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REMARKS

Proposed vugraphs for the September 11 meeting in P-118

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K/8

DISCUSSION OF A
GENERIC LETTER ON
BWR CONTAINMENT PERFORMANCE

SEPTEMBER 11, 1986

ROBERT M. BERNERO, USNRC

GENERIC LETTER ON
BWR CONTAINMENT PERFORMANCE

THE SETTING: PLANT EVALUATIONS UNDER THE SEVERE ACCIDENT POLICY STATEMENT

THE KEY REGULATIONS: GDC 16 AND GDC 50

THE SUBJECTS: 37 BWRS WITH PRESSURE SUPPRESSION CONTAINMENTS

THE METHOD: A GENERIC LETTER OF REQUIREMENTS TO IMPLEMENT CHANGES BASED ON GENERIC EVALUATION

NRC SEVERE ACCIDENT

POLICY STATEMENT

- THE MOST COST-EFFECTIVE OPTIONS FOR REDUCING THIS VULNERABILITY SHALL BE IDENTIFIED AND A DECISION SHALL BE REACHED CONSISTENT WITH THE COST-EFFECTIVENESS CRITERIA OF THE COMMISSION'S BACKFIT POLICY AS TO WHICH OPTION OR SET OF OPTIONS (IF ANY) ARE JUSTIFIABLE AND REQUIRED TO BE IMPLEMENTED.

- IN THOSE INSTANCES WHERE THE TECHNICAL ISSUE GOES BEYOND CURRENT REGULATORY REQUIREMENTS, GENERIC RULEMAKING WILL BE THE PREFERRED SOLUTION. IN OTHER CASES, THE ISSUE SHOULD BE DISPOSED OF THROUGH THE CONVENTIONAL PRACTICE OF ISSUING BULLETINS AND ORDERS OR GENERIC LETTERS WHERE MODIFICATIONS ARE JUSTIFIED THROUGH BACKFIT POLICY, OR THROUGH PLANT-SPECIFIC DECISION MAKING ALONG THE LINES OF THE INTEGRATED SAFETY ASSESSMENT PROGRAM (ISAP) CONCEPTION.

GDC 16:

CRITERION 16 - CONTAINMENT DESIGN. "--AN ESSENTIALLY LEAK-TIGHT BARRIER AGAINST THE UNCONTROLLED RELEASE OF RADIOACTIVITY TO THE ENVIRONMENT AND TO ASSURE THAT THE CONTAINMENT DESIGN CONDITIONS IMPORTANT TO SAFETY ARE NOT EXCEEDED FOR AS LONG AS POSTULATED ACCIDENT CONDITIONS REQUIRE."

GDC 50:

CRITERION 50 - CONTAINMENT DESIGN BASIS. "--AS REQUIRED BY SECTION 50.44, ENERGY FROM METAL-WATER AND OTHER CHEMICAL REACTIONS THAT MAY RESULT FROM DEGRADATION BUT NOT TOTAL FAILURE OF EMERGENCY CORE COOLING FUNCTIONING, (2) THE LIMITED EXPERIENCE AND EXPERIMENTAL DATA AVAILABLE FOR DEFINING ACCIDENT PHENOMENA AND CONTAINMENT RESPONSES, AND (3) THE CONSERVATISM OF THE CALCULATIONAL MODEL AND INPUT PARAMETERS."

U.S. BOILING WATER REACTORS

- 24 BWR 2/3/4 WITH MARK CONTAINMENT (ALL LICENSED)

- 9 BWR 4/5 WITH MARK II CONTAINMENT (7 LICENSED)

- 4 BWR 6 WITH MARK III CONTAINMENT (3 LICENSED)

INSTITUTIONAL PROCESS

- CLOSURE OF SEVERE ACCIDENT ANALYSIS FOR CONTAINMENT
  - NO FURTHER ANALYSIS UNLESS EXCEPTION IS TAKEN

- SPECTRUM OF OPTIONS
  - RULEMAKING
  - 50.54F LETTER FOLLOWED BY ORDER
  - GENERIC LETTER FROM DIRECTOR NRR OR DIRECTOR DBWRL

- PROCESS CHARACTERISTICS
  - BASED ON TECHNICAL WORK AVAILABLE, IDCOR, SOURCE TERM, PLANT SPECIFIC WORK, NUREG-1050, NUREG-1150
  - OPEN TO PUBLIC FOR COMMENT AND PARTICIPATION

A BWR - MARK I

FOR REFERENCE

BEFORE

- CORE MELT FREQUENCY: 1×10^{-4}/YR
  - A FULL SPECTRUM OF SEQUENCES INCLUDING BLACKOUTS

- CONTAINMENT CAPABILITY: UNCERTAIN AND VARIABLE BUT ASSUME 1 OUT OF 2 CORE MELTS GIVES FAIRLY LARGE RELEASE

AFTER

- CORE MELT FREQUENCY: 1×10^{-4}/HR
  - IPE FOR FRONT END MAY REDUCE BUT NO CREDIT IS TAKEN HERE

- CONTAINMENT CAPABILITY: SUBSTANTIAL ASSURANCE THAT CONTAINMENT WILL MITIGATE CONSEQUENCES, DEGREE VARIABLE FROM PLANT TO PLANT BUT 1 OUT OF 50 CORE MELTS GIVING A FAIRLY LARGE RELEASE SHOULD BE REPRESENTATIVE

HYDROGEN CONTROL

PROPOSED REQUIREMENTS

- CHANGE TECH. SPEC. AT END OF OPERATION FROM 24-HOUR ALLOWANCE TO 12-HOUR ALLOWANCE OF NON-INERTED OPERATION AT REDUCED POWER
- PERMIT 12-HOUR PERIOD AT REDUCED POWER WITHIN THE OPERATING CYCLE TO SEARCH FOR UNIDENTIFIED LEAKAGE

RATIONALE

- DEINERTING TYPICALLY TAKES 4-8 HOURS
- LEAKAGE INSPECTION AND MINOR REPAIR CAN BE REASONABLY ACHIEVED IN 4-8 HOURS
- REDUCED POWER ($\leq$ 33%) SUBSTANTIALLY REDUCES SHORT-LIVED FISSION PRODUCT INVENTORY AND DYNAMICS OF POSSIBLE ACCIDENTS

DRYWELL SPRAY

PROPOSED REQUIREMENTS

- REDUCE DESIGN SPRAY RATE (CHANGE NOZZLES) TO ABOUT 10% OF PRESENT VALUE
- PROVIDE AC-POWERED BACKUP WATER SUPPLY FOR SPRAY AND AC-INDEPENDENT WATER SUPPLY, AVAILABILITY BY REMOTE MANUAL OPERATION OR BY SIMPLE RELIABLE PROCEDURE

DESIRABLE

- MAKE ALTERNATE WATER SOURCES AVAILABLE TO COOL CORE DIRECTLY
- 90/10 MODE OF RHR OPERATION

RATIONALE

- WATER SUPPLIES AND EQUIPMENT ARE ALREADY AVAILABLE FOR LOWER FLOWS
- LOWER FLOWS PROVIDE ALL BENEFITS EXCEPT LOW ΔT DECAY HEAT REMOVAL AND DO NOT RAPIDLY FLOOD CONTAINMENT
- ASSURED DRYWELL SPRAY SUBSTANTIALLY REDUCES AND PROBABILITY SIGNIFICANCE OF DRYWELL FAILURE OR SUPPRESSION POOL BYPASS

PRESSURE CONTROL

PROPOSED REQUIREMENTS

- RELIABLE CAPABILITY TO VENT WETWELL AT EPG PRESSURE LEVEL WITH OR WITHOUT AC POWER. FOR VENTING WITHOUT AC POWER MANUAL PROCEDURE IN ADVANCE MAY BE USED IF NITROGEN PURGE IS AVAILABLE

- VENT OF 18-INCH DIAMETER OR GREATER

DESIRABLE

- ABILITY TO VENT SLOWER SEQUENCES THROUGH STANDBY GAS TREATMENT SYSTEM

- BURST RESISTANCE DUCTING IN REACTOR BUILDING TO MINIMIZE COMPLICATIONS

RATIONALE

- RELIABLE VENTING PREVENTS UNCONTROLLED OVERPRESSURE FAILURE WHICH CAN CAUSE CORE MELT

- VENTING WITH DRYWELL SPRAY GIVES GREAT ASSURANCE OF RELEASE MITIGATION

CORE DEBRIS

PROPOSED REQUIREMENTS

- ASSURE RETENTION OF WATER AT LEAST 3 FEET DEEP IN TORUS ROOM IF TORUS LEAKS ENTIRE CONTENTS

DESIRABLE

- CONCRETE CURBS OR OTHER BARRIERS WHICH WOULD RETARD DEBRIS ATTACK OF DRYWELL SHELL
- AVOID LOSS OF RECOVERY SYSTEMS FROM WETTING BY TORUS ROOM WATER

RATIONALE

- DRYWELL FAILURE BY DEBRIS ATTACK IS MADE LESS LIKELY AND LESS SIGNIFICANT BY DRYWELL SPRAY AND VENTING
- RETENTION OF TORUS WATER ENSURES DEBRIS QUENCHING AND SHOULD FACILITATE ACCIDENT RECOVERY

BOSTON EDISON'S GOALS

OUTAGE GOAL: Identify And Implement Containment Improvements Responsive To Bernero's Draft Policy Prior To Startup. Revise Procedures And Train Operators To Improve Operational Readiness For Effective Severe Accident Management.

LONG RANGE GOAL: Perform A Comprehensive Probabilistic Risk Assessment Supported By Deterministic Analyses of Severe Accidents To Ensure That:

1. The Pilgrim Specific Response To Severe Accidents Is Well Understood.
2. These Insights Are Effectively Used For Severe Accident Risk Management Within Boston Edison.
3. These Insights Are Available In Support Of Emergency Preparedness Planning.

BERNERO'S PROPOSED

SEVERE ACCIDENT CONTAINMENT POLICY FOR BWR'S

- **Hydrogen Control**
  - Prevent Containment Failures In Expected Accident Sequences Including Station Blackout
- **Reliable Containment Drywell Sprays**
  - Operable For Broad Spectrum Of Accident Sequences Including Station Blackout
- **Reliable Containment Wetwell Venting**
  - Operable At Design Pressure Conditions To Prevent Containment Overpressure Failure And To Maximize Use Of Suppression Pool As Filtering Medium
- **Core Debris Barriers**
  - Evaluate Paths For Core Debris Travel
  - Add Barriers If Needed To Protect Suppression Pool
- **Emergency Procedures And Training**
  - Review And Modify As Necessary To Ensure Operators are able to recognize Severe Accident Conditions And Use Plant Equipment To Best Advantage

ACTIONS PLANNED AT PILGRIM

HYDROGEN CONTROL

- No Need For Major Design Change
- Administrative Control To Limit The Amount Of Time Containment Is Deinerted Is Under Study
- Decision Will Be Made After Reliability Evaluation Of Nitrogen Inerting Function Is Completed

ACTIONS PLANNED AT PILGRIM

CORE DEBRIS BARRIERS

- Core Debris Behavior Will Be Evaluated Using The IDCOR MAAP Code

- Based Upon Industry Studies To Date, No Modifications Are Anticipated

ACTIONS PLANNED AT PILGRIM

CONTAINMENT DRYWELL SPRAYS

- Fire Water System Will Be Interconnected to RHR System To Provide Additional Source Of Water For Drywell Spray
- Options To Provide For Control Of Essential Valves Under Station Blackout Conditions Are Being Evaluated
- Modifications To The Existing Spray Spargers Are Planned Based On Evaluation Of Severe Accident Events

ACTIONS PLANNED AT PILGRIM

CONTAINMENT WETWELL VENTING

- Capability For Venting Wetwell at Containment Design Pressure Using Supression Pool As Filtering Medium Will Be Provided
- Criteria And Guidance For Operational Use Of This Vent Will Be Prepared To Ensure That The Intended Risk Reduction Benefits Are Achieved
- Options For Routing The Vent Discharge Are Under Evaluation. A Selection Will Be Made After Reliability And Risk Evaluations Are Available

ACTIONS PLANNED AT PILGRIM

EMERGENCY PROCEDURES AND TRAINING

- Revise Procedures To Incorporate Plant Changes Made This Outage
- Revise Procedures For Station Blackout Event
- Review Procedures Against Current ATWS Event Analysis For Possible Improvements
- Revise Procedures For Combustible Gas Control
- Operators Will Be Trained In All Emergency

ACTIONS PLANNED AT PILGRIM

OTHER IMPROVEMENTS

- Modifications Will Permit Fire Water To Be Supplied To The Reactor Thru The RHR Low Pressure Coolant Injection Valves Under Station Blackout Conditions
- Capabilities To Backup The On-Site Fire Water Pumps Using Fire Trucks Will Be Available. Pilgrim Now Has A Fire Truck At The Site.
- Operators Will Be More Knowledgeable Of The Plant Behavior And Capability To Control Degraded Conditions Beyond Design-Basis Events As A Result Of Severe Accident Studies Being Done
- We Are Evaluating The Cost And Benefits Of Installing A Third Diesel Generator During This Outage

9/11

SEVERE ACCIDENT CONTAINMENT PCLICY

INDUSTRY EVALUATIONS

SEPTEMBER 11, 1986

BETHESDA, MD.

FOIA-10
K/11

OBJECTIVES:

- o PRESENT RESULTS OF IDCOR/BWROG & UTILITY EVALUATIONS OF PROPOSED SEVERE ACCIDENT CONTAINMENT POLICY ELEMENTS

- o DISCUSS CONCLUSIONS REACHED BY UTILITIES

CONCLUSIONS - BWR EXECUTIVE MEETING

- AUGUST 19 MEETING - 20 OF 23 BWROG UTILITIES REPRESENTED

- CONTINUE CONSTRUCTIVE DIALOGUE WITH NRC

- AGREEMENTS:
  - COMMIT TO IMPLEMENTATION OF REVISION 4 TO EPGS
  - NUMARC CONTACTED TO CONSIDER SEVERE ACCIDENT CONTAINMENT ISSUE AS A GENERIC INDUSTRY ISSUE
  - CONTINUE WORKING WITH NRC TO BETTER DEFINE ISSUES FOR RESOLUTION
  - PROPOSE TO BWROG SEVERE ACCIDENT INSIGHT REVIEW OF EPG REV. 4

IDCOR/BWROG/UTILITY EVALUATIONS:

- WIDE VARIATION IN ESTIMATED COST
- EVALUATIONS PERFORMED ON SMALL NUMBER OF PLANTS
- OBJECTIVES APPEAR GENERIC - ENHANCEMENTS APPEAR PLANT SPECIFIC
- QUALITATIVE ASSESSMENTS OF BENEFITS AND NEGATIVE IMPACTS PERFORMED

ELEMENT 1 - HYDROGEN

OBJECTIVE: PREVENT HYDROGEN COMBUSTION CAUSED FAILURE

REQUIREMENTS:

A. OXYGEN CONTROL (MARK I AND MARK II)

B. HYDROGEN CONTROL (MARK III)

IDCOR/BWROG EVALUATIONS:

- OXYGEN CONTROL BY NITROGEN INERTING ADEQUATE FOR MARK I AND MARK IIs.
- LIMITING THE TIME DEINERTED UNDER REVIEW.
- MARK III HYDROGEN CONTROL BEING ADDRESSED BY HYDROGEN CONTROL OWNERS' GROUP (HCOG).

HYDROGEN CONTROL OWNERS GROUP

STRATEGY TO ADDRESS SEVERE ACCIDENTS AND STATION BLACKOUT

MEETING WITH THE NRC ON SEVERE ACCIDENTS

BETHESDA

SEPTEMBER 11, 1986

BACKGROUND

* Established HCOG Programs to address Hydrogen Rule requirements for "Degraded Core Accidents".

  - Quarter Scale Testing Program

  - Analytical effort

* Station Blackout as a hydrogen generation event (HGE) within the context of recoverable degraded cores is an issue being addressed.

  - Current HCOG evaluation indicates that SBO is not a credible HGE

  - HCOG responding to NRC questions

* The need for an independent power supply for igniters in the event of an SBO identified by the NRC in the context of Severe Accidents.

HCOG PROPOSAL

HCOG to supply the design criteria for a backup power supply to the hydrogen igniters.

* Backup power supply need not be safety related

* Identify impact of addressing Severe Accidents on the design of a backup power supply.

* Number of igniters required in the event of an SBO

  - Make use of existing data base and criteria

  - Additional testing, only if necessary, to follow completion of current Test Program - end of this year

Responsibility of individual Mark III Owners with support of HCOG as required.

  - Define backup power supply source

  - Define associated costs

  - Meet and discuss with the NRC the details of the design, costs, and benefits of a backup power supply to the igniters

  - Decision and timing for proceeding

ELEMENT 2 - SPRAYS

OBJECTIVE: SPRAY WATER TO:

1. QUENCH DEBRIS (PRIMARY)
2. SCRUB AEROSOLS (SECONDARY)
3. LOWER PRESSURE (SECONDARY)
4. COOL VULNERABLE EQUIPMENT (SECONDARY)

REQUIREMENTS:

1. SPRAY IN DRYWELL
2. BACKUP WATER SOURCES AND PUMPS
   - HOSE CONNECTIONS
   - USE OF FIREMAINS

IDCOR/BWROG EVALUATIONS:

- o TYPICAL SPRAY CAPACITY 5 - 10,000 GPM/HEADER
- o CONCEPTS CONSIDERED
  - o CONNECTION TO HOSE STATION IN REACTOR BUILDING
    - APPROXIMATE FLOW PROVIDED 200 GPM
    - DOES NOT PROVIDE SPRAY

ELEMENT 2 - SPRAYS (Continued)

- CROSS TIES FROM DIESEL FIRE PUMPS TO RHR
  - APPROXIMATE FLOW PROVIDED 1/2 OF FIRE PUMP RATING
  - CLOSE OFF APPROXIMATELY 70% OF NOZZLES TO ACHIEVE SPRAY
- FLOW RATES IDENTIFIED APPEAR ADEQUATE
- DEBRIS QUENCHING DOES NOT REQUIRE SPRAY
- POTENTIAL BENEFIT/RISK WARRANTS FURTHER STUDY

ELEMENT 3 - PRESSURE

OBJECTIVES: 1. AVERT UNCONTROLLED OVERPRESSURE FAILURE

2. CONTROL RELEASE PATH (SCRUBBING)

REQUIREMENTS:

1. SUBSTANTIAL CAPABILITY TO VENT WETWELL
2. REMOTE/RELIABLE CONTROL OF VENT VALVE
3. ABILITY TO RECLOSE VENT

IDCOR/BWROG EVALUATIONS:

- o CONCEPTS CONSIDERED
  - o UPGRADE DUCTING AND STANDBY GAS TREATMENT SYSTEM (SBGTS) TO CONTAINMENT DESIGN PRESSURE CAPABILITY
    - NOT FEASIBLE TO UPGRADE SBGTS
  - o HARDPIPED BYPASS AROUND SBGTS
  - o HARDPIPED DEDICATED VENT
- o COSTS ARE COMPARABLE FOR HARDPIPED OPTIONS

<u>ELEMENT 3 - PRESSURE</u> (Continued)

- VENT SIZING UNDER REVIEW
  - ATWS
  - DHR
- NEGATIVE IMPACTS NEEDING FURTHER REVIEW
  - SECONDARY CONTAINMENT CONTAMINATION
  - DELIBERATE RELEASE

ELEMENT 4 - CORE DEBRIS

OBJECTIVE: REDUCE LIKELIHOOD OF FAILURE BY DIRECT ATTACK

REQUIREMENTS:

1. USE PRACTICAL DEBRIS RETARDING BARRIERS
2. CONSERVE SUPPRESSION POOL WATER AS A QUENCHING POOL

IDCOR/BWROG EVALUATIONS:

- DRYWELL
  - CONCEPTS CONSIDERED
    - PLUG IN PEDESTAL OPENING
    - INCREASE SUMP SIZE INSIDE PEDESTAL
    - CURB OUTSIDE PEDESTAL OPENING
    - CURB AT DRYWELL LINER/FLOOR JUNCTION
    - ADDITIONAL PEDESTAL OPENINGS TO PROMOTE EVEN DISTRIBUTION
- NEGATIVE IMPACTS
  - LOCA CONSIDERATIONS
  - SEISMIC INTERACTIONS
  - ALARA CONCERNS

- o QUALITATIVE BENEFIT LOW
  - DEPENDENT ON ANALYTICAL MODELS OF DEBRIS MOBILITY WHICH ARE VERY UNCERTAIN

- o WETWELL
  - o MOST PLANTS CURRENTLY HAVE CAPABILITIES TO HOLD WATER IN TORUS CHAMBER OR PROVIDE PROTECTION TO CRITICAL EQUIPMENT IN CORNER ROOMS
  - o QUALITATIVE BENEFITS LOW
  - o NO NEGATIVE IMPACTS
  - o NO FURTHER STUDY WARRANTED

ELEMENT 5 - TRAINING AND PROCEDURES

OBJECTIVE: ENSURE OPERATORS ARE READY TO USE PLANT FEATURES TO BEST ADVANTAGE IN SEVERE ACCIDENTS

REQUIREMENTS:

1. CLEAR SYMPTOM BASED STRATEGIES (INTEGRATED)
2. REMOVAL OF UNNECESSARY INHIBITIONS
3. TRAINING/PROCEDURES

IDCOR/BWROG EVALUATIONS:

- REV. 4 IMPLEMENTATION BY ALL UTILITIES CONSISTENT WITH PREVIOUS POST-TMI COMMITMENT
- PROPOSE REVIEW OF REV. 4 WITH INSIGHTS FROM SEVERE ACCIDENT STUDIES

CONCLUSIONS - BWR EXECUTIVE MEETING

- o AUGUST 19 MEETING - 20 OF 23 BWROG UTILITIES REPRESENTED
- o CONTINUE CONSTRUCTIVE DIALOGUE WITH NRC
- o AGREEMENTS:
  - COMMIT TO IMPLEMENTATION OF REVISION 4 TO EPGS
  - NUMARC CONTACTED TO CONSIDER SEVERE ACCIDENT CONTAINMENT ISSUE AS A GENERIC INDUSTRY ISSUE
  - CONTINUE WORKING WITH NRC TO BETTER DEFINE ISSUES FOR RESOLUTION
  - PROPOSE TO BWROG SEVERE ACCIDENT INSIGHT REVIEW OF EPG REV. 4

9/11

NAME	AFFILIATION
R. BERNERO	NRC / DBL
H. DENTON	NRC / NRR
C. REED	COMMONWEALTH EDISON
John Raulston	Tenn. Valley Auth.
Vincent Boyer	Phila. Electric Co
RICHARD DIEDERICH	PHILADELPHIA ELECTRIC
JAMES C. CARTER	IT CORP / IDCOR
EDWARD HOWARD	BOSTON EDISON
R. E SKAVDAHL	GENERAL ELECTRIC
John M. FULTON	BOSTON EDISON CO.
T. H. Landers	New York Power Authority
J. A. GRAY, JR.	NEW YORK POWER AUTHORITY
S. D. Floyd	Carolina Power & Light Co.
AB CUTTER	CAROLINA POWER & LIGHT
H. W. KEISER	PENN POWER & LIGHT CO
BRIAN MCCAFFREY	LONG ISLAND LIGHTING CO.
CHARLES DAVERIO	LONG ISLAND LIGHTING CO.
L. T. GUCWA	GEORGIA POWER CO.
S. H. CHESNUT	GEORGIA POWER COMPANY
J. R. Langley	Gulf States Utilities / HCOG
TERRY PICKENS	NORTHERN STATES POWER
KEVIN HOLTZCLAW	GENERAL ELECTRIC CO.
Dean Houston	NRC/ACRS Staff
FAROUK ELTAWILA	NRR/DSRO/RIB
JIM DEDDENS	GULF STATES UTILITIES CO.
Dennis B. Hacking	HCOG / Enercon Services
Wayne Hodges	NRC/DBL/RSB

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Jerry Hulman	NRR/DBL/PSB
ANGELO MARINOS	NRR/DBL
S.W. WILCZEK, JR.	Niagara Mohawk
BRUCE A. PRESTON	PSE & G Co.
Richard F. Murray	PSE&G
Michael J May	TVA - BFN
WOODY STROUPE	NUCARD INC. / ENERCON
Pat Powell	Washington Public Power Supply System
Lynn Connor	Doc-Search Associates
LARRY GIFFORD	GE
John Stang	NRR/BWD1/LaCrosse Project Manager
Thomas S. Rotella	NRR/DBL/BWD1 Big Rock Point PM
Raman Pichumani	NRR/DBL/FOB
DONALD R HOFFMAN	CLEVELAND ELECTRIC ILLUMINATING
Vernon L Rooney	NRR/DBL/BWD-2
JOHN LARKINS	OCM/LZ
KAZIMIERAS CAMPE	NRR/DBL/PSB
Gerald E. Gears	NRR/DBL/PD#2
JACK KUDRICK	NRR/DBL/PSB
JAY THAYER	YANKEE ATOMIC
RJ Lodwick	Vermont Yankee
Stephen P. Schultz	Yankee Atomic Electric
David H Wagner	NRR/DBL/BWD-3
Domenic B. Vassallo	NRR/DBL/FOB
DANIEL R. MULLER	NRR/DBL/PD#2

JOHN A Zwolinski	NRR/DBL/BWD1
Raj Auluck	NRR/DBL/BWD1
Jack Donohew	" "
E. G. ADENSAM	NRR/DBL/BWD3
EARL PAGE	DETROIT EDISON
Phillip L. Paull	Vermont Public Service Dept
G. TARRANT	VT. Dept. Pub. Service
N.W. EDWARDS	NUTECH
C. L. REID	BECHTEL
OWEN M. SCOTT	SOUTHERN CO. SERVICES
G.C. LAINAS	NRR/DBL
Eve Fotopoulos	SERCH Licensing, Bechtel
Akira Omoto	The Tokyo Electric Power
TETSI IMAI	JEPIC
SENGBO KIM	NRR/DBL/PSB
Mohan C. Thadani	NRR/DBL/BWD2
KATHLEEN H. SHEA	Newman & Holtzinger
David Wilson	Iowa Elec. Light and Power
Stephen Maloney	DEVONRUE
Don Lewis	MP&L
Paul Leech	NRC/BWD-1

STATE OF VERMONT
DEPARTMENT OF PUBLIC SERVICE
120 STATE STREET
STATE OFFICE BUILDING
MONTPELIER 05602
TEL. 802 828-2811

Transmittal Sheet

To: Vernon Rooney

From: Phillip Paull

Date Telecopied: Nov. 3, 1986

No. of Pages, Including This Transmittal Sheet: 60

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30 pages - 1st transmittal.

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Fairing —

Lessmont Consultant Report

This was telecopied by the state of Vermont yesterday

Vern

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K/16

PRD CONSULTING
P. R. Davis, President
1935 Sabin Dr.
Idaho Falls, ID 83401
(208) 529-2861
Oct. 30, 1986

Mr. Gerald R. Tarant
Commissioner, Dept. of Public Service
State of Vermont
120 State St.
Montpelier, VT 05602

Dear Commissioner Tarant,

Transmitted herewith is one copy of my report, "A Review of the Vermont Yankee Containment Safety Study." Upon your acceptance, this report constitutes fulfillment of the provisions in Contract No. 0938124. However, I will endeavor, as time permits, to examine YAEC-1564 ("BWR Mark I Containment Evaluation-Vermont Yankee, Oct. 1986) to determine if the results are consistent with and support the Vermont Yankee Containment Safety Study. I will provide you office with a letter if I find any important discrepancies. As you know, I did not receive the report until Oct. 29, one day before I was obligated to mail the enclosed report in order to meet the Oct. 31 deadline. As we discussed, I did not consider it mandatory for my review to evaluate YAEC-1564 since my review was based primarily (as stated in the attached report) on comparison of the VYCSS with similar contemporary severe accident assessments and related information, with less emphasis on an in-depth review of the VYCSS analysis.

I hope my report is useful to you and the State. I enjoyed working with you and Phil Paull on this project. I particularly appreciated the cooperation given by you and Phil in setting up the arrangements for the review and also the objective and competent evaluation of the material as the project progressed.

Sincerely,

P.R. Davis

P. R. Davis

A REVIEW OF THE VERMONT YANKEE CONTAINMENT SAFETY STUDY

prepared for

State of Vermont
Department of Public Service

by

P. R. Davis
principle author
PRD Consulting
1935 Sabin Dr., Idaho Falls, ID, 83401

and

Michael L. Corradini, PhD
contributing author
University of Wisconsin

Oct. 31, 1986

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TABLE OF CONTENTS

A REVIEW OF THE VERMONT YANKEE CONTAINMENT SAFETY STUDY

I. INTRODUCTION This report presents the results of a review of the Vermont Yankee Containment Safety Study prepared by the Vermont Yankee Nuclear Power Corporation. The review was undertaken for and funded by the State of Vermont, Dept of Public Service. The objectives of the review were as follows:

1. Perform an overall evaluation of the Vermont Yankee Containment Safety Study.

2. Determine whether the containment study provides a reasonably accurate estimate of the probability of containment failure from severe accidents. If the estimate is judged to be not accurate, determine if it is too high or too low and by what magnitude.

3. Identify technical shortcomings in the study and estimate their effect on the containment failure probability.

4. Determine what changes can be done which would increase the probability that the Vermont Yankee plant containment would not fail in the event of a severe accident. Included will be an evaluation of changes that Vermont Yankee Nuclear Power Corp. has identified in its report, changes suggested by the Nuclear Regulatory Commission but not accepted by Vermont Yankee, and other potentially effective changes.

On Sept. 2, 1986, the Vermont Yankee Nuclear Power Corporation submitted to the Nuclear Regulator Commission a document entitled "Vermont Yankee Containment Safety Study". This Study evaluated and estimated the probability of a large release of radioactivity from the Vermont Yankee Nuclear Power Plant as a result of a severe accident. The study employed data and methodology from a technical discipline commonly referred to as probabilistic risk assessment. A brief description of this discipline along with an evaluation of its strengths and limitations is provided in the following subsection.

<u>I.A. Probabilistic Risk Assessment</u> - In 1975, the U.S. Nuclear Regulatory Commission published the Reactor Safety Study (WASH-1400) This document provided the results of the first serious attempt in the U.S. to quantify the risks from the operation of nuclear power plants. The study employed a relatively new technical discipline refered to as probabilistic risk assessment (PRA). In this approach, accident intitiating events are postulated and their frequency is estimated based on actual

plant operating experience or application of other data. These initiating events are used as starting events on "event trees", which are basically logic diagrams which consider the plant response to various combinations of safety system success or failure following the postulated initiating event. Fig. 1 provides an example of an event tree prepared for a nuclear power plant PRA study. The headings on the event tree represent safety systems (or safety functions which are supplied by systems within the plant). The horizontal lines across the event tree represent the various possible accident sequences which can occur following the initiating event. The lines enter each event tree heading from the left and branch into two segments. The branches represent success or failure of the system represented by the event tree heading, with the top branch indicating success and the bottom branch indicating failure. In some instances, there is no branching, which indicates that the system has failed as a consequence of an earlier failure of a different system. The column on the far right of Fig. 1 indicates whether each of the various sequences results in successful cooling of the reactor core (s), the core overheates and releases radioactivity to the containment (cm), or the sequence is transferred to another event tree due to a failure (LOCA).

The next step in the PRA process is to quantify the probability of the accident sequences delinIated on the event trees. This is done by combining the frequency of the initiating event with the success or failure probabilities, as appropriate, for each branch point of the individual accident sequences. The failure probabilities for each of the event tree headings is generally determined by one of two methods. If the event tree heading represents a system for which sufficient data exists from testing or actuation, then failure probabilities may be obtained from this source. If insufficient data exists, then failure probabilities are established by the fault tree technique. In this instance, fault trees are constructed for the event tree system of interest. The fault trees may also be thought of as logic diagrams which subdivide the system into its mechanical, electrical, and hydraulic components which are arranged through the use of logic gates such that the manner in which individual components contribute to system failure are depicted. An example of a fault tree is given in Fig. 2. By assigning each individual component a failure rate which is obtained from applicable data bases, the failure rate of the system can be calculated.

After the probability of each accident sequence has been computed, the

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probability of various containment failure modes is estimated for each sequence or group of sequences which have similar characteristics in the context of imposing a threat to the containment integrity. The containment failure probabilities are estimated by analyzing the containment response to each of the accident sequences, and assessing the liklihood of various failure modes. This process may utilize containment event trees in which the event tree headings are represented by the containment safety systems.

Following the assessment of containment failure modes and associated probabilities, a radioactive source term is estimated for each containment failure mode. This source term defines the amount of each important radioactive species in the core which is expected to be released to the environment following each of the containment failure modes. This source term is then used to calculate the public health effects given a specific containment failure mode.

By combining the accident sequence probabilities with the containment failure mode probability and the public health effects associated with each containment failure mode, an estimate of public risk can be obtained. This risk is generally expressed as either an early fatality which results from large doses of radiation, or a latent health fatality (cancer) which can result from lower doses of radiation.

other risks should also be identified, but the two primary of interest are indicated.

This description is a very simplified overview of the PRA process. In reality, the process can be extremely complex and exhaustive. A major nuclear plant PRA may evaluate millions of accident sequences. Such an effort typically requires several million dollars and requires 40 or more man years of effort. As will be discussed in later sections, the Vermont Yankee Containment Safety Study is a limited PRA study which does not, and was not intended to, include all of the elements of a full scope PRA.

Since the completion of the first major PRA study in 1975, there have been some 35 additional PRA efforts completed for a variety of plants in the U.S., and several PRAs have also been completed for foreign plants. In general, these studies have estimated low public risks from the operation of nuclear power plants. Typically, the probability of a serious (core damage) accident has been found to be on the order of one chance in ten thousand per year (usually written as 1×10^{-4}/yr.) or less, and early fatality risks have generally been estimated at less than one chance in a million per yr. ($1X10^{-6}$/yr.) for a person residing within one mile of the plant. Similarly, latent cancer fatalities have been generally estimated at less than one chance in a million per yr. For comparison, the average risk of death from accidents and adverse effects for an individual in the U.S. is about one chance in 2500 ($4X10^{-4}$/yr.) and the average risk of death from cancer is $1.9X10^{-3}$/yr. based on recently published statistics for the year 1982 (most recent year compiled)[15].

The advent of PRA in the assessment of public risks from nuclear power has represented a major and significant advancement in the assessment of nuclear power safety. Prior to 1975, no realistic and comprehensive estimate of the risk of nuclear power generation existed. Nuclear power safety was based on designing the plants to withstand a spectrum of design basis accidents which were selected in an effort to encompass what were thought to be all important accidents. This was combined with the defense in depth philosophy which required that multiple barriers (including the fuel cladding, the primary system, and the containment) exist to inhibit the transport of radioactive material from the core to the environment. The PRA approach, on the other hand, provides a systematic and integrated assessment of all conceivable accident sequences coupled with a comprehensive evaluation of plant systems resulting in a numerical estimate of public risk.

this is a gross overstatement

In addition to being used to provide overall risk estimates, the PRA approach has been used extensively to provide some risk perspective on numerous reactor safety issues, and has been a major factor in several NRC decisions regarding nuclear power plant safety

Despite the wide acceptance and extensive use of PRA in nuclear power plant safety and risk evaluations, there remain shortcomings and limitations to the PRA approach. Further, several important areas exist where controversy and disagreement can be found among PRA researchers. Some of the more significant of these problems is discussed below:

a. Inadequate Data- The accurate estimate of risk from the PRA approach requires that statistically significant and applicable data exist to evaluate the frequency of initiating events, system failure rates, and component failure rates. In numerous instances, such data is limited, and in some cases, such as large seismic initiating events, it is lacking altogether.

b. Human Error- During the course of a postulated severe accident, opportunities exist for the plant operators to intervene in an attempt to improve the potential for adequate core cooling. It is also possible for these attempts to have a deleterious effect if the operator commits an error or acts with inadequate or incomplete information from plant instruments. The potential for and probability of such action is extremely difficult to evaluate quantitatively since human behavior under various stress levels is unpredictable.

c. System Dependencies- Detailed system analysis performed as part of past PRAs has revealed the existence of important system dependencies, or instances where a system failure or an initiating event can cause additional system failures by virtue of links between the systems. Sometimes these links can be subtle and troublesome to analyze properly. It is also difficult to demonstrate that all such important dependencies have been found as part of a PRA since there does not exist a validated methodology which is universally accepted for finding dependencies.

d. External Event Initiators- In some PRA studies, accidents which are initiated by events external to or not associated with the operation of the plant have been estimated to be important contributors to risk. External event initiators include earthquakes, fires, and high winds. It has been extremely difficult to assess the frequency of such events because, in general, they have to be more severe than any event recorded at the plant site. Thus, frequency evaluations must rely on extrapolations and judgement. Furthermore, it is not always easy to evaluate the plant response to such events since the important contributors are frequently beyond the design basis of the plant and no experience exists.

e. Severe Accident Phenomena- Since there has never been a severe accident in a nuclear power plant of U.S. design which has progressed beyond core damage within the reactor vessel, the assessment of such progression must depend on analysis and a limited amount of small scale, highly idealized experiments. The physical and chemical processes in these accident progressions are predicted to be extremely complex interrelated events. In several areas, analytical techniques are limited, and debate and controversy surrounds the most likely accident progression scenarios.

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f. Uncertainties- Due to the limitations described in a. through e. preceding, as well as additional factors, all PRA assessments contain a considerable amount of uncertainty. The quantification of, and appropriate acounting for, this uncertainty in the utilization and intrepretation of the PRA results has been an area of controversy. It is usually not possible to rigorously quantify the uncertainties involved in a PRA assessment.

True, but it is possible to identify the areas.

It should be noted that additional research and anlysis is underway in all of the above areas, and it is anticipated that PRA assessments will become more accurate and complete as this work progresses

The foregoing areas describing major limitations and uncertainties with respect to PRA assessments will be described in more detail in subsequent sections of this report as they apply to and influence the results of the Vermont Yankee Containment Safety Study. The next subsection of this introduction provides a brief overview of the Vermont Yankee Containment Safety Study.

I.B Overview of the Vermont Yankee Containment Safety Study- The subject of this report is a review of the document entitled " Vermont Yankee Containment Safety Study" which will be hereafter referred to as the VYCSS. This document describes the results of an assessment of the Vermont Yankee nuclear power plant to determine the probability of a severe release of radioactivity from the plant. The study was a limited PRA assessment performed principally to examine the integrity of the containment structure under severe accident conditions. It does not, and was not intended, to provide an explicit estimate of public risk from operation of the plant. The approach taken in the study was to modify the analysis and results of the Reactor Safety Study (WASH-1400) in an attempt to render them applicable to the Vermont Yankee reactor. The plant used in the Reactor Safety Study was the Peach Bottom nuclear power plant, a plant similar, but not identical to, the Vermont Yankee plant. The modification process in the VYCSS involved adjusting the probability of the Reactor Safety Study accident sequences to: 1) account for Vermont Yankee plant specific design features and 2) utilize more current data and accident sequence progression analysis. A further modification involved a more realistic assessment of the containment response to severe accidents based again on Vermont Yankee plant specific design features as well as increased knowledge regarding containment response and failure modes from research and anlysis conducted since 1975. A more thorough description of the methods, assumptions, and procedures utilited in the VYCSS will be provided in subsequent sections of this report, particularly in those areas that are important to supporting the results.

The principle results stated in the VYCSS are that the probability of core melt for the Vermont Yankee plant is $3\times10^{(-5)}$ per yr. or about one chance in 33,333 per year. The probability of having a large release of radioactivity from the plant was estimated at $2\times10^{(-6)}$/yr., or about one chance in 500,000 per year. In other words, the VYCSS concluded that there was a 7% chance of having a large release following a core melt accident sequence. Conversely, there was a 93% chance that the containment and its support systems would prevent a large release following a core melt accident. The purpose of the review described in this report, as stated in more detail in the first part of this section, was to attempt to ascertain if this result is valid, and, if not, what a more realistic value might be. The summary and major conclusions from this review is presented in the following section.

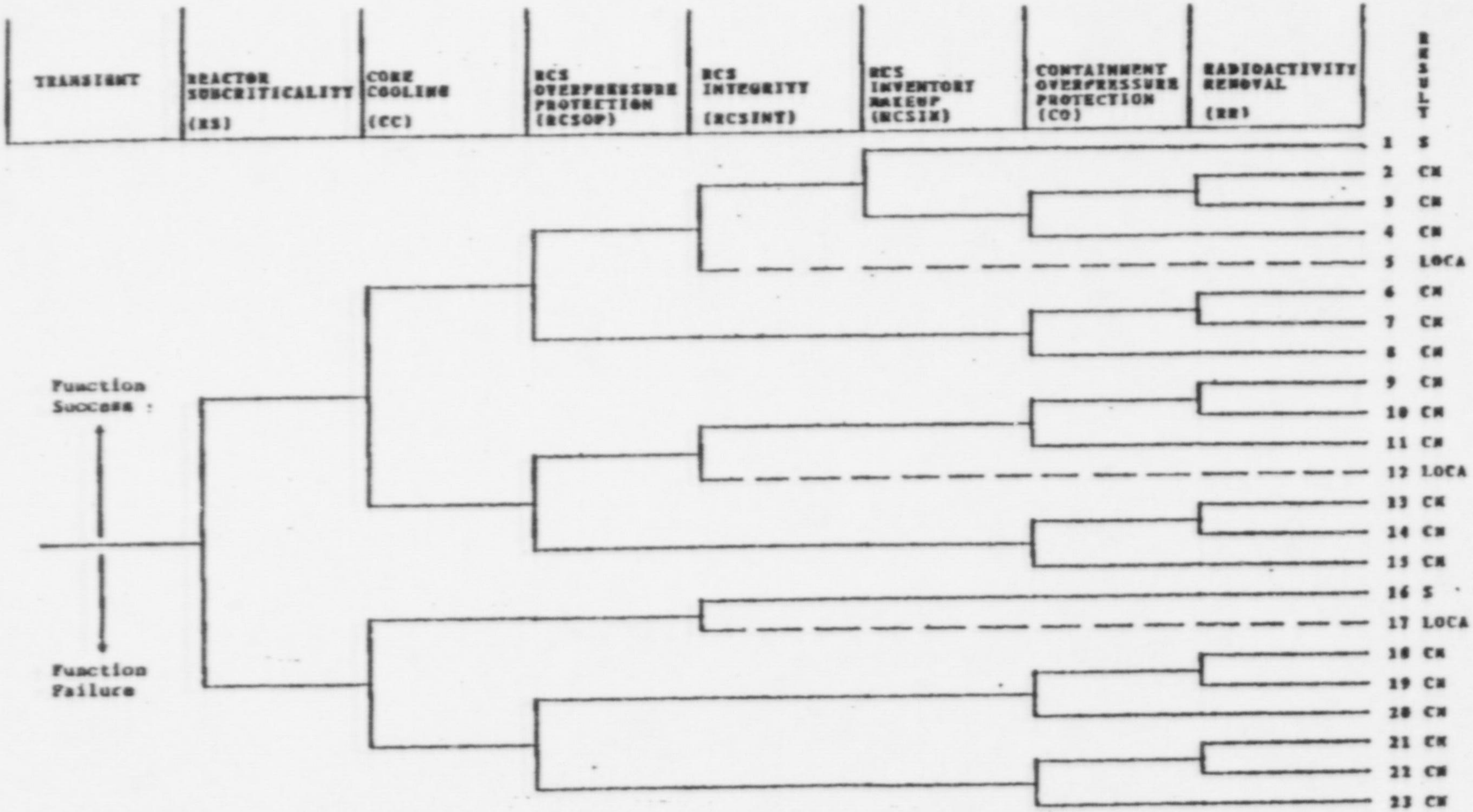


Fig. 1 - Example of PRA Event Tree (from Ref. 16)

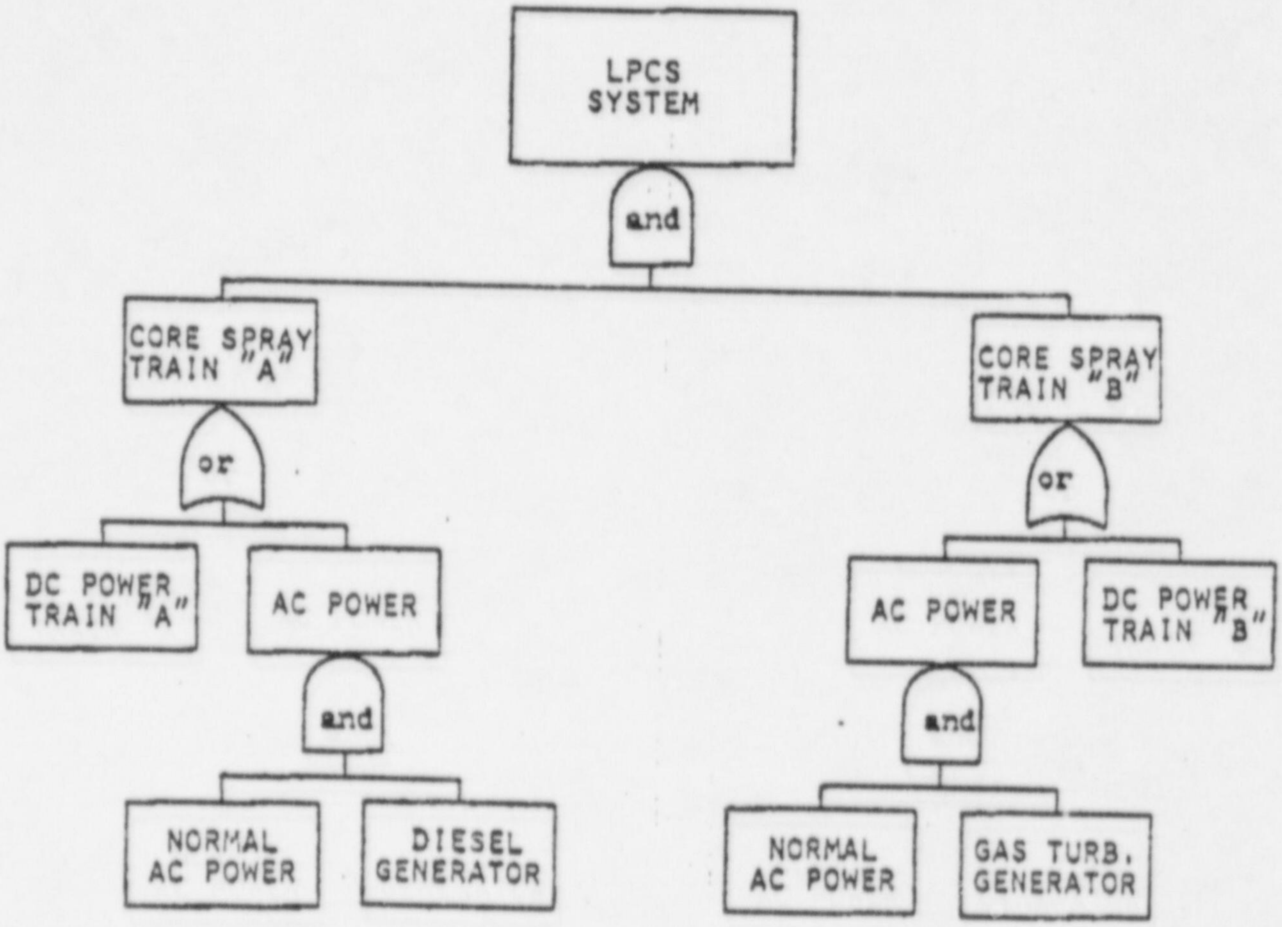


Fig. 2 - Example of PRA Fault Tree (from Ref. 17)

2. SUMMARY AND CONCLUSIONS- The Vermont Yankee Containment Safety Study (VYCSS) provides an estimate of the probability of severe core damage accidents at the Vermont Yankee nuclear power plant and the likelihood and mode of containment failure and resulting radioactive release as a result of such accidents. This review of the VYCSS consisted primarily of comparing the assumptions, analysis, data, and results with current available information regarding the probability and progression of severe accidents in order to determine if the study was consistent with this current knowledge. The report was also examined to determine if any important omissions might exist. Where deficiencies or omissions were found, an attempt was made to evaluate their significance. The primary findings and conclusions of this review are as follows:

A. Overall results - The VYCSS is considered to present a reasonable estimate of containment failure probabilities from severe accidents which could be initiated from internal events only. Although the VYCSS does not provide any uncertainty analysis, and this is considered a major deficiency as noted in Item C following, based on selected sensitivity studies and comparisons performed as part of this review, the VYCSS probability results for internal events appear to be generally consistent with present knowledge regarding severe accident behavior and within the range of large uncertainty associated with such behavior. With respect to external event initiators, it is concluded that seismic events have the potential to increase the probability of containment failure in conjunction with severe accidents at the Vermont Yankee plant.

B. Inadequate characterization of results with respect to limitations regarding omission of external events- The VYCSS is considered deficient in not adequately qualifying the results with respect to omission of external events. An evaluation of the potential contribution from external events is given in Sect. 8.

C. Inadequate characterization of results with respect to uncertainties- The VYCSS does not adequately 1) qualify the results in terms of uncertainties, 2) discuss the extent and implications of uncertainty in the results, or 3) provide a basis for ranges given on the results. This aspect is discussed in detail in Sect. 8, item 2.

D. Inadequate basis for some frequencies and probabilities- The report does not provide an adequate basis for some of the frequencies and probabilities used. Based on a response from Vermont Yankee Nuclear Power Corporation, the basis for these frequencies and probabilities is now considered to be adequately documented (Sect. 8, Item 7).

E. Inadequate description and implication of radioactive release following containment failure- The description of potential consequences from, and characterization of, radioactive release ranges used in the VYCSS are considered inadequate. Details are provided in Sect. 8, item 6.

F. Reasonable assessment of containment failure issues- The VYCSS provides a reasonable assessment of containment failure scenarios and related probabilities. However, the report is considered deficient in not exploring the effect and likelihood of alternative scenarios which would encompass the uncertainty involved evaluating important containment failure modes.

G. Generally adequate assessment of containment intregrity improvements- With a few exceptions with respect to 1) Operation of drywell sprays (Sect. 8), and 2) Drywell liner failure from molten core interaction (Sect. 6,8), the VYCSS appears to adequately address NRC suggestions regarding improving containment integrity as well as other recommendations identified in the VYCSS. Two additional plant changes and modifications with the potential for reducing the probability of a significant release were identified during the review. These changes were 1) increase capacity and reliability of drywell coolers, and 2) changes to the logic and control for the main steam isolation valves. These aspects are discussed in detail in Sect. 7.

3. PROCEDURE- This section describes the procedure utilized in reviewing the VYCSS. It is important to note that resources available for the review were not sufficient to perform an independent assessment of the VYCSS by performing a comprehensive PRA from basic methodology. Indeed, the VYCSS itself did not use this approach as noted in Section 1 preceding. Further, some of the supporting data and analysis referred to in the VYCSS was not obtained in time for the review, and a visit to inspect the plant was not carried out. Such a visit would have been of limited value since the plant is operating, rendering access to the containment impossible. Due to these limitations, the review relied significantly on comparison of the VYCSS assumptions, data, and results with the substantial body of similar assessments (other than the Reactor Safety Study) currently available in an attempt to determine if the VYCSS results were consistent with and supported by this additional information. It should be emphasized and recognized that in several areas, to be noted in subsequent sections, data and information are lacking to the extent that major differences of opinion exist among responsible PRA researchers. Thus, it is not possible to determine which approach or result is correct. These unknowns do not render the VYCSS results (or any PRA results) invalid. They do, however, contribute to uncertainties in the results, and the significance of such uncertainties can be important in qualifying and interpreting the results.

The basic approach employed in the review consisted of five steps, as follows:

a. Accident sequence probabilities- The VYCSS important accident sequences were compared with several other PRA studies for similar nuclear power plants to determine if the assessed probabilities were consistent, and if not, what valid reasons might exist to render them inconsistent.

b. Accident sequence completeness- The VYCSS important accident sequences were compared with several other PRA studies for similar plants to determine if any sequences found to be important in other studies are missing from the VYCSS and, if any were found, to determine if the missing sequences would be applicable to VYCSS and what their effect would be on the results if they were included.

c. Conditional Containment Failure Probability- The VYCSS conditional containment failure probability from dominant accident sequences were compared with two very recent independent assessments for a similar plant design. Differences were evaluated, and an estimate of the effect of any changes in the VYCSS judged to be valid on the basis of the comparison was made.

d. Containment Failure Issues- A review was performed of the current state of knowledge regarding important containment failure modes and processes from severe accidents to determine if the VYCSS assessment reflects appropriate consideration of this information, and if the VYCSS results in this regard are properly qualified and uncertainties acknowledged or accounted for. The potential effect of any discrepancies in this area were evaluated.

e. Containment Integrity Improvements- A review was made of various suggestions and analysis for providing plant modifications to improve the ability of the Vermont Yankee containment to sustain the loads imposed by severe accidents. Such improvements include those suggested by the NRC, the VYCSS, and others found potentially beneficial as part of this review.

f. General Deficiencies- Any additional deficiencies, not directly related to the above areas, were identified, described, and their implication evaluated to the extent practical.

The following Section provides the results of the first two review areas (a. and b.), while Section 5 provides the results of the review of item c. Section 6 considers item d, Section 7, item e, and Section 8, item f.

4. REVIEW OF VYCSS DOMINANT ACCIDENT SEQUENCES

This section presents the results of a comparison between dominant accident sequences from several PRA studies to those found to be dominant in the VYCSS. The main focus of the comparison was to determine if any important sequences might have been left out of the VYCSS, or if any inconsistencies appeared to exist in establishing accident probabilities. These two aspects are considered separately in subsections which follow. It was considered important to make this comparison primarily because the VYCSS relies on the sequences found in the Reactor Safety Study which is now some 11 years old. Several assessments of plants similar to VYCSS have been completed in the interim, and these are used as the basis for the comparison.

Prior to proceding with the results of the comparison, it is useful to re-arrange the VYCSS results. This will facilitate subsequent evaluations on the significance of differences which may be found, and will provide some perspective on the VYCSS results which are not currently directly displayed in the report.

Table 1 is a re-arrangement of the VYCSS results, and has been prepared based on information contained in Table 4.7, Pg.108 of the VYCSS report. Table 1 lists the accident classes found to be important in the VYCSS study. These classes contain accident sequences which have common features in terms of their impact on the containment, and as such, are characterized by similar initiating events and system failures which are summarized in the second column. The third column provides the computed probability of each accident class. This value is followed by a percentage contribution (in parenthesis) of each accident class to the overall core melt probability. The next column is the fraction (expressed in percent) of of each accident class that was found to fall in the EH or EM radionuclide release categories. EH designates a containment failure mode which results in an early release with high levels of radioactivity, while EM is an early release with medium levels of radioactivity. These designations are important because they represent the two classes of release which were considered to be high release levels in the VYCSS report. All other release categories were considered to be low (including no containment failure) or much later releases implying much lower public health consequences. The last column in Table 1 is the overall fraction (again expressed in percent) that each accident class contributes to the total probability of either an EH or EM release.

Table 1 serves to illustrate some important features relative to the VYCSS results in terms of the overall relationship between accident classes and significant releases. For example, the most significant accident class in terms of contribution to core melt probability is class IA, with a 43% contribution. However, this class is an insignificant contributor to the overall EH and EM categories. On the other hand, class IC is a relatively low contributor to core melt probability (8%), but a very significant contributor to the overall EH and EM categories. This is because these accident sequences were determined to have a relatively high probability of damaging the containment and therefore resulting in a significant release. This is illustrated by the fractional release columns which indicate that 20% of the class IC accidents are expected to result in an EH release. Thus, as illustrated by the table, there is a significant variation in the relative contribution of each accident cateagory to both core melt probability and contribution to significant release. These relationships will become important in evaluating the influence of potential changes to the VYCSS results. For example, if it were found that a more realistic estimate of class IA core melt probability were 1.33E-4, a very significant ten-fold increase, the core melt probability would be increased to 1.63E-4, a factor of about 5. However, the increase in EH and EM release probaiblity would be very modest, under 20% in both cases. Conversely, a modest increase in probability of IC accident class sequences, or an increase in the fractional contribution of these sequences to EH or EM categories would result in a significant increase in IH or EM release probabilities.

4.A. Review of Accident Sequence Probabilities- This review consisted of two phases. In the first phase, the VYCSS results were examined to determine if the adjustments and modifications to the accident sequence probabilities used from the Reactor Safety Study[1] (RSS) appeared to be valid. As indicated previously, the VYCSS did not undertake an independent assessment of core melt probability, but rather used the RSS results as a baseline from which modifications were made. The second phase of the review consisted of comparing the VYCSS results with numerous results from other PRA studies completed for plants similar to Vermont Yankee. These results from these two phases are described below

4.A.a. Modifications to RSS results -

The RSS estimated that the core melt probability for the Peach Bottom reactor is 3.2E-5/yr. The VYCSS result is 3.0E-5/yr. These two results can be considered identical since the uncertainties in the estimates are significantly greater than the difference. However, the accident sequence probabilities which make up the total core melt probability for the two studies were not identical in that the VYCSS made, in some cases, significant modifications to the RSS sequence probabilities on the basis of plant design differences or what was considered to be more applicable data. This difference can be important since different sequences, as noted previously in this section, can have markedly different influences on the likihood and mode of containment failure leading to large releases.

Section 4 of the VYCSS provides a requantification of the RSS accident sequence probabilities. A review of this section revealed no significant errors or omissions, and it is therefore concluded that the VYCSS requantification of RSS accident sequence probabilities is reasonable and no significant changes to any accident sequence probability were found to be warranted. However, two general deficiencies of note were found: 1) The basis for some of the numerical changes made to the RSS sequences were not adequately delinlated, and 2) There was an inadequate treatement of uncertainties in the requantification process. The first problem was largely resolved on the basis of answers to questions provided by the Utility (Vermont Yankee Nuclear Power Corp.) as given in Appendix A of this report, and the second deficiency is explored further in Sect. 8. Further comments on Section 4 of the VYCSS are also provided in Sect. 8, but none of these are judged to be significant in the context of the overall VYCSS results and conclusions.

4.A.b. Comparison with Recent PRA Studies

Since the publication of the RSS, as noted previously, there have been numerous additonal PRA studies completed for U.S. plants. Several of these have been for plants very similar to Vermont Yankee, which is a Boiling Water Reactor, Type 4, with a Mark I containment. For the purposes of this comparison, four other PRAs for Type 4 BWRs will be used. Table 3 lists the plants for which these PRAs were completed, gives the organization which sponsored the study, and the performing organization. As noted in the Table, the four plants are Peach Bottom, Limerick, Browns Ferry, and Shoreham. Two of these plants (Peach Bottom and Limerick) have Mark I containments, while Limerick and Shorenam have Mark II containments. However, for purposes of core melt comparisons, the difference in containment design is not considered important. The BWR Type 4 design is essentially identical in terms of systems included which can influence the probability and timing of core melt accidents[7].

As noted previously, there have been two independent assessments for the Peach Bottom plant since the RSS was published. The first such Peach Bottom results, as indicated in Table 3, are from an NRC updating of the RSS assessment. These results are not yet published, but were presented at a recent meeting of the Advisory Committee on Reactor Safeguards(2). The second Peach Bottom assessment is from the Industry Degraded Core Rulemaking (IDCOR) program which is a program to provide an independent (from NRC) evaluation of severe accident response in nuclear power plants.

The Limerick and Shoreham PRA studies (references 8 and 9) have undergone NRC review. These reviews (references10 and11) resulted in a requantification of accident sequence probabilities in those instances where the reviewers felt the PRA study was deficient.

The results of all of the assessments described in the preceding are presented in table 3. In this table, the dominant accident sequence probabilities have been arranged into the same accident classes, as appropriate, that were used in the VYCSS in order to facilitate comparisons. The table thus shows the accident class probabilities for each study, as well as the total core melt probability (in the last row).

Comparing the total computed core melt probability (last row of table 3) shows that the VYCSS estimate is about in the middle of the results given, with three assessments lower than VYCSS, and four greater. The VYCSS result is about four times greater than the lowest value (IDCOR-Peach Bottom), and about seven times less than the higest (Browns Ferry). This comparison illustrates two observations; 1) the VYCSS results is within the range, and therefore consistent with, other core melt probability assessments, and 2) there is substantial variation among the results used in this comparison (the range extends to almost a factor of 26 from lowest to highest result). The reasons for these substantial variations are not considered in depth in this study. The differences are primarily related to credit given for operator action, the use of different data bases, and, to a lesser extent, plant specific design differences and site specific initiating events (such as loss of off site power).

In examining individual accident class probabilities, it was found helpful to combine some of the VYCSS classes in order to render the results comparable, since they were considered as single sequence classes in some of the other studies. These include classes IA and ID as well as IC and IV as indicated in the first column. Further, some of the studies included accident sequence classes which were not considered as separate in the VYCSS. These include those designated as TPQI and TPQE on the table. In these cases, the sequence resulted in a loss of coolant accident (stuck open relief valve) and were apparently considered as LOCAs in the VYCSS (and RSS) study, or accident class III.

In comparing the accident class proabability results from table 3, it is noted that, in some cases, substantial differences exist. For classes IA, ID (loss of make-up water to the core) the VYCSS results are within and nearer the higher end of the probability range, and thus are consistent with the other assessments. For class IB, the VYCSS results are also within the probability range of the other assessments, although near the lower end of the range. This sequence is loss of all AC power (plant blackout), and was assessed in the VYCSS to have a somewhat lower probability than the RSS because of the proximity of, and electrical connection to, the Vernon hydro electrical generating station. The credit assessed in the VYCSS for this additional power source was found to be reasonable in terms of reducing the probability of AC power loss.

For the next combination of classes. IC and IV (anticipated transients without scram) the VYCSS probability assessment is also within the range of other assessments, being about a factor of 5 higher than the lowest (NRC assessment for Peach Bottom) and a factor of about 12 less than the highest (Browns Ferry). The major reason for the difference between the VYCSS ATWS probability estimate and that for Browns Ferry as assessed by EG&G is credit taken in the VYCSS for implementation of the ATWS rule[8]. This rule is being implemented at BWR plants and contains features which will have a significant influence in reducing the probability of ATWS. The rule was being formulated at the time the Browns Ferry assessment was published (1982). Based on the review of the VYCSS and this comparison of other assessments, it is concluded that the VYCSS estimate of ATWS probability is reasonable.

Do we know when?

For accident class II (loss of containment heat removal), the VYCSS assessment is again within, although closer to, the higher probability end of the range of other assessments. The VYCSS result is a factor of about 200 higher than the lowest (Peach Bottom, NRC), and a factor of 50 lower than the highest (Browns Ferry). This class therefore has a very large range. The primary reason for the difference among the assessments is the credit taken for operator action and use of plant equipment not specifically designed for containment heat removal. For example, the two assessments for Peach Bottom take credit for manual venting of the containment drywell. This accident develops very slowly (many hours), and time is available for positive operator intervention and alignment of other equipment which may be available at the plant. The VYCSS result is considered reasonable and consistent with other results although substantial uncertainty, to be considered later, is associated with the probability estimate for this class.

Table 3 shows for accident class III (loss of coolant accidents) that the VYCSS result is at the high end of the probability range but within a factor of 7 of all assessments except the very low Limerick result. This is considered reasonable agreement.

The V accident class (containment bypass) was not found to be a significant contributor for any of the assessments except that the Shoreham PRA assigned this class a probability of 2E-7, which is not a significant contributor (less than 1%) to the Shoreham core melt probability of 4.4E-5. Since the VYCSS did not provide a basis for the conclusion that this sequence would be a negligible contributor, the utility (Vermont Yankee) was asked for the basis. Their response is included in Appendix A as response #10. The response indicates that this accident class is estimated to have a probability of 1E-7, quite comparable to the Shoreham result, and insignificant in terms of the overall VCYSS core melt probability (less than a 1% contributor). It should also be noted that, based on Table 1 results, this accident class would not be a significant contributor to the probability of a large release even if a large fraction of the accident class contributed to high releases.

As shown in Table 3, the two accident classes (TPQI and TPQE) not considered separately in the VYCSS were not found to be significant contributors (less than 10%) for any of the studies.

From this comparison, it is concluded that the VYCSS assessment of core melt probability is consistent with other independent studies and PRA reviews. Further, all important accident classes appear to have been considered. As noted in the Table 3 comparisons, the overall core melt probabilities ranged from a factor of 7 higher than the VYCSS result to a factor of about 4 lower.

TABLE I–Distribution of Significant Radionuclide Release per Accident Class from the Vermont Yankee Containment Safety Study

ACCIDENT Class	Type	Core Mlt. Prob(%)	Fractional % EH	Fractional % EM	Overall % EH	Overall % EM
IA	Loss of Makeup-High Pressure	1.33E-5 (43)	0	0	1	2
IB	Plant Blackout	6.20E-6 (20)	0	2	1	7
IC	ATWS-Loss of Makeup	2.60E-6 (8)	20	11	96	17
ID	Loss of Makeup- Low Press	3.90E-6 (13)	0	1	0	3
II	Loss of Containment Heat Removal	2.10E-6 (7)	0	26	0	33
III	Loss of Coolant Accidents	7.30E-7 (2)	0	0	0	0
IV	Anticipated Transient W/O Scram	2.20E-6 (7)	0	30	0	40

Table 2. PRA Studies for Boiling Water Reactors, Type 4

Plant	Sponsor	Performing Org.	Date Published	Reference
Peach Bottom	U.S. NRC	Sandia National Labs	(early 1987)	(2)
Peach Bottom	IDCOR	IT Corp; various contractors	1986	(3)
Limerick	Phil. Electric Co	Science Applications, General Electric	1983	(4)
Shoreham	Long Is. Lighting Co.	Science Applications	1983	(5)
Browns Ferry	U.S. NRC	EG&G Idaho Inc.	1982	(6)

TABLE 3-COMPARISON OF DOMINANT ACCIDENT SEQUENCE PROABILITIES

including external events?

CLASS	VYCSS	PEACH BOTTOM IDCOR	PEACH BOTTOM NRC	BROWNS FERRY	LIMERICK PRA	LIMERICK NRC	SHOREHAM PRA	SHOREHAM NRC
IA,ID	1.72E-5	4.1E-8	2.4E-7	5.5E-7	6.0E-5	6.0E-5	6.8E-5	5.0E-5
IB	6.2E-6	4.5E-7	8.7E-6	2.9E-5	3.1E-5	3.1E-5	1.3E-5	1.3E-5
IC,IV	4.8E-6	7.3E-6	1.0E-6	5.5E-5	3.7E-6	3.7E-6	4.5E-5	4.5E-5
II	8E-7	1.5E-7	1.0E-8	1.0E-4	3.2E-6	3.2E-6	1.1E-5	9.0E-6
III	7.3E-7	1.4E-7	1.1E-7	(1)	2.4E-9	(1)	5.5E-7	(1)
V	(2)	(1)	(1)	(1)	(1)	(1)	2.0E-7	(1)
TPQI(3)	(4)	(5)	(5)	1.1E-7	9.0E-7	9.0E-7	1.7E-7	1.7E-7
TPQE(6)	(4)	(5)	(5)	1.0E-7	(5)	(5)	(5)	(5)
TOTAL CMP	3.0E-5	7.9E-6	9.8E-6	2.0E-4	1.5E-5	9.9E-5	4.4E-5(7)	1.2E-4

(1) Not found to be a dominant accident class
(2) Estimated by Vermont Yankee to be 1.0E-7/yr., see Appendix A, Answer #10
(3) Transient accident with stuck open relief valve followed by loss of containment heat removal
(4) Considered, as in the RSS, to be a loss of coolant accident and included under III
(5) Not found to be a separate dominant accident class
(6) Transient accident with stuck open relief valve followed by core injection failure
(7) This value is actually described as a "core vulnerable" condition in the Shoreham PRA, and a small factor was applied to estimate the core melt probability given a core vulnerable condition.

5. COMPARISON OF CONTAINMENT FAILURE PROBABILITY WITH RECENT RESULTS- This section provides a comparison between the VYCSS assessment of containment failure probability and mode with two recent similar assessments for BWR Type 4 plants with Mark I containments. These assessments were chosen because they are the most recent available, they were done for a reactor and containment design (Peach Bottom in both cases) similar to Vermont Yankee, and they were performed by two different agencies, namely the IDCOR group and the NRC and their contractors. It should be noted that many of the assessments used in the preceding section are not suitable for this comparison for various reasons. The Limerick and Shoreham assessments, for example, are for plants with Mark II containment designs and would not be suitable for the Mark I design of Vermont Yankee. Further, the Browns Ferry assessment used in Sect. 4 did not include an independent assessment of containment failure probability.

The IDCOR and NRC assessments of Mark I containment failure probability assessment have been recently summarized in a draft report(9) the results of which were presented at a recent ACRS meeting(2). These results were used, with some intrepretation, to prepare table 4. It should be noted that the results, particularly the NRC results, are preliminary and subject to revision as their assessment is refined. Table 4 considers two accident sequence classes; Station Blackout (designated as Class IB in Table 3 and in the VYCSS assessment) and ATWS (Classes IV and IC). These sequences have been found in previous PRA assessments to be the dominant contributors to the proability of containment failure resulting in a significant release. Examination of Table 1 reveals that these two sequences were dominant contributors in the VYCSS assessment also. For example, ATWS sequences contributed 96% to the probability of the most serious release (EH), and ATWS sequences contributed 57% to the EM release probability. The Station Blackout sequences were less significant, contributing only 1% to the EH category, and 7% to EM.

Table 4 shows the fractional release probability for the two sequences for each of the studies. It should be emphasized that the descriptors high, med. and low are qualitative and are based on judgement regarding the similarities of the releases. The footnotes are provided to give additional information regarding the mechanics assumed for each release. Thus, these categories are equivalent only in a rough sense and are used in an attempt to assess any significant differences rather than as the basis for a requantification of the CYVSS results.

The table indicates that for the Station Blackout accident, the VYCSS would estimate a much lower conditional failure probability of a high or medium release than either of the IDCOR or NRC studies. As noted on the table, the primary containment failure mode contributing to these releases from the IDCOR and NRC studies are overtemperature failure causing drywell leakage, and drywell liner failure from contact with molten core debris. The primary reason for this difference appears to stem from two factors: 1) The VYCSS assumed a lower likihood of drywell liner failure because of the smaller amount (about 1/2) of molten core material available due to the lower power level coupled with the fact that the drywells for both the Peach Bottom and Vermont Yankee plants are similar in size, and 2) The VYCSS assumed that drywell sprays would more likely be available because of the possibilty of using a diesel driven fire pump which can be connected through the RHR system and used to pump water to the sprays under Station Blackout conditions. As pointed out in the VYCSS, the use of drywell sprays would inhibit both drywell liner/molten core interactions as well as overtemperature drywell failures.

The CYVSS reasoning is considered valid but not conclusive in terms of the effect of less core material reducing the likihood of molten core/drywell liner interactions.(This issue is explored in depth in the following section). However, the use of the drywell sprays appears to be questionable due to the procedures involving their use. This issue is explored in Sect. 8 of this report. Given these considerations, and the further discussion in this report, it is extremely difficult to assess the conditional probability of drywell failure for the station blackout sequences. There is very little applicable experimental data, and any estimate relies heavily on assumptions and analysis, neither of which have been verified. To assist in evaluating the significance of this issue, a simplified sensitivity study was undertaken. By use of the information in table I in conjunction with table 4.7 of the CYVSS assessment it can be readily shown that raising the VYCSS conditional probability of a high release to .55 (consistent with the NRC estimate) would raise the conditional probability of a high or medium release (given a core melt) from the current CYVSS estimate of 7% to about 20%. This is a rather modest increase for such a drastic change in the conditional probability of a high release from the station blackout sequence. The .55 conditional probability is considered an upper limit for CYVSS because of the features mentioned which should inhibit the drywell liner failure mode. On the other hand, it does not seem possible with present knowledge to rule out such a conditional probability value. Thus, while the CYVSS assessment in this regard is considered reasonable, the uncertainties are quite large.

For the ATWS sequences, the CYVSS assessment is between the IDCOR and NRC assessments. Given an ATWS sequence, the conditional probability of a high or medium release is .22, .51, and .31 for the IDCOR, NRC, and CYVSS assessments respectively. This is considered reasonably good agreement. To assess the sensitivity of these values and to provide some insight on the potential range of uncertainty, the CVYSS results were recomputed using the NRC results. The NRC fractions were applied to both types of ATWS sequences (accident classes IC and IV in the VYCSS). The results showed that the conditional release probability for an EH or EM release would be raised to about 10% compared with the 7% for the existing VTCSS estimate. This change is considered insignificant in view of the overall uncertainties.

Thus, it is concluded from this comparison that the Vermont Yankee conditional containment failure probability for a moderate or high early release could be as high as about 20%, but this is considered an upper limit. Further, no deficiences were found in the CYVSS which were judged to lead to a significant potential for changing the best estimate results. However, there were some problems encountered in assessing the use of drywell sprays to mitigate drywell containment failures. These are discussed and evaluated in Sect. 7.

6. ASSESMENT OF ISSUES RELATED TO CONTAINMENT FAILURE MODES AND EFFECTS- In this Section, the important containment failure modes considered in the VYCSS are evaluated. It should be noted that the time available for this review was limited, and therefore the focus is on the key modes of containment failure (see Sect. 5 preceding for the indentification of these important modes).

6.A. Specific Containment Failure Modes- The important containment failure modes are those with potential for causing early failure when the fission products might be suspended in the containment. Because the complete details of the particular Vermont Yankee reactor geometry were not available, use was made of data provided in the VYCSS Appendices, in conjunction with the Peach Bottom BWR4, Mark I design as a reference.

Based on the discussion in the VYCSS, important containment failure modes are those which do not meet the following criteria:

> The containment should remain intact without excessive leakage for at least 24 hours and the following limits must be satisfied to insure this condition: 1) minimize spread of molten core material to the drywell liner, and 2) do not exceed over-temperature or over-pressure limits of the containment drywell.

These two limits should be considered in conjunction with both high pressure and low pressure core melt sequences. The containment failure modes with the potential to violate these criteria are considered separately, for high and low pressure sequences, in the following subsections.

6.A.a. Drywell Liner Melt-through- The melt-through of the drywell steel liner is a mechanism for failure for the Browns Ferry plant that was identified during the NRC sponsored containment loads working group meetings held in 1984. In the problem that was considered as part of this effort, the whole core of the reactor was assumed to be released from the reactor vessel. The geometry of the Browns Ferry plant is similar to Vermont Yankee in that it has one doorway exiting from the pedestal region (sub-pile room), and has floor sumps located within the sub-pile room. The analysis of the thermal attack for this case indicated that the drywell liner melt-through would be assured under conditions of: a) large melt pool contacting the drywell wall (6" depth), b) relatively high molten pool temperatures (2550K), and c) no significant heat losses from the molten pool boundary. The situation appears to be different than this for Vermont Yankee as alluded to on page 136 of the VYCSS due primarily to the smaller core providing only a 1" molten core pool depth for uniform spreading conditions out to the drywell wall. However, details of the quantitative basis for the VYCSS evaluation are not clear, and the range of uncertainty is not addressed. Three important aspects that affect the actual behavior will be considered: 1) specific geometry of the Vermont Yankee floor, 2) actual temperature of the melt at time of contact with the drywell liner, and 3) effect of pressure in the primary system at time of vessel melt-through.

First, the specific geometry is important because this will have a large effect on the material motion. If there are sumps within the sub-pile room, they they may be capable of accomodating a significant melt mass following molten core melt-through of the reactor vessel. For example, in recent analysis by Sandia National Labs for the NRC, the Peach Bottom sumps were found to have a capacity of 235 cubic feet, which is capable of accomodating the equivalent mass of 20% of the core. Furthermore, the melting process is now not considered to be instantaneous, but a more realistic continuous meltdown over tens of minutes. In this case the initial melt mass may be completely accommodated by the pedestal sumps, and as the molten melt continues to exit the reactor vessel, molten core-concrete interaction would provide a cavity minimizing the spread of the melt out of the sub-pile room. This scenario is considered a reasonable best-estimate scenario at this time for those accidents in which the primary system is at low pressure during core melting. As an upper limit, if the effect of the sumps and the progressive melting behavior is ignored, then a molten pool can form, spreading out of the sub-pile room. In this case, for even distribution over the floor, the 1" depth used in the VYCSS is valid, and this is considered insufficient to cause liner failure. However, the melt must exit through the sub-pile room doorway, and may preferentially pool up at the doorway exit and attack the drywell liner adjacent to the doorway. Therefore, a reasonable modification to prevent this scenario could be to enhance the uniform spreading of the melt by, for example, small concrete dams on the drywell floor. Given uniform spreading of the melt, drywell liner melt-through seems quite improbable.

The second consideration is the melt temperature. At this time, the initial temperature of the melt exiting the core is not well known. Therefore, it is prudent to consider relatively high initial melt temperatures (2800-3100K) and then consistently take into account the various ways the melt will lose energy as it proceeds toward the liner wall. Some of the heat sinks would be: 1) heat loss to the water and steel in the reactor vessel lower plenum, 2) heat loss to the concrete floor, and 3) heat loss by radiation and convection to the atmosphere and structures over the pool. This final consideration is one of the boundary conditions that must be also considered for temperature conditions in the drywell. Given the 1" depth of the pool from the VYCSS analysis, the conclusion that the melt will be relatively cool at drywell liner contact is considered reasonable. However, due to the uncertainties as noted previously, it is considered prudent to consider thicker localized pools to address the sensitivity of the analysis to these conditions.

6.A.b. Direct Heating- Finally, the effect of pressure in the reactor vessel at the time of lower plenum failure is important. When the pressure vessel is at low pressures, then the previous qualitative description and considerations apply. If the pressure vessel is at high pressure, the melt will be ejected at high pressures and dispersed throughout the drywell. In this situation, the melt will contact a number of solid surfaces including the drywell liner wall as dispersed droplets. Therefore, as the VYCSS states, the heat load on structures will be more uniform. However, the specific geometry of the drywell is again important and must be considered to evaluate this case. For example, the location of the pedestal doorway exit should be considered. As discussed in the preceding, there is only one doorway exit from the sub-pile room. This suggests that the melt ejected with the high pressure gasses will preferentially impact the drywell liner wall adjacent to the door. It is not clear that this design specific situation was considered in the VYCSS analysis. It would be useful to quantitatively examine this scenario to determine if it could be a consideration for a reasonable range of core melt masses and temperatures.

For this particular high pressure case, overpressure is probably not a concern because the pressure rise will be accommodated by the suppression pool. The rise time for direct heating pressurization is slower than the time it takes to clear the vents in the suppression pool. Therefore, the direct heating energy is transformed into suppression pool heating (and possibly vaporization) which means that overpressure failure of the drywell is unlikely from this phenomenon.

6.A.c. Over-temperature/Over-pressure Failures- Failures from excessive pressure and/or temperature may also occur. These failure considerations seem to be reasonably considered in the VYCSS. However, the effect of uncertainties on the heat up and pressure build-up in the drywell was not specifically discussed.

With a uniform pool depth, the over-temperature failure potential may be important and is influenced by the effect of the heat loss from the surface of the pool to the drywell structures above the pool. For example, it is important to consider the presence of miscellaneous structures in the drywell near the pool surface. It is not clear to what extent these were included in the VYCSS analysis, and the effect of related uincertainties on the heat up of the drywell structures. Also, in the structural analysis there was no specific mention made of the locations for the critical drywell penetrations (except for the vents to the suppression pool) that could be affected by the liner temperature rise. If these penetrations are low in the drywell, near where the molten pool could be, then the effect of heating could be important to drywell integrity. It would be useful to provide sensitivity calculations to correlate penetration temperatures with heat flux from the molten pool.

In the case of the molten pool concentrated within the sub-pile room, the over-pressure condition would be of concern particularly if the metallic concentration in the pool is high (for example,from molten steel structures in the reactor vessel). In this case, the molten-core concrete interaction would hold the pool temperatures at initially high levels causing more noncondensible gas production and fission product release from the pool. In the VYCSS, this possible configuration did not seem to be evaluated, and would appear to be a useful additonal consideration. It should be noted that the large containment volume relative to core volume of the Vermont Yankee plant will tend to delay the over-pressure failure.

6.8 Conclusions- The important containment failure modes in the VYCSS were reviewed. The review indicates that for the cases considered in the study, reasonable conclusions have been reached. However, ther are a number of variations of the scenarios that were not considered that could have an impact on the three modes of early containment failure. Consideration of these scenarios would provide additional support that the VYCSS containment failure probabilities are reasonable, and would provide some perspective on the effect of alternative possible scenarios.

7. CONTAINMENT INTEGRITY IMPROVEMENTS- This section evaluates improvements to the Vermont Yankee plant which have the potential for decreasing the probability of containment failure in the event of a severe accident. These improvements fall into three categories which are discussed in separate subsections which follow. The three categories are: 1) improvements suggested by the NRC, 2) improvements identified in the VYCSS, and 3) improvements identified during this review of the VYCSS.

7A. Improvements to Containment Integrity Suggested by the NRC - Section 5 of the VYCSS discusses recent improvements which have been suggested by the NRC for BWR plants with Mark I containments. These improvements were recently (Sept. 23, 1986) discussed in detail during a meeting between the NRC staff and the Advisory Committee on Reactor Safeguards. A review of the transcripts from that meeting[18] indicates that the VYCSS properly describes and characterizes all of the NRC suggestions. The VYCSS evaluates each of the five NRC suggestions and derives an assessment of their applicability in the context of the Vermont Yankee plant design and severe accident evaluation. The following table lists the five NRC suggestions and summarizes the VYCSS response (as described in Sects. 5 and 7) to each:

NRC SUGGESTION	VYCSS RESPONSE
1. Hydrogen combustion control	No additional consideration required
2. Control of molten debris in drywell	No additional consideration required
3. Improve reliability of containment sprays	Further study recommended
4. Containment venting	Further study recommended
5. Augment training/procedures for severe accidents	Further study recommended

The remainder of this sub-section evaluates the VYCSS response to the NRC suggestions and provides a judgement on the appropriateness of the response in the context of providing additional measures to ensure containment integrity in the event of severe accidents. Each of the suggestions is considered separately.

7Aa. Hydrogen Combustion Control- The NRC suggestion for this issue consists of two parts; minimize time that containment is not inerted during operation, and minimize the potential for oxygen ingress during the course of severe accidents. With respect to the former, the VYCSS indicates that the Vermont Yankee plant has historically been deinerted only 1.1% of the time with the plant at power, and this is considered acceptable (Pg. 115) by the VYCSS although no basis is provided. In order to evaluate this conclusion, a sensitivity study was done in an attempt to determine the potential influence on containment failure probability by having the containment de-inerted 1% of the time. If it is conservatively assumed that a de-inerted containment will always cause containment failure with either a high or moderate radioactive release, then, using the results of Table 4.7 (Pg. 108) of the VYCSS it can be seen that this contribution would raise the early high or moderate releases from 7% to 8% of the CMP. This is obtained by multiplying the total CMP by .01 and adding the result to the existing VYCSS estimate of EH and EM release probabilities. This conservative sensitivity study shows that the effect of having a de-inerted containment 1% of the time is negligible, and a substantial increase would be required to have a significant effect.

With respect to oxygen ingress during severe accidents, the VYCSS concludes that due to plant modifications in using nitrogen for those systems that could inject air in the drywell, coupled with existing limits for the use of drywell sprays (to prevent depressurization and air ingress through vacuum breakers), oxygen ingress is not a concern.

The VYCSS response to the hydrogen control issue is considered adequate and valid with one exception. The VYCSS does not provide an adequate description and analysis of when and how drywell spray actuation will be inhibited, and how such inhibition will assure that air ingress will not occur. The references to drywell spray inhibition (for example, see Appendix A, Answer #9) in the VYCSS indicate that the drywell spray is inhibited on the basis of excessive drywell pressure and temperature to avoid the potential containment collapse from negative pressure. The inhibition is apparently not based on any consideration of air ingress due to opening of vacuum breakers. The issue of drywell spray actuation during severe accidents at Vermont Yankee is considered further in Sect. 8.

7Ab. Control of Molten Debris in Drywell - The NRC concern regarding molten debris control in the drywell relates to the possibility that the molten debris may flow to the intersection of the drywell floor and steel wall liner. If this occurs, the liner may fail, and a radioactive release may occur which bypasses the suppression pool. The VYCSS concludes that this issue is not pertinent to the Vermont Yankee plant because the small core size will likely prevent the occurrance of molten debris-liner interaction, and the use of drywell sprays will inhibit migration of molten debris to the liner. It is further argued (Pg. 136) that barriers to molten debris migration would be counter productive since they would inhibit drywell spray flow to the sub-pile room.

The issue of molten debris-drywell liner interaction is a controversial aspect of severe accident behavior for Mark I plants. As such, the issue is considered separately in Sect. 6. While the VYCSS arguments on this issue appear pertinent, two considerations appear incompletely assessed. First, as pointed out in the previous sub-section, the use of drywell sprays under severe accident conditions is inhibited by high drywell pressure and temperature. As a result, it is not clear from information in the VYCSS to what extent these sprays can be relied upon to arrest the progression of molten debris across the drywell floor. Second, it does not appear that the VYCSS has fully explored the alternate placement of physical barriers (for example, near the drywell liner wall) which would not inhibit drywell spray flow to the sub-pile room, or the use of small local barriers to assure uniform distribution of the molten debris (See Sect. 6).

7Ac. Improve Reliability of Drywell Sprays- The VYCSS concurs with the NRC suggestion that this aspect needs further study, and several areas are identified, starting on Page 116, to improve drywell spray reliability. These areas are considered appropriate and valid. The problem of inhibiting the spray actuation when high drywell pressures and temperatures exist is not addressed, however (see Sect. 8 for additional disucssion) Further, the possibility of low NPSH for the ECCS pumps identified on page 117 is not considered further in the discussion on reliability. It does not appear, however, that this is a major concern for Vermont Yankee since alternate sources of water (other than the suppression pool) are available.

7Ad. Containment Venting- The VYCSS identifies this suggestion as an issue requiring further study. Various aspects of the benefits and potential detrimental effects are discussed throughout the VYCSS, and these considerations appear valid. The discussion regarding further study of the issue (starting on Pg. 121) also appears consistent with the NRC postion, and encompasses those areas which appear to be important with one exception. The only deficiency noted in the VYCSS with respect to this issue is the lack of any consideration regarding the addition of a completely separate venting system; the evaluation considers only modification to existing plant equipment.

7Ae. Augment Training/Procedures for Severe Accidents- The NRC objective for this suggestion is to assure that the plant operators make the best use of plant systems for the prevention and mitigation of severe accidents. The VYCSS response to this suggestion (starting on Pg. 139) appears appropriate and complete. The only deficiency found was the lack of any discussion regarding the schedule for the implementation of the changes identified for future consideration.

7B. Containment Integrity Improvements Identified in the VYCSS-

Scattered throughout the VYCSS are suggestions and recommendations for additional analysis and study regarding potential plant improvements for reducing the risk from severe accidents. In Sect. 6.2 on page 178, the VYCSS explicitly discusses "continuing efforts regarding severe accident analysis". These efforts are grouped into three categories. The first category covers "procedure changes (which) are recommended for final evaluation and implementation" These changes consist of six items. The first five items all pertain to improved recovery from station blackout scenarios, and the sixth pertains to improved procedures to respond to the ATWS accident. As noted previously in Sect. 5 of this report, the station blackout and ATWS sequences are dominant contributors to containment failure leading to a significant release. Thus, the VYCSS recommendations address those sequences of significance, and they are considered appropriate

The second category of VYCSS improvements are described as "recommended for further study augmented by detailed analysis to ensure that the positive and negative impacts of potential changes are well understood." Three items are recommended, all of which are considered to be appropriate and potentially important. The three items are upgrading of the Emergency Operating Procedures, enhancement of containment spray/reactor injection capability, and enhanced capability for venting of the containment. As the VYCSS points out, all of these items have positive and negative aspects which need to be carefully evaluated before any changes are implemented.

The third category is described as "areas which merit further study", and consists of four items. These include evluation of RHR pump response to high suppression pool temperature, evaluation of nitrogen supply to safety/relief valves, reliability evaluation of the service water system, and reliability evaluation of the standby liquid control system. All of these items appear to be appropriate items for consideration.

While all of the above items appear appropriate and recognize the important severe accident issues, it is not clear which items Vermont Yankee Nuclear Power Corporation has committed to accomplish and on what time schedule. The VYCSS implies that all of the items will be addressed, but they are described as "recommendations", not commitments.

7C. Improvements Identified During the Review of the VYCSS- In reviewing the VYCSS, one of the objectives was to identify additional improvements not considered by the study or suggested by the NRC which appeared to have the potential for improving the probability of maintaining containment integrity in the event of a severe accident. Two such improvements were identified, as follows:

1. Upgrade the capacity and reliability of the drywell coolers- At present, according to the VYCSS, the drywell coolers will be isolated during severe accidents. It also appears that their capacity is insufficient to remove decay heat until tens of hours after core shutdown. However, if the capacity of these units could be upgraded, and their reliability assured during severe accidents, it appears they could be effective in reducing the liklihood of drywell failure from overtemperature or overpressure. They also could remove some of the heat loading on the suppression pool for transients, and extend the time of, or prevent, overpressure containment failure.

2. Reduce the frequency of main steam isolation valve (MSIV) closures for transients and/or make provisions to reopen these valves following transients. This modification could improve the likelihood that the main condensor would be available as a heat sink for some accident sequences. The VYCSS does acknowledge (Pg. 151) the possibility of re-opening the MSIVs for the ATWS event. However, there is no discussion regarding the likely success of such a strategy, or what operator actions would be required. Further, there appears to be no consideration for changing the MSIV closure logic, or re-opening the valves for other, more likely, transient events.

