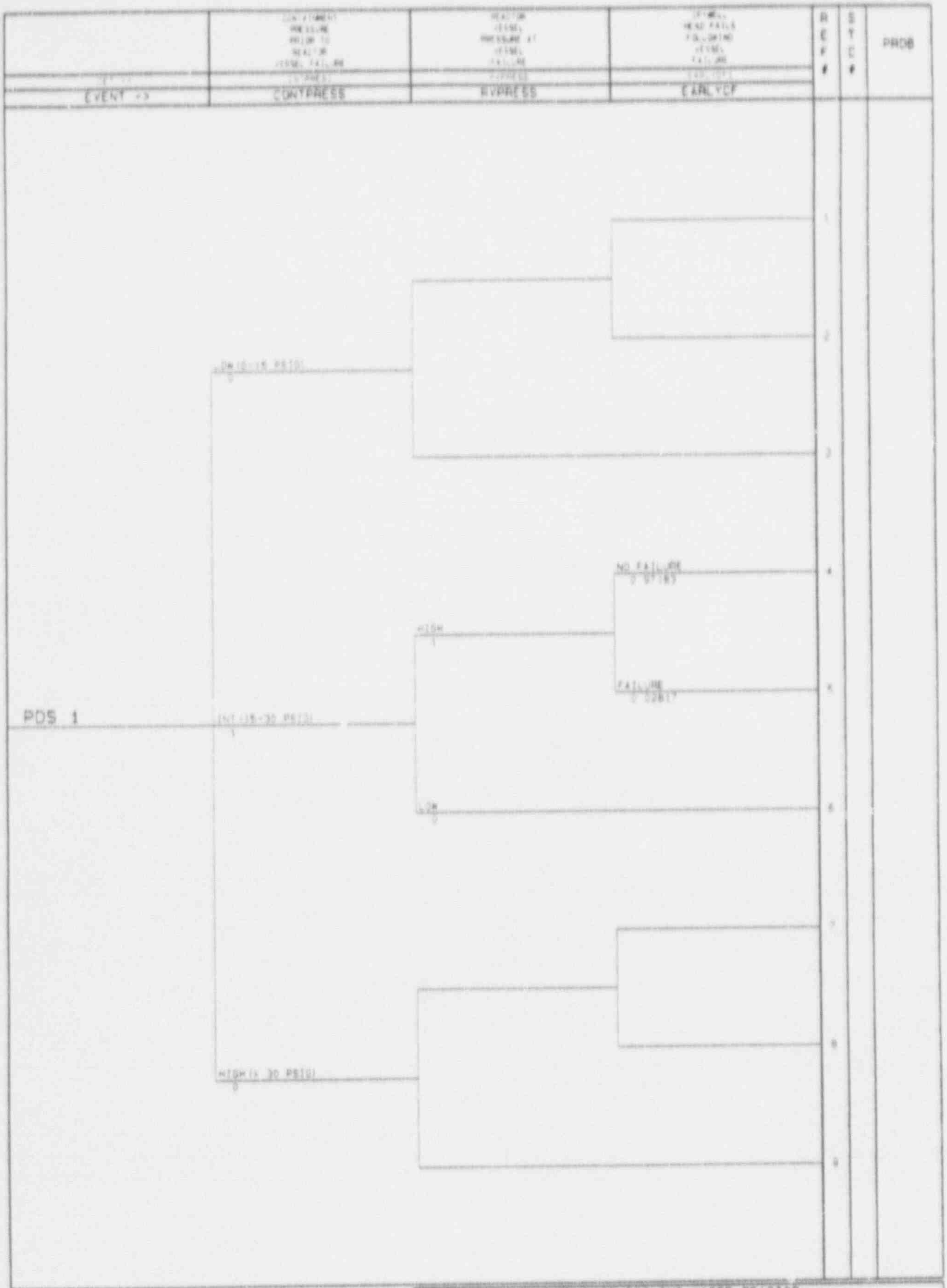
HPME and DCH Uncertainty Modelling

- *Consists of a main tree and three decomposition event trees (DETs)*
  - *Main tree sorts sequences by important plant damage state attributes*
  - *DETs assess probability of drywell head failure resulting from HPME and DCH*
- *Deterministic analysis performed to determine peak containment pressure*
  - *Plant damage state information used as fixed conditions*
  - *Uncertainty informatin varied for each plant damage state*
- *Peak pressure compared to drywell head fragility curve to determine probability of early containment failure*
- *Long term failure due to high temperature effects will be considered elsewhere*



ADVANCED BOILING WATER REACTOR
PROBABILISTIC RISK ASSESSMENT

CONTAINMENT EVENT EVALUATION
EVENT TREE FOR
PROBABILITY OF EARLY CONTAINMNET FAILURE (PAGE 1)



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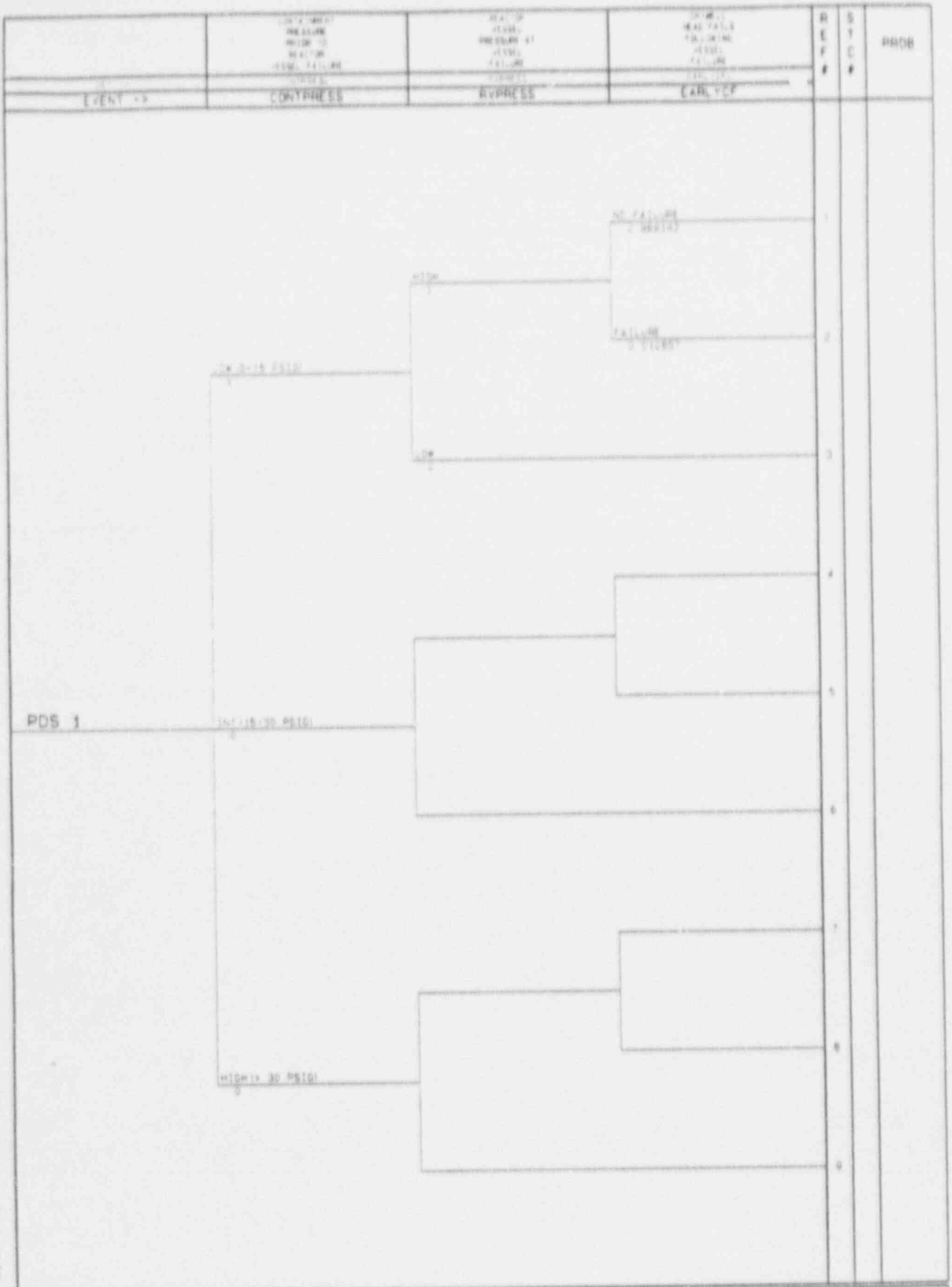