It should be emphasized that neither the cost, feasibility, nor quantitative risk reduction potential was evaluated in formulating the above suggestions for potential improvements. Further, it is not meant to imply by making the suggestions that any plant improvements are necessary to decrease severe accident risks.

8. Miscellaneous Comments and Discussion of Issues

This section presents a listing of comments which were developed in conjunction with the review of the VYCSS report which are not directly related to items discussed in previous sections. Some of these comments were considered significant and an expanded discussion is provided. Those considered significant in terms of impacting the important results are identified and an evaluation of their impact is addressed where appropriate. Editorial and other minor comments not relevant to the interpretation or validity of the results are not included.
The comments and a discussion, where appropriate, follows:

1. Characterization of results- The conclusions given in the VYCSS (page 178) state that "...the best estimate single-point conditional containment failure probability is 7%". Also, the cover letter transmitting the CYVSS (Weigand to Denton, Sept. 2, 1986) indicates (third paragraph) that "Vermont Yankee has concluded that the best estimate of probability of containment failure is once every 500,000 years, which corresponds to a 7 percent containment failure probability in the unlikely event of a serious accident resulting in core melt." These statements appears to be improper characterizations of the VYCSS results for two reasons. First, the probability of containment failure given a core melt is not 7%, but actually 32% based on the results presented on Pg. 108 of the report. The 7% value is the probability of containment failure resulting in an early radioactive release of high or medium magnitude. Second, the one in 500,000 (2E-6/yr.) probability quoted in the cover letter appears to be based directly on combining the VYCSS core melt probability result (3.0E-5) times the containment failure probability with significant release, or 7% (.07). However, on page 58 of the VYCSS it is stated that "As noted earlier (Section 4.1), this approach (referring to the approach used in the VYCSS) is not capable of supporting a "bottom line" value for the core melt frequency at Vermont Yankee" While the meaning and intent of this statement is not entirely clear (there is no corresponding admonition "noted" in Section 4.1, contrary to the statement contention) it seems to imply that overall core melt probability numbers calculated in the study are insufficiently accurate to use in a bottom line sense. While this statement on page 58 may be too severe a characterization of the overall core melt probability results, it seems to undermine the unqualified statement of results in the transmittal letter. This apparent discrepancy is closely related to the uncertainty discussion which follows.

2. Uncertainties- The report is considered deficient in not providing a reasonable assessment of uncertainties. The assessment of uncertainties is said to be beyond the available resources. However, this feature of PRA analyses has been found to be so important and controversial that omitting any discussion detracts from the usefulness and credibility of the assessment. Even a judgemental estimate of uncertainties with appropriate discussion of the basis (as opposed to a rigorous statistical based assessment) would have been an important addition. Instead, the report does not even include much qualitative insight on the significance of uncertainties which would add some perspective to the results. To compound the problem, there appear to be several inconsistent and confusing references to the results in the context of their perceived accuracy. For example, on page 37 it is stated that "Because this parameter (conditional failure probability of the containment given a severe accident as calculated in the VYCSS) was not the intended purpose of earlier analyses, the containment conditional failure probability that can be inferred from such analysis is a conservative upper bound estimate (emphasis added) adequate only for use in the integrated public risk estimates." The VYCSS assumptions, analysis, and data do not appear to support the contention that the results are a conservative upper bound estimate, nor is it clear how these results are judged to be adequate only for use in the (which?) integrated public risk estimates. Further, on page 39 it is conceded that "Because of the limited time available for this Vermont Yankee specific analysis, this uncertainty was not explicitly quantified; rather a 'best estimate' value was developed. However, despite the attempt at a best estimate calculation, there may still be excessive conservatism in the evaluation.

Another characterization of the results in this context appears on page 48, to wit; "In the text which follows the term "best estimate" is used to characterize the processes and results (for the containment failure probability evaluation). This term is intended to indicate that the evaluation was performed in an objective manner that attempted to balance uncertainties... The analysis provides a formal means for investigating possible challenges leading to core melt or containment failure and hence characterizing in approximate terms the probability of containment failure." Again, on page 81, it is stated that "The process has led to a number of procedural insights which could be enhanced to preserve the best-estimate nature of the numerical assessment." Also on page 81, there appears to be a range of uncertainty provided in some of the results, but there is no basis provided. For example, "The best-estimate of successful containment performance for such a definition (early or intermediate failure times with high or moderate release) is in the range of 90 to 98%...Other ranges for various measures of containment performance appear on page 81, and they have ranges of 10%, encompassing the best estimate value. Further, on page 178 (6.0, CONCLUSION) it is stated that "For all severe accident types involving core damage and having the potential of causing a significant radionuclide release, the conditional containment failure probability is 2-10%." These results do not seem to support the earlier contention ("conservative upper bound estimate"), and no discussion is provided indicating how such ranges were derived.

In the comparisons provided in Sect. 5 of this report, some discussion of the ranges of results found in PRAs for BWR4s was given. On the basis of that evaluation, and utilizing other judgements, it is estimated, roughly, that the VYCSS core melt probability has an uncertainty of about a factor of 5, and the containment conditional failure probability for an early release of high or moderate magnitude has a range of perhaps 4% to an upper bound of 20% (vs. the VYCSS "best estimate" of 7%).

3. External Events- The VYCSS excludes consideration of external events (Pg. 54). While this limitation does not invalidate the results for the events considered, it does open the question of the potential influence of external event considerations on the results. Further, the results stated in the report (particularly in Section 6.0, CONCLUSIONS), and in the cover letter, do not express this limitation, nor is there any qualitative judgement offered regarding the potential effect of this limitation. In order to provide some perspective on the possible significance of external events for the Vermont Yankee plant risks a literature search was undertaken to determine if risk assessments for plants in the region of the Vermont Yankee site (i.e. Northeast U.S.) found external events to be significant. It should be emphasized that such a survey cannot conclude that external events are or are not important for Vermont Yankee because 1) As pointed out in Reference 10, external event risks are very plant and

site specific, and therefore external event risks from one site cannot be extrapolated, in general, to another, 2) external event risk estimates are exceedingly uncertain and controversial, and not much confidence can be placed in the results[10], and 3) no PRA has been published for a BWR4 with a Mark I containment in the Northeast U.S. Notwithstanding these limitations, it is useful to assess the overall significance of external event risks from other PRA studies in order to obtain some perespective on trends and potential significance as applied to the Vermont Yankee case.

The results of six published PRAs for plants in the Northeast U.S. which have included an assessment of external event risks were examined in this survey. These include Millstone 3[11], Limerick[4], Shoreham[5], Seabrook[13], and Indian Point Units 2&3[12]. The Millstone and Indian Point plants are PWRs (with reactors built by Westinghouse), while the Limerick and Shoreham plants are BWRs, Type4, with Mark II containments. In summary, in all of these PRAs, external events were found to be significant (in most cases dominant) in terms of contribution to the probability of serious release. In most cases the dominant external event contributor in this regard was seismic events. In all cases, however, the overall risks computed were very low. For example, in the Shoreham case, the estimated frequency of a significant release (capable of causing one or more early fatalaties) was 2.5E-7/yr., and was dominated by seismic events.

The only valid conclusion from this survey is that external events have been found to be important contributors to nuclear plant risks in the region of the U.S. in which Vermont Yankee is located. This implies that such risk contributions could be important for Vermont Yankee.

4 Use of Drywell Sprays- The VYCSS assessment of the potential use and effectiveness of drywell sprays is somewhat confusing and apparently incomplete. For example, there does not appear to be an adequate consideration of the potential for a steam overpressure spike when the spray water may contact the pool of molten debris late in some accident sequences. If this contact does not occur, as might be the case earlier in the sequence, it may be possible for the sprays to condense and cool the dyrywell atmosphere to an extent that the reduction of pressure could open vacuum breakers and allow oxygen to flow into the drywell, raising

the potential for hydrogen combusions. While this possibility is recognized, it does not seem to be evaluated for some of the accident sequences. On Page 115, it is stated that drywell spray actuation limits would actually preclude operator actuation in cases where excessive negative drywell pressures could result. However, it is not clear for which accidents, and at what times this inhibit condition would occor, nor is it clear how this factor has been considered in evaluating the containment response for various sequences.

5. Release Magnitude Perspective -The VYCSS divides release magnitudes into 6 categories (pg. 108), and these magnitudes are briefly described in terms of time intervals and fraction of "equivalent" iodine release. However, there in no discussion provided concerning why these particular time periods are significant, or why the particular iodine fractions were chosen to characterize different release. It would be particularly helpful if the report could discuss the significance of these releases and release times in terms of public health consequences. Without this information, it is not possible to evaluate the detailed significance of the results in terms of release paramaters vs. probability provided in table 4.7. Many other radionuclides (i.e., Cs, Te, Ba, Sr,) can be important contibutors to public risk and their release potential is not explicitly identified in describing the releases.

6. Benefits of Smaller Core Power Level than Reference Plant- On page 48 and 49 several benefits are listed which are said to derive from the fact that the Vermont Yankee plant has a lower core power level than the WASH-1400 plant (Peach Bottom) while the containment volume is essentially identical. Indeed, the Vermont Yankee power level of 1593MWt is only about half that of Peach Bottom (3293MWt), and the containment drywell volumes are comparable (134,000 cu. ft. for Vermont Yankee vs. 159,000 for Peach Bottom according to Appendix A of the VYCSS). These design parameters mean that the core power to containment volume ratio is considerably less for Vermont Yankee than for Peach Bottom, and this difference is said (Pg. 68) to result in much delayed containment failures by overtemperature or overpressure compared to Peach Bottom. However, the primary energy absorbing media for all severe accidents is the water in the suppression pool. The ability of this water volume to absorb decay heat (or core power for ATWS) will be the primary determining factor in temperature and pressure rise rate. From Appendix A of the VYCSS it can

be seen that the Vermont Yankee plant enjoys only a very modest benefit (about 14% greater) when the two plants are compared on the basis of core power to suppression pool volume ratio. Thus, while some benefits would definitely accrue from the relatively low Vermont Yankee core power level, they may not be as significant as implied in the VYCSS.

7. Basis for Frequency and Probability Values - In several instances throughout the report, it was judged that an inadequate basis was provided for the frequency and probability values. The more significant of these instances were as follows:

PAGE NO.	DESCRIPTION
62	Unavailability for Vernon Hydro given station blackout
68	Calculated times for containment failure
74	Conditional failure probabilities of containment failure
75	Unavailability of water injection

The basis for the values used in each of these four instances were requested from the utility during the review. The basis was promptly supplied in all instances, and they appear reasonable and adequate. The utility response to this request is provided in Appendix A to this report.

9. REFERENCES-

1. Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400, U.S. Nuclear Regulatory Commission, Oct. 1975.

2. Presentation by Brookhaven National Laboratories to ACRS Subcommittee on Class 9 Accidents, Sept. 24, 1986.

3. Peach Bottom Atomic Power Station-Integrated Containment Analysis, IDCOR Technical Report T23.1PB, Mar. 1985.

4. Probabilistic Risk Assessment-Limerick Generating Station, Philadelphia Electric Co., 1984.

5. Probabilistic Risk Assessment-Shoreham Nuclear Power Station, Long Island Lighting Co., June 1983.

6. Interim Reliability Evaluation Program: Analysis of the Browns Ferry, Unit 1, Nuclear Plant, EG&G Idaho, Inc., NUREG/CR-2802, July, 1982.

7. Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors, NEDO-24708, Aug. 1979.

8. 10CFR50.62, NRC Rule on Anticipated Transients Without Scram for Nuclear Power Reactors.

9. Prevention and Mitigation of Severe Accidents in a BWR4 with a Mark I Containment (DRAFT), Brookhaven National Laboratory, Aug. 1986.

10. Probabilistic Risk Assessment (PRA) Reference Document, NUREG-1050, U.S. Nuclear Regulatory Commission, Sept, 1984.

11. Millstone Unit 3 Probabilistic Safety Study, Northeast Utilities, Aug. 1983.

12. Indian Point Probabilistic Safety Study, Consolidated Edison Co. and New York Power Authority, Mar. 1982.

13. Seabrook Station Probabilistic Safety Assessment, Public Service of New Hampshire, Dec. 1983

14. Severe Accident Risk Assessment, Limerick Generating Station, Philadelphia Electric Co., April 1983.

15. Statistical Abstract of the United States-1986, 106th edition, U.S. Dept. of Commerce, Bureau of Census, Sept. 1986.

16. Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One-Unit 1 Nuclear Power Plant, NUREG/CR-2787, Sandia National Laboratories, June 1982.

17. Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant, Science Applicatons, Inc., NUREG/CR-3085, Jan. 1983.

18. Transcripts from the Advisory Committee on Reactor Safeguards meeting of the Subcommittes on Containment Requirements and Reactor Safeguards, Sept. 23, 1986.

APPENDIX A

Responses from Vermont Yankee Nuclear Power Corporation to Questions Developed During the Review of the Vermont Yankee Containment Safety Study

PRELIMINARY QUESTIONS ON THE VERMONT YANKEE

CONTAINMENT SAFETY STUDY

QUESTION 1

Page 12 - The paragraph at the top of the page states that failure of the HPCI in the case of a small LOCA would require operation of the ADS to reduce vessel pressure. The first paragraph on Page 11 states that the RCIC may be considered as a backup to HPCI in the event of a very small LOCA. Please indicate for which break sizes the RCIC, acting alone, can be considered a viable means of core cooling.

ANSWER 1

The Reactor Core Isolation Cooling (RCIC) System can deliver 400 gallons per minute makeup water to the reactor. The High Pressure Coolant Injection (HPCI) System can deliver 4,250 gallons per minute of coolant injection to the reactor. Although RCIC's design requirement is to provide hot shutdown core cooling for transients in which the main condenser is unavailable, its capacity of 400 gpm (or approximately one-tenth that of HPCI) can provide makeup to maintain adequate reactor level for very small leaks in the reactor coolant pressure boundary. The actual leak size is a function of the decay heat level, the leak location, and the leak geometry. Typically, steam line leaks of less than two inches in diameter are within RCIC System capacity.

QUESTION 2

Page 62 - Please provide the basis for the Vernon Hydro unavailability values listed on this page.

ANSWER 2

The best available information indicates that the plant was unavailable for a total of 2 hours and 24 minutes in a 21-year period.* The average unavailability is, therefore, 1.3×10^{-5}. In response to a grid collapse, the hydroelectric station must separate from the grid to allow the tieline to remain available. The only active action identified was the automatic opening of a single normally closed feed breaker (b). Therefore, H can be approximated as:

$$H = 1.3 \times 10^{-5} + b \text{ (fails to open)***}$$

$$= 6.6 \times 10^{-4}$$

The estimates for operator inappropriate action are as follows:

		O **
Phase I	0 - 2 hours	.1
Phase II	2 - 4 hours	.05
Phase III	4 - 10 hours	.01
Phase IV	10-24 hours	.01

---

* It should be noted that only two events were recorded: one of 2 hours and 20 minutes and one of 4 minutes. The station recovered quickly from the latter event, which was initiated by a lightning strike.

** (1) BWR Individual Plant Evaluation Methodology.

(2) A. D. Swain and H. E. Guttman, Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications, NUREG/CR-1278.

*** b (fails) to open) = 6.5×10^{-4}/demand, Seabrook PRA.

Assuming that required action (opening) of the normal feed breaker to each emergency bus is addressed in the electric power system model, then two breakers have to close to feed either bus. In addition, it is assumed that the diesel breaker must open (this is conservative since failure of this breaker to close may have been the cause of "diesel failure to supply emergency bus"). So that V can be approximated as:

$$V = 1.7 \times 10^{-3}$$

The factor C reflects those loss of off-site power events that would also render the Vernon Hydroelectric Station Unavailable. A review of 114 off-site power events identified 24 that were caused by extreme external phenomena (e.g., lightning, ice storms, heavy snow, tornadoes, etc.). Events such as saltwater spray and Florida grid instabilities were assessed not be be applicable to the Vermont Yankee site. More detailed analysis of the particular arrangement at Vernon and Vermont Yankee is needed to assess C. However, if it is assumed that not all of the events are applicable to Vermont Yankee, then C could range from 4×10^{-3} to .1.

The upper estimate (0.1) is used for the point estimate quantification in this analysis.

In summary the unavailability of Vernon Hydro as an effective AC power source to the emergency buses given a station blackout is:

$$U = H + O + V + C$$

$$U = 6.6E-4 + 0 + 1.7E-3 + 1E-1$$

	Vernon Hydro Unavailability For Extreme External Phenomena Events	Vernon Hydro Unavailability For Events Other Than Extreme External Phenomena
Phase I 0 - 2 hours	.2	.1
Phase II 2 - 4 hours	.15	.05
Phase III 4 - 10 hours	.11	.01
Phase IV 10-24 hours	.11	.01

The Vernon Hydro unavailability for extreme external phenomena events has been incorporated into the quantification of Vermont Yankee station blackout sequences in B.3.

QUESTION 3

Page 63 - Provide a reference for the "previous evaluations" referred to in the last paragraph.

ANSWER 3

References:

1. SLI-8218, "Inadequate Core Cooling Detection in Boiling Water Reactors," prepared for BWR Owner's Group, S. Levy, Inc., November 1982.

2. SLI-8221, "Review of Shoreham Water Level Measurement Systems," S. Levy, Inc., November 1982.

3. Shoreham PRA Report.

QUESTION 4

Page 68 - The first paragraph refers to "deterministic calculations" to support late containment failures. Please supply or reference these calculations.

ANSWER 4

The severe accident phenomenology was analyzed for several accident sequences using both MARCH/RMA and MAAP 3.0 codes, performed by Risk Management Associates and Fauske and Associates, Inc., respectively. For the station blackout accident sequences, the results of both MARCH/RMA and MAAP 3.0 analyses indicated that the containment failure by overpressure would not occur within 24 hours after the initiation of an accident. Reference: YAEC-1564, Appendices C and D.

QUESTION 5

Page 74 - Please elaborate on the basis for the conditional containment failure probabilities provided at the bottom of the page.

ANSWER 5

The bases for the conditional probability of early containment failure (CI) for each class of accident are as follows:

Accident Class	CI	Basis
IA	10^{-3}	a
IB	10^{-3}	a
IC	.31	b
ID	10^{-3}	a
II	1.0	c
III	10^{-3}	a
IV	1.0	a

Notes

a. The appropriate failure mode was judged to be hydrogen burn. The probability is based on the Shoreham PRA study.

b. The failure probability was estimated as the fraction of accident sequences in Class IC that would result in elevated containment pressure (≥40 psia) before reactor vessel failure.

c. For these sequences, containment is assumed to have failed prior to core melt..

QUESTION 6

Page 75 - Please provide the basis for the estimates of unavailability for water injection given on this page, and indicate why Classes IE, IV, and V are not included.

ANSWER 6

The bases for the estimates of unavailability of effective active mitigation, water injection (I) for each class of accident, are as follows:

Accident Class	I (Given no early containment failure)	Basis
IA	0.01	a
IB	0.14	b
IC	0.01	a
ID	0.1	c
IE	1.0	e
II	1.0	a
III	0.05	d
IV	1.0	e
V	1.0	e

Notes

a. Credit for drywell spray, Low Pressure System, Feedwater System, Core Spray System, and Control Rod Drive System.

b. Failure probability of off-site/on-site power recovery and unavailability of diesel fire pump, weighted by the core melt frequency of each phase in Class IB.

c. Credit for Control Rod Drive System.

d. Credit for Control Rod Drive System and Feedwater System.

e. No credit taken; failure probability of 1.0 was assigned.

f. Credit for Feedwater System.

g. No credit taken.

QUESTION 7

Page 77 - Please provide the basis for the 50% probability for leakage failures given in the last paragraph.

ANSWER 7

Based on the PRA studies of Shoreham and Limerick, a 50% probability is assigned to the leakage failure given a slow overpressure challange to the containment.

QUESTION 8

Page 80 - Please provide a reference for the "recent investigations" referred to in the first full paragraph.

ANSWER 8

Reference: NUREG/CR-4550, "Plant Accident Sequence Likelihood Characterization - Peach Bottom Unit," Draft, Volume III, May 1986.

QUESTION 9

Page 115 - Item 4 indicates that procedures include limits to prevent initiation of drywell sprays to avoid excess negative drywell pressures. Elsewhere in the report, the use of drywell sprays is shown to be an effective severe accident mitigation measure. Please elaborate on when procedures would allow use of drywell sprays during severe accident progressions.

ANSWER 9

Drywell sprays are initiated whenever:

1. Drywell temperature cannot be maintained below the drywell temperature design limit, or

2. Torus airspace pressure exceeds the Suppression Chamber Spray Initiation Pressure (SCSIP), or

3. Containment pressure cannot be maintained below the containment pressure design limit.

These actions are predicated on the torus airspace temperature and pressure being below the drywell spray initiation pressure limit curve. This curve assures that the containment will not collapse or otherwise fail due to negative pressure resulting from the spray initiation and the subsequent evaporative and convective cooling of the containment atmosphere.

QUESTION 10

Page 106 - Table 4.5 indicates that "interfacing LOCA" (Class V) events are estimated to have a "negligible" frequency. There does not seem to be any justification provided in the report for this conclusion. Please provide additional information.

ANSWER 10

Reference: "Peach Bottom Individual Plant Evaluation." The frequency of interfacing LOCA (Class V) was estimated as 10^{-7}/year in PBIPE. This frequency for Class V is assumed to be applicable to Vermont Yankee, and it is negligible for the purpose of evaluating containment conditional failure probability.

QUESTION 11

General - Please provide Reference 11 (YAEC-1564).

ANSWER 11

YAEC-1564 was developed as a detailed description of the analyses used and results obtained in the Vermont Yankee Containment Safety Study. The methodologies selected and their specific application to the Vermont Yankee plant are considered by Vermont Yankee to be new work in the field, and have potentially significant commercial value. Vermont Yankee would be pleased to provide this material, if necessary, for review provided its confidentiality can be assured through formal agreement. We are available to discuss the details of such an agreement at your convenience.

PROPOSED BWR
SEVERE ACCIDENT
CONTAINMENT REQUIREMENTS

--R. M. BERNERO

DECEMBER 9, 1986

FOIA-87-10
K/21

NRC SEVERE ACCIDENT POLICY

- AUGUST 8, 1985
- PRESENT REACTORS ARE SAFE ENOUGH, BUT...
- SEARCH FOR OUTLIERS
- CONSIDER BALANCE OF PREVENTION AND MITIGATION
  - SPECIAL CONSIDERATION OF CONTAINMENT PERFORMANCE

THE SEARCH FOR OUTLIERS

- SEARCH FOR SIGNIFICANT VULNERABILITY
  - FIND OUTLIERS NOT NECESSARILY QUANTIFY INLIERS

- INDIVIDUAL PLANT EXAMINATION
  - UNLESS ALREADY DONE
  - IDENTIFY OUTLIERS
  - BACKFIT AS APPROPRIATE

- WHERE TECHNICAL ISSUE GOES BEYOND CURRENT REGULATORY REQUIREMENTS
  - GENERIC RULEMAKING PREFERRED
  - ALSO USE BULLETINS, ORDERS OR GENERIC LETTERS

GDC 16:

CRITERION 16 - CONTAINMENT DESIGN. "--AN ESSENTIALLY LEAK-TIGHT BARRIER AGAINST THE UNCONTROLLED RELEASE OF RADIOACTIVITY TO THE ENVIRONMENT AND TO ASSURE THAT THE CONTAINMENT DESIGN CONDITIONS IMPORTANT TO SAFETY ARE NOT EXCEEDED FOR AS LONG AS POSTULATED ACCIDENT CONDITIONS REQUIRE."

GDC 50:

CRITERION 50 - CONTAINMENT DESIGN BASIS. "--AS REQUIRED BY SECTION 50.44, ENERGY FROM METAL-WATER AND OTHER CHEMICAL REACTIONS THAT MAY RESULT FROM DEGRADATION BUT NOT TOTAL FAILURE OF EMERGENCY CORE COOLING FUNCTIONING, (2) THE LIMITED EXPERIENCE AND EXPERIMENTAL DATA AVAILABLE FOR DEFINING ACCIDENT PHENOMENA AND CONTAINMENT RESPONSES, AND (3) THE CONSERVATISM OF THE CALCULATIONAL MODEL AND INPUT PARAMETERS."

IDCOR/NRC PROCESS

- TWO PARALLEL PROGRAMS TO STUDY SEVERE ACCIDENTS IN REFERENCE PLANTS
  - NRC SEVERE ACCIDENT PROGRAM
  - IDCOR
- COMPARE AND RESOLVE TECHNICAL ISSUES
- IDCOR PREPARE AND SUBMIT IPE METHODOLOGY FOR NRC REVIEW
- NRC GENERIC LETTER TO DO IPE
  - WITH GUIDELINES & CRITERIA
  - BY APPROVED METHODOLOGY
- CONDUCT IPE
- IDENTIFY AND EVALUATE OUTLEIRS
- ORDER FIXES

U. S. BOILING WATER REACTORS

- 24 BWR 2/3/4 WITH MARK I CONTAINMENT (ALL LICENSED)

- 9 BWR 4/5 WITH MARK II CONTAINMENT (8 LICENSED)

- 4 BWR 6 WITH MARK III CONTAINMENT (4 LICENSED)

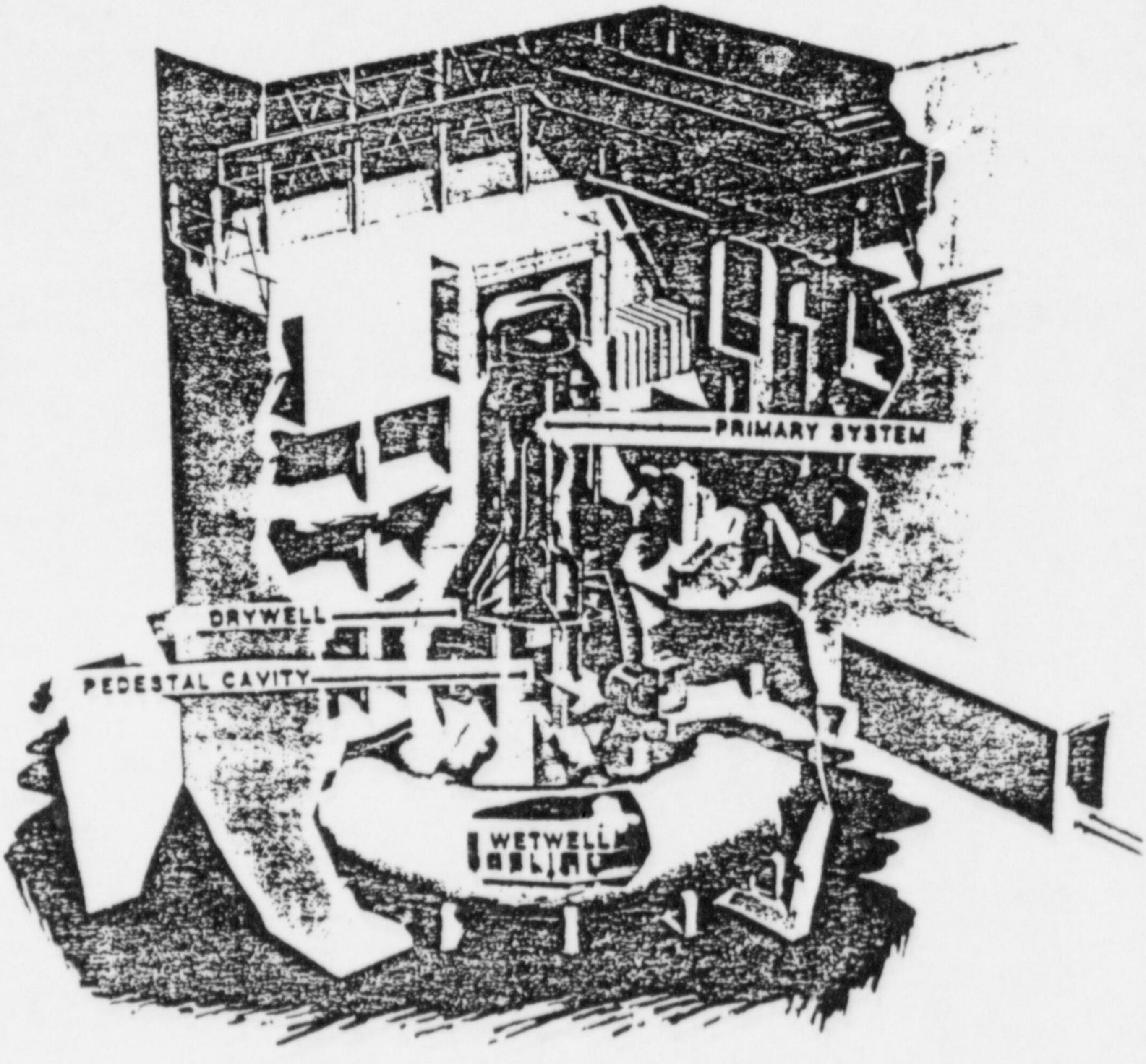
TABLE 1 - U.S. BWR PLANT-SPECIFIC PRA STUDIES

PLANT	PROGRAM NAME	REPORT	REPORT YEAR	CORE/ CONTAINMENT	REACTOR POWER (MWT)	CORE-DAMAGE FREQUENCY PRA ESTIMATE	EVENTS CONSIDERED	MEDIAN, MEAN OR POINT ESTIMATE	CONTAINMENT CONDITIONAL FAILURE PROBABILITY
Peach Bottom	RSS	WASH-1400	1975	BWR-4/MK I	3293	3×10^{-5}	Internal/ External	Median	Not evaluated
Peach Bottom	IDCOR	Tech Summary Task 21	1984	BWR-4/MK I	3293	4×10^{-5}	Internal	Mean	0.2
Peach Bottom	IPE	IPE	1986	BWR-4/MK I	3293	2×10^{-5}	Internal	Mean	Not evaluated
Millstone	IREP	NUREG/CR 3085	1983	BWR-3/MK I	1727	3×10^{-4}	Internal	Median	Not evaluated
Millstone	NUSCO	Millstone 1 PSS	1986	BWR-3/MK I	1727	5×10^{-4}	Internal	Mean	Not evaluated
Browns Ferry	IREP	NUREG/CR 2801	1982	BWR-4/MK I	3293	2×10^{-4}	Internal	Point Estimate	Not evaluated
Vermont Yankee	VYCSS	VYCSS	1986	BWR-4/MK I	1593	3×10^{-5}	Internal	Mean	0.07
Big Rock Point	Consumers	Big Rock Point PRA	1981	BWR-1/Dry	158	1×10^{-3}	Internal/ External	Mean	0.25
Big Rock Point	EG&G/BNL	EG&G-EA-5533 Rev. 1	1982	BWR-1/Dry	158	1×10^{-3}	Internal/ External	Mean	0.25
Limerick	PEPCO	Limerick PRA	1981	BWR-4/MK II	3293	7×10^{-5}	Internal/ External	Mean	1.0
Limerick	BNL	NUREG/CR-3028	1983	BWR-4/MK II	3293	1×10^{-4}	Internal/ External	Mean	1.0
Shoreham	LILCO	Shoreham PRA	1983	BWR-4/MK II	2436	5×10^{-5}	Internal	Point Estimate	Not evaluated
Shoreham	BNL	NUREG/CR-4050	1985	BWR-4/MK II	2436	1×10^{-4}	Internal	Point Estimate	Not evaluated
Shoreham	IPE	Shoreham IPE	1986	BWR-4/MK II	2436	8×10^{-5}	Internal	Mean	Not evaluated
Susquehanna	IPE	IPE	1986	BWR-4/MK II	3293	2×10^{-7}	Internal	Mean	Not evaluated
Grand Gulf	RSSMAP	NUREG/CR-1659	1981	BWR-6/MK III	3833	4×10^{-5}	Internal	Median	Not evaluated
Grand Gulf	IDCOR	Tech Summary Task 21	1984	BWR-6/MK III	3833	8×10^{-6}	Internal	Mean	Not evaluated
GESSAR	GE	GESSAR II PRA		BWR-6/MK III	3579	4×10^{-6}	Internal/ External	Mean	Not evaluated

KEY RESULTS FOR BWR CONTAINMENTS

- REACTOR SAFETY STUDY - PEACH BOTTOM 90% EARLY RELEASE
- IDCOR - PEACH BOTTOM 20% EARLY RELEASE
- VERMONT YANKEE - 7% EARLY RELEASE
- NUREG-1150

DRAFT



TYPICAL MARK I CONTAINMENT DESIGN

FIGURE 1

TABLE 2

BOILING WATER REACTORS
WITH MARK I CONTAINMENT CONFIGURATION

PLANT NAME	PROD LINE	LIC PWR (MWT)	ECCS CONFIGURATION	LPCI RATED FLOW (GPM)	CONT DES PRESSURE (PSIG)	DRYWELL VOLUME (CU FT)	WETWELL AIR VOL (CU FT)	WETWELL WATER (CU FT)
Browns Ferry 1	BWR 4	3293	RCIC/HPCI/LPCI/LPCS	10000	56.0	159,000	129,300	123,000
Browns Ferry 2	BWR 4	3293	RCIC/HPCI/LPCI/LPCS	10000	56.0	159,000	129,300	123,000
Browns Ferry 3	BWR 4	3293	RCIC/HPCI/LPCI/LPCS	10000	56.0	159,000	129,300	123,000
Brunswick 1	BWR 4	2436	RCIC/HPCI/LPCI/LPCS	5775	62.0	164,100	124,000	87,600
Brunswick 2	BWR 4	2436	RCIC/HPCI/LPCI/LPCS	5775	62.0	164,100	124,000	87,600
Cooper Station	BWR 4	2381	RCIC/HPCI/LPCI/LPCS	7700	56.0	132,465	106,850	87,660
Dresden 2	BWR 3	2527	IC/HPCI/LPCI/LPCS	3625	62.0	158,236	116,645	112,203
Dresden 3	BWR 3	2527	IC/HPCI/LPCI/LPCS	3625	62.0	158,236	116,645	112,203
Duane Arnold	BWR 4	1658	RCIC/HPCI/LPCI/LPCS	3600	56.0	118,000	94,270	58,900
Fermi 2	BWR 4	3292	RCIC/HPCI/LPCI/LPCS	7500	56.0	163,730	130,900	117,450
FitzPatrick	BWR 4	2436	RCIC/HPCI/LPCI/LPCS	4375	56.0	154,500	113,000	105,600
Hatch 1	BWR 4	2436	RCIC/HPCI/LPCI/LPCS	5775	56.0	146,010	112,900	87,300
Hatch 2	BWR 4	2436	RCIC/HPCI/LPCI/LPCS	5775	56.0	146,266	109,800	87,300
Hope Creek	BWR 4	3293	RCIC/HPCI/LPCI/LPCS	10000	62.0	169,000	133,500	118,000
Millstone 1	BWR 3	2011	IC/FWCI/LPCI/LPCS	7500	62.0	146,900	109,900	98,000
Monticello	BWR 3	1670	RCIC/HPCI/LPCI/LPCS	5000	56.0	134,200	103,510	68,000
Nine Mile Point 1	BWR 2	1850	IC/LPCS	N/A	62.0	180,000	120,000	89,000
Oyster Creek 1	BWR 2	1930	IC/LPCS	N/A	62.0	180,000	127,000	83,400
Peach Bottom 2	BWR 4	3293	RCIC/HPCI/LPCI/LPCS	10900	56.0	175,800	127,700	122,900
Peach Bottom 3	BWR 4	3293	RCIC/HPCI/LPCI/LPCS	10900	56.0	175,800	127,700	122,900
Pilgrim 1	BWR 3	1998	RCIC/HPCI/LPCI/LPCS	5600	56.0	147,000	120,000	84,000
Quad Cities 1	BWR 3	2511	RCIC/HPCI/LPCI/LPCS	3625	62.0	158,236	116,645	112,203
Quad Cities 2	BWR 3	2511	RCIC/HPCI/LPCI/LPCS	3625	62.0	158,236	116,645	112,203
Vermont Yankee	BWR 4	1593	RCIC/HPCI/LPCI/LPCS	7200	56.0	134,000	112,500	68,000

BWR CONTAINMENT ISSUES - MARK I

- SMALL VOLUME
  - MORE RAPID OVERPRESSURE
  - ESPECIALLY VULNERABLE TO HYDROGEN BURN

- SMALL DRYWELL FLOOR
  - LOWER HEAD AREA CLOSE TO DRYWELL WALL
  - POTENTIAL FOR DIRECT DEBRIS ATTACK
  - DIRECT RADIATION AND CONVECTION HEATING

- LIMITED PASSIVE CAPABILITY BUT OPTIONS FOR ACTIVE RESPONSE

- 5-ELEMENT APPROACH
  - HYDROGEN CONTROL
  - SPRAY IN DRYWELL
  - PRESSURE RELIEF
  - DEBRIS CONTROL
  - PROCEDURES AND TRAINING

CONTAINMENT IMPROVEMENT STRATEGY

- PREVENT HYDROGEN COMBUSTION BY INERTING

- REDUCE DRYWELL SPRAY FLOW RATE
  - PERMITS ALTERNATE SUPPLIES TO PRODUCE SPRAY
  - EXTENDS WATER SUPPLIES

- PROVIDE RELIABLE BACKUP SUPPLIES FOR DRYWELL SPRAY
  - PROVIDES SHALLOW POOL OF WATER ON DRYWELL FLOOR
  - DIRECT SPRAY COOLING OF ANY CORE DEBRIS LEAVING LOWER HEAD AREA
  - SPRAY SCRUBBING OF DRYWELL VOLUME
  - DIRECT COOLING OF WALLS

- WETWELL PRESSURE RELIEF TO STACK
  - POOL SCRUBBING
  - ELEVATED RELEASE

- DEBRIS CONFINEMENT

- TRAINED OPERATORS

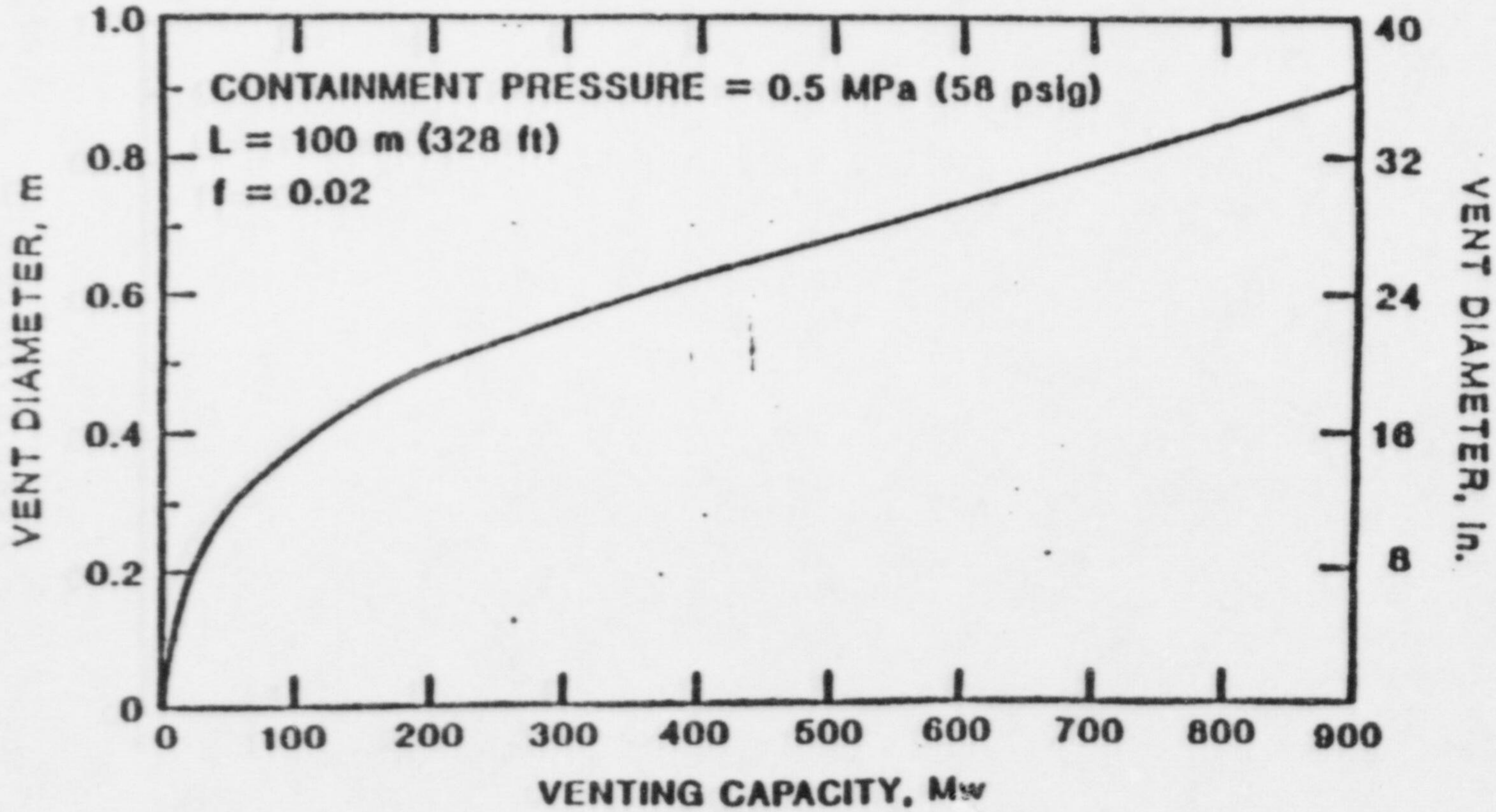


Figure 2. **Vent Size Requirement as a Function of Power**

(This figure is reproduced from IDCOR Report, "Evaluation of BWR Accident Mitigation Capability Relative to proposed NRC Changes," August 1986.)

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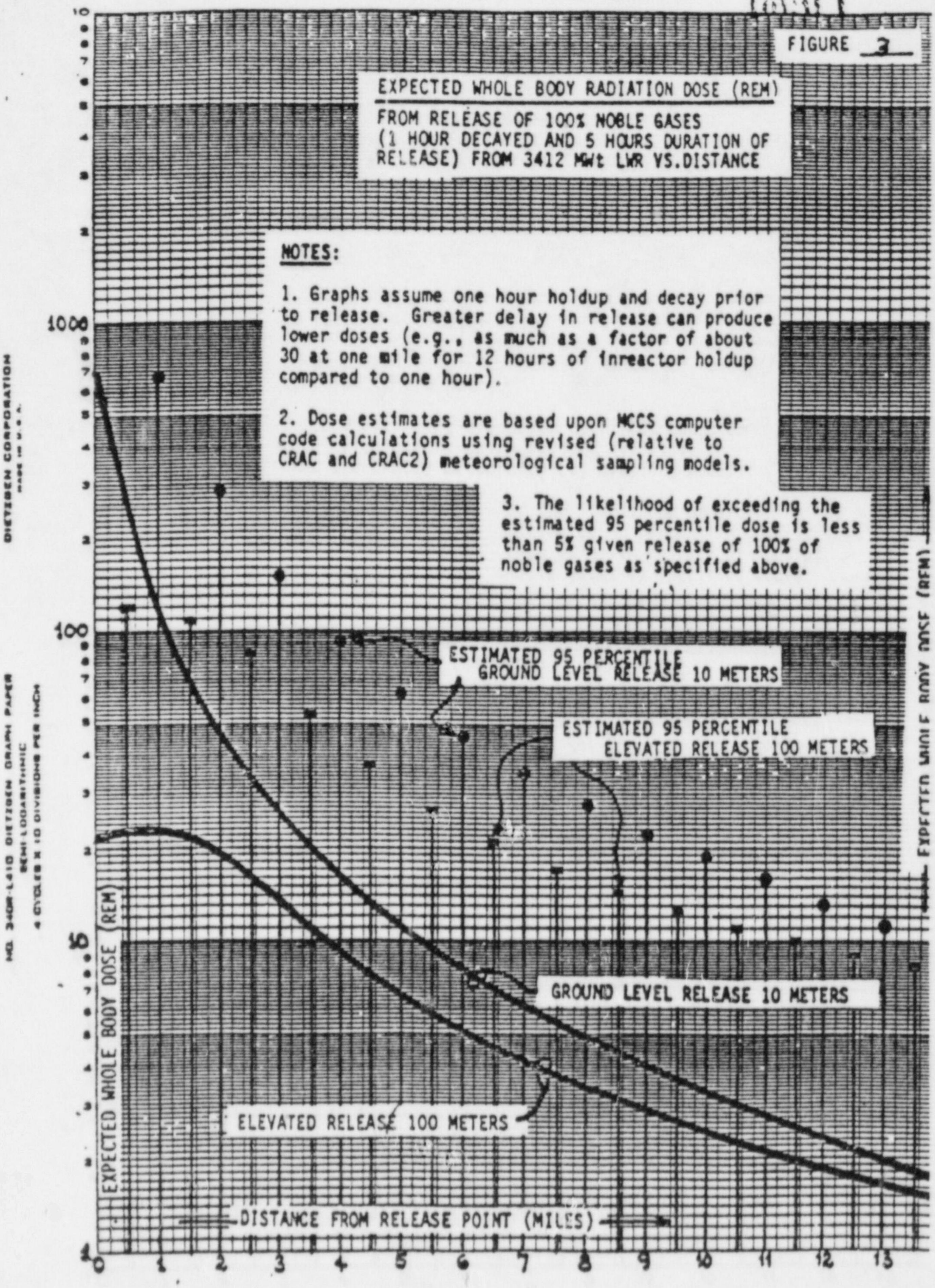


FIGURE 3
EXPECTED WHOLE BODY RADIATION DOSE (REM)
FROM RELEASE OF 100% NOBLE GASES
(1 HOUR DECAYED AND 5 HOURS DURATION OF
RELEASE) FROM 3412 MWt LWR VS. DISTANCE
NOTES:
1. Graphs assume one hour holdup and decay prior to release. Greater delay in release can produce lower doses (e.g., as much as a factor of about 30 at one mile for 12 hours of inreactor holdup compared to one hour).
2. Dose estimates are based upon MCCS computer code calculations using revised (relative to CRAC and CRAC2) meteorological sampling models.
3. The likelihood of exceeding the estimated 95 percentile dose is less than 5% given release of 100% of noble gases as specified above.
ESTIMATED 95 PERCENTILE
GROUND LEVEL RELEASE 10 METERS
ESTIMATED 95 PERCENTILE
ELEVATED RELEASE 100 METERS
GROUND LEVEL RELEASE 10 METERS
ELEVATED RELEASE 100 METERS
EXPECTED WHOLE BODY DOSE (REM)
EXPECTED WHOLE BODY DOSE (REM)
DISTANCE FROM RELEASE POINT (MILES)
1000
100
10
0
1
2
3
4
5
6
7
8
9
10
11
12
13
DIETZGEN CORPORATION
MADE IN U.S.A.
NO. 340R-L410 DIETZGEN GRAPH PAPER
SEMI-LOGARITHMIC
4 CYCLES X 10 DIVISIONS PER INCH

PROPOSED REQUIREMENTS

1. HYDROGEN CONTROL

   PRESENT REQUIREMENTS IMPOSED BY 10 CFR PART 50.44 AND THE TECHNICAL SPECIFICATIONS SHALL BE ADHERED TO, NO ADDITIONAL REQUIREMENTS ARE PROPOSED.

2. CONTAINMENT SPRAY

   ALL BWRs WITH MARK I CONTAINMENT SHALL PROVIDE AT LEAST TWO BACKUP WATER SUPPLY SYSTEMS FOR THE CONTAINMENT DRYWELL SPRAY, ONE OF WHICH SHALL BE FUNCTIONAL DURING STATION BLACKOUT. WATER TO THE SPRAY SYSTEM FROM THESE BACKUP SUPPLIES SHALL BE AVAILABLE BY REMOTE MANUAL OPERATION OR BY SIMPLE PROCEDURES FOR CONNECTION AND STARTUP WHICH CAN BE IMPLEMENTED DURING A SEVERE ACCIDENT SCENARIO.

   IN ADDITION, THE SPRAY NOZZLES SHALL BE ADJUSTED SO THAT AN EVENLY DISTRIBUTED SPRAY PATTERN WILL BE DEVELOPED IN THE DRYWELL WHETHER WATER IS SUPPLIED BY THE PRIMARY SOURCE OR EITHER OF THE BACKUP SOURCES. A FLOW RATE ON THE ORDER OF 1/10 OF THE PRESENT FLOW RATE IS CONSIDERED TYPICAL, THE LICENSEE SHALL SELECT THE FLOW BASED ON AN ANALYSIS OF PLANT SPECIFIC PARAMETERS.

PROPOSED REQUIREMENTS (CONT'D.)

3. PRESSURE RELIEF

THE LICENSEE SHALL SELECT A PRESSURE BETWEEN DESIGN PRESSURE AND 1½ TIMES DESIGN PRESSURE AT WHICH TO OPEN AN EXHAUST PATH FROM THE WETWELL VAPOR SPACE TO THE HIGHEST VENT POINT (STACK OR PIPE) AVAILABLE. THIS LINE SHOULD BE CAPABLE OF HANDLING WATER VAPOR FLOW EQUIVALENT TO 1% DECAY HEAT AT THE VENT PRESSURE SELECTED WITHOUT SIGNIFICANT CHANCE OF RUPTURE BEFORE THE DESIRED RELEASE POINT. THE LINE SHALL BE EQUIPPED WITH ISOLATION VALVES WHICH CAN BE OPENED AND RECLOSED BY REMOTE MANUAL OPERATION OR BY SIMPLE PROCEDURES WHICH CAN BE IMPLEMENTED DURING SEVERE ACCIDENT SCENARIOS INCLUDING STATION BLACKOUT.

4. CORE DEBRIS CONTROL

THE LICENSEE SHALL ENSURE THAT THE WATER IN THE SUPPRESSION POOL IN THE EVENT OF TORUS FAILURE IS HELD WITHIN THE CONFINES OF THE TORUS ROOM AND THE CORNER ROOMS AND CANNOT FLOW OUT TO OTHER PARTS OF THE PLANT.

5. PROCEDURES AND TRAINING

THE LICENSEE SHALL IMPLEMENT EMERGENCY OPERATING PROCEDURES AND OTHER PROCEDURES BASED ON ALL SIGNIFICANT ELEMENTS APPROPRIATE TO ITS PLANT OF EMERGENCY PROCEDURE GUIDELINES, REVISION 4.

CONDITIONS

QUALITY AND DESIGN STANDARDS

SINCE THESE REQUIREMENTS ARE INTENDED TO BE AN OPTIMIZED USE OF EXISTING EQUIPMENT IT IS EXPECTED THAT ADDED EQUIPMENT, OF ITSELF, NEED NOT MEET THE QUALITY OR DESIGN STANDARDS OF SAFETY RELATED EQUIPMENT. NEVERTHELESS, MODIFICATIONS TO OR NEAR EQUIPMENT OR SYSTEMS WHICH ARE ALREADY SAFETY RELATED SHALL NOT COMPROMISE THE QUALITY OF SUCH EQUIPMENT OR SYSTEMS.

IMPLEMENTATION

THE EQUIPMENT CHANGES REQUIRED HEREIN SHALL BE INSTALLED DURING THE FIRST REFUELING OUTAGE WHICH BEGINS NINE (9) MONTHS AFTER THE EFFECTIVE DATE OF THIS LETTER. THE PROCEDURES AND TRAINING REQUIRED SHALL BE IMPLEMENTED ON A SCHEDULE REVIEWED AND APPROVED BY THE NRC. GIVEN THE IMPLEMENTATION OF THE GENERIC IMPROVEMENTS OF MARK I CONTAINMENTS THERE IS NO NEED FOR AN INDIVIDUAL PLANT EVALUATION (IPE) FOR CONTAINMENT PERFORMANCE. THIS DOES NOT REMOVE THE NEED FOR AN IPE WHICH COVERS THE SYSTEM RELIABILITY OR CORE MELT FREQUENCY PORTION OF THE SEVERE ACCIDENT QUESTION.

SEVERE ACCIDENT POLICY STATEMENT

- OPERATING NUCLEAR POWER PLANTS REQUIRE NO FURTHER REGULATORY ACTION TO DEAL WITH SEVERE ACCIDENT ISSUES UNLESS SIGNIFICANT NEW SAFETY INFORMATION ARISES TO QUESTION WHETHER THERE IS ADEQUATE ASSURANCE OF NO UNDUE RISK TO PUBLIC HEALTH AND SAFETY.

- IN THE LATTER EVENT, A CAREFUL ASSESSMENT SHALL BE MADE OF THE SEVERE ACCIDENT VULNERABILITY POSED BY THE ISSUE AND WHETHER THIS VULNERABILITY IS PLANT OR SITE SPECIFIC OR OF GENERIC IMPORTANCE.

- THE MOST COST-EFFECTIVE OPTIONS FOR REDUCING THIS VULNERABILITY SHALL BE IDENTIFIED AND A DECISION SHALL BE REACHED CONSISTENT WITH THE COST-EFFECTIVENESS CRITERIA OF THE COMMISSION'S BACKFIT POLICY AS TO WHICH OPTION OR SET OF OPTIONS (IF ANY) ARE JUSTIFIABLE AND REQUIRED TO BE IMPLEMENTED.

- IN THOSE INSTANCES WHERE THE TECHNICAL ISSUE GOES BEYOND CURREN REGULATORY REQUIREMENTS, GENERIC RULEMAKING WILL BE THE PREFERR SOLUTION. IN OTHER CASES, THE ISSUE SHOULD BE DISPOSED OF THROUGH THE CONVENTIONAL PRACTICE OF ISSUING BULLETINS AND ORDERS OR GENERIC LETTERS WHERE MODIFICATIONS ARE JUSTIFIED THROUGH BACKFIT POLICY, OR THROUGH PLANT-SPECIFIC DECISION-MAKING ALONG THE LINES OF THE INTEGRATED SAFETY ASSESSMENT PROGRAM (ISAP) CONCEPTION.

COMMISSION RESPONSE TO A HEARING QUESTION

JULY 16, 1986

QUESTION

IS A 90 PERCENT CHANCE OF FAILURE IN THE EVENT OF A CORE MELTDOWN AN ACCEPTABLE FAILURE RATE?

ANSWER

THE NRC HOLDS THE POSITION THAT THE LIKELIHOOD OF CORE MELT ACCIDENTS IN ANY PLANT SHOULD BE VERY LOW AND, IN ADDITION, THAT THERE SHOULD BE SUBSTANTIAL ASSURANCE THAT THE CONTAINMENT WILL MITIGATE THE CONSEQUENCES OF A CORE MELT SHOULD ONE OCCUR IN ORDER TO ENSURE LOW RISK TO THE PUBLIC. IT IS NOT MERELY A QUESTION OF HAVING LOW RISK BUT OF HAVING AS WELL THE DEFENSE-IN-DEPTH ASSURANCE OF COMBINED PROTECTION BY PREVENTION AND MITIGATION...

TABLE 3 COST-BENEFIT ANALYSIS

COST: $0.7-2.2M

BENEFIT:(1)	FCM	CCFP BEFORE	CCFP AFTER	AVERTED LOSS/YR	AVERTED LOSS PRES. VALUE
BASE CALCULATION	$1x10^{-4}$/yr	0.5	0.05	$4x10^{5}/yr	$3M/$12M
LOWER FCM	$1x10^{-5}$/yr	0.5	0.05	$4x10^{4}/yr	$0.3M/$1.2M
LESS CHANGE IN CONTAINMENT	$1x10^{-4}$/yr	0.5	0.1	$4x10^{5}/yr	$3M/$12M
BETTER CONTAINMENT TO START	$1x10^{-4}$	0.2	0.05	$2x10^{5}/yr	$2M/$6M
"OPTIMISTIC" CALCULATION	$1x10^{-5}$	0.2	0.05	$2x10^{4}/yr	$0.2M/$0.6M
"PESSIMISTIC" CALCULATION	$3x10^{-4}$	0.9	0.1	$2x10^{5}/yr	$16M/$60M

(1) FCM = Frequency of Core Melt
CCFP = Conditional Containment Failure Probability
AVERTED LOSS PRESENT VALUE expressed as A/B where A is the averted loss per year times 8 (roughly equivalent to discount at 12%/yr rate) and B is the averted loss per year times 30 (no discount).

PROPOSED ACTION

- DEC. 9 & 12, 1986 ACRS REVIEW
- DEC. 19, 1986 CRGR REVIEW
- JANUARY 1987 REVIEW OF ACRS AND CRGR REACTION WITH COMMISSION
- FEB. 1, 1987, PUBLISH PROPOSED GENERIC LETTER FOR COMMENT
- MAY 1987 ISSUE FINAL GENERIC LETTER

SIMILAR LETTERS ON MARK II AND MARK III TO FOLLOW

CRGR BRIEFING ON NUREG-1150

DEC 15 1986

- APPROACH
- PRELIMINARY RESULTS - ACCIDENT FREQUENCIES
- PRELIMINARY RESULTS - RISK
- RISK UNCERTAINTY ANALYSIS
- CONCERNS
  - PRESENTATION OF RESULTS
  - PREMATURE USE

J. A. MURPHY

FOIA-87-10
K/22

OBJECTIVES

- TO PROVIDE A GREATER UNDERSTANDING OF FREQUENCY, RISKS, AND UNCERTAINTIES DUE TO SEVERE CORE DAMAGE ACCIDENTS AT NUCLEAR POWER PLANTS, BASED ON THE ASSESSMENT OF INTERNALLY INITIATED ACCIDENT SEQUENCES AT FIVE REFERENCE PLANTS THAT HAVE DIFFERENT PLANT AND CONTAINMENT DESIGNS;

- TO ASSESS THE USEFULNESS OF THE METHODS USED AND THE INFORMATION GAINED IN: (1) EVALUATING AND PROVIDING INSIGHTS TO DECISION MAKERS ON VARIOUS PLANT-SPECIFIC AND GENERIC SEVERE-ACCIDENT REGULATORY MATTERS, AND (2) HELPING FOCUS THE LIMITED RESOURCES OF THE NUCLEAR REGULATORY COMMISSION IN A MORE EFFECTIVE MANNER; AND

- TO OBTAIN COMMENTS ON THE ABOVE FOR CONSIDERATION IN PREPARING THE FINAL NUREG-1150.

NOTE: NUREG-1150 PROVIDES DATA AND INSIGHTS REGARDING THE RISKS ASSOCIATED UNCERTAINTIES. IT DOES <u>NOT</u> RECOMMEND REGULATORY ACTIONS

<u>METHODOLOGY</u>

- ABBREVIATED ANALYSIS OF FREQUENCY OF SEVERE ACCIDENTS, EMPLOYING INSIGHTS FROM PREVIOUS STUDIES TO ALTER DEPTH OF ANALYSIS, REVIEWED BY INTERNAL QA, NRC, AND SENIOR CONSULTANT GROUP

- EXTREMELY DETAILED CONTAINMENT EVENT TREES, REVIEWED BY UNIVERSITY OF WISCONSIN

- ANALYSIS OF SEVERE ACCIDENT PHENOMENOLOGY USING STATE-OF-THE-ART TOOLS, REVIEWED BY EXTERNAL PEER REVIEW GROUPS

- CONSEQUENCE ANALYSES EMPLOYING IMPROVED MODELING AND LATEST HEALTH EFFECTS MODELS, SUBJECTED TO EXTENSIVE PEER REVIEW

- RISK ESTIMATION WITH COMPREHENSIVE UNCERTAINTY ANALYSES o DRAFT REPORT IS INCOMPLETE

o RISK AND RISK REDUCTION RESULTS OF SEQUOYAH, ZION, PEACH BOTTOM AND GRAND GULF TO BE ADDED

o DETAILED DISCUSSION OF EXTRAPOLATION TO SIMILAR PLANTS IN PREPARATION

ACCIDENT FREQUENCY UNCERTAINTY ANALYSIS

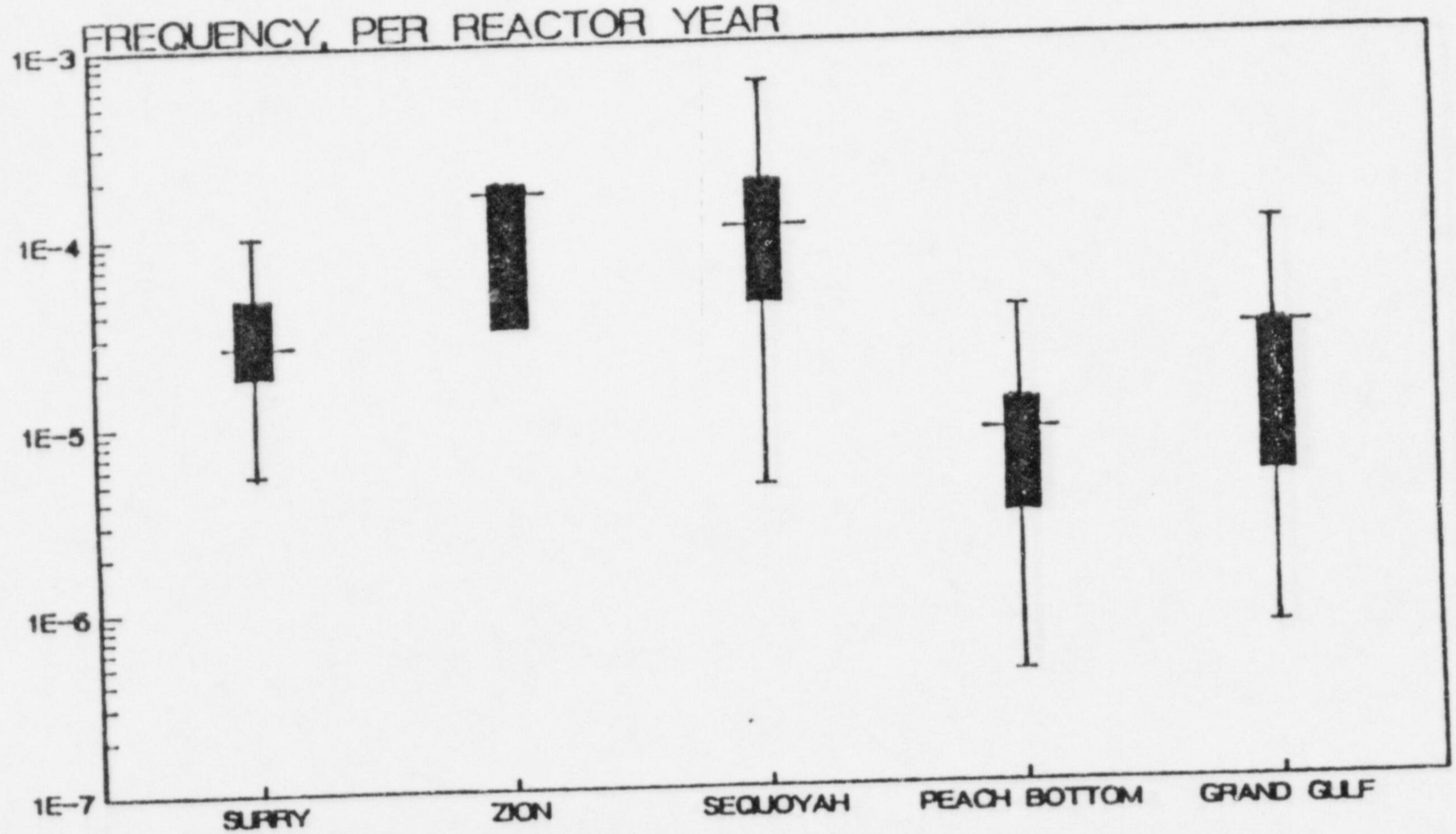
- PROPAGATION OF DATA UNCERTAINTIES THROUGH RELIABILITY MODELS IN STANDARD PRA FASHION
- SENSITIVITY STUDIES TO INVESTIGATE DIFFERENT MODELING ASSUMPTIONS

NUREG—1150 LEVEL OF DETAIL COMPARED TO OTHER STUDIES

ISSUE	WASH—1400	RSSMAP	IREP	RMIEP	IDCOR
Accuracy of Info.	Same	More	Same	Same	More*
Data	More	More	More	Less	More
Initiating Events	More	More	Same	Same	More
Human Factors	Same	More	Same	Less	More
Recovery	More	More	Same	Less	Same
Actuation/ Control	Less	More	Less	Less	More
Electrical Systems	Same	More	Less	Less	More
Other Support Systems	More	More	Less	Less	More
Common Cause	More	More	More	Less	More

* [illegible] IDCOR

COMPARISON OF SEVERE CORE DAMAGE FREQUENCIES OF REFERENCE PLANTS



PRINCIPAL CONTRIBUTORS TO CORE DAMAGE FREQUENCIES FOR REFERENCE PLANTS*

Percent Contribution to Core Damage Frequency	PLANT				
	SURRY	PEACH BOTTOM	SEQUOYAH	GRAND GULF	ZION
Station Blackout	38	86	5	99	2
Loss of Offsite Power	4	-	-	-	1
Loss of Bus	20	-	3	-	-
LOCA	28	1	59	-	18
Interfacing LOCA	4	-	-	-	-
ATWS	6	12	1	1	-
Loss of Component Cooling Water	-	-	-	-	79
Severe Core Damage Frequency (Yr^{-1})	2.6×10^{-5}	8.2×10^{-6}	1.0×10^{-4}	2.7×10^{-5}	1.5×10^{-4}

*Principal contributors to core damage frequency are not necessarily principal contributors to risk.

RISK UNCERTAINTY ANALYSIS

- UNCERTAINTIES NOT ADDRESSED ADEQUATELY IN RSS
- UNCERTAINTIES LARGE SINCE UNDERSTANDING OF SEVERE ACCIDENT PHENOMENA IS NOT COMPLETE
- KNOWLEDGE BASE HAS IMPROVED SUBSTANTIALLY SINCE RSS, BUT STILL MORE TO BE LEARNED

<u>THEREFORE</u>

- SUBJECTIVE JUDGMENT OF EXPERTS, BASED ON EXTANT KNOWLEDGE, REQUIRED TO DEFINE REASONABLE RANGES OF POTENTIALLY IMPORTANT ISSUES
- SUBJECTIVE JUDGMENTS BY EXPERTS NEVER SUBSTITUTE FOR GOOD SCIENCE, <u>EITHER</u> EXPERIMENTAL VALIDATION IS NEEDED <u>OR</u> REGULATORY DECISION MUST BE SUFFICIENTLY CONSERVATIVE TO ENCOMPASS UNCERTAINTIES

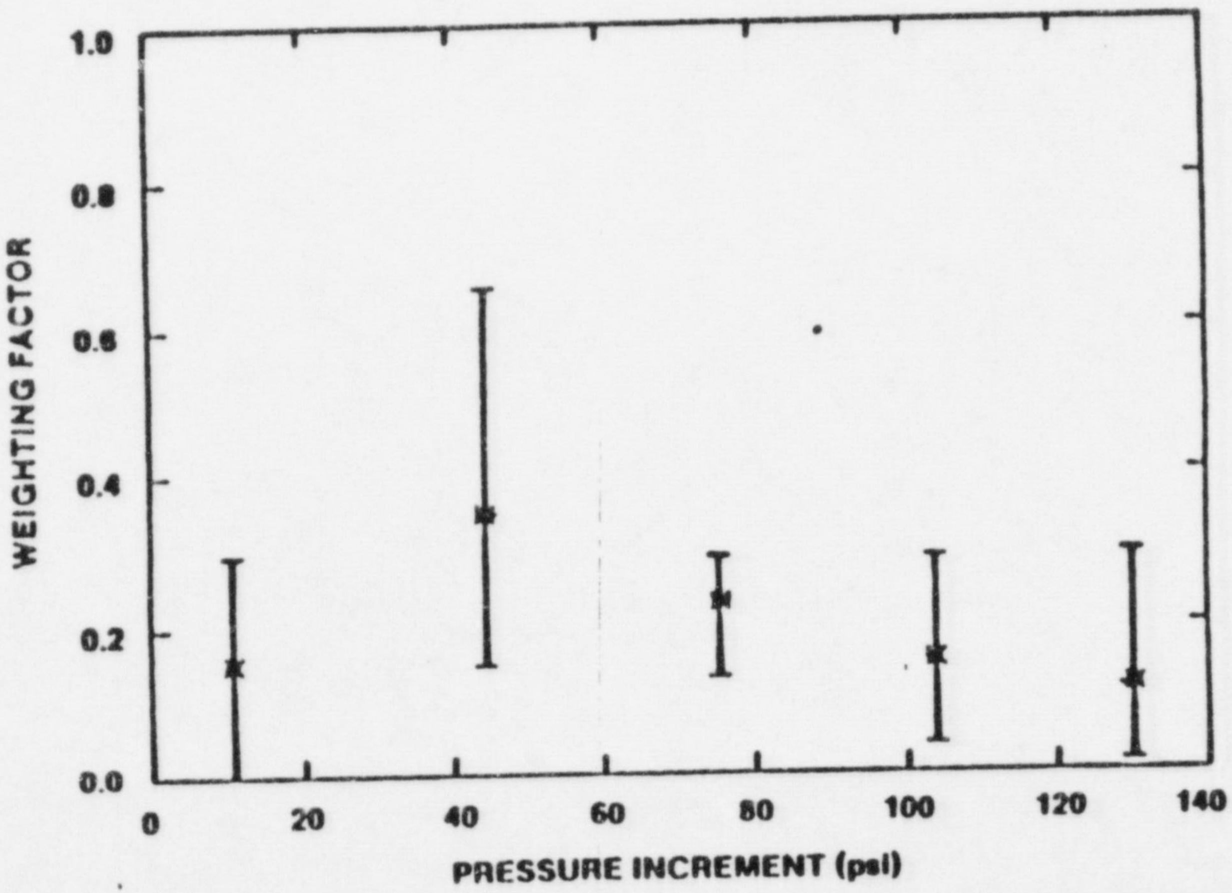
METHOD FOR RISK UNCERTAINTY ANALYSES

- o SELECT POTENTIALLY RISK IMPORTANT FACTORS, AND PARAMETERS DRIVING THEM
  - FREQUENCY
  - CONTAINMENT PERFORMANCE
  - SOURCE TERMS
- o JUDGEMENTALLY SELECT DISCRETE VALUES AND WEIGHTS FOR PARAMETERS, USING AVAILABLE DATA AND ANALYSES
- o DEVELOP REASONABLE RANGE WITHIN WHICH MEAN VALUE OF RISK WOULD LIKELY LIE, USING STATISTICAL SAMPLING TECHNIQUE

EXPERT INPUT FOR LLH ISSUES

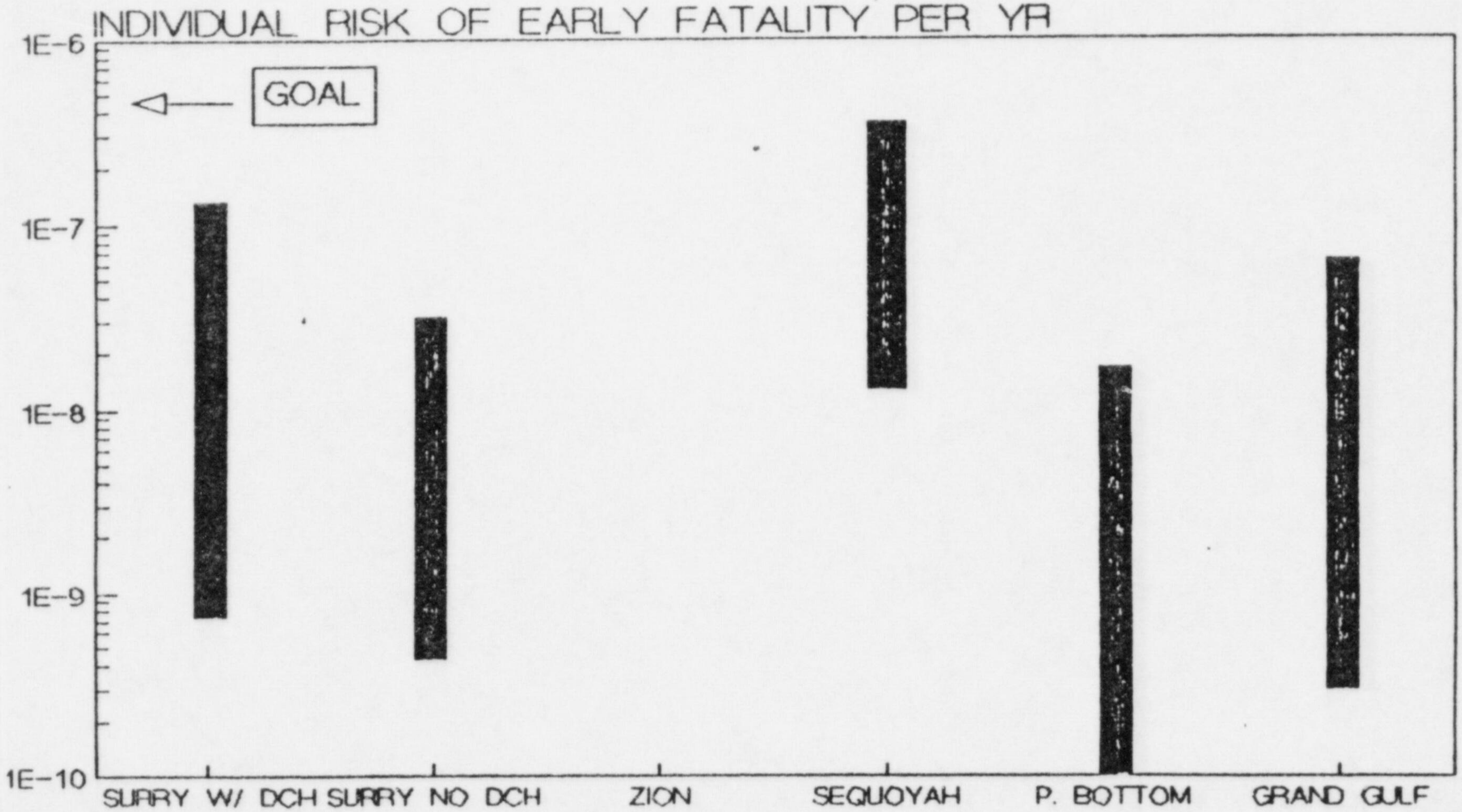
MAGNITUDE OF DIRECT HEATING PRESSURE INCREMENT

PRECONDITIONS:
- HIGH RCS PRESSURE
- DRY CAVITY
- NO CONTAINMENT HEAT REMOVAL

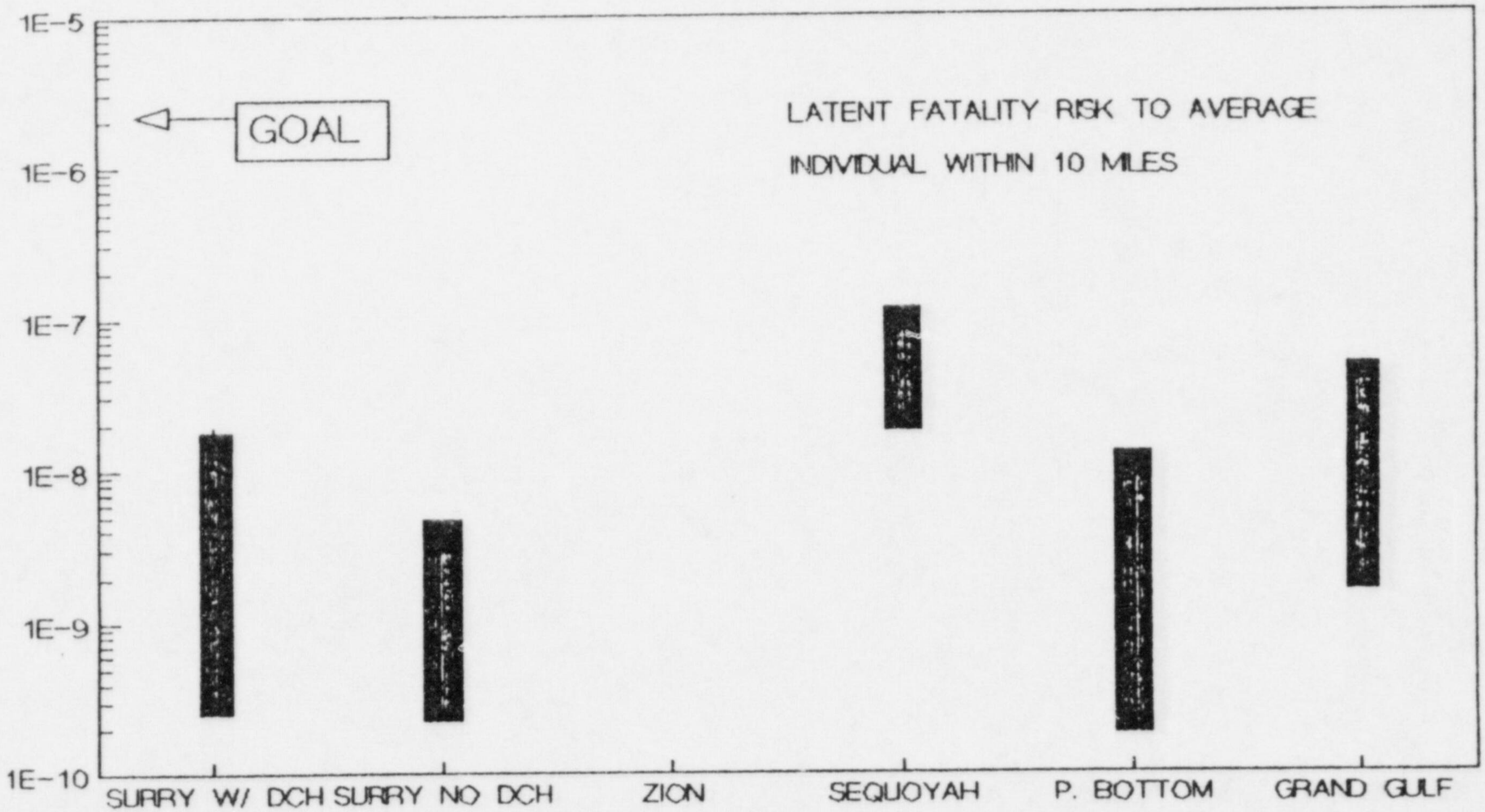


- METHOD FOR DISPLAYING RISK RESULTS UNDER STUDY
- HOW DO WE BEST CONVEY THE INFORMATION OBTAINED WITHOUT MISLEADING THE INITIATIVES?
- HOW SHOULD WE CONSIDER UNCERTAINTY EXPLICITLY IN AN INTEGRAL FASHION?