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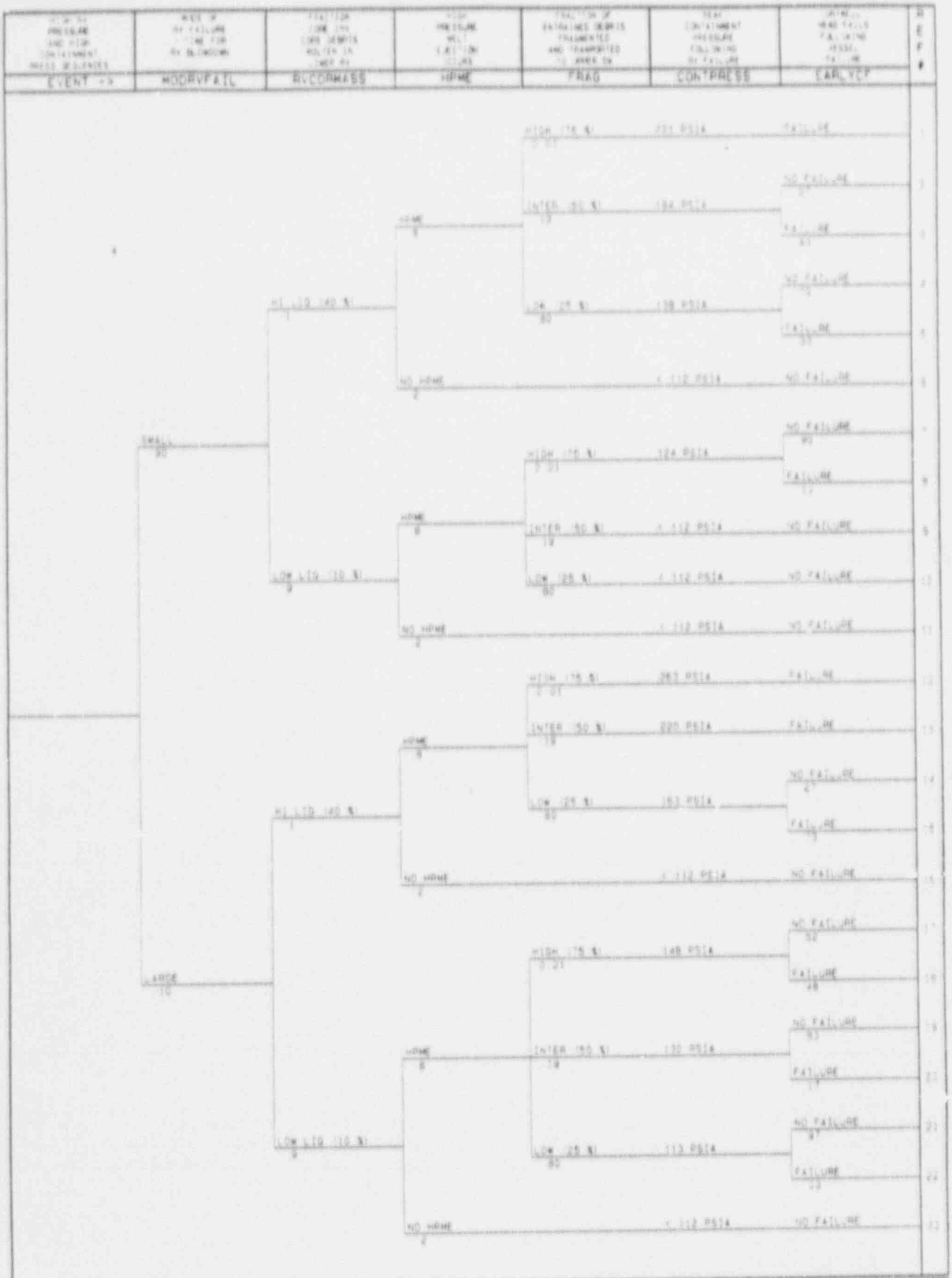