SUMMARY COMPARISONS WITH SAFETY GOALS



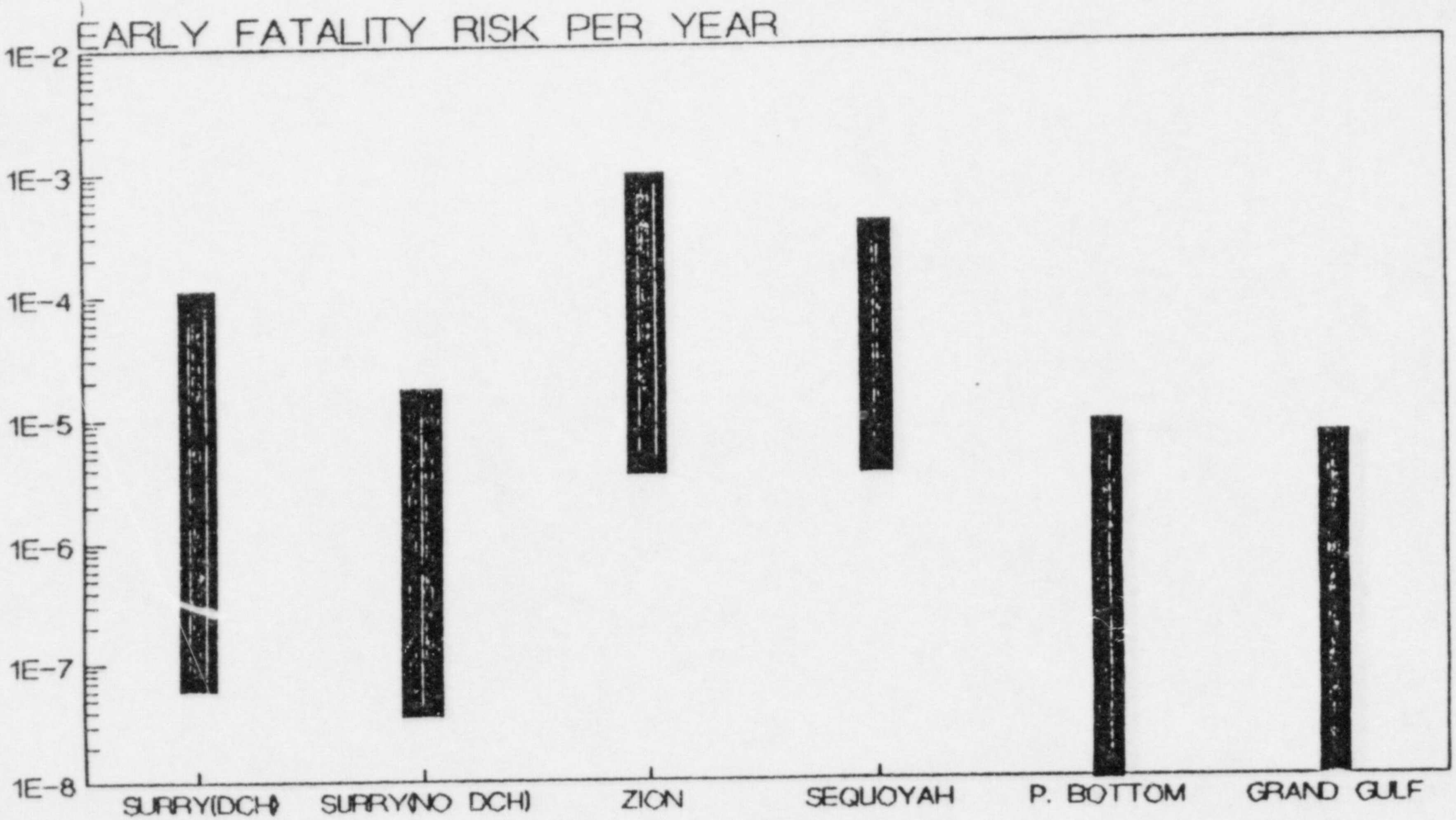
SUMMARY COMPARISON TO SAFETY GOALS

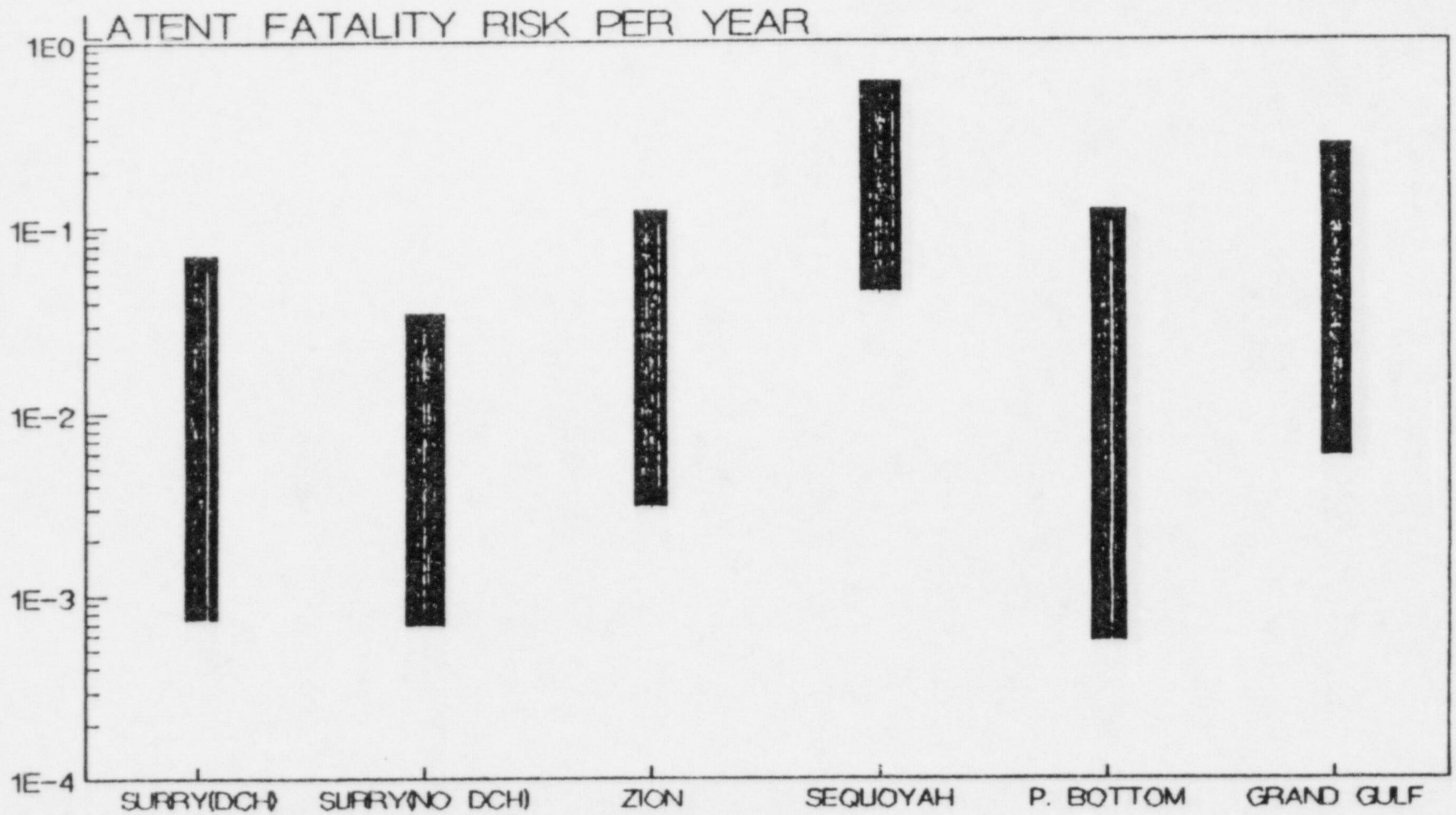


RISK FROM INTERNAL EVENTS ONLY

DCH = DIRECT CONTAINMENT HEATING

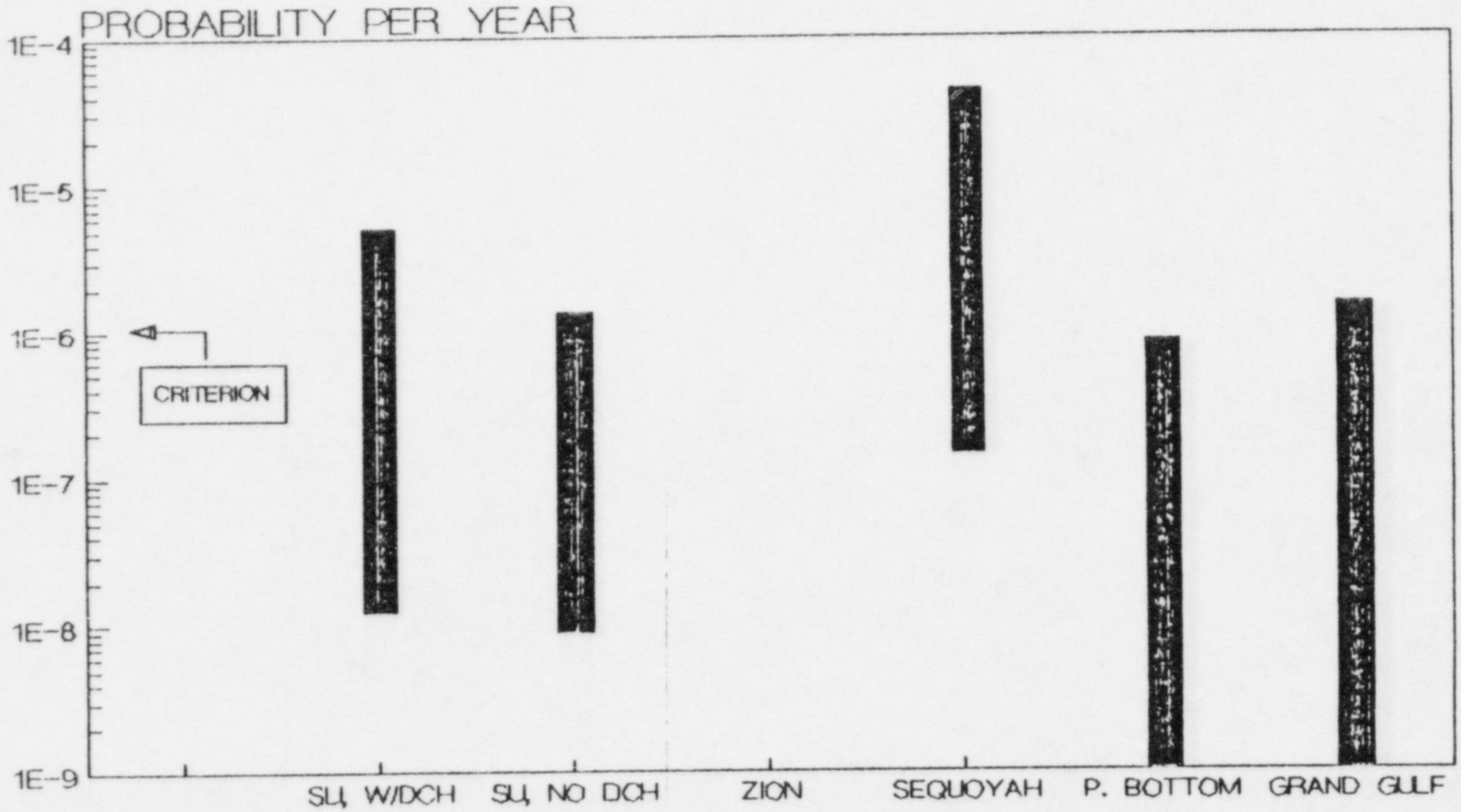
SUMMARY COMPARISONS OF PLANT RISKS





DCH = DIRECT CONTAINMENT HEATING

PROBABILITY OF LARGE RADIOACTIVE RELEASE

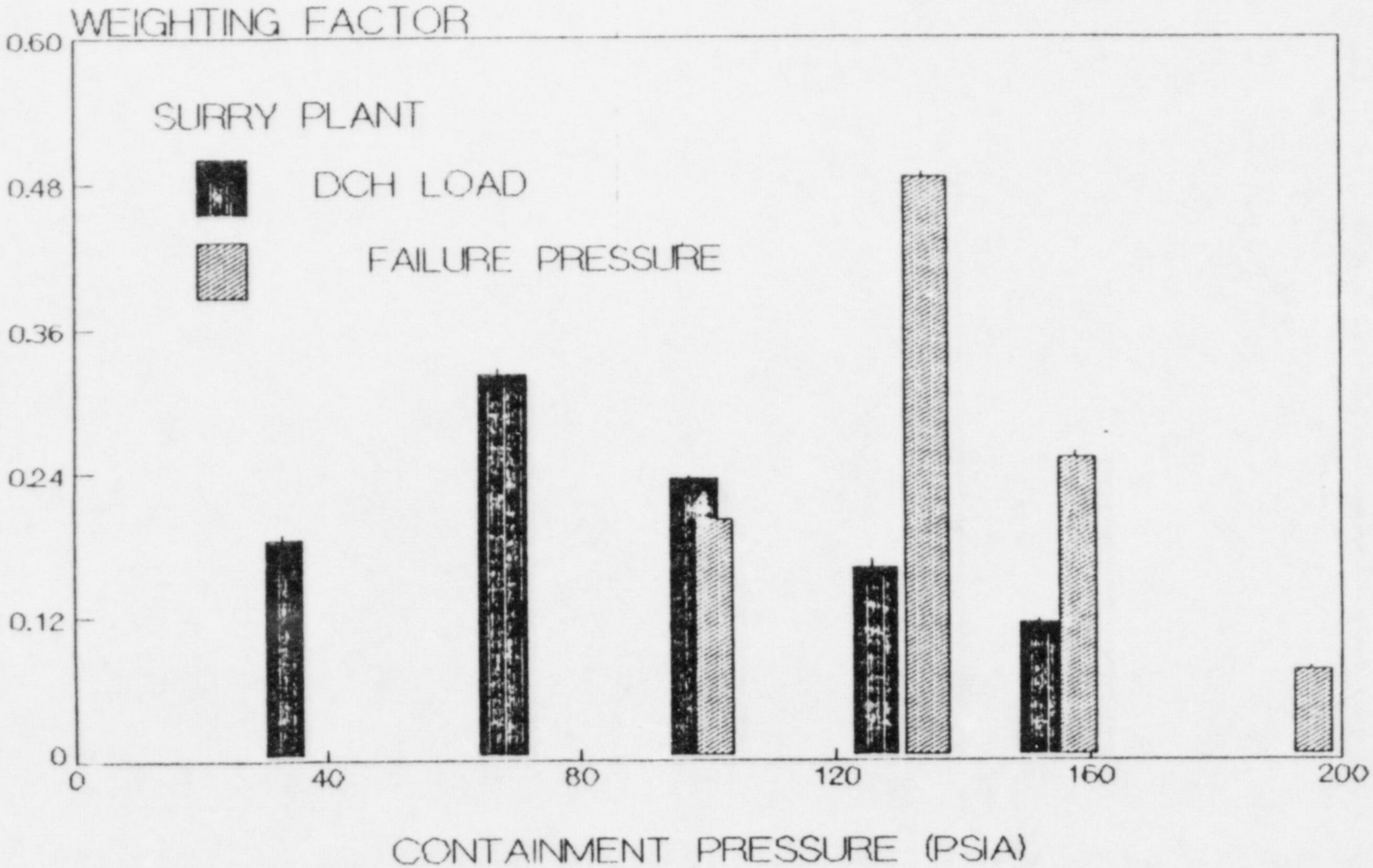


LARGE RELEASE - THAT RESULTING

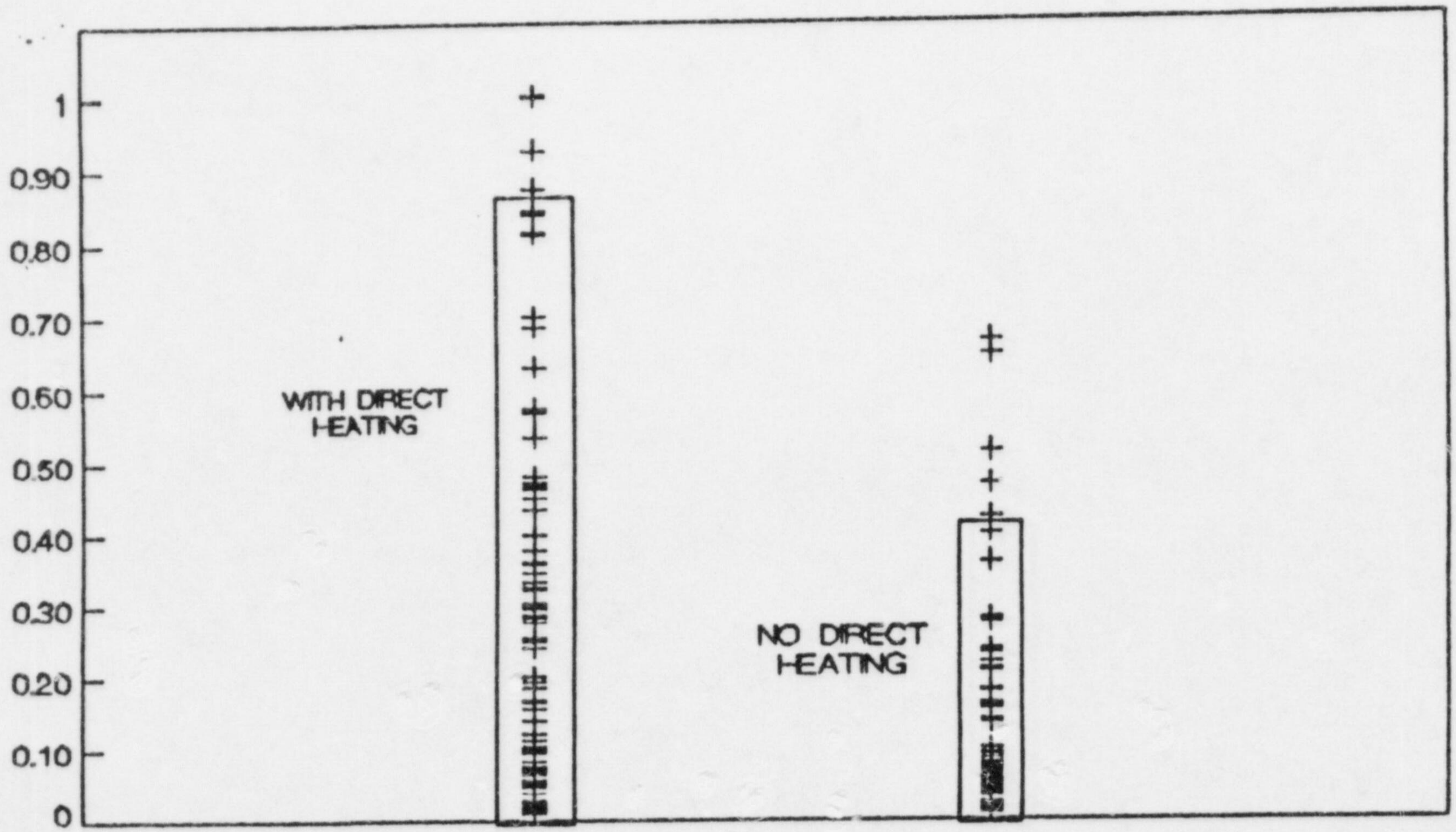
IN 1 OR MORE EARLY FATALITIES

RISK FROM INTERNAL EVENTS ONLY
SU = SURRY
DCH = DIRECT CONTAINMENT HEATING

COMPARISON OF DCH LOADS AND CONTAINMENT FAILURE PRESSURE RANGE

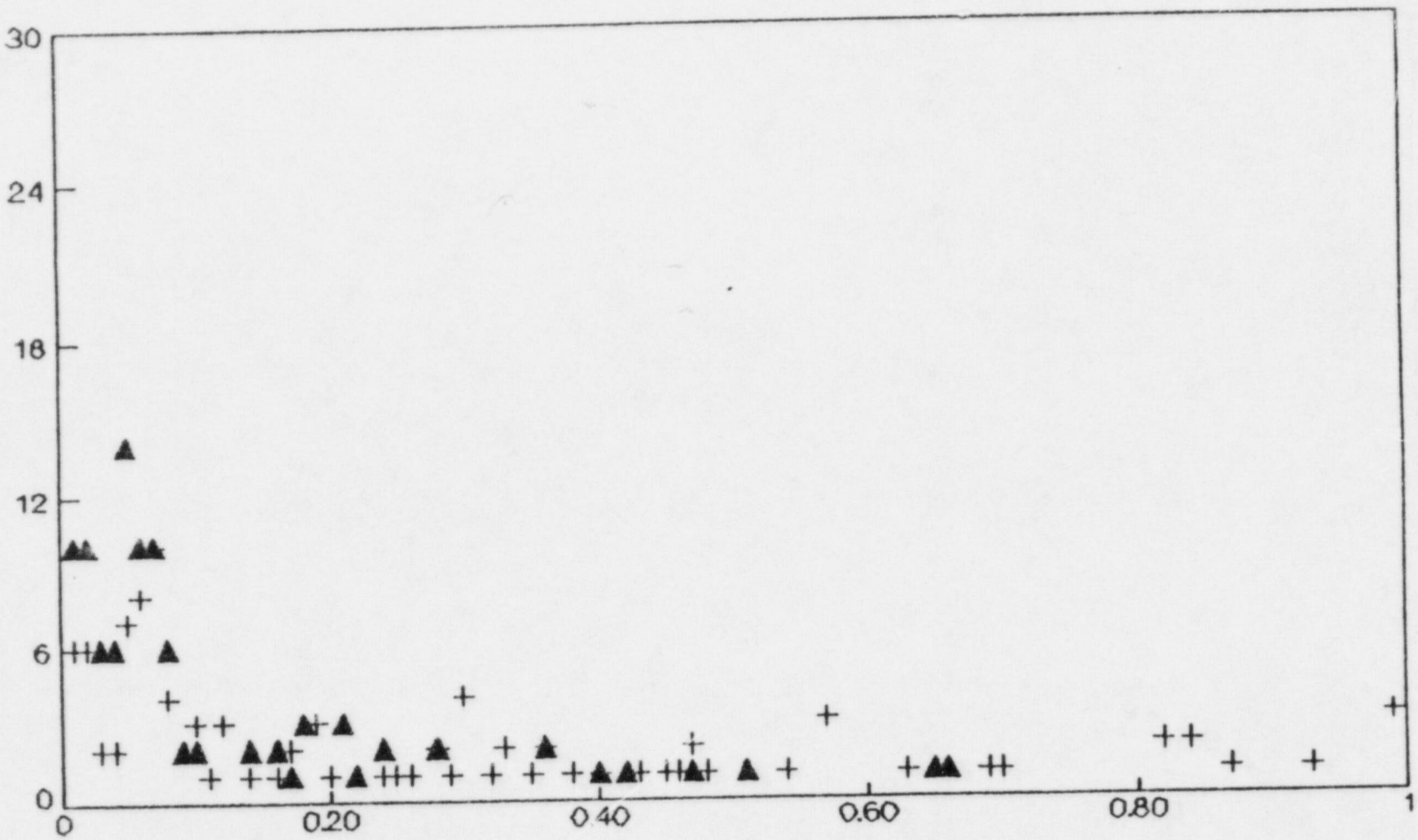


PROBABILITY - EARLY CONTAINMENT FAILURE
SURRY PLANT

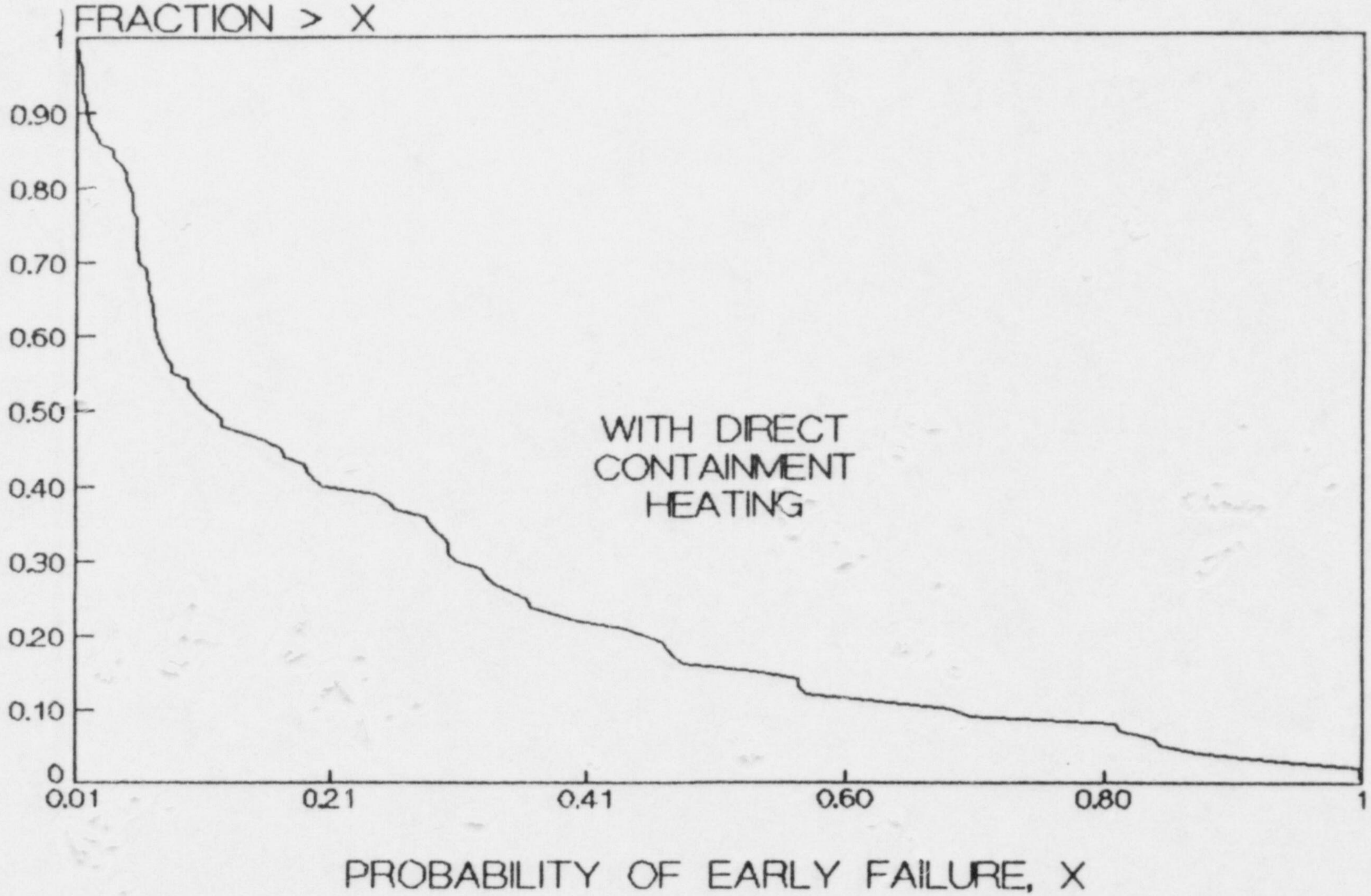


PROBABILITY – EARLY CONTAINMENT FAILURE
SURRY PLANT

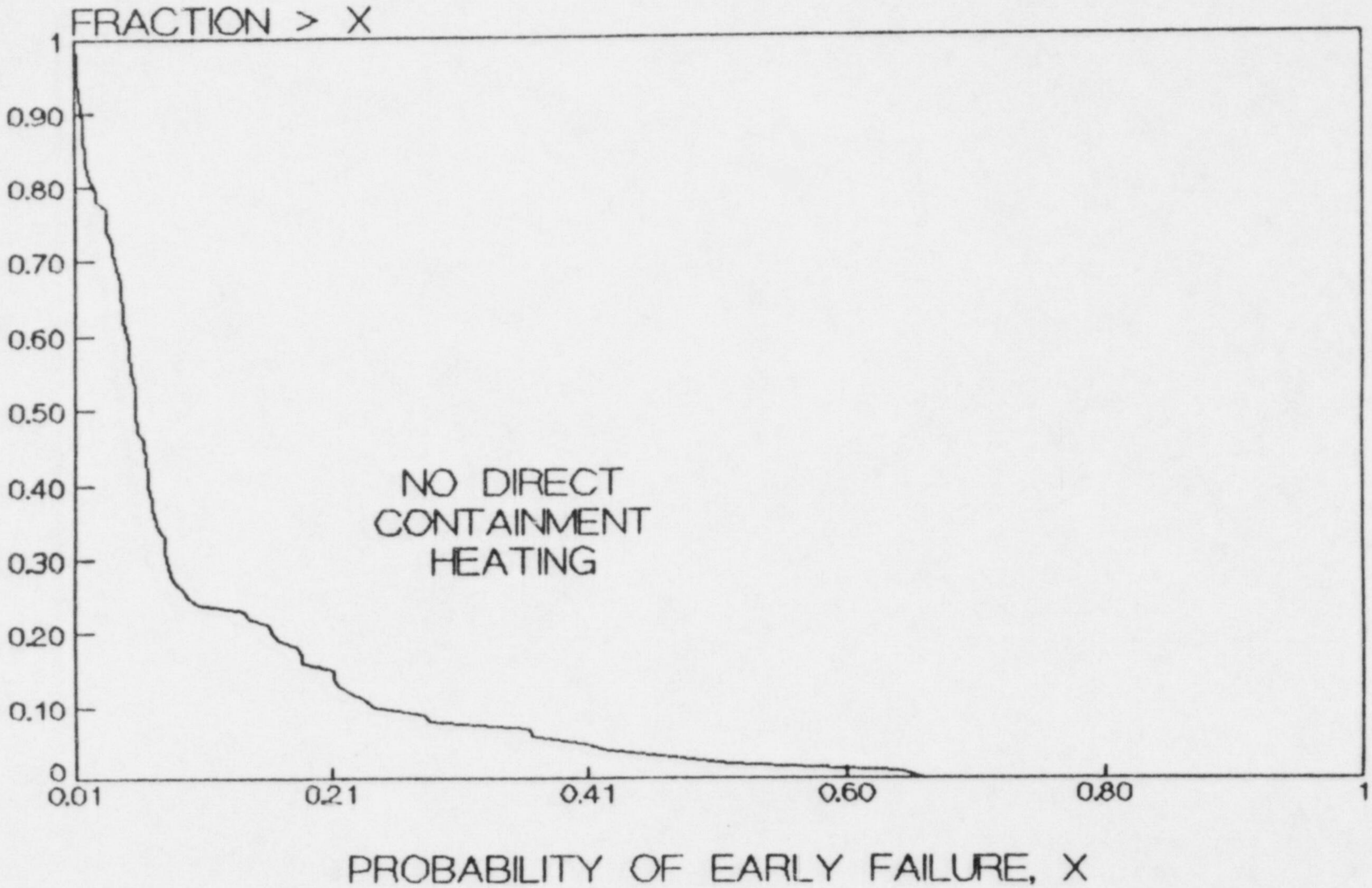
+ WITH DCH ▲ NO DCH



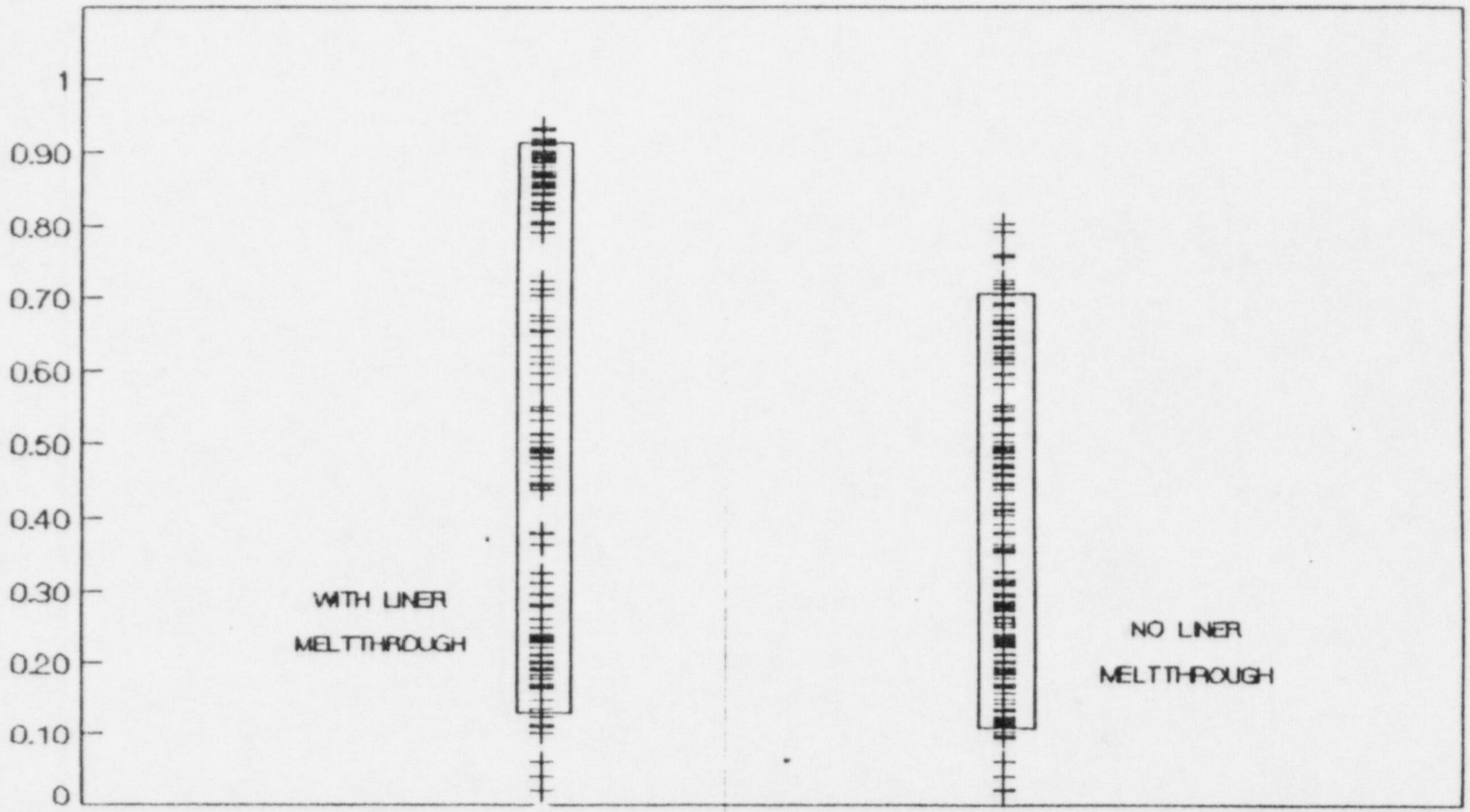
PROBABILITY - EARLY CONTAINMENT FAILURE
SURRY PLANT



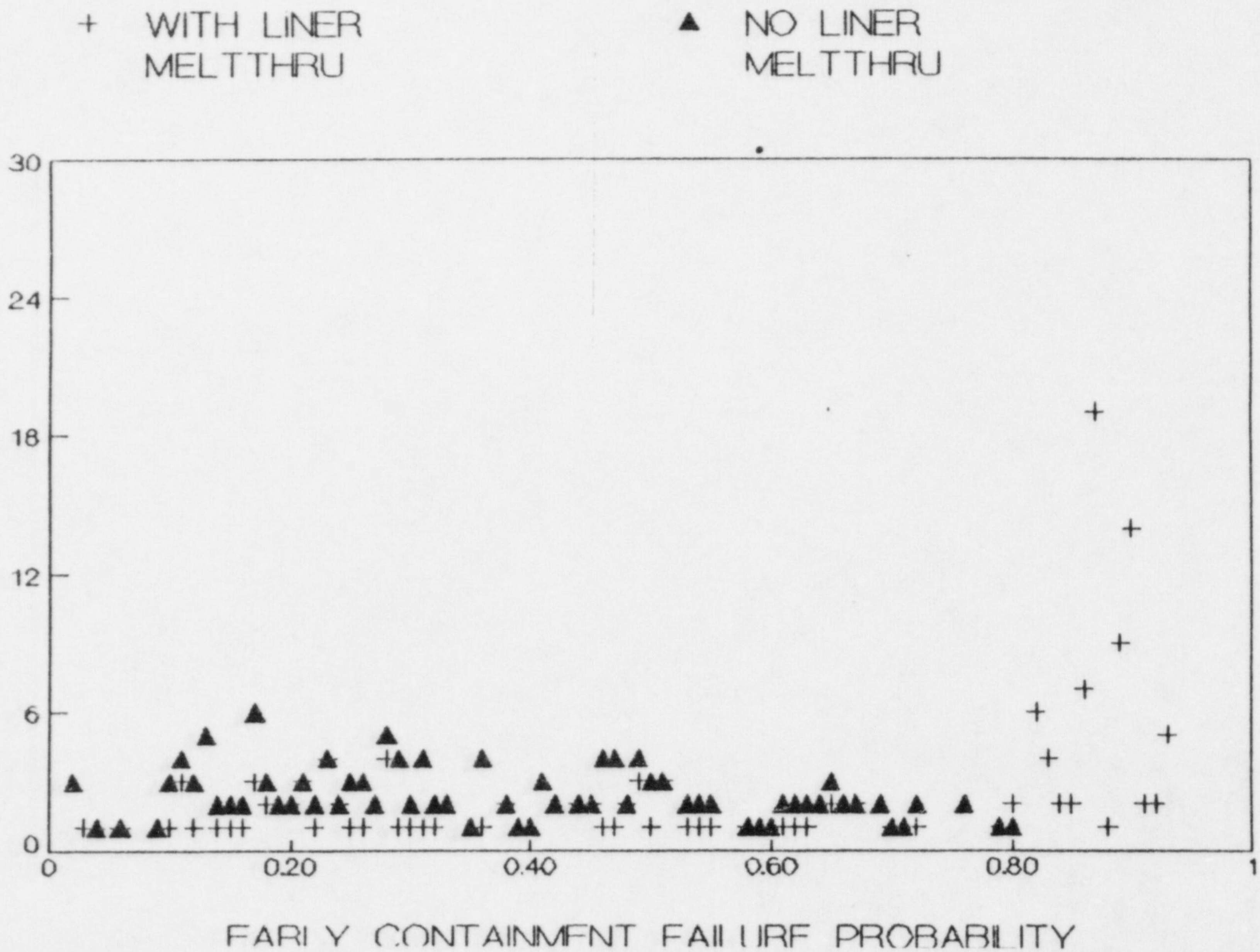
PROBABILITY - EARLY CONTAINMENT FAILURE
SURRY PLANT



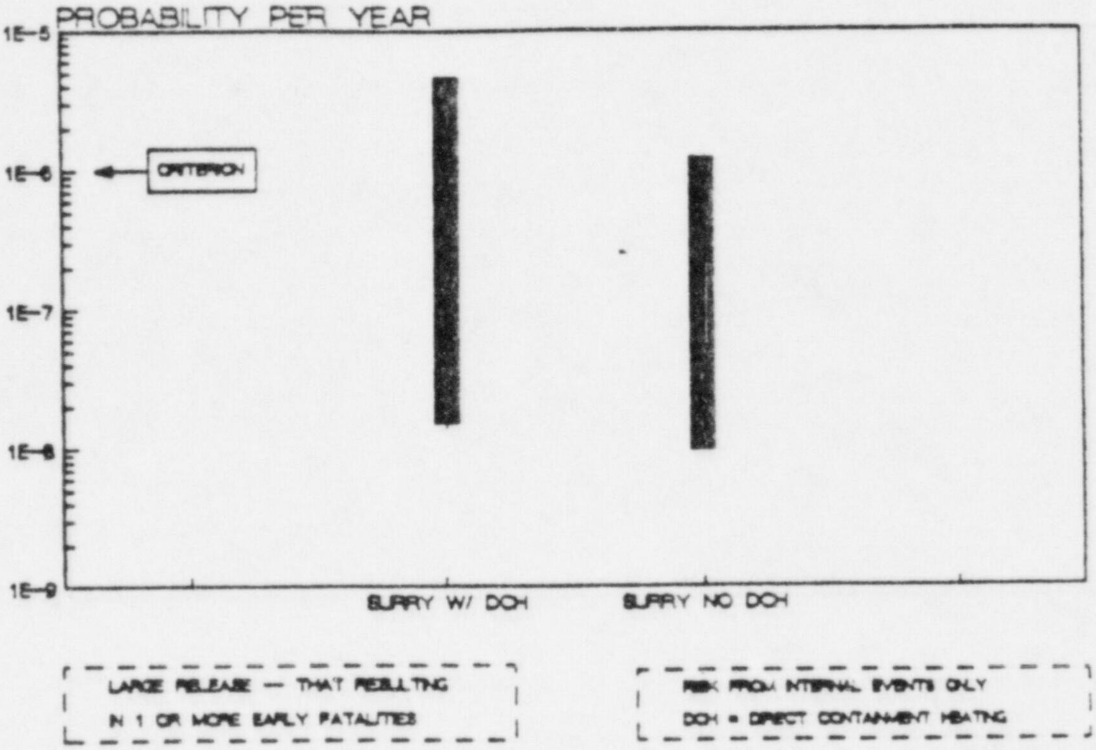
PROBABILITY - EARLY CONTAINMENT FAILURE
PEACH BOTTOM PLANT



PROBABILITY - EARLY CONTAINMENT FAILURE
PEACH BOTTOM PLANT



PROBABILITY OF LARGE RELEASE -- SURRY



SURRY RISK REDUCTION POTENTIAL

- o REDUCTION OF RISK TO ZERO EQUATES TO A BENEFIT (AVERTED COST) OF NOT MORE THAN 2 MILLION DOLLARS
- o GIVEN THIS, NONE OF THE IDENTIFIED RISK-REDUCTION FEATURES IS COST-EFFECTIVE
- o IMPROVED RCS DEPRESSURIZATION CAPABILITY IS MOST COST EFFECTIVE FEATURE WHICH OFFERS SIGNIFICANT RISK REDUCTION
  - COST: 2 MILLION DOLLARS
  - ACTS TO REDUCE RISK UNCERTAINTY BY PRECLUDING DIRECT CONTAINMENT HEATING, STEAM SPIKE
- o CAUTION: RISK AND RISK REDUCTION BENEFIT NOTED HERE ARE BASED ON MID-1985 "SNAPSHOT." FUTURE PERFORMANCE OF PLANT CAN BE BETTER OR WORSE; COST-BENEFIT ANALYSES WOULD CHANGE PROPORTIONALLY

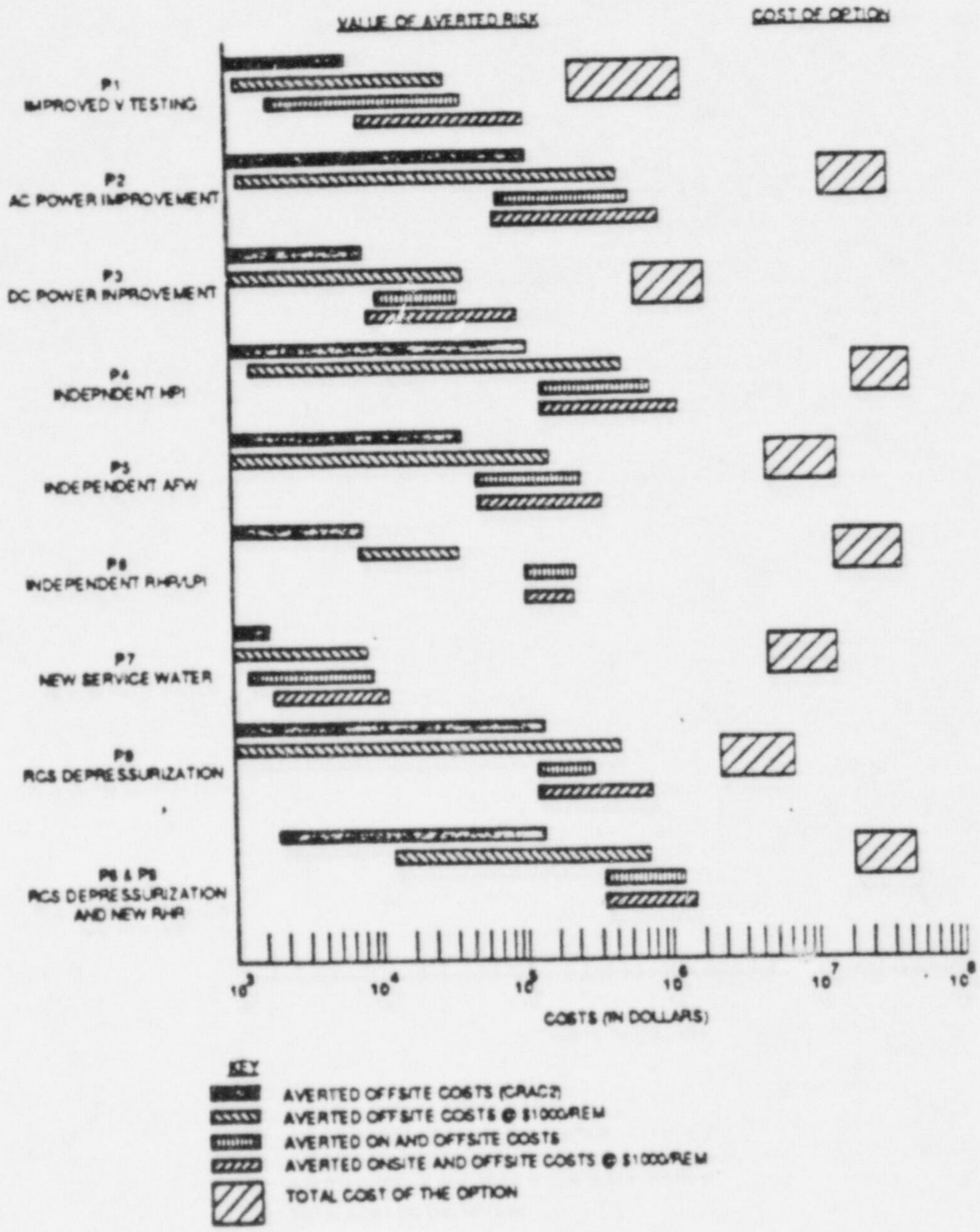


Fig. 9.5

COMPARISON OF COSTS AND BENEFITS FOR PREVENTION OPTIONS

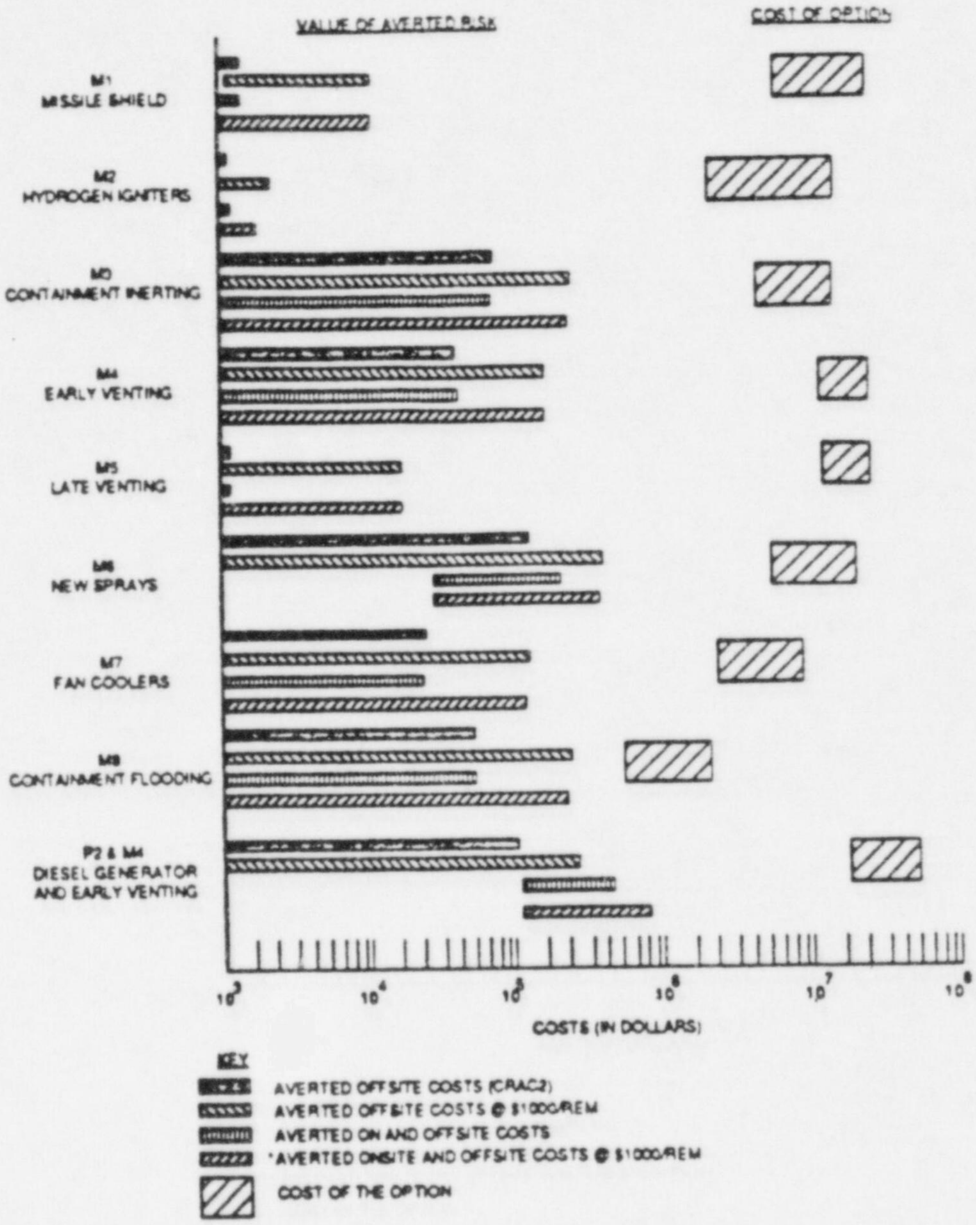
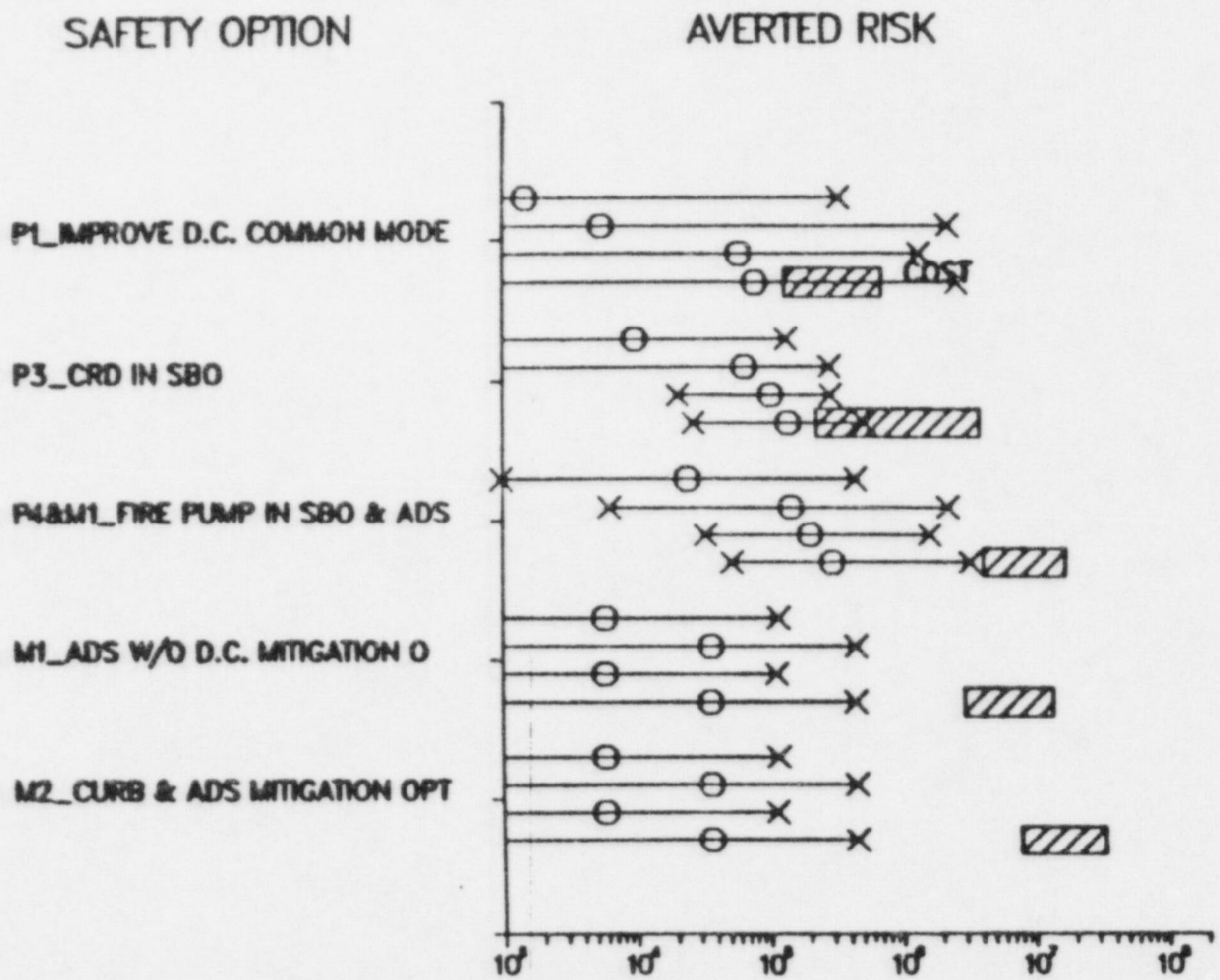
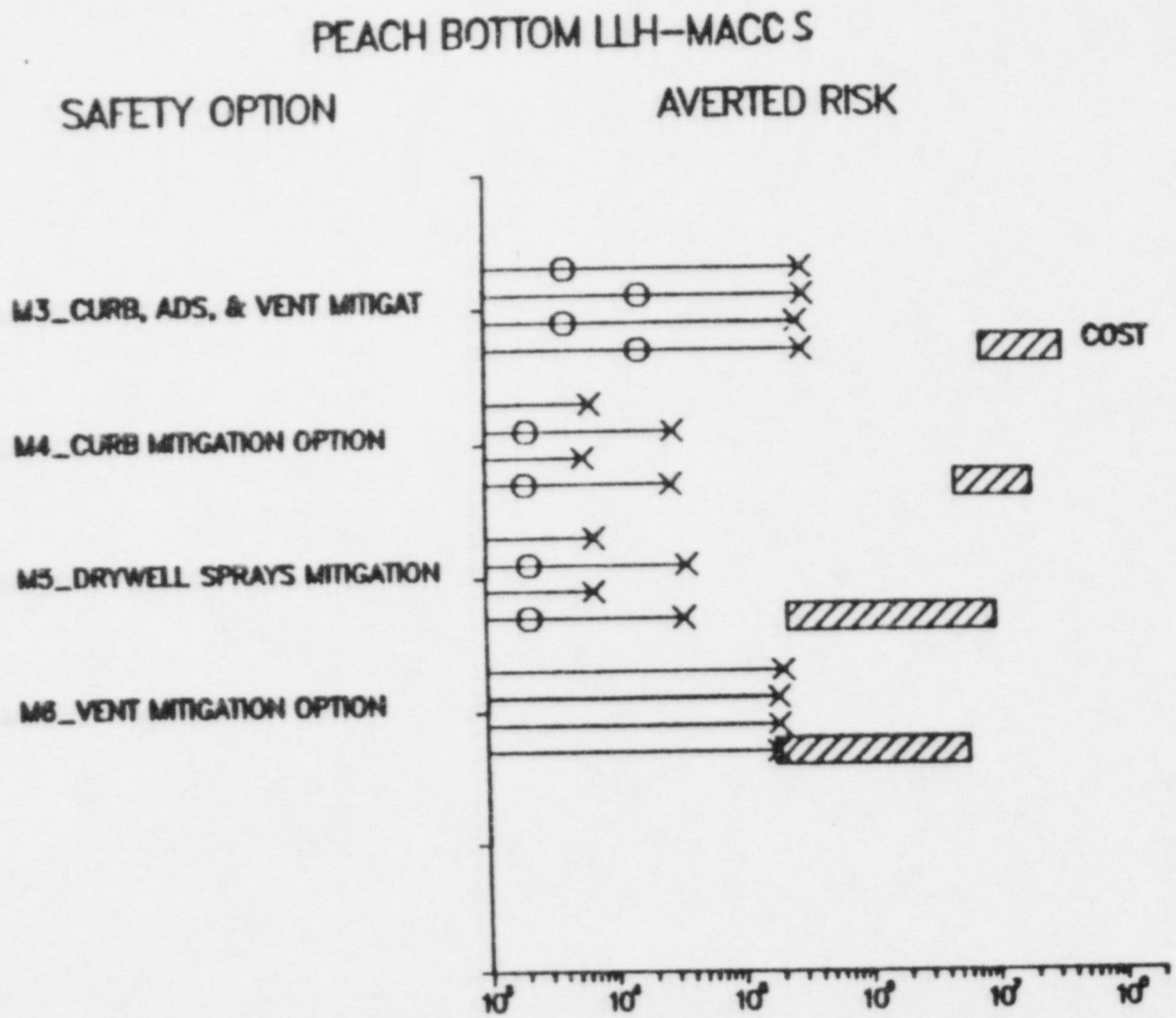


Fig. 9.6
COMPARISON OF COSTS AND BENEFITS FOR MITIGATION OPTIONS

PRELIMINARY RESULTS -- SUBJECT TO CHANGE

PEACH BOTTOM LLH-MACCS




PEACH BOTTOM LLH-MACC S
SAFETY OPTION
AVERTED RISK
M3_CURB, ADS, & VENT MITIGAT
M4_CURB MITIGATION OPTION
M5_DRYWELL SPRAYS MITIGATION
M6_VENT MITIGATION OPTION
COST
10³
10⁴
10⁵
10⁶
10⁷
10⁸

RELATIVE EFFECTIVENESS OF EMERGENCY RESPONSE ACTIONS

CONDITIONAL PROBABILITY OF EXCEEDING 200 REM WHOLE BODY DOSE

EMERGENCY ACTION	1 MILE FROM PLANT HIGH*	1 MILE FROM PLANT LOW*	2 MILES FROM PLANT HIGH*	2 MILES FROM PLANT LOW*	5 MILES FROM PLANT HIGH*	5 MILES FROM PLANT LOW*
CONTINUE NORMAL ACTIVITIES (RELOCATE AFTER 24 HOURS)	.95	.90	.90	.25	.25	0
CONTINUE NORMAL ACTIVITIES (RELOCATE AFTER 6 HOURS)	.95	.45	.45	.10	.03	0
BASEMENT SHELTER	.55	.15	.12	0	0	0
EVACUATE AT 2.7 MPH STARTING:						
-- 0.5 HRS. BEFORE RELEASE	.05	0	0	0	0	0
-- 1 HR. AFTER RELEASE	.70	.10	.25	0	0	0
EVACUATE AT 10 MPH STARTING:						
-- 0.5 HOURS BEFORE RELEASE	0	0	0	0	0	0
-- 1 HOUR AFTER RELEASE	.50	.05	.10	0	0	0
-- 2 HOURS AFTER RELEASE	.80	.15	.15	.05	0	0

* HIGH - SOURCE TERM AT UPPER END OF RANGE OF MEANS
LOW - SOURCE TERM AT LOWER END OF RANGE OF MEANS

CONCLUSION

REFERENCE PLANT STUDIES INDICATE:

- o FREQUENCY OF SEVERE CORE DAMAGE LOWERED DUE TO CHANGES IN RECENT YEARS
  - DESIGN
  - PROCEDURES
  - SYSTEM AND HUMAN RELIABILITY
- o EARLY CONTAINMENT FAILURE CANNOT BE RULED OUT
- o UNCERTAINTIES ARE LARGE, BUT NUREG-1150 PROVIDES GOOD CHARACTERIZATION
- o THE SEVERE ACCIDENT RESEARCH PROGRAM IS DESIGNED TO REDUCE THE REMAINING UNCERTAINTIES IDENTIFIED IN NUREG-0956 AND NUREG-1150
- o IMPORTANT PHENOMENOLOGICAL UNCERTAINTIES HAVE UNKNOWN STATISTICAL DISTRIBUTION; THEREFORE, PROVIDING POINT ESTIMATE MEAN VALUE WOULD BE MISLEADING
- o INFORMATION IS USEFUL TO THE REGULATOR, BECAUSE FULL DISCLOSURE IS MADE REGARDING OUR CURRENT STATE OF KNOWLEDGE

CAUTION

- REPORT IS BEING ISSUED AS A "DRAFT FOR COMMENT"
- THERE ARE KNOWN AREAS WHERE IMPROVEMENTS ARE REQUIRED
- EXTERNAL EVENT ANALYSES COULD ALTER RESULTS SIGNIFICANTLY
- REFERENCE PLANT STUDIES LARGELY RELY ON GENERIC DATA BASE
- GREATER USE OF PLANT-SPECIFIC DATA COULD SUBSTANTIALLY AFFECT RESULTS
- PLANT-SPECIFIC DATA BASE ALSO COULD CHANGE, BECAUSE IT IS DEPENDENT ON EFFECTIVENESS OF PLANT MANAGEMENT IN ACHIEVING RELIABLE SAFETY SYSTEMS AND HUMAN PERFORMANCE
- USE OF THESE RESULTS FOR REGULATORY DECISIONS BEFORE PEER REVIEW AND PUBLIC COMMENT ARE COMPLETED SHOULD BE DONE WITH GREAT CARE.

PROPOSED BWR
SEVERE ACCIDENT
CONTAINMENT REQUIREMENTS

R. M. BERNERO

DECEMBER 22, 1986

FOIA-87-10
K/23

BACKGROUND

- O MARCH 28, 1979: TMI 2 ACCIDENT
- O OCTOBER 2, 1980 FEDERAL REGISTER
  - INTERIM DEGRADED CORE RULE
  - ADVANCE NOTICE OF PROPOSED RULEMAKING LONG TERM DEGRADED CORE
- O DECEMBER 1980: IDCOR FOUNDED
- O NRC SEVERE ACCIDENT RESEARCH PROGRAM
- O JANUARY 6, 1982: SECY 82-1
  - PROPOSED SEVERE ACCIDENT POLICY
- O APRIL 13, 1983: PROPOSED POLICY FOR COMMENT
- O AUGUST 8, 1985: NRC SEVERE ACCIDENT POLICY

SEVERE ACCIDENT POLICY STATEMENT

- OPERATING NUCLEAR POWER PLANTS REQUIRE NO FURTHER REQULATORY ACTION TO DEAL WITH SEVERE ACCIDENT ISSUES UNLESS SIGNIFICANT NEW SAFETY INFORMATION ARISES TO QUESTION WHETHER THERE IS ADEQUATE ASSURANCE OF NO UNDUE RISK TO PUBLIC HEALTH AND SAFETY.

- IN THE LATTER EVENT, A CAREFUL ASSESSMENT SHALL BE MADE OF THE SEVERE ACCIDENT VULNERABILITY POSED BY THE ISSUE AND WHETHER THIS VULNERABILITY IS PLANT OR SITE SPECIFIC OR OF GENERIC IMPORTANCE.

- THE MOST COST-EFFECTIVE OPTIONS FOR REDUCING THIS VULNERABILITY SHALL BE IDENTIFIED AND A DECISION SHALL BE REACHED CONSISTENT WITH THE COST-EFFECTIVENESS CRITERIA OF THE COMMISSION'S BACKFIT POLICY AS TO WHICH OPTION OR SET OF OPTIONS (IF ANY) ARE JUSTIFIABLE AND REQUIRED TO BE IMPLEMENTED.

- IN THOSE INSTANCES WHERE THE TECHNICAL ISSUE GOES BEYOND CURREN REGULATORY REQUIREMENTS, GENERIC RULEMAKING WILL BE THE PREFERR SOLUTION. IN OTHER CASES, THE ISSUE SHOULD BE DISPOSED OF THROUGH THE CONVENTIONAL PRACTICE OF ISSUING BULLETINS AND ORDERS OR GENERIC LETTERS WHERE MODIFICATIONS ARE JUSTIFIED THROUGH BACKFIT POLICY, OR THROUGH PLANT-SPECIFIC DECISION-MAKING ALONG THE LINES OF THE INTEGRATED SAFETY ASSESSMENT PROGRAM (ISAP) CONCEPTION.

IDCOR/NRC PROCESS

- TWO PARALLEL PROGRAMS TO STUDY SEVERE ACCIDENTS IN REFERENCE PLANTS
  - NRC SEVERE ACCIDENT PROGRAM
  - IDCOR
- COMPARE AND RESOLVE TECHNICAL ISSUES
- IDCOR PREPARE AND SUBMIT IPE METHODOLOGY FOR NRC REVIEW
- NRC GENERIC LETTER TO DO IPE
  - WITH GUIDELINES & CRITERIA
  - BY APPROVED METHODOLOGY
- CONDUCT IPE
- IDENTIFY AND EVALUATE OUTLEIRS
- ORDER FIXES

U. S. BOILING WATER REACTORS

- 24 BWR 2/3/4 WITH MARK I CONTAINMENT (ALL LICENSED)
- 9 BWR 4/5 WITH MARK II CONTAINMENT (8 LICENSED)
- 4 BWR 6 WITH MARK III CONTAINMENT (4 LICENSED)

TABLE 1 - U.S. BWR PLANT-SPECIFIC PRA STUDIES

PLANT	PROGRAM NAME	REPORT	REPORT YEAR	CORE/ CONTAINMENT	REACTOR POWER (MWT)	CORE-DAMAGE FREQUENCY PRA ESTIMATE	EVENTS CONSIDERED	MEDIAN, MEAN OR POINT ESTIMATE	CONTAINMENT CONDITIONAL FAILURE PROBABILITY
Peach Bottom	RSS	WASH-1400	1975	BWR-4/MK I	3293	$3x10^{-5}$	Internal/ External	Median	Not evaluated
Peach Bottom	IDCOR	Tech Summary Task 21	1984	BWR-4/MK I	3293	$4x10^{-5}$	Internal	Mean	0.2
Peach Bottom	IPE	IPE	1986	BWR-4/MK I	3293	$2x10^{-5}$	Internal	Mean	Not evaluated
Millstone	IREP	NUREG/CR - 3085	1983	BWR-3/MK I	1727	$3x10^{-4}$	Internal	Median	Not evaluated
Millstone	NUSCO	Millstone 1 PSS	1986	BWR-3/MK I	1727	$5x10^{-4}$	Internal	Mean	Not evaluated
Brown, Ferry	IREP	NUREG/CR 2801	1982	BWR-4/MK I	3293	$2x10^{-4}$	Internal	Point Estimate	Not evaluated
Vermont Yankee	VYCSS	VYCSS	1986	BWR-4/MK I	1593	$3x10^{-5}$	Internal	Mean	0.07
Big Rock Point	Consumers	Big Rock Point PRA	1981	BWR-1/Dry	158	$1x10^{-3}$	Internal/ External	Mean	0.25
Big Rock Point	EG&G/BNL	EG&G-EA-5533 Rev. 1	1982	BWR-1/Dry	158	$1x10^{-3}$	Internal/ External	Mean	0.25
Limerick	PEPCO	Limerick PRA	1981	BWR-4/MK II	3293	$7x10^{-5}$	Internal/ External	Mean	1.0
Limerick	BNL	NUREG/CR-3028	1983	BWR-4/MK II	3293	$1x10^{-4}$	Internal/ External	Mean	1.0
Shoreham	LILCO	Shoreham PRA	1983	BWR-4/MK II	2436	$5x10^{-5}$	Internal	Point Estimate	Not evaluated
Shoreham	BNL	NUREG/CR-4050	1985	BWR-4/MK II	2436	$1x10^{-4}$	Internal	Point Estimate	Not evaluated
Shoreham	IPE	Shoreham IPE	1986	BWR-4/MK II	2436	$8x10^{-5}$	Internal	Mean	Not evaluated
Susquehanna	IPE	IPE	1986	BWR-4/MK II	3293	$2x10^{-7}$	Internal	Mean	Not evaluated
Grand Gulf	RSSMAP	NUREG/CR-1659	1981	BWR-6/MK III	3833	$4x10^{-5}$	Internal	Median	Not evaluated
Grand Gulf	IDCOR	Tech Summary Task 21	1984	BWR-6/MK III	3833	$8x10^{-6}$	Internal	Mean	Not evaluated
GESSAR	GE	GESSAR II PRA		BWR-6/MK III	3579	$4x10^{-6}$	Internal/ External	Mean	Not evaluated

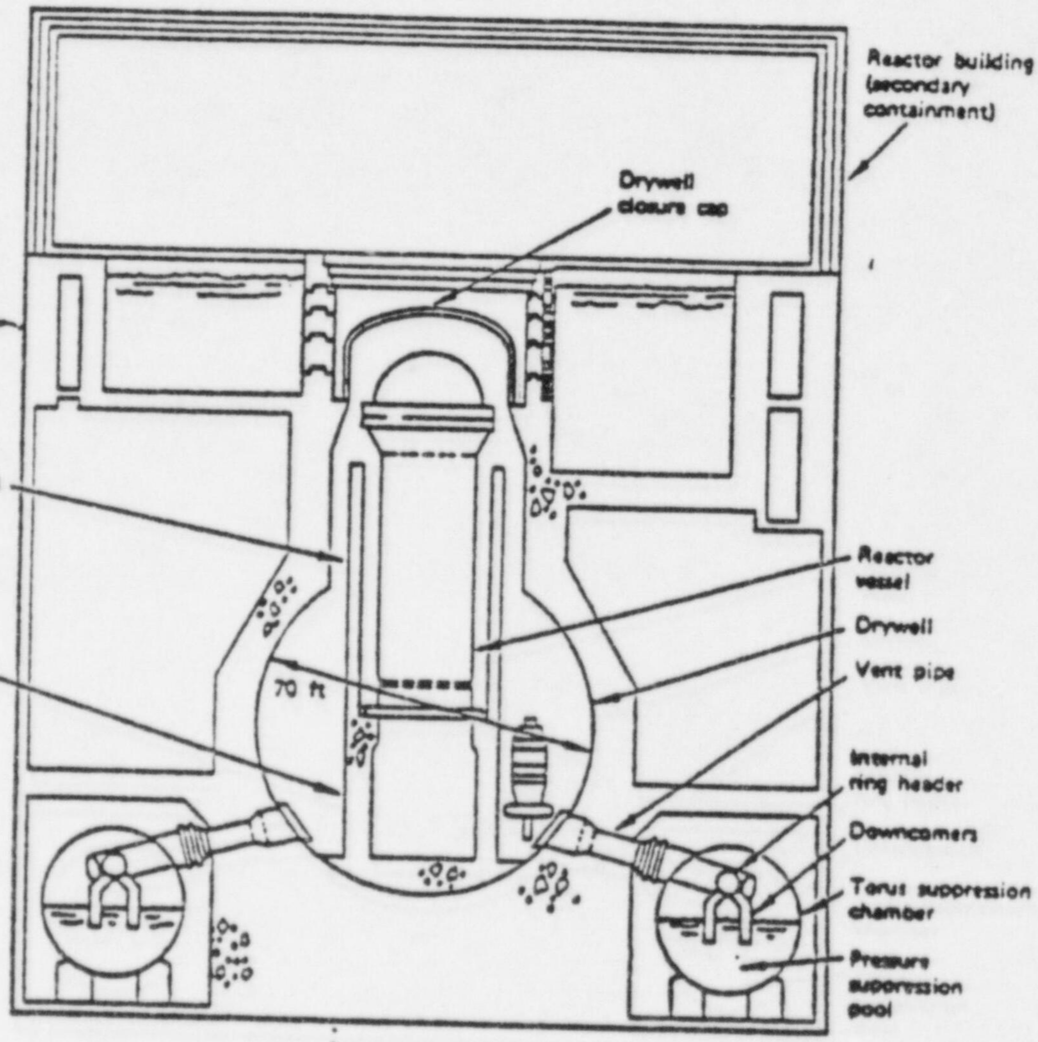


Figure 4. BWR Mark I containment configuration

BWR CONTAINMENT ISSUES - MARK I

- SMALL VOLUME
  - MORE RAPID OVERPRESSURE
  - ESPECIALLY VULNERABLE TO HYDROGEN BURN

- SMALL DRYWELL FLOOR
  - LOWER HEAD AREA CLOSE TO DRYWELL WALL
  - POTENTIAL FOR DIRECT DEBRIS ATTACK
  - DIRECT RADIATION AND CONVECTION HEATING

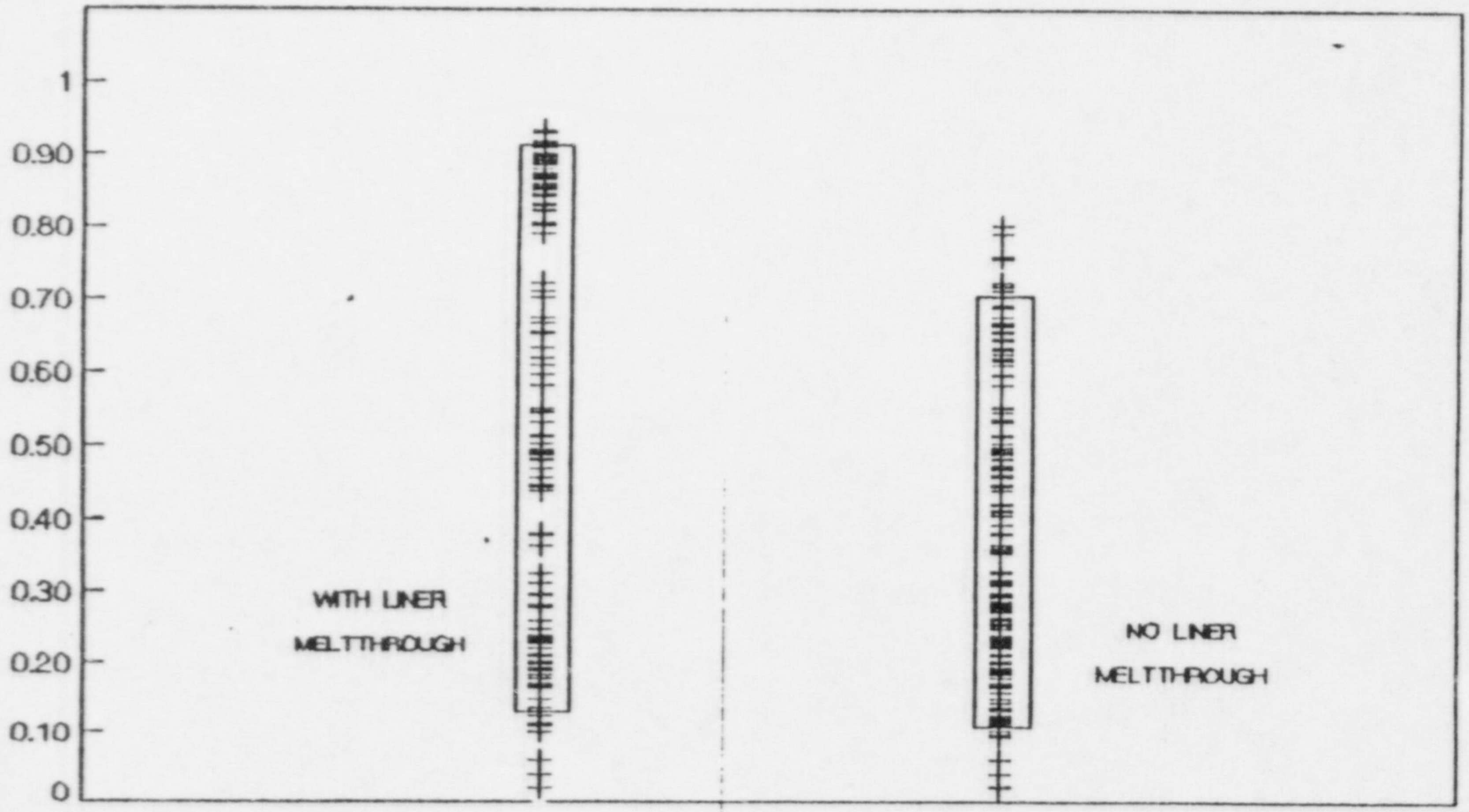
- LIMITED PASSIVE CAPABILITY BUT OPTIONS FOR ACTIVE RESPONSE

- 5-ELEMENT APPROACH
  - HYDROGEN CONTROL
  - SPRAY IN DRYWELL
  - PRESSURE RELIEF
  - DEBRIS CONTROL
  - PROCEDURES AND TRAINING

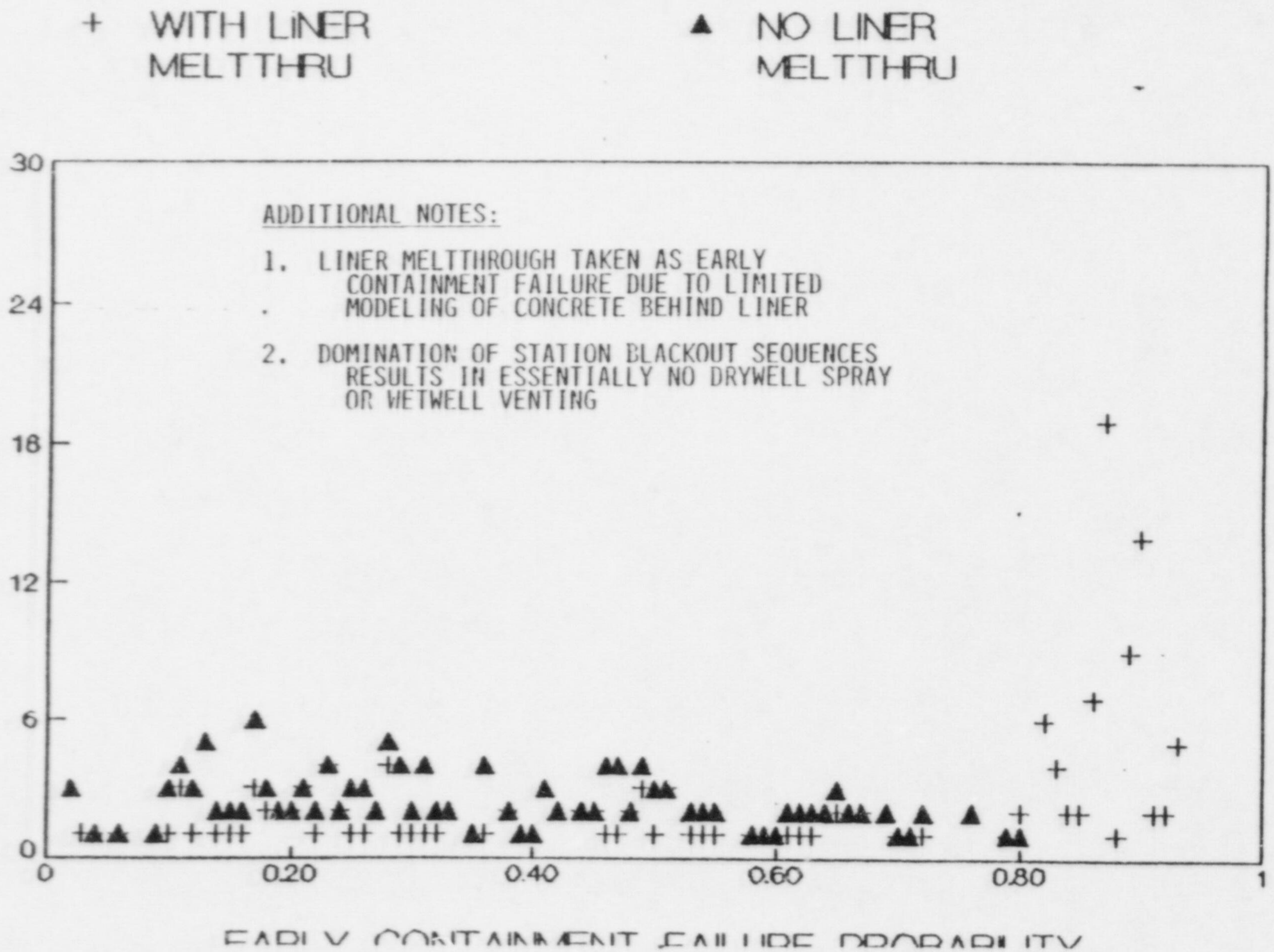
KEY RESULTS FOR BWR CONTAINMENTS

- REACTOR SAFETY STUDY - PEACH BOTTOM 90% EARLY RELEASE
- IDCOR - PEACH BOTTOM 20% EARLY RELEASE
- VERMONT YANKEE - 7% EARLY RELEASE
- NUREG-1150

PROBABILITY - EARLY CONTAINMENT FAILURE
PEACH BOTTOM PLANT



PROBABILITY – EARLY CONTAINMENT FAILURE
PEACH BOTTOM PLANT



BWR CONTAINMENT VENTING

- O TO PREVENT OVERPRESSURE FAILURE
  - POSSIBLE RELEASE OF NOBLE GAS ACTIVITY
- O FIRST "APPROVED" IN GENERIC LETTER 83-05 (FEBRUARY 8, 1983)
  - EPG REV. 2 SAFETY EVALUATION
  - SUPPLEMENTAL DEVELOPMENT NEEDED
- O SECOND "APPROVAL" IN LETTER TO BWROG (NOVEMBER 23, 1983)
  - EPG REV. 3 SAFETY EVALUATION
  - STILL AWAITING FUTURE SUBMITTALS ON CRITERIA FOR CONTAINMENT VENTING PRESSURE
  - REQUEST TO CONSIDER FURTHER TECHNICAL STRATEGIES FOR A DEGRADED CORE CONDITION. (SEVERE ACCIDENT MANAGEMENT)
- O ALL LARGE BWRS HAVE VENTING PROCEDURES
  - VARIOUS PRESSURES (0.5-2 x DESIGN)
  - EXISTING DUCTS

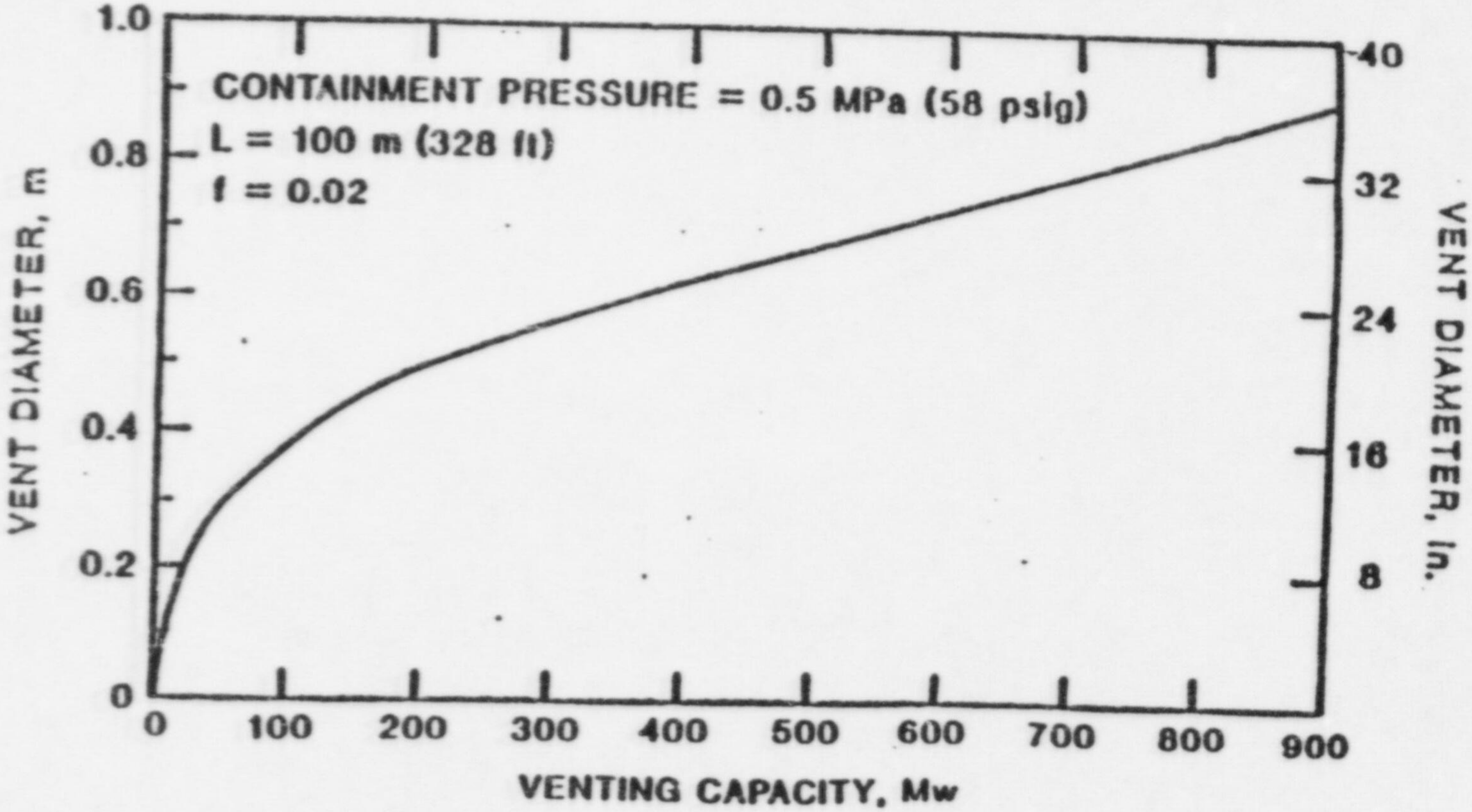
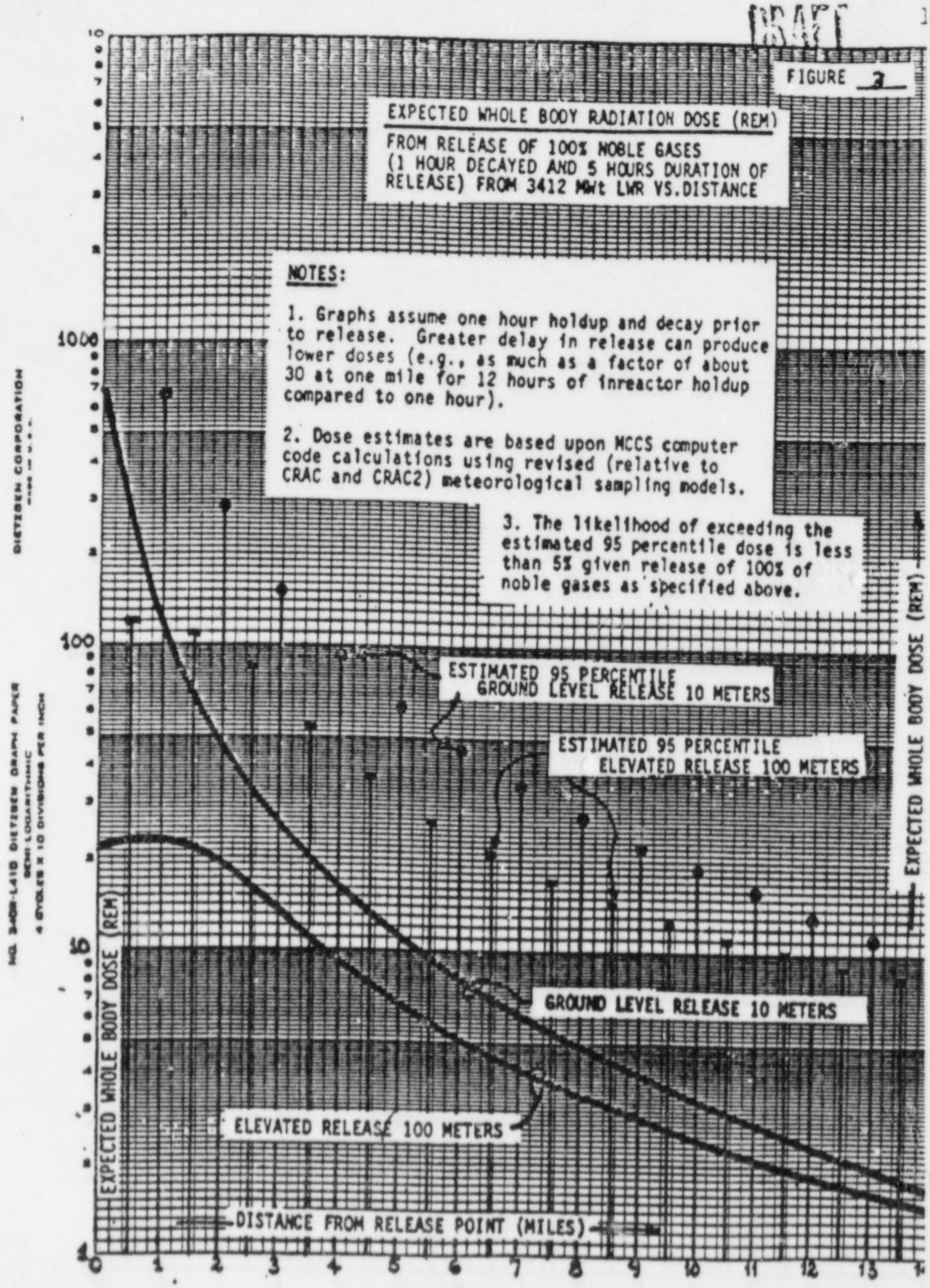


Figure 2. Vent Size Requirement as a Function of Power

(This figure is reproduced from IDCOR Report, "Evaluation of BWR Accident Mitigation Capability Relative to proposed NRC Changes," August 1986.)

DRAFT


DRAFT
FIGURE 3
EXPECTED WHOLE BODY RADIATION DOSE (REM)
FROM RELEASE OF 100% NOBLE GASES (1 HOUR DECAYED AND 5 HOURS DURATION OF RELEASE) FROM 3412 MWt LWR VS. DISTANCE
NOTES:
1. Graphs assume one hour holdup and decay prior to release. Greater delay in release can produce lower doses (e.g., as much as a factor of about 30 at one mile for 12 hours of inreactor holdup compared to one hour).
2. Dose estimates are based upon MCCS computer code calculations using revised (relative to CRAC and CRAC2) meteorological sampling models.
3. The likelihood of exceeding the estimated 95 percentile dose is less than 5% given release of 100% of noble gases as specified above.
ESTIMATED 95 PERCENTILE GROUND LEVEL RELEASE 10 METERS
ESTIMATED 95 PERCENTILE ELEVATED RELEASE 100 METERS
GROUND LEVEL RELEASE 10 METERS
ELEVATED RELEASE 100 METERS
EXPECTED WHOLE BODY DOSE (REM)
DISTANCE FROM RELEASE POINT (MILES)
1000
100
10
4
0
1
2
3
4
5
6
7
8
9
10
11
12
13
14
DIETZGEN CORPORATION
NO. 340R-L410 DIETZGEN GRAPH PAPER
SEMI-LOGARITHMIC
4 CYCLES X 10 DIVISIONS PER INCH

CONTAINMENT IMPROVEMENT STRATEGY

- PREVENT HYDROGEN COMBUSTION BY INERTING
- REDUCE DRYWELL SPRAY FLOW RATE
  - PERMITS ALTERNATE SUPPLIES TO PRODUCE SPRAY
  - EXTENDS WATER SUPPLIES
- PROVIDE RELIABLE BACKUP SUPPLIES FOR DRYWELL SPRAY
  - PROVIDES SHALLOW POOL OF WATER ON DRYWELL FLOOR
  - DIRECT SPRAY COOLING OF ANY CORE DEBRIS LEAVING LOWER HEAD AREA
  - SPRAY SCRUBBING OF DRYWELL VOLUME
  - DIRECT COOLING OF WALLS
- WETWELL PRESSURE RELIEF TO STACK
  - POOL SCRUBBING
  - ELEVATED RELEASE
- DEBRIS CONFINEMENT.
- TRAINED OPERATORS

PROPOSED REQUIREMENTS

1. HYDROGEN CONTROL

   PRESENT REQUIREMENTS IMPOSED BY 10 CFR PART 50.44 AND THE TECHNICAL SPECIFICATIONS SHALL BE ADHERED TO, NO ADDITIONAL REQUIREMENTS ARE PROPOSED.

2. CONTAINMENT SPRAY

   ALL BWRs WITH MARK I CONTAINMENT SHALL PROVIDE AT LEAST TWO BACKUP WATER SUPPLY SYSTEMS FOR THE CONTAINMENT DRYWELL SPRAY, ONE OF WHICH SHALL BE FUNCTIONAL DURING STATION BLACKOUT. WATER TO THE SPRAY SYSTEM FROM THESE BACKUP SUPPLIES SHALL BE AVAILABLE BY REMOTE MANUAL OPERATION OR BY SIMPLE PROCEDURES FOR CONNECTION AND STARTUP WHICH CAN BE IMPLEMENTED DURING A SEVERE ACCIDENT SCENARIO.

   IN ADDITION, THE SPRAY NOZZLES SHALL BE ADJUSTED SO THAT AN EVENLY DISTRIBUTED SPRAY PATTERN WILL BE DEVELOPED IN THE DRYWELL WHETHER WATER IS SUPPLIED BY THE PRIMARY SOURCE OR EITHER OF THE BACKUP SOURCES. A FLOW RATE ON THE ORDER OF 1/10 OF THE PRESENT FLOW RATE IS CONSIDERED TYPICAL, THE LICENSEE SHALL SELECT THE FLOW BASED ON AN ANALYSIS OF PLANT SPECIFIC PARAMETERS.

PROPOSED REQUIREMENTS (CONT'D.)

3. PRESSURE RELIEF

THE LICENSEE SHALL SELECT A PRESSURE BETWEEN DESIGN PRESSURE AND 1½ TIMES DESIGN PRESSURE AT WHICH TO OPEN AN EXHAUST PATH FROM THE WETWELL VAPOR SPACE TO THE HIGHEST VENT POINT (STACK OR PIPE) AVAILABLE. THIS LINE SHOULD BE CAPABLE OF HANDLING WATER VAPOR FLOW EQUIVALENT TO 1% DECAY HEAT AT THE VENT PRESSURE SELECTED WITHOUT SIGNIFICANT CHANCE OF RUPTURE BEFORE THE DESIRED RELEASE POINT. THE LINE SHALL BE EQUIPPED WITH ISOLATION VALVES WHICH CAN BE OPENED AND RECLOSED BY REMOTE MANUAL OPERATION OR BY SIMPLE PROCEDURES WHICH CAN BE IMPLEMENTED DURING SEVERE ACCIDENT SCENARIOS INCLUDING STATION BLACKOUT.

4. CORE DEBRIS CONTROL

THE LICENSEE SHALL ENSURE THAT THE WATER IN THE SUPPRESSION POOL IN THE EVENT OF TORUS FAILURE IS HELD WITHIN THE CONFINES OF THE TORUS ROOM AND THE CORNER ROOMS AND CANNOT FLOW OUT TO OTHER PARTS OF THE PLANT.

5. PROCEDURES AND TRAINING

THE LICENSEE SHALL IMPLEMENT EMERGENCY OPERATING PROCEDURES AND OTHER PROCEDURES BASED ON ALL SIGNIFICANT ELEMENTS APPROPRIATE TO ITS PLANT OF EMERGENCY PROCEDURE GUIDELINES, REVISION 4.

CONDITIONS

QUALITY AND DESIGN STANDARDS

SINCE THESE REQUIREMENTS ARE INTENDED TO BE AN OPTIMIZED USE OF EXISTING EQUIPMENT IT IS EXPECTED THAT ADDED EQUIPMENT, OF ITSELF, NEED NOT MEET THE QUALITY OR DESIGN STANDARDS OF SAFETY RELATED EQUIPMENT. NEVERTHELESS, MODIFICATIONS TO OR NEAR EQUIPMENT OR SYSTEMS WHICH ARE ALREADY SAFETY RELATED SHALL NOT COMPROMISE THE QUALITY OF SUCH EQUIPMENT OR SYSTEMS.

IMPLEMENTATION

THE EQUIPMENT CHANGES REQUIRED HEREIN SHALL BE INSTALLED DURING THE FIRST REFUELING OUTAGE WHICH BEGINS NINE (9) MONTHS AFTER THE EFFECTIVE DATE OF THIS LETTER. THE PROCEDURES AND TRAINING REQUIRED SHALL BE IMPLEMENTED ON A SCHEDULE REVIEWED AND APPROVED BY THE NRC. GIVEN THE IMPLEMENTATION OF THE GENERIC IMPROVEMENTS OF MARK I CONTAINMENTS THERE IS NO NEED FOR AN INDIVIDUAL PLANT EVALUATION (IPE) FOR CONTAINMENT PERFORMANCE. THIS DOES NOT REMOVE THE NEED FOR AN IPE WHICH COVERS THE SYSTEM RELIABILITY OR CORE MELT FREQUENCY PORTION OF THE SEVERE ACCIDENT QUESTION.

WHY DO THIS

- "COMPLIANCE" WITH THE SAFETY GOAL? - MOST CONTAINMENTS NOT NEEDED
- NEEDED FOR SAFETY?
  - POSSIBLE ARGUMENT
  - RESPONSE TO MARKEY QUESTION
- JUSTIFIABLE BACKFIT? - YES

COMMISSION RESPONSE TO A HEARING QUESTION

JULY 16, 1986

QUESTION

IS A 90 PERCENT CHANCE OF FAILURE IN THE EVENT OF A CORE MELTDOWN AN ACCEPTABLE FAILURE RATE?

ANSWER

THE NRC HOLDS THE POSITION THAT THE LIKELIHOOD OF CORE MELT ACCIDENTS IN ANY PLANT SHOULD BE VERY LOW AND, IN ADDITION, THAT THERE SHOULD BE SUBSTANTIAL ASSURANCE THAT THE CONTAINMENT WILL MITIGATE THE CONSEQUENCES OF A CORE MELT SHOULD ONE OCCUR IN ORDER TO ENSURE LOW RISK TO THE PUBLIC. IT IS NOT MERELY A QUESTION OF HAVING LOW RISK BUT OF HAVING AS WELL THE DEFENSE-IN-DEPTH ASSURANCE OF COMBINED PROTECTION BY PREVENTION AND MITIGATION...

TABLE 3 COST-BENEFIT ANALYSIS

COST: $0.7-2.2M

BENEFIT:(1)	FCM	CCFP BEFORE	CCFP AFTER	AVERTED LOSS/YR	AVERTED LOSS PRES. VALUE
BASE CALCULATION	$1x10^{-4}$/yr	0.5	0.05	\$$4x10^{5}$/yr	$3M/$12M
LOWER FCM	$1x10^{-5}$/yr	0.5	0.05	\$$4x10^{4}$/yr	$0.3M/$1.2M
LESS CHANGE IN CONTAINMENT	$1x10^{-4}$/yr	0.5	0.1	\$$4x10^{5}$/yr	$3M/$12M
BETTER CONTAINMENT TO START	$1x10^{-4}$	0.2	0.05	\$$2x10^{5}$/yr	$2M/$6M
"OPTIMISTIC" CALCULATION	$1x10^{-5}$	0.2	0.05	\$$2x10^{4}$/yr	$0.2M/$0.6M
"PESSIMISTIC" CALCULATION	$3x10^{-4}$	0.9	0.1	\$$2x10^{5}$/yr	$16M/$60M

(1) FCM = Frequency of Core Melt
CCFP = Conditional Containment Failure Probability
AVERTED LOSS PRESENT VALUE expressed as A/B where A is the averted loss per year times 8 (roughly equivalent to discount at 12%/yr rate) and B is the averted loss per year times 30 (no discount).

METHOD OF ACTION

O RULEMAKING?

- COMMISSION CLOSED SEVERE ACCIDENT RULEMAKING
- COMMISSION CHOSE DEVELOPMENT OF A SAFETY GOAL FOR GENERAL PERFORMANCE STANDARD
- TEDIOUS TO START A SERIES OF 50.44 TYPE RULES

O GENERIC LETTER/ORDER?

- CLASS SPECIFIC
- PUBLISH FOR COMMENT
- OPTIMUM TREATMENT OF DETAILS
- PREFERRED FOR PROMPT EFFECTIVE USE OF RESOURCES

T. Pickens
12-12-86

BWR OWNERS' GROUP

SEVERE ACCIDENT CONTAINMENT ISSUES

NUMARC APPROVED SEVERE ACCIDENT CONTAINMENT ISSUES APPROACH

1. OBJECTIVE
   EVALUATE CONTAINMENT INTEGRITY. IF APPROPRIATE, ASSESS POTENTIAL IMPROVEMENTS TO MINIMIZE OFFSITE RELEASES FOR SEVERE ACCIDENT CONDITIONS (BEYOND DBA) WITHIN AN APPROPRIATE COST/BENEFIT GOAL.

2. IDENTIFY CHALLENGES TO CONTAINMENT *
   1.) H_2 GENERATION
   2.) OVERPRESSURE
   3.) TEMPERATURE
   4.) CORE DEBRIS ATTACK
   5.) FISSION PRODUCT CONTROL
   6.) HUMAN ACTIONS
   7.) DIRECT CONTA~~INM~~ENT HEATING

3. IDENTIFY INITIATORS TO EACH CHALLENGE FROM EXISTING ANALYSES
   PICK KEY EVENTS/INITIATORS (MOST SEVERE - LESS SEVERE)
   SEQUENCE THAT PRODUCES MOST SEVERE CHALLENGE

4. ASSESS PLANTS' ABILITIES TO MEET CHALLENGES

5. ASSESS PLANT VULNERABILITIES

6. PROPOSE ALTERNATIVES TO ADDRESS VULNERABILITIES

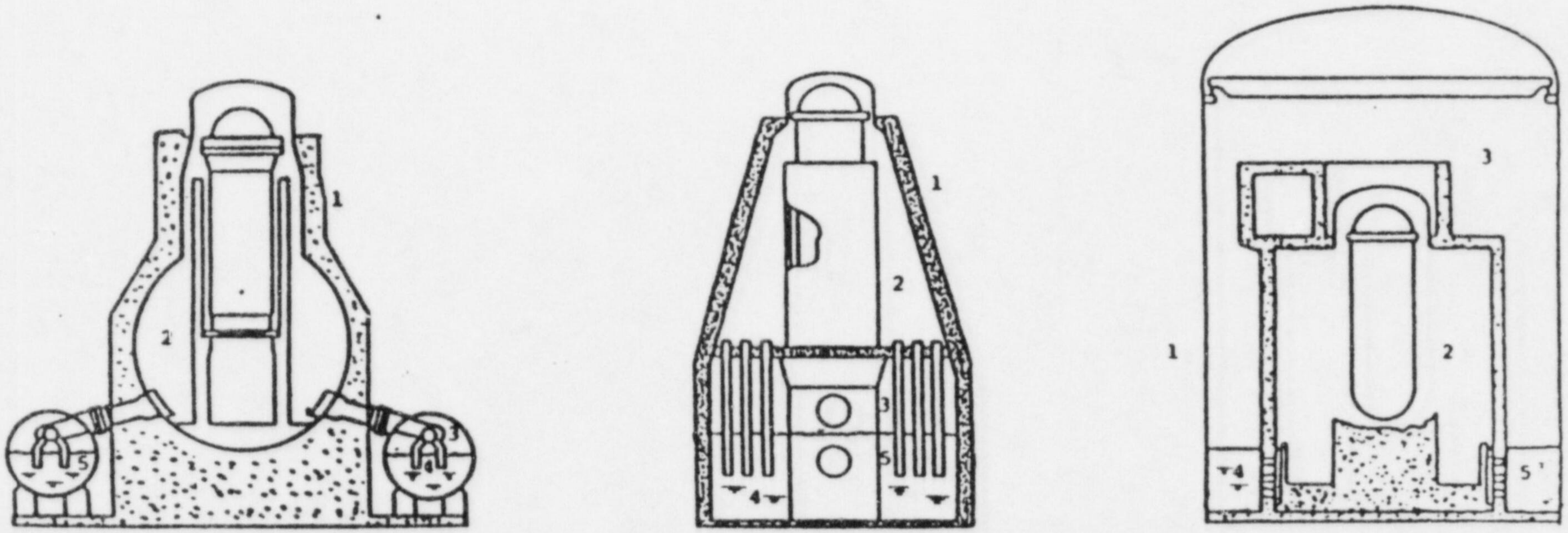
7. EVALUATE ALTERNATIVES

8. REACH DECISIONS

* OTHER ISSUES TO BE ADDED

OG37/12.12

1 = PRIMARY CONTAINMENT
2 = DRYWELL
3 = WETWELL
4 = SUPPRESSION POOL
5 = VENT SYSTEM



MARK I　　MARK II　　MARK III

FIG. 2.1 GENERAL ELECTRIC PRESSURE SUPPRESSION SYSTEM DESIGNS

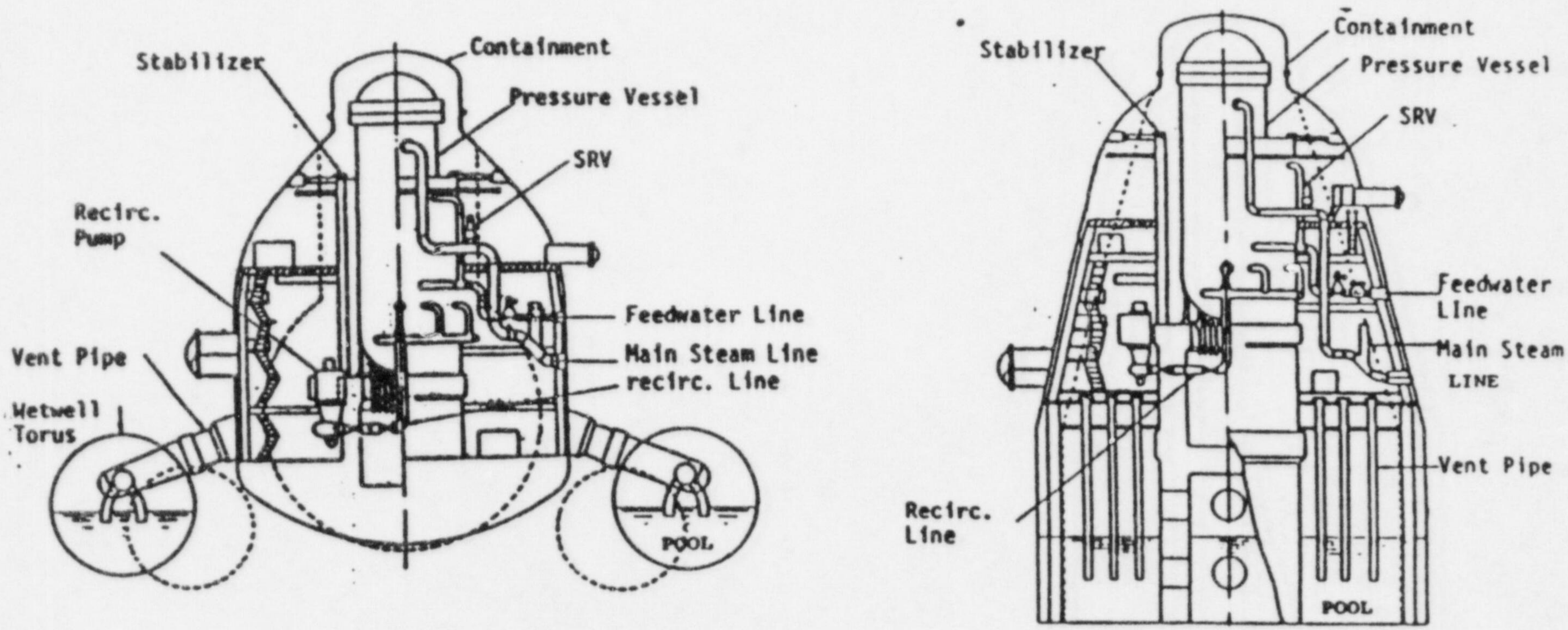
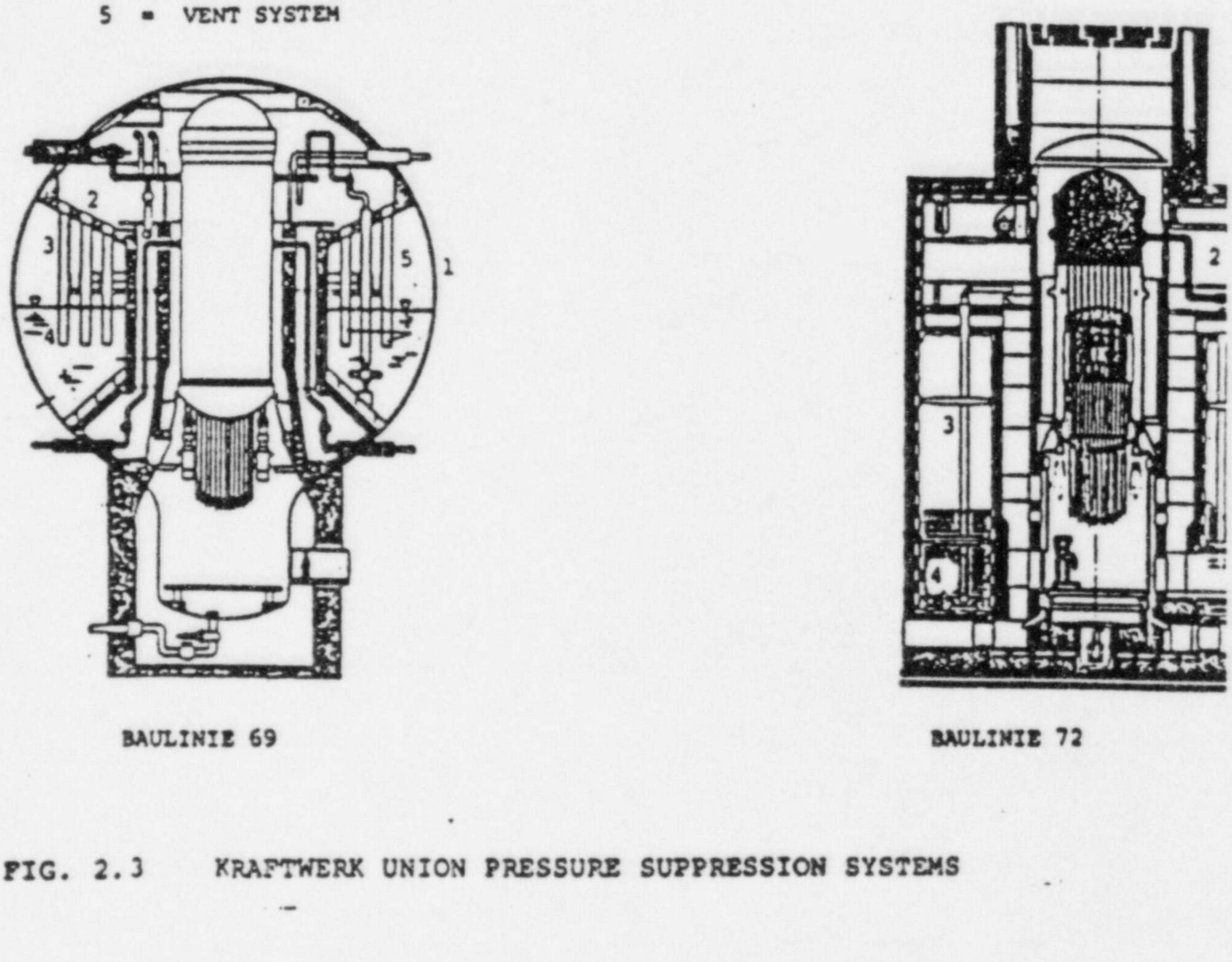


FIG. 2.2 JAPANESE CONTAINMENT DESIGNS - MODIFIED MARK I AND II CONFIGURATIONS
(DOTTED LINES SHOW ORIGINAL CONFIGUP^TIO!

1 = PRIMARY CONTAINMENT
2 = DRYWELL
3 = WETWELL
4 = SUPPRESSION POOL
5 = VENT SYSTEM



BAULINIE 69

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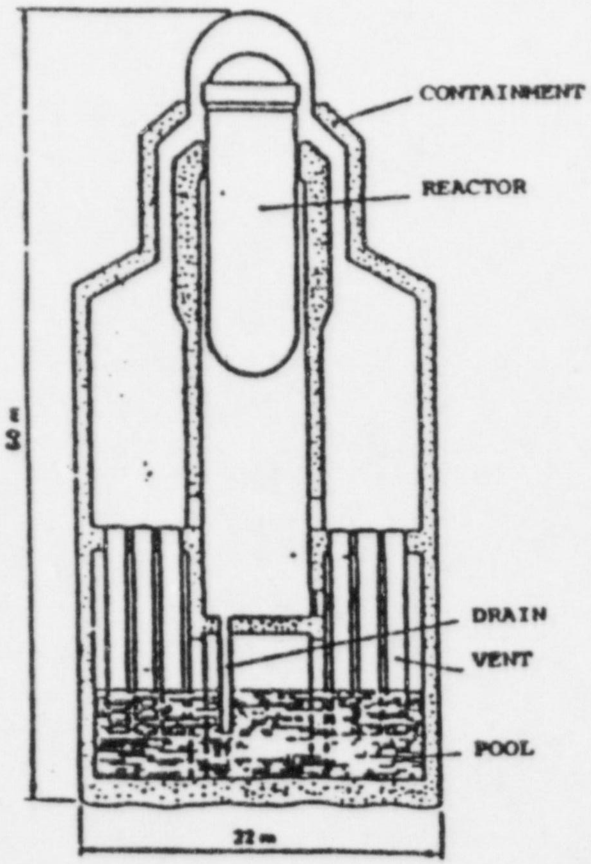
FIG. 2.3 KRAFTWERK UNION PRESSURE SUPPRESSION SYSTEMS

CONTAINMENT FOR EXTERNAL
RECIRCULATION PUMP
REACTOR



CONTAINMENT FOR INTERNAL
RECIRCULATION PUMP
REACTOR

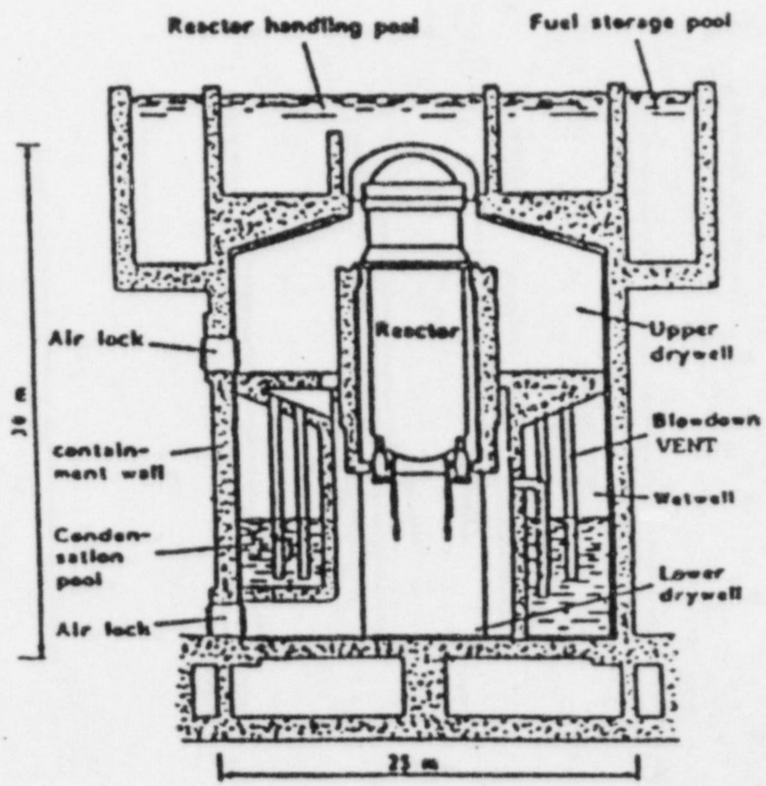
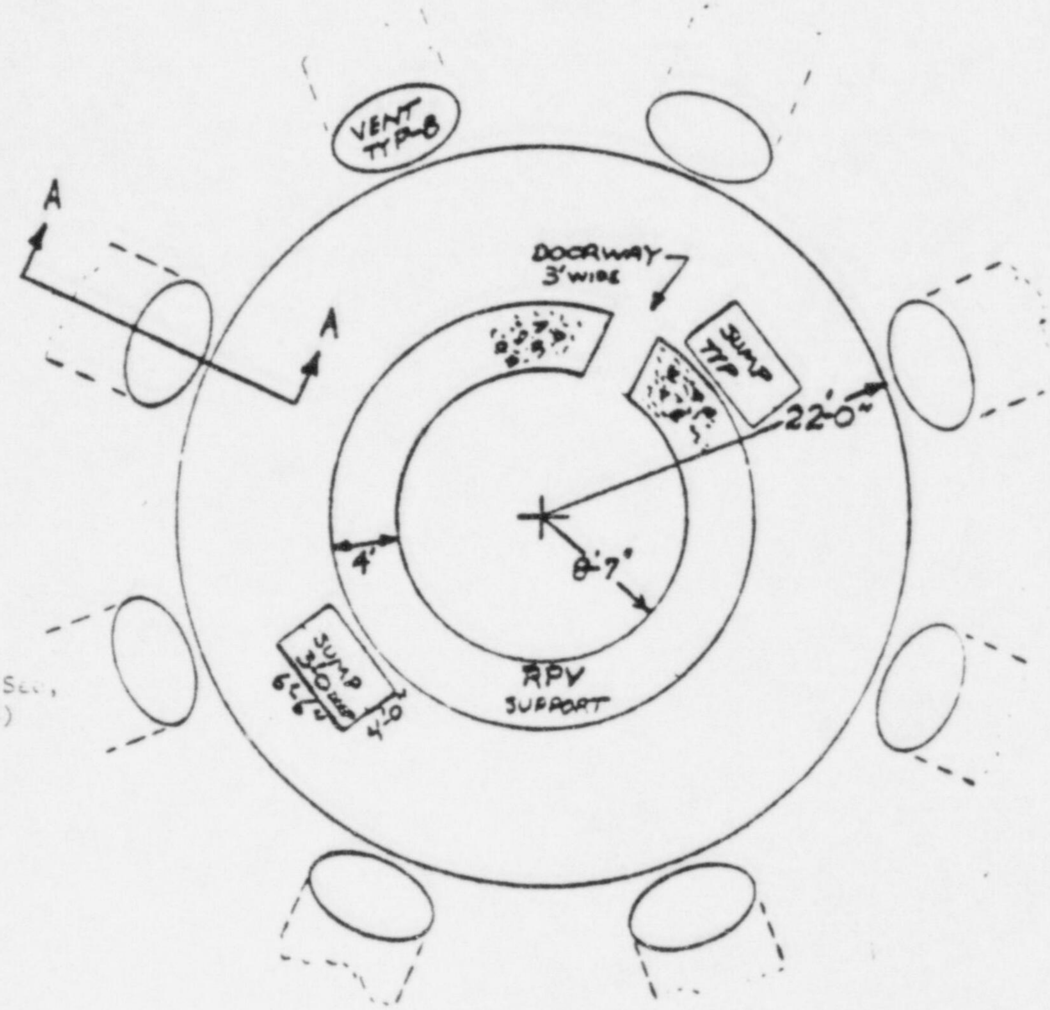


FIG. 2.4 ASEA-ATOM PRESSURE SUPPRESSION SYSTEMS

Drywell Floor Area



(For Detail See, Figure 5-14)

PLAN - DRYWELL FLOOR

Figure 5-13

Drywell Shell (Detail A-A of Figure 5-13)

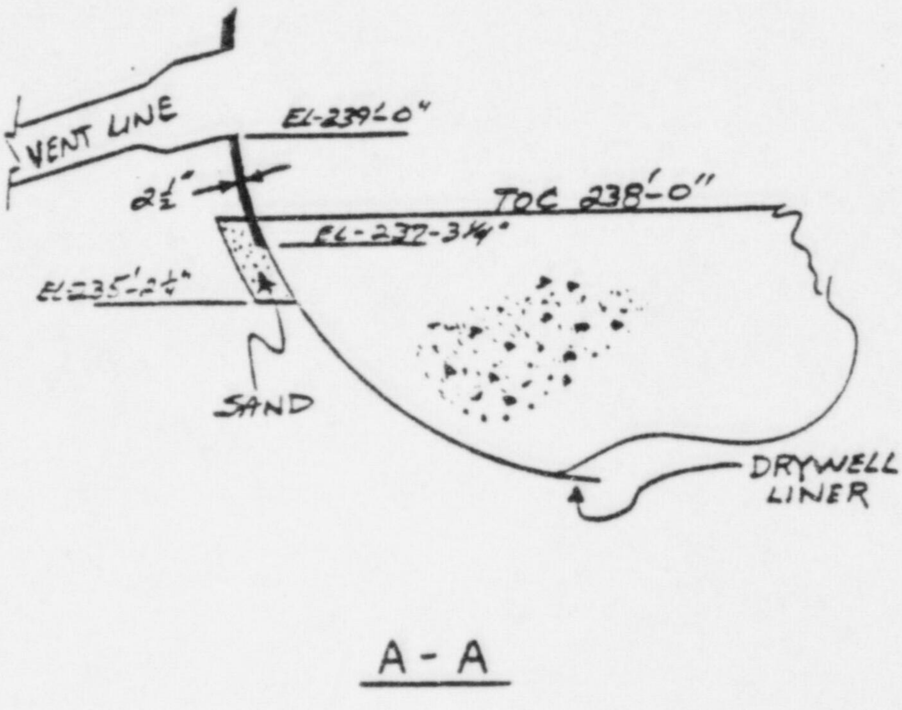


Figure 5-14

HOLMAN

Evaluation of BWR Accident Mitigation Capability Relative to Proposed NRC Changes

Submitted by
IDCOR
to
BWROG

August 1986

FOIA-87-10

K/C

8609290088XA

ABSTRACT

The NRC has suggested five changes to operating BWRs which could have the potential for providing additional containment mitigation capability. This report summarizes these changes and identifies the feasibility, safety improvement, and potential competing risks associated with these changes. The five changes are related to the following:

- o Hydrogen combustion control
- o Molten debris coolability
- o Containment pressure relief
- o Direct containment attack by molten material
- o Training and procedures under severe accidents

The evaluation has been carried out by an IDCOR team making use of the analytic models developed within the IDCOR program and the insights gained from previous analyses. Individual utilities participated in the development of the plant specific containment description and in the cost estimates of design options. The evaluation is being submitted to the BWROG for information and use by BWR owners in making decisions regarding these suggested changes.

The approach taken in the evaluation is as follows:

- o Define the basic issues behind the proposed changes in terms of effects on key BWR containment functions.
- o Develop several conceptual design options.
- o Look at the changes in the context of
  - the basic issues
  - dominant severe accident sequence impacts
  - containment failure modes
- o Summarize the results in a manner that can aid the BWR owners in the decision making process.

A value-impact evaluation has not been performed although costs for various conceptual designs are included. A value impact evaluation would require quantification of the public risk changes associated with the proposed changes. This has not been performed as part of this evaluation.

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(continued)

ACKNOWLEDGEMENTS

The IDCOR team that provided the technical evaluation included:

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W.A. Brinsfield (Delian Corporation)

R.G. Brown (Delian Corporation)

Additional valuable technical input was provided from:

T. Pickens (BWROG Chairman)

K.W. Holtzclaw (GE)

A.F. Deardorff (Nutech)

Section 1
INTRODUCTION

1.1 BACKGROUND

A substantial amount of work has been performed by IDCOR [1-1] and the NRC regarding the acceptability of plants under severe accident conditions. These efforts are currently ongoing, but the initial phases of this work have been completed. At this time no plants have been identified as presenting unacceptable risk to the health and safety of the public. Calculations of the public health risk for both BWRs and PWRs demonstrate that each type meets the proposed NRC safety goals [1-2] for the plants analyzed to date (see Table 1.1-1). The conclusions can be summarized as follows:

- o The probabilities of severe nuclear accidents occurring are extremely low.
- o The source terms are likely to be much less than calculated previously.
- o Risks and consequences to the public of severe accidents are much smaller than the risk levels incorporated in the NRC interim safety goals.

These conclusions are based upon evaluations performed on a limited number of plants. In order to ensure that these generic conclusions also apply to specific individual plants, the industry in cooperation with the NRC has formulated a method to examine each plant for possible weaknesses which could cause these generic conclusions to not hold. This method, called the Individual Plant Evaluation (IPE), provides a structured approach to systematically examine a specific plant for core melt frequency and source term outliers. Although the specific requirements for compliance with the NRC's Severe Accident Policy have yet to be published formally, it is probable that each plant will be required to utilize the IPE (or another equivalent or more in-depth tool like a PRA) to demonstrate that there are no risk outliers.

1.2 REGULATORY CONCERNS

The Individual Plant Evaluation will be an important tool in satisfying the requirements of the NRC's Severe Accident Policy. The Severe Accident Policy

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Table 1.1-1

U.S. BWR PLANT SPECIFIC PRA OR PRA-LIKE STUDIES PUBLISHED FROM 1975 TO 1986

Plant	Operating License Year	Rating MWe	Plant Containment	A/E	Program Name[a]	Report	Report Year
Peach Bottom	1974	1065	BWR/4 MKI	Bechtel	RSS	NUREG 75/014 (WASH-1400)	1975
Peach Bottom	1974	1065	BWR/4 MKI	Bechtel	IDCOR	Task 21; Technical Summary Report	1984
Peach Bottom	1974	1065	BWR/4 MKI	Bechtel	PECo.	BWR IPE	1986
Grand Gulf	1982	1250	BWR/6 MKIII	Bechtel	RSSMAP	NUREG/CR-1659	1981
Grand Gulf	1982	1250	BWR/6 MKIII	Bechtel	IDCOR	Task 21; Technical Summary Report	1984
Browns Ferry 1	1974	1065	BWR/4 MKI	TVA	IREP	NUREG/CR-2801	1982
Millstone 1	1970	650	BWR/3 MKI	Ebasco	IREP	NUREG/CR-3085	1983
Millstone 1	1970	650	BWR/3 MKI	Ebasco	NUSCo.	Probabilistic Safety Study	1985
Limerick 1	1985	1065	BWR/4 MKII	Bechtel	PECo.	USNRC Docket Nos. 50-352, 50-353	1981
Big Rock Point	1962	71	BWR/2 DRY	Bechtel	Consumers	USNRC Docket No. 50-155	1981
Shoreham	-	820	BWR/4 MKII	S & W	LILCO	USNRC Docket No. 50-322	1983
Shoreham	-	820	BWR/4 MKII	S & W	LILCO	BWR IPE	1986
Gessar (Proprietary)		1100	BWR/6 MKIII	-	GE		1984
Susquehanna	1984	1065	BWR/4 MKII	Bechtel	PP&L	BWR IPE	1986

[a] RSS: Reactor Safety Study

RSSMAP: Reactor Safety Study Methodology Applications Program

IREP: Interim Reliability Evaluation Program statement (NUREG-1070) [1-3] also contains an additional objective, which identifies the avoidance of containment vulnerability as a part of the decision making process in severe accident evaluations. Recently, the NRC (Bernero) [1-4] has identified five items which may be generically applicable to BWRs, and may provide an increase in containment capability under severe accident conditions (particularly for plants with Mark I containment designs). These five items are as follows:

1. Prevent containment failure due to hydrogen combustion (using igniters, or by inerting containment and controlling oxygen ingress);
2. Assure reliable operation of containment spray during accident conditions, including station blackout;
3. Avert containment overpressure failure and uncontrolled radionuclide release by ensuring adequate wetwell venting capability and reliable operation of vent/purge valves;
4. Reduce the likelihood of containment failure from direct contact with molten core material; and
5. Provide emergency procedures and training to ensure operators can recognize and respond to severe accident conditions.

At this time, the NRC is considering the establishment of a generic severe accident policy for BWR containments which would embody these five elements. Furthermore, the NRC is considering acting on these issues at an accelerated pace, separate from the Individual Plant Evaluations. Bernero [1-4] indicated that the justification for this approach is that:

- The issues are considered to be generic.
- The actions to achieve the aforementioned objectives provide substantial assurance that containment failure can be prevented given a core melt accident.
- The actions appear to be relatively easy to incorporate.

The NRC's stated objective [1-4] for this proposed policy is to provide a minimum set of plant changes* which will provide substantial assurance that

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* It should be noted that these changes may be subject to the backfit rule [1-6].

early containment failure can be prevented given a core melt, and to implement these changes as quickly as possible.

1.3 APPLICABILITY AND APPROACH TO ADDRESS REGULATORY CONCERNS

The intent of these five items as stated by the NRC (Bernero) [1-4] is to provide a minimum, generic package which should be implemented by each plant. However, the five areas identified by the NRC have varying degrees of applicability to existing BWR containments. Table 1.3-1 provides an overview of the applicability of these five items to each BWR containment design. In some cases, this package may not provide the necessary and sufficient set of modifications that could be required to meet the objective of the NRC's proposed policy. On the other hand, it may not represent an appropriate expenditure of individual utility resources when viewed in the overall context of improving plant safety.

The purpose of this report is to provide IDCOR and the BWR Owners Group with technical information to support a logical approach to future regulatory interactions by utility decision makers on these BWR containment capability issues. The primary focus of this report is on BWR plants with Mark I containment designs since these plants have been the focus of more regulatory attention. In addition to addressing the items proposed by the NRC, this report also supports consideration of other plant or procedural changes that would offer significant benefits for little cost.

In evaluating the NRC's proposed items, several key questions must be answered about each element. These include:

- Does it reduce risk?
- Are there significant competing risks introduced?
- Does it make technical sense?
- Is there a lack of information which precludes a clear definition of the safety benefit?

Table 1.3-1

SUMMARY OF APPLICABLE ISSUES
TO
BWRs REGARDING SEVERE ACCIDENT ISSUES

NRC IDENTIFIED ITEMS	BWR CONTAINMENTS		
	MARK I	MARK II	MARK III
Hydrogen Control or Inerting System	Inerted (May be plant specific)	Inerted	Igniters (plant specific)
Including Blackout	Inerted	Inerted	Open
DW Spray High Reliability	Yes	Yes	NA
Including Blackout	Yes	Yes	NA
WW Venting	Yes	Yes	Yes
Blackout	Yes	Yes	Yes
ATWS	Yes	Yes	Yes
Debris Barrier	Yes	NA	NA
EPG for Severe Accident Management	Yes	Yes	Yes

- o Does it resolve all pertinent issues?

- o Is the resolution generic or plant specific?

- o What does it cost?

To answer these questions, an approach has been developed to examine BWR containment capability issues in a systematic fashion. The elements of this process and the section of this report where they are discussed are noted below:

- o Identify the functional requirements needed to mitigate accidents and which are associated with or affected by the containment issues (Section 2). The basic assumption underlying the approach is that there is a limited set of key safety functions which, if successfully performed automatically or through manual action, result in a "safe" condition for the plant. Therefore, before any evaluation of individual plants and plant systems can be performed, those "critical safety functions" which are associated with containment response and which may be impacted by the regulatory issues must be identified.

- o Identify the existing BWR mitigation capability (Section 2). Each safety function identified during the previously defined task should have associated with it at least one system or procedure to be used to satisfy the function and mitigate the effects of any accident. Coupling the existing system or systems, and/or procedures, with the functional requirements enables an assessment to be made as to the adequacy of the individual plant's mitigative capability.

- o Identify potential options for addressing the NRC's concerns about existing BWR containment mitigation capability (Section 3). As with most issues there may be more than one option for successful resolution of the concern. These options may have varying degrees of success when applied to the underlying safety issues. By systematically identifying and studying these options, prudent choices as to the best solution in terms of safety, cost, and other factors can be made.

- o Identify the feasibility of the proposed NRC modifications or procedural enhancements (Section 4). This section addresses two issues related to the feasibility of implementing the options:

  - First, the current plant configurations which may influence the implementation.

  - Secondly, the analytic support for the determination of positive or negative safety impacts associated with the proposed changes.

- Identify potential safety benefits and competing risks of the proposed changes (Section 5). A "safety evaluation" of the identified options in terms of previously identified BWR dominant severe accident sequences is performed. This evaluation provides an indication of the potential benefits which may be associated with each option. Negative impacts, such as decreased safety in other parts of the plant's overall design brought about by changes which address the specific containment concerns may also occur.

- Estimate the costs of the proposed changes (Section 6). The reality of the situation facing nuclear power plants today is that they must operate within limited, real budgets. Therefore, a very important factor in the decision-making process is the cost of any proposed change. Estimated costs of identified options are provided.

The approach is structured to identify logically the benefits and competing risks associated with the proposed changes for use by decision makers. PRA has been utilized as a framework for evaluation of the technical issues. Figure 1.3-1 identifies the pieces of a probabilistic risk assessment which evaluates the risk to the public. For this evaluation, the first two parts of a PRA evaluation are examined in a qualitative manner to ascertain whether the suggested changes could provide a beneficial risk reduction. Representative postulated accident sequence classes are evaluated qualitatively to establish the mitigation capability over the spectrum of core melt accident sequences.

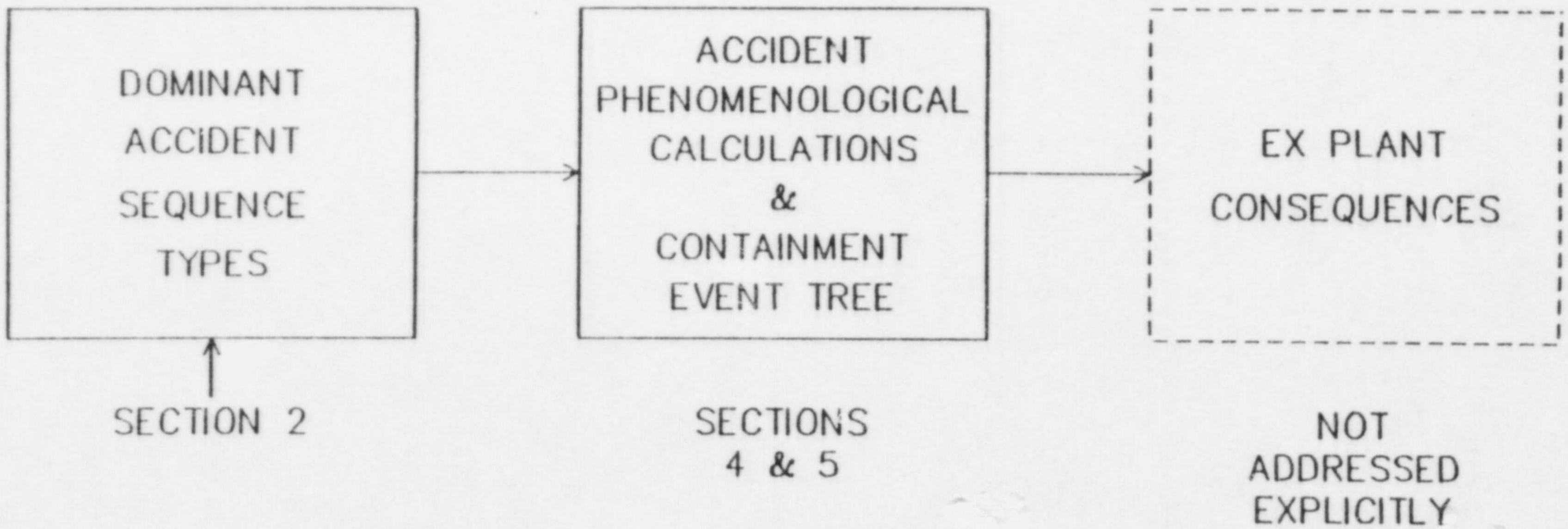

Figure 1.3-1 PRA Framework For Issue Evaluation

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References

[1-1] Technology for Energy Corp., Nuclear Power Plant Response to Severe Accidents, IDCOR Technical Summary Report, November 1984.

[1-2] Safety Goals for Nuclear Power Plant Operation, NUREG-0880, Revision 1, For Comment, May 1983.

[1-3] NRC Policy on Future Reactor Designs: Decisions on Severe Accident Issues in Nuclear Power Plants, NUREG-1070 (Draft), April 1984.

[1-4] Meeting; Presentation by R. Bernero (NRC) to BWR Utilities and Public regarding additional plant changes which might be requested by NRC for BWRs.

[1-5] Indian Point Probabilistic Safety Study, 1982.

[1-6] NRC Backfit rule.

Section 2
CURRENT CONTAINMENT DESIGN FEATURES AND ACCIDENT MITIGATION CAPABILITY

This section summarizes the following BWR related features that place into context the five suggested changes proposed by the NRC:

- o Containment functional description (Section 2.1)
- o Severe Accident Sequence Event Tree End State Types (2.2)
- o Dominant BWR Severe Accident Sequences
- o Containment Event Tree

2.1 EXISTING PLANT CONFIGURATIONS: GENERAL DISCUSSION, CONTAINMENT EFFECTIVENESS

This section contains a discussion of existing BWR plant configurations for containing the radionuclides possibly released following postulated accidents. The various design and operating characteristics are included along with some discussion of the estimated effectiveness of various systems and design features in mitigating radionuclide release. The use of probabilistic risk assessment (PRA) techniques as tools in the evaluation of containment responses is also described in this section.

Containments are used to prevent or mitigate the radionuclide releases to the environment. As such, the containment consists of not only the primary containment which has been designed for a specific set of accident sequences but also all the associated passive and active systems which allow for the effective mitigation of radionuclides in the event of a severe accident.

Probabilistic risk assessment (PRA) techniques are useful as tools for logically displaying and evaluating the integrated responses of plant system to various conditions. As such, PRAs have been used to evaluate containment system responses following postulated severe accidents.

Previous PRAs [2-1, 2-2] and IDCOR [2-3] have demonstrated that BWR containments are capable of preventing radionuclide releases to the environment under severe accident conditions. This capability derives from a series of multiple barriers. Figures 2.1-1 through 2.1-3 provide pictures and schematics of the various types of BWR Containments: Mark I, II, and III. Although much of the discussion also applies to the other BWR containments, the remainder of this section will focus attention on the Mark I containment.

BWR Mark I containments encompass a multitude of barriers, both active and passive, to prevent or reduce any radionuclide releases to the environment.

These barriers include the following:

Passive Mitigation

- Primary containment
- Secondary containment: Reactor Building

Active Mitigation

- Post core melt coolant injection
- Post core melt containment spray
- Containment heat removal, including containment venting from the wetwell
- Suppression pool scrubbing through venting

All of these features provide real barriers or fission product retention capability to minimize any health effects to the public. While WASH-1400 [2-4] was a quantum jump in thinking regarding the protection of the public against severe core damage accidents, one of the failings of WASH-1400 was the excessive conservatism used in modeling the BWR mitigation capability of the secondary containment and the active mitigation measures. The IDCOR program has built upon the foundation laid by later BWR PRAs, which have identified the considerable effectiveness of post core damage BWR containment mitigation capability, to present a more realistic evaluation.

The following discussion conceptually identifies the relationship of BWR containment systems with their capability to mitigate severe accidents.

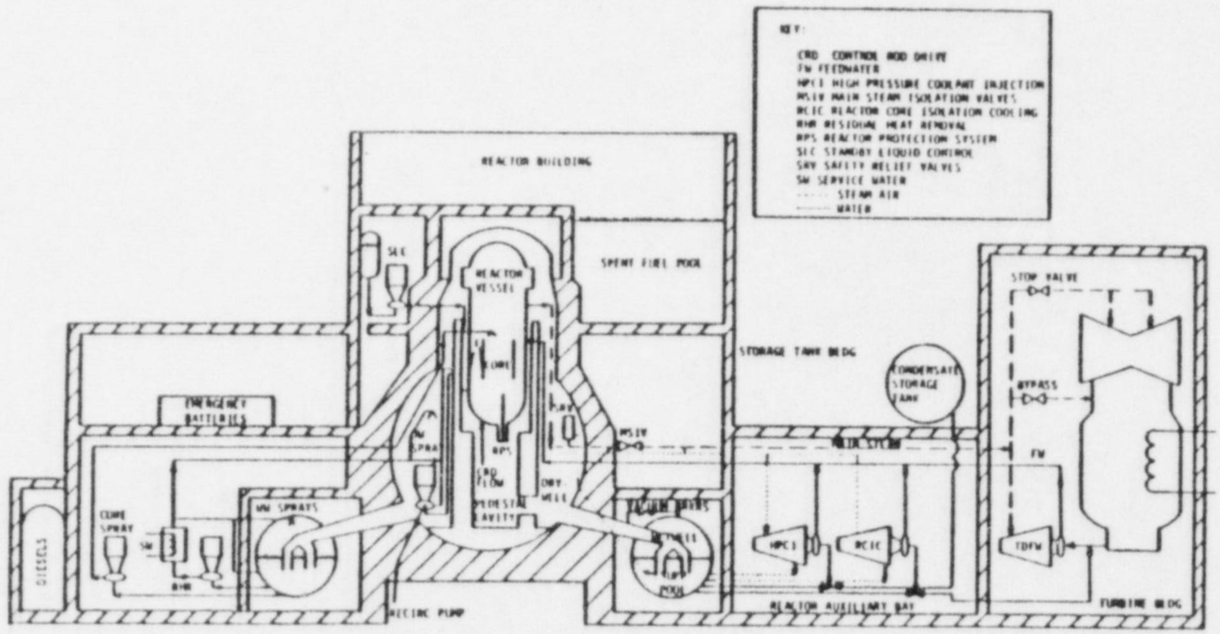

Figure 2.1-1a Peach Bottom System Features.

1. Secondary Containment Building
2. Refueling Floor
3. Drywell
4. Reactor Vessel
5. Pedestal
6. Wetwell/Torus
7. Downcomer
8. Concrete Shielding

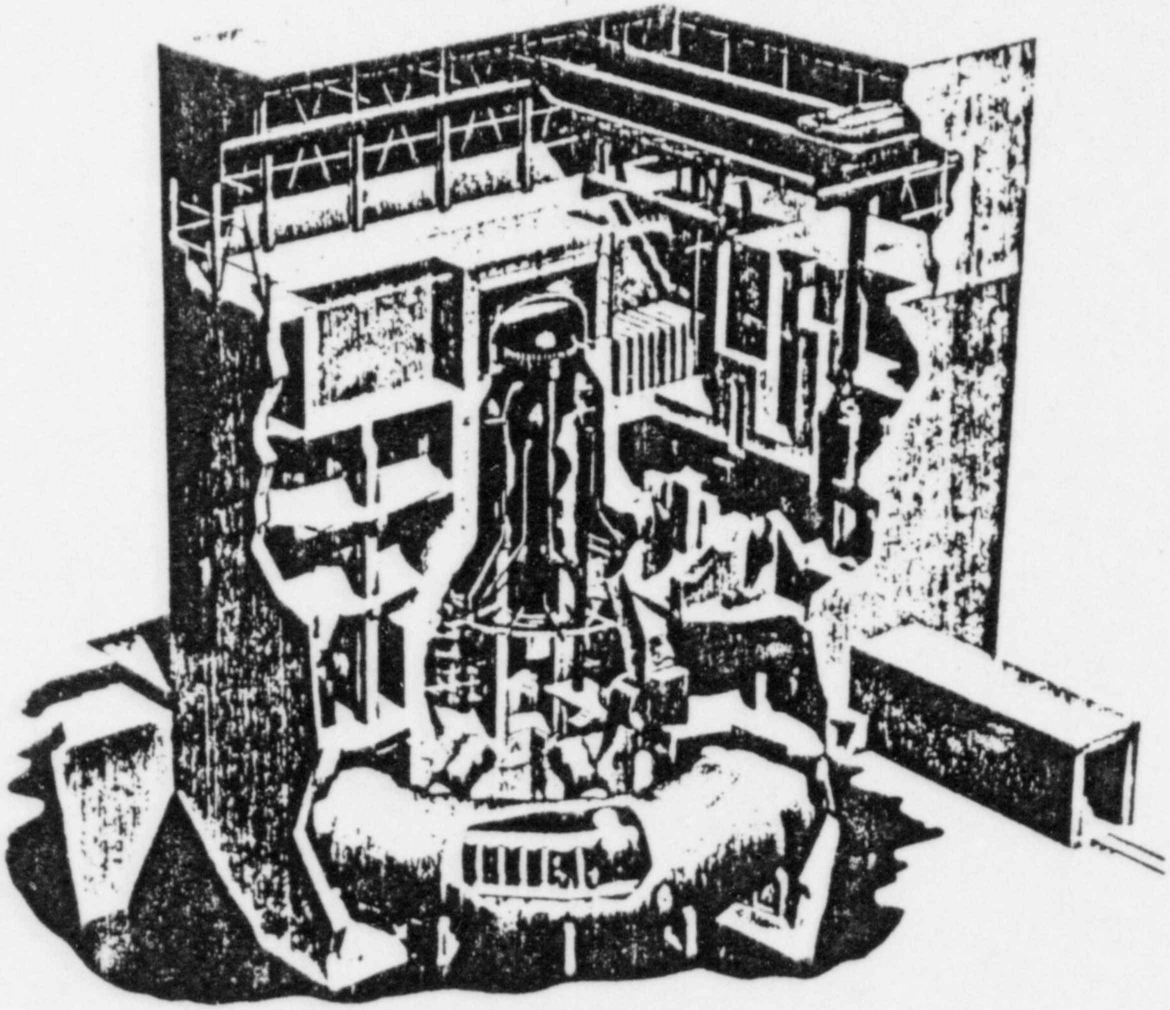

Figure 2.1-1b Boiling Water Reactor Mark I Containment.

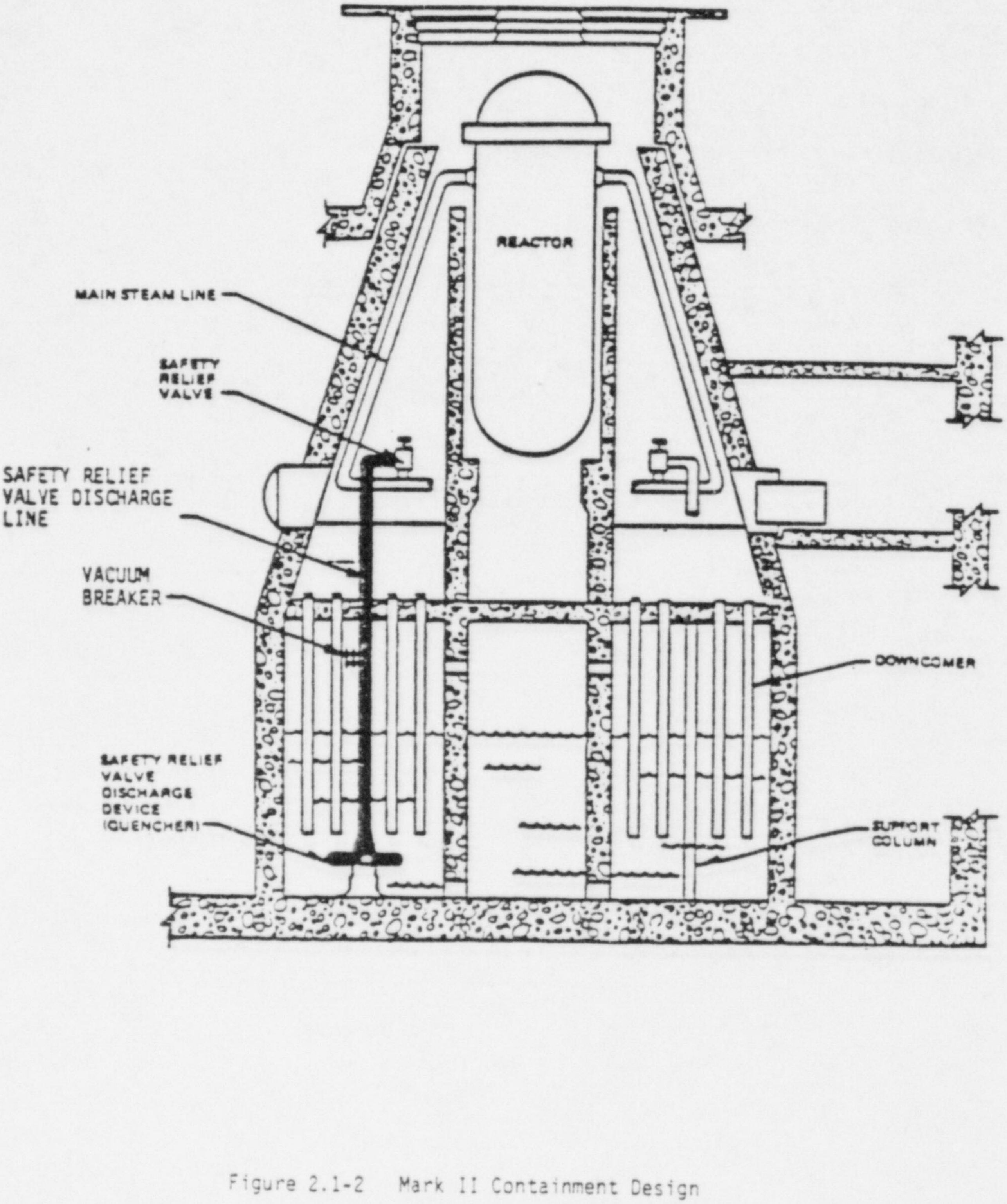

Figure 2.1-2 Mark II Containment Design

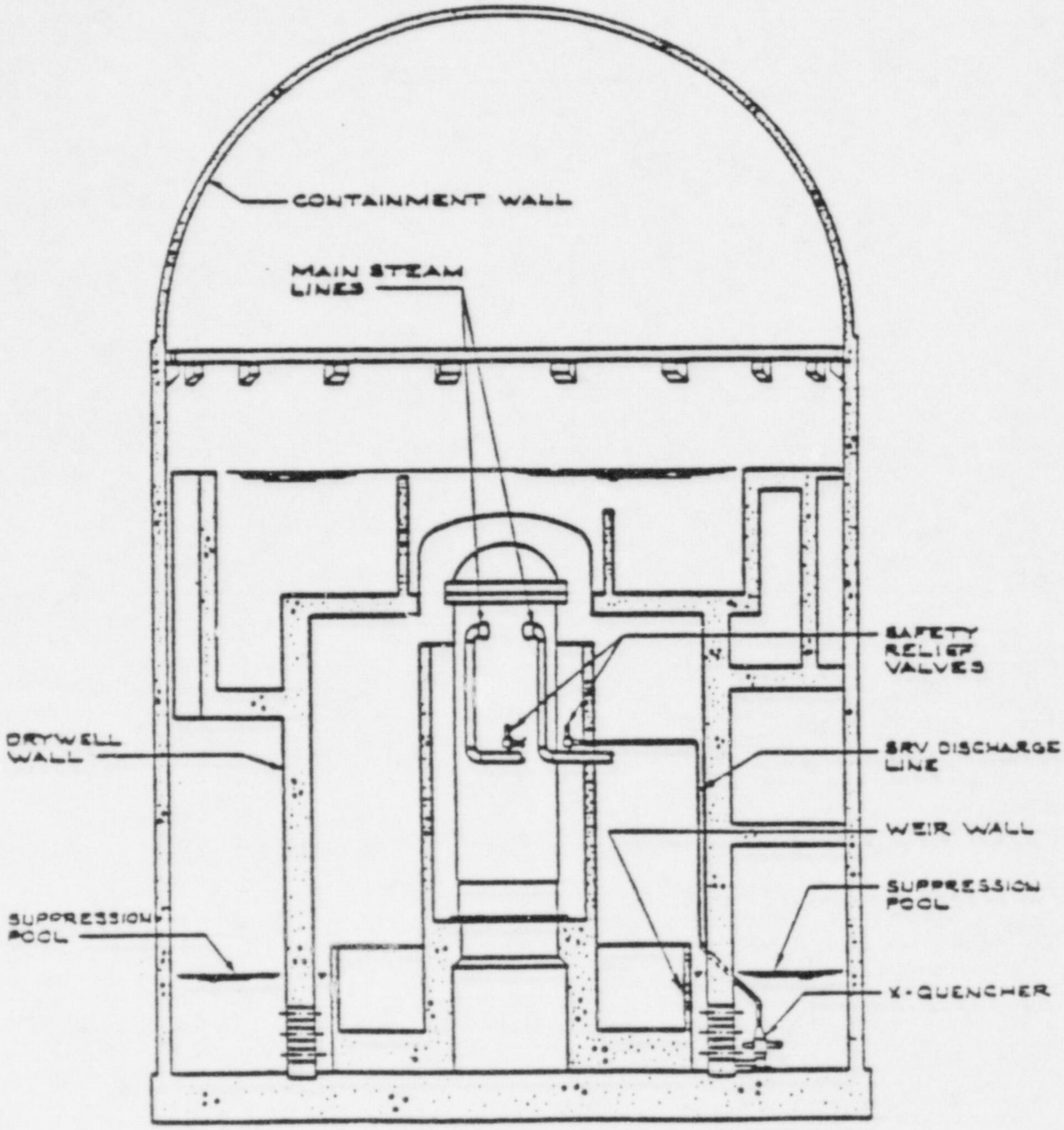

Figure 2.1-3 Mark III Containment Design

Primary Containment

BWR primary containments are used to both prevent severe accidents and mitigate severe accident consequences. The design of the BWR primary pressure suppression containment makes use of large quantities of water inside containment to provide a passive heat sink for heat that may be generated following a postulated accident. The principal design basis accident governing many of the design features of the containment is the "worst case" design basis accident (DBA) loss of coolant accident (LOCA). This double ended shear of a primary system pipe is explicitly included in the deterministic calculations supporting the design basis of the containment. BWR containment is therefore explicitly designed to function adequately with coolant injection occurring to the reactor pressure vessel (RPV), and containment heat removal occurring via the residual heat removal (RHR) system despite the possibility of a large break in the primary system piping. Large margin exists beyond this design basis for the containment on a realistic basis.

The BWR primary containment has features which provide:

- High internal pressure capability (56 psig design and greater than 130 psia (Mark I) ultimate). The larger volume Mark III containments have a lower design and ultimate pressure - Passive.
- Suppression of H_2 burning through inerting of the primary containment - Passive.
- Scrubbing of potential fission products through the use of the large containment water inventory - Passive or active mode.
- Long term containment temperature and pressure control through RHR and internal or external containment water injection sources - Active.
- Alternative containment heat removal methods also exist (including venting) - Active.

The BWR primary containment has a number of fundamental response requirements to ensure that radionuclides are effectively mitigated even for severe accident conditions. These functional response features which may be fulfilled passively or through active measures include the following:

- Isolation
- Pressure control
- Temperature control
- Prevention of direct containment contact with molten material
- Prevention of explosive or energetic reactions
- Radionuclide scrubbing
- Fission product scavenging.

These functional elements are described in more detail in the discussion of the containment event tree development (Section 2.4) along with the interrelationships with other mitigation features such as the reactor building and containment injection.

Secondary Containment

The BWR secondary containment (reactor building) provides an effective secondary radionuclide barrier to trap fission products if they escape from the primary containment barrier. The secondary containment is a massive structure surrounding the Mark I and II containments. The secondary containment atmosphere is exhausted to the stand-by gas treatment system (SGTS). The SGTS provides a filtered release to the environment. Therefore, even for severe accidents which might cause leakage from the primary containment, there remains the additional filtration system of SGTS to further reduce the radionuclides released to the environment. In addition, the secondary containment structure has a large interior volume (2 million ft^3) which will be relatively cold even in the event of a severe accident. This large volume represents a large site for scavenging and depositing radionuclides that might leak from the primary containment. Figures 2.1-4 and 2.1-5 provide simplified pictures of the reactor building mitigation features.

Post Core Melt Coolant Injection

Post core melt coolant injection to the containment provides a method for:

BWR PRIMARY CONTAINMENT IS PROVIDED BY THE DRYWELL AND WETWELL; SECONDARY CONTAINMENT IS PROVIDED BY THE SURROUNDING STRUCTURE

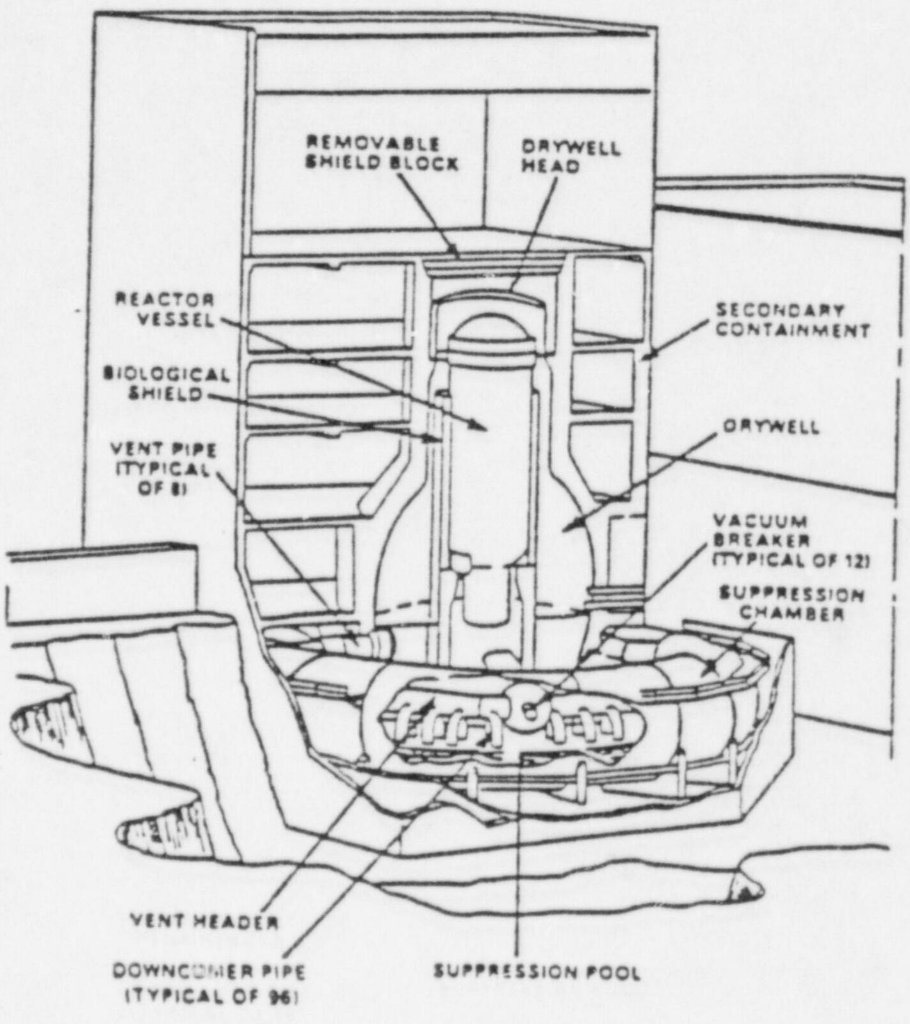

Figure 2.1-5 Primary and Secondary Containment for BWR Mark I and II Designs

- Temperature control
- Pressure control
- Debris coolability: limitation of debris movement in the containment
- Scavenging of fission products from debris or containment atmosphere.

The BWR plant has several coolant injection systems which are capable of supplying water to the reactor from water sources internal or external to the primary containment inventory. The systems include:

- RHR
- Low pressure core spray (LPCS)
- Condensate pumps
- Control rod drives (CRD)
- Condensate transfer pumps
- Other BOP systems such as fire pumps (diesel driven fire pumps).

These water injection sources may be useful in satisfying all of the above containment functions.

Core melt accidents are postulated for a large number of reasons. From a risk perspective two of the more likely causes of a core melt accident involve:

- o Loss of high pressure coolant makeup and failure to depressurize the RPV
- o Failure of adequate AC power.

These two examples involve cases in which the coolant injection sources can be made effective after RPV melt-through due to either depressurization, recovery of AC power, or completion of required operator actions, e.g., within the time following core melt but before containment failure (e.g., 6 hours). Therefore, for certain accident sequences the likelihood of containment coolant injection (such as drywell sprays) is very high despite a core melt occurring, e.g., reliabilities of .5 to .99 are estimated depending upon the accident sequence.

Post Core Melt Containment Spray

The BWR containments for Mark I and II both have the capability to spray the drywell from a number of sources with a variety of pumps. The containment drywell sprays provide a means for both pressure and temperature control. When RHR pumps are available, the containment sprays can be manually initiated from the control room in a very short period of time, i.e., less than 15 minutes. The operator is trained for and has procedures to perform these actions. While this type of response is expected for all BWRs when the RHR pumps are available, less uniform condition may exist when the RHR pumps are disabled. For example, many, but not all, plants have the capability to spray the drywell with a connection to service water and/or the diesel fire pump through a cross tie into the RHR system, i.e. the "ultimate cooling connection." This capability could afford an advantage in maintaining long term containment te. ,.rature control and debris coolability given a long duration severe accident condition. Accident sequences for which this backup means of containment cooling could be made available include the following:

- Station blackout where no AC power to the RHR pumps is available
- Flooding of the RHR pump room
- Loss of access to the water in the torus
- Anticipated transient without scram (ATWS) events.

At present the BWROG emergency procedure guidelines (EPGs) [2-5] are receiving a "final" revision to include many of the insights from severe accident analyses. Although the "final" EPGs may mention the use of non-safety systems, previous EPG revisions and plant implementations have not uniformly suggested, recommended, or directed operator actions to spray the drywell under these severe accident conditions. This inhibition is generally related to other safety or operational concerns.

Containment Heat Removal

Containment heat removal can be accomplished through a diverse and redundant set of methods including:

Safety Related

- RHR (see discussion below)

Non-Safety Related

- Main condenser
- Isolation condenser
- Reactor water clean up (RWCU)
- Spent fuel pool heat exchanger
- Venting (see discussion below)
- Others

The principal method of containment heat removal under accident conditions is through the use of RHR in the suppression pool cooling mode. This mitigation function can be effective in preventing containment overpressure challenges either prior to core melt or during core melt. The RHR system is an extremely flexible and redundant system.

Table 2.1-1 summarizes the RHR heat removal capabilities for most of the BWR Mark I containments.

Containment Venting

Containment venting has been identified as a potential operator action for certain postulated accident sequences which are far beyond the plant licensing design basis. The BWROG EPGs specify that containment venting should be used as a last resort operator action in the event of rising containment pressure. The action is taken with the purpose of preventing an uncontrolled breach of containment, while maintaining the containment function as a fission product barrier through deposition and scrubbing.

The methods to accomplish containment venting and the containment pressure at which one would initiate such a procedure are still under development. As with most engineering decisions there are a number of trade-offs which need to be made to establish a rational procedure. These trade-offs are summarized in Section 3 under Venting Options.

2.2 SEVERE ACCIDENT SEQUENCE EVENT TREE END STATE TYPES

A large number of accident sequences can be postulated that could lead to core damage and potentially challenge containment. These sequences can result in different types of unique challenges to containment. These types of challenges

Table 2.1-1

BWR RHR DESIGN CAPACITIES PER LOOP*

$$Q = K\,(T_{pool} - T_{sw})$$

Where T_{pool} = Suppression Pool Temperature, °F

T_{SW} = Service Water Inlet Temperature, °F

	K (Btu/sec °F)	$T_{service\ water}$ (°F) Max.	Min.
Browns Ferry	239.6	95	33
Dresden 3	416.7	105	85
Duane Arnold	145.2	95	NA
Enrico Fermi II	321	89	40
Fitzpatrick	142.6	77	35
Hatch I	213.9	95	32.4
Hatch II	213.9	95	NA
Dresden II	416.7	105	85
Monticello	199.6	86	37
Brunswick I	213.9	86	35
Brunswick II	213.9	85	35
Peach Bottom II	271.6	88	NA
Cooper	228.9	84	NA
Peach Bottom III	271.6	88	NA
Pilgrim	218.8	95	NA
Millstone Pt. 1	123.7	75	32
Quad Cities I	404.7	105	85
Quad Cities II	404.7	105	85
Vermont Yankee	200	95	32
Browns Ferry II	239.6	95	33
Hope Creek II	288.9	95	65

*Information provided by GE.

to containment have been characterized on a realistic basis through the IDCOR program. In addition, previous BWR PRAs [2-1, 2-2] have also categorized these containment challenges into a finite, discrete group of accident sequences. This section provides an overview of these BWR accident sequence classifications.

The types and frequencies of postulated accident sequences in a nuclear power plant cover a broad spectrum. A PRA utilizes system level event trees to model these accident sequences. The end states of these event tree are either a safe shutdown condition, or a condition in which the core is considered vulnerable to damage. To limit this spectrum of end states to a manageable number of sequences with similar characteristics (e.g. similar initiator and system failures) sequences are generally grouped together (i.e. binned). Figure 2.2-1 illustrates schematically this grouping process as practiced in many PRAs and as performed in the IPE methodology. As can be seen, this grouping is performed at several levels. The consolidation occurs both in the structuring of the model such that some of the fine details are incorporated together with other sequence variations; and in a post-processing step of summing similar sequences together.

Examples of sequence-specific detail that can often be consolidated include the exact timing, precise system failure modes (i.e. cut sets), and the operator's time varying response. In addition to this consolidation, it has been noted in past PRAs that the accident sequence end states leading to core melt can be further collapsed. This grouping or binning of similar core melt sequences is generally performed based upon the following:

- o Decay heat level
- o Integrity of the containment
- o Relative timing of the core melt.

In addition to these parameters, the binning can be further subdivided based upon the conditional unavailability of key frontline support systems in each of these end states.

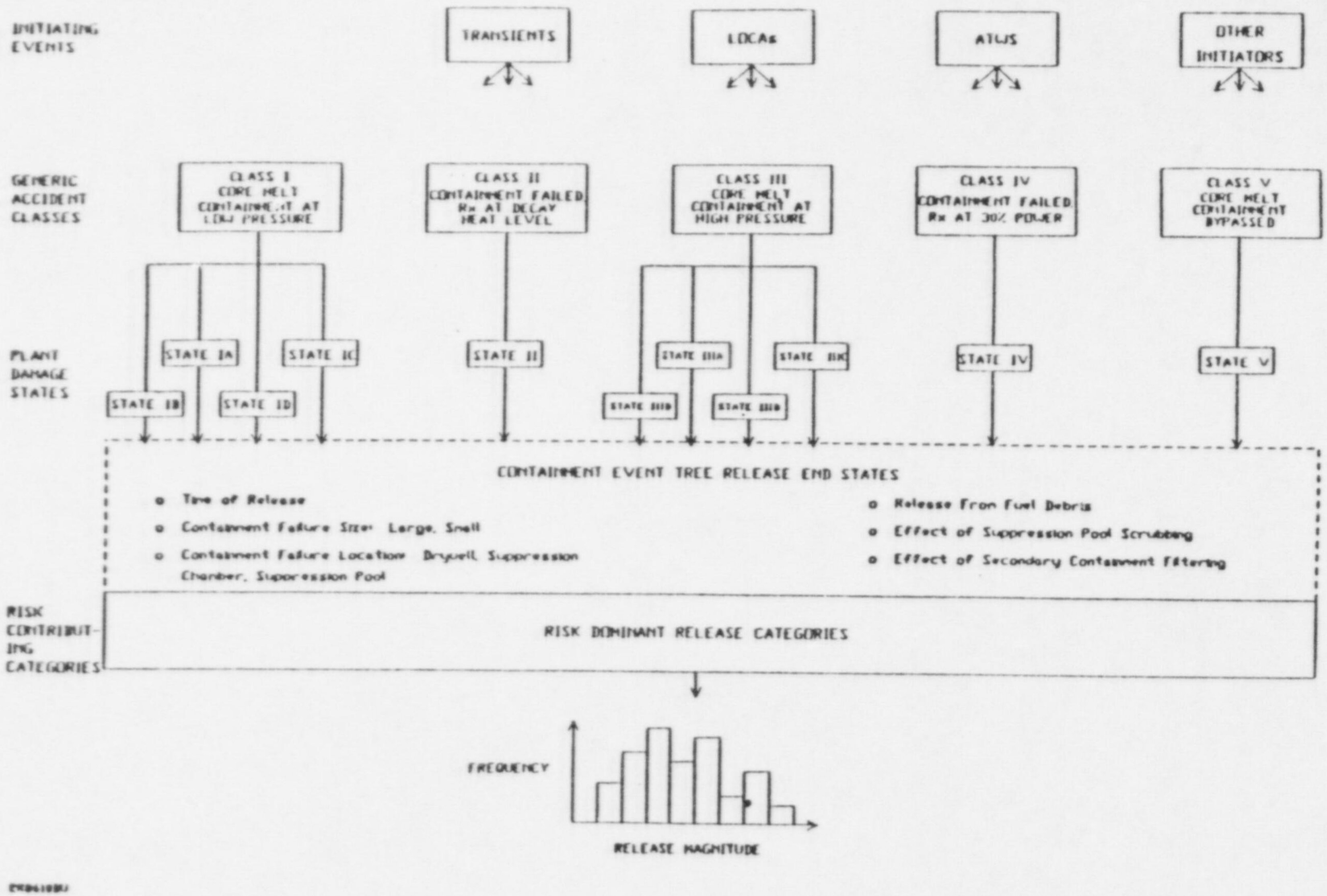

Figure 2.2-1 PRA Accident Sequence Binning Scheme

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Based on the above, an event sequence classification into five accident sequence classes is performed. These five representative accident sequence classes are described in Table 2.2-1. The sequences included in each of the five classes are based upon the above accident characteristics. The description of classes is presented here to introduce the terminology to be used in characterizing the basic types of challenges to containment.

The reactor pressure vessel condition and containment condition for each of these classes is noted below:

CLASS	RPV CONDITION	CONTAINMENT CONDITION
I	Loss of Effective Coolant Inventory	Intact
II	Loss of Effective Containment Heat Removal	Breached
III	LOCA	Intact
IV	Failure of Effective Reactivity Control	Breached
V	LOCA Outside Containment	Breached (Bypassed)

References 2-1 and 2-2 have identified that the highest frequency of postulated core melt accidents belong to Class I, in which the BWR containment is intact. The other accident classes, while having potentially more severe impacts on primary containment, are of lower calculated frequency.

In assessing the ability of the containment and other plant systems to prevent or mitigate radionuclide release, it is often desirable to further subdivide these general categories. In the <u>second level</u> classification process, the similar accident sequences grouped within each accident class are further divided into several subclasses (e.g., Classes IA,B,C,D) such that the potential for systems' recovery in the short term, subsequent to a core/containment vulnerable condition, or in the long term, subsequent to a core melt condition and possibly containment failure, can be modeled. The interdependencies which exist between plant system operation and the core melt and radionuclide release

Table 2.2-1
ACCIDENT SEQUENCE CLASSIFICATION

ACCIDENT CLASS DESIGNATOR	PHYSICAL BASIS FOR CLASSIFICATION	SYSTEM LEVEL CONTRIBUTING EVENT SEQUENCE	REPRESENTATIVE SEQUENCE FOR CLASS
Class I (C_1)	Fuel will melt rapidly if cooling systems are not recovered; containment intact at core melt and at initially low pressure; highest probability release pathway is from the vessel to the suppression pool	Transients involving loss of inventory makeup; small LOCA events involving SRV actuation with loss of inventory makeup; transients involving loss of scram function and inability to provide sufficient coolant makeup	Transient with coolant makeup unavailable
Class II (C_2)	Fuel will melt relatively slowly due to lower decay heat level if cooling systems are not recovered; containment is breached prior to core melt; highest probability release pathway is from the vessel to the suppression pool	Transients or LOCAs involving loss of containment heat removal; in advertent SRV opening accidents with inadequate heat removal capability	Transient with loss of residual heat removal
Class III (C_3)	Fuel will melt rapidly if cooling systems are not recovered; containment intact at core melt, but at initially high internal pressure; involves a release from the vessel to the drywell	Large and medium LOCAs with insufficient coolant makeup; small and medium LOCAs with failure of the SRVs to actuate and long-term loss of inventory makeup; RPV failures with insufficient coolant makeup	Large LOCA with loss of low pressure ECCS
Class IV (C_4)	Fuel will melt rapidly if cooling systems are not recovered; containment fails prior to core melt due to overpressure; highest probability release pathway is from the vessel to the suppression pool	Transients involving loss of scram function and loss of containment heat removal or all reactivity control; transients with loss of scram function followed by rapid depressurization	Transient with failure of RPS and failure of SLCS
Class V (C_5)	Fuel will melt rapidly if cooling systems are not recovered; containment failed from initiation of accident due to equipment failure; involves a release pathway from the vessel which bypasses the containment	LOCAs outside containment with insufficient coolant makeup to core; interfacing system LOCAs with insufficient coolant makeup	LOCA in main steam lines with failure of MSIV closure and loss of ECCS

phenomena are adequately represented in the release frequencies through a binning process involving these subclasses. In addition to the better definition of operator actions and success probability, the subclasses also allow a refinement in the consequence calculation for radionuclide release, timing, and transport. Table 2.2-2 provides a description of the subclasses which have been established.

Containment event trees can be developed which make the transition from system event tree plant damage states to potential radionuclide release end states. The containment event tree paths and quantification are based upon the entry conditions derived from the systematic event tree plant damage states (i.e. Classes IA, B, C, D through Class V). Appendix A and Section 2.4 provide a description of the simplified containment event trees which can be used to display the key containment phenomena and the mitigation system capability.

2.3 DOMINANT ACCIDENT SEQUENCES

The accident sequences which control the calculated core melt frequency or the public risk may be referred to as the dominant accident sequences.

Spectrum of Accident Sequences to be Considered

Published BWR PRAs have identified that there may be a spectrum of potential contributors to core melt or containment challenge that can arise for a wide variety of reasons. In addition, sufficient analysis has been done to indicate that these sequences are highly uncertain and therefore they may be more or less important on an absolute scale <u>and</u> relative to each other depending upon assumptions, training, equipment response, and other items that have limited modeling sophistication. This uncertainty means that we can neither dismiss portions of the spectrum from consideration or emphasize a portion of the spectrum to the exclusion of other sequence types. This is particularly true when trying to assess the benefits and competing risks associated with a change or a plant feature, i.e. we cannot be too narrow in our focus.

Table 2.2-2

SUMMARY OF THE CORE VULNERABLE ACCIDENT SEQUENCE SUBCLASSES (PLANT DAMAGE BINS)

ACCIDENT CLASS DESIGNATOR	SUBCLASS	DEFINITION	EXAMPLE
CLASS I	A	Accident Sequences Involving Loss of Inventory Makeup in which the Reactor Pressure Remains High	TQUX
	B	Accident Sequences Involving a Loss of Off-Site Power and Loss of Coolant Inventory Makeup	T_EQUV
	C	Accident Sequences Involving a Loss of Coolant Inventory Induced by an ATWS Sequence	$T_T{}'C_MQU$
	D	Accident Sequences Involving a Loss of Coolant Inventory Makeup in which Reactor Pressure has been Successfully Reduced to 200 psi.; Accident Sequences Initiated by Common Mode Failures Disabling Multiple Systems (ECCS) Leading to Loss of Coolant Inventory Makeup	TQUV
	E	Accident Sequences caused by Common Mode Failures which Result in Multiple Frontline System Failures with the Reactor at High Pressure	$T_DDC2DC3$
CLASS II	-	Accident Sequences Involving a Loss of Containment Heat Removal	TW
CLASS III	A	Accident Sequences Leading to Core Vulnerable Conditions Initiated by Vessel Rupture where the Containment Integrity is not Breached in the Initial Time Phase of the Accident	R
	B	Accident Sequences Initiated or Resulting in Small or Medium LOCAs for Which the Reactor Cannot be Depressurized	S_1QUX
	C	Accident Sequences Initiated or Resulting in Medium or Large LOCAs for which the Reactor is at Low Pressure	AQUV
	D	Accident Sequences which are Initiated by a LOCA or RPV Failure and for which the Vapor Suppression System in inadequate, Challenging the Containment Integrity	AD
CLASS IV	-	Accident Sequences Involving Failure to Insert Negative Reactivity Leading to a Containment Vulnerable Condition Due to High Containment Pressure	$T_TC_MC_2$
CLASS V	-	Unisolated LOCA Outside Containment	

The spectrum of accident sequences can sometimes be bounded by a single surrogate accident sequence (i.e. dominant sequence) for the purposes of limiting the deterministic calculation that must be performed. Because of this and the fact that even the probabilistic treatment tends not to look at partial failures, results in identifying a discrete set of sequences which to some extent are bounding. This philosophical problem with discretizing the continuous spectrum of severe accident sequences leads to a certain amount of oversimplification in the PRA representation. The challenge is to avoid allowing this oversimplistic representation to adversely impact the decision making process by eliminating or removing from consideration risk significant sequences because of "low probability" arguments or slight changes in deterministic calculations.

For an example, consider the issue of venting. The bounding deterministic calculations may show marginal or inadequate capability for some narrow set of predefined boundary conditions such as the worst case ATWS. In reality, if this "worst case" ATWS were to occur, there would likely be competing effects such as some rod insertion, some partial successes, and late boron injection. Therefore the venting scheme may provide just the extra margin required to hold containment together until reactivity control can be regained. The level of sophistication of PRA is such that it does not support excluding safety improvements because they will do <u>no</u> good. It supports identifying qualitatively those engineering insights that may be important in certain sequences (even if the importance cannot be adequately quantified) and supports the identification of improvements that can be made to enhance those insights.

Table 2.3-1 summarizes the dominant accident sequences which are reported in the published BWR PRAs. The sequence identification scheme (alphanumeric identifiers of failure events) follows the terminology used in the IDCOR BWR IPE methodology. In addition, the dominant sequences are assigned to the accident sequence class which most closely represents the physical processes that would occur in the primary system and the containment. These classes are described above in Section 2.2.

Table 2.3-1

COMPARISON OF THE DOMINANT ACCIDENT SEQUENCES
FROM PUBLISHED PROBABILISTIC ANALYSES

	PEACH BOTTOM			GRAND GULF		BROWNS FERRY	MILLSTONE 1	
	WASH-1400	IDCOR TSR	IPE UTILITY	IDCOR TSR	RSSMAP	IREP	IREP	UTILITY
CLASS I	TC_MC_2 TQUV	T_EQUV	T_EQUV T_IQUV T_IQUX T_MQUX $T_{MS}QUX$ T_CQUX	T_EQUV	T_EQUV TPQUV T_EPQUV	T_M, T_CWQUV $T_M, T_CC_MC_2QUV$ T_EWQUV T_MPWQUV T_TC_MRQUV T_EPWQUV	T_EPQX T_EPQUV T_EPQUV T_EQUV $T_EPQUX(W)$ T_EQUV T_EQUX	TPUV T_EQUV T_EQUX T_MQUX,UV
CLASS II	TW S_2W	TW	T_CW $T_{SW}W$	T_MW	TQW T_EQW S_1W TPW T_EPW	--	T_EQPW T_EW	S_2W T_EW T_MT_CW $T_{SW}W$,PW T_CPW AW
CLASS III	AV	S_1QUV AQUV	S_1QUV	AV		--		S_2, S_1AQUV
CLASS IV		TC_MC_2	$T_TC_MC_2$ $T_IC_MC_2$ $T_FC_MC_2$ $T_MC_MC_2$	T_MC	TC_MC_2	--	T_M, T_CC	$T_MC_MC_2$
CLASS V			$A_{OUT}QUV$			--		

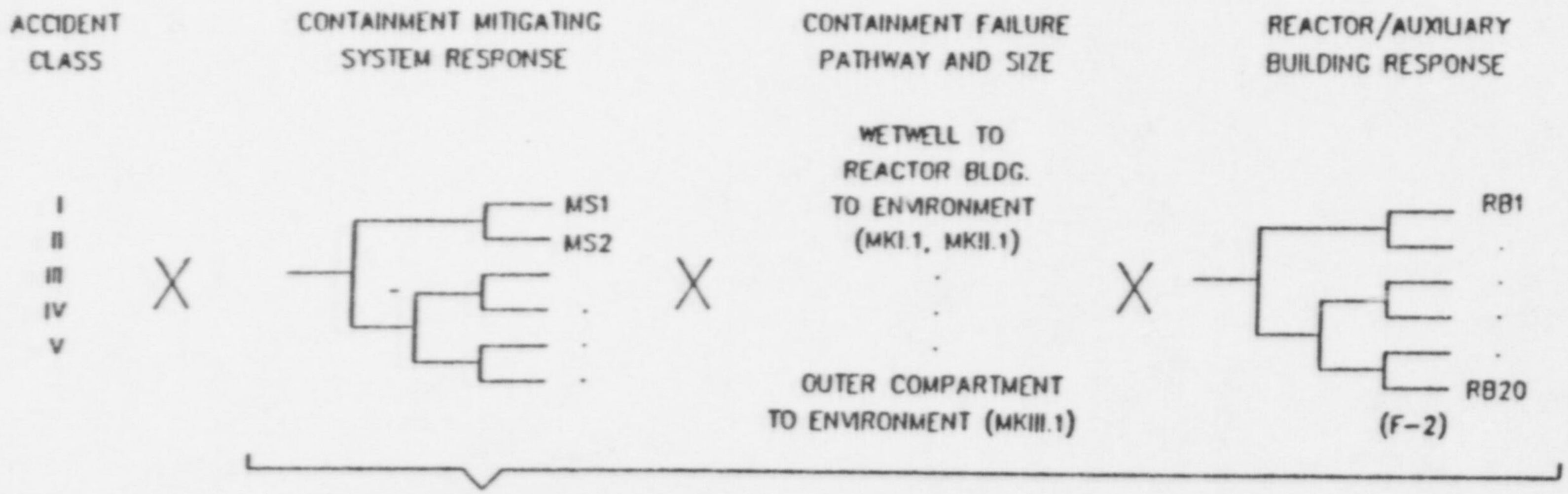

Figure 2.4-1 Overview of the Containment Event Tree Evaluation Process for Radionuclide Release Determination

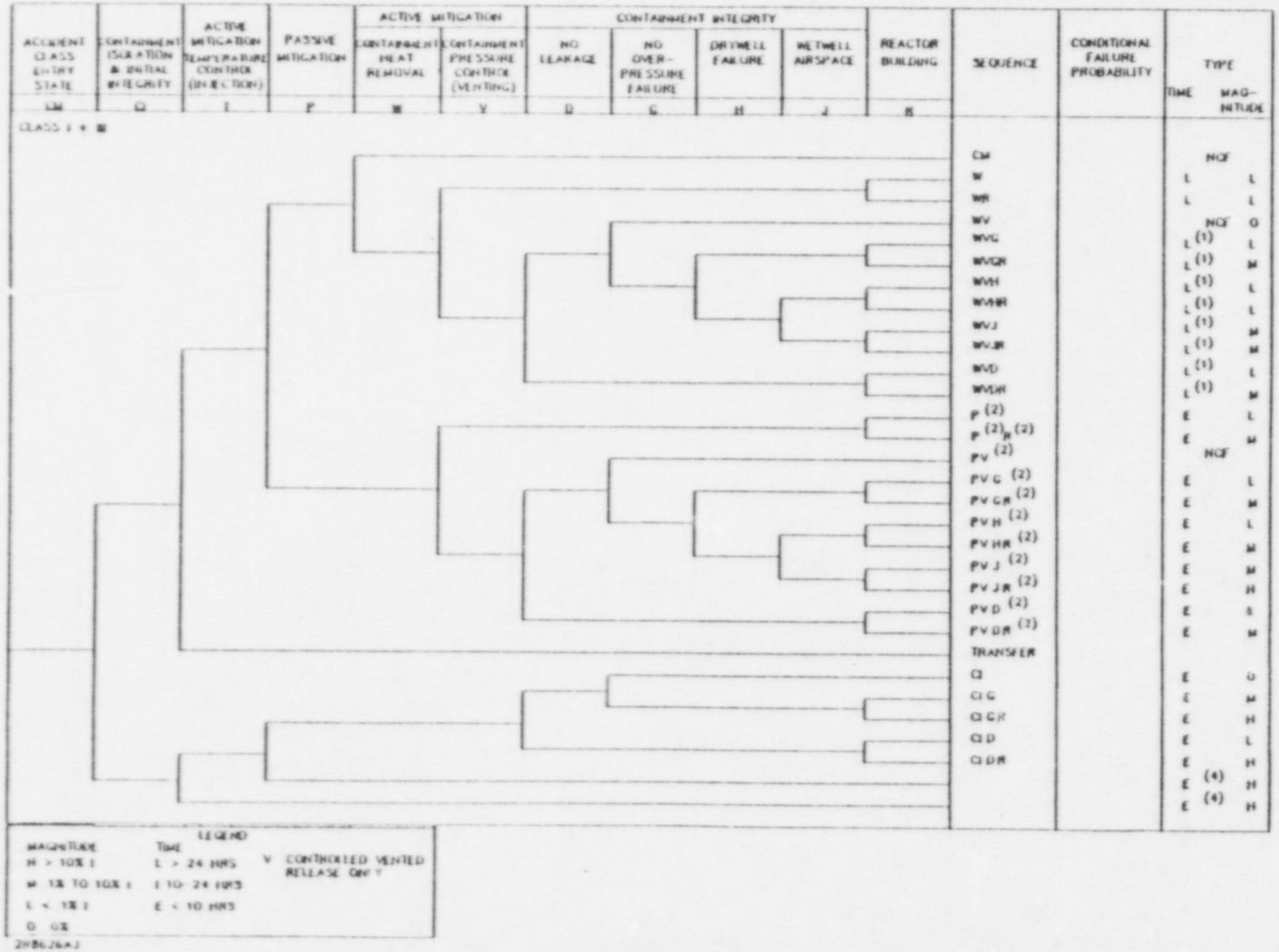

Figure 2.4-2 Containment Event Tree for Classes I and III

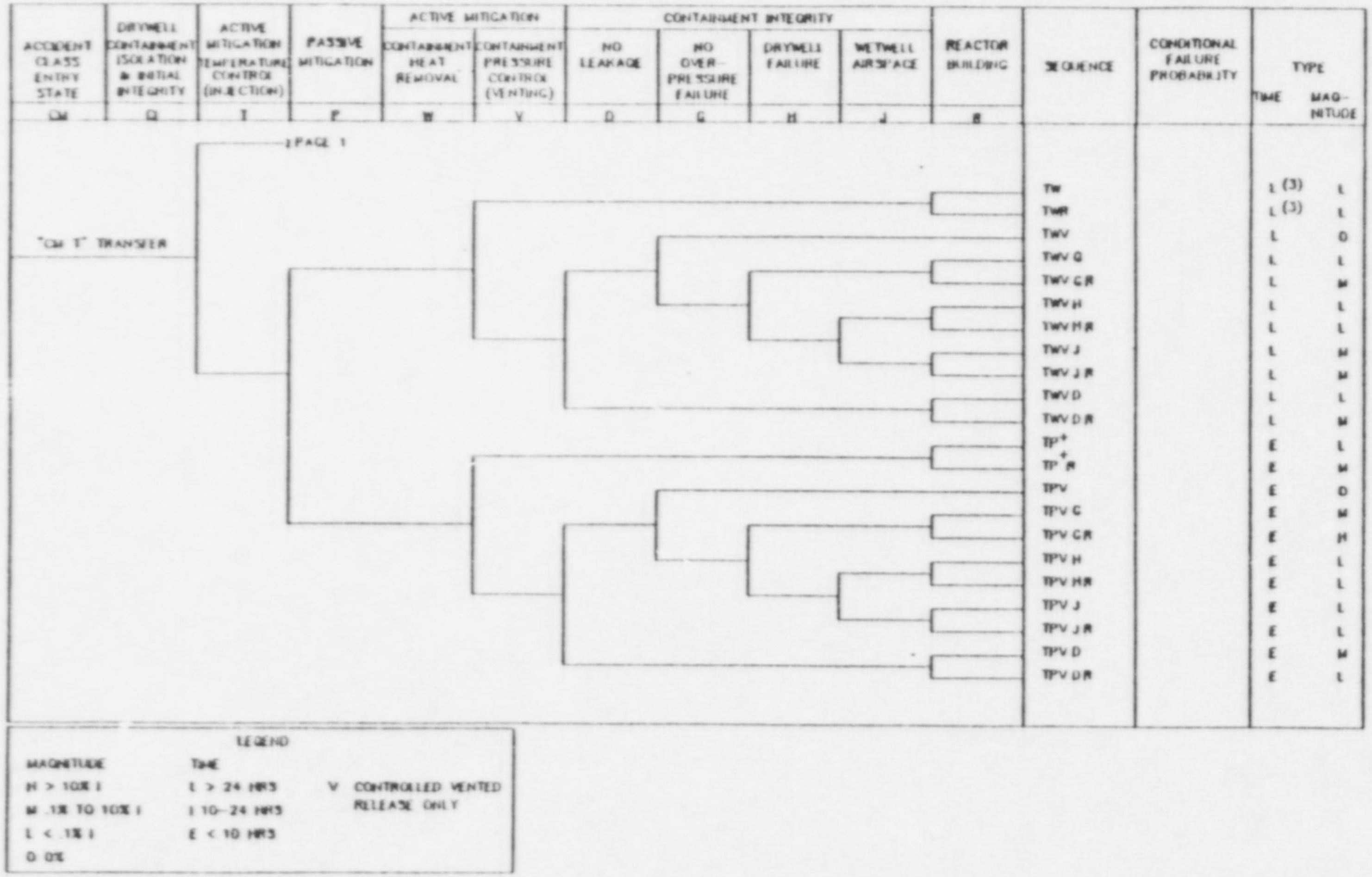

Figure 2.4-2 (continued) Containment Event Tree for Classes I and III

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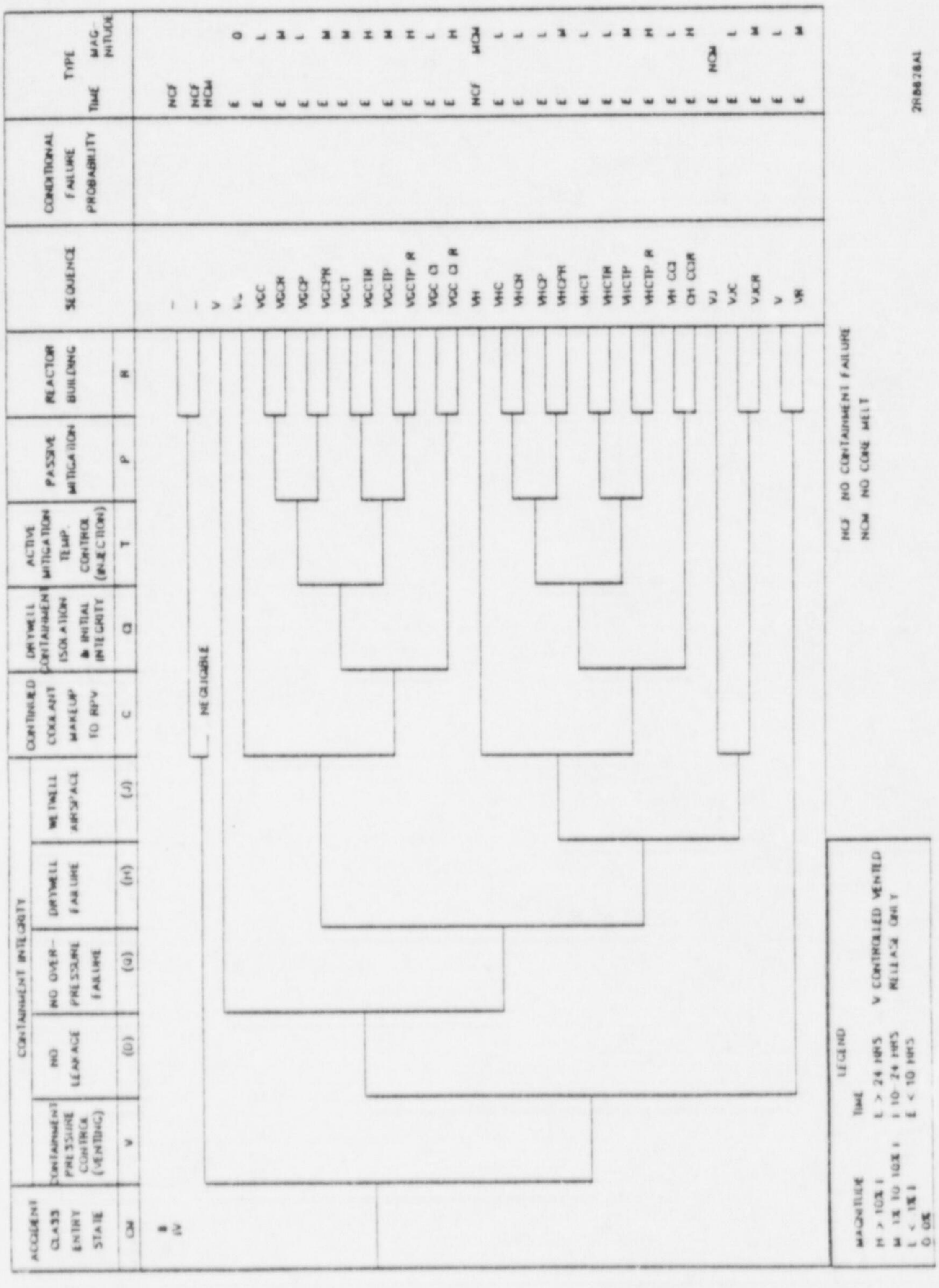

Figure 2.4-3 Containment Event Tree for Classes II and IV

1. Isolation

The CET can be used to estimate the probability of successful isolation. These estimates, which include leaks, can be made for both inerted containments as well as containments constructed prior to the change to inerted containments.