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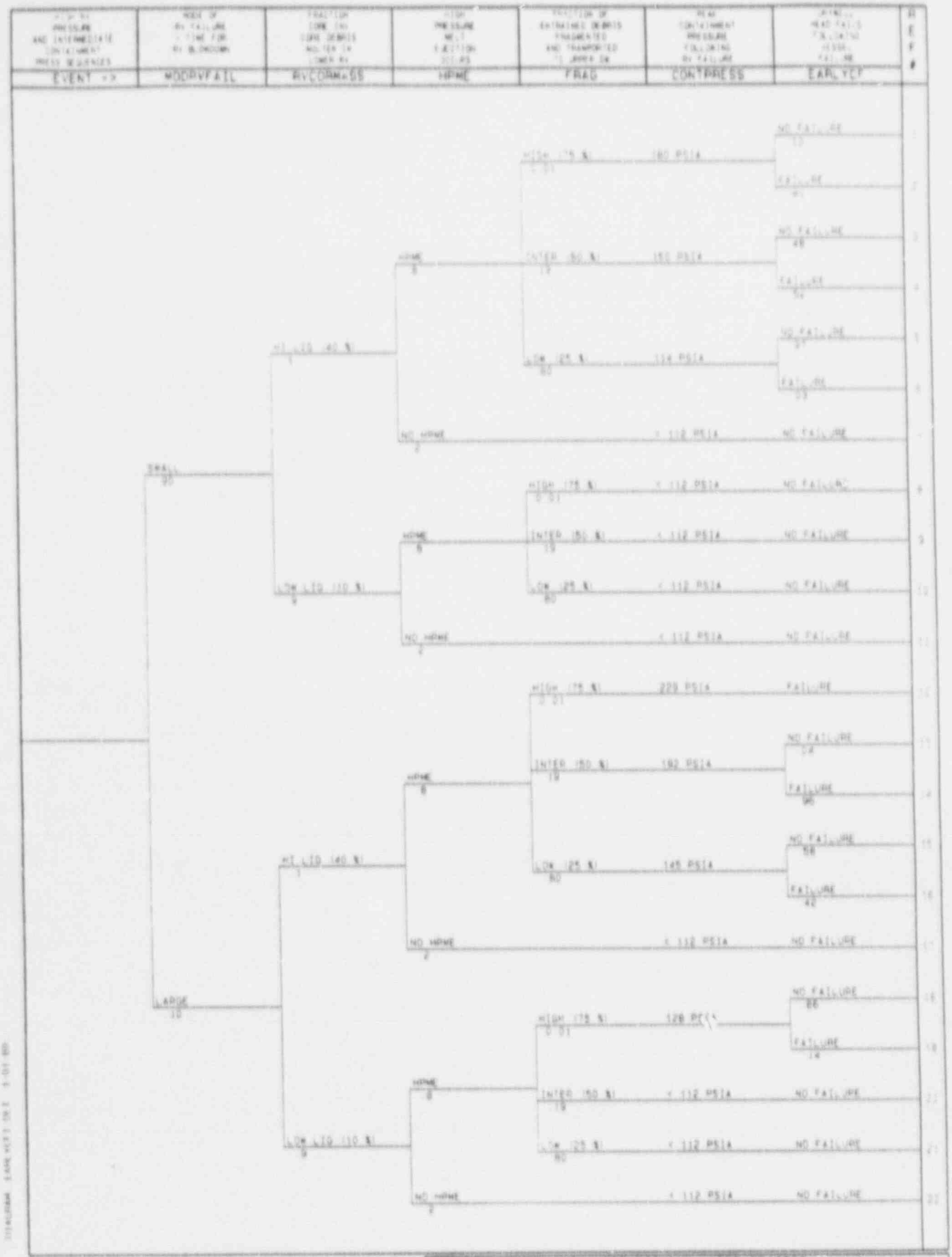
PROBABILISTIC RISK ASSESSMENT
CONTAINMENT EVENT EVALUATION
EVENT TREE FOR PROBABILITY OF EARLY CONTAINMNET FAILURE
PAGE (2A) - HIGH RV PRESS AND HIGH CONT PRESS SEQUENCES



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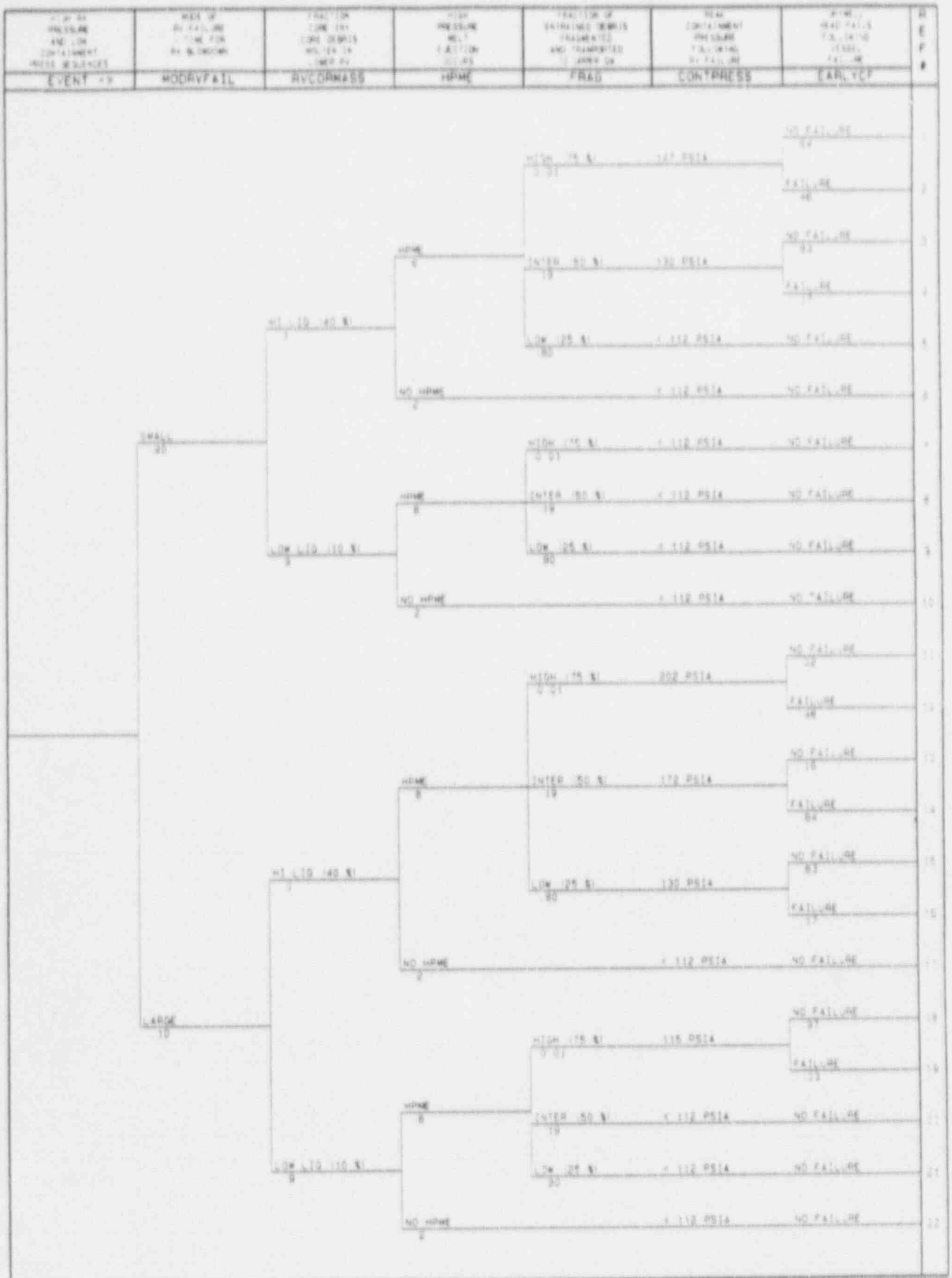
PROBABILISTIC RISK ASSESSMENT

CONTAINMENT EVENT EVALUATION

EVENT TREE FOR PROBABILITY OF EARLY CONTAINMENT FAILURE

PAGE (2B) - HIGH RV PRESS AND INTER CONT PRESS SEQUENCES



ADVANCED BOILING WATER REACTOR
PROBABILISTIC RISK ASSESSMENT

CONTAINMENT EVENT EVALUATION
EVENT TREE FOR PROBABILITY OF EARLY CONTAINMNET FAILURE
PAGE (2C) - HIGH RV PRESS AND LOW CONT PRESS SEQUENCES

Main DCH Trees

- Containment pressure prior to reactor vessel failure

  Three pressure regimes considered

  - Low (15-20 psia) Non-ATWS Sequences with operable containment heat removal or fast core damage
  - Intermediate (30-45 psia) LBLOCAs with no containment heat removal, SBO with RCIC and loss of containment heat removal
  - High (> 45 psig) ATWS with RCIC

- Reactor vessel pressure at time of vessel failure

  Two pressure regimes considered

  - Low (< 200 psia) Entrainment does not occur
  - High (≥ 200 psia) Entrainment can occur