2. Explosive or Energetic Reactions

Failure nodes related to high energy events such as steam explosions, hydrogen deflagration, etc., are addressed. Although these events are of low likelihood, containment integrity questions are still raised in the evaluation of the CET.

3. Containment Temperature Maintenance

The ability to prove containment capability by maintaining containment shell temperature within acceptable limits is also addressed by the CET described in Appendix A. Concerns include core heatup, molten material, and hot non-condensible gases.

4. Direct Structural Damage

Direct heating of the structure is addressed by asking questions about the dispersal of the molten material from the RPV to the sump and drywell.

5. Containment Pressure Maintenance

Similar concerns as those existing for containment temperature control are also considered for containment pressurization.

The containment event trees are presented in Appendix A to establish the types of failure modes that have arisen in past analyses. Therefore, the CETs are meant only as a check to ensure that the proposed changes address those failure modes of greatest interest to the regulators, the public, and the utility. In this context the CET is used qualitatively to ensure that:

- o Combinations of systems necessary to function are identified for certain sequences
- o Failure modes which are not addressed by these fixes are delineated.

The last objective of the CET addresses new potential containment issues which could surface in the future and require additional BWR changes. These types of changes are discussed in later sections.

References

[2-1] Limerick Generating Station Probabilistic Risk Assessment, Philadelphia Electric Company, Docket 50-352, 50-353, September 1982.

[2-2] Shoreham Nuclear Power Station Probabilistic Risk Assessment, Long Island Lighting Company, Docket 50-322, June 1983.

[2-3] Nuclear Power Plant Response to Severe Accident, IDCOR Technical Summary Report, November 1984.

[2-4] Reactor Safety Study: An Assessment of Accident Risks on U.S. Commercial Nuclear Power Plants, U.S. Nuclear Regulatory Commission, NUREG 75/014, WASH-1400, October 1975.

[2-5] "BWR Emergency Procedure Guidelines, Revision 3", BWROG-8262, December 1982.

[2-6] Containment Event Trees, IDCOR Task 4.1 Technical Reports, October 1983.

[2-7] Amos CN, Griesmeyer JM, Kolaczkowski AM, Containment Event Analysis for Postulated Severe Accidents at the Peach Bottom Atomic Power Station, (SAND 86-1135) (Preliminary DRAFT for Review, May 12, 1986.

Section 3
NRC PROPOSED BWR CHANGES

The NRC on June 16, 1986 [3-1] requested BWR owners to review the feasibility of immediate implementation of a select group of changes (hardware and procedural) in order to further reduce the perceived conditional containment failure probability given a severe accident. The items identified by the NRC include the following items:

- Hydrogen Control: Prevent containment failure due to hydrogen combustion (using igniters, or by inerting containment and controlling oxygen ingress)
- Containment Sprays: Assure reliable operation of containment spray during accident conditions, including station blackout
- Wetwell Venting: Avert containment overpressure failure and uncontrolled radionuclide release by ensuring adequate wetwell venting capability and reliable operation of vent/purge valves
- Debris Barrier: Reduce the likelihood of containment failure from direct contact with molten core material
- Severe Accident Management Procedures: Provide emergency procedures and training to ensure operators can recognize and respond to degraded core conditions

The mitigative functional characteristics of each change which are most important in severe accident mitigation are identified in Section 3.1. Section 3.2 summarizes each of the proposed changes and identifies variations on these changes in the form of options. The engineering characteristics of each change which must exist in order that the mitigative function is fulfilled are also defined in Section 3.2.

3.1 INTENDED MITIGATIVE FUNCTION OF EACH PROPOSED ITEM

The purpose of this section is to identify, in the terminology of the simplified containment event tree and containment functional response, the key objectives of the proposed NRC modifications.

In Section 2, the containment functional response items and postulated containment failure modes which have been identified in previous BWR PRA evaluations were discussed. These functional response features which may be fulfilled passively or through active measures include the following:

- o Isolation
- o Pressure control
- o Temperature control
- o Prevention of direct containment contact with molten material
- o Prevention of explosive or energetic reactions
- o Radionuclide scrubbing
- o Fission product scavenging.

The containment failure modes can be classified in ways in which they affect the containment integrity as follows:

- o Pressure induced failures
  - \- Immediate
    - -- Vapor suppression system failures
    - -- Steam explosions (invessel and exvessel)
    - -- Hydrogen deflagration
    - -- Implosion
    - -- RPV blowdown + H_2 generation invessel
  - \- Short term to long term
    - -- Non-condensible gas generation
    - -- ATWS power levels greater than RHR or main condenser capacity
  - \- Long term
    - -- Loss of decay heat removal
- o Temperature induced failures (other than through direct contact of molten material)
  - \- Radiation shine from molten material

- o Direct contact of molten material

  - Molten material may be distributed across the drywell floor such that it reaches the drywell shell. This failure is judged not to be a catastrophic containment failure mode

  - Molten material reaching the downcomer potentially failing the downcomer in the torus room

  - Molten material reaching the torus shell and causing catastrophic failure of the torus loss of the torus water and loss of scrubbing capability of the suppression pool

- o Failure of isolation

- o Failure size and location

Each of these failure modes may have different impacts on the resultant fission product pathway and remaining mitigative capability of containment regardless of prior leakage or rupture breaches of containment; that is, the above failure modes may override in importance more benign failures of containment.

The NRC proposed changes can be viewed simply in terms of their functional purposes and the failure modes they may prevent as follows:

- o Hydrogen control (Inerting)

  -- hydrogen deflagration or detonation

- o Drywell sprays

  -- containment temperature control
  -- debris coolability
  -- containment pressure control
  -- localization of debris
  -- fission product removal

- o Containment venting

  -- pressure control
  -- fission product removal path

- o Debris barrier

  -- prevention of drywell wall direct failure

The cross correlation between previously identified containment failure modes and their potential resolution in terms of the NRC proposed changes is provided in Table 3.1-1.

The discussion in Section 3.2 for each of these features provides options for satisfying the functional requirements with varying postulated conditions, assumptions, and models.

Table 3.1-1

COMPARISON OF CONTAINMENT FAILURE MODE AND NRC PROPOSED CHANGE

FAILURE MODE	NRC PROPOSED CHANGE
1. Steam Over Pressure	
- ATWS	Vent
- RPV Rupture	--
- RPV Melt Through	Vent
- Loss of Effective Containment Heat Removal (TW)	Vent
- Pool Bypass	--
- Steam Explosion	--
2. Non-Condensible Over Pressure	Vent (includes SBO)
3. Direct DW Contact	Drywell Debris Barrier
4. High Shell Temperature	DW Sprays (includes SBO)
5. Direct Contact of Torus w/Molten Material	Drywell Debris Barrier
6. Direct Impingement	--
7. Direct Heating	--
8. Hydrogen Detonation	Inerting Tech Spec
9. Containment Isolation Failure	--

3.2 DEFINITION OF PROPOSED CHANGES: ENGINEERING CHARACTERISTICS OF EACH CHANGE WHICH MUST EXIST TO MAKE THE CHANGE EFFECTIVE

The impact of the proposed BWR containment mitigation system changes can be bounded by considering several options. The potential benefits, the negative safety impact, and the associated costs of each of these options may then be used in assessing the feasibility of the individual choices.

A number of ways of implementing these concepts have been developed. Appendix B provides some different conceptual designs from those offered in this section.

Design Criteria

Some general "design criteria" have been formulated to identify the NRC proposed changes in more deterministic terms. These "design criteria" are provided for the following:

- All changes
- Venting
- Debris coolability
- Debris Barriers

All Changes

- The backup drywell sprays, wetwell vent and core debris barriers are not "safety-related" in the regulatory sense. Their function need not be seismically qualified, and is not subject to the single-failure criterion and is not subject to the quality assurance requirements of 10CFR50, Appendix B.

Venting

- The wetwell venting scheme will accommodate ATWS as an option.
- Wetwell venting is to be reliably operable for Station Blackout, but the vent sized for ATWS need not be. Estimates for the non-ATWS vent should include the option for non-Blackout operability. Reliability of .9 to .95 is adequate.

- Design conditions at time of venting are as follows:

  Case 1 ("blackout" vent)

  WW Pressure - 76.0 psia
  WW Temp - 247. °F
  Mole frac steam - 0.27
  Mole frac H_2 - 0.06
  Mole frac CO_2 - 0.02
  Mole frac N_2 - 0.65
  dP/dt - 4.6 psi/hr

  Case 2 ("ATWS" vent)

  WW Pressure - 76 psia
  Steam flow equal to maximum HPCI flow
  Case 1 - 33 lbm/sec saturated steam (for 3300 Mw(t) core)

- Airborne activity at time of venting is equal to 100 percent NG distributed in the drywell and wetwell airspace w/no particulate (assumed scrubbed).

- Venting is needed within 1/2 hour for ATWS and within 10 hours for Blackout (w/and w/o AC power, respectively).

- Vent and reclosure capability should function for 24 hours w/o AC power restoration.

- For the case where the wetwell vent terminates outside the reactor building, it is unacceptable for the purge system to fail in the reactor building during venting.

- No vent throttling is required. However, it is desirable to first open a vent path that is within the capability of the SGTS to process the flow. If this vent path is insufficient to maintain acceptably low containment pressure, then the full vent should be opened. Once open, it is necessary to reclose the vent one time.

- No filtration of the vent discharge is required.

- Wetwell venting must be operable without entering the reactor building. If operator action to establish the vent is from outside the control room, it must be in a sheltered location permitting rapid return to the control room.

Debris Coolability

- An injection system to the breached reactor vessel can fulfill the objectives of the backup drywell sprays as long as it meets the other system requirements.

- Backup spray/injection systems must be capable of injecting 250 gpm (for a 3300 Mw(t) core) with 100 psid between the pump suction and the drywell and operate reliably under station blackout conditions for 24 hours. Reliability of .9 to .95 is adequate.

- It is acceptable to use a "non-integrated" source (e.g., a fire engine) to provide the backup spray/injection as long as it is on-site.

Debris Barrier

- Core debris retention within the MKI/MKII pedestal is not desirable.
- Temperature, mass and volume of core debris will be based on information given in BMI-2104, Volume II for a 3300 Mw(t) core. Assume the viscosity to be "water". (This criteria is used only as an upper bound and does not reflect the best estimate.)
- A thickness of refractory material equivalent to one foot of concrete is required to prevent failure of the core debris barrier for a minimum of 24 hours.

While these "design criteria" are viewed as the best that may be needed, there is a range of existing plant systems or configurations that may already provide adequate systems to eliminate the issues behind the NRC proposed changes. Existing plant specific hardware and procedures may be able to be fashioned in such a way that a reliable mitigation system can be demonstrated using existing plant equipment and under the specific scenarios that can be postulated at each plant.

3.2.1 Drywell Spray Options

The drywell spray arrangement suggested by the NRC has several objectives; namely:

- To lower the containment pressure
- To cool vulnerable equipment
- To quench debris
- To scrub aerosols

In addition, it was strongly suggested that this function be reliably provided during station blackout events.

The NRC has identified drywell sprays as a method of providing drywell temperature control and assisting in preserving containment integrity. Detailed calculations by the IDCOR team using the MAAP codes had identified that any source of water injection to the Mark I drywell (greater than 250 gpm) is effective debris coolability and for preserving the containment within its realistic temperature limits, i.e., less than 1200°F [3-2]. Although drywell sprays may be more effective for "instant" pressure reduction and radionuclide scrubbing, the calculations indicate that water injection through the drywell sprays, the breached reactor vessel, or any other injection source capable of reaching the debris is adequate for debris coolability. In addition, injection through the RPV after breach has the side benefits of maintaining somewhat lower primary system temperatures and inhibiting revolatilization. Further, some Mark II plants would have ineffective debris coolability with drywell sprays. Therefore in the discussion of this item the use of the term "drywell sprays" will be replaced with the term drywell injection or debris coolability (note that drywell sprays may still be the most important source of water in many plants). Instead of drywell sprays, active water injection mitigation from any available coolant injection source will be assessed to satisfy the desire for containment temperature control.

The current BWRs provide an extensive network of coolant injection sources which can provide adequate debris cooling capability whether injected into the RPV or the containment. These coolant injection sources include the following:

Pump	Power Requirements: Pump	Power Requirements: Valves
o RHR pumps	AC offsite or onsite	Same or manual
o CS	AC offsite or onsite	Same or manual
o Condensate pumps	AC offsite	None
o CRD pumps	AC offsite (some onsite)	None
o Service water pumps	AC offsite or onsite	Same or manual
o Diesel fire pump	None	Same or manual
o Condensate transfer pumps	AC offsite	Same

There are drywell sprays on the Mark I and II containments. These sprays have flow supplied principally by the RHR pumps. However, there is a wide spectrum of plant specific designs when it comes to "backup" methods of providing drywell spray capability. Section 4 discusses some of the possible differences. Note for the RPV breach and post-core melt scenarios, any drywell injection location is judged to fulfill the intent of the functional requirements cited for drywell sprays.

In order to satisfy the intended functions of accident mitigation given in Section 3.1, the identified item must have certain engineering characteristics. These characteristics must be defined for both evaluation of the benefits to be obtained and the costs of the modifications. Table 3.2-1 provides an interpretation of the options which might be proposed in terms of their engineering characteristics.

In Table 3.2-1 are listed the pumps available to supply the source of water, the valves through which the water may reach the drywell, the minimum power requirements for the valves and pumps, and the instrumentation required for the operator to successfully complete the task of cooling the debris. A description of the conceptual designs similar to that labeled "Options C and D" follows for a reference Mark I plant. Component and location identifiers are included in Figure 3.2-1, a schematic for the described option.

The concept for providing a 250 gpm source of water to the existing primary containment spray header could be to utilize the existing diesel driven fire pump as the source of water and pressure. A tie would be made between existing reactor building fire protection piping and the RHR system, at penetration X-39A (see Figure 3.2-1), downstream of existing valve MOV-31A. The proposed system is shown schematically on Figure 3.2-1. If spray (flow) into the reactor pressure vessel were required a similar system could be installed using the core spray header. The tie in would be at penetration X-16A, downstream of existing valve MOV12A. Alternatively, some plants may have the capability to inject into the RPV through LPCI lines.

Table 3.2-1

ENGINEERING DESCRIPTION OF PROPOSED CHANGES TO BE USED
IN FEASIBILITY AND COST ESTIMATE
FOR DRYWELL SPRAYS

Drywell Coolant Injection Option	Pumps	Drywell Injection Valves	Power Required	Instrumentation Required(1)
A	RHR	DW Sprays	AC	DW Temperature DW Pressure Contain. Wtr Lvl.
		LPCI		
	CS	CS Injection	AC	"
	Condensate	FW Injection	AC	"
B	Option A HPSW	Option A Ultimate core cooling valves and LPCI or DW sprays	Option A AC	Option A Option A
C	Option B Diesel Fire Pump	Option B Ultimate core cooling valves	Option B None	Option A
		and LPCI or DW sprays + Spool piece	AC(2)	
D	Option C	Option C (w/o spool piece)	Option C	Option A

(1) There is an EPG requirement to avoid flooding the containment. A procedure may be required to fulfill this requirement.

(2) The injection valves may be located such that coolant injection late (after core damage) would be precluded if manual valve operation were required during a station blackout situation. Option C requires remote operation even with no offsite or onsite AC or DC power.

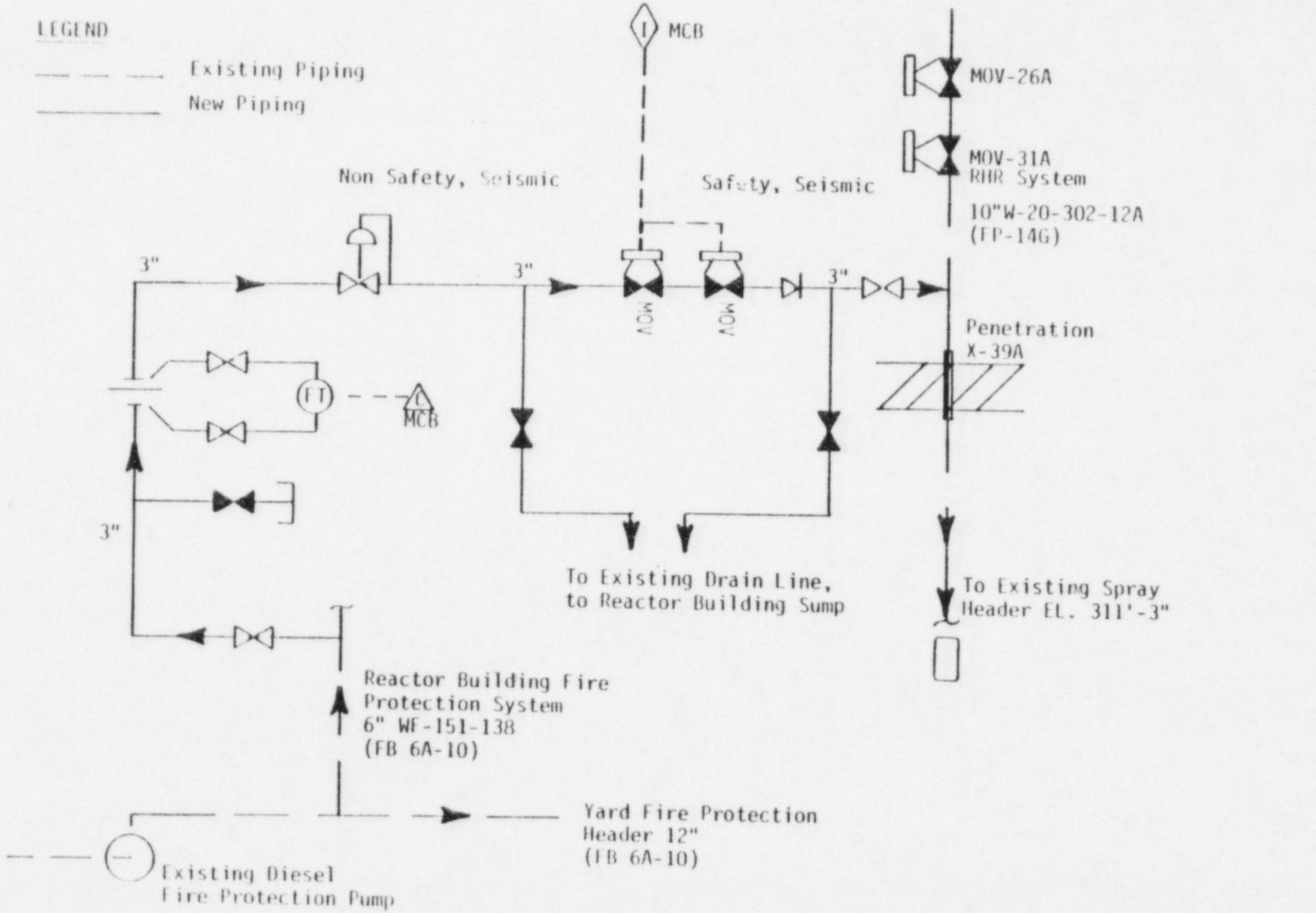

Figure 3.2-1 Primary Containment Spray Header System

Two normally closed, motor operated containment isolation valves are provided to manually initiate system operation. Opening both valves electrically, with power being provided by an independent D.C. source, will drop the fire protection header pressure, automatically starting the diesel driven fire pump. Controls for these two valves, along with open and shut indication, will be provided in the control room. A pressure control valve would be installed to assure delivery of 250 gpm to the spray header depending on existing fire pump demand. The limiting condition for this operation could be the diesel fuel supply for long term operation. This flow could vary with containment pressure, but would not be less than 250 gpm. A flow transmitter with input to the control room would be provided to monitor flow. A check valve installed downstream of the last MOV of the system would preclude inadvertent flow from the existing RHR system to the fire protection system. Two system isolation valves installed in the system would allow for normal system maintenance while the plant is in operation. Additionally, test connections provided down stream of the pressure control valve would allow periodic verification of the new system readiness and operability. The test line would be routed to an existing drain line to one of the reactor building sumps, allowing flow to be verified during plant operation. The difference between options C and D is only that Option C requires a hose connection or spool piece be manually aligned during the accident.

Of the four postulated options, it appears that Option C, because it is viable during station blackout events, is the minimum acceptable to meet the intent of the NRC objectives as stated orally at the June 16 meeting [3-1]. Option D also satisfies the objectives but would require more effort than Option C to implement.

Table 3.2-2 provides a summary of the options which may be possible in operating plants for various levels of expenditure.

3.2.2 Containment Pressure Control (Containment Venting)

The NRC suggestion for containment pressure control has the objectives to:

- o Avert uncontrolled overpressure failure.
- o Maintain a scrubbed release path.

Table 3.2-2

DRYWELL SPRAY OPTIONS

OPTION	DESCRIPTION
A	This option provides drywell spray capability with the RHR pumps. It requires realignment of certain valve and piping arrangements from those which exist currently at most plants. This option requires a source of offsite or onsite AC power. Drywell injection capability is provided for by using the RHR pumps or by using the CS and/or CRD+ system. This injection mode also requires plant reconfiguration, as well as a source of AC power.
B	This option is similar to Option A, except that service water or HPSW are also used as backup systems. A source of AC power is still required.
C	Same as B except DFP++ also provides a successful drywell injection mode during severe accidents and would be operable without dependence on offsite or onsite AC power. However, this could require operator actions in the reactor building prior to the release of radionuclides in the vessel. Plant configuration changes (valve, piping, electrical) are required before this option would be available.
D	Same as C except no spool piece or hose connection operation required for installation and no access to the reactor building is required following loss of AC power. By eliminating the need for operator access this option minimizes the changes of exposure to the plant staff. However, more physical and procedural changes are required.

+May be plant specific.
++Diesel fire pump.

These objectives were more fully defined by further specifying that the vent:

- Have substantial capability to vent the wetwell (specifically for ATWS events).
- Be remotely and reliably controlled (including station blackout).
- Have the ability to be reclosed (this was indicated to be of lower priority).

Containment venting has been identified as a potential operator action for certain postulated accident sequences which are far beyond the plant licensing design basis. The BWROG EPGs currently specify that containment venting should be used as a last report operator action in the event of rising containment pressure. The action is taken with the purpose of preventing an uncontrolled breach of containment, while maintaining the containment function as a fission product barrier through deposition and scrubbing.

The methods to accomplish containment venting and the containment pressure at which one would initiate such a procedure are still under development. As with most engineering decisions there are a number of trade-offs which need to be made to establish a rational procedure. Some of these considerations are summarized in the following discussion.

The BWROG EPGs provide operator guidelines and other design considerations for the implementation of a scheme for containment venting. The purpose of the containment venting guidelines is to provide capability to the plant to preserve containment functionability in the face of potential overpressure events.

Flexibility is provided to each utility to make use of the hardware within each plant. The venting procedures are designed to be compatible with the existing plant hardware. No new plant hardware is visualized as being installed as part of the EPG implementation.

The containment structural acceptance test (SAT) is performed at 1.15 design pressure. In addition, the containment is periodically tested up to a certain pressure (P_A)+ (by means of an integrated leak rate test, or ILRT), and there

+ P_A = Peak calculated LOCA pressure.

are postulated design basis accidents which could lead to these pressures. It seems prudent to allow the containment to perform its design function under such conditions without resorting to venting. Therefore, the containment design pressure may be the minimum pressure which should be considered before venting is attempted. As the pressure rises above the containment design pressure, there is a decreasing confidence in the high integrity of the containment. Therefore, containment venting becomes more desirable as the pressure continues to rise.

There are a number of considerations in the thought process which may influence the choice of the containment pressure window over which the plant specific procedures would specify operator venting of containment. These considerations are the following:

- o Containment structural capability
- o Containment vent valve operability
- o Measured parameters and instrumentation
- o System operability
  - SRV
  - HPCI/RCIC
  - Low pressure pumps
- o Reactor Building environment
- o Radionuclide activity
- o Types of accident mitigation.

Based upon a generic review of the concept of containment venting and the plant specific applicability of the concept, several conclusions have been reached. These conclusions are summarized as follows:

- o The concept of containment venting is specified for use only in postulated accident scenarios far beyond the plant design basis.
- o The concept of venting provides the operating staff an alternative which:
  - maintains the containment function

- minimizes any radionuclide release to the public while preventing an uncontrolled containment breach

- provides an alternative decay heat removal path

o Generic procedural guidance exists

o The plant specific implementation may require consideration of the types of items highlighted in the evaluation

o Typical considerations which are judged appropriate for inclusion in the decision making process include:

  - Vent Initiation Pressure

    -- Calculated values of the ultimate structural capability may not be the limiting factor in setting the containment vent pressure, i.e. some margin may be wished to be retained.

    -- SRV operability is a principal goal of long term accident mitigation and this may lead to setting the vent initiation pressure in the range of 1.0 to 1.5 times design pressure for Mark I and II containments.

    -- Low pressure systems (exceptions: HPCS, CRD, and motor driven feedwater) may be required for accident mitigation during the venting procedure and therefore NPSH limits need to be addressed under venting conditions.

  - Vent Termination

    -- Termination of venting is judged prudent in the range of 10 to 20 psi below the vent initiation pressure.

  - Vent Pathways

    -- There are a wide spectrum of plant specific vent pathways available to the operator with a number of considerations affecting their desirability such as valve capability. However, the following conclusions are appropriate:

      1. The lines which originate in the wetwell airspace and exhaust to (a) the SGTS or (b) to outside the reactor building are the highest desirability level. However, it should be noted that SGTS filter trains may be incapable of handling full wetwell vent flow, i.e. if all the decay heat immediately after shutdown is being removed through the vent. In addition, their may be other features of the SGTS that could be adversely affected by such vent conditions.

2. Maximize control over the amount released, i.e., small lines preferred.

3. Larger lines from the wetwell air space to outside the reactor building are next in priority.

4. Next, larger lines from the wetwell air space to the reactor building may be used.

5. The lowest priority lines are those that originate from the drywell and exhaust to (a) the reactor building or (b) outside the reactor building.

o Open Items Regarding Containment Venting: Current Status

- A plant specific implementation plan is needed.
- A decision is needed on the use of very large vents to cope with ATWS.
- A decision is required on the advisability of venting during events with high radiation alarms.
- Use of ductwork during venting needs to be considered as a potential hazard to personnel and equipment in the event venting is included.
- Power requirements and need for or ability to access various plant locations for venting implementation are plant specific.

An extensive amount of work has been performed by FAI to identify alternate ATWS mitigation strategies which would not require containment venting. Such strategies are just now being developed, supported by analytic work at GE and EPRI. If these strategies prove successful, the use of a very large vent for ATWS can be reduced in importance (i.e. safety benefit) or completely eliminated. However, this effort (described in Section 4.2) is not yet complete.

Given the above background on venting, the options identified to accomplish the NRC objectives of station blackout and ATWS mitigation capability (or portions of these objectives) for containment venting are summarized in Table 3.2-3. Table 3.2-4 provides an engineering interpretation of these options. Figures 3.2-2a and 3.2-2b provide simplified schematics of containment venting conceptual diagrams for Mark I and II plants, respectively. Option C" appears

Table 3.2-3
CONTAINMENT VENTING OPTIONS

OPTION	DESCRIPTION
A	A small wetwell vent capable of removing decay heat approximately 8 hours after control rod insertion. The capability to operate the vent would be remotely provided from offsite or onsite power sources. The valves would be capable of opening at the specified opening pressure (taken to be 60 psig for this analyses).
A'	Option A plus the capability to open the vent valves under station blackout conditions. Based upon postulated failure modes this may be a dedicated DC power supply and/or air operated valve.
A"	Option A' with hard pipe inside the reactor building allowing the release to be directed to the environment.
B(1)	A wetwell vent capable of removing approximately 6-8% power associated with controlling reactor water level below the TAF for ATWS related events. The capability would be remotely provided from both offsite and onsite power sources.
B'	Option B except SBO included.
B"	Option B' except hard pipe in the reactor building to allow direct release to the environment.
C	A wetwell vent capable of removing 18-30% power associated with controlling reactor water level near the TAF for ATWS related events. The capability would be remotely provided from both offsite and onsite power sources.
C'	Option C except SBO included.
C"	Option C' except hard pipe in the reactor building to allow direct release to the environment.

(1) This may also have the definition of providing the capability to relieve the non-condensible gas generation from core concrete reaction given that a core melt occurs for other sequences such as a station blackout.

Table 3.2-4

ENGINEERING DESCRIPTION OF PROPOSED CHANGES TO BE USED IN FEASIBILITY AND COST ESTIMATE

VENT	APPROX. SIZE	POWER REQ'D	Rx BLDG VENT CONSTRUCTION	INSTRUMENTATION/POWER	DOES IT EXIST IN TYPICAL BWRs (See Section 4)
A	6"	AC	Duct	DW Pressure/DC or AC	Yes
A'	6"	DC and/or Air	Duct	DW Pressure/DC	No
A"	6"	DC and/or Air	Pipe	DW Pressure/DC	No
B	6-18"	AC	Duct	Reactor Water Level/DC or AC DW Pressure/DC or AC	Yes
B'	6-18"	DC and/or Air	Duct	Reactor Water Level/DC DW Pressure/DC	No
B"	6-18"	DC and/or Air	Pipe	Reactor Water Level/DC DW Pressure/DC	No
C	36"	AC	Duct	Reactor Water Level/DC or AC DW Pressure/DC or AC	No
C'	36"	DC and/or Air	Duct	Reactor Water Level/DC DW Pressure/DC	No
C"	36"	DC and/or Air	Pipe	Reactor Water Level/DC DW Pressure/DC	No

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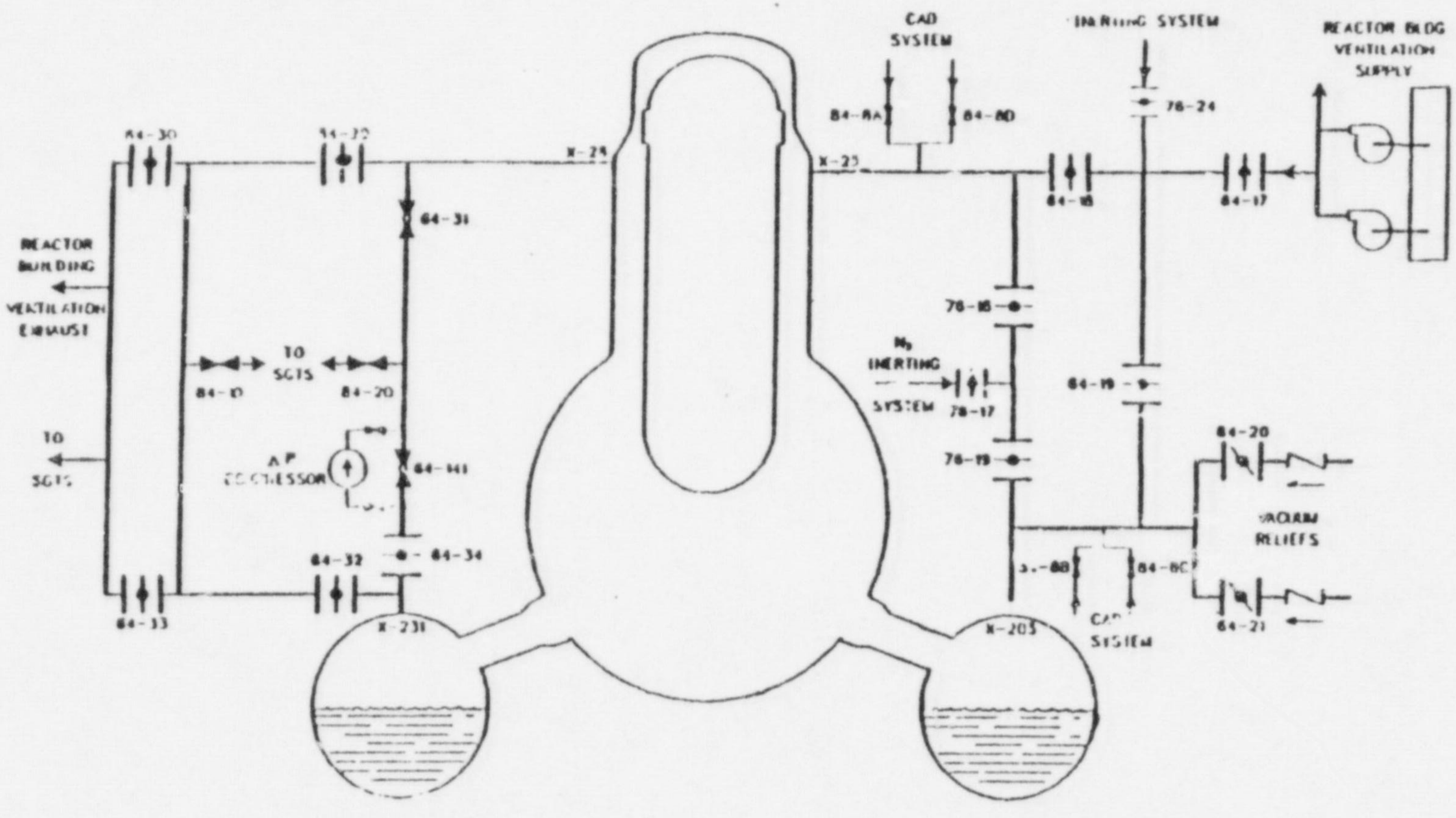

Primary containment ventilation and inerti.g systems.

Figure 3.2-2a Schematic of Containment Venting Lines

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3-21

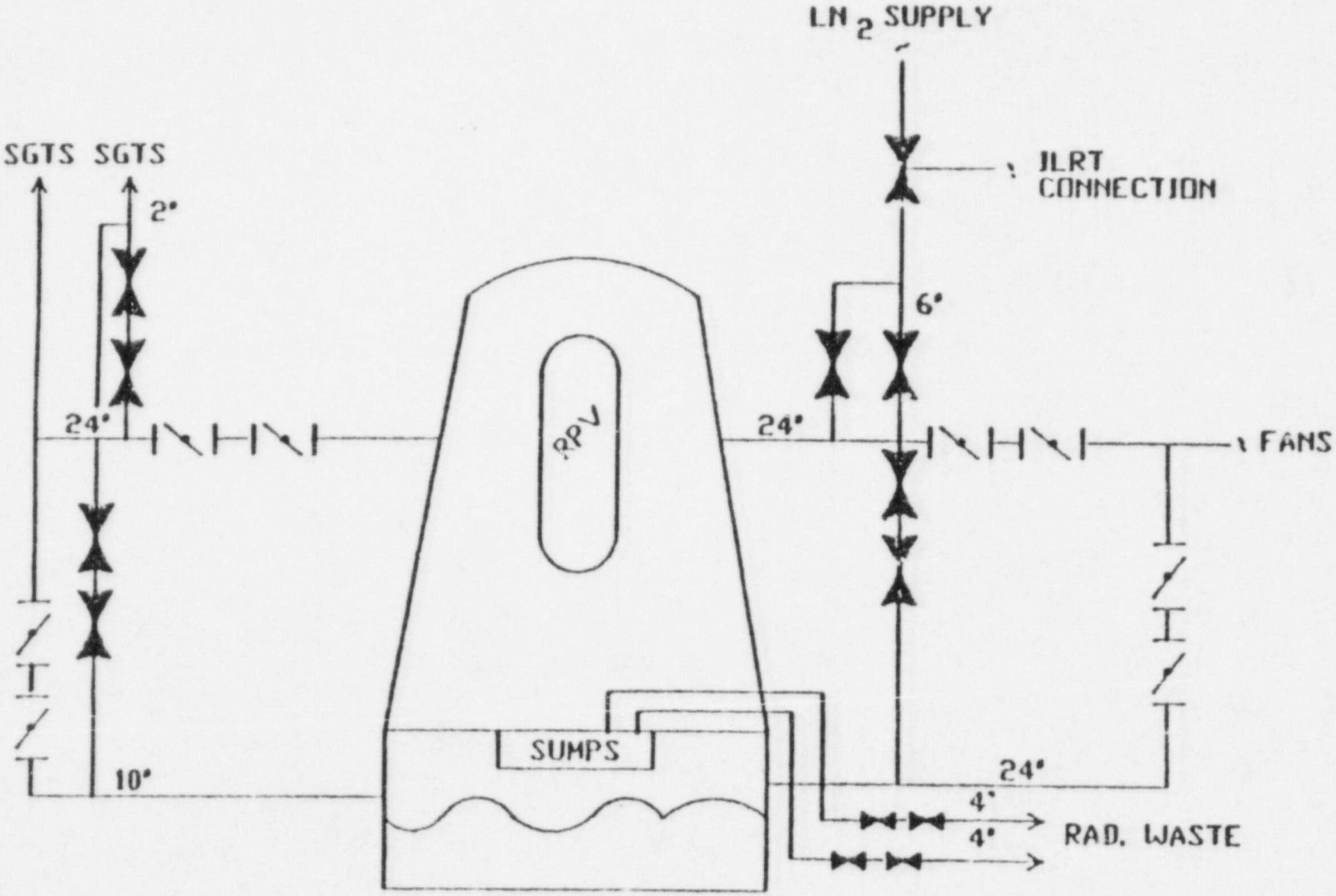


Figure 3.2-2b Containment Venting Provisions

to meet all the requirements inferred from the NRC request [3-1]. A description of a conceptual design similar to Option C" follows:

> The venting system is designed to vent steam and other gases to the atmosphere from the wetwell to prevent its overpressurization. The system will be manually operated, and will function without AC power. No filtration or other treatment of the vented gases is provided.
>
> This system is designed to vent up to 600 lb/sec during an ATWS. It will also be capable of venting the lesser flows that would result from a severe accident. As shown on Figure 3.2-3 a 36 in. line would be required.
>
> Two DC-powered, fail closed, motor-operated butterfly valves are included. The vent system would connect to the drywell vacuum relief line immediately above the torus. (Note that only a few Mark I plants have the external vacuum breaker lines. Therefore this plant specific design difference may require more expensive penetration modification to be made.) The vent line would pass through the floor at ground level elevation, through the two butterfly block valves, and then to the outside of the reactor building.

3.2.3 Barrier Options

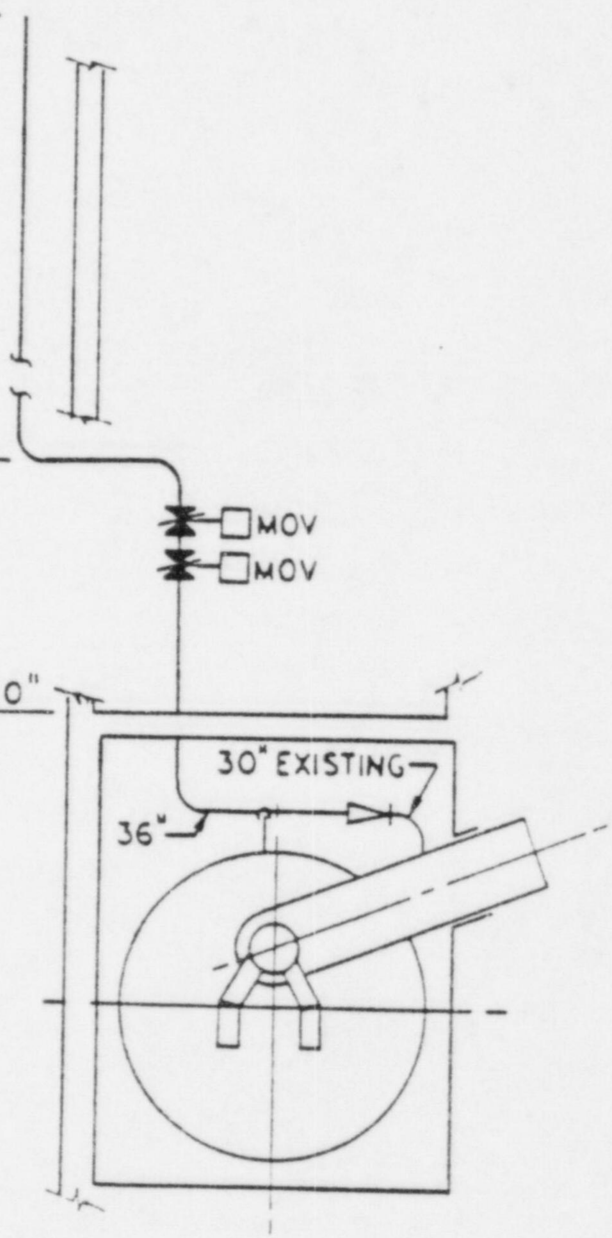
The NRC proposed containment modifications appear in two general categories:

- o Debris Barrier: Barrier to prevent contact of molten material with the containment boundary.
- o Torus Water Barrier: Barrier to preserve the torus water in a location that will maintain containment functionability as an effective fission product scrubber.

Debris Barrier Options

The debris barrier suggested by the NRC has the objective to "Reduce the likelihood of failure (of containment) by direct attack (by the molten fuel)." An additional objective is to also ensure that molten material does not enter the torus. Each of these are principally Mark I related issues.
Note that an initial fundamental decision was made between two options (see Figures 3.2-4 and 3.2-5. Also see Appendix B for other conceptual designs):

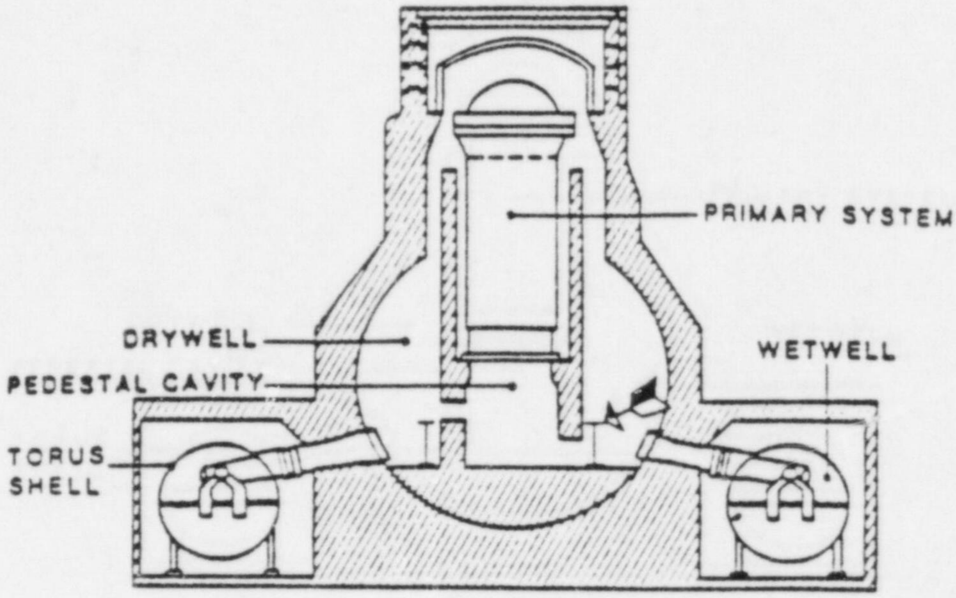
1. Plug the pedestal under the RPV with a barrier (Type A).
2. Position the barrier adjacent to the drywell wall (Type B).



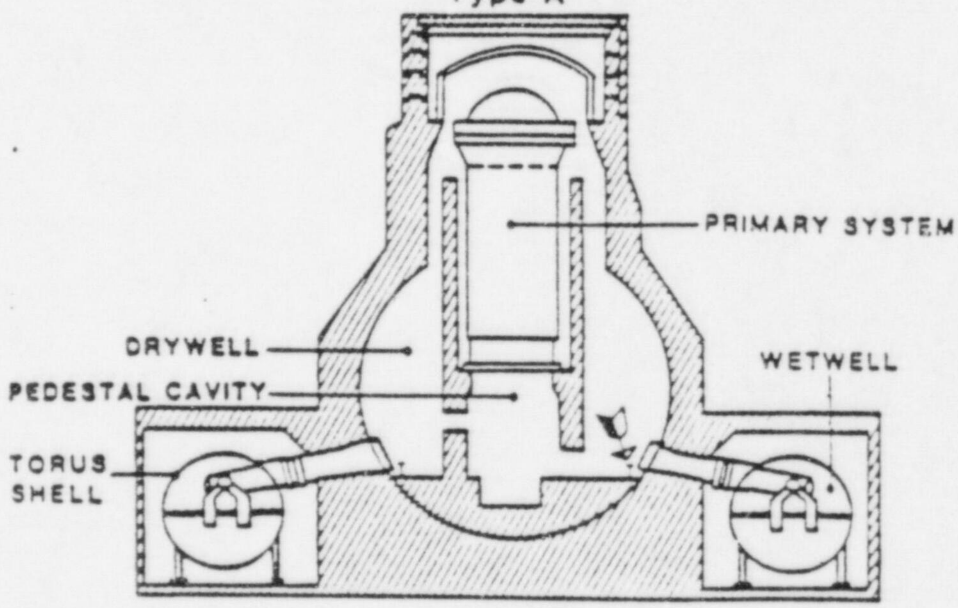
NOTE: VALVES TO BE TESTED AS PART OF CONTAINMENT LEAK TEST.

		TITLE: FIGURE 3.2-3 WETWELL VENTING SYSTEM (ATWS OPERATION)	SCALE
CHECKED			DATE
CORRECT			
APPROVED			SKETCH NUMBER
REVISIONS	② ③ ④ ⑤		

TYPICAL MARK I CONTAINMENT DESIGNS



Type A



Type B

Figure 3.2-4 Debris Barrier Options as Envisioned for BWR Mark I Containments

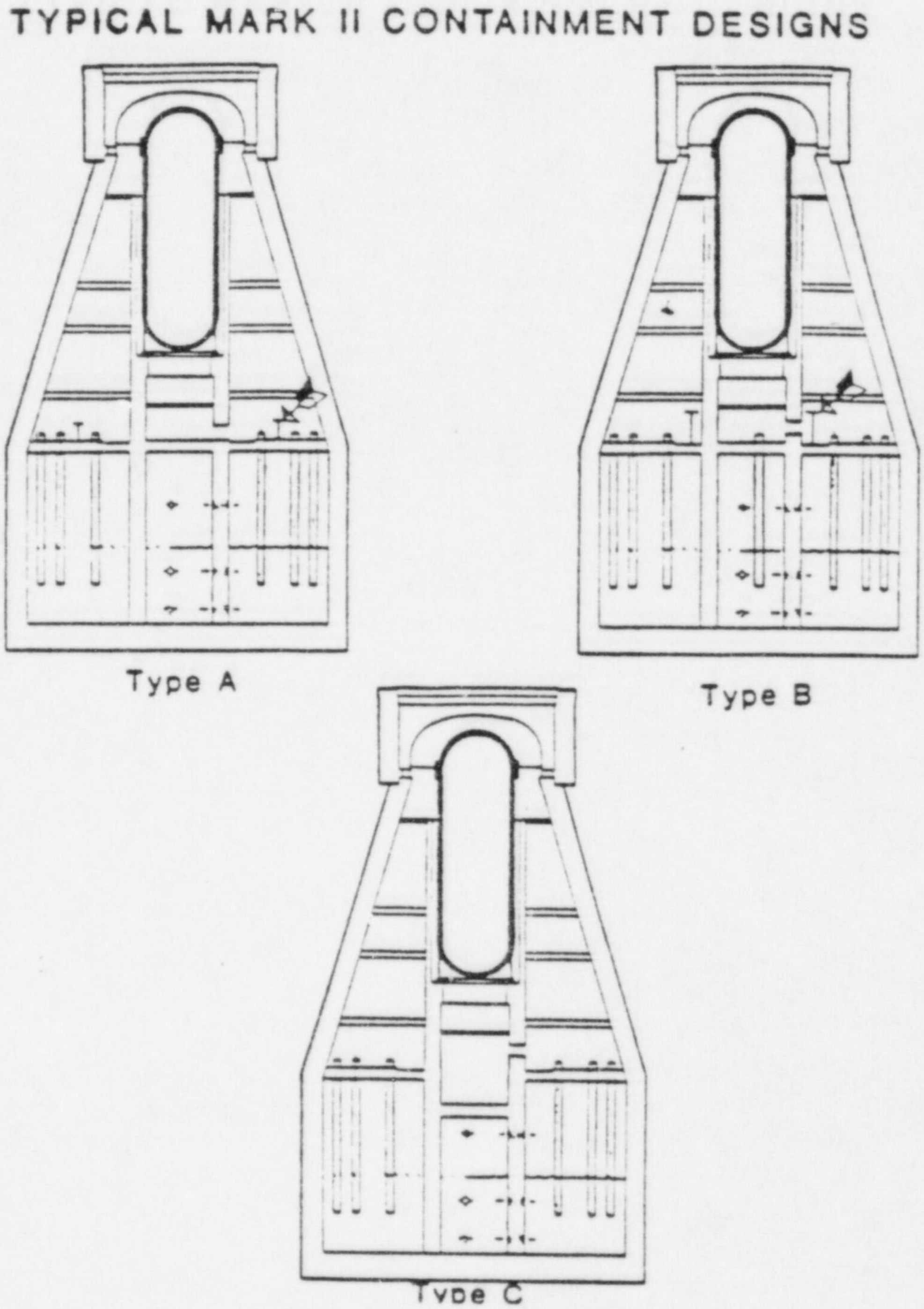


Figure 3.2-5 Debris Barrier Options as Envisioned for BWR Mark II Containments

The first option may be rejected for Mark I containments on the basis that the molten material should not be confined to the pedestal region because of potential attack on the pedestal support and prevention of drywell spray effectiveness. For Mark II containments either option can be feasible if the molten material has access to the suppression pool water through drain lines or downcomer vent pipes (see Shoreham and Nine Mile Point 2 designs Type B Mark II).

The second option has the advantage of allowing the drywell sprays to be effective in direct debris coolability by providing intermixing of spray water with debris. The remaining report discussion focuses on the second option for Mark I containment.

Because there is not a general consensus regarding the actual progression of accident sequences or the code capability to predict the phenomena there may be a range of barrier designs developed to address variations in assumptions and code models. Section 4.2 addresses some of the modeling issues involved in the decision process. A set of options to address such issues is provided in Table 3.2-5, while Table 3.2-6 contains an engineering description of the options.

A brief description of a conceptual design for a debris barrier follows:

> General Design: A debris barrier is provided to contain the core melt and prevent contact with the containment shell or liner. This barrier can be provided at the RPV pedestal or at the extremities of the drywell floor, depending upon specific plant arrangements.
>
> Specific Design: The example core debris barrier is shown on Figure 3.2-6. This barrier would fit well below the downcomers to the torus and would interfere minimally, if at all, with the downcomers baffle supports.

Torus Water Barrier

In the event that the torus integrity is violated due to the direct contact of molten material (i.e., molten material is not quenched), then the containment functionability as a fission product scrubber could be compromised. A barrier might be devised which would act as a secondary containment around the torus pool which would prevent the water level from going below the downcomers. Such

Table 3.2-5

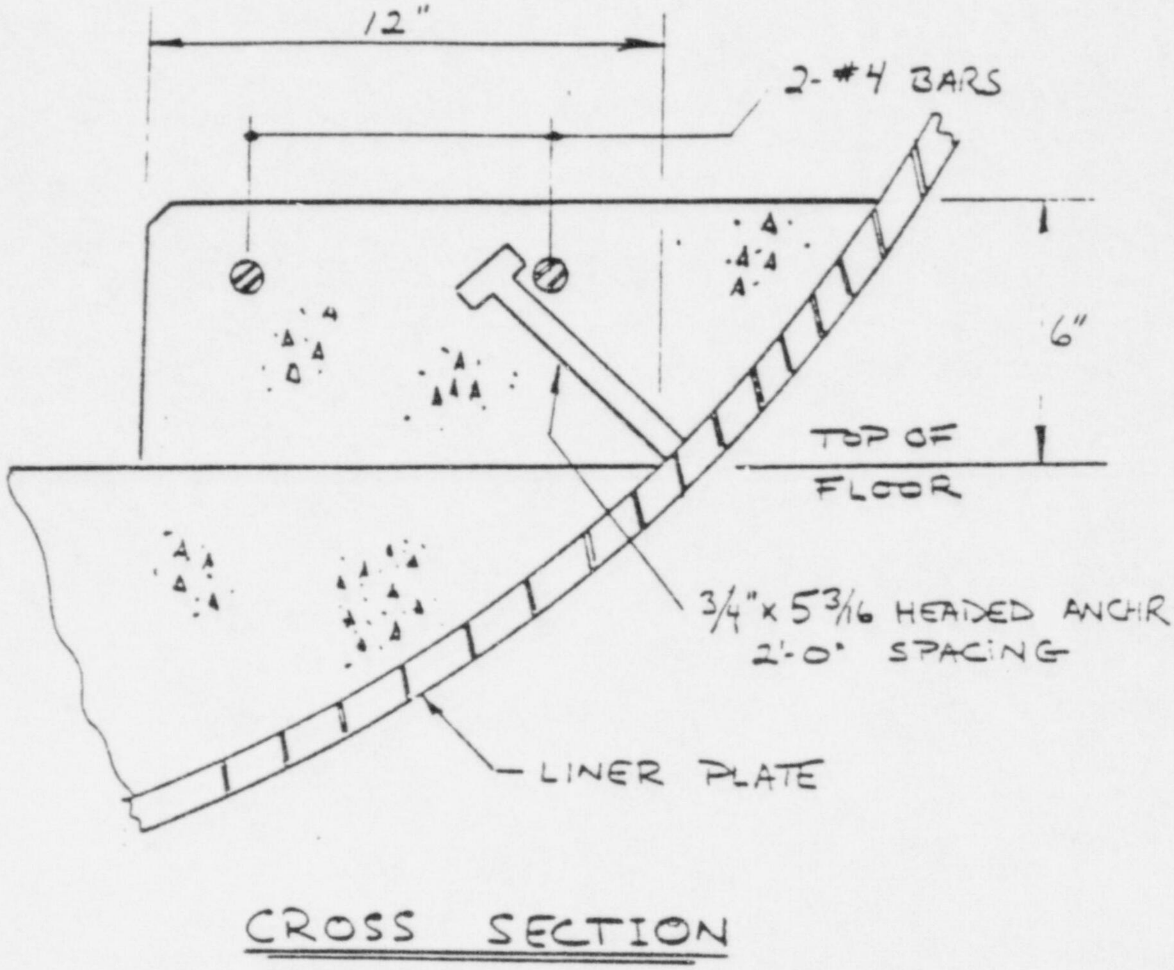
DEBRIS BARRIER

OPTION	DESCRIPTION
A	Small barrier in the range of 2-4 in. height to prevent small amounts of molten material that may be fluid enough to reach the outer part of the drywell floor contacting the drywell wall. A nominal thickness of 6 in. to 1 ft. is estimated to stop the initial flow of material to the wall.
B	This differs from Option A in the amount of molten fuel to be contained. The barrier for Option B is a barrier of sufficient size to contain the entire molten core if it is assumed to be all fluid and on the drywell floor. A nominal thickness of 6 in. to 1 ft. is estimated to stop the initial flow of material to the wall.
C	Option A with a greater thickness to provide a high probability of preventing molten material from flowing to the drywell wall even if extended times are required.
D	While Option B is a "tall" barrier, Option D is a "tall and thick" barrier. Option D is Option B with a barrier of sufficient size to contain the entire molten core if it is assumed to be all fluid and on the drywell floor. A nominal thickness of 6 in. to 1 ft. is estimated to stop the initial flow of material to the wall.
E	Option D welded to the wall of the drywell.
F	A maze barrier which provides protection of the DW shell from slow moving molten material but also provides protection form high flow rate ejection by deflecting the flow. (see Appendix B)

Table 3.2-6

ENGINEERING DESCRIPTION OF PROPOSED CHANGES TO BE USED
IN FEASIBILITY AND COST ESTIMATE

Debris Barrier	Height of Barrier	Thickness of Barrier
A	2-4 in.	6 in. to 1 ft.
B	8-12 in.	6 in. to 1 ft.
C	2-4 in.	2 ft.
D	8-12 in.	2 ft.
E	8-12 in.	2 ft.
F	2-4 in.	2 ft.



Note: Approximately 140lf of curb is required.

Figure 3.2-6 Core Debris Barrier Mark I Containment a stringent requirement may not be met by any but a few of the existing Mark I containments. Alternatively, a torus water barrier could be constructed, and exists at most plants to prevent torus water leakage or rupture from spreading to adjacent compartments. Since adjacent compartments contain safety related equipment such as CS and LPCI pumps such a barrier could have the side benefit of preventing damage to ECCS equipment for any loss of torus water event. (Suction to these pumps from the suppression pool could however be lost in some plants.)

The torus room barrier would provide a water overburden on any molten material that made its way to the torus room floor (this is expected to be zero or very small amounts), but because the downcomers are not covered the remaining radionuclide releases may not be scrubbed.

The torus room barrier is not discussed further in this report. It does not appear to have significant containment related benefits to justify further consideration (although it may have benefits related to prevention of core damage sequences, but that is another investigation).

3.3 CONSIDERATIONS ON THE EFFECTIVENESS OF CHANGES INDIVIDUALLY: AND IN COMBINATION

This section looks at dominant accident sequences which may be effectively mitigated through the use of a single change or combination of changes.

As noted in Section 2, there is a wide spectrum of potential severe accidents. As noted in Section 2.4, most types of accident sequences can be successfully contained or mitigated within the BWR containment with a high reliability. There are, however, several low frequency accident sequence types for which this mitigation may be less reliable. These sequences include:

- Station blackout sequences for which no AC power is available.
- ATWS sequences for which containment failure could occur in a catastrophic manner affecting the coolant injection sources and causing a preexisting containment failure before core melt.

Table 3.3-1 summarizes the effects of the proposed changes as they affect these two low frequency sequences, additional discussion of the sequence mitigation capability is provided in Section 5. This brief summary is to note that the changes may depend for their success to some extent upon the successful implementation of the other changes in order to provide assurance that the perceived dominant failure modes are adequately addressed. Take, for example, the following combinations of actions:

1. Debris Coolability+: for temperature control (2-3 hrs).
2. Venting: for pressure control (6-10 hrs).
3. Debris barrier prevents direct contact failure of the drywell shell.

For a conservative "design basis" approach, it would be necessary to install capability for all three features. However, with debris coolability alone, containment failure may be delayed for an extended period of time (more than 24 hours in some plants++) similar to the PWR arrangement. Water injection post

+ Or equivalent.
++ Plant specific value. Some Mark Is have substantially higher free air and water volumes relative to decay heat power than other Mark Is.

Table 3.3-1

ACCIDENT SEQUENCE SPECIFIC
POSTULATED
BENEFITS

SEQUENCE TYPE	INERTING	DEBRIS COOLABILITY	VENTING	DEBRIS BARRIER
SBO (Class IB)	Reduce the likelihood of explosive mixture inside containment, i.e., reduce the probability of an hydrogen deflagration that fails containment.	Post core melt: Effective in DW temperature control and debris coolability. Would assist in prevention of early containment failure due to: (1) over temperature of drywell shell; (2) drywell shell contact; and, (3) excessive non-condensible gas generation rates.	Provides a method to: - Prevent over-pressure failure of containment, i.e. effective means of removing heat and non-condensibles (long term heat removal and non-condensible gas control) - provides a release path which allows scrubbing of the radionuclide releases	Prevents the spread of molten material to the DW wall and possibly to the torus (Early Failure Prevention)
$T_M C_M C_2$ (ATWS) (Class IV)	Similar to above but containment failure or venting may change the effectiveness of inerting.	Debris coolability and fission product scrubbing from the DW atmosphere.	Provides a method of preventing core melt, if: - EQ in Rx Bldg is not affected - Water is available to provide continued makeup long term	Same as above.

RPV breach provides debris coolability which controls the drywell temperature, and limits the core concrete reaction reducing the non-condensible gas generation. In other words, the venting requirement and the debris barrier would not be required for accident scenarios such as station blackout. However, venting and debris barriers would not by themselves prevent containment failure in the intermediate time frame due to possible overtemperature or overpressure failure in the 2-10 hour time frame. Venting may by itself offer a method to reduce the radionuclide releases by relieving the pressure driving force from the drywell.

The complex nature of the processes deserves an integrated evaluation on a plant specific basis. Some detailed calculations of event timing, phenomenological effects, etc., would be required to determine all of the interactions between the various actions. Such detailed investigation is beyond the scope of this present analysis; however, the knowledge that such interactions may exist has been factored into subsequent discussions.

References

[3-1] Meeting between NRC (Bernero, Denton, Speis, Rosztozy) and BWR Owners, on June 16, 1986, Washington D.C.

[3-2] Technical Report 23.1 PB: Peach Bottom Atomic Power Station - Integrated Containment Analysis, Philadelphia Electric Co., Mark 1985.

Section 4
FEASIBILITY OF NRC IDENTIFIED ITEMS

As part of the evaluation of BWR changes raised by the NRC and summarized in Section 3, a summary of existing plant capability in these areas was undertaken by the BWROG. The summary of existing plant configurations can be used to:

A. Identify existing accident mitigation potential.

B. Identify the feasibility of the proposed changes by virtue of back fitting into the existing plants.

4.1 PLANT SPECIFIC EFFECTS

This section reviews some of the impacts of the plant specific factors on the proposed changes. The specific items addressed in this section include:

- o Procedures
- o Existing physical hardware
- o Existing plant configurations.

Based upon the review of participating utilities, a set of tables providing comparative information on plant specific features affecting the efficacy of the proposed changes are presented. The BWROG response was primarily from Mark I owners and therefore the tables summarize only the Mark I information.

These tables were assembled by the participating utilities over a very short time frame with minimal guidance. The IDCOR questionnaire provided to the BWR owners was not precise and did not explicitly request some of the information identified in the enclosed matrices. Therefore in most cases, values were not reported because of poor question format or complete absence of questions. In addition, the data has not been verified and should therefore be treated cautiously. It does, however, provide an indication of the relative diversity of containment designs which exists. These differences can make substantial differences in severe accident progression calculations depending upon the assumptions and models used.

The following tables are presented to summarize the results of the existing plant capability:

- o Existing plants (Table 4.1-1)
- o Hydrogen control (Mark I) (Table 4.1-2)
- o Debris coolability alternatives (Table 4.1-3 and Table 4.1-4)
- o Torus room configuration (Table 4.1-5)
- o Drywell configuration/debris barrier (Table 4.1-6)
- o Containment parameters (Table 4.1-7)
- o Containment venting (Table 4.1-8)
- o Emergency procedures (Table 4.1-9).

The responses cover 17 of the 25 Mark I plants.

Observations derived from the questionnaires to BWR owners and tabulated in the above tables can be summarized as follows:

Hydrogen Control (Mark I) (Table 4.1-2)

All Mark I's that responded are inerted. The technical specification requirements appear similar for nearly all Mark Is. For those plants with the information, inerting unavailability was reported at approximately 1%. The only differences were in the description of the power requirements to maintain an inerted condition for extended periods of time. These apparent differences may have arisen due to the ambiguous questionnaire.

Debris Coolability (Tables 4.1-3 and 4)

These tables provide the following insights:

- o There are a significant number of injection sources to the RPV that could provide debris coolability, post RPV breach
- o These sources are proceduralized and surveillance tested (in general)
- o AC power is required for all these sources
- o Drywell sprays are generally reported as available only from RHR pumps

Table 4.1-1
SUMMARY OF BWR TYPES

Plant	Product Line	Contmt Type	ECCS Complement		Containment Heat Removal Systems		
			High Pressure	Low Pressure	Safety Related	Normal	Contingency
Nine Mile Pt.	BWR/2	MK I	2 IC + FWCI	4 LPCS	2 IC	PCS(% bypass) shutdown heat removal HX(3)	
Oyster Creek	BWR/2	MK I	2 IC + EDFW	4 LPCS		PCS(% bypass)	
Millstone	BWR/3	MK I	1 IC + FWCI	2 LPCS + 2 LPCI	IC	PCS(120% bypass) RHR shutdown heat removal HX(2)	
Dresden 2 Dresden 3	BWR/3	MK I	1 IC + HPCI	2 LPCS + 4 LPCI	IC; RHR(2HX)	PCS(40% bypass); RHR shutdown heat removal	
Monticello	BWR/3	MK I	RCIC + HPCI	2 LPCS + 4 LPCI	RHR(2HX)	PCS(15% bypass); RHR shutdown heat removal	
Pilgrim	BWR/3	MK I	RCIC + HPCI	2 LPCS + 4 LPCI	RHR(2HX)	PCS(% bypass); RHR shutdown heat exchanger	
Quad Cities 1 Quad Cities 2	BWR/3	MK I	RCIC + HPCI	2 LPCS + 4 LPCI	RHR(2HX)	PCS(40% bypass); RHR shutdown heat removal	
Vermont Yankee	BWR/4	MK I	RCIC + HPCI	2 LPCS + 4 LPCI	RHR(2HX)	PCS(105% bypass); RHR shutdown heat removal	
Browns Ferry 1 Browns Ferry 2 Browns Ferry 3	BWR/4	MK I	RCIC + HPCI	2 LPCS + 4 LPCI	RHR(4HX)	PCS(25% bypass); RHR shutdown heat removal	
Peach Bottom 2 Peach Bottom 3	BWR/4	MK I	RCIC + HPCI	2 LPCS + 4 LPCI	RHR(4HX)	PCS(25% bypass); RHR shutdown heat removal	Containment Venting
Fitzpatrick	BWR/4	MK I	RCIC + HPCI	2 LPCS + 4 LPCI	RHR(2HX)	PCS(25% bypass); RHR shutdown heat removal	
Shoreham	BWR/4	MK II	RCIC + HPCI	2 LPCS + 4 LPCI	RHR(2HX)	PCS(25% bypass); RHR shutdown heat removal	
Cooper	BWR/4	MK I	RCIC + HPCI	2 LPCS + 4 LPCI	RHR(2HX)	PCS(% bypass); RHR shutdown heat removal	
Hatch 1 Hatch 2	BWR/4	MK I	RCIC + HPCI	2 LPCS + 4 LPCI	RHR(2HX)	PCS(25% bypass); RHR shutdown heat removal	
Brunswick 1 Brunswick 2	BWR/4	MK I	RCIC + HPCI	2 LPCS + 4 LPCI	RHR(2HX)	PCS(25% bypass); RHR shutdown heat removal	Spent fuel pool Hx; S.P. makeup and drain systems; RWCI systems (non-regenerative HX); HPCI recirc. to CST;

Table 4.1-1 (continued)
SUMMARY OF BWR TYPES

Plant	Product Line	Contmt Type	ECCS Complement		Containment Heat Removal Systems		
			High Pressure	Low Pressure	Safety Related	Normal	Contingency
Duane Arnold	BWR/4	MK I	RCIC + HPCI	2 LPCS + 4 LPCI	RHR(2HX)	PCS(25% bypass); RHR shutdown heat removal	
Hope Creek 1 Hope Creek 2	BWR/4	MK I	RCIC + HPCI	2 LPCS + 4 LPCI	RHR(2HX)	PCS(% bypass); RHR shutdown heat removal	
Fermi 2	BWR/4	MK I	RCIC + HPCI + MOTOR DRIVEN	2 LPCS + 4 LPCI	RHR(2HX)	PCS(25% bypass); RHR shutdown heat removal	
Limerick 1 Limerick 2	BWR/4	MK II	RCIC + HPCI	4 LPCS + 4 LPCI	RHR(2HX)	PCS(25% bypass); RHR shutdown heat removal	Containment venting;
Susquehanna 1 Susquehanna 2	BWR/4	MK II	RCIC + HPCI	4 LPCS + 4 LPCI	RHR(2HX)	PCS(25% bypass); RHR shutdown heat removal	
La Salle 1 La Salle 2	BWR/5	MK II	RCIC + HPCS	1 LPCS + 3 LPCI	RHR(2HX)	PCS(25% bypass); RHR shutdown heat removal	
Bailly	BWR/5	MK II	RCIC + HPCS	1 LPCS + 3 LPCI	RHR(2HX)	PCS(% bypass); RHR shutdown heat removal	
Hanford 2	BWR/5	MK II	RCIC + HPCS	1 LPCS + 3 LPCI	RHR(2HX)	PCS(% bypass); RHR shutdown heat removal	
Nine Mile Pt. 2	BWR/5	MK II	RCIC + HPCS	1 LPCS + 3 LPCI	RHR(2HX)	PCS(% bypass); RHR shutdown heat removal	
Grand Gulf 1 Grand Gulf 2	BWR/6	MK III	RCIC + HPCS	1 LPCS + 3 LPCI	RHR(2HX)	PCS(35% bypass); RHR shutdown heat removal	
Perry 1 Perry 2	BWR/6	MK III	RCIC + HPCS	1 LPCS + 3 LPCI	RHR(2HX)	PCS(35% bypass); RHR shutdown heat removal	
River Bend	BWR/6	MK III	RCIC + HPCS	1 LPCS + 3 LPCI	RHR(2HX)	PCS(10% bypass); RHR shutdown heat removal	
Clinton	BWR/6	MK III	RCIC + HPCS	1 LPCS + 3 LPCI	RHR(2HX)	PCS(35% bypass); RHR shutdown heat removal	

Table 4.1-2

HYDROGEN CONTROL POST CORE MELT (MARK I & II)

Plant	Containment Inerted	Tech Spec+	Dependency of Inerting on Power for 24 hrs	Unavailability of Inerted Containment (per yr) Planned	Unplanned	Oxygen Sources: Vacuum Breaker Rx Bldg - WW	Other In-leakage
Oyster Creek	Yes	24 hrs	No	Not Reported		X	Containment makeup line Containment spray test line Instrument air lines
Pilgrim MK I	Yes	24 hrs	None	2%	.06%	X	Service Air
Duane Arnold	Yes	24 hrs	None	Not Reported		X	
Dresden 2&3	Yes	24 hrs	Yes/Manual	Not Reported		2	
Quad Cities 1&2	Yes	24 hrs	Yes/Manual	Not Reported		X	
Vermont Yankee	Yes	24 hrs	None	1.2%		X	
Monticello	Yes	24 hrs	Yes	Not Reported		X	Service Air
Nine Mile Pt. 1	Yes	10 hrs	None	Not Reported		X	
Peach Bottom 2&3	Yes	24 hrs	Yes		1%++	2 (Actuation at .5 psid)	
Cooper	Yes	24 hrs	No	Not Reported		No	Service Air
Hatch 1	Yes	24 hrs	Yes	Not Reported		X	
Hatch 2	Yes	72 hrs	Yes	Not Reported		X	
Fitzpatrick	Yes	24 hrs	No	not Reported		X	Breathing Air
Fermi 2	Yes	24 hrs	Yes	No Experience		X	

\+ No. of hours after reactor is placed in RUN mode when containment must be inerted or when prior to shutdown it can be deinerted.

++ Limerick PRA.

Table 4.1-3

DEBRIS COOLABILITY

Plant	Drywell Injection RPV	DW Sprays	Emergency Procedural Guidance (Symptoms)	Power	Proceduralized	Surveillance	Inhibits on Injection Sources
Pilgrim	ECCS CRD	RHR Pumps	Yes Hi DW T&P	AC	Yes	Yes	Rx Level < 2/3
Duane Arnold	ECCS	RHR Pumps	Yes	AC	Yes	Yes	Not Reported
	CRD	Diesel Fire Pump		None	No	No	Not Reported
	DFP+	Fire Truck on-site		None	No	No	Not Reported
		off-site		None	No	No	Not Reported
Dresden	1 HPCI	RHR Pumps (4)	Yes	AC	Yes	Yes	Cont Level=30ft
	2 CS			AC	Yes	Yes	
	4 LPCI			AC	Yes	Yes	
	MFW			AC	Yes	Yes	
	2 DFP			None	Through MFW (yes)	No	Cont Level=30ft
	Fire Truck			None	(No)	No	Cont Level=30ft
Quad Cities	LPCI	RHR Pumps (4)	Yes	AC	Yes	Yes	Level < 27.5ft to permit S.P. Spray
	CS			AC	Yes	Yes	
	MFW			AC	Yes	Yes	
	(DFP to condenser only)						

+ Diesel Fire Pump.

Table 4.1-3 (Cont)

DEBRIS COOLABILITY

Plant	Drywell Injection RPV	DW Sprays	Emergency Procedural Guidance (Symptoms)	Power	Proceduralized	Surveillance on Injection Sources	Inhibits
Vermont Yankee	CRD LPCI CS MFW COND	RHR	Yes	AC	Yes	Yes	Not Reported
	DFP	DFP	No	LPCI Inj. Has dedicated DC Backup; all Other Valves Manual	Yes	Not Reported	
Monticello	LPCI CS COND. CRD RHRSW Keep Full	RHR	Yes (1-11-87)	AC	Yes	Yes	High Torus Temperature
Nine Mile Pt 1	CS CRD Condensate Raw Water	Cont. Spray + Raw Water	Yes	AC	Not Reported	Yes Yes Yes Yes	None
	DFP			None Reported	Yes; Spool Piece	No	None

Table 4.1-3 (Cont)

DEBRIS COOLABILITY

Plant	Drywell Injection RPV	Drywell Injection DW Sprays	Emergency Procedural Guidance (Symptoms)	Power	Proceduralized	Surveillance on Injection Sources	Inhibits on Spray
Peach Bottom	ECCS SLC CRD HPSW	RHR	EOPs	AC	Yes	Yes	-High Cont. Level
Cooper	ECCS CRD	RHR	EOPs	AC	Yes	Yes	DW Vacuum
Hatch 1&2	ECCS CRD	RHR	EOPs	AC	Yes	Yes	-2/3 Core HT Interlock -Torus Level Above Vacuum Brkrs. -Lack Of Instr.
Fitzpatrick	ECCS	RHR	EOPs	AC	Yes	Yes	
Fermi 2	RHR SW LPCI CS COND. SB FWP HTR. FD	RHR	EOPs	AC	Yes	Yes	None Reported
	DFP					No	
Oyster Creek	CRD COND/FW Core Spray SLC	RHR	EOPs	AC	Yes	Yes	High Drywell Pressure
	DFP					No	

Table 4.1-4

ALTERNATIVE INJECTION SOURCES TO CONTAINMENT

Plant	Diesel Fire Pump	To Spray	To MFW or RPV	GPM @ 60 psig In Cont.
Pilgrim	No	No	No	NA
Duane Arnold	Yes	Yes	Yes	2500
Dresden	Yes	No	Yes	NR+
Quad Cities	Yes	No	No	NR
Vermont Yankee	Yes	Yes	Yes	2500
Monticello	No	No	No	NA
Nine Mile Pt 1	Yes	No	No	2500
Peach Bottom	Yes	No	No	NR
Cooper	No	No	No	NA
Hatch	No	No	No	NA
Fitzpatrick	No	No	No	NA
Oyster Creek	Yes	No	Yes	NR
Fermi 2	Yes	NR	NR	NR

+ Not reported.

Table 4.1-5

TORUS ROOM CONFIGURATION

Plant	Torus Flood Barriers To Contain Water	Flood Equilibration Level Above T Quenchers	Above Down Comers	Above HPCI/RCIC Exhaust
Pilgrim	Yes	Yes	No	No
Duane Arnold	Yes	No	No	No
Dresden	Yes	Yes (marginal)	No	No
Quad Cities	Yes	Yes (Marginal)	No	No
Nine Mile Pt. 1	Yes	No	No	NA
Vermont Yankee	Yes			
Monticello	No	No	No	No
Peach Bottom	Yes	Yes	No	No
Cooper	Yes	Yes	No	No
Hatch 1	Yes	Yes+	No	No
Hatch 2	Yes	Yes+	No	No
Fitzaptrick	Yes	Yes	Yes	No
Ferml 2	Yes	No	No	No

+ Inferred

Table 4.1-6

DRYWELL CONFIGURATION DEBRIS BARRIER

Approximate Containment Parameters
Displaying Plant To Plant Variations

Power MW(e)	Plant	Radial Dist. To Shell (ft)	Entire DW Floor Area+ (ft^2)	Pedestal Floor Area (ft^2)	Sump Volume (ft^3)	Downcomer Height Above Floor	Possible Pathways to Torus for Molten Fuel	DW Floor Volume Below Down-Comer (ft^3)	Pedestal Curb ++
670	Pilgrim	23	1682	259	388	11"	None	1860	None
545	Duane Arnold	19	1174	214	NR(5)	1.3'	None	1526(3)+++	None
794	Dresden	22	1562	314	298	4"	None	515(3)	None
789	Quad Cities	22	1562	314	298	4"	None	515(3)	None
514	Vermont Yankee	20	1244	231	88(4)	9"	None	(933)(3)	None
536	Monticello	20	1252	231	245	11"	None	1148	None
610	Nine Mile Pt 1	23	1677	254	535	1.63'	None	2683	None
1065	Peach Bottom	20	1264	322	203	2'	None	2731	None
778	Cooper	20.4	1305	250	NR	1.75'	None	2283	NR
806	Hatch 1&2	19	1139	252	NR	1.5'	None	1705	None
620	Oyster Creek	~24	1550	250	NR	2'	None	3100	6" Curb at doorway

\+ Includes Pedestal Floor Area.
++ Includes sump volume.
+++ Sump volume not reported.
(3) Without any sump volume included.
(4) An additional 500 ft^3 sump exists on the opposite side of the DW.
(5) Not reported.

Table 4.1-6 (Cont)

DRYWELL CONFIGURATION DEBRIS BARRIER

Approximate Containment Parameters
Displaying Plant To Plant Variations

Power MW(e)	Plant	Radial Dist. To Shell (ft)	Entire DW Floor Area+ (ft^2)	Pedestal Floor Area (ft^2)	Sump Volume (ft^3)	Downcomer Height Above Floor	Possible Pathways to Torus for Molten Fuel	DW Floor Volume Below Down-Comer (ft^3)	Pedestal Curb
821	Fitzpatrick	21.6	1459	250		1'	None	1459	Not Rep.
1100	Fermi 2		1365	349	307	1'7"	None	2161	None

Table 4.1-7(a)

CONTAINMENT COMPARISONS

Plant	Bulb Radius ft	Gap Thickness in.	Free Air Volume ft^3	Radius of Torus ft	Torus Thickness in	Water Volume ft^3
Pilgrim	32	2	267,000	14.75	5/8	94,000 max
Duane Arnold	31.5	2	203,700	12.9	.5 to .534	58,900(normal) 61,500(accident)
Dresden	33	2	275,000	15	.585(Top) .653(Bottom)	114,000 112,200 (min)
Quad Cities	35	2	275,500	15	.585(Top) .653(Bottom)	114,000 112,000 (min)
Vermont Yankee	31	NR+	247,000	13.8	NR	69,000
Monticello	31	NR	242,450(max) 237,540(min)	13.8	2/3	72,910 max 68,000 min
Nine Mile Pt 1	NR	2	344,000	NR	.46	85,000
Peach Bottom	33.5	2	291,000	15.5	.6	123,000
Cooper	32.5	1 3/4	NR	NR	.6	NR
Hatch 1 & 2	32.5	2	NR	14	NR	NR
Fitzpatrick	32.5	NR	NR	14.75	NR	NR
Fermi 2	34	2	NR	15.25	.587(Top) .658(Bottom)	NR
Oyster Creek	35	3	NR	15	NR	NR

+ Not Reported

Table 4.1-7(b)

CONTAINMENT COMPARISONS: RATIOS OF CONTAINMENT DESIGN PARAMETERS TO MW(t)

Plant	Power MW(t)	Free Air Volume(ft^3) / Power (MWt)	Torus Water(ft^3) / Power(MWt)	Pedestal Sump Volume(ft^3) / Power(MWt)	Floor Volume(ft^3) / Power(MWt)
Pilgrim	1998	134	47	NR	.93
Duane Arnold	1593	127	37	NR	.96
Dresden	2527	109	45	NR	.20
Quad Cities	2511	109	45	NR	.21
Vermont Yankee	1593	155	43	NR	.60
Monticello	1670	144	42	NR	.69
Nine Mile Pt 1	1850	206	50	NR	1.60
Peach Bottom	3293	88.	37	NR	.82

From a severe accident standpoint only, larger ratios would appear to be better. Larger ratios imply the possibility of additional passive containment capability.

Table 4.1-8
CONTAINMENT VENTING

Plant	Do Procedures Exist to Vent Containment	Vent Valves (Prioritized)	Effective Venting by Sequence: ATWS	TW	SBO	All Severe Accidents (pres/temp)	Power	Location for Opening	Max Opening Pressure (psig)	Release in Rx Bldg
Pilgrim	Yes	2 @ 2"		X			AC	Local/Remote	70	Duct
		2 @ 1"					DC	Local/Remote	56	At SGTS
		2 @ 8"*	Marginal				AC	Local/Remote	75	Suction
Duane Arnold	Yes	24" *	X	X	X		DC/Air	Remote	?	Not
		2 @ 2"		X	X		DC/Air	Remote	?	Reported
Dresden	Yes	2"					120VAC	Control Room	125 psig	Yes
		18"	X	X			120VAC	Control Room	125 psig	
Quad Cities	Yes	2"					120VAC	Not Reported	125 psig	Yes
		18"	X	X			120VAC		125 psig	
Vermont Yankee	Yes	1"/2" **		X	X				62 psig	No
		3" **	X	X					No	
		18" *	X	X	X	X				
Monticello	Yes	Torus 2"	X			X	Air +	Control Room	56	Yes
		18" (Blocked at 40°) DW same	Marginal	X	X	Marginal	120VAC	Control Room	56	
Nine Mile Pt.	Yes	4" *		X	X		N2	Control Room	Not Reported	65 psig No
		1" & 4" *	X	X		Air	Control Room	Not Reported	65 psig No	
		3" & 4" *	X	X		Air	Control Room	Not Reported	35 psig ATWS Yes	
		12" & 24"*	X	X	X	X	AOV + MOV	Control Room	Not Reported	
Peach Bottom	Yes	Torus 2"					AC	CR	Not Reported	Stack
		6" (ILRT)		X	X	X	Manual	Local	Not Reported	Environment
		18" (Seals WW)					Manual	Local		
		18" (Seals WW)	X	X	X	X	AC/MOV	CR or Local		Rx Bldg
		DW same								

*Identifies valves which are not explicitly included in the current venting procedure.
**Venting precluded if high radiation exists.

Table 4.1-8 (continued)
CONTAINMENT VENTING

Plant	Do Procedures Exist to Vent Containment	Vent Valves (Prioritized)	Effective Venting by Sequence: ATWS	TW	SBO	All Severe Accidents (pres/temp)	Power	Location for Opening	Max Opening Pressure (psig)	Release in Rx Bldg
Fermi	No	None					NA	NA	NA	NA
Cooper	Yes	WW 1"					125VDC	Local & Remote	50 psi	SGTS
		DW 1"					125VDC	Local & Remote	50 psi	SGTS
		WW 2"					AC	Local & Remote	62 psi	Rx Bldg
		DW 2"					AC	Local & Remote	62 psi	Rx Bldg
		WW 24"	X	X	X	X	AC/Air	Local & Remote	62 psi	Rx Bldg
		DW 24"	X	X	X	X	AC/Air	Local & Remote	62 psi	Rx Bldg
Hatch	Yes	DW 2"					Onsite AC	Control Room	56 psig	SGTS
		DW 2"					Onsite AC	Control Room	56 psig	SGTS
		WW 2"					Onsite AC	Control Room	56 [illegible]	[illegible]GTS
		WW 2"					Onsite AC	Control Room	56 psig	SGTS
Fitzpatrick	Yes	24"	X	X	X	X	AC/Air	Remote	56 psi	Rx Bldg
		24"	X	X	X	X	AC/Air	Remote	56 psi	Rx Bldg
		20"	X	X	X	X	AC/Air	Remote	56 psi	Rx Bldg
		20"	X	X	X	X	AC/Air	Remote	56 psi	Rx Bldg
		2"					DC	Remote	>56 psi	Rx Bldg
		2"					AC	Remote	>56 psi	Rx Bldg
		2"					AC	Remote	>56 psi	Rx Bldg
		2"					DC	Remote	>56 psi	Rx Bldg
Oyster Creek	Yes	2" **					DC/Air	Control Room + Local	58 psi	SGTS
		2" **					DC/Air	Control Room + Local	58 psi	SGTS

...ntifies valves which are not explicity included in the curren... ...ting pocedure.
...ating precluded if high radiation exists.

Table 4.1-9

EMERGENCY PROCEDURE STATUS

Plant	Current	Planned Rev. 4 Implementation	Significant Differences of Plant Specific & EPGs
Pilgrim	Rev 2	Not Planned	NR
Duane Arnold	Rev 3	W/in 1 yr of Approval	NR
Dresden	Rev 3	W/in 1 yr of Approval	NR
Quad Cities	Rev 4S		More explicit
Vermont Yankee	Rev 3L before 1-11-87	Not Planned	PC/H Not implemented
Nine Mile Pt. 1	Rev 4AC	After Approval	NR
Peach Bottom	Rev 2	Fall 1987	None
Cooper	Rev 3I	After Approval	None
Hatch	None	Not Reported	NA
Fitzpatrick	Rev 3	After Submittal	None
Fermi 2	Rev 3A	April 1987	None
Oyster Creek	Rev 2-3 (Hybrid)	October 1986	None

*Identifies valves which are not explicitly included in the current venting procedure.
**Venting precluded if high radiation exists.

the containment free volume airspace and torus water volumes are lower at Peach Bottom and Browns Ferry on a per MW(t) basis than other Mark I plants. IDCOR and the NRC, therefore, appear to be examining the most restrictive Mark I parameters for those phenomena governed by available containment air and water volumes. Nevertheless, Peach Bottom has been shown in IDCOR to have acceptable calculated public risk and severe accident response. Other Mark I plants are therefore expected to have even better response to severe accidents in those areas which would be governed by available free air volume and torus water inventory.

Also included in this section as Figure 4.1-1 is a typical drywell configuration. This figure can be used in conjunction with information contained in Table 4.1-6 to examine plant-to-plant variations in the drywell design. Selected parameters which influence the impact of drywell designs are illustrated on Figure 4.1-1 and include the following.

- Distance between downcomer and DW floor
- Distance between bottom of sump and shell
- Depth of sump
- Location(s) of doorway(s)
- Arrangement of sumps

4.2 INFLUENCES OF SEVERE ACCIDENT PHENOMENA ON THE EVALUATION OF THE PROPOSED NRC ITEMS

The phenomena of severe accidents affects not only the types of concerns to be addressed, but the possible options within each concern well. This section discusses the influence of the severe accident phenomena upon such issues as venting capacity and barrier options. Much of this discussion is taken from Reference 4-1.

4.2.1 Hydrogen Control

The issues related to hydrogen control which have been addressed in the safety evaluation are as follows:

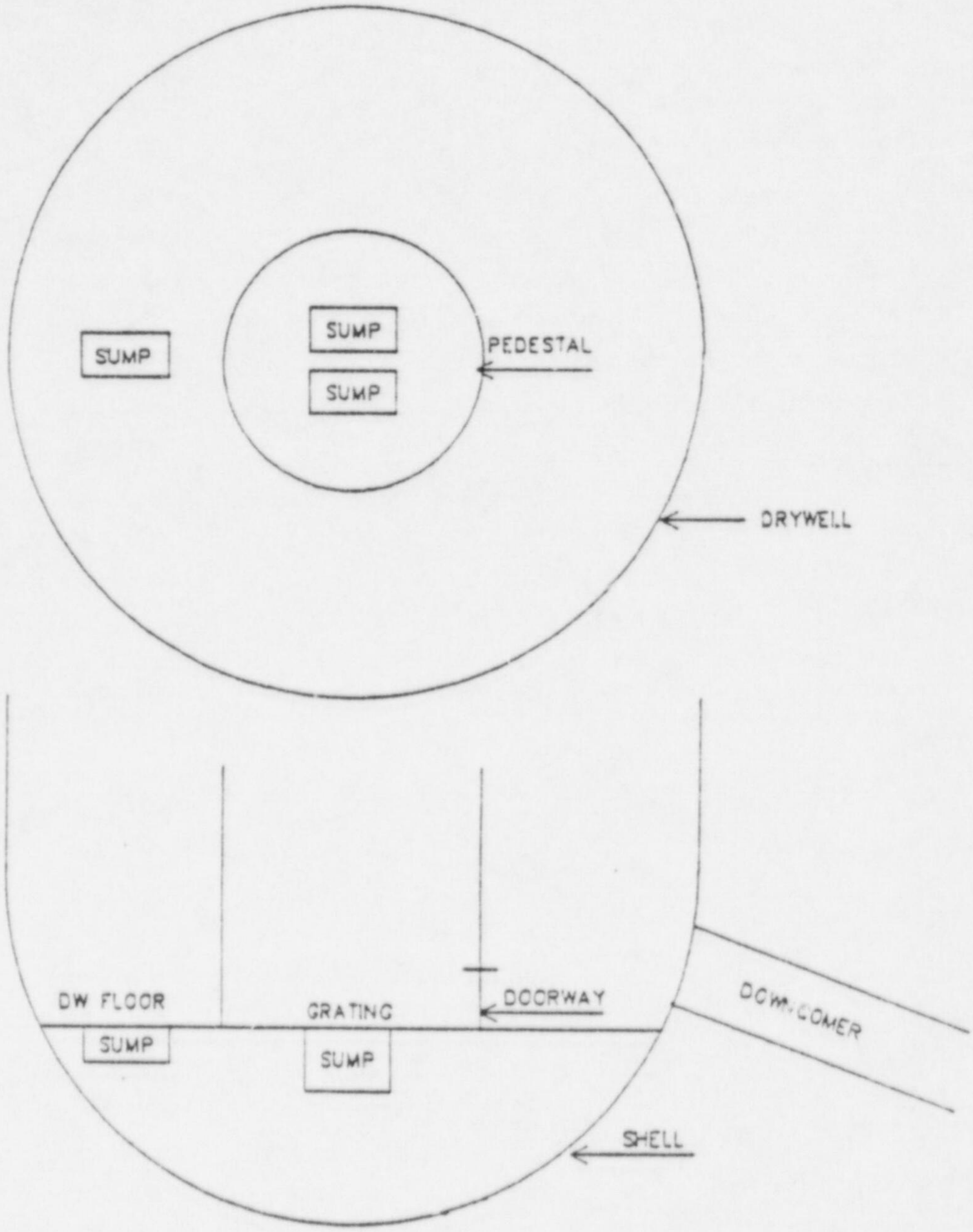


Figure 4.1-1 Typical Drywell Configuration

- Mark III hydrogen control: HCOG evaluation provided separately

- Mark I and II hydrogen control:

  - during inerted conditions
  - during Mark I reactor building to wetwell vacuum breaker operation

This section on phenomenology addresses only the Mark I and II plants.

The inerting of the Mark I and II plants provides a reliable capability to prevent hydrogen burning during severe accidents by virtue of the fact that combustible mixtures are inhibited in the inerted containment. NRC experiments show that greater than 5% oxygen is required to support combustion.

Therefore, the inerting of Mark I and II plants coupled with the minimum observed unavailability of such inerted operation in operating plants results in acceptable severe accident response, i.e., small probability of hydrogen combustion leading to containment failure.

This conclusion holds as long as there are not alternative means of inducing oxygen addition during the course of severe accidents. The three principal mechanisms which have been postulated are:

- <u>Service air</u> ruptures inside containment (causing or induced by severe accidents) that cannot be isolated

- <u>Venting</u> allowing reverse flow from outside into the containment

- <u>Reactor building to wetwell vacuum breakers operation</u>.

These postulated cases are dealt with as follows:

- Service air ruptures inside containment are viewed as a plant specific system feature that should be assessed on a plant specific basis, and is not judged to be generically applicable.

- Venting allowing reverse flow is not judged to be a realistic scenario if the operator follows the proposed guidelines of not reducing containment pressure via venting significantly below containment design pressure. In addition, it is judged appropriate to have restrictions on the simultaneous spraying and venting operations. These matters can be addressed in severe accident management procedures.

- Reactor building to wetwell vacuum breaker operation (Mark I containments only) has been demonstrated through conservative calculations [4-1] to allow the introduction of up to a maximum of 5% oxygen in the worst case. Such calculations suggest that even for the worst postulated severe accident conditions only 5% oxygen can be reintroduced through the reactor building to wetwell vacuum breakers. Because there has not been a complete sensitivity study for all severe accident conditions, and because the conservative modeling has not included all the details of a dynamic spray model, there remains some uncertainty regarding worst case oxygen introduction. Nevertheless, there appears to be high confidence [4-1] that only non-combustible quantities of oxygen can be introduced.

In addition to the above observation, it has been observed that for slow hydrogen there is potentially no threat to the Mark I and II containments. Therefore, even for cases involving combustible mixtures in the containment, containment failure would still be maintained if combustion occurred at a constant rate limited by the hydrogen generation rates [4-9].

In summary, hydrogen burning is not evaluated as a potential containment issue for inerted Mark I and II containments.

4.2.2 Venting Capacity

Containment venting may have strong impacts on the calculated radionuclide releases to the environment.

IDCOR analyses [4-3] have shown a large reduction in the fission product release to the environment as a result of venting though the wetwell gas space. Table 4.2-1 summarizes some of the cases run with venting for the Peach Bottom TC sequence. These are estimated source terms developed by FAI using available MAAP calculations for Mark I plants [4-1]. Tables 4.2-2 and 4.2-3 [4-1], are also estimated radionuclide releases for dominant types of severe accident sequences. These estimates are for two different postulated locations of containment failure:

- Drywell and
- Wetwell (bellows)

Containment venting can be used in a number of postulated severe accidents to prevent uncontrolled radionuclide releases. The required containment vent size varies with the accident sequences type. For the spectrum of accidents considered in the IDCOR reference plant analyses, the venting requirements include ATWS conditions, like those addressed in the Reactor Safety Study (WASH-1400), venting of decay heat and venting of noncondensible gases generated during core-concrete thermal attack. Of these, the ATWS conditions are the most severe since the venting capability is required to meet from 20-30% of the nominal core power vented as steam at pressures in the range of 0 to 100 psig. As will be shown, this requires a large vent. The required venting capacity can be calculated based upon the vent line configuration (see Figure 4.2-1) and the containment pressure when venting is initiated.

The required vent size as a function of the power to be vented is shown in Figure 4.2-2. For this sample calculation, the vent line is assumed to be 100 m long with an average friction factor of 0.02. As illustrated, vent sizes for power levels of several hundred megawatts are 60 cm (2 ft) in diameter or larger, whereas those required for decay heat would be 10-15 cm (4-6 in.). These calculations clearly call attention to the definition of the ATWS sequence and its relevance to the plant specific Emergency Operating Procedures.

The use of containment venting as an effective mitigator in reducing radionuclide source terms has been shown in IDCOR MAAP analyses to offer potentially significant advantages even for ATWS related sequences. There may, however, be enough uncertainty in the alternative mitigation measures now under development, that the exact methods of implementing: (1) containment venting and (2) the alternate ATWS mitigation strategies, should await the final analytic calculations and generic procedural development. The alternative ATWS strategies relate to starving the core and thereby drastically reducing core power, potentially within the RHR capacity.

This area of uncertainty makes immediate implementation of an ATWS venting scheme a potential resource expenditure that will have to be duplicated when the further analytic work is complete. (See discussion in [4-1] and under "other considerations" in Section 5.)

Table 4.2-1

CsI DISTRIBUTION AND RELEASE
(FRACTION OF INITIAL CORE INVENTORY)

	TC - Vent No Debris Cooling	TC - Vent Debris Cooling	TC - No Vent No Debris Cooling	TC - No Vent Debris Cooling
(Time, hrs.)	60	60	60	15*
RPV	.01	.25	.04	.27
Drywell	0.0	0.0	0.0	.02
Suppression Pool	.60	.75	.27	.55
Secondary Containment	.36	2×10^{-4}	.56	.13
Environment	.03	6×10^{-4}	.13	.03 +

*At 15 hours the major fission product release has occurred.

+Note that the current MAAP modeling does not account for fission product retention due to drywell sprays which may represent a significant conservatism in the ATWS source term evaluation.

Containment Overpressure Failure Location

→ Case I : DW
Case II: WW (Bellows)

Table 4.2-2
RADIONUCLIDE RELEASES

	No Venting		Venting	
Sequence	No Water Injection	Water Injection	No Water Injection	Water Injection
$T_MC_MC_2$	13% at 4 hours (Ref.1)	3% at 4 hours (Ref. 1)	Noble Gas	Noble Gas
T_MQUX	5% at 18 hours (Ref. 1)	< 2% (Ref. 3)	Noble Gas	Noble Gas
T_EQUV	1% at 35 hours (Ref. 2)	< 2% (Ref. 3)	Noble Gas	Noble Gas
TW	20% at 40 hours (Ref. 1)		Noble Gas	Noble Gas

*CsI release fraction to environment.

Ref. 1: IDCOR Task 23.1 Report (MAAP 2.0).
Ref. 2: Recent MAAP 3 calculations.
Ref. 3: Estimation based on other MAAP runs.

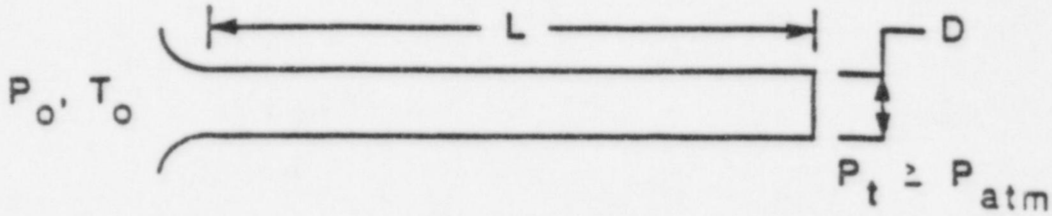
Table 4.2-3

RADIONUCLIDE RELEASES

Containment
Overpressure
Failure
Location

Case I : DW
→ Case II: WW (Bellows)

	No Venting		Venting	
Sequence	No Water Injection	Water Injection	No Water Injection	Water Injection
$T_M C_M C_2$	Noble Gas	Noble Gas	Noble Gas	Noble Gas
$T_M QUX$	Noble Gas	Noble Gas	Noble Gas	Noble Gas
$T_E QUV$	Noble Gas	Noble Gas	Noble Gas	Noble Gas
TW	Noble Gas	Noble Gas	Noble Gas	Noble Gas



$$\Delta P_{inertial} = \frac{W^2}{2\,\rho_g\,c_d\,A^2}$$

$$\Delta P_{friction} = f\,L/D\,\frac{W^2}{2\,\bar{\rho}_g\,A^2}$$

$$P_o - P_t = \frac{W^2}{2\,\bar{\rho}_g\,A^2}\left[\frac{1}{c_d} + f\,L/D\right]$$

Figure 4.2-1 Vent Capacity Determination

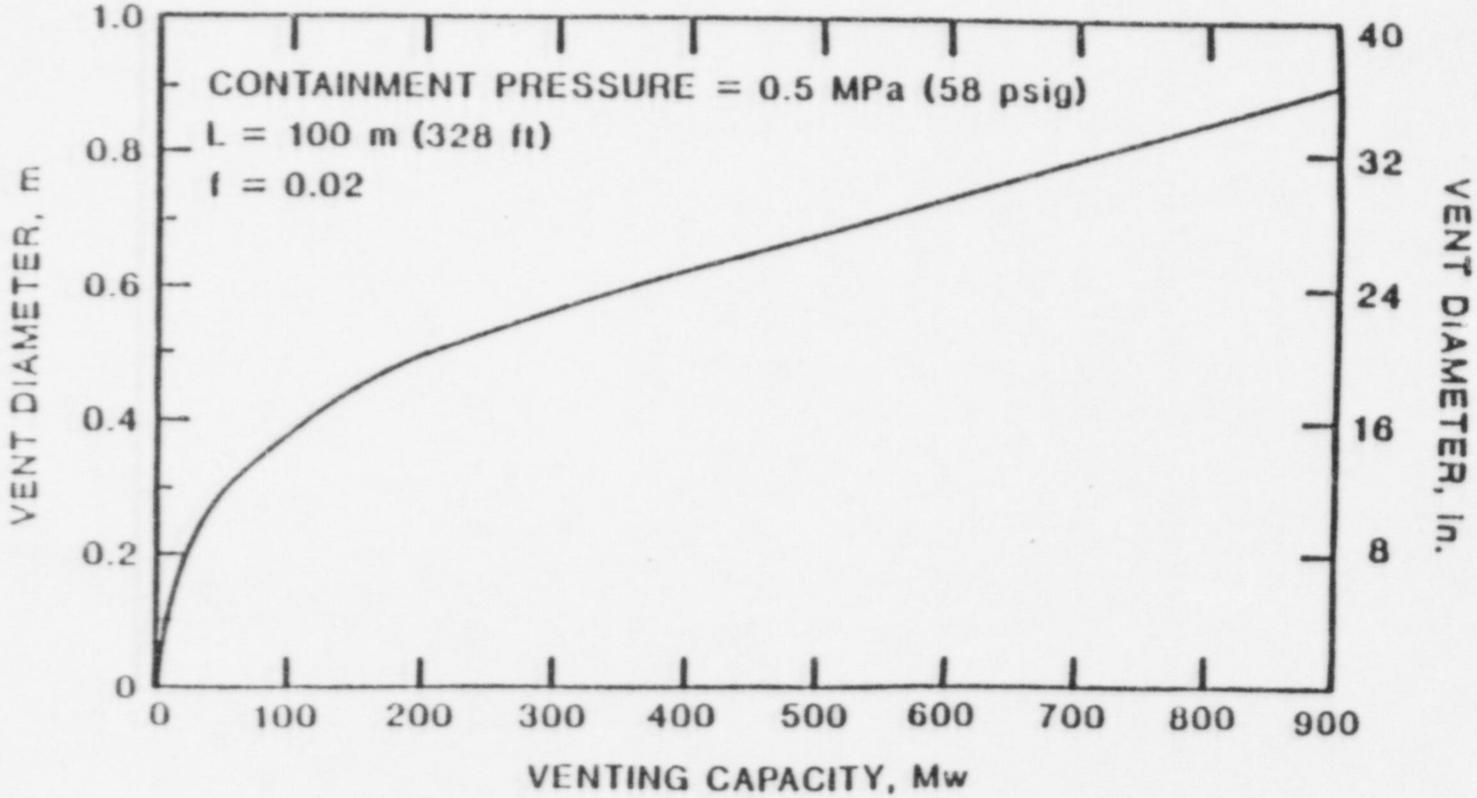


Figure 4.2-2 Vent Size Requirement as a Function of Power

4.2.3 Debris Coolability

This sub-section discusses the following:

- o General debris coolability requirements
- o Specific issues related to drywell sprays

General Considerations

For accident sequences in which the containment is intact but the core has melted, the drywell coolant injection has one major function: cooling the structure and the debris. For a typical Mark I plant a flow of 250 gpm is required to remove decay heat from the entire inventory of core material. This assertion is of course somewhat dependent upon the geometry of the core debris in the drywell (and sumps) and the relative location of coolant injection discharge into the drywell. This is discussed below.

The steam generated by debris cooling will force any airborne fission products into the suppression pool where they will be successfully scrubbed. Debris cooling can be achieved not only by the initiation of drywell sprays but by the injection of water either into the failed reactor pressure vessel or into the drywell. Systems that are potential candidates for spraying or injecting into either the vessel or drywell are:

- o Feedwater
- o Emergency Core Cooling Systems (ECCS)
- o Control Rod Drive (DRC)
- o Service water
- o Fire pump
- o Condensate pump
- o Standby Liquid Control (SLC)

For sequences which progress to reactor vessel failure, debris would be released to the pedestal and drywell. Depending upon the accident scenario water may or may not be accumulated prior to this time. If water is available, the heat removal from the debris typically is 1-2 Mw/m^2. Hence, if the debris covers about 100 m^2 (1200ft^2) of drywell floor, the heat removal far exceeds decay heat. However, water must be available on a continual basis to cool the debris.

This could be accomplished by injecting to the drywell at a rate which exceeds decay heat. Flow rates more than this amount would begin to accumulate water in the drywell which would spill over into the torus, and debris coolability would be assured.

Fission product removal can be achieved through several mechanisms. First cold water injected through the containment sprays would condense steam and this would sweep vapors and aerosols to the spray droplets. Secondly, the spray droplets would impact aerosols during their fall and this would also capture fission product aerosols. Lastly, steam formed by debris quenching or decay heat would flow to the suppression pool and sweep airborne material along as well. These vapors and aerosols would be scrubbed by the suppression pool.

If water is available via drywell injection, both the containment boundary and the debris could be cooled by the water. Consequently, significant heating of the containment shell would not occur and containment integrity would be maintained.

Lastly, if drywell injection is available to cool debris and heat removal is activated in the suppression pool, a heat transport path would be established between the debris and the pool. In this case the containment could be maintained intact.

<u>Drywell Spray Related Issues</u>

One possible method of debris coolability is through the use of drywell sprays. The drywell sprays have a number of considerations which have not yet been fully explored. These considerations include:

- Positive influences:

  - Remove fission products from the atmosphere: This may require a high flow rate to get an adequate spray configuration.

  - Cool the debris: A relatively slow flow rate is acceptable as described above. This is true unless the pedestal is isolated from the drywell. Then flooding may be a required option.

o Negative influences:

- rate and magnitude of depressurization: Rapid drywell depressurization could result in vacuum breaker cyclic operation. NUTECH [4-7] has performed sensitivity studies which indicate that drywell spray initiation under moderate containment conditions (compared with severe accidents) could create postulated conditions for which "containment capabilities would be at marginal limits..." This of course is with some conservative modeling assumptions. FAI [4-1,4-3,4-9] has performed an extensive library of sensitivity calculations and performed a number of bounding calculations to isolate the phenomena which may result in potentially adverse conditions. The results of these evaluations can be summarized briefly that in order for drywell sprays to induce a potential serious problem during a severe accident all of the following are required:

  -- a high spray flow rate
  -- a nearly instantaneous spray startup
  -- cold spray water
  -- adverse containment conditions.

  Therefore, while it is judged that drywell spray initiation would not adversely impact vacuum breaker operation or containment integrity, nevertheless this remains at this time a potential negative safety impact for full flow drywell sprays during a severe accident. (Note that current procedural limitations are sufficiently restrictive to both preclude spray operation in worst case scenarios and preclude spray during other severe accidents.)

- implosion (see above)

- introduction of O_2 into the containment: rapid drywell or suppression pool spray initiation from external sources (not currently considered in the EPGs) could lead to the reactor building vacuum breakers opening out allowing the introduction of O_2 into the containment. If flammable mixture could be induced, then a hydrogen deflagration could occur. While the precise condition (H_2 and O_2 mixtures, no steam inerting, and sudden ignition source) are unlikely, this must also be listed as a potential phenomena which could complicate severe accident mitigation.

4.2.4 Barrier Options

Analyses performed by Brookhaven National Laboratory with support by Oak Ridge National Laboratory [4-4] concluded that under specific accident conditions for a 100 MWe there would be a potential in a Mark I containment to have an early drywell shell failure due to hot core debris contacting the containment wall.

Assumptions in the sensitivity analyses that led to such a conclusion were:

1. 60-80% of the core inventory spread uniformly over the pedestal and drywell floor over a short time (seconds), during the initial RPV breach,

2. both basaltic and limestone concrete types, and

3. initial corium temperatures of 1775K, 1900K, and 2550K.

It should be noted that not each of the sensitivity cases performed by BNL led to drywell shell failure from this phenomena. For the lower assumed debris temperature cases with basalt concrete no drywell liner failure is calculated (even for the case with 80% of the core initially dispersed in the drywell).

IDCOR analyses performed to date have shown that at the time the lower portion of the core melts and slumps into the lower plenum of the reactor pressure vessel approximately 20-30% of the core is molten. This means that at vessel failure approximately 20-30% of the original core inventory could flow out into the containment. This sequence progression could be followed by melting of the remaining core over the next 3-5 hours. This core melt progression differs from the 60-80% instantaneous melt discharge used in the above Brookhaven calculation and has a very strong influence on drywell shell failure. Furthermore, after the core material is expelled from the RPV, water remaining in the lower plenum (~ 200,000 lbs.) will flow out and cool the core debris. The correct debris temperature history would therefore be one in which the initially expelled debris mass is cooled by the lower plenum water and then slowly heats up as the water is boiled away. This scenario influences the calculations performed by Brookhaven and would tend to result in lower debris temperatures. As shown by BNL, lower debris temperatures tend to result in no shell failure. The IDCOR analyses suggest that the Brookhaven assumptions on debris mass and temperature for the drywell shell failure analyses are overconservative and should be replaced with more realistic boundary conditions.

There are also other mitigating features of BWR plants which tend to suggest that the BNL models may be overly conservative because they conservatively model phenomena such as:

- In-vessel non-melted segments of the core, particularly the outer ring of fuel assemblier, may persist for extended periods of time following RPV breach.

- The large mass of solid structural and CRD material that exists below the RPV and which is at relatively cool temperatures when the molten material is assumed to be released from the RPV can have a dramatic impact on reducing the debris temperature.

- The approximately 30 foot drop from the RPV to the drywell floor during which time the molten debris will be cooling can also cause low debris temperatures.

- Water may be present in the lower plenum that would also be released at the time of RPV breach and would act as an additional quenching agent on the drywell floor.

Simplistic calculations by Cohen [4-6] have identified two phenomena which could be expected to result in releases of the molten material into the drywell which would not be sufficient to flow to the drywell shell wall. These two phenomena are as follows:

- Because of radiative heat transfer from the hotter regions of the core to the cooler regions, it seems reasonable to conclude that at best the outer fuel assemblies would not melt, and similarly there would be at least 1 ft of unmelted fuel at the top and bottom of the core. This latter would seem to represent something of an obstruction to large masses of molten fuel falling out of the core.

- The molten fuel distribution in the drywell may result in spreading material over a large area, "if there were plenty of water available, the fuel could be effectively cooled... However, if all of the fuel collects in relatively small sumps, thermal conductivity through the thickness of the fuel is not sufficient to prevent remelting of its inner portion."

The above observations support the similar IDCOR calculations that molten material on the drywell will quickly solidify as its radial mass decreases to thin layers.

Despite the above phenomena which argue against molten material from reaching the drywell shell, such an evaluation is performed [4-1] which includes the influence of direct contact of the core debris on the steel shell as well as an assessment of the quenching potential for the core debris, should water be available as a result of the accident sequence via discharge from the reactor system at the time of vessel melt-through or from activation of the drywell sprays. In the assessment [4-1], the core debris is first assumed to be in direct contact with the steel containment shell in the absence of water. This simplified, conservative calculation demonstrates that the conduction along the steel shell is sufficient to keep the temperature of the shell substantially below the stainless steel melt point. An evaluation of the thermal response assuming that core debris comes in direct contact with the steel shell and is simultaneously quenched by an overlying pool of water is also performed. As demonstrated in this calculation, the steel shell would have substantial margin to ablation failure.

Using the "worst case" models and calculating a containment shell failure at the drywell floor joint does not mean that the public health and safety are unacceptably affected. In fact, from an overall risk perspective, such containment failure modes can be characterized as:

- o Refreezing of molten material outside the drywell shell
- o Release of fission products through constricted spaces with high deposition rates and potential for plugging
- o Release of fission products into the torus room where the most effective fission product decontamination associated with the reactor building can take place.

These are all factors which result in radionuclide source terms equivalent or below those assessed for radionuclide leakage release categories in the IDCOR Task 21.1 for BWR Mark I containments. These were shown to have small associated risk components.

Debris coolability from any drywell injection source has the ability to remove uncertainties associated with drywell shell failure resulting from core debris contact. Water injection onto the debris ex-vessel will avoid the threat of shell failure by cooling the debris, keeping the drywell cool, and forcing fission products into the suppression pool.

In summary, the deterministic calculation by FAI [4-1] demonstrate:

- o Small amounts of core material are anticipated to be released at RPV breach.

- Rapid cooldown of the material is calculated on the drywell floor, which would inhibit the chances of reaching the drywell shell.

- Heat transfer calculations indicate the drywell shell remains structurally adequate because of high heat conductions rates away from the debris interface.

- Any water availability would enhance the capability of the drywell shell to remain at low temperatures.

4.3 FEASIBILITY SUMMARY

The information compiled to date indicates that there is a great deal of similarity among BWRs in the NSSS equipment. However, the BOP systems (and to a moderate degree containment systems) can vary among the Mark I and II plants by some significant yet subtle differences. Some of these differences were previously noted during the Mark I containment loads program [4-10,4-11]. Again in the severe accident evaluation it can be found that there can be substantial differences among Mark I and II plants that can affect the assessment of severe accident phenomena and timing. These features include but are not limited to the following:

- ratio of core power to drywell volume

- ratio of core power to drywell floor area

- volume of drywell sumps and floor volume below the downcomers in Mark I

- arrangement of pedestal in Mark II plants

- diesel fire pump capability to inject to the RPV and/or drywell with a high reliability during all postulated sequences including station blackout.

Based upon the review of the responding plants, the following qualitative conclusions can be reached regarding the feasibility of the suggested changes:

<u>Hydrogen Control</u>

From the sampled plants examined, all Mark I plants are inerted and therefore have an effective mitigation against hydrogen deflagration during severe accidents by limiting the presence of oxygen. The introduction of oxygen into the containment during a severe accident through the reactor building to wetwell vacuum breakers is shown through simple, conservative calculations to not present a realistic threat. The issue however, is included to identify potential areas of uncertainty.

No additional action is judged necessary for the Mark I and II plants other than inerting the containments to meet the technical specifications.

Containment Venting

The plants responding to the questionnaire have varying capability to vent via installed and analyzed hardware coupled with variability in the methods incorporated into procedures.

From an analytic viewpoint, the IDCOR calculations have indicated the benefits associated with venting to be:

- Preservation of containment functionability (prevent drywell breach and maintain a scrubbed fission product pathway)
- Prevention of core melt

However, there are some uncertainties regarding hazards induced by venting radionuclides (noble gases) and venting during an ATWS. These two areas may require additional technical evaluations (possibly plant specific) before an optimized conclusion could be reached. In the interim, either situation appears to have public safety advantages although the existing IDCOR risk calculations would tend to support the position of venting prior to reaching containment failure pressure [4-7, 4-8]. Thus venting is a feasible alternative which is within the capability of some plants.

Debris Coolability

All the plants examined have redundant and diverse methods to inject water into the drywell through the breached RPV following a postulated core melt. In addition, all plants examined have the ability to spray the drywell directly using RHR pumps.

The phenomenological calculations [4-1, 4-3] indicate that either of the above methods of debris coolability can fulfill moderate the drywell temperatures and minimize any concrete attack or molten debris spreading on the drywell floor. Therefore, the intent of debris coolability as established in the deterministic calculations [4-1, 4-3] can be supported by:

- o RPV coolant injection post breach
- o Drywell sprays at relatively low flow rates.

First, addressing the drywell spray capability:

EPG Revision 4AE specifies that drywell spray initiation (RHR pumps only) could be triggered by any of the following:

- o H_2 or O_2 concentration
  - undetermined
  - > 6%
- o Suppression chamber pressure > 17.5 psig
- o Temperature (DW) < 340°F (281° for Peach Bottom), but approaching 340°

Based upon Revision 4AE of the EPGs, which appear to have somewhat relaxed the restrictions on DW spray usage, termination of drywell sprays occurs for <u>any</u> of the following reasons:

- o DW pressure below 2 psig
- o DW temperature high (> 340°F)
- o Suppression pool level above 17', i.e., above the wetwell to drywell vacuum breakers
- o No RHR pumps
- o Violation of DW spray curve (essentially DW temperature > 340°F)

Some select plants also have the capability to use external water sources via the service water or diesel fire pump to inject into the drywell sprays. However, very few plants have the procedures, access, or power capability to perform these actions under the conditions where other sources would not be available.

Therefore, the generic BWR condition is that drywell spray capability exists; but for the limiting station blackout scenarios this capability may be defeated because of lack of power to the necessary valves and the lack of access to those valves due to high radiation levels.

In addition, based upon a review of existing procedures by participating utilities there may be a number of existing restrictions on the implementation of drywell sprays. Table 4.3-1 provides a sample of the types of restrictions which currently exist versus typical severe accident scenarios.

> Procedural Interface of Drywell Sprays: Feasibility
>
> Generalizing, the "Primary Containment Control Emergency Procedure" directs the operator to initiate containment sprays (suppression pool/drywell) based on high containment pressure or temperature after assuring that LPCI diversion will not preclude adequate core cooling. However, the use of drywell sprays is also restricted by a requirement to satisfy the Drywell Spray Initiation Pressure Limit. The initiation pressure limit is a function of either the suppression chamber or drywell temperature, which is defined to be the maximum temperature at which drywell sprays may be initiated. In the case of suppression pool sprays, initiation is allowed only if suppression pool water level is below a specified limit. This restriction is intended to prevent high wetwell air space pressure which could prompt downcomer chugging. Suppression pool sprays are initiated before the suppression chamber pressure reaches approximately 16 psig but drywell sprays are precluded at these pressures. If the suppression chamber pressure exceeds approximately 16 psig, drywell sprays could be initiated. Drywell and suppression chamber sprays may also be initiated if the suppression chamber pressure cannot be maintained below the Primary Containment Pressure Limit of approximately 60 psig provided that the temperature pressure plot is not in the unsafe region. Since at this point the containment pressure has already exceeded the design limit of 48 psig, the sprays are initiated irrespective of whether adequate core cooling is assured if the initiation curve is satisfied.
>
> Accident Sequence Effects on Sprays
>
> The principal uncertainties associated with the implementation of the containment spray system are the timing of the spray actuation and the availability of the spray system. By far the most restrictive condition imposed on the use of the drywell spray system is the Drywell Spray Initiation Pressure Limit. This limit precludes the use of the drywell sprays once the suppression chamber temperature approaches the saturation temperature.

Table 4.3-1
SUMMARY OF LIKELY OPERATOR RESPONSE GIVEN EXISTING PROCEDURES FOR POSTULATED SEVERE ACCIDENT CONDITIONS

	CLEAR DIRECTION TO INITIATE DW SPRAYS		
	SBO BEFORE RPV FAILURE DW TEMP = 200-350°F WW TEMP = 90-200°F DW PRES = 2 PSIG-15PSIG	SBO AFTER RPV FAILURE DW TEMP = 600°F-1500°F WW TEMP = 200°F-400°F DW PRES = 60-130	ATWS DW TEMP = 90-340°F WW TEMP = 90-340°F DW PRES = 130
Rev 4AC	NO	NO	YES
Pilgrim	NO	NO	NO
Duane Arnold	NO	NO	NO
Dresden	NO	NO	YES
Quad Cities	NO	NO	YES
Vermont Yankee	NO	MARGINAL	MARGINAL
Monticello	NO	YES-NO	MARGINAL
Nine Mile Pt 1	NOT REPORTED	NOT REPORTED	NOT REPORTED
Peach Bottom	NO	MARGINAL	OFFSCALE
Cooper	NO	MARGINAL	OFFSCALE
Hatch 1 & 2	YES	NO	OFFSCALE
Fitzpatrick	NOT REPORTED	NOT REPORTED	NOT REPORTED

- Class I: Switching to a spray mode given significant radioactivity in the primary containment (prior to containment failure) may provide additional scrubbing effectiveness for Class I events.

- Class II: As for a Class II TW sequence, which is a slow pressurization event, the spray systems being a subsystem of RHR is not expected to be available based on sequence definition.

- Class III: A Class III sequence, like the AE, assumes a loss of all coolant injections and hence the spray systems are again not available.

- Class IV: The drywell spray system may not be available long enough to reduce containment pressure during rapid pressurization sequences such as a Class IV ATWS.

In summary, a reliable method of drywell spray under severe accident conditions does not exist generically for all accident scenarios. In order to make the drywell sprays effective for debris coolability, the following is required:

- Less restrictive procedures

- Confirmation that adverse impacts do not exist or are minimal

- Power or demonstrated procedures to open required valves and power pumps during station blackout scenarios.

The alternative of debris coolability via injection through the breached RPV has all the same inhibits as discussed above, except there are significantly fewer procedural inhibits which could interfere with the operator desire to inject.

Debris Barrier Feasibility

No specific evaluation, except as performed by the participating utilities (Northern States Power and Philadelphia Electric Company), was performed to establish the feasibility of the debris barrier concept within Mark I containments. These utility specific evaluations developed design concepts and costs (see Section 6) which identified that debris barriers could be physically incorporated in those two current designs. In addition, deterministic calculations performed by FAI indicate that the debris barrier is unnecessary, but that it could minimize the uncertainty associated with one of the identified containment failure modes.

Section 5 describes the safety benefits and competing risks of such a decision.

References

[4-1] Fauske & Associates, Inc., IDCOR/NRC Integral Evaluation of NRC Proposed Containment Performance Items: FAI Response to Selected Issues", FAI/86-32 Draft, August 1986.

[4-2] Deardorff, AF., Evaluations and Conceptual Designs for Assuring Severe Accident Containment Integrity of Monticello, NUTECH, NSP-37-001, (Preliminary), August 1986.

[4-3] IDCOR Technical Report 23.4, MAAP Uncertainty Analyses, May 1985.

[4-4] Impact of Core-Concrete Interactions in the Mark I Containment Drywell on Containment Integrity and Failure of the Drywell Liner, G.A. Greene, et al., IAEA-SM-281/36, Int. Symposium on Source Term Evaluation for Accident Conditions, Columbus, Ohio, OctoLer 1985.

[4-5] Wierman, R.W., Experimental Study of Hydrogen Jet Ignition and Jet Extinguishment, HEDL-TME 78-80, April 1979.

[4-6] Cohen, BL, Physics of the Reactor Meltdown Accident, Nuclear Science and Engineering 80, 47-60 (1982).

[4-7] IDCOR Technical Report 23.1PB, Peach Bottom Integrated Containment Analysis.

[4-8] IDCOR Technical Report 21.1, Summary of Risk Evaluation of Peach Bottom.

[4-9] Personal conversation FAI (GABOR) and Delian (Burns), August 12, 1986.

[4-10] Mark I Containment Short Term Program, NUREG-0408, dated December 1977.

[4-11] Mark I Containment Long Term Program Resolution of Generic Technical Activity A-7, NUREG 0661, dated July 1980.

Section 5
SAFETY EVALUATION OF THE IDENTIFIED NRC ITEMS

This section provides a qualitative summary of the safety evaluation related to the NRC proposed changes. Both the potential safety benefits and possible competing risks are discussed with respect to the changes. This qualitative approach results in significant judgement being included in the evaluation. A more quantitative approach could be performed, but it would most likely require a plant specific evaluation to accurately reflect the plant unique characteristics that can substantially alter the quantitative estimates of risk reduction. Use of a generic or surrogate plant for quantitative risk-benefit assessment is judged to lead to average or biased results. Given the substantial plant specific variations among operating plants (see Section 4), plant unique features may dominate the quantitative assessment; therefore, average surrogate quantitative values could be misleading. This qualitative safety evaluation is supplemented by estimates of the potential costs involved in the implementation and maintenance of these items in Section 6.

The five changes suggested by the NRC are as follows:

- Hydrogen control
- Debris Barriers in the drywell
- Debris cooling
- Containment overpressure relief
- Severe accident management procedures

Severe accident management procedures and Mark III hydrogen control are discussed in separate submittals by the BWROG and the Hydrogen Owners Group (HCOG), respectively. Therefore, these items are not discussed in this report.

This section provides the following information regarding these proposed changes:

- Safety Benefits - Section 5.1
- Competing Safety Impacts - Section 5.2
- Open Issues - Section 5.3
- Summary Comparison - Section 5.4

5.1 POTENTIAL BENEFITS OF PROPOSED MITIGATION SYSTEMS

5.1.1 Overview of Safety Benefits

The concepts embodied in the five proposed BWR changes have features which can improve the BWR containment response for selected severe accidents or can reduce the uncertainty associated with severe accident phenomena. The examination of these proposed changes relative to potential positive influences on postulated severe accidents is carried out in this section.

5.1.2 Qualitative Summary of Benefits

The proposed changes have the potential for positive benefits given selected severe accident sequences.

The safety benefits identified here are those which can preclude containment failure or substantially mitigate radionuclide releases. Table 5.1-1 summarizes some of the benefits anticipated to be gained by the incorporation of these changes in existing BWRs. These benefits are derived assuming that severe accident management procedures governing their optimum operation are implemented. Otherwise no benefit may be derived for some of the active system mitigation measures.

There are a number of accident sequences which have been identified as potential contributors to core melt. This is shown in Section 2. These types of sequences may have varying degrees of impact on the containment. Similarly, the proposed changes have varying degrees of mitigation improvement depending upon the sequence type.

Appendix A discusses the postulated containment failure modes which have been identified in various BWR analyses. These containment failure modes can be affected differently, as a function of the types of accident sequences which may occur. In addition, the mitigation features which may be able to preclude a containment failure or limit the radionuclide release are also influenced by the types of accident sequences, i.e., the sequence cutsets.

Table 5.1-1

SUMMARY OF THE BENEFITS TO BE GAINED FROM THE PROPOSED NRC CHANGES

Proposed Change	Benefits
Hydrogen Control (Inerting)	Reduction in the possibility of hydrogen deflagration in containment which could cause early containment failure.
Debris Barrier	Prevention of molten material from flowing smoothly to the drywell shell and inducing thermal failure of the drywell shell or the Mark I downcomers.
Debris Cooling	o Arrest the spread of molten material across the drywell floor and prevention of drywell shell failure. o Scrub fission products from the core debris concrete interaction and from the drywell atmosphere
Venting	o Successful prevention of core melt due to maintenance of containment integrity. o Successful controlled venting (including scrubbing) of fission products during core damage or core melt accidents. o Successful mitigation of ATWS events by virtue of preventing containment failure and therefore, allowing an open cycle BWR until rods can be inserted.
Severe Accident Management Procedures	Explicit guidance is provided such that operations personnel, Technical Support Center and NRC know what procedural options can be exercised.

Extreme examples of the accident sequence dependence of the proposed changes can be reported as follows:

- o <u>No Benefit</u>: For an ATWS event which induces drywell overpressure failure and subsequent core melt, the debris barrier has no effect on the containment mitigation capability or the public health and safety
- o <u>Substantial Benefit</u>: For loss of containment heat removal accident sequences, small venting capability can completely preclude a core melt and any adverse consequence to the public health and safety.

Using these specific examples, it seems prudent that the generalized benefits should be established in terms of those sequences which may dominate risk at a given plant. Specifically, because of plant unique features different, plants may have different risk controlling sequences. This observation is based upon the PRAs (PWR and BWR) that have been performed to date.

Therefore, the approach taken here is to examine the proposed changes as they affect the dominant accident sequences which have been calculated in published BWR, PRAs, for three of the proposed changes:

- o Venting
- o Debris coolability
- o Debris barrier.

Tables 5.1-2 through 5.1-4 identify the impact of the proposed changes on selected dominant accident sequence types. The approach taken in these tables is to establish a matrix of options versus accident sequence types. This matrix display offers the ability to distinguish the potential for safety benefits among the various types of accident sequences. For example, the following option of a small, decay-heat-size, containment vent for "clean"* steam only is discussed relative to its impact on each of the sequence types:

* The example chosen is for a vent option which prohibits the use of the vent if any significant radionuclides are present in the containment.

EXAMPLE OF THE TYPE OF INFORMATION TO BE GAINED FROM TABLES 5.1-2-4		
Class	Description	Clean Decay Heat Sized Vent Option Impact
I	Loss of makeup	
A	Reactor at high pressure	None
B	Station blackout	None
C	ATWS loss of makeup	None
D	Reactor at low pressure	None
II	Loss of containment heat removal	Prevent
III	LOCA	
A	RPV rupture (containment intact)	None
B	Large LOCA	None
C	Medium LOCA	None
D	RPV rupture/Large LOCA (containment failure	None
IV	ATWS	None
V	Interfacing LOCA	None

Tables 5.1-2 through 4 also provide differences in mitigation capability as a function of the way the change is implemented, i.e., the options chosen. (Section 3 discusses the options in more detail than is presented here.)

Most accident scenarios will receive a beneficial impact from each of the proposed changes. Any adverse impacts are discussed in the next subsection. This section identifies the degree of expected benefit as follows:

- H - high benefit: potential complete mitigation of some failure modes by virtue of the single modification; or prevention of core damage and adverse plant conditions which could impact all safety systems.
- M - moderate benefit: potential complete mitigation of some failure modes if other active systems operate successfully
- L - low benefit: mitigation of less important failure modes or requirement for multiple active systems in addition to the proposed change or mitigation of radionuclide releases but not prevention of containment failure.

o (-) no benefit identified.

The observations from each of the accident sequence summary tables is provided below:

Venting (Table 5.1-2):

The description of existing plant venting configurations is discussed in Section 4. The types of options available for venting are discussed in Section 3. The matrix of options versus accident sequence types is provided in Table 5.1-2. However, one of the variations in the venting options which is not discussed explicitly in Table 5.1-2 is provided above as an example of the type of information to be gained from the matrix format. The example provides in simplified form the potential impact on dominant accident sequence types for Option "A" with the exception that a procedural restriction is imposed to prohibit the use of venting if significant radionuclides exist in the containment. From the example, it can be seen that the "clean" steam vent has its only impact on the Class II sequence types. These sequences are conceptually the same as one of the dominant accident sequences identified in WASH-1400.

Table 5.1-2 then describes the potential safety benefits which can be associated with venting using various hardware options for these same accident sequence types, but assuming that no such procedural prohibition exists. In fact, a strong severe accident mitigation procedure and training is assumed for effective venting.

Option A which is a set of vent lines (single or multiple lines) with a 6" equivalent diameter has low (L) to moderate (M) mitigation capability for a number of different types of accident scenarios including the loss of containment decay heat removal sequences. As such, this change offers some very positive potential safety benefits to the public. Upgrading Option A to be able to operate in a station blackout (Option A') and further to be hard piped in the reactor building (Option A") enhances the safety benefits by allowing mitigation of additional sequences, such as station blackout and providing higher assurance that adverse impacts on equipment in the reactor building will not occur.

Option B provides some improvement in that it also allows the possibility of mitigation of some spectrum of ATWS events.

Option C provides the highest identified safety benefits. (Note competing risks are discussed in Section 5.2). This is because such a vent size offers the potential to mitigate a substantial fraction of the postulated worst case ATWS scenarios.

If one examines each of the dominant accident sequence types, the A" and C" options received the highest safety ratings (qualitative) for each of the accident sequence types (Classes I through III). The difference is that Class IV has some potential for effective mitigation via the Option C".

Table 5.1-2

VENT:

SUMMARY OF THOSE SEQUENCES WHICH CAN BE MITIGATED BY THE PROPOSED CHANGES IN VENTING

Vent Options	Description (see Section 3 for more complete)	Dominant Accident Sequences Mitigated[1]						
		Class I		Class II	Class III		Class IV	Class V
		TQUX,QUV A,C,D	SBO B	TW	A B C	D	$T_M C_M C_2$	
A	6" Duct	L[(2)]	-	M	L	-	-	-
A'	6" Duct/DC	L	L	M	L	-	-	-
A"	6" Pipe/DC	M	M	H	M	-	-	-
B	6-18" Duct	L	-	M	L	-	L	-
B'	6-18 Duct/DC	L	L	M	L	-	L	-
B"	6-18" Pipe/DC	M	M	H	M	-	L	-
C	36" Duct	L	-	M	L	-	M	-
C'	36" Duct/DC	L	L	M	L	-	M	-
C"	36" Pipe/DC	M	M	H	M	-	M	-

(1) Where mitigated here means either prevention of core melt or preservation of the containment design function of effective scrubbing of radionuclide releases. It should also be recognized that not all containment failure modes are mitigated. Section 5.3 discusses other postulated containment failure modes.

(2) Legend included in text.

Class V sequences which involve a LOCA outside of containment are judged to be marginally affected by the options related to venting, at least as the worst case sequences are calculated. More realistic calculations would provide some benefits similar to Class I associated with vent schemes.

Drywell Sprays & Debris Coolability (Table 5.1-3):

Debris coolability offers a number of potential safety benefits as identified in the summary Table 5.1-1. Table 5.1-3 provides an indication of where these benefits would be applicable as a function of the option available and the type of severe accident.

Option A is the implementation of existing+ plant design capability with effective severe accident procedures. This has a moderate benefit (M) for accident sequences with the containment intact (Classes I and III A,B,C) because it provides:

- o Debris bed coolability to limit core concrete interaction and non-condensible gas generation
- o Limitation of the progression of molten material travel on the drywell floor
- o Scrub fission products from the debris.

While this capability currently exists in BWRs, published BWR PRAs have not always effectively quantified the benefit of these mitigation features. The estimates of benefit are therefore made without regard to existing plant capability which may vary from plant to plant and sequence to sequence.

The judgment based upon available MAAP and STCP calculations is that the fission product removal capability of drywell sprays is not sufficient to clearly justify its choice over injection into the RPV to provide:

- RPV internal cooling, i.e., preventing revolatilization within the primary system
- debris coolability on the drywell floor++
- fission product scrubbing and/or retention by forcing debris melt releases through an overburden pool of water.

+"Existing" is used generically to reflect the apparent extensive hardware capability identified by the reporting Mark I plants (see Section 4).

++The plant specific issue raised here is that certain Mark IIs and all Mark IIIs would not have effective debris coolability through the use of "drywell sprays".

Table 5.1-3
DEBRIS COOLABILITY:
SUMMARY OF SEQUENCES WHICH CAN BE EFFECTIVELY MITIGATED BY THE PROPOSED CHANGES

Drywell Spray Options	Description	Dominant Accident Sequences Mitigated[1]						
		Class I		Class II	Class III		Class IV	Class V
		TQUX A,C,D	SBO B	IW	AE A B C	D	$T_mC_mC_2$	
A	RHR Injection Spray	M	-	L	M	-	L	-
B	RHR/HPSW Injection Sprays	M	-	L	M	L	L	L
C	RHR/HPSW/ Diesel Fire Pumps(2)	M	L	L	M	L	L	L
D	Option C with no access req'd to the Rx Bldg(2)	M	M	L	M	L	L	L

(1) Where mitigated here means either prevention of core melt or preservation of the containment design function of effective scrubbing of radionuclide releases. It should also be recognized that not all containment failure modes are mitigated. Section 5.4 discusses other postulated containment failure modes.

(2) No AC power required for operation.

Option A also has an impact on the other sequence types in which the containment may have been breached (e.g., Class II or IV sequences). The phenomenological effects are the same as above, however, the result is that radionuclide releases are not prevented because containment is breached. The result is a mitigated radionuclide release. Using the above definition this is a low safety benefit (L) for Classes II and IV.

Option B provides a slight increase in the likelihood of success by including an additional water source, particularly since it is external to the containment. However, the net assessment is that no significant improvement is obtained. This is not true if there are severe RHR pump operating limitations on suppression pool temperature which could preclude successful sprays or injections when pool temperature is high - a situation in many postulated severe accident types.

Option C extends the capability of the debris coolability to include some of the postulated loss of station AC power accidents, particularly those with delayed core melts (i.e., the most probable.)

Option D further enhances the benefits of the debris coolability by extending the effectiveness of the debris coolability to most station blackout type sequences.

The debris cooling options considered all have the capability to provide effective debris coolability benefits for the spectrum of accidents. The reliability and the risk reduction benefit associated with each option varies. However, the existing BWR injection capability is effective+ for nearly all the accident sequence types identified. The addition of the diesel fire pump capability (Option C and D) could provide equivalent mitigation capability for long duration station blackout scenarios.

Debris Barrier (Table 5.1-4):

Core melt progression phenomenology has a number of areas of uncertainty. Within this band of uncertainty there are models which assume that large quantities (> 80%) of the molten fuel can be released to the drywell over very short times and at elevated temperatures. Using such models, it can be postulated that molten material can come in contact with the drywell shell, at least in some plants. Therefore, in lieu of demonstrating that these modeling assumptions are conservative or not applicable, one approach which can be taken is to install physical barriers in the drywell to inhibit the molten material from contacting the drywell shell.

Table 5.1-4 summarizes the impact on safety for several debris barrier options considered.

+ High reliability for most accident sequences within the grouping.

Ta 5.1-4
DEBRIS BARRIER:
SUMMARY OF THOSE SEQUENCES WHICH CAN BE EFFECTIVELY MITIGATED(1) BY(2) THE PROPOSED DEBRIS BARRIERS

Debris Barrier Options	Description	Dominant Accident Sequences Mitigated+							
		Class I			Class II	Class III		Class IV	Class V
		TQUX A,D	SBO B	ATWS C	TW	AE A,B,C	AD D	$T_M C_M C_2$	
A	Short Barrier	-	-	-	-	-	-	-	-
B	Tall Barrier	L(3)	L	L	-	L	-	-	-
C	Short/Thick Barrier	-	-	-	-	-	-	-	-
D	Tall/Thick Barrier	L	L	L	-	L	-	-	-
E	Drywell Shell Coating	L	L	L	-	L	-	-	-
F	Maze Barrier	M	M	M	-	M	-	-	-

(1) Where mitigated here means preservation of the containment design function; effective scrubbing of radionuclide releases, or preclude releases. It should also be recognized that not all failure modes are mitigated. Section 5.3 discusses other postulated containment failure modes.

(2) Note that the current BWR design is anticipated to be adequate to effectively mitigate all or nearly all of the postulated direct contact containment failure modes. This table therefore addresses only the incremental safety benefit associated with the addition of a debris barrier.

Option A is a small barrier installed near the drywell wall to inhibit molten material from reaching the drywell wall during an initial release of material. Because it may not substantially alter the existing resistance of the Mark I containments, it is judged to be a negligible improvement.

Option B is designed such that the slow release of molten material from the RPV will not result in drywell shell failure if adequate temperature control (sprays) and containment heat removal exist. Therefore, a positive benefit can be achieved under certain modeling assumptions.

Option C has the same characteristics as Option A except it is thicker to provie longer term resistance to core concrete attack. No significant improvement from Option A is identified using the MAAP code estimates for debris cooling and core concrete attack.

Option D has the same characteristics as Option B, except with a thicker barrier. No significant improvement relative to Option B is identified.

Option E is a different concept for the barrier, i.e., a coating applied to the drywell wall. No significant safety benefits relative to Option B were identified.

Option F: Maze Barrier: This option affords another dimension to the debris barrier. Not only does the maze barrier afford a barrier comparable to D, it also has the potential to fully or partially mitigate other failure modes of containment such as direct heating and direct impingment (see Section 5.4). These failure modes are judged to be of low conditional probability of potential severe consequences to the drywell shell. Therefore, the maze design (see Appendix B) would have a higher potential positive safety benefit for enhancing the containment integrity.

In summary for the debris barriers, the positive benefits to be accrued to the debris barrier can be strongly plant specific. This plant unique dependence is related particularly to the design details of the pedestal, the drywell, the drywell lip, and the sumps (arrangement, depth, proximity to the shell). Therefore, the debris barrier benefits will vary from zero for Mark Is with large shallow sump volumes inside the pedestal, to potentially significant for plants with small or inaccessible sump volume <u>and</u> no lip between the downcomers and the drywell floor.

<u>Summary</u>

Table 5.1-1 summarizes qualitatively the types of benefits that are foreseen if the NRC proposed changes are implemented. Tables 5.1-2 through 4 identify how these benefits could affect the level of risk associated with the dominant accident sequences types in BWRs.

Because the accident scenario of station blackout has received increased attention in this discussion, Table 5.1-5 is used to identify a typical time line of key events that might be identified for a station blackout scenario and the impact of the proposed changes.

The following matrix of selected options versus effectiveness is meant to provide a summary of some of the most beneficial results for implementing these options in terms of severe accident prevention or mitigation.

EXAMPLE SUMMARY CHART OF BENEFITS BY ARBITRARILY SELECTED DESIGN OPTION			
NRC Proposed Change	Option	Description	Safety Benefit
Vent Containment	A"	6" Pipe w/ DC Power	Potential for risk reduction assessed as moderate or high for all accident sequences except ATWS and LOCAs outside containment.
Debris Coolability	D	RHR/SW/and diesel fire pump (w/DC control to valves)	Existing capability is already adequate for most severe accidents. The incremental improvement is shown for station blackout sequences. All other sequences remain unchanged as a result of this plant modification.
Debris Barrier	B	6"-12" barrier	Potential exists to reduce the uncertainty associated with a postulated failure mode of direct DW shell contact with molten fuel. This could also be accomplished by DW injection. In fact without DW injection a high temperature shell failure may occur despite the barrier (due to radiative heat transfer to the shell).

Table 5.1-5
STATION BLACKOUT SCENARIO FOR PEACH BOTTOM

Event	Time of Key Event	Benefits of Items Addressed Individually-Not In Combination			
		Inerting	DW Sprays	Venting	Debris Barrier
SBO Initiated	0	-	-	-	-
S.P. Temp. 140°F		-	-	-	-
HPCI/RCIC Fail	6 Hr.	-	-	-	-
Core Uncover TAF	8.4 Hr.	-	-	-	-
Core Melt Initiator	11.4 Hr.	-	-	-	-
RPV Failure (High Pressure)	12 Hr.	No H_2 Detonation	Debris Coolability	Cont. Pressure Control, Scrubbing, earlier release of N.G.	Prevent Direct Contact of Molten Material with steel shell
Molten Material Spreads out in Pedestal		No H_2 Detonation	Effective in: DW temperature control; concrete attack in pedestal sump may occur	Same	No Effect
Molten Material Spreads out on DW Floor			Debris Coolable	Same	Prevent Direct Contact
Containment Fails On High DW Shell Temperature	18 Hr.		No Containment Failure (Possible late containment failure if no cont. heat removal)	Maybe (Temperature or direct contact)	Maybe (Temperature over pressure)

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5.2 POTENTIAL ADVERSE IMPACT OF PROPOSED MITIGATION SYSTEMS

5.2.1 Overview of Negative Safety Impact

Any decision regarding a change to a complex group of systems should involve a balance between the desired benefits and the potential competing risks. For the interrelated systems which make up a nuclear power plant, this trade-off may be between increased safety in one area versus reduced safety in another. In any case, for the proper evaluation of any design change it is always prudent to include in the trade-off evaluation the potential negative safety impact which may result from the proposed changes. This section summarizes some of the possible areas of competing risk associated with the proposed changes.

A mitigation system to be used in the response to a severe accident should not adversely impact the safety of the plant in its response to other accident sequences. However, it appears that for some alternative situations, the proposed mitigation systems (if not properly designed and implemented) may lead to some small increase in the risk associated with certain accident sequences.

This section summarizes qualitatively those competing risks which may be incurred in the implementation of the proposed NRC changes. No attempt has been made to quantitatively compare the relative risk changes associated with the proposed changes.

5.2.2 Qualitative Description of the Competing Risks

The competing risks identified for the proposed changes are discussed qualitatively in this section.

5.2.2.1 Hydrogen Control

Hydrogen control has two different applications:

1. Inerting for the Mark I and II containments.
2. Reliable igniters for the Mark III containment.

Inerting in the Mark I and II containments does not appear to have a negative safety impact on the public. It provides an additional layer of protection for certain severe accidents and some DBAs and does not contribute to an increase in public risk.

The only competing risk identified in this review is the potential personnel hazard that the N_2 inerting system may introduce to the plant operation and maintenance personnel. However, operating experience to date has indicated that U.S. BWR utilities have successfully managed this personnel risk. (Note that a BWR in India (Tarapur) did have an incident in which operating staff personnel were asphyxiated by the nitrogen atmosphere.)

Upgrading igniters in Mark III containments to reliably cope with station blackout conditions does not appear to have a negative safety impact beyond that which exists with igniters for other sequences for which delay exists in initiating the igniters after hydrogen has been released to the containment. The igniter evaluation has been deferred to HCOG.

5.2.2.2 Debris Coolability with High Reliability (including Station Blackout scenarios)

One of the options suggested for debris coolability is the use of the drywell sprays. Other options for debris coolability including direct RPV injection also present viable means of obtaining, most or all, of the benefits identified in Section 5.1. However, drywell sprays may induce some phenomena which are not desirable. Such phenomena may be very plant specific and include the potential for the following:

- o Containment Implosion
- o Equipment damage, i.e. vacuum breaker or SRV solenoid failures
- o Introduction of oxygen into the containment atmosphere through the reactor building to wetwell vacuum breaker (plant specific)
- o Flood levels above vacuum breakers.

The following discussion provides additional discussion of the above negative safety impacts.

Low Containment Pressure (Cyclic Vacuum Breaker Operation)

The operation of the drywell sprays may induce a differential pressure across the wetwell to drywell vacuum breakers and the reactor building to wetwell vacuum breakers. This differential pressure would then lead to challenges to these vacuum breakers. The drywell sprays may be cycled on and off depending on the perceived drywell temperature and pressure, core status, and containment water level. This cycling may produce rather severe loads on the vacuum breakers and may then induce eventual failure of the vacuum breakers (i.e., stuck open). The severity of the challenges to the vacuum breakers and the cyclic nature of the events can lead to rather high probabilities of vacuum breaker failures. Such failures of the drywell to wetwell vacuum breakers which occur prior to RPV blowdown could result in early containment failure due to the blowdown pressure surge not directed to the suppression pool but being shunted through stuck open vacuum breakers, then leading to an early containment over pressure failure at one of the "worst" times, i.e., at the time of RPV blowdown. Current EPGs restrict drywell spray operation under conditions which could induce these failures. The use of RPV injection would minimize this effect.

Introduction of Oxygen

Similar to the discussions above regarding the induced negative pressure differential resulting from the drywell spray initiation, a related issue is the potential for the Reactor Building to wetwell vacuum breakers operating as designed to minimize the differential pressure across the containment. Such actions could result in the introduction of oxygen into containment. This oxygen presents some small incremental chance of causing a hydrogen burn. The current judgement based upon deterministic calculations [see Section 4.2] is that:

- The level of oxygen introduced would be relatively small
- The containment would likely be steam inerted or spray inerted which would preclude a substantial hydrogen deflagration.

Nevertheless, there is a remote possibility that a negative safety impact could be induced by the operation of the drywell sprays leading to the admittance of oxygen to the drywell. Current procedures generally restrict spray operation to minimize or prevent the introduction of oxygen into the containment. The use of RPV injection for debris coolability would minimize this effect.

Energetic Core Coolant Interactions

Accumulation of water in the drywell pedestal and on the drywell floor prior to core melt and RPV breach could result in hot molten fuel being ejected into a cold pool of water. If this occurs, there are hypotheses which state that energetic steam explosions may occur which could fail the primary containment during this portion of the accident. This would be classified as an early containment failure, and possibly a large drywell failure. The operation of sprays would result in this possibility nearly always existing in a severe accident. In this case, the DW sprays would act to increase the modeling uncertainty. On the other hand, RPV injection would be a lesser contributor to these conditions, i.e., only LOCAs would result in the same deterministic set of circumstances as described for drywell sprays.

Exceed Containment Negative Pressure Capability

Conservative calculations of drywell pressure response for saturated steam environments and very high temperature environments in the drywell when drywell sprays are initiated can result in rather high pressure drops in the drywell in a time frame before the vacuum breakers have time to open and equalize pressure. These conservative calculations indicate that at extremely high spray flow rates that the drywell pressure could drop sufficiently to exceed the design differential pressure of -2 psig for Mark I containments. While the Mark I ultimate capability is judged to be higher than this and the calculations which show this high drywell pressure drop are judged conservative, the question of whether the drywell spray initiation may introduce a negative safety impact that could cause an early large containment failure remains an area of uncertainty.

A synergistic effect may also exist with drywell sprays operated intermittently following the time at which the vacuum breakers are covered with water (see discussion below on external water sources). The intermittent drywell spray operation may induce even higher negative pressures under the conditions with the vacuum breakers submerged.

It is judged that for the drywell sprays to result in sufficient negative differential pressure across containment to cause a failure, all of the following conditions would need to occur:

- Very high spray flow rates (greater than 2000 gpm)
- Relatively cold water
- Sudden spray flow initiation
- Adverse drywell conditions (probably including a failure to have initiated wetwell sprays).

External Water Sources

One of the options considered for reliable drywell spray operation is the use of water external to containment such as from the service water intake, CST, or Diesel Fire Pump water source. When using such water sources care must be taken to monitor the containment water level and provide a means to remove excess water. The negative safety impact which could occur would be associated with too much water and a flooding of the torus.

Figure 5.2-1 provides a schematic of the Peach Bottom torus with some key dimensions listed. The negative safety impacts may arise if the flooding inundates the torus to drywell vacuum breakers. Table 5.2-1 summarizes the approximate times to covering the vacuum breakers. (This same issue would also apply to RPV injection for debris coolability.)

The consequences of such flooding of the vacuum breakers could be that if RPV blowdown from high pressure through an RPV breach occurs subsequent to covering the vacuum breakers, then vapor suppression may not be fully functional and containment overpressure failure could occur.

In addition, from a hardware and procedural view point, there are no physical interlocks which would prevent containment spray operation from causing unacceptable pressure differentials across key containment members. The operators would have to assume the responsibility to throttle the drywell spray valve to reduce the rate of pressure decrease (containment vacuum design limit is -2 to -10 psid depending on the containment design).

Competing Risks for Drywell Sprays: Summary

The identified competing risks (i.e., potential adverse safety impacts) which may occur as a result of specifying drywell sprays are as follows:

- o Rapid depressurization leading to the introduction of oxygen into the primary containment via the Reactor Building to wetwell vacuum breaker.
- o Excessive water in the containment leading to: (a) reduced volume to absorb non-condensibles and, (b) may preclude wetwell venting.
- o Precipitate containment failure given non-condensible gas generation.
- o Vacuum breaker failure induced by rapid pressure drop in DW
- o Vacuum breakers ineffective when flooded, causing large negative pressure when sprays cycled on.
- o Containment implosion during rapid spray initiation.

While drywell spray hardware currently exists in Mark I and II plants, the plant specific procedures do not currently allow the sprays to be operated in the regime of some severe accidents, e.g., debris on the drywell floor and the drywell at very high temperatures. In addition, they do not specify spraying

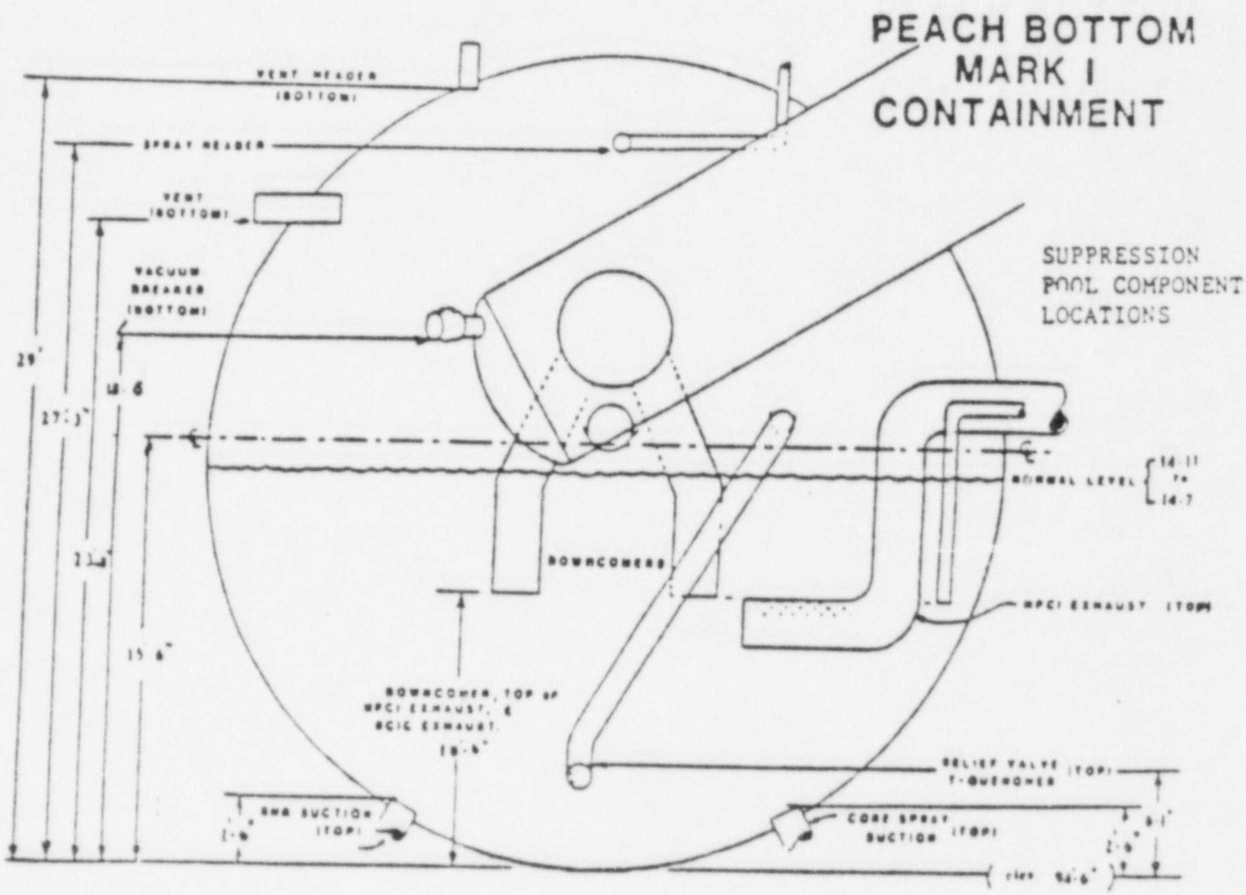

Figure 5.2-1 Simplified Drawing of the Mark I Torus Design: Suppression Pool Component Locations

Table 5.2-1

SUMMARY OF APPROXIMATE TIMES TO FLOOD THE TORUS AND CONTAINMENT AT PEACH BOTTOM

FLOOD LEVEL	TIME (Hrs)	
	Flow Rate 740 gpm	Flow Rate 10,000 gpm
Wetwell to DW Vacuum Breakers	3.6 Hrs.	30 Min.
Drywell Floor	20 Hrs.	1.5 Hrs.

from external sources. Therefore, the competing risks associated with existing procedures and drywell spray hardware are not considered high. The assessment of competing risk for the proposed NRC changes assumes that a greater potential for drywell spray initiation would exist for very severe containment temperature and pressure conditions. It is for these severe containment conditions only that a substantial adverse impact is postulated. No analytic calculations have yet confirmed these postulated concerns regarding adverse impacts.

The cost estimates on drywell sprays did not investigate the possible costs associated with hardware modifications that could result if a low flow drywell spray was required for severe accidents, i.e., preclude a high flow rate. This could be accomplished by installing throttle valves or by plugging the drywell spray nozzles. Each of these options has not been addressed in the cost estimating process. Such modifications would, however, tend to reduce the identified negative safety impacts that have been identified (but not confirmed) in this evaluation.

Debris Coolability (Mark II)

Note that for some Mark II designs (each Mark II appears to have substantial differences in drywell/pedestal configurations) the use of drywell sprays may not be an effective means of debris coolability. In those cases RPV injection may be the only viable method of debris coolability.

5.2.2.3 Venting

Containment venting to prevent catastrophic overpressure failure of containment and preserve the suppression pool in the release pathway for purposes of scrubbing is a highly desirable function. BWRs in general have the capability to provide this venting from the torus or wetwell airspace for accident sequences with only decay heat present. The capability for venting for higher powers such as ATWS cases is less uniform and may result in failures of ductwork in the reactor building which in turn would cause high humidity (saturated steam) and high temperatures in the reactor building.

The potential negative safety impacts which can be postulated for some venting options are discussed below:

Adverse Impact on Equipment and Operator

For certain accident scenarios (ATWS) the containment vent using existing hardware would result in catastrophic ductwork failure in the reactor building (not all plants). These failures would then release steam and some radioactivity into the reactor building thereby eliminating access to the reactor building. The potentially more serious concern is the adverse impact on the electrical equipment in the reactor building which may then precipitate a core melt. This situation may still be more beneficial than the alternative which is to allow the containment to continue to pressurize until a potentially catastrophic failure occurs.

Premature Release of Radionuclides (All classes except Class II)

There may be a set of severe accidents which will result in containment pressures in the range higher than design pressure and lower than ultimate failure pressure. This window of severe accidents would then be successfully mitigated (very low releases) in the case without venting. On the other hand, venting will result in substantial releases of noble gases.

This impact is judged to be negligible on public safety based upon previous PRA evaluations.

Compromise Accessibility (All classes except Class II)

There are two aspects of this potential impact:

- o Reactor building contamination that may be created by venting steam through duct that may fail or substantially leak. This would limit accessability long term (see also remote shutdown panel discussion below) and could limit short term repair or recovery actions. (This is not judged to be a significant impact).
- o Control building accessibility and habitability. The vented release may be piped and ducted in the vicinity of where operators may be required (control room or remote shutdown panel). Therefore, venting may result in severe adverse impacts on the vital personnel required to safely shutdown the plant.

ATWS Scenario: Pool Depletion

By allowing the operating staff the capability and training to vent large amounts of steam, it can be postulated that ATWS accident sequences could degenerate into a vented condition in which the suppression pool is depleted, the downcomers and T quenchers are uncovered, and a core melt is induced without the suppression pool in the pathway of the radionuclide release. Such vented scenarios could arise if SLC and control rods cannot be inserted for extended periods of time, while RPV makeup is from the suppression pool.

5.2.2.4 Debris Barrier

There are some negative impacts perceived for the debris barrier concepts. These impacts include the following:

- o Personnel (ALARA) issues: Installation and maintenance
- o Prevention of collection of normal leakage detection (Technical Specification Requirement)
  - Prevent rapid leakage detection by inhibiting the collection of water, i.e., tech spec measurements are inhibited.
- o Seismic related issues regarding the impact of the barriers on other safety related equipment for deterministic evaluations required by regulation
- o Interference with the assumed downcomer flow area in LOCA calculations
- o Disintegration of the barrier during other severe accidents (seismic, LOCA, etc.) which may cause the barrier to be blown into the torus.

Insufficient evaluation has been performed to indicate if the design options considered can effectively eliminate these issues. It is anticipated that these will be plant specific issues.

5.2.3 Summary of Potential Negative Safety Implications

When estimating risk it is judged to be important to use the best estimate throughout the calculation (fully recognizing the potential uncertainties in the calculations) since biased values, either high or low, can lead to false conclusions resulting in:

1. Inappropriate resource allocation
2. Introduction of equipment and procedures to mitigate accident sequences which may never occur

3. Interference with mitigation systems for accident sequences which may be important in preserving the safety of plant operation.

The important point to note is that there are competing risks in the operation of a nuclear power plant, and emphasizing an unimportant sequence or set of scenarios diverts attention* from those sequences which may be more frequent and/or higher risk.

The implementation of design fixes to address certain accident sequences or issues may resolve some issues, reduce the uncertainty in the treatment of other issues, or may in fact aggravate other postulated accident sequences.

Table 5.2-2 summarizes some of the adverse impacts of the proposed changes which could substantially alter their net safety benefit.

The primary examples of possible competing risks can be found in the following:

- o Drywell sprays can both reduce radionuclide releases and can induce containment failure due to induced negative pressure.
- o Containment venting through the wetwell allows a viable way to preserve containment integrity but may release noble gases at a higher frequency than if no venting existed, and under the worst case ATWS scenarios may induce a core melt in a dry containment.

It is recommended that additional evaluation of these negative safety impacts be performed before a final commitment is made to make the changes evaluated.

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* Attention of the system designers, the safety analysis people, and, most importantly, the plant operators.

Table 5.2-2

SUMMARY OF THE COMPETING RISKS WHICH MAY RESULT FROM THE PROPOSED NRC CHANGES

PROPOSED CHANGE	NEGATIVE SAFETY IMPACT ON IDENTIFIED SEQUENCES	CREATE NEW SEQUENCES	PERSONNEL SAFETY
Hydrogen Control (Inerting)	None Identified	None Identified	Yes; Judged Minimal Based on Experience
Debris Barrier	-Seismic Related Issues -Interference With LOCA Downcomer Flow Area -Loose Material During Blowdown To Torus	None Identified	ALARA For Installation And Maint.
Debris Cooling	DW Sprays: -WW To DW Vacuum Breakers Challenged With High Velocity Challenges -Negative Delta P On Drywell Potentially Near Design Values	Induce Early Failure At Exact Time Of RPV Failure Due To Failed Vacuum Breakers	None Identified
Venting	-Violate NPSH -Introduce Steam To Rx Bldg. -Introduce Radioactivity To Rx Bldg. -Cause Core Melt -Deplete The Suppression Pool Inventory -Premature release of Radionuclides	Open Containment i.e., Another Containment Failure mode	Exposure Of Maintenance Personnel In The Rx Bldg. Potential Exposure To Control Room Operator Depending On Vent Arrangement Relative To Control Room
Severe Accident Management Procedures			

5.3 POTENTIAL REMAINING OPEN ISSUES

One of the areas of concern which may arise in the decision making process is the determination of whether the proposed changes are the "best" changes or whether they are the necessary and sufficient changes for BWRs to perform to provide resolution with the NRC. While the proposed changes appear reasonable it is shown in this section that the NRC proposed items are not sufficient to resolve all issues and that open items will remain even after these items would be incorporated.

Thus far, the discussion has focused principally on the proposed changes and the relative worth of the changes based upon our best available knowledge regarding the core melt progression. There are however, substantial modeling uncertainties which are being investigated by the industry and the national laboratories. These uncertainties may dramatically affect the perception of each issue in terms of its positive or negative safety benefits. This section presents a brief overview of some areas of uncertainty involving differences in accident sequences, modeling and assumptions.

The specific items identified are:

- o Containment failure modes (see the containment event tree in Appendix A)
- o Phenomenological modeling
- o Ex-plant consequences
- o Accident sequences
- o Procedural implementation

5.3.1 Containment Failure Modes

The containment event trees are presented here to establish the types of failure modes that have arisen in the past or could arise in future analyses or discussions. Therefore, the CETs are meant only as a check to ensure that the proposed fixes address those failure modes of greatest interest to the regulators, the public, and the utility.

Table 5.3-1 summarizes the postulated containment failure modes of a MARK I containment given a severe accident. These failure modes have varying degrees of belief associated with them. In addition, some are the result of analytic limitations, some are the result of different scenarios an accident may take, and some are the result of unique plant features. Table 5.3-2 summarizes the references for these postulated failure modes and the relative timing associated with these failure modes.

Table 5.3-3 summarizes the relationship between these postulated containment failure modes and the proposed NRC changes ("fixes").

An additional consideration regarding the containment failure modes relates to possible synergistic effects.

There are two competing Mark I drywell failure modes which are currently identified as having relatively high conditional probabilities based upon IDCOR or NRC evaluations but not both.

1. High drywell shell temperature of greater than 1000°F due to radiative heat transfer from the molten material in the absence of water injection to the drywell. This failure mode is projected by the MAAP code to lead to small leakage failures in the drywell shell.

2. Molten material attack on the drywell shell due to direct contact. This failure mode is perceived as a small leakage pathway up to a large 7 sq. foot break depending upon the hypotheses and modeling assumptions. This failure mode is projected based upon selected NRC modeling assumptions and calculations.

The debris barrier proposed change would effectively address a portion of the second of the above failure modes by preventing the smooth flow of molten material to the drywell wall. However, it would not address the first failure mode which the MAAP models still predict as a potential drywell failure mode.

On the other hand, the post RPV breach coolant injection provides a method of limiting: (1) the temperature in the drywell, (2) the flow of molten material to the shell.

Table 5.3-1

POSTULATED MARK I CONTAINMENT
FAILURE MODES UNDER SEVERE ACCIDENT CONDITIONS

(Alternative Models or Scenarios Defined Along with Approximate Times to Significant Containment Failure)

1. Steam Over Pressure

   - ATWS (1 hr)
   - RPV Rupture (0 hr) (Not a dominant contributor to core melt frequency)
   - RPV Melt Through (Generally agreed that the suppression pool can mitigate the CRD penetration RPV failure mode)
   - Steam Explosions

2. Long Term Non-Condensibles

   - March/CORCON/INTER(3-6 hr)
   - MAAP (Extended time)

3. Direct Contact of Molten Material w/DW

   - NRC (RPV Failure): Calculational Model - 80% initial core dispersal
   - IDCOR: Calculational Model - 20% initial core dispersal
   - Unpublished: Sump pr drain pipe penetration to the Rx building.

4. High Temperature Shell Failure

   - Radiative Heat Transfer IDCOR model only 3-6 hrs

5. Direct Contact of Molten Material with the Torus (RPV Failure): Probably not a Realistic Evaluation; Downcomer Failure May Exist

6. Direct Impingement of Molten Particles on DW Shell Due to High Pressure Blowdown (RPV Failure)

7. Direct Containment Heating Following High Pressure Blowdown (RPV Failure): No analytic proof, Failure Mode by Conjecture (immediate)

8. Hydrogen Detonation During Deinerted Operation

   - Low probability, treated by Technical Specification; shown to be small contributor in other PRAs

9. Failure to Isolate

   - Inerted
   - Positive over pressure

Table 5.3-2

IDENTIFICATION OF
REFERENCE CALCULATIONS
FOR POSTULATED CONTAINMENT FAILURE MODES

CONTAINMENT FAILURE MODE	REFERENCE	CONDITIONAL FAILURE PROBABILITY GIVEN SEVERE ACCIDENT	TIMING OF CONTAINMENT FAILURE
1. Steam Overpressure			
- ATWS	WASH-1400, LGS, SNPS PRA	1.0	1 hr
- RPV Rupture	WASH-1400, SNPS PRA, LGS SARA	1.0	0 min
- RPV Melt Through	MARCH	10^{-2}	2-3 hrs
- Loss of Effective Containment Heat Removal (TW)	WASH-1400, LGS PRA, SNPS PRA	1.0	40 hrs
- Pool By Pass	SRV Discharge Line Break, Vacuum Breaker Failure	10^{-3}	1 hr
- Steam Explosion	WASH-1400	10^{-2}	2-3 hrs
2. Long Term Non-Condensible Generation (Core Concrete Reaction)	LGS PRA	10^{-3}	3-10 hr
3. Direct Contact of Molten Material w/DW Wall	ORNL (Mark I only)	1.0	2-3 hrs
4. High Temperature Shell Failure	IDCOR TSR Task 21 (Mark I only)	.1-1.0	3-6 hrs
5. Direct Torus Failure Due to Molten Material Contact	None	None	None

Table 5.3-2 (continued)

IDENTIFICATION OF
REFERENCE CALCULATIONS
FOR POSTULATED CONTAINMENT FAILURE MODES

CONTAINMENT FAILURE MODE	REFERENCE	CONDITIONAL FAILURE PROBABILITY GIVEN SEVERE ACCIDENT	TIMING OF CONTAINMENT FAILURE
6. Direct Impingement of Molten Material Particles on DW Shell	None	None	None
7. Direct Containment Heating			
- Pressure Rise Due to Dispersed Corium Particles	PWR Issue Resolution, Sandia Peach Bottom CET	10^{-2}	2-3 hrs
8. Hydrogen Detonation During Deinerted Operation	WASH-1400, LGS PRA, SNPS PRA	10^{-3}	2-3 hrs
9. Failure to Isolate			
- Inerted	LGS PRA, SNPS PRA	10^{-2}	2-3 hrs
- Positive Overpressure	None	10^{-3}	2-3 hrs

Table 5.3-3

COMPARISON OF CONTAINMENT FAILURE MODE

AND

NRC PROPOSED CHANGE

FAILURE MODE	NRC PROPOSED CHANGE
1. Steam Over Pressure	
- ATWS	Vent
- RPV Rupture	--
- RPV Melt Through	Vent
- Loss of Effective Containment Heat Removal (TW)	Vent
- Pool Bypass	--
- Steam Explosion	--
2. Non-Condensibles	Vent (includes SBO)
3. Direct DW Contact	Drywell Debris Barrier
4. High Shell Temperature	Debris Cooling (includes SBO)
5. Direct Contact of Torus w/Molten Material	Drywell Debris Barrier
6. Direct Impingement	-------------
7. Direct Heating	-------------
8. Hydrogen Detonation	Inerting Tech Spec
9. Containment Isolation Failure	-------------
10. Direct DW Penetration Through Sump Or Drains	Debris Cooling

This can also be viewed as stating that the implementation of both changes would provide double assurance regarding prevention of the direct contact drywell shell failure (i.e., barrier and reliable coolant injection), and providing single level assurance that the high temperature shell failure mode would be mitigated via the reliable coolant injection.

5.3.2 Phenomenological Modeling

There are phenomenological modeling uncertainties associated with each of the proposed changes. These uncertainties are sufficiently broad that individual modeling assumptions can be taken which demonstrate deterministically that all the changes have positive safety impacts and no competing risks. Using those assumptions and models the decision would be relatively clear-cut. However, because of the uncertainties involved there are questions about the relative benefit that is gained from the changes, and certain competing risk factors may be introduced that would compromise the existing safety position of the BWRs.

Some of the phenomenological uncertainties which could influence the assessment of these proposed changes include:

- o Debris behavior during release from the RPV
- o In-vessel coolability (not addressed here)
- o Drywell spray adverse effects (see discussion under procedures)
- o Venting under ATWS (see discussion under procedures)

Debris Barrier

All Mark I plants responding to the survey appear to have lips between the drywell floor and the downcomer to the torus. Therefore, using the IDCOR MAAP models, molten material would not run out and directly down the downcomer except in some very unusual circumstances involving a large mass of material (100% of the core molten and the density increased due to gas bubbles or change in chemical composition, i.e., approximately 1.6 times the core solid mass).

Core melt progression and RPV breach are not well understood phenomena and there can be a wide spectrum of possible scenarios. Within the spectrum of such scenarios are variations due to modeling uncertainties and those due to real sequence differences. The best estimate of the sequence of events is described in the IDCOR MAAP models, although the following differences could alter these perceptions:

- The quantity of material initially released from the RPV given a core melt RPV breach
- The temperature at breach
- The heat losses during ex-vessel transport to the DW floor (CRD housing interaction)
- The chemical composition of the material
- The thermal expansion of the material
- The degree to which material is dispersed by pressure blowdown
- The degree of concrete reaction and the amount of gases
- The presence of water in the pedestal or sumps.

5.3.3 Ex-Plant Consequences

Noble gases which exceed the Environmental Protection Agency's Protective Action Guidelines (PAGs) will not necessarily be mitigated by the proposed NRC changes; in fact the NRC changes may cause the PAGs to be exceeded (i.e., venting) with higher frequency in an attempt to preclude more severe containment failures.

5.3.4 Accident Sequences

There are many types of accident scenarios which introduce different challenges to the containment and to the core. The proposed changes affect the integrity of containment during these challenges in different ways. Table 5.3-4 summarizes how the optimum implementation of the proposed changes might affect the identified accident sequence types. Many of the accidents show an improved containment integrity following the implementation of the changes. Other sequences would not have a significant change in containment integrity.

Table 5.3-4
POSSIBLE RESULT OF THE PROPOSED MODIFICATIONS
ON DOMINANT ACCIDENT SEQUENCE TYPES

Accident Sequences	Improvement in Primary Containment Integrity Through Proposed Mitigation Measures		
	Debris Barrier	Vent	Spray
I A	Yes	Yes (R)	Yes
B	Yes	Yes (R)	Yes
II	No	Yes	No
III	Yes	Yes (R)	Yes
IV	No	No (V)	No
V	No	No	No

(R) Radionuclides released directly to the environment.

(V) The ATWS vent has not been included in this table.

Therefore, one may conclude that the types of dominant sequences at a given plant would determine the relative cost benefit relationship which would be justifiable. This then becomes both a modeling uncertainty and a plant specific variation.

5.3.5 Procedural Implementation

The procedural uncertainties or potential open issues for drywell sprays and containment venting are discussed below.

Drywell Sprays

Drywell sprays have a series of open items which are generally plant specific in nature but which currently limit the availability of drywell sprays during some severe accidents. These open items include the following:

- Debris cooling via RPV injection can be implemented consistent with the current EPGs from water sources inside containment (i.e., the suppression pool) or external water sources.
- Current EPGs (prior to Rev. 4 and to some extent Rev. 4) as reflected in plant specific implementation do not give the operating staff a large leeway within which drywell spraying can be accomplished with these sources particularly at those times when debris could be on the drywell floor.
- The version of the BWROG EPG Revision 4 AE currently under development specify the use of RHR pumps to provide the DW spray. There is not a consideration of other DW spray sources such as diesel fire pump and HPSW (p. PC-8, PC-4). Some plant specific implementations of the symptom based procedures to allow this option.
- Severe accident management procedures to ensure that drywell sprays can be effectively used would be necessary to obtain full benefit from this hardware.
- Station blackout scenarios could be effectively mitigated if an AC power source can be restored between core melt and prior to containment failure. (Success probability .5 to .99 for recovery times of 4 to 24 hours, respectively can be gleaned from existing PRA evaluations.)

At present there are no procedures or guidelines outlining the operation of the spray system in severe accidents. Under those situations the use of the spray system outside its region of operability (e.g., outside the Drywell Initiation Pressure Limit) may be important in the control of fission products suspended in the primary containment. In general, there are no procedures or operator training for connecting external sources of water to the containment spray. Furthermore, the availability of the spray system is questionable subsequent to a containment breach.

Venting

Containment venting requires: (1) hardware; (2) procedures and training; and (3) instrumentation to effectively implement. Section 3 and 4 have touched on the hardware aspects of containment venting. One of the possible open items in the implementation of an effective venting scheme may be the procedural interface. Currently there is a difference of opinion among the BWR owners on if, and how, venting is to be implemented.

For simplistic discussion it may be appropriate to separate the venting regimes that are being discussed as follows:

- Decay heat venting (no radionuclides in containment)
- Core melt decay heat venting (e.g., SBO)
- ATWS venting.

As outlined in Sections 3 and 4 these venting regimes have different physical requirements needed in order to satisfy acceptable performance. The decay heat vent appears to be the simplest conceptually and procedurally. The venting of radionuclides in both (1) a technical issue regarding compromising recovery, repairs, and personnel safety (including the operating staff); and (2) a public safety issue regarding the purposeful release of radionuclides, possibly prematurely, to the environment.

The last regime identified for venting is in association with a failure of adequate reactivity control.

The ATWS scenarios developed as part of the ATWS rule and for NUREG 1150 severe accident source term evaluation identify those types of responses which may be required to successfully prevent core melt given a failure to scram. These scenarios are in general conservatively characterized to identify the minimum times for operator action to prevent severe containment and/or core damage. On the other hand, there may be significant additional time which would be available to the operator to successfully mitigate a containment failure for ATWS or partial ATWS events where the boundary conditions are less severe than imposed in the deterministic calculations used for these assessments. Specifically any reduction in power during the accident, for example, due to partial system success (control rod insertion, boron injection) could lead to substantially longer times for effective actions. Therefore, it becomes crucial to the decision making process to decide whether these sequences (worst case) are sufficiently likely to require an extraordinary hardware modification and procedural change to mitigate.

Alternatives to the large vent strategy suggested by the NRC as a mitigating action for ATWS are now being considered by ORNL, NYPA, EPRI, GE, and FAI.

An extensive amount of work has been performed by FAI to identify alternate ATWS mitigation strategies which would not require containment venting. Such strategies are just now being developed, supported by analytic work at GE and EPRI. If these strategies prove successful, the use of a very large vent for ATWS can be reduced in importance (i.e., safety benefit) or completely eliminated. The strategy involves identifying for the operator safe, stable states in which an isolated containment could remain intact (within its design temperature and pressure during an ATWS without SLC or control rod motion.)

5.4 SUMMARY

This section summarizes in brief tabular format the benefits and potential competing risks associated with the proposed changes. Table 5.4-1 provides the summary table.

Table 5.4-1

TRADE OFF OF BENEFITS VERSUS RISKS: SUMMARY OF BWR ITEMS

ITEM	BENEFITS	COMPETING RISKS
HYDROGEN CONTROL OR INERTING SYSTEM	o PREVENT H_2 DETONATION IN CONTAINMENT FOLLOWING LARGE LOCA OR SEVERE ACCIDENT	o PERSONNEL HAZARD (INERTING)
DRYWELL SPRAY HIGH RELIABILITY	o CONTROL DW TEMPERATURE o SCRUB FISSION PRODUCTS o DEBRIS COOLABILITY o MITIGATION OF SRV BREAK IN THE WW AIRSPACE	o INTERFACING LOCA (AN RPV CONNECTION WILL EXIST ANYWAY) o RAPID PRESSURE DROP (1) CAUSES NEGATIVE INTERNAL CONT. PRES., CONTAINMENT IMPLODES o VACUUM BREAKER FAILURES INDUCED BY CYCLIC OPERATION (1) o DIVERSION OF FLOW (1) FROM RPV INJECTION o ADVERSE IMPACTS ON DW EQUIPMENT o INGRESS OF OXYGEN THROUGH Rx BLDG TO WW VACUUM BREAKER (POTENTIAL H_2 DEFLAGRATION)
DEBRIS COOLABILITY (RPV INJECTION)	DEBRIS COOLABILITY	o INTERFACING LOCA

Note: (1) while drywell spray hardware currently exists in Mark I and II plants, the plant specific procedures do not currently allow the sprays to be operated in the regime of some severe accident, e.g., debris on the drywell floor and the drywell at very high temperatures. In addition, they do not specify spraying from external sources. Therefore, the competing risks associated with drywell sprays are not considered high. The assessment of competing risk for the proposed NRC changes assumes that a greater potential for drywell spray initiation would exist for very severe containment temperature and pressure conditions. It is for these severe containment conditions only that a substantial adverse impact is postulated. No analytic calculations have yet confirmed these postulated concerns regarding adverse impacts.