Quantification of DCH DETs

- *Mode of reactor vessel failure*
  - *Two regimes considered*
    - *Small failure - Nominal size 0.1 sq m*
      - *Instrument tube, control rode drive tube or drain line penetration*
    - *Large failure - Nominal size 2.0 sq m*
      - *Lower head creep rupture failure*
  - *Quantified using insights from NUREG/CR-4551 Grand Gulf and Recent INEL Analyses (Rempe)*

- *Fraction of core inventory molten in lower head*
  - *Defines the maximum amount of debris that can participate in HPME and DCH*
  - *Two regimes considered*
    - *Small (0-20% of core inventory)*
    - *Large (20-60% of core inventory)*
  - *Quantified using NUREG-4551 mean values*

Quantification of DCH DETs (continued)

- High pressure melt ejection (HPME)
  - Evaluates whether a substantial fraction of the debris is dispersed from the lower drywell floor by high velocity gas flow
  - HPME can occur only for high pressure sequences
  - Branch probabilities taken from NUREG-4551 Grand Gulf based on the similarities between the Mark III pedestal cavity and the ABWR lower drywell

Quantification of DCH DETs (continued)

- *Fraction of entrained debris fragmented and entrained to the upper drywell*
  - *Four factors considered*
    - *Trapping of debris in the lower drywell*
    - *Impaction and removal of debris in the gas transport pathway to the upper drywell*
    - *Partitioning of entrained debris between the upper drywell and wetwell (based on gas flow)*
    - *Debris dispersed by wave formation rather than as small particles*
  - *Fraction of molten debris participating in an HPME that is fragmented and dispersed into the upper drywell has been represented by*

$$f_{frag} = f_{downcomer} \times f_{impaction} \times f_{split\text{-}upw} \times f_{part}$$

    - *where distributions were assumed for each parameter in the above relationship*

Quantification of DCH DETs (continued)

- Fraction of entrained debris fragmented and entrained to upper cavity (cont)
  - The resulting distribution for f_{frag} has the following properties
    - Median .22
    - 90th percentile .42
    - 99th percentile .60
  - The following discretized distribution was then constructed to represent this parameter

	f_{frag}	Probability
Low	0. - .35	0.8
Intermediate	.35 - .60	.19
High	.6 - 1.0	.01

Quantification of DCH DETs (continued)

- *Peak containment pressure following vessel failure*
  - *This event has no branching - it is used to summarize the results of the deterministic DCH model. Predictions for peak containment pressure for each pathway through the tree*
  - *The pathways through the tree represents different sets of assumptions regarding the major phenomena which impact containment pressurization*
- *Drywell head fails following vessel failure*
  - *This event assesses the probability of drywell head failure given the pressure in the preceding event*
  - *The containment peak pressure is compared with the drywell head fragility curve*

Model for Computing Peak Containment Pressure for Direct Containment Heating

- Two node model represents drywell and wetwell
- Initial debris mass entrained into upper drywell based on user defined time constant
- Gas heat-up due to debris energy including oxidation heat
- Horizontal vent clearing time based on model by Moody

Major Model Assumptions

- *RPV and horizontal vent flow calculated as in MAAP GFLOW subroutine*
- *Debris and gas assumed in thermal equilibrium*
- *Gas behaves as an ideal gas*
- *No credit for containment heat sinks*
- *Amount of Zr oxidation limited by steam mass and Zr mass entering containment*
- *Simple Euler integration*

Required Input to DCH Model

- Containment volume
- Initial gas temperatures
- Initial steam fraction in containment
- Horizontal vent area
- Vent clearing depth

- Initial RPV pressure and temperature
- RPV gas volume
- Exit enthalpy for RPV gas

- Initial debris temperature
- Fraction of Debris Zr to be oxidized (20% of initial debris mass assumed to be Zr, 50% oxidized)
- Time constant for entrainment

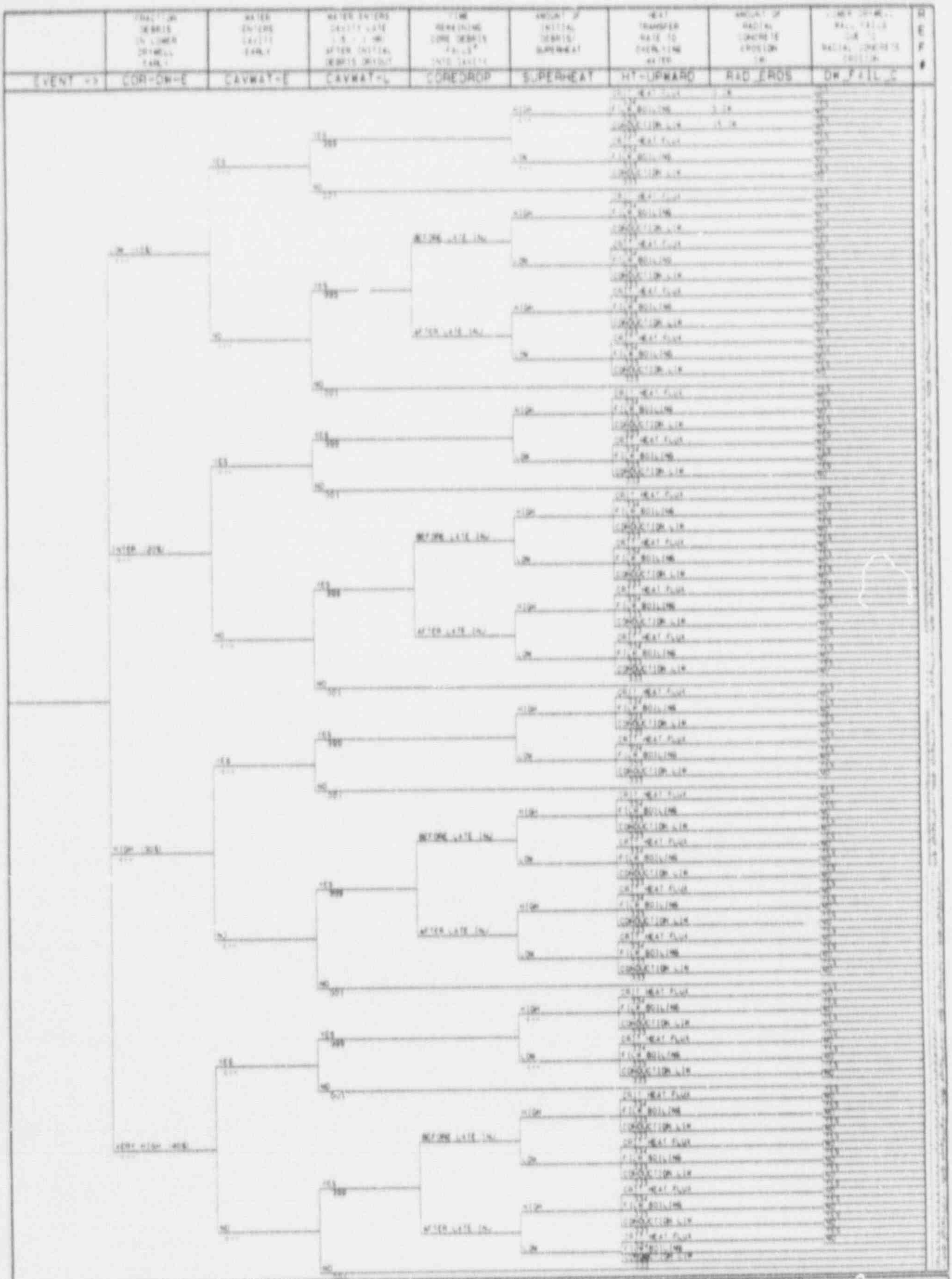
Case Dependent Input

- *Debris mass*
- *Vessel failure area*
- *Initial containment pressure*

Time Constant for Entrainment

- *CONTAIN: User input*
- *DHCVIM (BNL):*
  - *For comparison to SNL DCH-1, entrainment rate taken from experiment*
  - *Model not applied to full reactor scale*
- *HARDCORE (ANL):*
  - *Predicted entrainment time of about 0.1 seconds for small scale CWTI-13 and DCH-1*
  - *When applied to full scale ~ 2.5 seconds*
- *CORDE (UK): No information available*