Table 5.4-1 (continued)

TRADE OFF OF BENEFITS VERSUS RISKS: SUMMARY OF BWR ITEMS

ITEM	BENEFITS	COMPETING RISKS
WETWELL VENTING	o CONTAINMENT HEAT REMOVAL (ALTERNATE) (PREVENT CORE MELT-CLASS II) o RADIONUCLIDE SCRUBBING PATHWAY THROUGH SUPPRESSION POOL	o ADVERSE IMPACT ON Rx BLDG EQUIPMENT o ADVERSE IMPACT ON OPERATORS AT REMOTE SHUTDOWN PANEL AND CONTROL ROOM o INDUCE CORE MELT o LOSS OF CONTAINMENT INTEGRITY THROUGH STUCK OPEN VALVE o PREMATURE RELEASE OF NOBLE GASES o DEPLETION OF THE SUPPRESSION POOL
DEBRIS BARRIER	o PREVENT DOWNCOMER FAILURE IN MK I I.E., DIRECT RELEASE TO THE CONTAINMENT o DELAY DRYWELL SHELL FAILURE DUE TO DIRECT CONTACT FOR SIGNFICANT TIMES	o INTERFERENCE WITH LOCA FLOW AREA TO TORUS o SEISMIC ANALYSIS o MAINTENANCE INTERFERENCE o PERSONNEL o ALARA DISINTEGRATION DURING LOCA IMPACT ON RHR STRAINERS
EPGs FOR SEVERE ACCIDENT MANAGEMENT	o PROVIDE ADDED GUIDANCE FOR SEVERELY DEGRADED CONDITIONS -- HIGH RADIATION -- HIGH CONTAINMENT TEMPERATURE	o INAPPROPRIATE ACTION INDUCED BY FALSE INDICATION WHEN SEVERE ACCIDENT DOES NOT EXIST

Section 6
ESTIMATED COSTS

This section infers the costs from alternatives of similar schemes as they were developed for individual utilities. These alternatives are described in Section 3, Reference [6-1], and Appendix B. The cost estimates and their bases and assumptions may vary substantially and therefore the estimates presented here should be considered approximations which would require additional detailed investigation and better design definition before accurate cost estimates could be assembled. The variation in estimates is also judged to reflect potential for plant specific variability that could be encountered based upon the plant to plant differences noted in Section 4.

It can be anticipated that with further investigation into methods of optimizing the design for more severe accident conditions that the quoted costs may escalate further. This judgement is, however, not based upon an evaluation of such optimized designs.

Sections 6.1 through 6.3 summarize the possible cost categories which could influence the decision making process for choosing to implement the changes for debris coolability, venting, and debris barriers, respectively.

There are a number of considerations which may impact the "cost" of implementation which are not easily factored into traditional measures of costing. These considerations include the following:

- o The limited engineering resources available.
- o The limited funds available.
- o The limited time available during maintenance outages.

The implementation of the suggested NRC changes may result in lowering the priority of other issues, and may therefore result in other modifications or issues not being resolved in a timely manner. These kinds of effects are not included in this evaluation.

The cost considerations which are most easily developed can be broken down into the following areas:

- o Hardware
  - components
  - engineering
  - analysis required (50:59 considerations)
- o Installation
- o Test and maintenance
- o Plant unavailability
  - initial installation
  - periodic maintenance/LCO violations
- o ALARA.

Other costs not explicitly listed above include:

- procedural development
- personnel training
- future impacts on backfits or other required plant modifications.

In all of the quoted cost estimates, the installation, maintenance, plant availability, and ALARA "costs" were not separated out. Therefore, these values are either listed as "not reported" or are crudely estimated for illustration only.

6.1 DEBRIS COOLING

The ability to cool debris in the drywell can vary substantially in reliability, effectiveness, and the breadth of sequences which can be adequately mitigated when examined on a plant specific basis. Table 6-1 summarizes various options of proposed or existing designs that could be available to supply debris coolability in the drywell.

Options A and B are within most current plant capabilities. These options make use of RHR pumps or SW pumps to inject into the RPV or into drywell spray headers. While no separate costs are shown for implementation, the following cautions should be included:

Table 6-1
COST ESTIMATE SUMMARY ($ MILLION)
DRYWELL COOLANT INJECTION

DRYWELL COOLANT INJECTION	DESCRIPTION	HARDWARE	INSTALLATION	TEST AND MAINTENANCE	PLANT UNAVAILABLE[1]	ALARA[2]	TOTAL COST
Option							
A	RHR Spray	0[3]	Not Reported	Not Reported	Not Reported	0	0
B	RHR/HPSW Sprays	0[4]	Not Reported	Not Reported	Not Reported	0.1	0[4]
C	RHR/HPSW/ Diesel Fire Pumps[5]	.5 - 1	Not Reported	Not Reported	Not Reported	0.0	0.6 - 1.1
D	Option C with no access req'd to the Rx Bldg[5]	1 - 2	Not Reported	Not Reported	Not Reported	0.1	1.1 - 2.1

(1) $.5M/day assumed equivalent of extended outages for installation.

(2) $1000/man rem assumed equivalent for personnel exposure (both initial installation and subsequent maintenance). These values are not based on utility estimates, but are judgements made only for illustration.

(3) All costs in millions of dollars.

(4) Not true for all plants.

(5) No AC power required for operation; a separate independent DC power supply must be necessary for the isolation injection valves.

- o Debris cooling via RPV injection can be implemented consistent with the current EPGs from water sources inside containment (i.e., the suppression pool) or external water sources.

- o Current EPGs (prior to Rev. 4 and to some extent Rev. 4) as reflected in plant specific implementation do not give the operating staff a large leeway within which drywell spraying can be accomplished with these sources particularly at those times when debris could be on the drywell floor.

- o The EPGs do not specify the use of external sources for spraying the drywell. Some plant specific implementations of the symptom based procedures to allow this option.

- o Severe accident management procedures to ensure that drywell sprays can be effectively used would be necessary to obtain full benefit from this hardware.

- o Station blackout scenarios could be effectively mitigated if an AC power source can be restored between core melt and prior to containment failure. (Success probability .5 to .99 for recovery times of 4 to 24 hours, respectively can be gleaned from existing PRA evaluations.)

One of the desirable attributes established by the NRC for the debris coolability change is that reliable injection be provided even for cases in which a complete loss of AC power (station blackout) were to occur. Options C and D include Option B, but provide the added capability of a diesel fire pump that can supply injection to either the RPV or the drywell spray header. The principal costs estimated by the participating utilities are for the dedicated DC power supplies and the DC controllers for the isolation valves required so that the system would be operable under Station Blackout. The amount of hard pipe required to upgrade existing systems makes up for the remainder of the costs. A diesel fire pump is assumed to already be present onsite for the cost estimates presented. The first three cautions noted above for Options A and B apply also to Options C and D.

Some of the negative safety impacts identified in Section 5 related to the possibility of high flow rate sprays. A potential "optimization" of the debris coolability change could be a modification which installs low flow nozzles in the drywell spray header(s). The cost estimates on drywell sprays did not investigate the possible costs associated with hardware modifications that could result if a low flow drywell spray was required for severe accidents, i.e. preclude a high flow rate. This could be accomplished by installing throttle valves or by plugging the drywell spray nozzles. Each of these options has not been addressed in the cost estimating process. Such modifications would, however, tend to reduce the identified negative safety impacts that have been identified (but not confirmed) in this evaluation.

Not reflected in these estimates is one which identifies a separate bunkered injection system at a cost of $25 million.

6.2 CONTAINMENT VENTING

Containment venting is viewed as a last resort containment pressure control function. The options presented include several combinations which have incorporated to some degree the following considerations:

- Wetwell venting only
- Sufficient to remove the decay heat from containment at times of 10-24 hours (probably less than 1% decay heat)
- Adequate to operate under station blackout conditions (i.e., dedicated DC power supply)
- Adequate for maximum calculated power levels for certain ATWS scenarios.

These options are combined together to give the nine options presented in Table 6-2. The costs developed by the utilities are principally related to:

- DC power to the isolation valves
- Sufficiently large pipe to allow the heat generated from an ATWS of various sizes to be removed.

These costs appear to be applicable to Mark I and II plants. It is judged that Mark III plants would not be significantly different.

6.3 DEBRIS BARRIERS

The drywell debris barrier options are identified in Table 6-3. The costs for the reported options are included. The options generally progress in complexity and costs from A through F. These go from the simplest splash shield concepts

Table 6-2
COST ESTIMATE SUMMARY ($ MILLION)
VENTING

VENTING OPTION	DESCRIPTION	HARDWARE	INSTALLATION	MAINTENANCE	PLANT UNAVAILABLE[1]	ALARA[2]	TOTAL COST
A	6" Duct	0 - .2[3]	Not Reported	Not Reported	Not Reported	0 - .1	.0 - .3
A'	6" Duct/DC	.2 - 1	Not Reported	Not Reported	Not Reported	0 - .1	.2 - 1.1
A"	6" Pipe/DC	.2 - 1	Not Reported	Not Reported	Not Reported	.1	.3 - 1.1
B	6-18" Duct	0 - 1	Not Reported	Not Reported	Not Reported	.1	.1 - 1.1
B'	6-18 Duct/DC	1	Not Reported	Not Reported	Not Reported	.1	.1 - 1.1
B"	6-18" Pipe/DC	1 - 2	Not Reported	Not Reported	Not Reported	.1	1.1 - 2.1
C	36" Duct	1 - 2	Not Reported	Not Reported	Not Reported	.1	2.1 - 5.1
C'	36" Duct/DC	1 - 2	Not Reported	Not Reported	Not Reported	.1	2.1 - 5.1
C"	36" Pipe/DC	1 - 2	Not Reported	Not Reported	Not Reported	.1	2.1 - 5.1

(1) $.5M/day assumed equivalent of extended outages for installation.

(2) $1000/man rem assumed equivalent for personnel exposure (both initial installation and subsequent maintenance). These values are not based on utility estimates, but are judgements made only for illustration.

(3) All costs in millions of dollars.

Table 6-3
COST ESTIMATE SUMMARY ($ MILLION)
BARRIERS

DEBRIS BARRIER	DESCRIPTION	HARDWARE	INSTALLATION	MAINTENANCE	PLANT UNAVAILABLE(1)	SEISMIC ANALYSIS(2)	ALARA(3)	TOTAL COST
Option								
A	Short Barrier	.2(4)	Not Reported	Not Reported	Not Reported	.1	.1	.4
B	Tall Barrier	.5	Not Reported	Not Reported	Not Reported	.2	.2	.9
C	Short/Thick Barrier	-	Not Reported	Not Reported	Not Reported	-	-	-
D	Tall/Thick Barrier	-	Not Reported	Not Reported	Not Reported	-	-	-
E	Drywell Shell Coating	1-2	Not Reported	Not Reported	Not Reported	1	1	3-4
F	Maze Barrier	2-4	Not Reported	Not Reported	Not Reported	2	1	5-7

(1) $.5M/day assumed equivalent of extended outages for installation.

(2) No estimates for such analyses have been performed by individual utilities. The estimates presented are only for illustration.

(3) $1000/man rem assumed equivalent for personnel exposure (both initial installation and subsequent maintenance). These values are not based on utility estimates, but are judgements made only for illustration.

(4) All costs in millions of dollars.

to a more complex maze barrier. The more complex maze barrier could be effective over a range of postulated debris release rates to both slow down and divert the flow of molten debris away from the drywell shell wall. It can also be said that the negative safety impacts also increase as the complexity increases.

Section 7
SUMMARY

IDCOR has identified that public risk associated with severe accidents is low and far below the NRC safety goals [7-1]. Therefore, no major modifications appear necessary. However, arguments regarding "defense-in-depth" may still indicate that "minor" changes or procedural enhancements which could increase the effectiveness of existing hardware to mitigate radionuclide releases is a desirable objective.

The NRC has suggested five BWR changes which could further reduce the already low calculated risk for BWRs. These proposed changes are the following:

- Hydrogen Control: e.g., inerting or igniters (including during station blackout events)
- Drywell Sprays: i.e., assure reliable operation (including during station blackout events)
- Wetwell Venting: i.e., assure capability
- Debris Barrier: i.e., prevent direct contact containment failure due to downcomer or torus failure
- Severe Accident Management Procedures

The approach taken in the evaluation is as follows:

- Describe the proposed changes
- Assess current plant configurations
- Define the basic issues behind the proposed changes in terms of effects on key BWR containment functions
- Develop conceptual design options
- Look at these changes in the context of:
  - the basic issues
  - dominant severe accident sequence impacts
  - containment failure modes which can be affected.

While the items cited by the NRC appear to offer some advantages, reservations exist concerning these issues because these items may not:

- o Address all the underlying concerns which are behind these proposed changes.
- o Be the best choices for investment of resources and capital.
- o Eliminate the need for further plant modifications to address the same issues.
- o Be generic because each BWR has unique features which alter both the containment response and the dominant contributors to accident sequences and therefore the efficacy of the changes.
- o Have a minimum adverse impact on the existing level of plant safety in protecting the public.

The following discussion of Mark I, II, and IIIs summarizes by plant type, some of the relevant factors to be considered by the BWR owners in their decision making process. This report is provided as a summary of technical information and does not presume to make judgements on the conclusions regarding the value-impact tradeoff of the changes.

7.1 BWR MARK I OVERVIEW

7.1.1 Qualitative Summary

The BWR Mark I severe accident analyses performed to date indicate that the Mark I plants are within the proposed NRC safety goals and do not represent a disproportionate risk to the public. This conclusion is now being tested on a plant specific basis through implementation of the IPE.

In addition, no NRC documents exist which identify the Mark I as a risk outlier. There are issues that continue to need exploration but no evidence to date demonstrates that a safety concern exists. Mark I plants continue to meet the objectives of the NRC safety goals on low risk to the public. There are, of course, a number of programs which have assessed or are now assessing the potential for public health risks from nuclear power plants. However, many of these programs are not sufficiently refined or complete to support a decision making process different than the IPE method mentioned above. Consider the following:

- Core Melt Frequency: Accident sequence evaluations of BWRs continue to show core melt frequencies below the proposed NRC safety goal. The latest draft NUREG-1150 [7-2] section on Peach Bottom also supports this conclusion.

- Containment Structural Capability: The containment Loads Working Group and the Containment Performance Working Group both had limited resources to bring to bear on their respective concerns. Many of the results of their evaluations are based on bounding calculations or conservative assumptions used for expediency. Therefore, these evaluations are not presently adequate for use in challenging the adequacy of the Mark I containment.

- Phenomenology: The core melt phenomenology items, including thermal hydraulics, core slump, debris coolability, and Zr oxidation, are sufficiently controversial such that a technical consensus does not exist. Therefore, decisions based upon existing models may require further modifications to the proposed changes at some time in the future when more refined models are developed.

- Source Term: NUREG-0956 [7-3] and NUREG/CR-4624 [7-4] form the basis for NRC work on assessing source term response. NUREG-0956 received significant criticism from the industry and was judged not to represent an adequate, integrated characterization of factors influencing the source terms. The latest Battelle evaluation [7-4] includes only a limited number of sequences which may not be probabilistically relevant.

- Public Health: No definitive report on public health risks for BWR Mark I's has been published which shows anything but risks which are small relative to the safety goals.

7.1.2 MARK I Summary

Based upon the calculations performed under the IDCOR program, the plant-specific IPE analyses performed to date, and a review of on-going NRC work (including Sandia testing), the following conclusions are reached for the BWR Mark I:

- Mark I severe accident sequence frequencies are low and meet the formerly proposed NRC safety goal of less than 1E-4/yr for core melt.

- Mark I containments meet the design requirements.

- Mark I containments are adequate far beyond their design temperatures and pressures.

- Mark I containments are completely adequate for most of the postulated core melt and severe accident scenarios, including:

  - Loss of coolant inventory accidents (such as experienced at TMI-2)

  - Large break LOCAs leading to core melt.

- Even for very low frequency accidents for which the Mark I containment is postulated to be breached, there remain several passive and active mitigation measures to mitigate the radionuclide release.

- Calculated source terms for all sequences analyzed in IDCOR are very low.

- Public health risks are within the proposed NRC safety goals.

- Competing risks of additional plant modifications may induce failure modes that are more risk significant than those presently existing.

- Even though they are adequate for most events, Mark I containments do have some vulnerabilities during severe accidents. The proposed NRC conceptual changes address a number of these postulated vulnerabilities.

Table 7.1-1 summarizes the observations from this study which may affect the decision making process regarding the implementation of the proposed NRC changes to the BWR Mark I containment or its mitigating systems. The areas of investigation which are identified on the table are as follows:

- Safety benefit

- Generic resolution

- Competing risks

- Alternatives considered

- Uncertainty or lack of information

- Cost

The following discussion provides a brief summary of the basis for the entries in the table on an item by item basis.

Table 7.1-1
IDENTIFICATION OF KEY ITEMS IN THE
DECISION MAKING PROCESS FOR
THE IDENTIFIED NRC IT. 1S
MARK I

ITEM		BEST ESTIMATE SAFETY BENEFIT (RISK REDUCTION)	GENERIC RESOLUTION	COMPETING RISKS	ALTERNATIVES OR OPTIONS	LACK OF INFORMATION OR UNCERTAINTY	COST ($M)
H_2 Control Inerting (MK I,II)		M	Yes	Small; plant personnel	Not Considered	Uncertainty on Full Positive Impact	Complete
Debris Coolability (Including Station Blackout)		H	Concept-yes Hardware-no	Yes; potentially large if high flow spray	Options Assessed	Uncertainty on the Negative Safety Impacts of High Flow Sprays	$.2-$25M
Vent	Decay Heat	H	Yes	Small	Not considered	Yes, Small	$1M
	Core Melt	M	No	Moderate	Not considered	Yes, Moderate	$1M
	ATWS	M	No	Large	On-going work	On-going work	$1-2M
Barrier Drywell		L	No	Small to Moderate (Seismic, ALARA, and LOCA Issues)	Yes	Modeling Uncertainty May Be Reduced	$.2-5M
EPG Rev 4		H	Yes	No	Yes	No	$.3-.8
Severe Accident Management Procedures		M	No	Small	Yes	Scope Unknown	Unknown

Legend: Qualitative in nature, not based on detailed calculations

H - High benefit: potential complete mitigation of some failure modes by virtue of the single modification; or prevention of core damage and adverse plant conditions which could impact all safety systems.

M - Moderate benefit: potential complete mitigation of some failure modes if other active systems operate successfully.

L - Low benefit: mitigation of less important failure modes or requirement for multiple active systems in addition to the proposed change or mitigation of radionuclide releases but not prevention of containment failure.

Hydrogen Control (Inerting - Mark I)

- o Best Estimate Safety Benefit: A review of the accident sequences affected by this change indicates that DBA LOCAs and severe accidents are affected. The ability to add assurance regarding the limits on the combustion of H_2 adds safety margin to the containment.

- o Generic Resolution: The concept is generic; the implementations have been plant specific.

- o Competing Risks: The only identified competing risk is plant personnel hazards due to asphyxiation. US plants appear to treat this adequately.

- o Alternatives: Since inerting has already been implemented, these issues are currently of low interest.

- o Lack of Information: There is a substantial modeling uncertainty in the calculation of H_2 combustion, such that the actual benefits of the inerting are highly uncertain, but it does reduce the modeling uncertainty regarding H_2 deflagration induced failures.

- o Cost: This is not considered since inerting is already performed.

Debris Coolability

- o Best Estimate Safety Benefit: The ability to provide water to a molten debris bed in the drywell can be a significant safety improvement in preserving containment integrity (i.e., against molten material progression, non-condensible gas generation, and high drywell temperatures). All these advantages tend to preserve containment integrity. In addition, water injected into the drywell can provide additional fission product retention even if containment failure occurs. Substantial safety benefits are projected which can be satisfied with any coolant injection source to the drywell or breached RPV.

- o Generic Resolution: The concept appears generic but the hardware to be used, the arrangements, and the capabilities are very plant specific.

- o Competing Risks: Current analytic modeling techniques indicate potential failure modes of containment, vacuum breakers, and oxygen introduction all of which could lead to severe radionuclide releases given modeling assumptions related to timing of RPV blowdown, steam inerting of containment, and hydrogen generation and deflagration modeling. These competing risks are associated principally with using drywell sprays as the debris coolability source with high spray flow injection. Another possible risk is associated with interfacing LOCA. This is judged small but the plant specific implementation could alter this conclusion.

While drywell spray hardware currently exists in Mark I and II plants, the plant specific procedures do not currently allow the sprays to be operated in the regime of some severe accidents, e.g., debris on the drywell floor and the drywell at very high temperatures. In addition, they do not specify spraying from external water sources. Therefore, the adverse risks associated with the operation of current drywell spray hardware using existing procedures is not considered high.

The evaluation of competing risk for the proposed NRC changes assumed that procedures and hardware would be designed and operated to provide a greater potential for drywell spray initiation under very severe containment temperature and pressure conditions than exists in current procedures. It is for these severe containment conditions only that a substantial adverse impact is postulated. However, no analytic calculations have yet confirmed these postulated concerns regarding adverse impacts.

- o <u>Alternatives</u>: The options which appear to maximize the safety benefit and minimize the risk are those associated with assuring post RPV failure injection via the RPV injection lines with the drywell spray lines as a backup.

- o <u>Lack of Information</u>: There is some uncertainty as to the nature and extent of the negative safety impacts. The analysis of competing risk is based upon several assumptions, the extent of which may change as modeling techniques improve.

- o <u>Cost</u>: The cost estimates for reliable injection systems varied from the installation of two DC MOVs with power supply ($2M), to a fully bunkered dedicated system ($25M).

<u>Containment Wetwell Venting</u>

- o <u>Best Estimate Safety Benefit</u>: There are three principal situations for which venting has significantly different design requirements and potential impacts. Each of these are discussed below:

  - Decay Heat Removal: An alternate diverse way to remove heat from containment has a substantial safety benefit given the information previously developed for WASH-1400 and other published BWR PRAs. These benefits are more substantial if it can be assured that there are minimum impacts on equipment and personnel outside of the containment due to steam environment or loss of NPSH.

  - Core Melt Sequences: By providing a filtered pathway for radionuclide releases and non-condensibles, a positive safety benefit is again identified for the public. As before, these benefits are more substantial if it can be assured that there are minimum impacts on equipment and personnel outside the containment due to steam environment, loss of NPSH, or high radiation. Most notably recovery actions could be severely hampered.

- ATWS: Mitigation of an ATWS by providing a scrubbed release through the suppression pool is a possible outcome of these actions. This would likely result in substantial scrubbing of radionuclide releases and therefore a substantial reduction in potential health effects to the public.

o <u>Generic Resolution</u>: Hardware, operating philosophy, and local political requirements may all influence whether venting is a viable method of accident mitigation. It would appear that the following is true:

- Decay Heat Removal: Generic to a large extent, but NPSH requirements may dictate different emphasis.

- Core Melt Sequences: Because of political ramifications and potential adverse impact on operator (i.e., control room) and recovery actions (i.e., reactor building) this issue is not considered generically.

- ATWS: Most plants do not have the capability to adequately vent for a large part of the ATWS accident spectrum. This would require a new dedicated line of approximately 30" diameter or larger depending on the design basis requirements. In addition, some utilities have very strong opinions supported by analytic models which dispute the efficacy of such a venting scheme. Further, current work suggests that alternative ATWS mitigation schemes could make venting a less important option. Therefore, this issue is not considered to be generic.

o <u>Competing Risks</u>: There are substantial competing risks associated with each of the three situations:

- Decay Heat Removal: A potentially large plant specific concern exists regarding the capability of pumps taking suction from the suppression pool when venting the containment. However, this is judged on a generic basis, given the results of the Quad Cities and Browns Ferry tests, to be a concern that can be treated.

- Core Melt Sequences: In addition to the above, the release of radionuclides outside the containment may preclude recovery of key systems and may prevent operator actions to preserve or mitigate the situation.

- ATWS: Venting the containment may, with a not insignificant likelihood, induce a subsequent core melt due to loss of water in the suppression pool. Such a situation would be equivalent to the unscrubbed release now calculated for certain containment failure modes following an ATWS.

- o <u>Alternatives</u>: Alternative methods for achieving the venting functions are considered. They include duct vs. pipe, and variations in the sizes. The uncertainty surrounding modeling techniques (see next bullet) makes it difficult to assess which option is desirable.

- o <u>Lack of Information</u>: The question of mitigation of ATWS events in a BWR assuming failure of both the redundant control rod scram system and the backup reactivity control system has not been resolved. This postulated set of failures has previously been identified in PRAs and in deterministic analysis as resulting in severe core damage, but with extremely low frequencies (particularly after the NRC rule requirements are incorporated). There is at present extensive uncertainty regarding the optimum methods of mitigating such low probability conditions. While the current EPG procedures are used in the accompanying evaluation, future refinements in calculational models may provide assurance that slightly different procedures provide substantial improvement in the ability to mitigate these already low frequency ATWS sequences.

  Specifically, ORNL, PP&L, NYPA, EPRI, GE, and FAI have all indicated that venting would induce too many negative impacts or uncertainties to be considered for such cases. Rather suggestions for starving the core and using RHR capability as a mode of containment heat removal to buy time or completely mitigate the sequences have been suggested. This issue remains open.

- o <u>Cost</u>:

  - Decay Heat Removal: Approximately $1M
  - Core Melt Sequences: Approximately $1M
  - ATWS: Between $1M and $2M.

- o <u>Summary</u>:

  - Decay Heat Removal: The venting option dealing specifically with decay heat removal appears viable and may exist in some form at every BWR.
  - Core Melt Sequences: Before the venting option is considered for these situations a more detailed interface with severe accident management procedures appears to be required.
  - ATWS: More analysis is required to better define the best response.

Core Debris Barrier in the Drywell

- o Best Estimate Safety Benefit: For certain postulated core melt scenarios, molten material may be dispersed into the drywell and may flow toward the drywell shell. For such scenarios the resolution of this issue could prevent immediate failure of the drywell shell by placing a barrier in the path of the molten material.

- o Generic Resolution: The concept is generic. There are, however, significant differences among plants which would require significant plant specific variations.

- o Competing risks: The following adverse safety impacts may be associated with the debris barrier:

  - Interference with the LOCA flow area to the downcomers.

  - Impact on other accidents through interference if barrier pieces become dislodged.

  - Personnel ALARA for installation, repair, and interference with maintenance.

  - Possible seismic impacts and interface with seismic analyses.

- o Alternatives: There are several design alternatives for the barriers as discussed in Section 3. These options may vary as a function of the existing RPV, sump, and drywell shell designs (see Section 4).

- o Lack of Information: Modeling uncertainties have currently been the reason for consideration of the debris barrier changes. These uncertainties could be reduced in size either by additional analytic work or by installing the debris barriers. It should also be noted, however, that the debris barriers would affect only one of the postulated containment failure modes, other failure modes would persist within the modeling uncertainties that exist.

- o Cost: $.25 - $2M.

The remaining issues (Barrier (Torus Room), EPG Rev. 4, and Severe Accident Management Procedures) do not have detailed discussions at this time. They are being treated separately by the BWROG.

7.1.3 Other Considerations

This section summarizes some of the other conditions which could influence the decision making process including:

- o Coupling among changes;
- o Plant specific effects;
- o Accident sequence types.

Coupling Among Proposed Changes (Synergism)

The proposed changes and plant specific features are closely coupled and need to be treated in an integrated fashion. In some cases, the changes are inter-dependent, i.e., require the effective operation of all or most of the changes in order to succeed in providing effective additional mitigation beyond that which is already provided in the BWR containment. In addition, it is shown in Section 6.3 that there are other issues, phenomena, or failure modes that will still exist or be in the area of uncertain response; that is, not all containment failure modes would be eliminated even if all of these changes were implemented.

Plant Specific Features

The information compiled to date indicates that there is a great deal of similarity among BWRs in the NSSS equipment; the BOP systems, however, (and to a moderate degree containment systems) can vary among the Mark I and II plants by some significant yet subtle differences. Some of these differences were previously noted during the Mark I containment loads program. Again, in severe accident evaluations, it is found that there can be substantial differences among Mark I plants that can affect an assessment of severe accident phenomena and timing. The review of the plants which have responded to the BWROG questionnaire indicates that there is a wide diversity in the plant configuration for:

- o Containment venting
- o Debris coolability
- o Debris barrier

While these features may have similar conceptual fixes for all Mark I plants, it is clear that plant unique variations may result in substantially different details of design, costs, potential negative safety impacts, and benefits.

One of the clear pieces of information which results from this brief study is a conclusion similar to that for accident sequence quantification:

> The plants (mostly their BOP design) are sufficiently diverse that generic conclusions regarding the efficacy of certain actions cannot be made.

This conclusion applies to nearly all cases in which the design is outside the GE scope of work, i.e., is in the control of the architect-engineer. The backup for drywell sprays, the venting scheme, and the drywell debris barrier are all areas where a generic design may be difficult to implement, or the impact of the change may not be estimated generically. Features of plant design which influence the generic applicability of issues or solutions include the following:

- o Ratio of core power to drywell volume.
- o Ratio of core power to drywell floor area.
- o Volume and location of drywell sumps and floor volume below the downcomers in Mark I.
- o Arrangement of pedestal in Mark II plants.
- o Diesel fire pump capability to inject to the RPV and/or drywell with a high reliability during all postulated sequences including station blackout.
- o Proximity of sump bottom to drywell shell.

Accident Sequence Types

There are many types of accident scenarios which introduce different challenges to the containment and to the core. The proposed changes affect the integrity of containment during these challenges in different ways. Table 7.1-2 summarizes how the optimum implementation of the proposed changes might affect the identified accident sequence types. As seen in this table, only some accidents show an improved containment integrity following the implementation of the changes.

Table 7.1-2

POSSIBLE RESULT OF THE PROPOSED MODIFICATIONS ON DOMINANT ACCIDENT SEQUENCE TYPES

Accident Sequences	Improvement in Primary Containment Response to Severe Accidents Through Proposed Mitigation Measures		
	Debris Barrier	Vent	Spray
IA	Yes	Yes (R)	Yes
B	Yes	Yes (R)	Yes
II	No	Yes	No
III	Yes	Yes (R)	Yes
IV	No	No (V)	No
V	No	No	No

(R) Radionuclides released directly to the environment.

(V) The ATWS vent has not been included at this time because of the potentially severe downside risks that are not as yet fully assessed.

7.2 BWR MARK II OVERVIEW

While most of the discussion has been focused on the BWR Mark I plant, observations can also be reported regarding the Mark II plants. These observations are as follows:

Debris Barrier: The Mark II containments are so significantly different that a generic conclusion cannot be reached regarding the Mark II containment debris barrier. However, it appears that for most if not all, Mark II containments that a debris barrier of the concept discussed for the Mark I is not necessary, nor would it result in any measurable reduction in risk or containment failure probability, because of the following:

- o The drywell floor is significantly larger.
- o The downcomers provide a path to the wetwell before the material could reach the drywell shell.
- o The drywell shell is generally concrete, not a steel shell.

Debris Coolability: The containment drywell injection (or sprays) issues are very similar to those discussed for the BWR Mark I. The exceptions to this similarity are the following:

- o There are no reactor building to wetwell vacuum breakers on the Mark II. This results in the following:
  - Lower likelihood of the introduction of oxygen into the containment during spray operation(no venting assumed).
  - A potentially higher differential pressure from the reactor building to the drywell.
- o There is another difference in the designs which is reflected in the higher negative Δp pressure capability for the Mark II (i.e., approximately - 10 psid, although in some Mark IIs the upward Δp across the drywell floor may be limited to - 5 psid) vs. the Mark I (approximate - 2 psid).

Note that for some Mark II designs (each Mark II appears to have substantial differences in drywell/pedestal configurations) the use of drywell sprays may not be an effective means of debris coolability. In those cases RPV injection may be the only viable method of debris coolability.

Venting: These considerations are very similar between the Mark I and II.

Table 7.2-1 summarizes the key components of the decision making process as applied to the Mark II containments.

Table 7.2-1
IDENTIFICATION OF KEY ITEMS IN THE
DECISION MAKING PROCESS FOR
THE IDENTIFIED NRC ITEMS
MARK II

ITEM		BEST ESTIMATE SAFETY BENEFIT (RISK REDUCTION)	GENERIC RESOLUTION	COMPETING RISKS	ALTERNATIVES OR OPTIONS	LACK OF INFORMATION OR UNCERTAINTY	COST ($M)
H_2 Control Inerting (MK I,II)		M	Yes	Small; plant personnel	Not Considered	Uncertainty on Full Positive Impact	Complete
Debris Coolability (Including Station Blackout)		H	Concept-yes Hardware-no	Yes; potentially large if high flow spray	Options Assessed	Uncertainty on the Negative Safety Impacts of High Flow Sprays	$.2-$25M
Vent	Decay Heat	H	Yes	Small	Not considered	Yes, Small	$1M
	Core Melt	M	No	Moderate	Not considered	Yes, Moderate	$1M
	ATWS	M	No	Large	On-going work	On-going work	$1-2M
Barrier Drywell		L	No (Large Differences Among Plant DW Designs)	Small	Yes	No	$.6-5M
EPG Rev 4		H	Yes	No	Yes	No	$.3-.8
Severe Accident Management Procedures		M	No	Small	Yes	Scope Unknown	Unknown

Legend: Qualitative in nature, not based on detailed calculations

H - High benefit: potential complete mitigation of some failure modes by virtue of the single modification; or prevention of core damage and adverse plant conditions which could impact all safety systems.

M - Moderate benefit: potential complete mitigation of some failure modes if other active systems operate successfully.

L - Low benefit: mitigation of less important failure modes or requirement for multiple active systems in addition to the proposed change or mitigation of radionuclide releases but not prevention of containment failure.

7.3 MARK III OVERVIEW

Table 7.3-1 summarizes the possible effects on a Mark III plant but these have not been examined in detail.

Table 7.3-1
IDENTIFICATION OF KEY ITEMS IN THE
DECISION MAKING PROCESS FOR
THE IDENTIFIED NRC ITEMS
MARK III

ITEM		BEST ESTIMATE SAFETY BENEFIT (RISK REDUCTION)	GENERIC RESOLUTION	COMPETING RISKS	ALTERNATIVES OR OPTIONS	LACK OF INFORMATION OR UNCERTAINTY	COST ($M)
H_2 Control Igniters		M	Yes	None	Not Considered	Yes	Not Available
Debris Coolability (Including Station Blackout; Drywell Sprays Not Available)		H	Concept-yes Hardware-no	Small	Options Assessed	Not Identified	$.2-$25M
Vent	Decay Heat	H	Yes	Small	Not considered	Yes, Small	$1M
	Core Melt	M	No	Moderate	Not considered	Yes, Moderate	$1M
	ATWS	M	No	Large	Yes, On-going work	Yes, On-going work High	$1-2M
Barrier Drywell		NA	NA	NA	NA	NA	NA
EPG Rev 4		H	Yes	No	Yes	No	$.3-.8
Severe Accident Management Procedures		M	No	Small	Yes	Scope Unknown	Unknown

Legend: Qualitative in nature, not based on detailed calculations

H - High benefit: potential complete mitigation of some failure modes by virtue of the single modification; or prevention of core damage and adverse plant conditions which could impact all safety systems.

M - Moderate benefit: potential complete mitigation of some failure modes if other active systems operate successfully.

L - Low benefit: mitigation of less important failure modes or requirement for multiple active systems in addition to the proposed change or mitigation of radionuclide releases but not prevention of containment failure.

References

[7-1] Policy Statement on Safety Goals for the Operation of Nuclear Power Plants, June 18, 1986.

[7-2] Kolaczkowski, A., Reference Plant Accident Sequence Likelihood Characterization: Peach Bottom Unit 2, NUREG/CR-4550 (Draft)

[7-3] Silberberg, M., et. al., "Reassessment of the Technical Bases for Estimating Source Terms", NUREG-0956 (Draft), July 1985.

[7-4] Denning, R.S., et. al., "Radionuclide Calculations for Selected Severe Accident Scenarios", NUREG/CR-4624, dated July 1986.

Appendix A
CONTAINMENT EVENT TREE

A.1 INTRODUCTION

The bridge between the core melt accident sequences and establishing the "character" of sequences can be viewed as a containment event tree (CET) in which containment conditions affecting the radionuclide releases associated with a given core melt sequence are established. The CET also includes active and passive mitigation measures which would change the radionuclide release.

The Shoreham PRA [A-5] published in June 1983 was the first published BWR PRA to use a detailed CET for the calculation of public risk. IDCOR has developed a detailed containment event tree (CET) for BWR containments (see IDCOR Task 4.1 dated October 1983). This detailed CET can form the basis for precise calculations of risk by allocating containment failure modes into a finer mesh of bins for consequence analyses than is used in some published BWR PRAs. The IDCOR Task 4.1 CET has many parallels to the recently developed set of containment questions from Sandia [A-2]. For the precise determination of risk, it may be appropriate to use such detailed CETs. On the other hand, such detail can prove to be both difficult to use in explaining the process and resource intensive for quantification of risk.

For the purposes of defining major phenomena of interest and for use in understanding potential mitigation measures it may be more appropriate to have a simplified CET. A simplified CET, augmented by functional fault trees, can provide a clear way to display the information. The IDCOR TSR reported BWR risk results using a simplified CET and therefore reflects a significant conservatism in the quantitative risk measures. Despite these conservatisms, the net result of the evaluation was to conclude that the risk associated with BWR Mark I's is substantially below the safety goals and acceptable. However, because of the increase in attention being focused on one segment of the probabilistic risk analysis, i.e., the containment performance, it is judged appropriate to provide slightly more detail than is included in the IDCOR TSR containment event tree. However, it is also judged important to preserve the relative simplicity of the

BWR containment in its response to severe accidents. This more straight forward approach has the side benefit that it facilitates technical discussions among users and reviewers. Thus the simplified CET is chosen as appropriate.

The objective of this appendix is to establish a framework for identifying a spectrum of accident scenarios which may occur following core melt and lead to varying containment challenges and radionuclide releases to the environment. Two reasonable boundary conditions are established: 1) the MAAP code accurately models the physical processes it addresses, and 2) the entry condition to the containment event tree (CET) is any possible severe accident sequence and its corresponding containment, i.e. intact or failed containment prior to initiation of core melting.

The conditions necessary for fission products to reach the environment can be represented by two general event trees: 1) accident sequence event tree defining severe accident scenarios; 2) a CET, including containment failure modes and timing. The spectrum of scenarios represented by these two trees can be treated by evaluating the impact of four aspects of an accident. These four aspects, which directly affect the nature of the sequence are as follows:

- The core melt accident class (i.e. the entry state to the CET);
- The containment mitigating system response (post core melt);
- The containment failure path and size;
- The reactor/auxiliary building response.

Knowledge of these four aspects of an accident will allow an analyst to assess the radionuclide releases and containment failure probability. Subsequently, an environmental transport and offsite consequence analysis could be performed if the public health consequences were to be determined.

Figure A.1-1 presents a pictorial representation of how the component parts of this sequence selection model are combined. As the figure illustrates, the outcomes of this combination process are accident progression scenarios following core melt which could result in varying levels of radionuclide release.

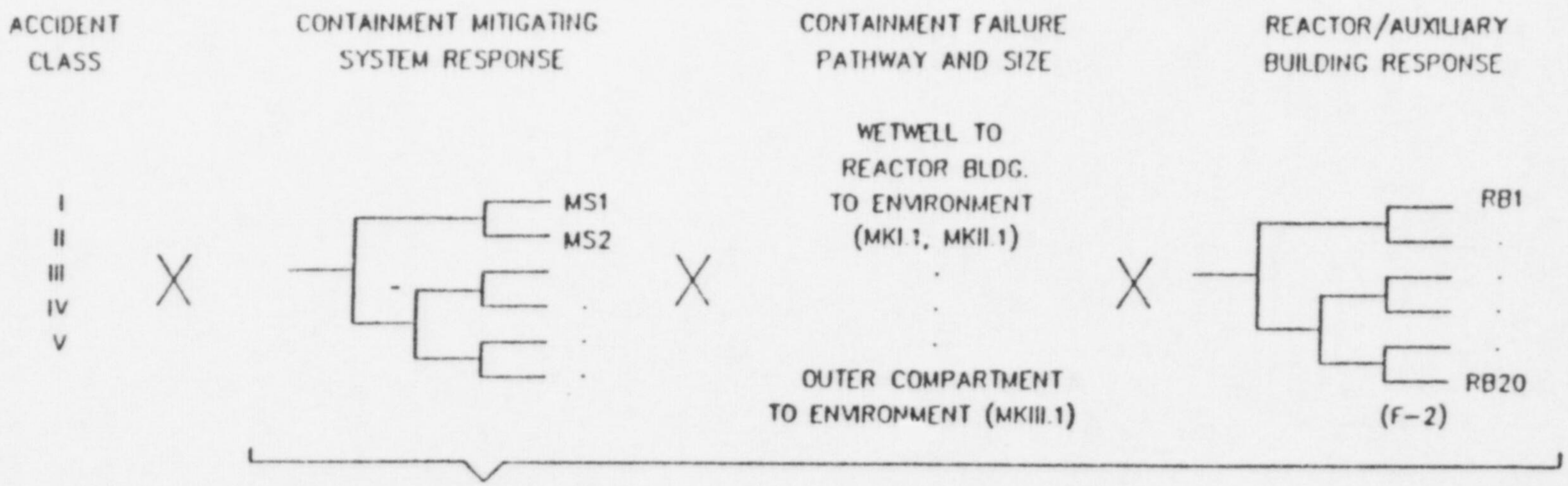

Figure A.1-1 Overview of the Containment Event Tree Evaluation Process for Radionuclide Release Determination

A.2 ACCIDENT CLASS

An accident class is the core melt end state in which the condition of the reactor pressure vessel, the containment, and the status of the mitigating systems are defined. Table A.2-1 defines the types of accident scenarios in terms of eleven accident class bins and gives sequence examples for each. This is the entry state for the containment event tree (CET).

A.3 CONTAINMENT EVENT TREE FOR CLASSES I AND III

This section includes the following items:

- The containment event tree for Class I and III. (A.3.1)
- The functional fault trees describing each mode of the containment event tree. (A.3.2)
- The quantitative summary of each of the functional fault trees for each accident class and a qualitative description plus references to support the quantification. (A.3.2)
- A summary of the quantification of the containment end states. (A.5)

A.3.1 <u>The Containment Event Tree</u>

The following are the key aspects of the CET development which dominate the determination of the radionuclide releases.

The scope of the mitigating systems assessment is to evaluate those systems which cannot prevent core melting or are not available when needed to prevent core melting, but can reduce the public health impact of a core melt if available, i.e. during release and/or containment challenge. This evaluation is strongly sequence-dependent, i.e. both the impact on containment conditions due to the system operating and the conditional availability of the mitigating system are found to vary among sequences.

Each of nodal decision points in the CET represents a group of systems or phenomena which can reduce the consequences of a core melt accident. In many cases, containment failure can be prevented, thereby preventing all but a negligible amount of fission products from being released to the environment.

Table A.2-1
SUMMARY OF THE CORE VULNERABLE ACCIDENT SEQUENCE SUBCLASSES
(PLANT DAMAGE BINS)

ACCIDENT CLASS DESIGNATOR	SUBCLASS	DEFINITION	EXAMPLE
CLASS I	A	Accident Sequences Involving Loss of Inventory Makeup in which the Reactor Pressure Remains High	TQUX
	B	Accident Sequences Involving a Loss of Off-Site Power and Loss of Coolant Inventory Makeup	T_EQUV
	C	Accident Sequences Involving a Loss of Coolant Inventory Induced by an ATWS Sequence	T_TC_MQU
	D	Accident Sequences Involving a Loss of Coolant Inventory Makeup in which Reactor Pressure has been Successfully Reduced to 200 psi.; Accident Sequences Initiated by Common Mode Failures Disabling Multiple Systems (ECCS) Leading to Loss of Coolant Inventory Makeup	TQUV
	E	Accident Sequences caused by Common Mode Failures which Result in Multiple Front-line System Failures with the Reactor at High Pressure	$T_DDC2DC3$
CLASS II	-	Accident Sequences Involving a Loss of Containment Heat Removal	TW
CLASS III	A	Accident Sequences Leading to Core Vulnerable Conditions Initiated by Vessel Rupture where the Containment Integrity is not Breached in the Initial Time Phase of the Accident	R
	B	Accident Sequences Initiated to Resulting in Medium or Large LOCAs for which the Reactor Cannot be Depressurized	S_1QUX
	C	Accident Sequences Initiated or Resulting in Medium or Large LOCAs for which the Reactor is at Low Pressure	AQUV
	D	Accident Sequences which are Initiated by a LOCA or RPV Failure and for which the Vapor Suppression System in Inadequate, Challenging the Containment Integrity	AD
CLASS IV	-	Accident Sequences Involving Failure to Insert Negative Reactivity Leading to a Containment Vulnerable Condition Due to High Containment Pressure	$T_TC_MC_2$
CLASS V	-	Unisolated LOCA Outside Containment	

Figure A.3-1 is the CET developed for accident sequence types in which core melt occurs into an intact containment. The following discussion provides (1) a qualitative discussion of the CET functional events; (2) functional level fault trees describing the principal failure modes of each node; and (3) the basis for the quantification of each branch in terms of the functional fault tree.

Figure A.3.1 (Sheets a, b, and c) is the Class I and III CET. This CET portrays those containment mitigation features (active and passive) which can affect the integrity of containment or the radionuclide releases from containment.

The functional events that are included are the following:

- Containment isolation and initial integrity
- Active Mitigation: drywell temperature control
- Passive Mitigation: containment of the debris
- Containment Heat Removal
  - RHR
  - venting
- Containment breach size
  - leakage
  - overpressure failures
- Location of containment breach
  - drywell
  - wetwell airspace
- Reactor building effectiveness

These top level functional events are described in more detail below. The accident sequence designators are provided for every accident sequence. Adjacent to each sequence end state are the conditional probability of that state and the type of end state. The end state types can be used to describe varying levels of containment performance depending on the purpose of the investigation. Risk evaluations would require this knowledge of the spectrum of releases which reflect both the magnitude and frequency of the radionuclide releases.

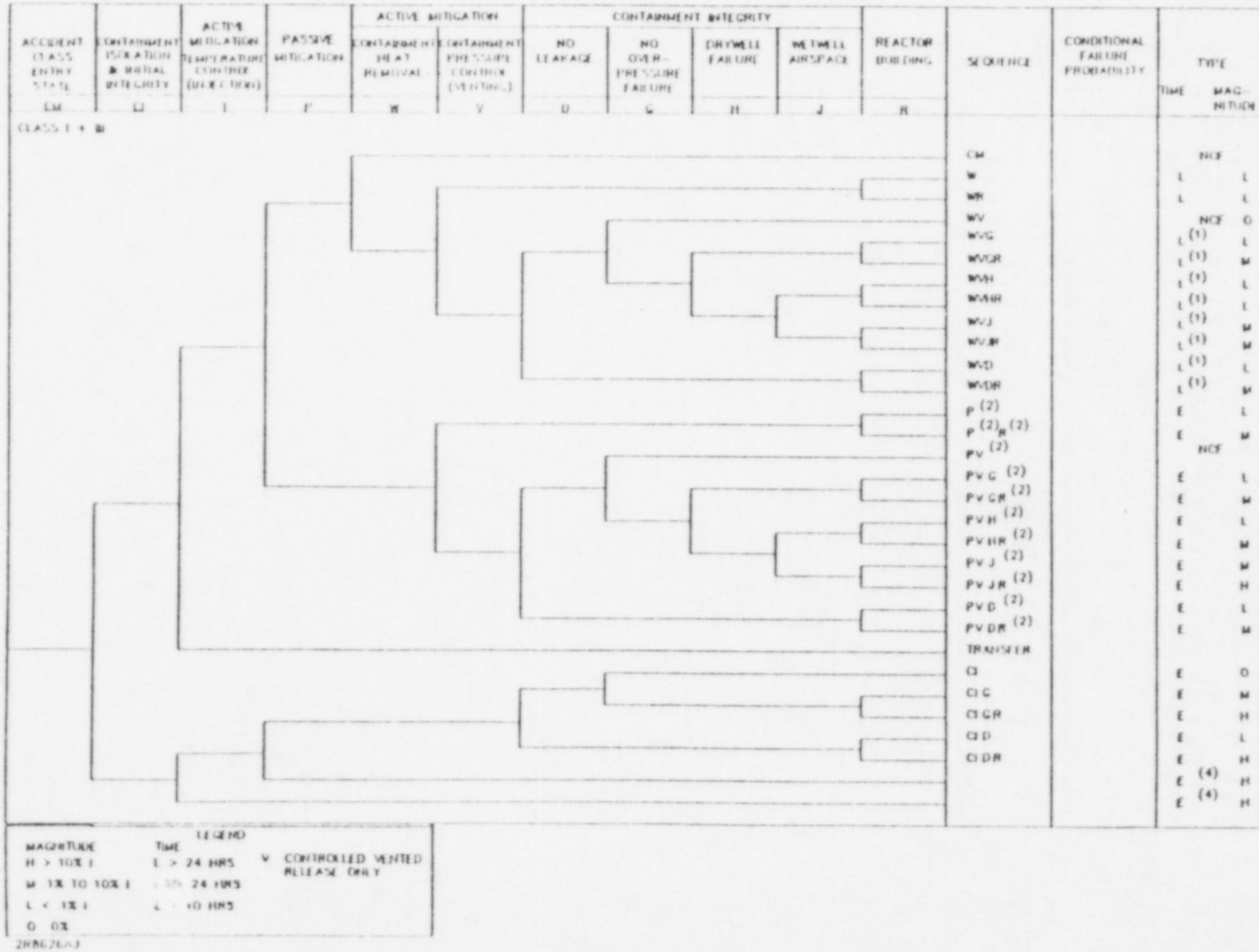

Figure A.3-1 Containment Event Tree for Classes I and III

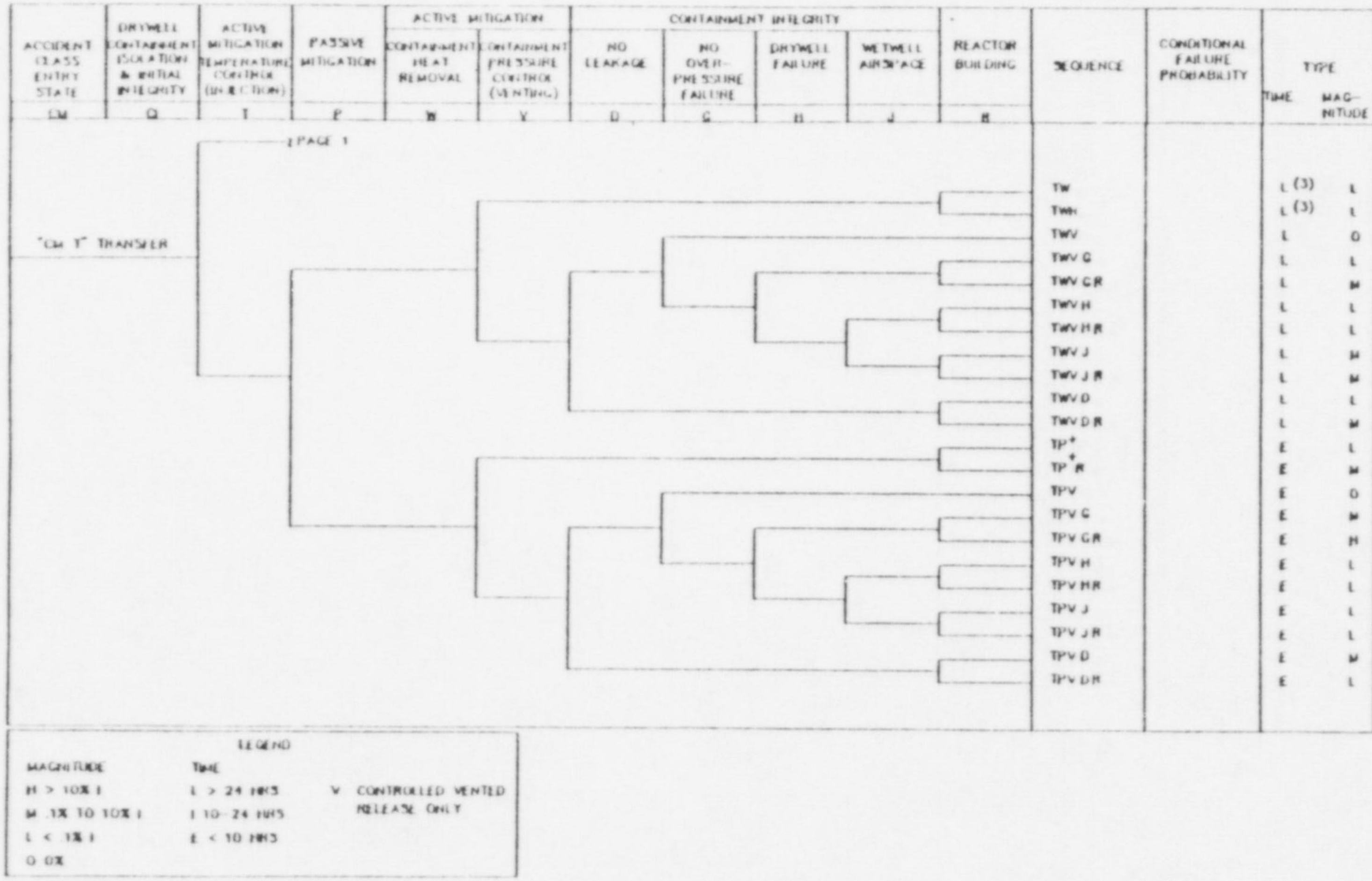

Figure A.3-1 (cont.) Containment Event Tree for Classes I and III

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Notes for Figure A.3-1

(1) Debris coolability is found to have a high likelihood of success if coolant injection can be successful.

(2) Consequences for failure of passive mitigation is determined by the plant specific configuration and the possibility of molten material spreading across the drywell floor.

MAAP calculations indicate that molten material will not flow to the drywell shell therefore the conditional probability of passive failure of the containment due to direct contact with molten material is assessed as negligible. (i.e., 1E-1).

(3) Temperature induced failures of 6 hours following RPV breach. It is judged based on Task 17.5 that this is a small leakage for vented cases. However, plant specific consideration could alter this perception.

(4) While these sequences should be transferred to additional event trees for evaluation it is judged adequate for our purposes to establish that these sequences are of potentially high radionuclide source terms and early containment failures such that these sequences represent unacceptable results for the current evaluation.

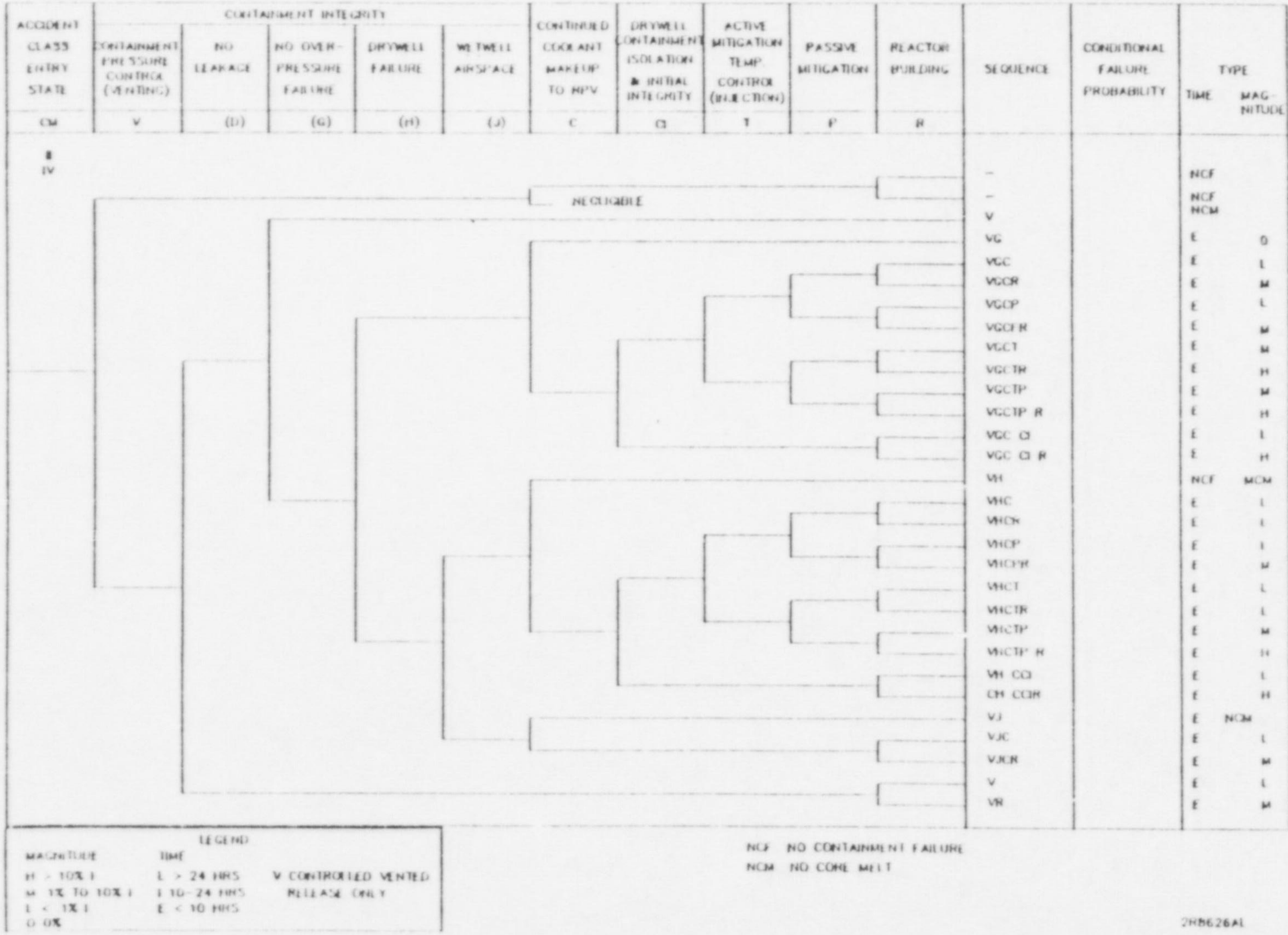

Figure A.3-2 Containment Event Tree for Classes II and IV

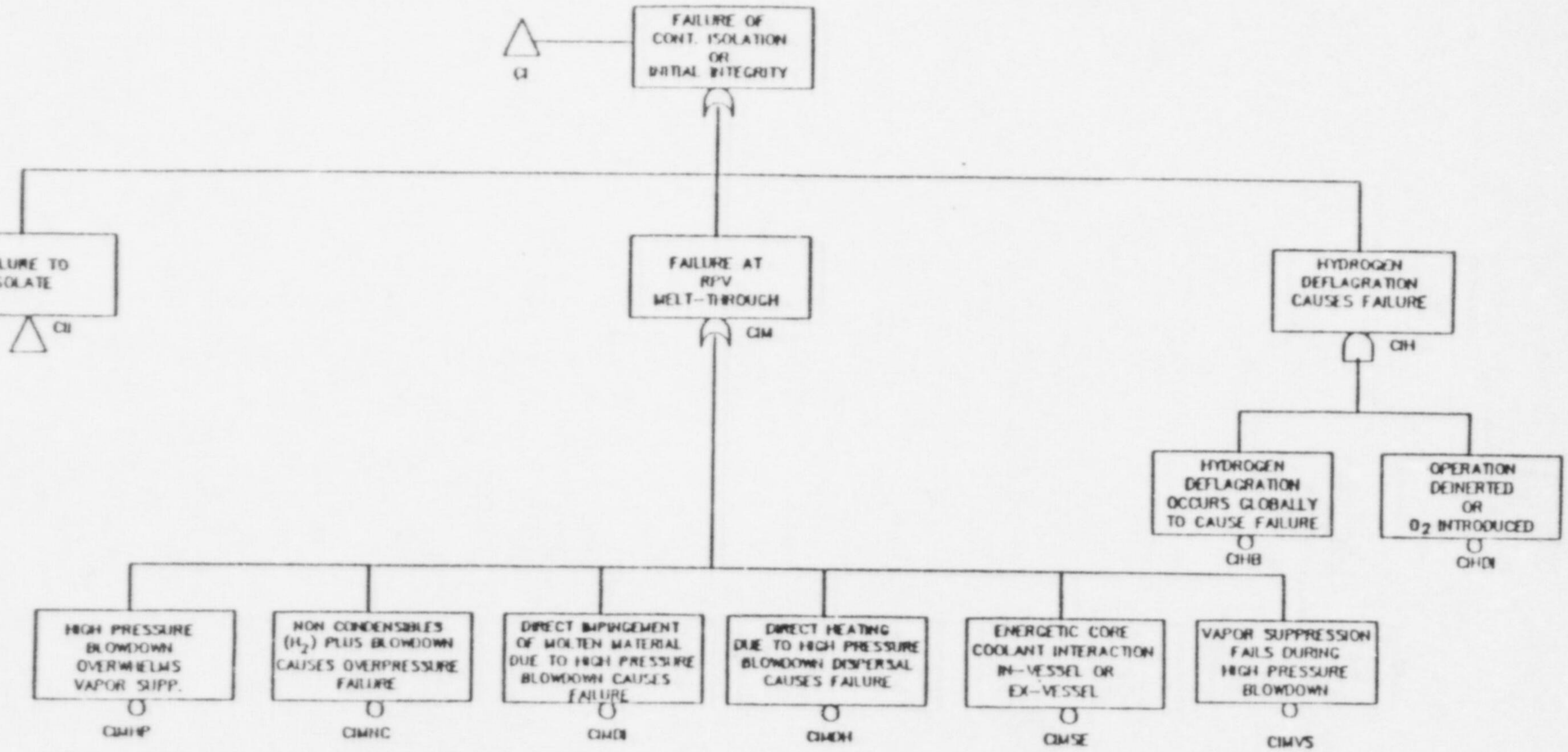

Figure A.3-2 Early Containment Failure Modes Included in the Event Tree Quantification

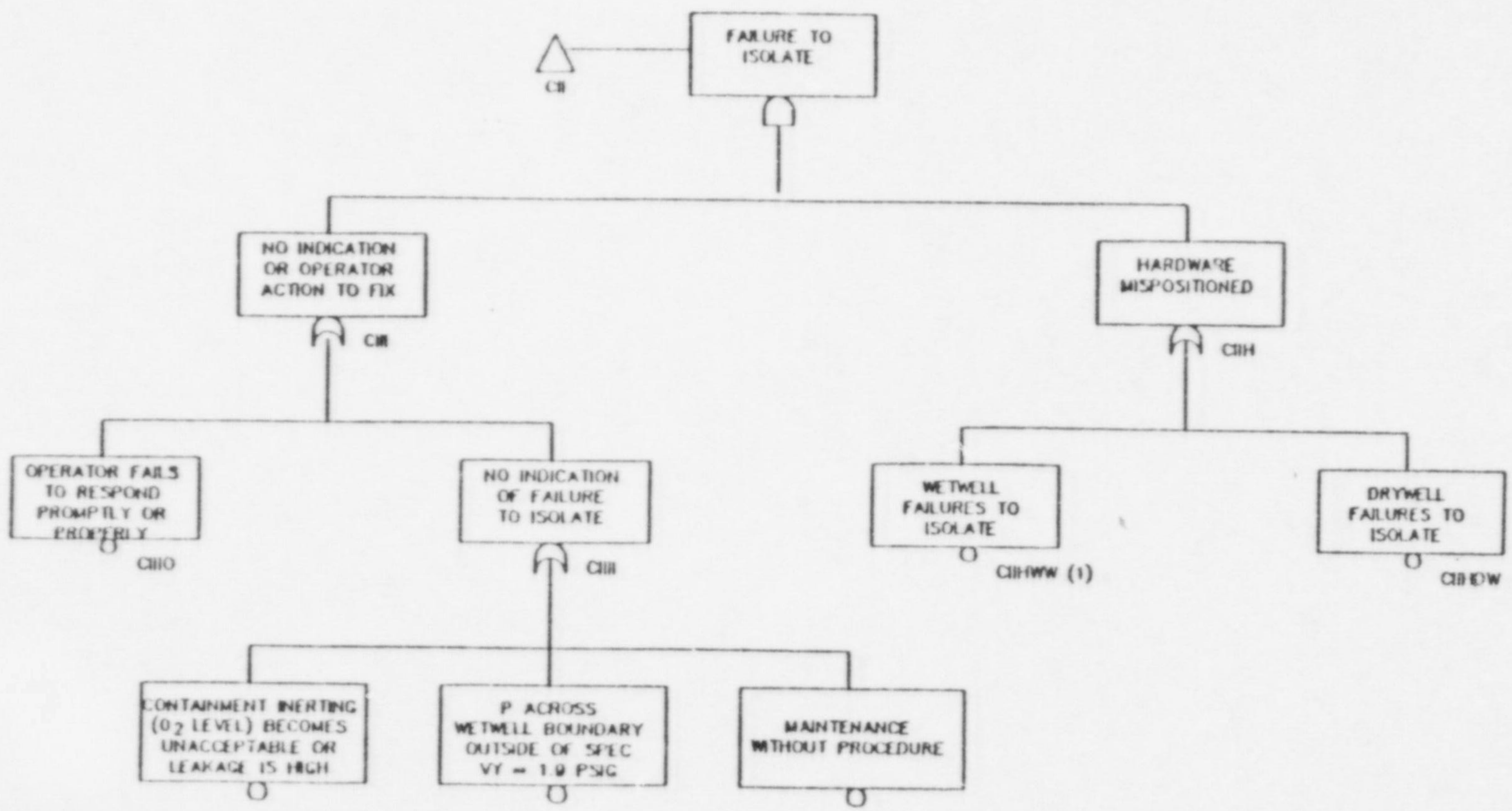

Figure A.3-2 (continued)

(1) Not addressed because failure would prove to be a beneficial containment state with minimum radionuclide release.

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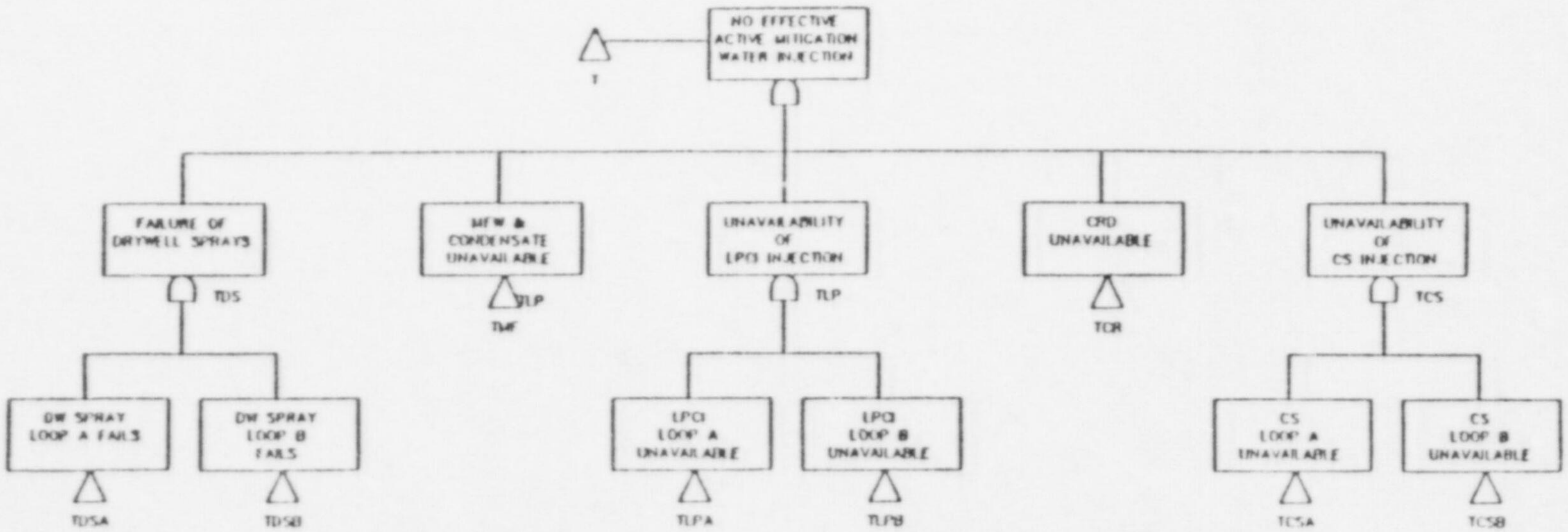

Figure A.3-3 Containment Temperature Control Using Active Mitigation Through Water Injection

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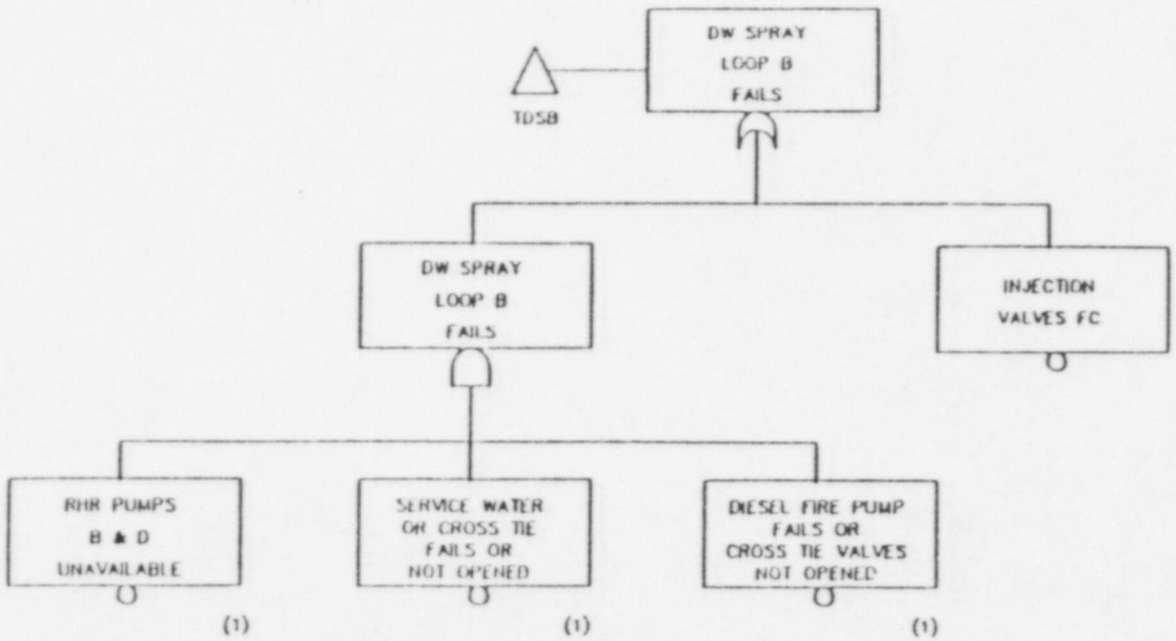
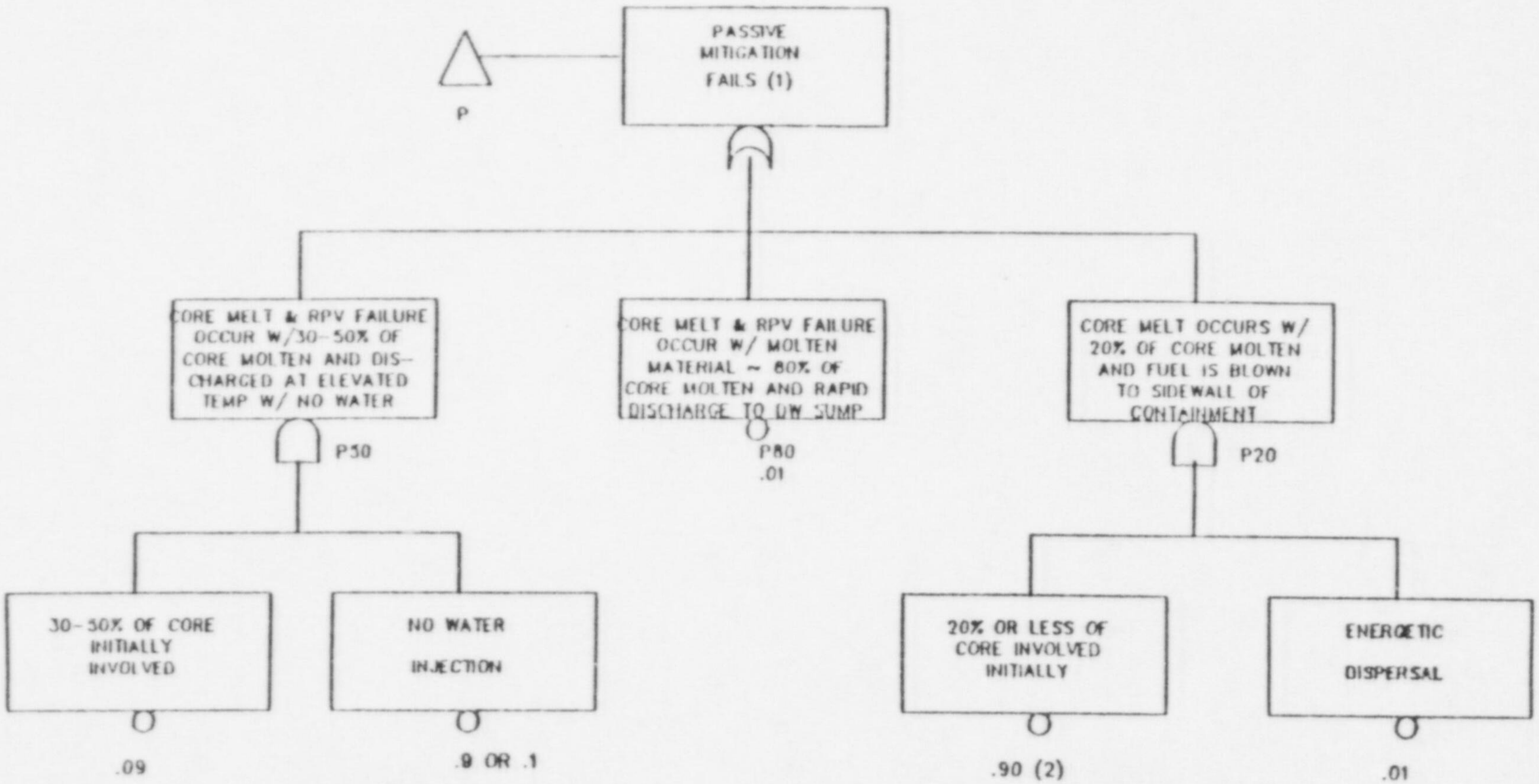


(1) These are the same supplies to the LPCI injection capability identified under TLPB.

Figure A.3-3 (continued)



(1) Direct contact between DW shell and molten material or high temperature induced shell failure
(2) IDCOR best estimate

Figure A.3-4 Passive Containment Mitigation Fails

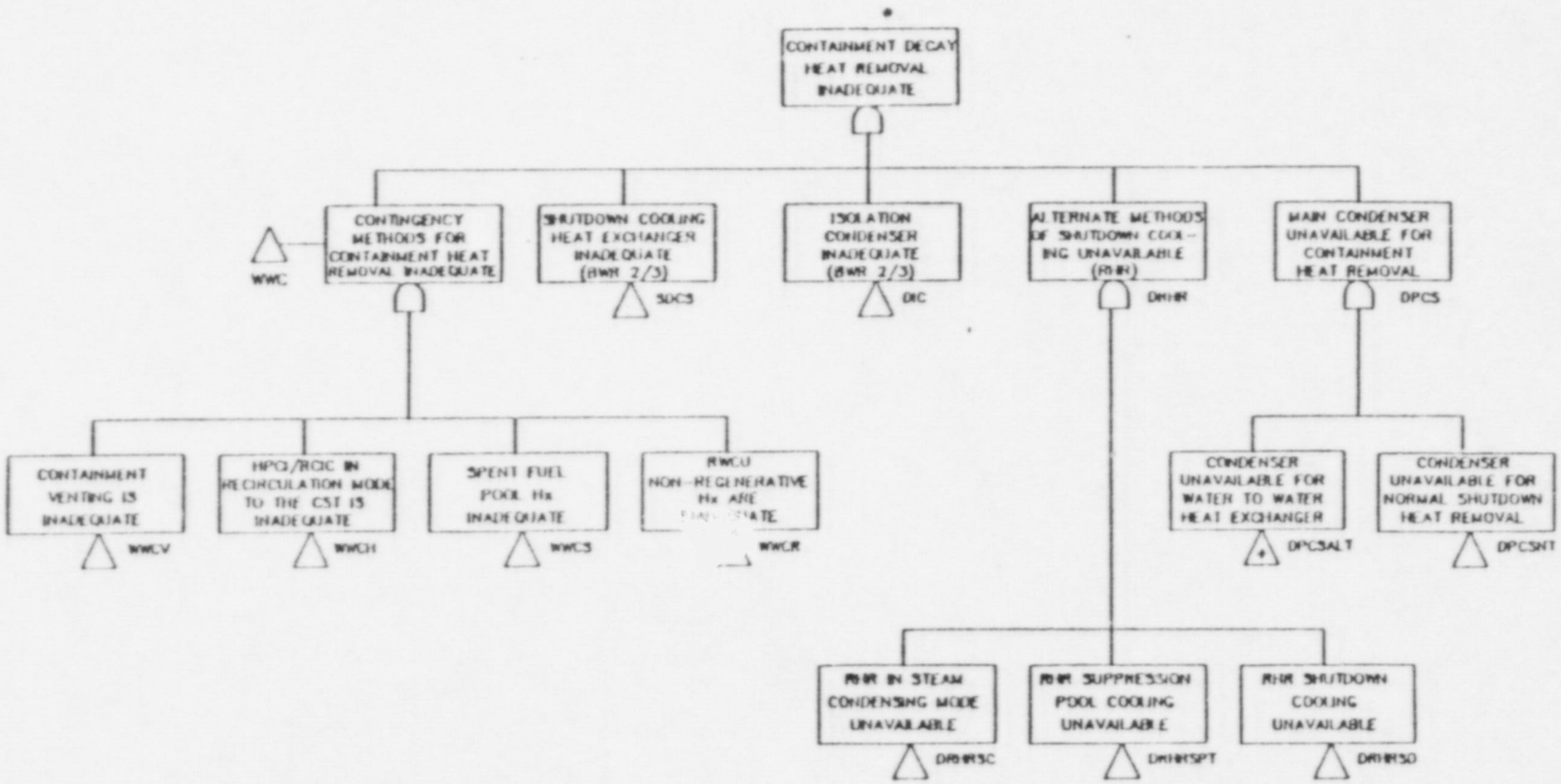


Figure A.3-5 Functional Fault Tree Describing the Possible Containment Heat Removal Options

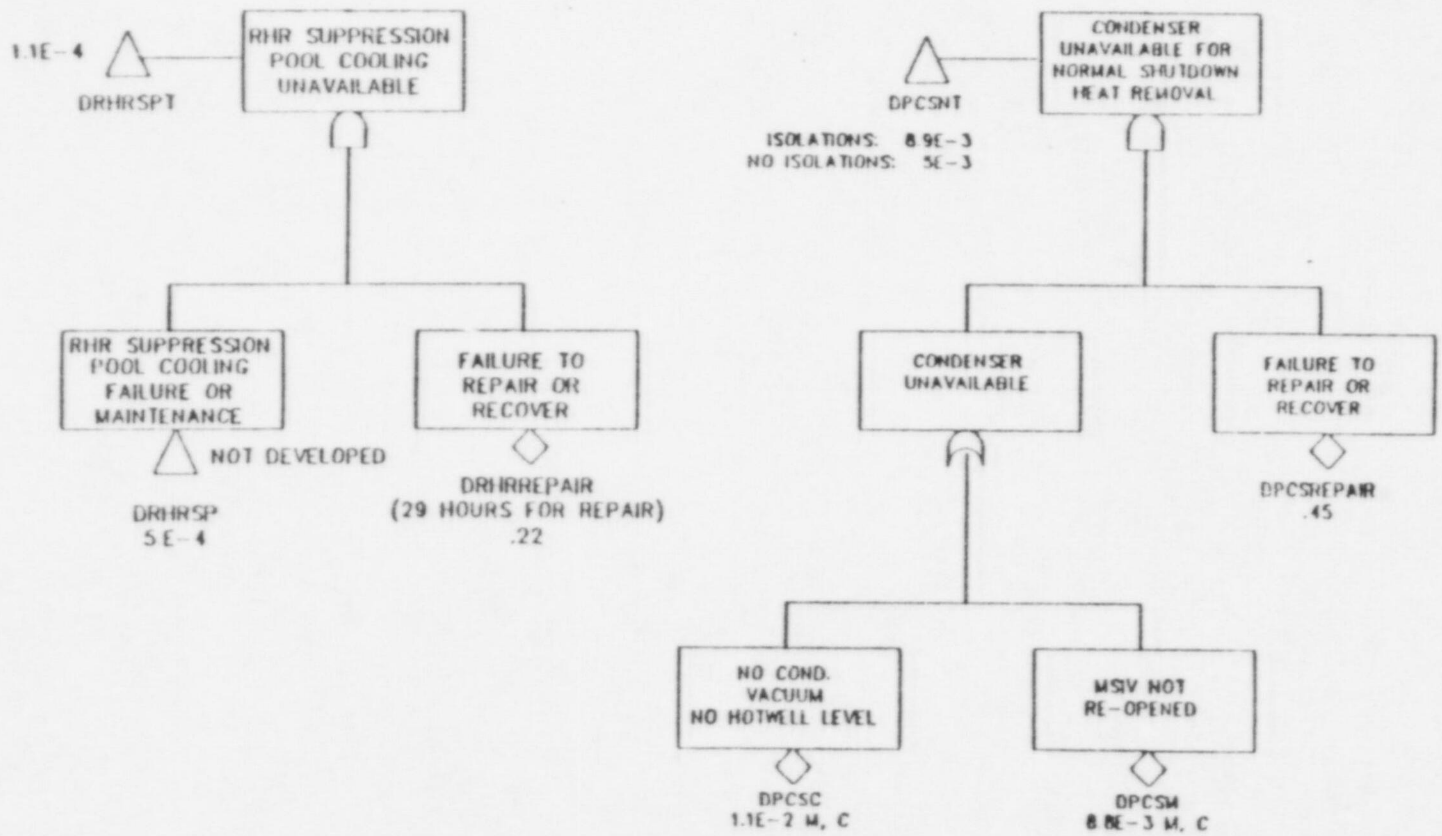
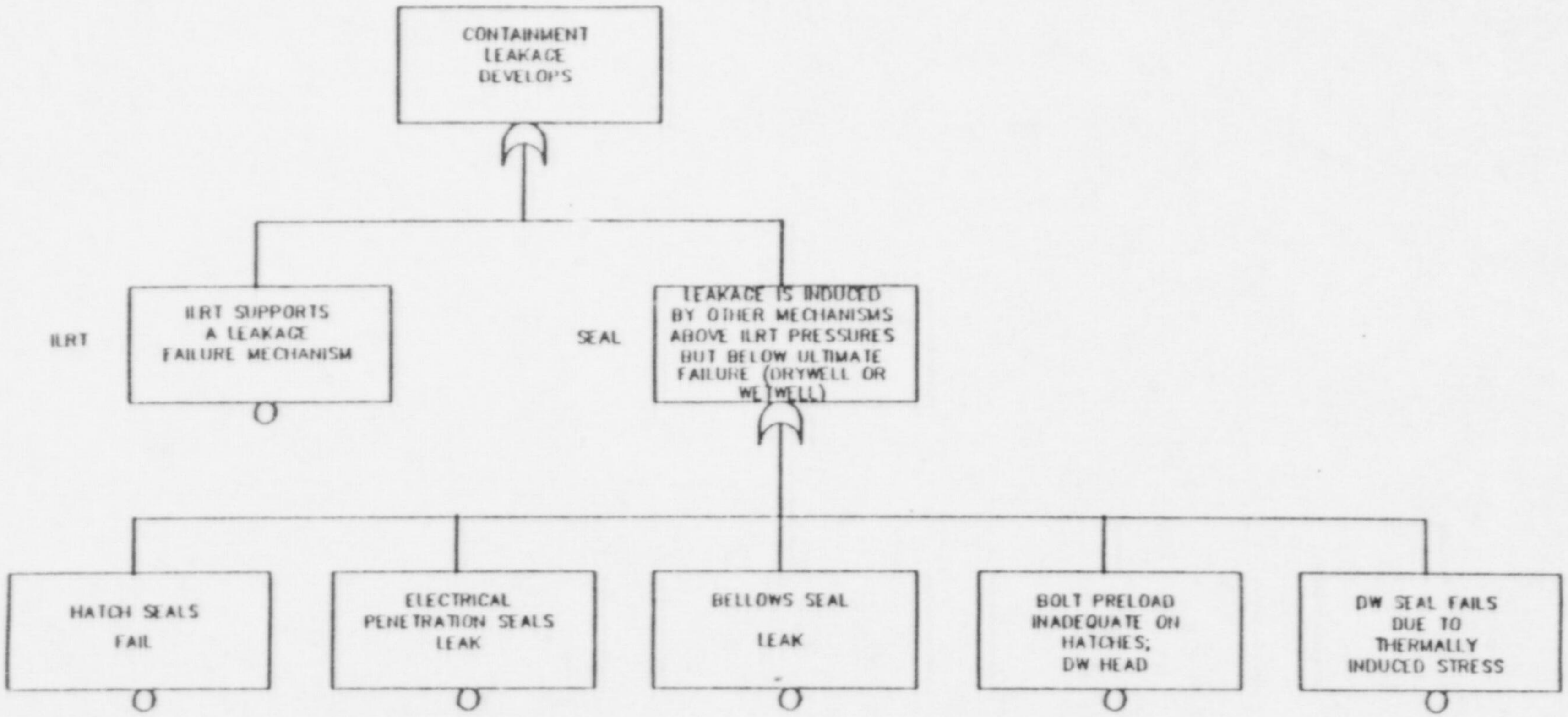


Figure A.3-5 (cont.) Functional Fault Tree Describing the Possible Containment Heat Removal Options



Note: Containment leakage as defined here is a small containment leakage pathway or pathways which limit the amount of material released to the reactor building and force the material through a tortuous path causing relatively high deposition rates along the pathway. The leak is sufficient to terminate the containment pressure rise and is within the capacity of the SGTS.