Based on ANL work, an e-folding time of .5 seconds or a linear removal time of 2 seconds is appropriate for the full scale case.



ADVANCED BOILING WATER REACTOR
PROBABILISTIC RISK ASSESSMENT

LOWER DRYWELL WALL FAILS DUE TO CCI DET

DET for Pedestal Failure due to CCI

- *Fraction of core debris in lower drywell early*
  - *Based on:*
    - *Amount of debris molten in vessel lower head at vessel failure*
    - *Extent of high pressure melt ejection*
- *Water enters cavity early*
  - *Very low probability of water in lower drywell prior to vessel failure*
  - *Residual water in vessel*
  - *Low pressure systems injecting into vessel after vessel failure for high pressure sequences*
- *Water enters cavity late (1/2 to 1 hour after initial debris dryout)*
  - *Lower drywell passive flooder system*
  - *Firewater addition to drywell leading to overflow of suppression pool*
  - *Late power or system recovery*

DET for Pedestal Failure due to CCI (continued)

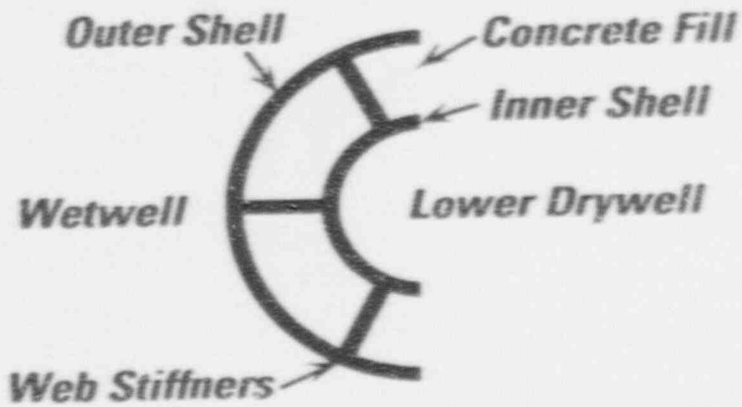
- *Time remaining core debris falls into reactor cavity*
- *Amount of initial debris superheat*
  - *Based on time/mode of vessel failure and amount of debris molten at vessel failure*
- *Heat transfer rate to overlying water*
  - *Area of large uncertainty*
  - *Equal probability assigned to three distinct regimes*
    - *Critical heat flux*
    - *Film boiling*
    - *Conduction limit*

Amount of radial concrete erosion

  - *No branches on this event*
  - *Event will display results of advanced DECOMP CCI calculation for the amount of radial erosion*

Ability of Pedestal to Withstand Ablation

- ABWR pedestal can withstand at least 38 cm of ablation
- Calculated by comparing the design load to residual pedestal area
- Analysis is conservative:
  - Design load contains significant margin
  - Strength of fill concrete ignored
  - Stress kept below yield stress
- After ablation, pedestal is still highly resistant to buckling because of fill concrete and geometry



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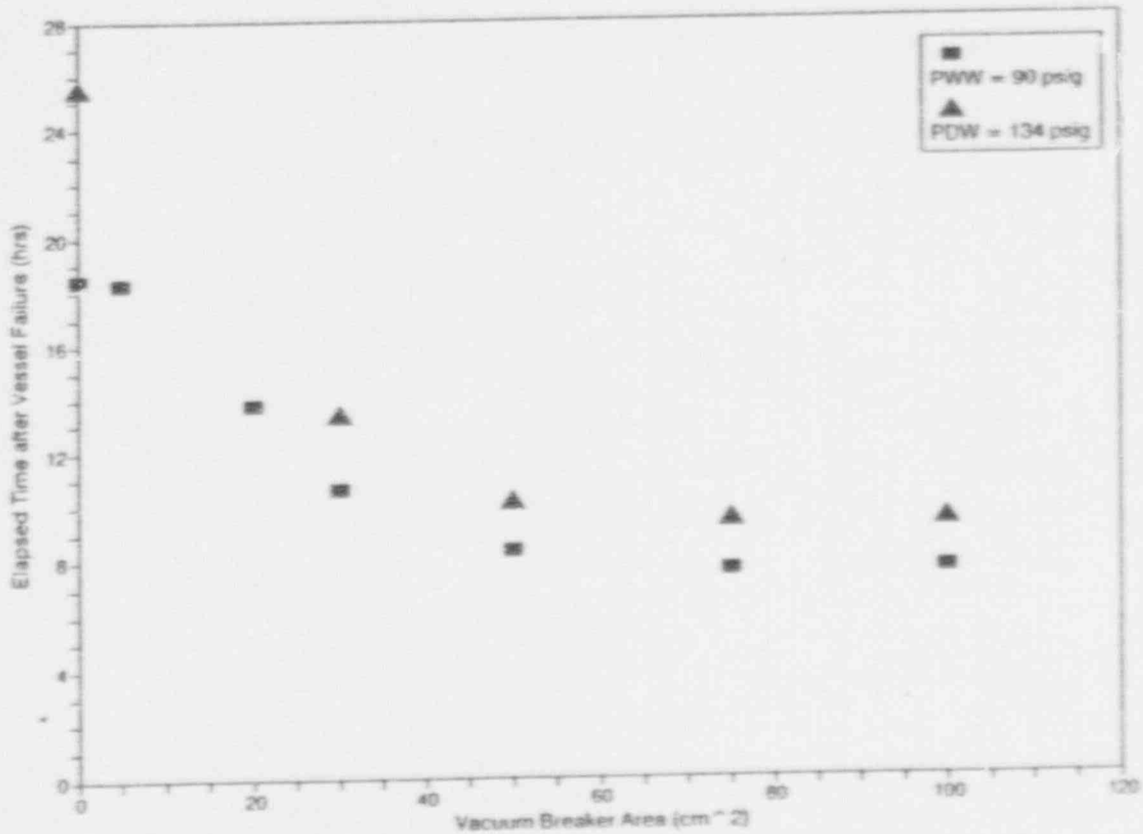
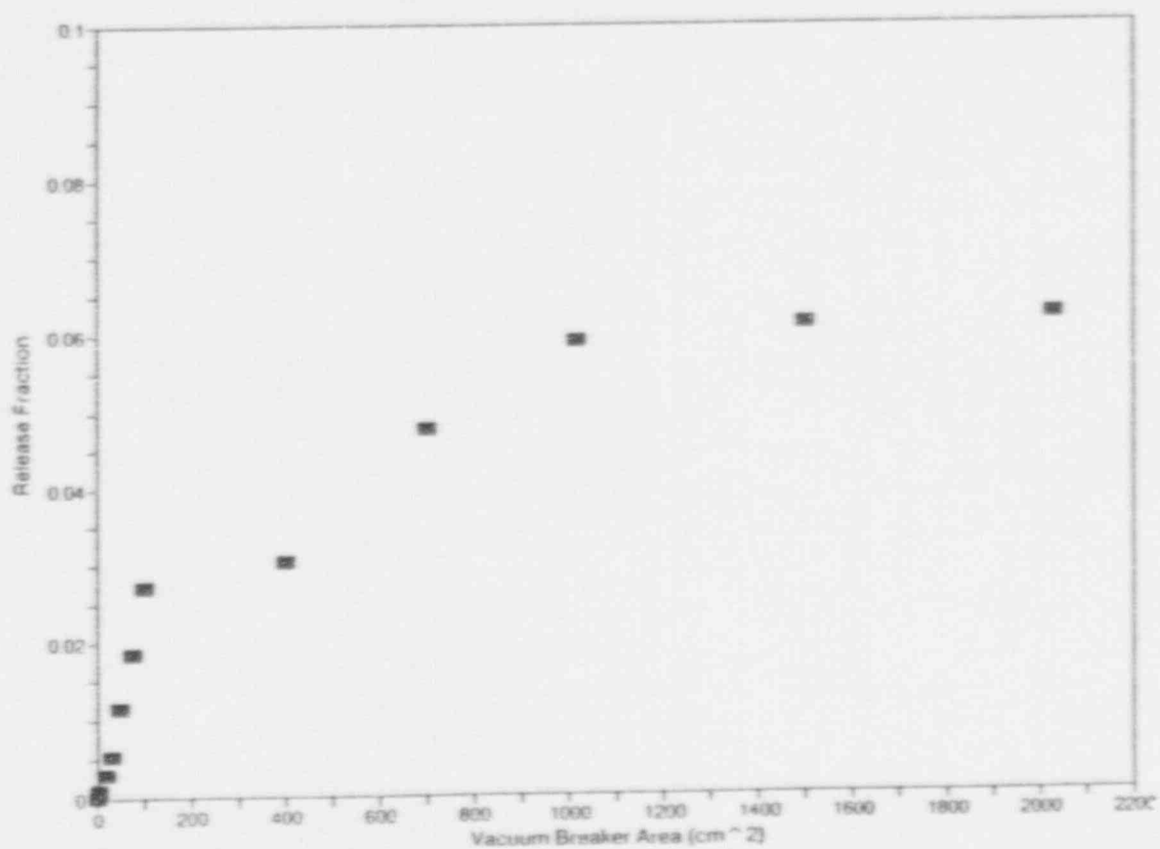
Bypass Leakage and you

- *Sensitivity analyses performed assuming vacuum breaker leakage between 0 and 2000 cm^2 (one vacuum breaker full open)*
- *Bypass leakage has been identified as an area which could have significnat impact on offsite dose*
  - *Release time for rupture disk opening approaches a limit of about 8 hours after vessel failure*
  - *Release time drywell head failure (at median value) approaches about 9.5 hours after vessel failure*
- *Release fractions also approach an asymptotic limit of about 6% for rupture disk opening*
- *Evidence of revaporization differences are observed*

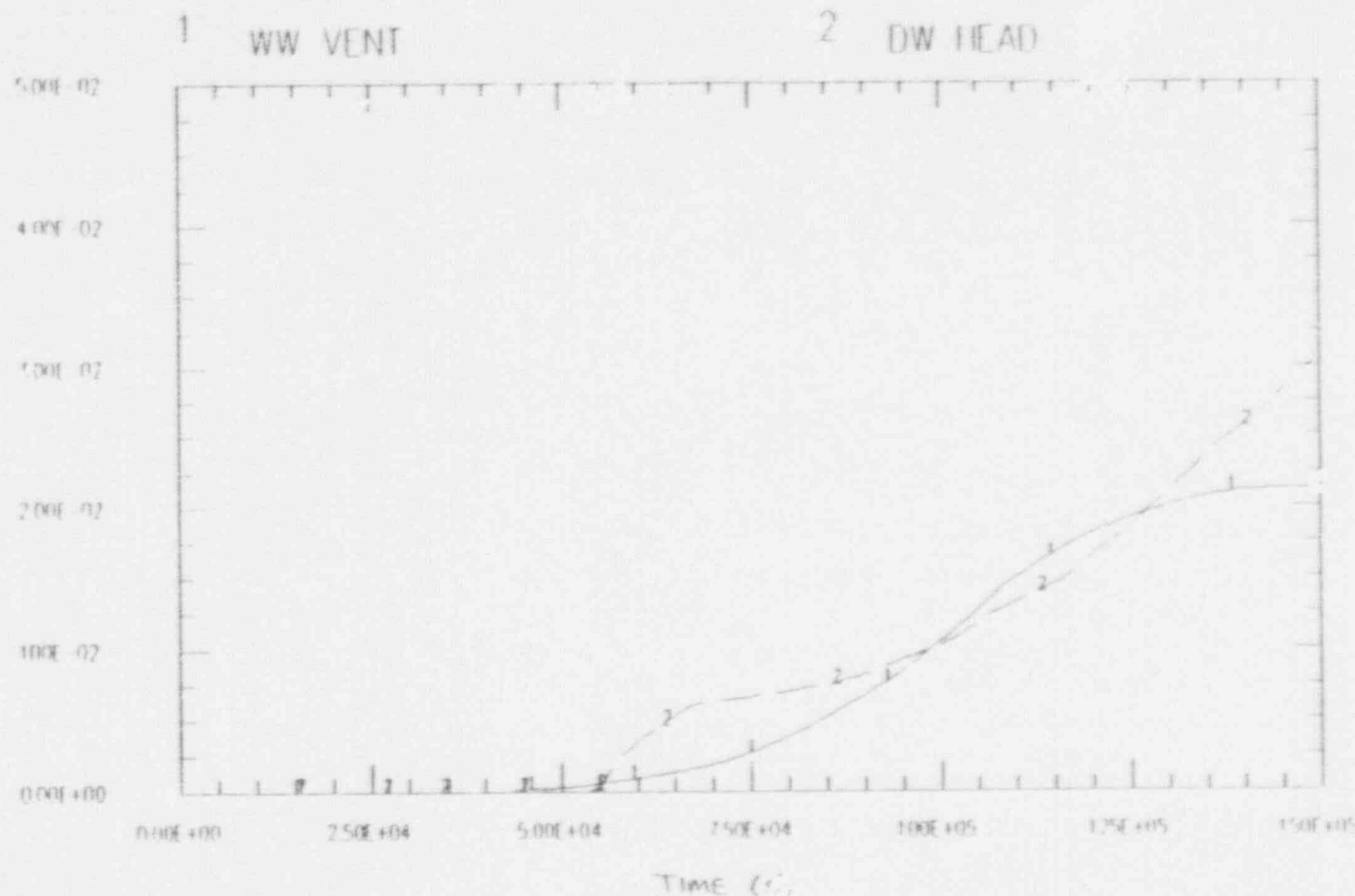
Release time for varying bypass leakage



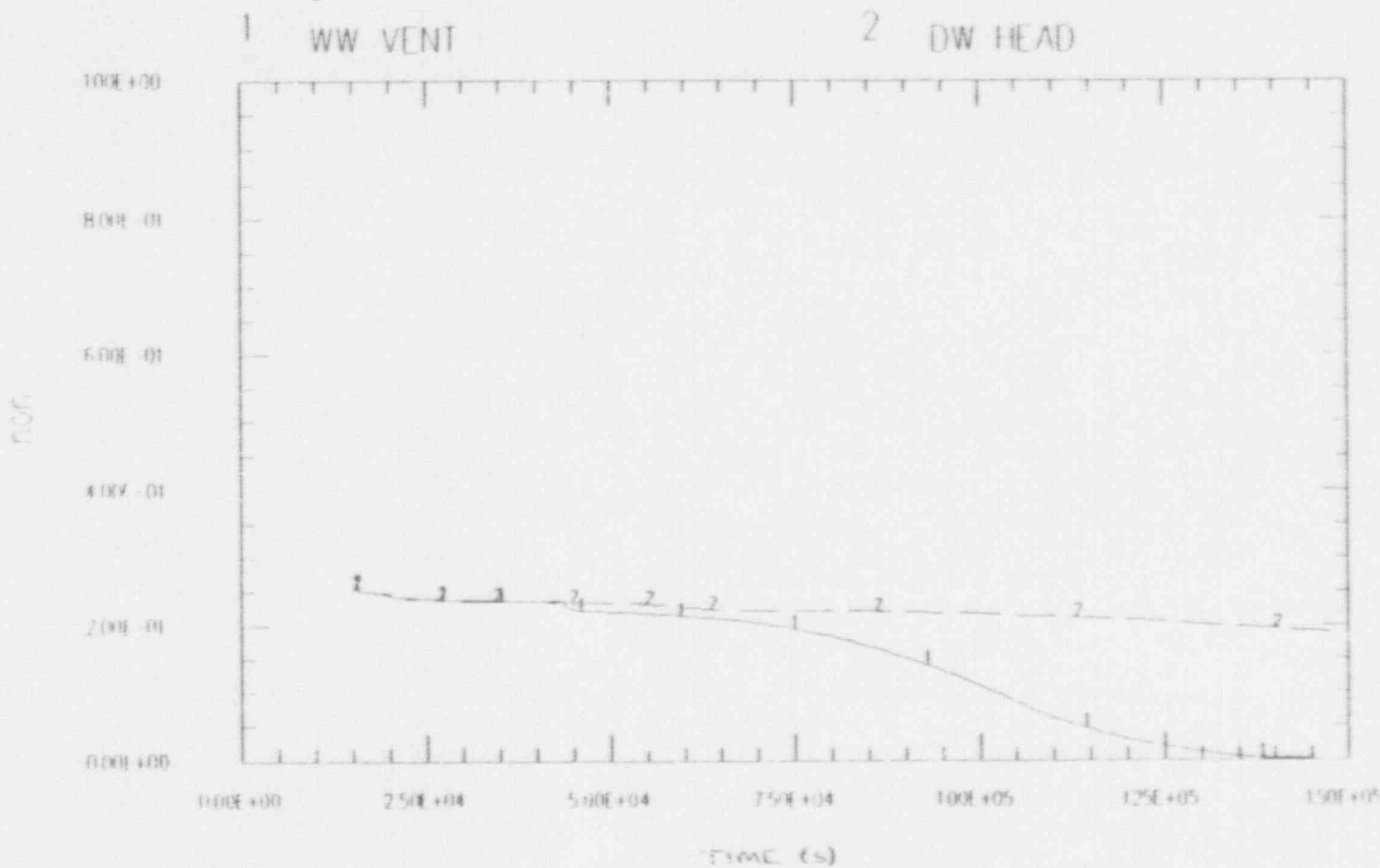
Rupture Disk Release Fractions



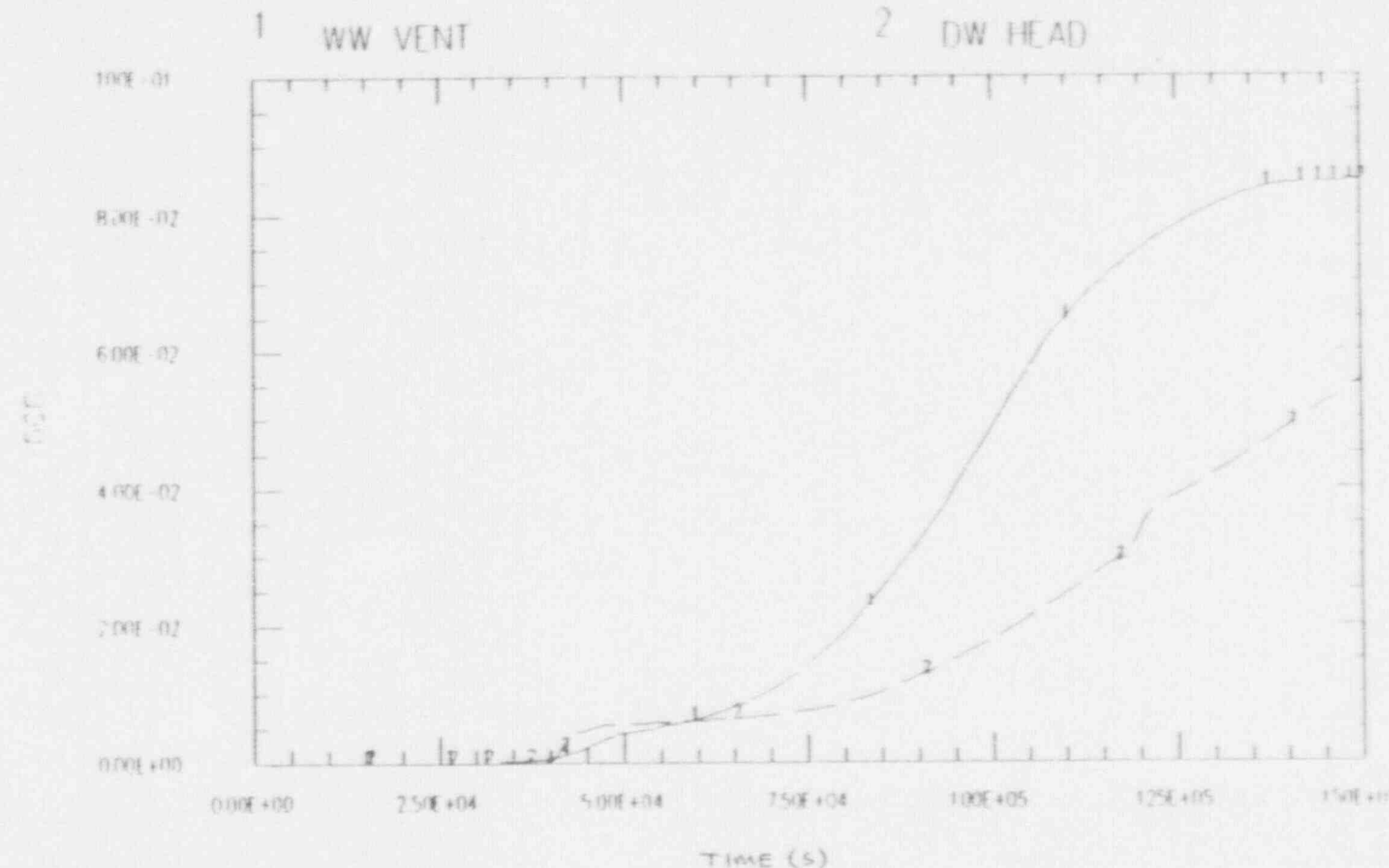
Release Fractions for AVB = 30 cm^2



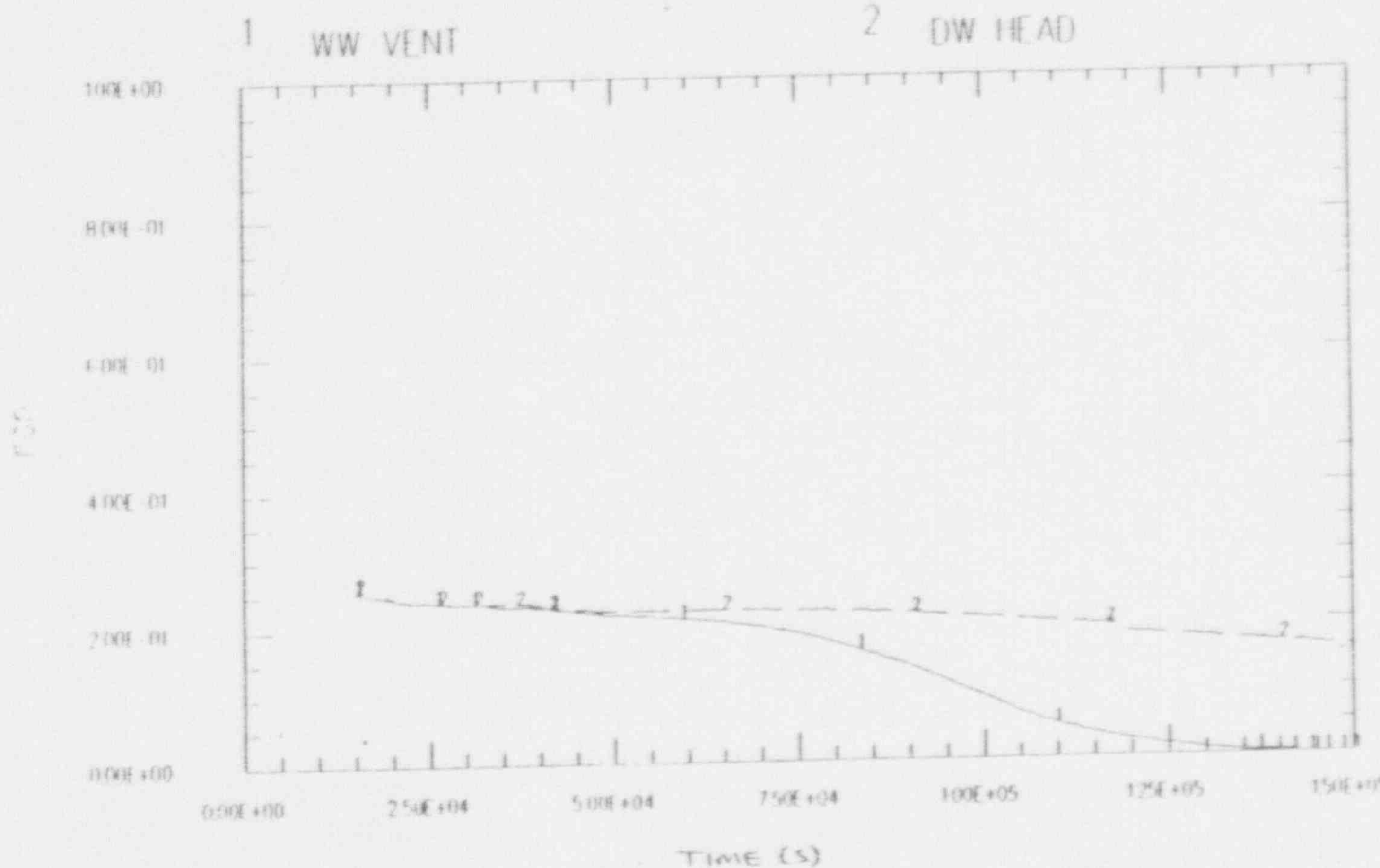
Vessel Retention Fractions for AVB = 30 cm^2



Release Fractions for AVB = 100 cm^2



Vessel Retention Fractions for AVB = 100 cm^2



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ATWS without Boron Injection

Issue: *INEL Report DOE/ID10211 interpreted as showing that 3.45 heat exchangers are necessary to remove heat from the containment during an ATWS event with no boron injection.*

INEL analysis indicates that:

- *3.45 heat exchangers are needed to maintain containment pressure below design limit.*
- *3 heat exchangers limits containment pressure to 72 psig.*
- *With 3 heat exchangers containment pressure will not exceed design for almost 6 hours.*

ATWS Without Boron Injection (continued)

GE position:

- *This event is a seriously degraded event - well beyond the design basis*
- *Success criteria for the PRA use realistic limits rather than licensing or design basis limits*
- *Peak pressure predicted using three heat exchangers is below service level C*
- *Further, time until design level is reached is adequate for the alternate insertion of boron, so it is extremely unlikely that even the design limit will be reached*

Three heat exchangers have adequate heat removal capacity to mitigate ATWS

Observations on DSER text for ATWS success criteria

Page 19-24

- *Text states that staff raised questions about the adequacy of a single train of RHR to remove heat for an ATWS with failure of boron injection*
- *Referenced RAI discusses only ATWS with successful boron injection*
- *Text goes on to say that GE's response confirmed staff's finding that two RHR systems are required*
  - *If intent was ATWS without boron*
    - *The previous discussion indicates three RHRs are required*
  - *If intent was ATWS with boron*
    - *GE analysis (in RAI 725.65) indicates a peak containment pressure of 49 psig*
    - *As previously, the appropriate success criteria for the PRA are not limited by the design condition*
    - *GE analysis does not confirm staff finding*

Observations on DSER Text for ATWS

Page 19-69

- *The DSER notes an apparent inconsistency between the seismic analysis in Figure 19J.5-7 and the text on page 19.4-11.*
  - *GE concurs and will amend the text*
- *The DSER comments that the power will be high in the event of ATWS with failure of boron injection and flow control, and states that core damage may ensue despite continued cooling*
  - *Maximum flow is 4760 m^3/hr*
    - *3 LPCI pumps at 1000 m^3/hr each,*
    - *2 HPCF pumps at 865 m^3/hr each, and*
    - *CRD at 30 m^3/hr*
  - *Feedwater flow about 9000 m^3/hr*
  - *Maximum power level well below that which could cause core damage*