Figure A.3-6 Functional Fault Tree to Describe the Conditional Probability of Containment Leakage Failure

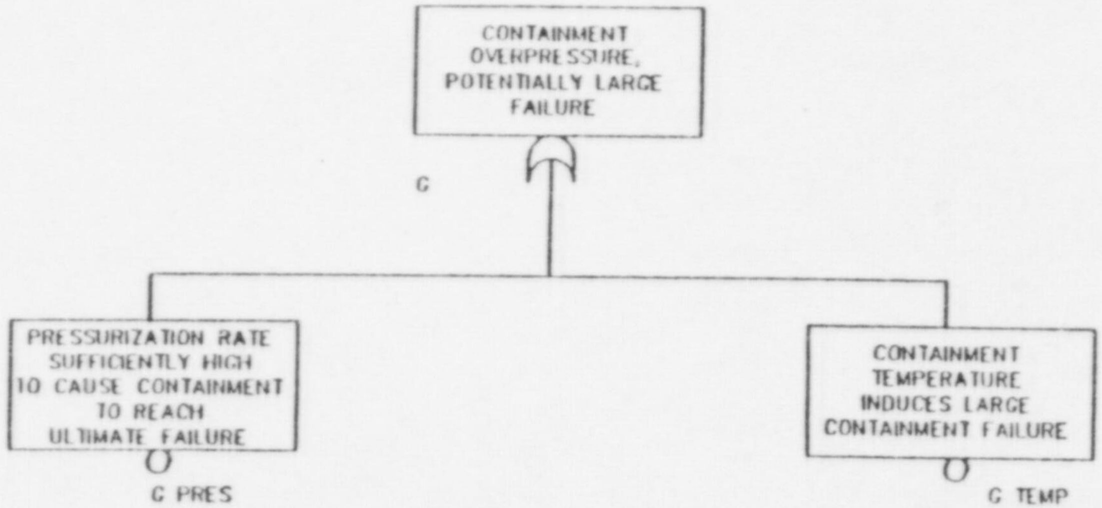


Figure A.3-7 Functional Fault Tree to Describe the Conditional Probability of Containment Overpressure Given no Leakage for Cases Without:

- Containment Heat Removal
- Coolant Injection
- Passive Resistance

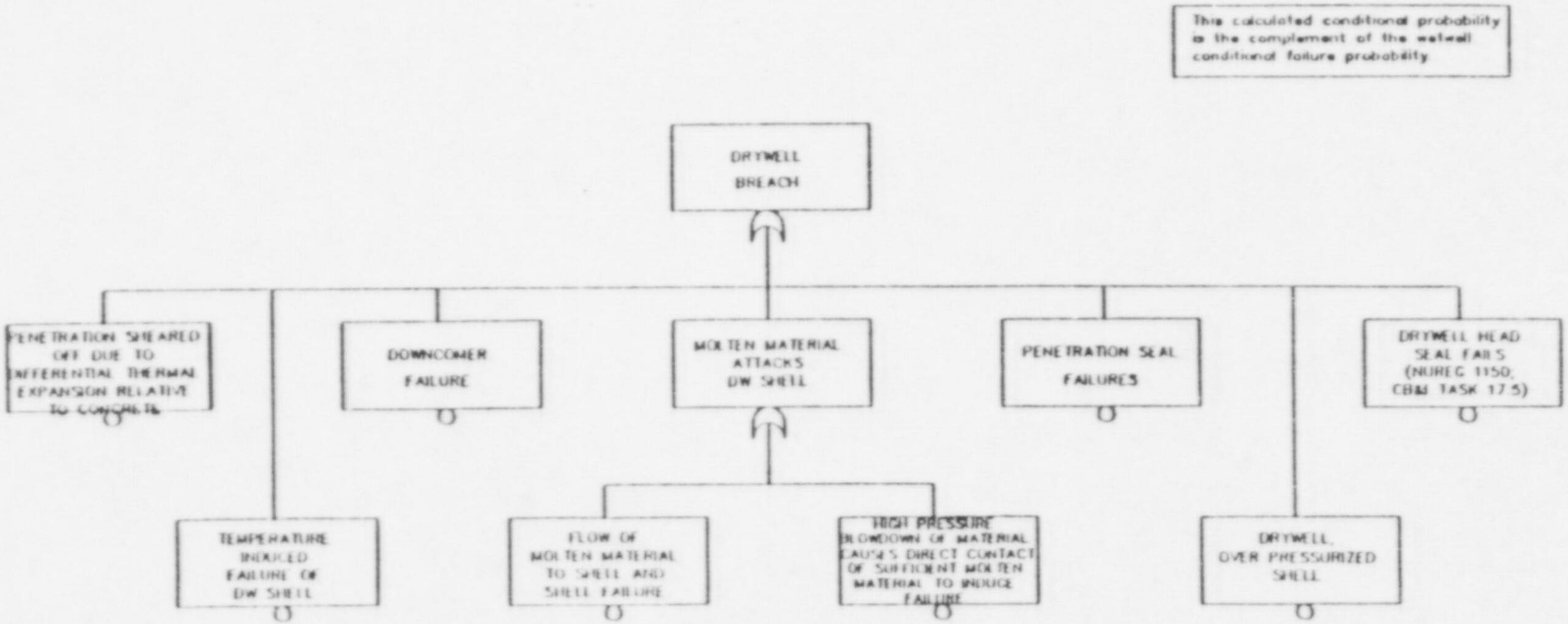
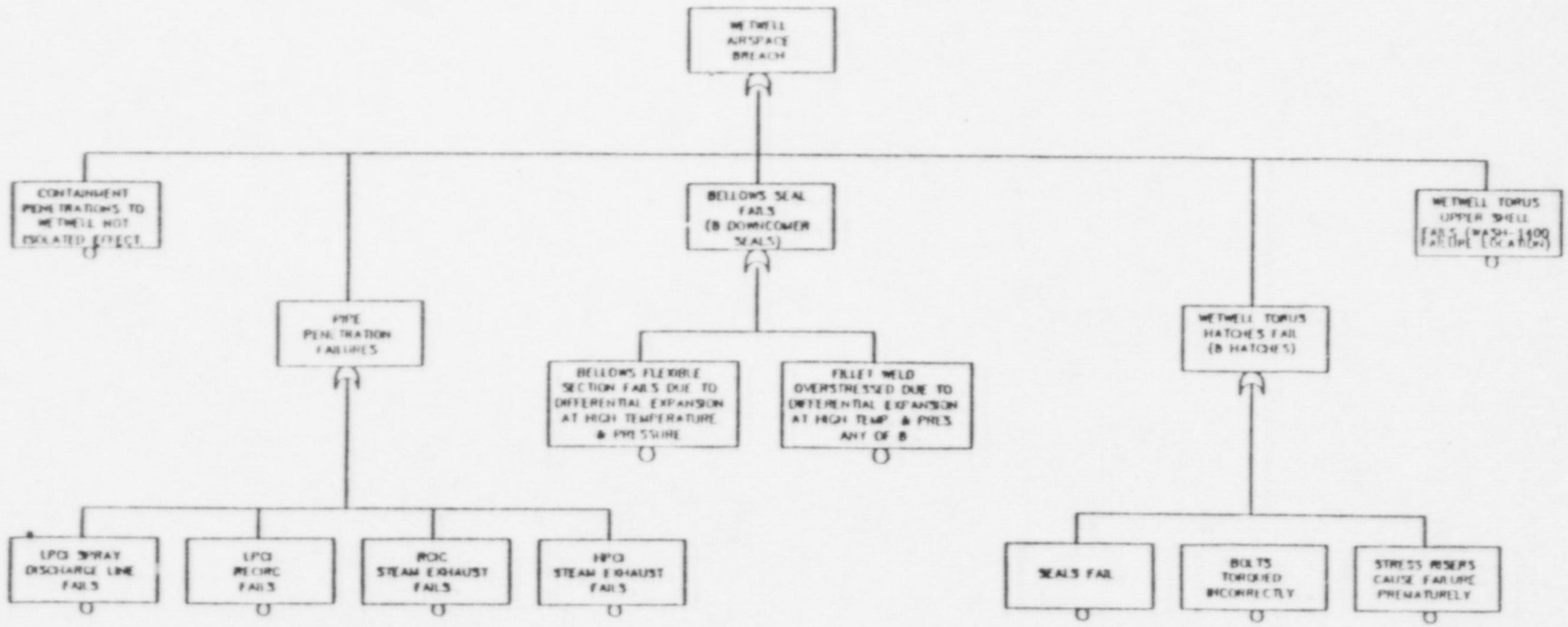


Figure A.3-8 Functional Fault Tree to Describe the Conditional Failure Probability of the Wetwell versus the Drywell



Note that the drywell is pressurized at Vermont Yankee. This would assist in identifying pre-existing leakage in the drywell.

Figure A.3-8 (cont.) Functional Fault Tree to Describe the Conditional Failure Probability of the Wetwell versus the Drywell

2-R-85-024

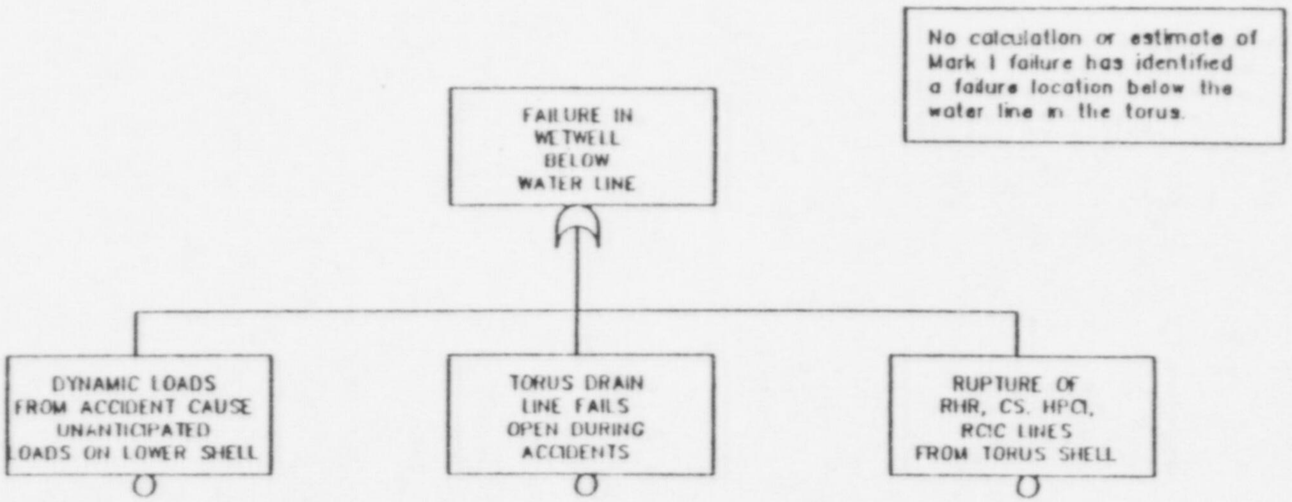


Figure A.3-9 Functional Fault Tree to Describe the Conditional Probability of a Wetwell Failure below the Waterline given a Failure not in the Drywell

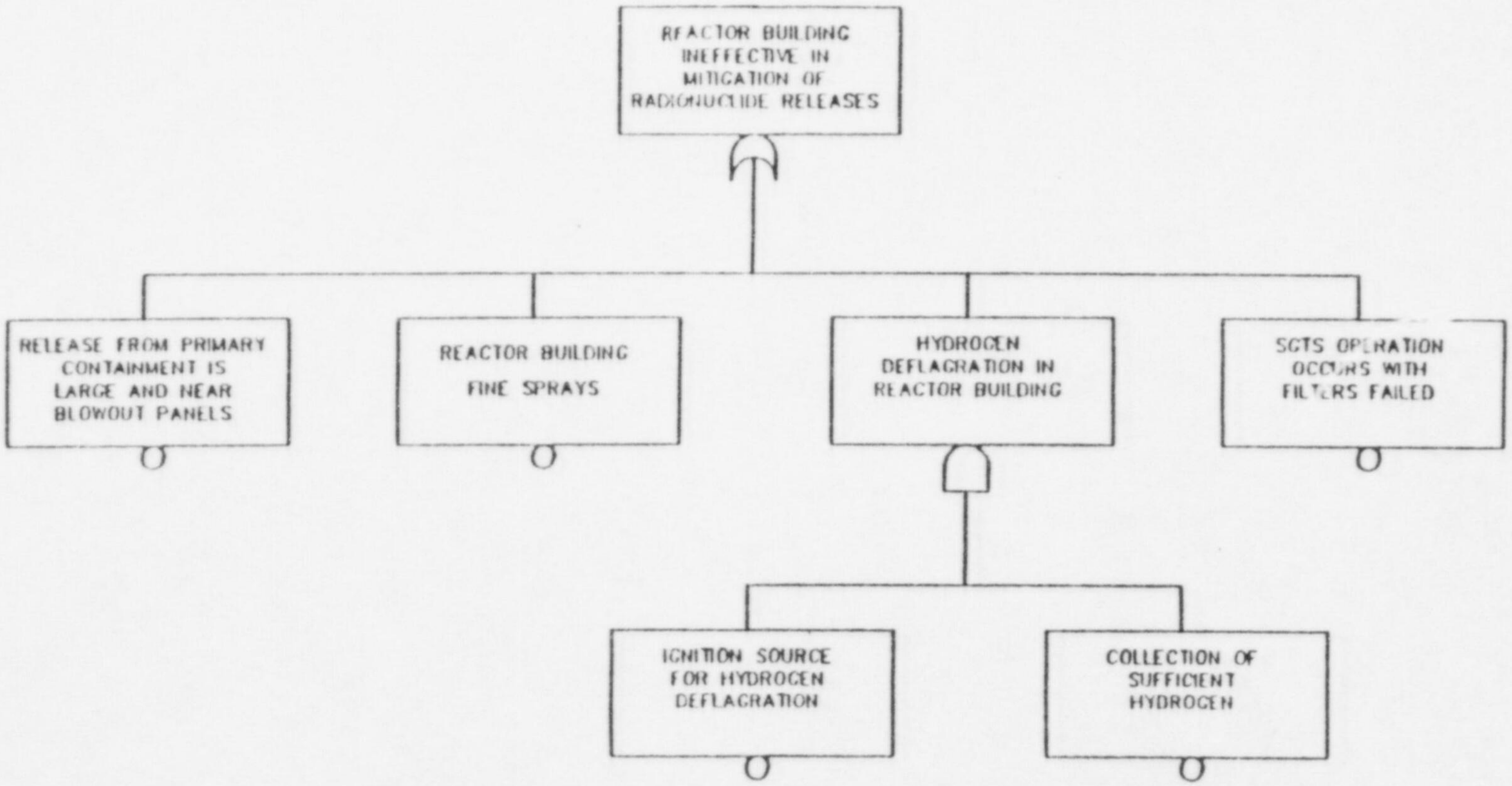


Figure A.3-10 Functional Fault Tree to Describe the Conditional Failure Probabilityof Effective Reactor Building Scrubbing of Fission Products

The following description of the CET top functional events can be used on a plant specific basis to define the plant capability.

- Early Containment Failure (CI) (Figure A.3-2)

  Early containment failures are characterized by:

  - early releases, except the majority of the release may be delayed and occur only if revaporization cannot be prevented;

  - low energy at time of release, because early containment failure will preclude coincident release and containment failure at high pressure and energy (low energy releases cause higher likelihood of fatalities near the site);

  - short warning time for first releases (however, may still be long warning time for the major portion of the release).

  Two potential failure modes which can lead to early containment failure are the following:

  - Containment Isolation Failure (CI) (CII)

    A containment isolation failure is an early containment failure mode, possibly existing or caused at the time of accident initiation.

  - Overpressure Failure Early (CIM, CIH)

    An overpressure failure early is a containment failure mode resulting from the accident initiator and/or the subsequent core melt phenomena causing rising containment pressure.

Containment/Drywell Sprays

Containment or drywell sprays can mitigate the consequences of a potential core melt accident. The sprays can perform three functions, the two most important of which are scrubbing fission product releases which are not otherwise scrubbed (i.e. in the case where the suppression pool is bypassed), and providing water to cool the core debris on the drywell floor after the vessel fails (Mark I & II). In this latter function, cooling the core debris can occur in two time phases. The first phase is cooling the core debris relatively soon after vessel failure to prevent a vaporization release. The second phase would be operation of the sprays after the vaporization has occurred. In this mode of operation, containment failure could be prevented by termination of drywell wall heating and the associated thermal stresses which can lead to containment failure. The third function which can be performed by the containment sprays is to remove heat during a hydrogen burn.

Vessel Water Injection

RPV water injection can perform some of the same functions as spray operation mentioned above, i.e. it can terminate a vaporization release and prevent containment overtemperature failure. Many systems exist which can perform this function; however, these systems may be failed if a core damage accident has occurred at low pressure. Therefore they may have to be recovered or long-term water sources aligned. However, if the core melt accident occurred because of inability to depressurize, then the core melt could induce RPV depressurization and make coolant injection possible (Classes IA, IB, and IE, and possibly IV).

The systems which might perform the function of coolant injection post core melt are dependent on the plant design, but may include:

- Electric or diesel fire water pumps
- Control Rod Drive pumps
- Low Pressure Coolant Injection
- Core Spray
- HPCS
- Condensate Pumps
- Feedwater (motor-driven pumps)

These same systems can perform another function as well. Water injected into the vessel will cool the RPV downcomer. This cooling will prevent fission product revaporization from the RPV downcomer and therefore prevent them from being released to containment where they might pass through to the environment.

Operation of the vessel water injection systems mentioned above after vessel failure in a Mark I will act to cool the core debris that remains on the drywell floor. Cooling the debris will terminate any vaporization release from the debris. It will also cool the drywell atmosphere and minimize or eliminate revaporization. The post core melt water injection and associated steam also will prevent the drywell from reaching very high temperatures. For Classes II, IV and V, with the containment already failed, preventing the drywell from overheating will prevent revaporization of the fission products deposited on drywell surfaces. For Classes I and III, with the containment intact, preventing the drywell from overheating will prevent overtemperature failure of either the drywell liner or the penetrations.

Containment Heat Removal

If containment or drywell sprays are operating, or if vessel water injection is available to cool the core debris, containment heat removal can provide a heat transport path to complete the requirements for a stable condition which dramatically reduces the potential for radionuclide releases to the environment.

If containment heat removal is available or recovered before containment overpressure failure in the case of Classes I and III, the containment will remain intact and releases to the environment will be negligible.

Venting

This event represents venting of the drywell or wetwell vents to the Standby Gas Treatment System (SGTS), directly to the atmosphere, or to the reactor building. Venting will limit containment pressure and, therefore, may prevent an uncontrolled release of fission products to the environment by preventing containment overpressure failures. Alternatively wetwell venting may be used to control the release and ensure that any radionuclide release must traverse the suppression pool.

Containment Failure Path and Size

As shown in Figure A.1-1, the next consideration imposed on this process after having investigated the mitigating system possibilities, is to focus on potential pathways out of containment. The containment failure location and its size may have dramatic impacts on the calculated radionuclide releases. The determination of these items may depend on the core melt accident sequence and the mitigating systems available coupled with the deterministic evaluation of the containment failure location.

Evaluations by Ames Laboratory and CB&I have developed the following conclusions:

- The drywell head seal could be a failure location on overtemperature and pressure.
- The bellows seal in the wetwell airspace is another potential overpressure failure site.

Reactor/Auxiliary Building Model

The next step in the CET process as shown in Figure A.1-1 is the consideration of the reactor building status during the postulated core melt sequences. Contributors to Reactor Building effectiveness include the following:

- Reactor/Auxiliary Building Integrity after Containment Failure
- SGTS Operation
- Fire Sprinkler Operation
- Hydrogen Combustion in the Reactor Building
- Reactor/Auxiliary Building Integrity after Combustion

Figure A.3-11 provides a simplified schematic of potentially important fission product removal mechanisms in the reactor building.

These events are discussed below.

THE MK I SECONDARY CONTAINMENT DESIGN PROVIDES THREE FISSION PRODUCT REMOVAL MECHANISMS: SETTLING AND DEPOSITION; WASH OUT BY FIRE PROTECTION SYSTEM SPRAYS; AND THE ACTION OF THE STANDBY GAS TREATMENT SYSTEM (SGTS)

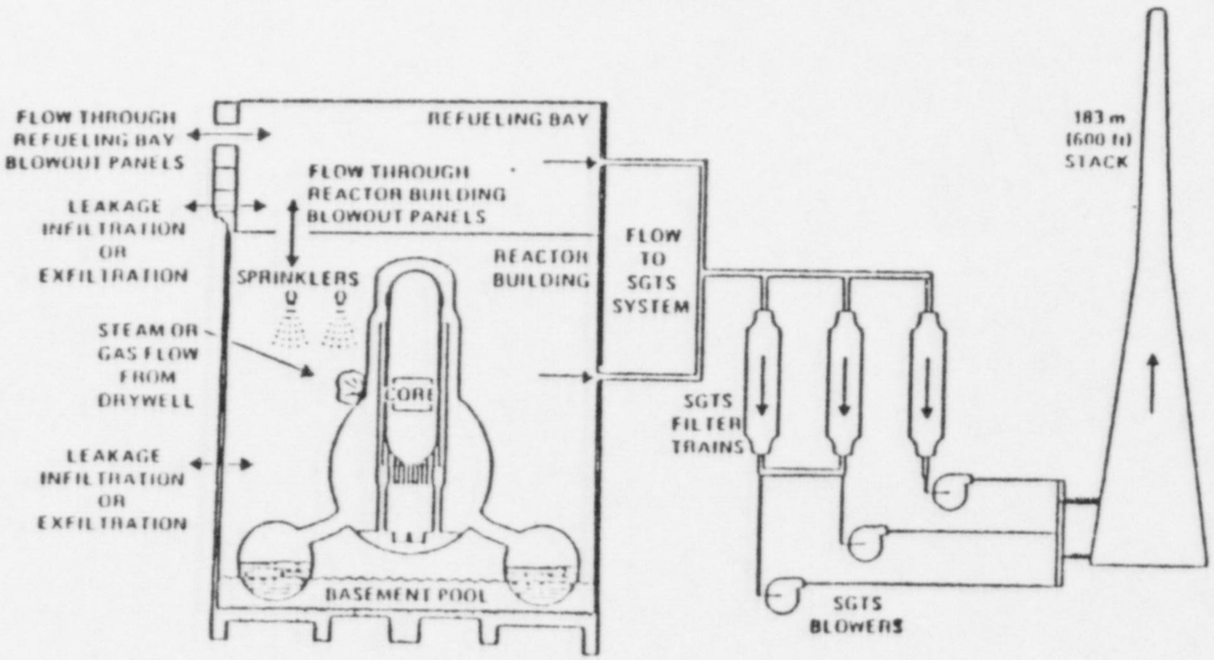


Figure A.3-11 Potentially Important Fission Product Release Mechanisms in t Reactor Building

- Reactor/Auxiliary Building Integrity After Containment Failure

  The reactor building surrounds the primary containment for a Mark I design (See Figure A.3-12). In either case, if failure causes the containment to leak to this "secondary" containment, the fission products will have another opportunity for natural removal before release to the environment. However, because the reactor/auxiliary building may have blowout panels with a low tolerance for change in pressure, the primary containment failure may cause a substantial release pathway for airborne radionuclides. The location of the failure will vary depending on the plant and the location and size of the primary containment failure. Even if the reactor/auxiliary building has a blowout panel open, the radionuclide removal processes in the reactor building can be very important to minimizing overall fission product releases to the environment. IDCOR's analysis of Peach Bottom (Mark I design) resulted in factors of between 10 and 20 reduction in releases for some containment failure sizes. Some sizes had smaller reduction factors, however. The size (and possibly the location) of primary containment failure will determine the "certainty" with which one can expect to maintain reactor or auxiliary building integrity and the corresponding expectation for a large amount of fission product retention. The size of the containment failure will also impact the aerosol loading in the reactor or auxiliary building and therefore the natural removal rate of releases through that building.

- Standby Gas Treatment System (SGTS) Operability

  The SGTS includes fans and filters which process fission product releases from the primary and secondary containment. Because they were not designed for the conditions existing in severe accident sequences, the SGTS may fail as a result. If the SGTS is not operating, filters do not remove fission products; however, the fans increase the flow rate through the building, reduce the residence time and reduce the natural removal. If the SGTS is operating, the filters will collect aerosols, both fission product bearing aerosols and inert aerosols. After a certain limit is reached the filter may overload and tear. Before tearing, removal rates by the filters are high, much higher than natural removal. However, if the filters tear, the fans will remain running and the residence time in the building may be substantially reduced. Because the filters are no longer available, this may substantially reduce the natural removal rate over the case where no system was operating. Consequently, SGTS operation can alter fission product releases sufficiently to merit consideration in sequence selection.

- Fire Sprinkler System Operation

  Figure A.3-13 provides a simplified schematic of the possible effects of fire sprinklers in the reactor building.

BWR PRIMARY CONTAINMENT IS PROVIDED BY THE DRYWELL AND WETWELL; SECONDARY CONTAINMENT IS PROVIDED BY THE SURROUNDING STRUCTURE

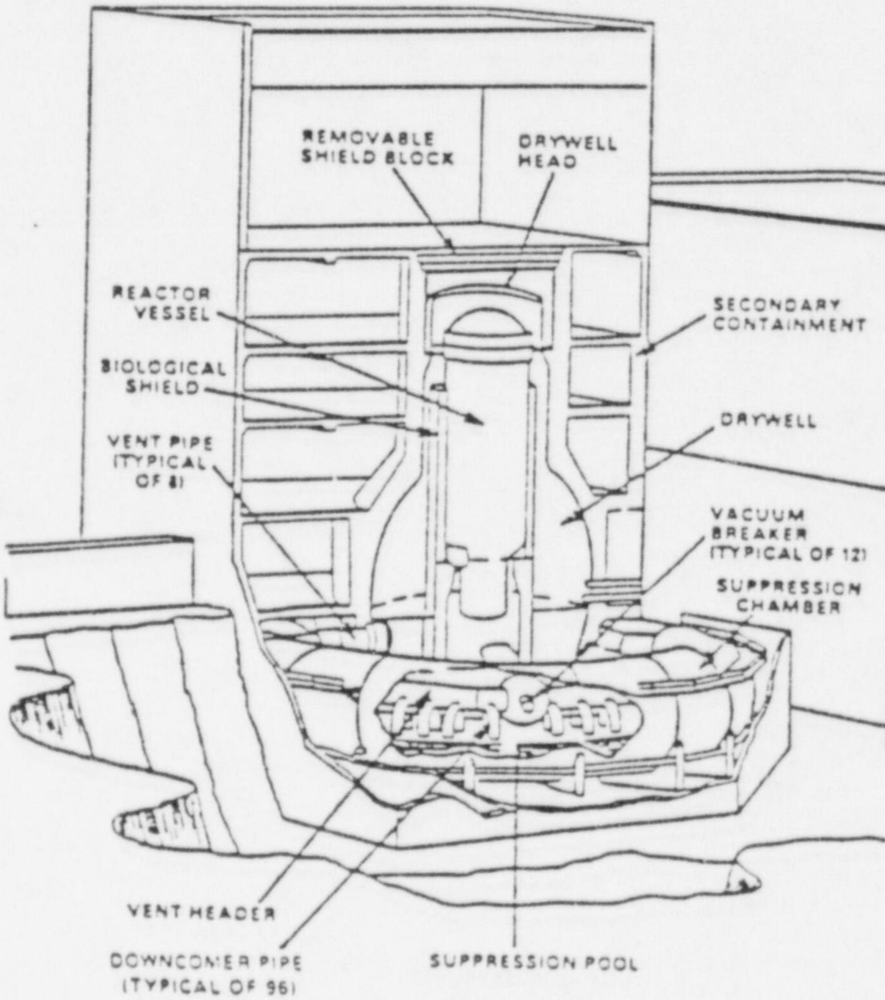
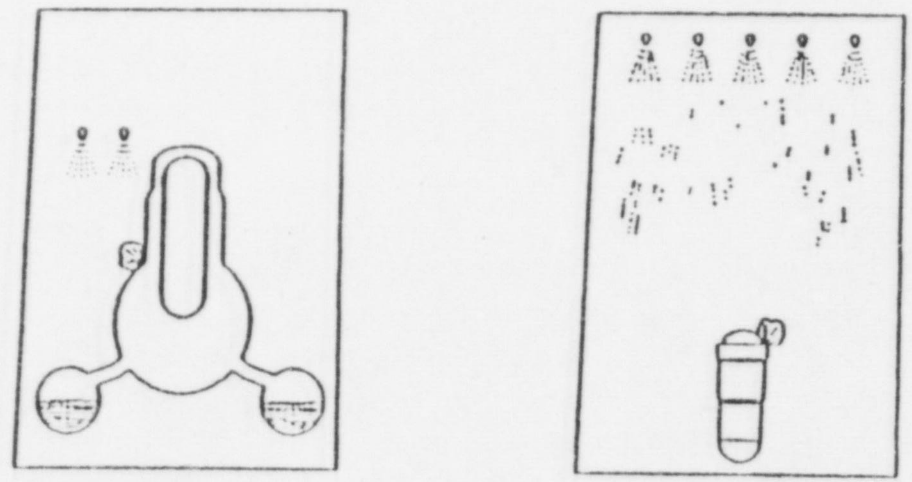


Figure A.3-12 Primary and Secondary Containment for BWR Mark I and II Designs

THE EFFECT OF BROWNS FERRY FIRE PROTECTION SYSTEM SPRAYS IS SIMILAR TO THAT OF THE SPRAYS IN THE LARGE PWR CONTAINMENTS



PEACH BOTTOM HAS ONLY CABLE TRAY SPRAYS

Figure A.3-13 Possible Effects of Fire Sprinklers in the Reactor Building

Hydrogen Burn in the Reactor/Auxiliary Building

A hydrogen burn in the reactor building can cause failure of an intact reactor building or alter the chemical form of cesium iodine to elemental iodine. The burning, however, would tend to be oxygen limited.

The possibility of hydrogen combustion is heavily sequence and plant specific. ORNL has performed extensive BWR evaluations. Some of the work performed by ORNL includes evaluation of reactor building effectiveness including questions regarding hydrogen burning. Figures A.3-14, 15, and 16 summarize some of this work. ORNL indicates that for Peach Bottom during the station blackout sequence analyzed that insufficient mixtures would be available to result in hydrogen burning. While at Browns Ferry it may be possible to have burning in the reactor building.

A.4 CONTAINMENT EVENT TREE FOR CLASSES II AND IV

This section includes the following items:

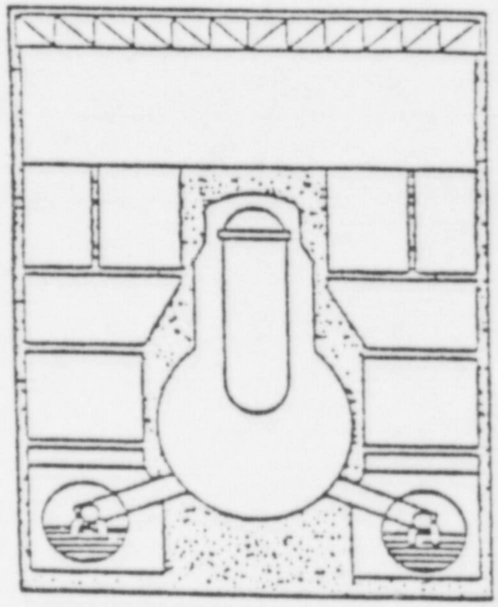
- o The containment event tree for Classes II and IV. (A.4.1)
- o The functional fault trees describing each mode of the containment event tree. (A.4.1)
- o The quantitative summary of each of the functional fault trees for each class and a qualitative description plus references to support the quantification. (A.4.2 or A.3.2)
- o A summary of the quantification of the containment end states. (A.5)

A.4.1 The Containment Event Tree for Classes II and IV

Classes II and IV are significantly different challenges to containment than those of Classes I and III. In Class II or IV challenges the containment have extremely high pressures, far above the design pressure, which results in the possibility of a containment failure prior to any significant core damage. Such a severe accident can be successfully mitigated by the BWR Mark I containment if:

THE PEACH BOTTOM UNITS HAVE A SMALLER REFUELING BAY–TO–REACTOR BUILDING RATIO

PEACH BOTTOM



BROWNS FERRY

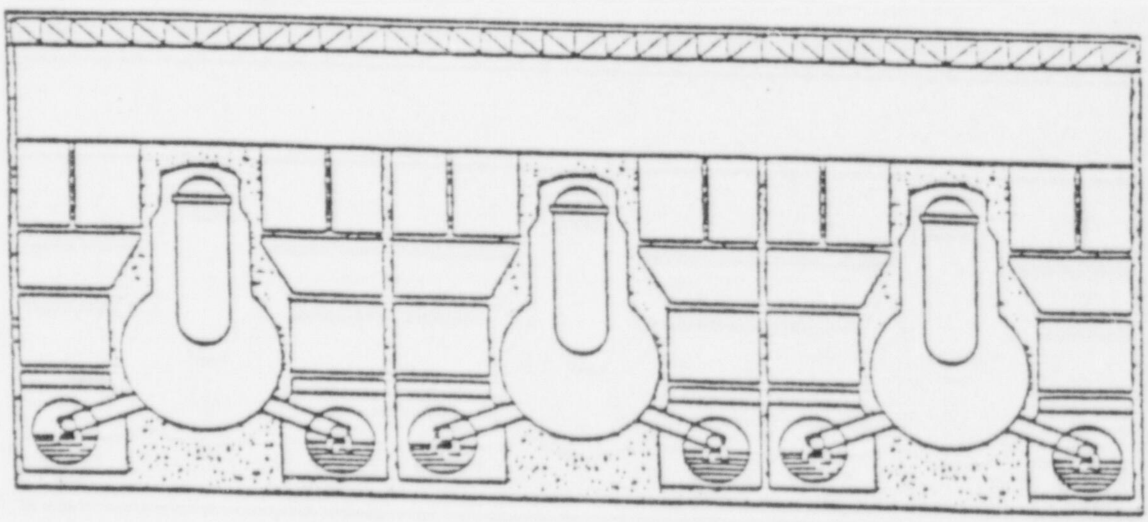


Figure A.3-14 Likelihood of Hydrogen Burn Depends Upon Ratio of Refueling Bay to Reactor Building Volume

HYDROGEN BURNS WOULD OCCUR IN THE BROWNS FERRY SECONDARY CONTAINMENT IF AN IGNITION SOURCE IS AVAILABLE

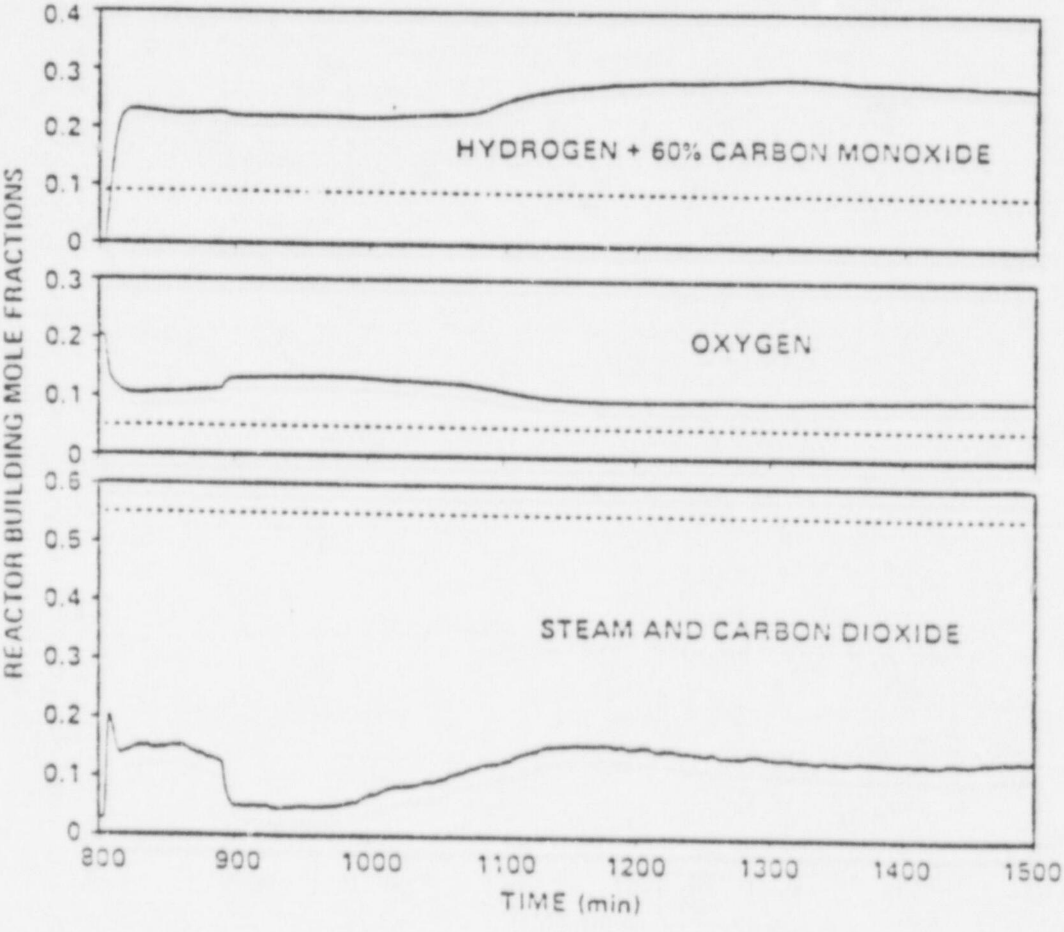


Figure A.3-15 Results of ORNL Reactor Building Hydrogen Burn Analyses

CALCULATIONS PREDICT INSUFFICIENT OXYGEN IN THE PEACH BOTTOM REACTOR BUILDING TO PERMIT BURNING DURING MOST OF THE SEQUENCE

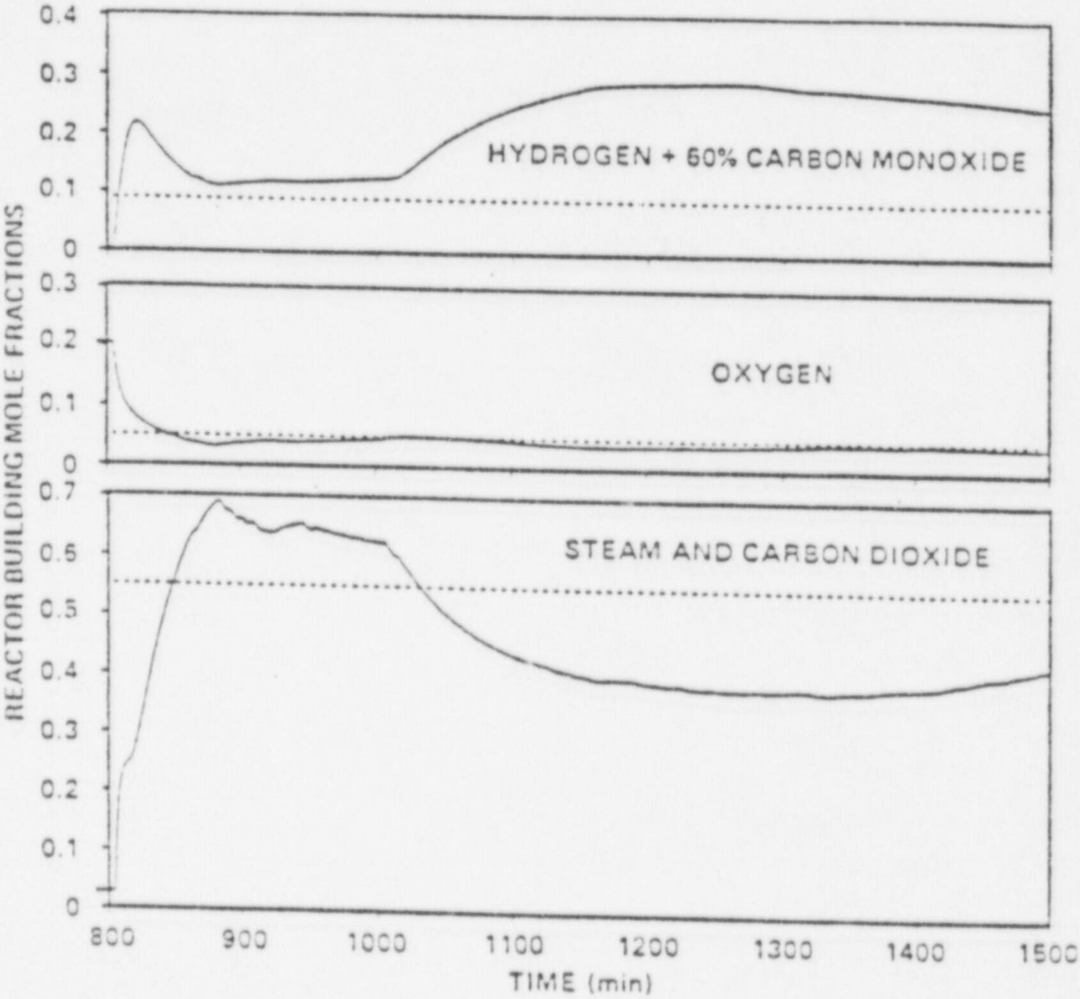


Figure A.3-16 Results of ORNL Reactor Building Hydrogen Burn Analyses

- containment heat removal can be reestablished
- containment venting capability exists and can be implemented
- core cooling can be effectively maintained regardless of whether containment failure can be prevented or not.

Figure A.4-1 is the Class II and IV CET. This CET portrays those containment mitigation features (active or passive) which can affect the integrity of containment or the radionuclide releases from containment.

The functional events which are included in the Class II and IV CET are as follows:

- o Containment pressure control (venting)
- o Containment breach size
- o Containment breach location
- o Coolant makeup likelihood
- o Containment integrity
- o Active mitigation capability: temperature control
- o Passive mitigation
- o Reactor building effectiveness

These top level functional events are described in more detail below.

The accident sequence designators are provided for every accident sequence. Adjacent to each sequence end state are the conditional probability of that state and the type of end state.

The end state types can be used to describe varying levels of containment performance depending on the purpose of the investigation. Risk evaluations would require this knowledge of the spectrum of releases which reflect both the magnitude and frequency of the radionuclide releases.

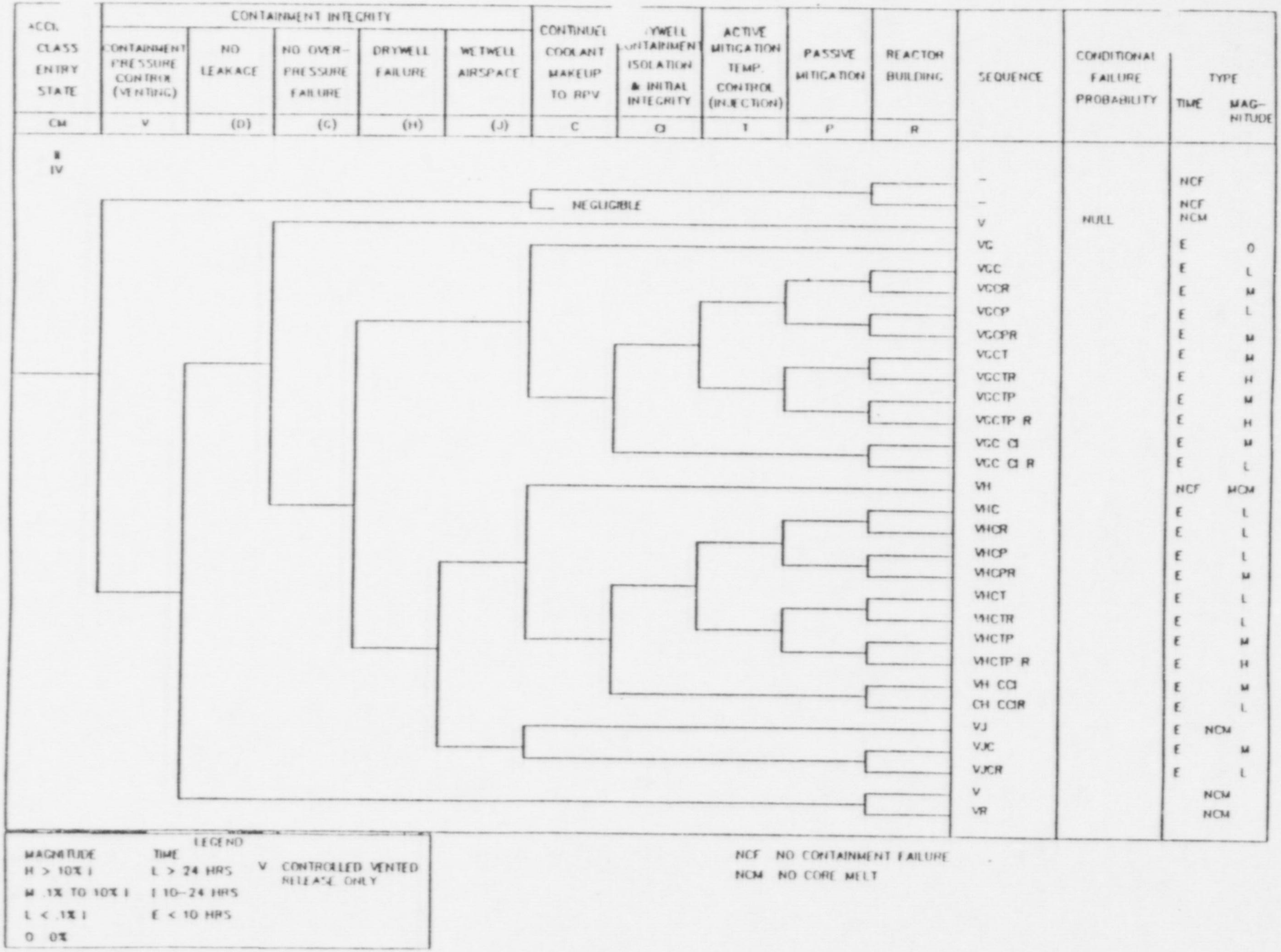


Figure A.4-1 Containment Event Tree for Classes II and IV

References

[A-1] Containment Event Trees, IDCOR Task 4.1 Technical Reports, October 1983.

[A-2] Amos CN, Griesmeyer JM, Kolaczkowski AM, Containment Event Analysis for Postulated Severe Accidents at the Peach Bottom Atomic Power Station, (SAND 86-1135) (Preliminary DRAFT for Review, May 12, 1986.

[A-3] Reactor Safety Study: An Assessment of Accident Risks on U.S. Commercial Nuclear Power Plants, U.S. Nuclear Regulatory Commission, NUREG 75/014, WASH-1400, October 1975.

[A-4] Limerick Generating Station Probabilistic Risk Assessment, Philadelphia Electric Company, Docket 50-352, 50-353, September 1982.

[A-5] Shoreham Nuclear Power Station Probabilistic Risk Assessment, Long Island Lighting Company, Docket 50-322, June 1983.

[A-6] Amos CN, Griesmeyer JM, Kolaczkowski AM, Containment Event Analysis for Postulated Severe Accidents at the Peach Bottom Atomic Power Station, (SAND 86-1135) (Preliminary DRAFT for Review, May 12, 1986.

[A-7] Engineering Judgement

[A-8] IDCOR Task 17.5, Chicago Bridge and Iron, 1986 (DRAFT).

[A-9] Griemann, et.al, Review of the Containment Structural Analysis, NUREG/CR

[A-10] Modular Accident Analysis Program (MAAP), IDCOR.

[A-11] Wooton RO, and Avci HI, 1980. MARCH (Meltdown Accident Response Characteristics) Code Description and User's Manual, NUREG/CR-1711, Battelle Columbus Laboratories, Ohio.

[A-12] Nuclear Power Plant Response to Severe Accident, IDCOR Technical Summary Report, November 1984.

[A-13] BWR Individual Plant Evaluation Methodology (IPE), Delian Corporation for IDCOR, May 1986.

[A-14] Greene GA, Perkins KR, Hodge SA; Impact of Core-Concrete Interactions in the Mark I Containment Drywell on Containment Integrity and Failure of the Drywell Liner, International symposium on Source Term Evaluation for Accident Conditions, IAEA-SM-281/36, 28 October/November 1985.

Appendix B

SUMMARY OF POTENTIAL ALTERNATIVES AND OPTIONS

This appendix summarizes some of the options and alternatives which have been considered as potential candidates for application to the five NRC issues. Cost estimates are included.

The summary is presented in the form of a series of "viewgraphs" prepared by the Nutech Engineers. These viewgraphs are presented here unchanged and in their entirety. In addition, Nutech has developed a preliminary report for Northern States Power Company (Monticello) which incorporates these conceptual designs along with their cost estimates (NSP-37-001, dated August 1986).

GROUND RULES FOR CONCEPT DEVELOPMENT

*** CORE DEBRIS BARRIERS ***

- 100% OF FUEL AND RELATED INTERNALS HAVE MELTED
- BOTTOM HEAD OF RPV HAS BEEN "OPENED"
- CORIUM FLOWS MORE OR LESS UNIFORMLY
- DO NOT RELY UPON COOLING/SOLIDIFICATION ARGUMENTS
- PREVENT DAMAGE TO CONTAINMENT PRESSURE BOUNDARY

GROUND RULES FOR CONCEPT DEVELOPMENT

*** CONTAINMENT VENTING ***

A. EARLY IN ACCIDENT

- CONTAINMENT PRESSURE HIGH AND RISING
- EARLY ENOUGH THAT FUEL CLADDING HAS NOT BEEN BREACHED
- STATION BLACK-OUT
- MULTIPLE VENTING ACTUATIONS MAY BE REQUIRED
- AVOID CONTAINMENT RUPTURE

B. LATER IN ACCIDENT

- CONTAINMENT PRESSURE HIGH BUT RATE OF PRESSURE INCREASE IS "LOW"
- FUEL HAS MELTED THROUGH RPV
- STATION BLACK-OUT
- MULTIPLE VENTING ACTUATIONS MAY BE REQUIRED
- AVOID CONTAINMENT RUPTURE

GROUD RULES FOR CONCEPT DEVELOPMENT

*** ALTERNATE WATER SOURCES FOR SPRAYS ***

- STATION BLACKOUT
- CONTAINMENT SPRAY OPERATION FOR EXTENDED DURATION
- REACTION TIME AVAILABLE TO ALLOW FOR "MAKE-SHIFT" HOOK-UP
- ACCESS TO REACTOR BUILDING NOT PRACTICAL

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MONTICELLO EVALUATION

- CORE DEBRIS BARRIERS
- VENTING
- ALTERNATE WATER SOURCES FOR SPRAY

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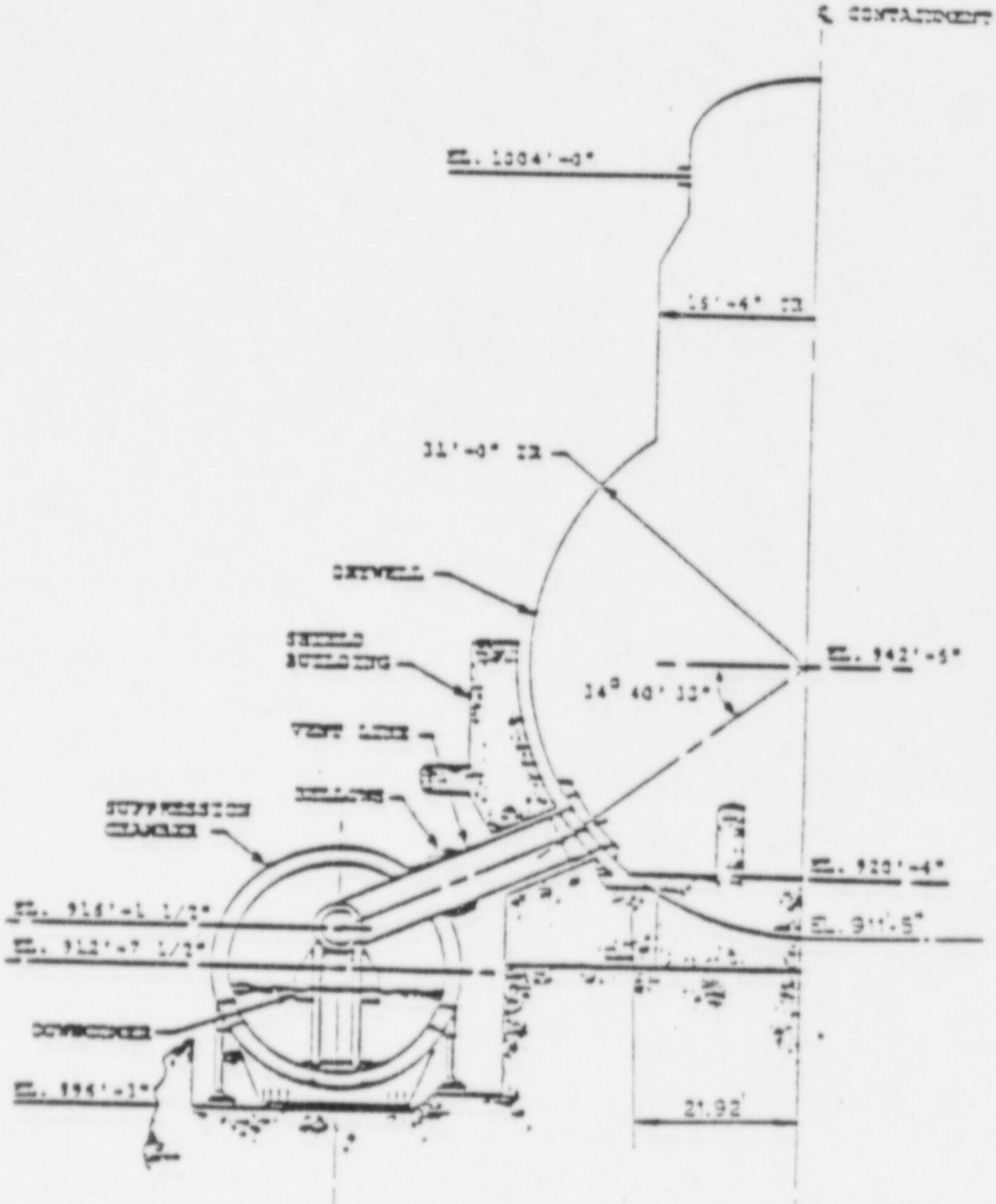

VENT TO DRYWELL FLOOR = 7 inches
SUMP VOLUME = 216 ft^3
PEDESTAL VOLUME = 205 ft^3/ft
DRYWELL FLOOR VOLUME = 1000 ft^3/ft
(OUTSIDE PEDESTAL)

ELEVATION VIEW OF CONTAINMENT

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CORE DEBRIS BARRIERS

CONCEPT 1 - CURB IN ACCESS OPENING

OBJECTIVE:

- CONTAIN MOLTEN CORE WITHIN PEDESTAL

EVALUATION:

- MOLTEN CORE DIFFICULT TO COOL
  - DEEP POOL OF MATERIAL
  - NON-ACCESSIBLE TO SPRAY
  - DRYWELL TO VENT HEIGHT LOWER THAN CURB

COST: $200,000

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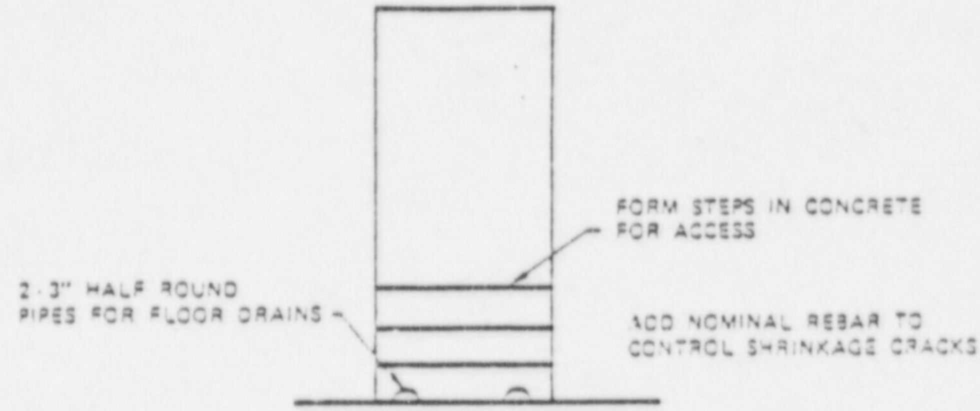
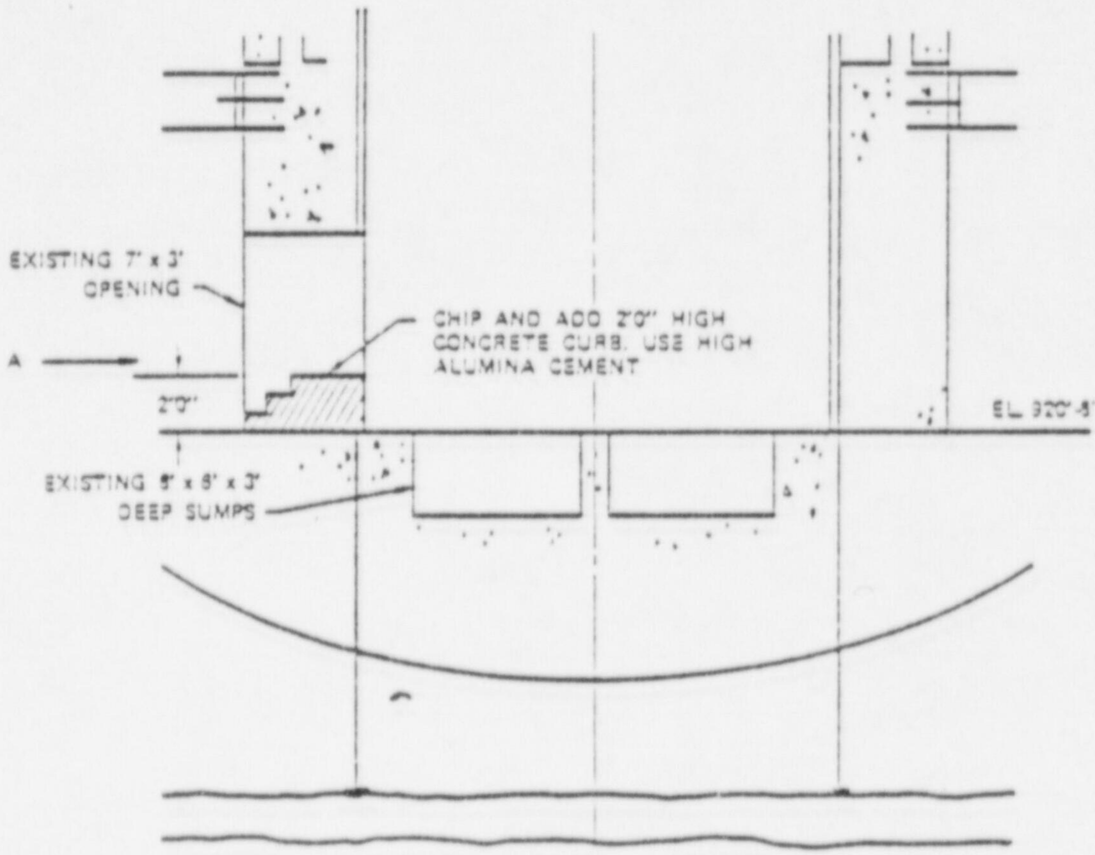


ELEVATION A



ALTERNATE 1 - ADD CURB TO ACCESS OPENING

CORE DEBRIS BARRIERS

CONCEPT 2 - EXCAVATE WITHIN PEDESTAL

EVALUATION:

- AS CONCEPT 1 EXCEPT PROVIDE MORE VOLUME BELOW FLOOR LEVEL

EVALUATION:

- REASONABLE AMOUNTS OF MOLTEN MATERIALS CAN BE CONTAINED
- COULD ADD STEEL DOOR TO CONTAIN INITIAL MATERIAL
- COOLING STILL DIFFICULT DUE TO DEEP POOL, BUT ACCUMULATED SPRAY COULD COVER TOP OF MOLTEN MATERIAL

COST: $500,000

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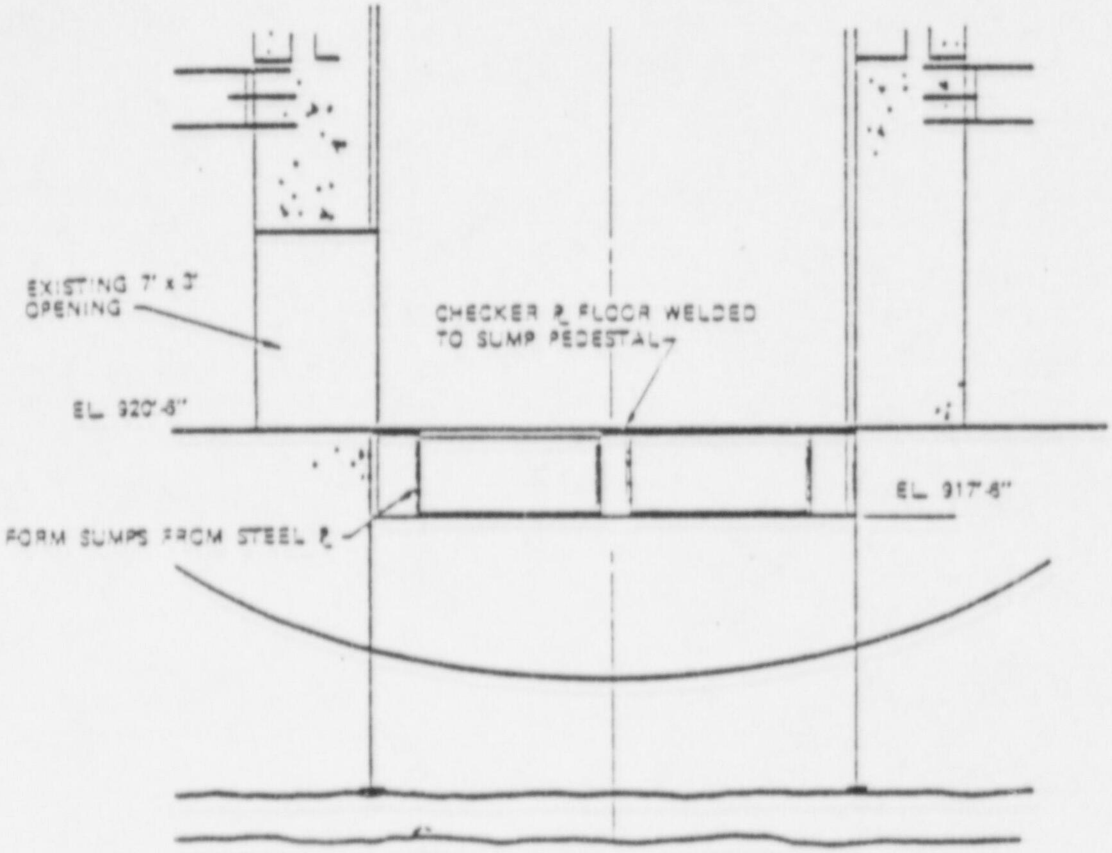

ALTERNATE 2 - EXCAVATE CONCRETE TO EL. 917'-6"
FORM SUMPS AND FLOOR FROM STEEL PLATE

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CORE DEBRIS BARRIERS

CONCEPT 5 - REFRACTORY FILLED CAVITY WITHIN PEDESTAL

OBJECTIVE:

- CONTAIN MOLTEN MATERIALS AND PROVIDE ADDITIONAL PROTECTION TO CONCRETE

EVALUATION:

- REFRACTORY MATERIALS WOULD PROVIDE MORE TIME PRIOR TO SIGNIFICANT CONCRETE DAMAGE

- COOLING OF MOLTEN POOL DIFFICULT DUE TO DEPTH

COST: $2 - $5 MILLION

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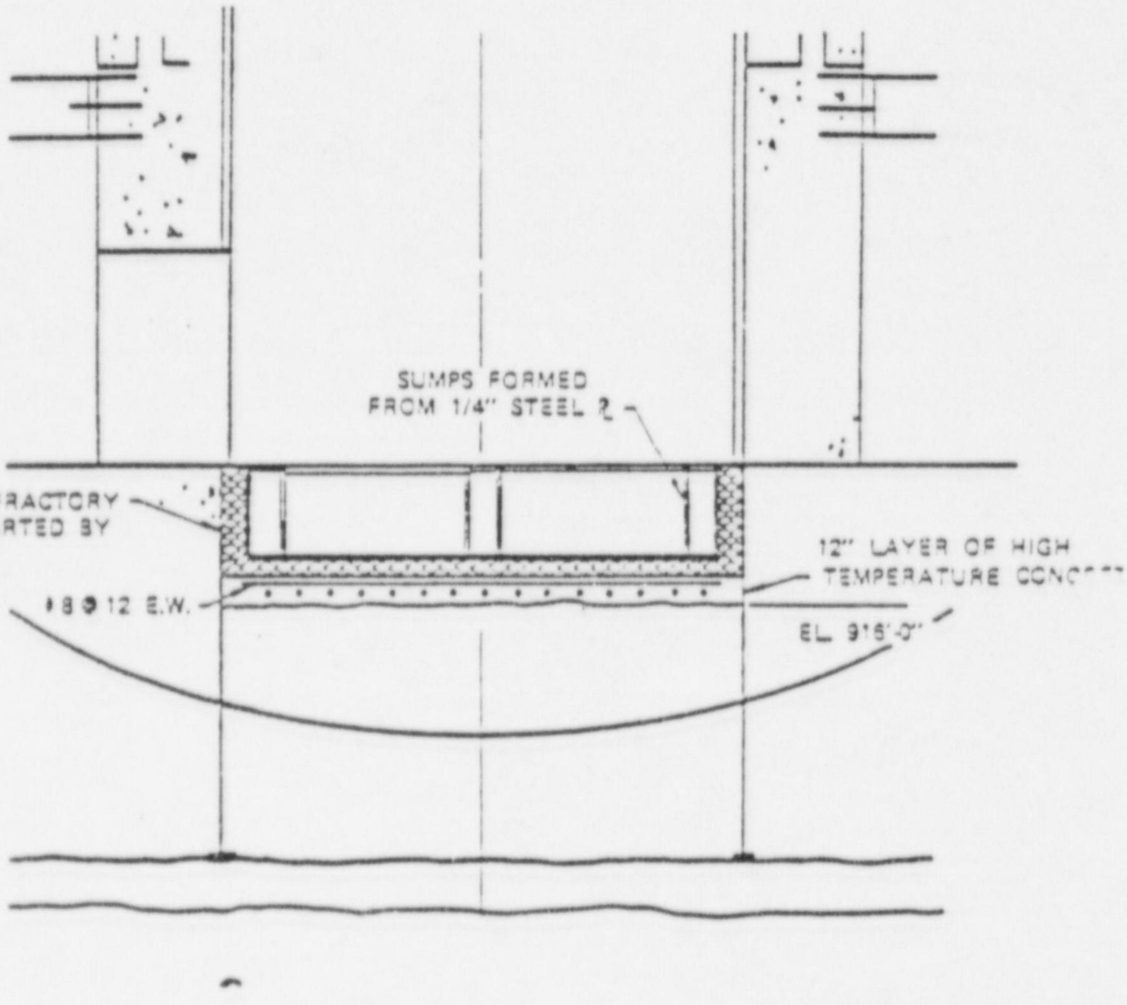

ALTERNATE 3 - EXCAVATE CAVITY, ADD HIGH TEMPERATURE CONCRETE AND REFRACTORY INSULATION

CORE DEBRIS BARRIERS

CONCEPT 4 - DRYWELL CURB

OBJECTIVE:

- PREVENT MOLTEN CORE FROM DAMAGING DRYWELL SHELL

DESIGN CONSIDERATIONS:

- PROVIDE CURB TO SUFFICIENT DEPTH TO CONTAIN CORE MELT AND SOME SPRAY FOR SURFACE COOLING
- ASSURE THAT DRYWELL LEAKAGE CAN GET TO SUMP
- PREVENT MOLTEN CORE FROM ENTERING VENT

COST: $1 - $2 MILLION

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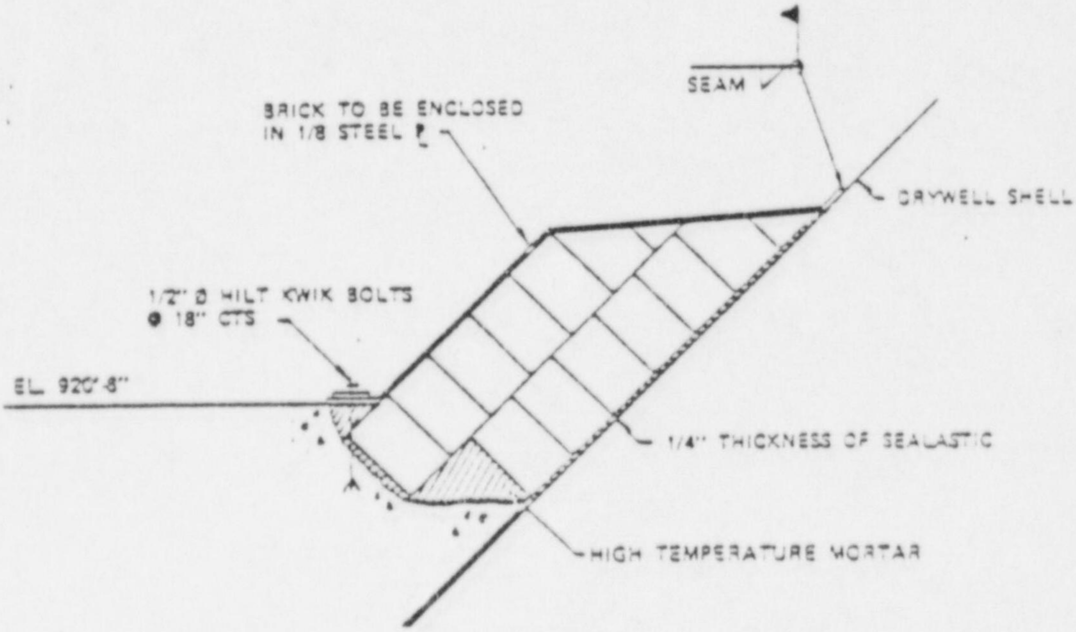
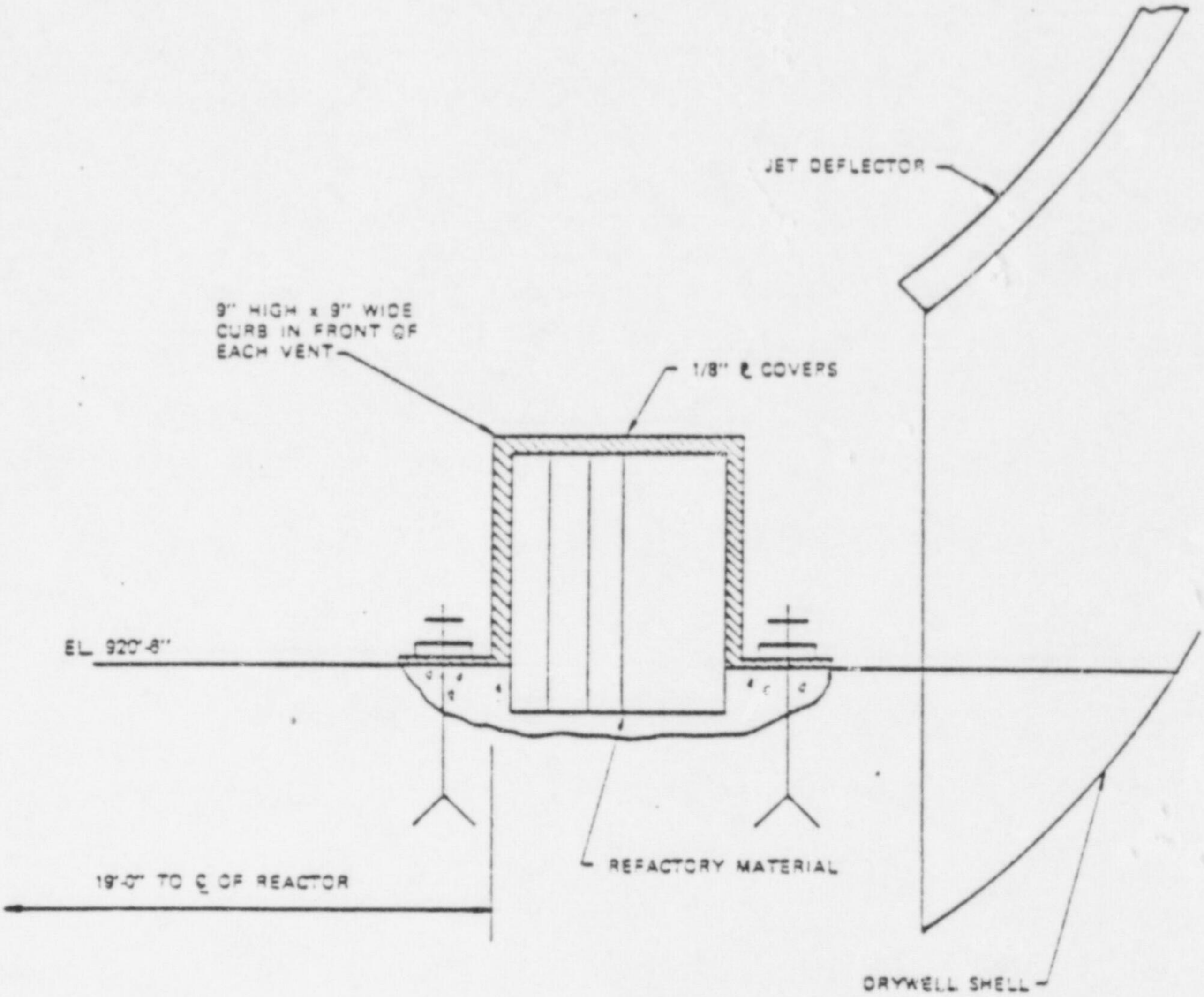


DRYWELL CURB TYPICAL DETAIL



TYPICAL DETAIL OF DRYWELL
CURB AT DRYWELL VENT

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CORE DEBRIS BARRIERS

CONCEPT 5 - BARRIERS ON DRYWELL FLOOR

OBJECTIVE:

- SPREAD MOLTEN CORE MATERIALS ON DRYWELL FLOOR TO ASSURE COOLING

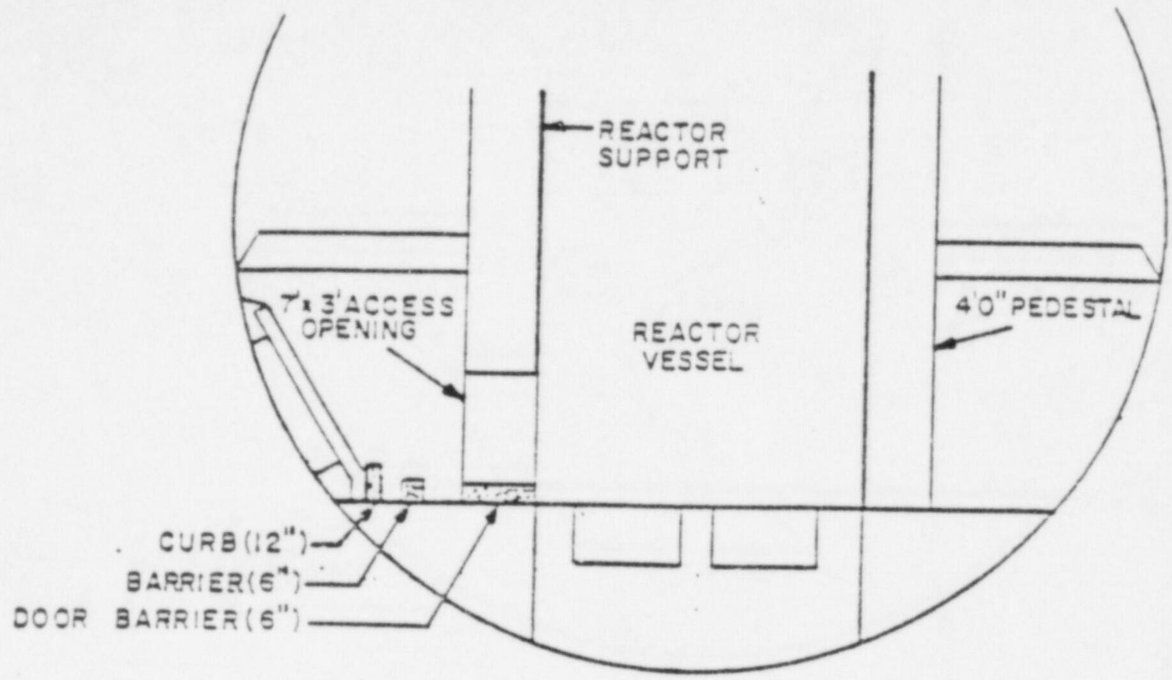
CONSIDERATIONS:

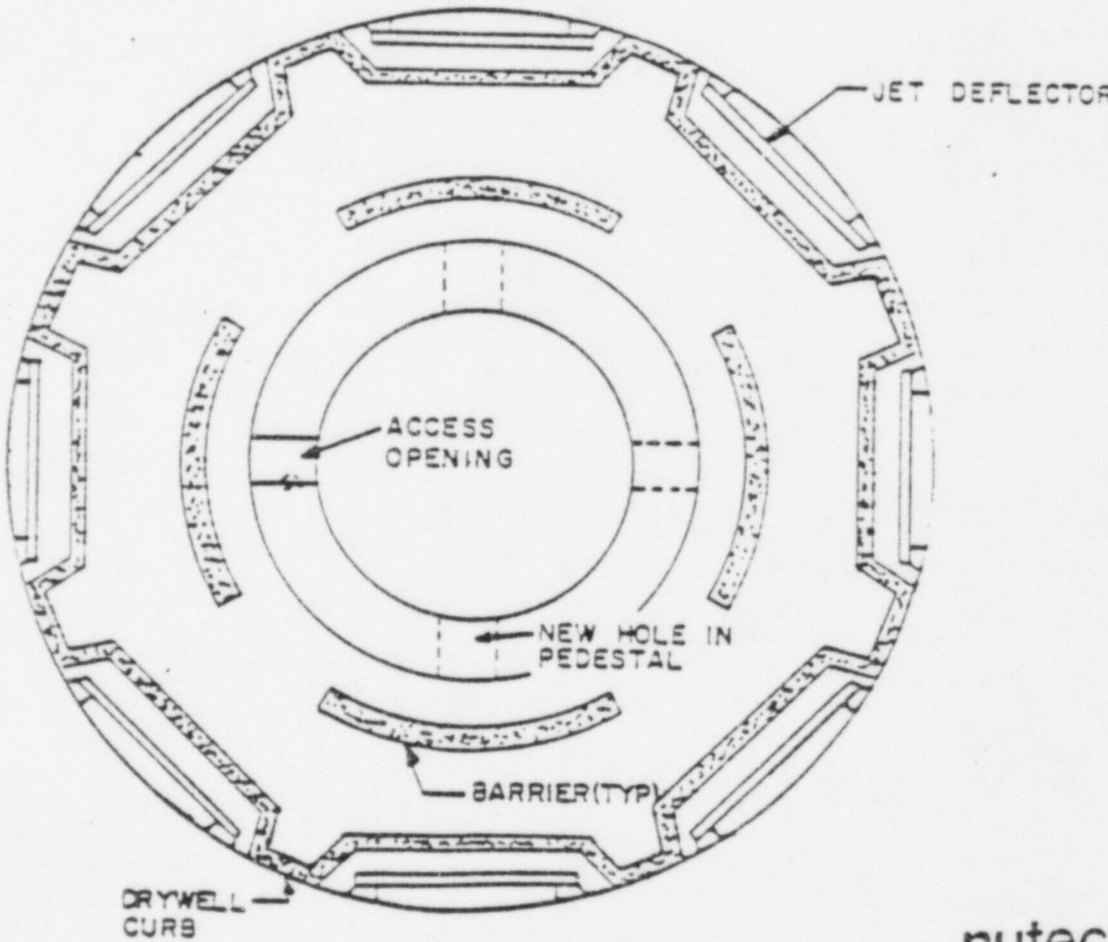
- NOMINAL (6 TO 12-INCHES HIGH) INSTALLED TO SPREAD FLOW OF CORE MATERIALS

- USE IN CONJUNCTION WITH DRYWELL CURB AND/OR PEDESTAL EXCAVATION

- ADDITION HOLES IN PEDESTAL PROMOTE A MORE UNIFORM DISTRIBUTION ON FLOOR

COST: $2 - $4 MILLION (INCLUDING DRYWELL CURB)





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ALTERNATE 5—BARRIERS ON DRYWELL FLOOR

CORE DEBRIS BARRIERS

OTHER CONSIDERATIONS

- NOT CONSIDERED FEASIBLE TO PROTECT DRYWELL-TO-WETWELL VENT PIPES IF CONTACTED BY MOLTEN CORE MATERIALS

- DRYWELL VOLUME CHANGES AND ADDITIONAL DRYWELL WATER HOLD-UP WILL AFFECT LOCA ANALYSIS

CONTAINMENT VENTING

CONCEPT 1 - VENT TO SECONDARY CONTAINMENT

OBJECTIVE:

- ASSURE PRIMARY CONTAINMENT INTEGRITY

EVALUATION:

- FOR ATWS-TYPE SCENARIO, RELEASE INSIDE SECONDARY CONTAINMENT JUDGED TO BE UNACCEPTABLE SINCE RECOVERY PRIOR TO SIGNIFICANT PLANT DAMAGE IS LIKELY
- FOR CORE-MELT SITUATION, RELEASE SHOULD BE FILTERED BY STANDBY GAS TREATMENT SYSTEM TO MAXIMUM EXTENT

COST: NOT EVALUATED

CONTAINMENT VENTING

CONCEPT 2 - HARD PIPE STANDBY GAS TREATMENT SYSTEM

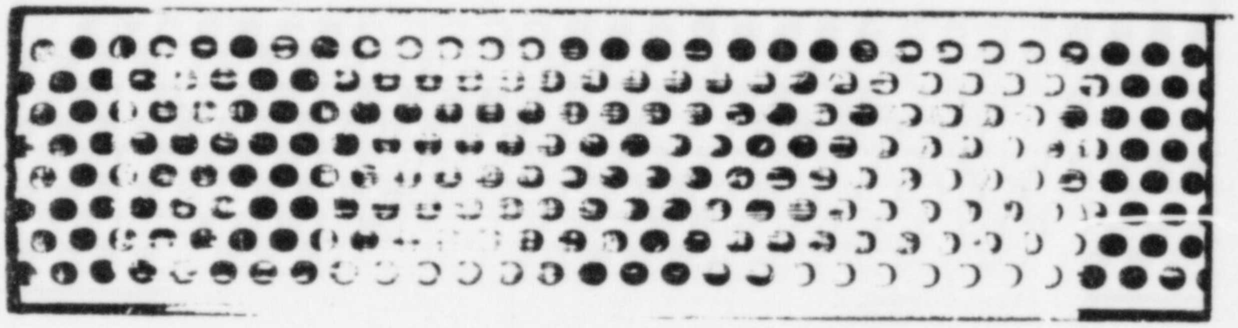
OBJECTIVE:

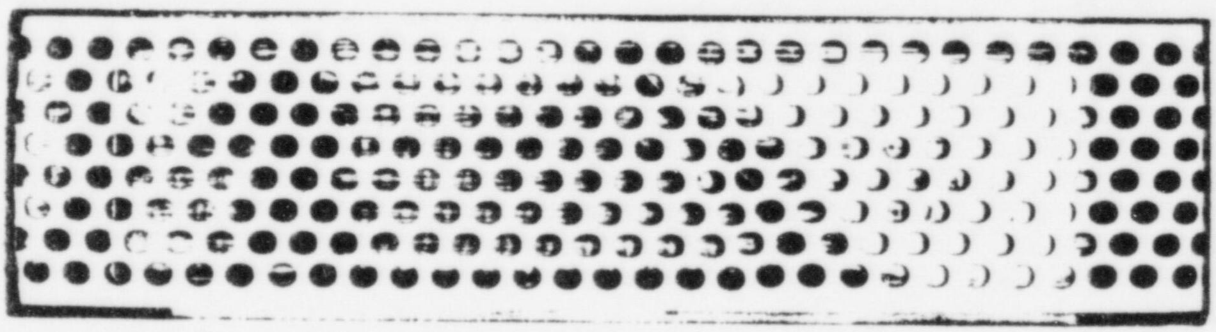
- ASSURE CONTAINMENT INTEGRITY AND PROVIDE FILTERED RELEASE

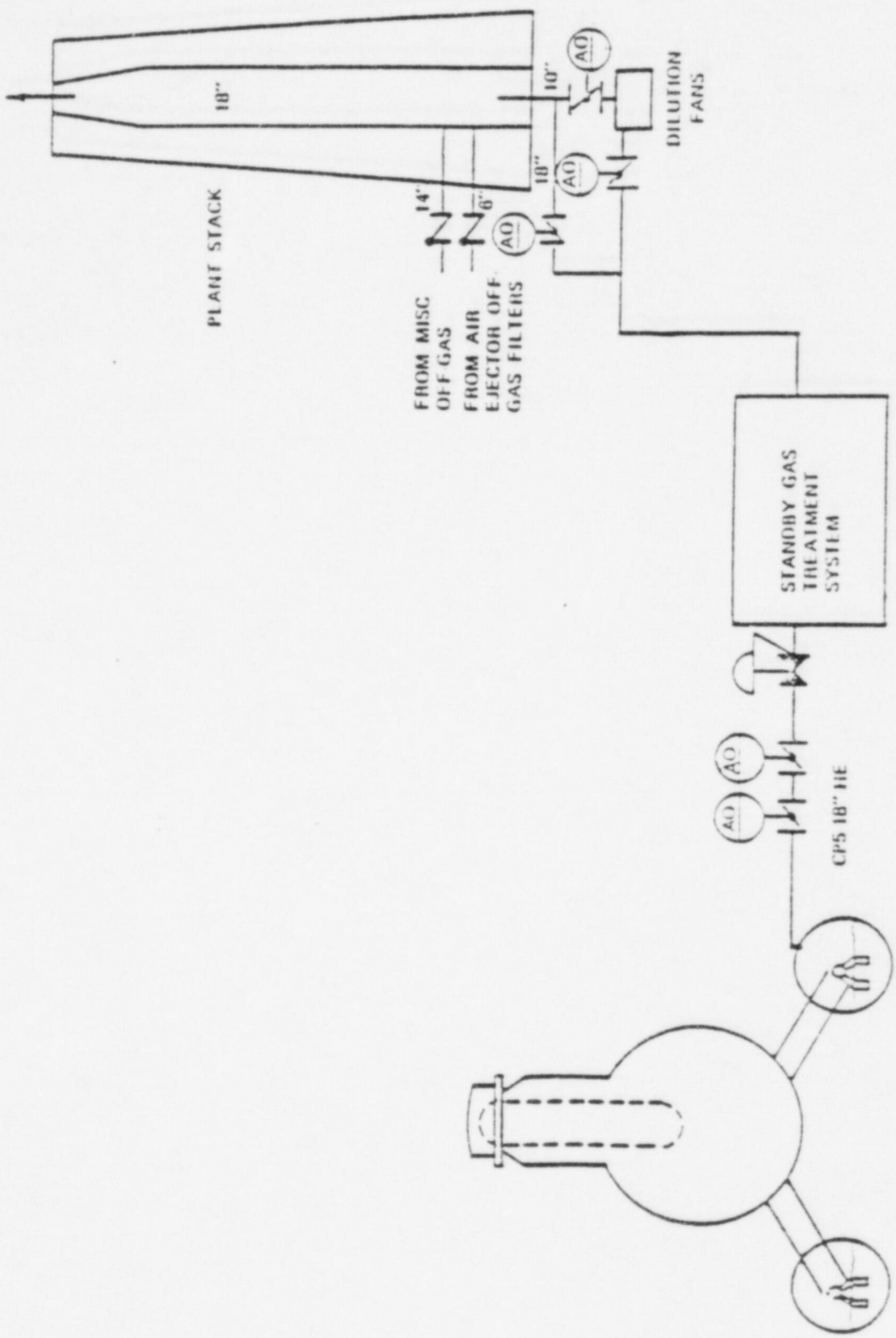
EVALUATION:

- MODIFIED SYSTEM MUST REMAIN SAFETY-RELATED
- CURRENT DUCTING WOULD HAVE TO BE REPLACED WITH PIPE
- EQUIPMENT UPGRADE FOR HIGHER PRESSURES REQUIRED
- EQUIPMENT PROBABLY NOT CAPABLE OF FULL FLOW RATE OF 18-INCH VENT AND PURGE VALVES

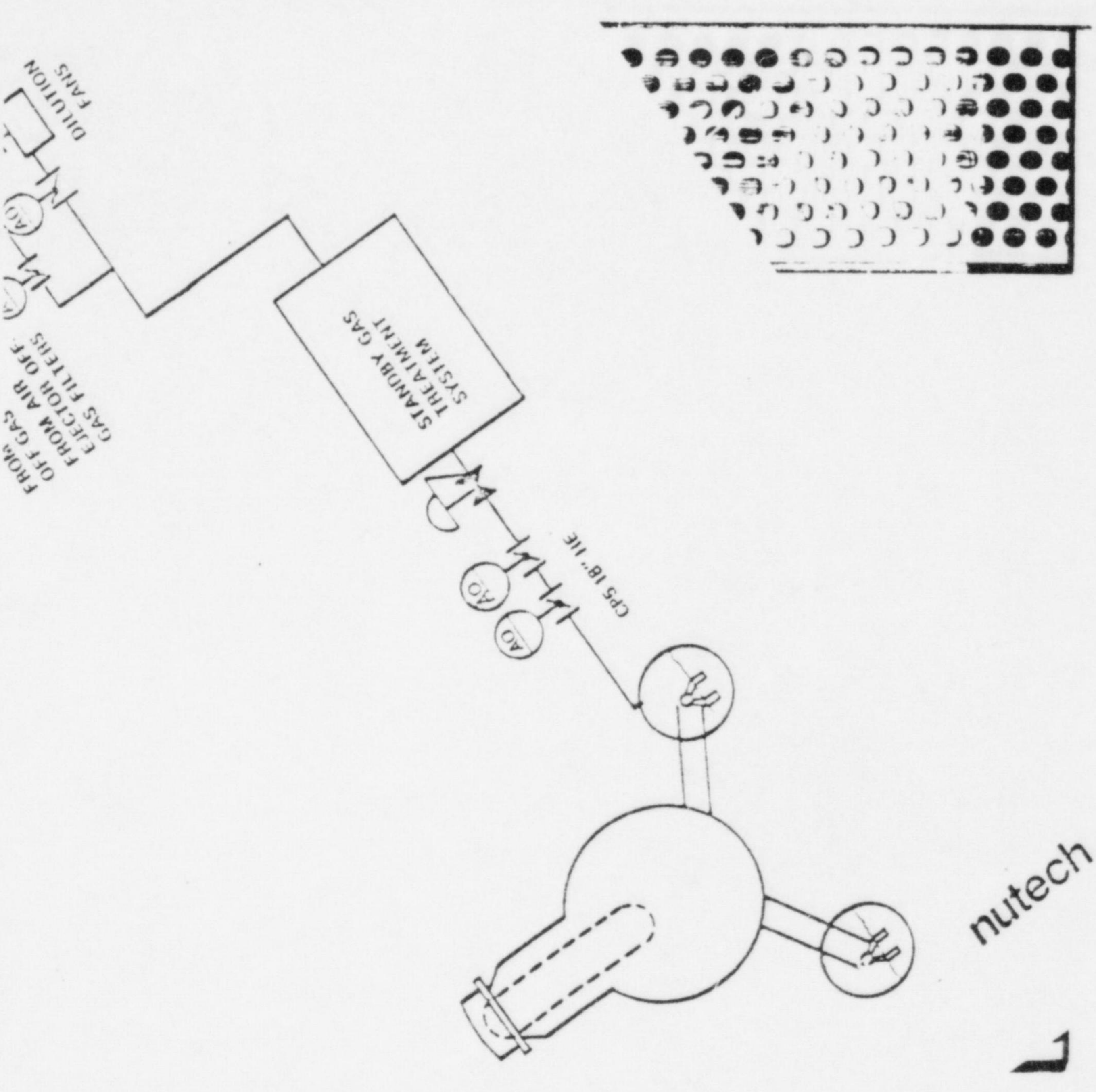
COST: $1 - $2 MILLION






PLANT STACK
18"
10"
18"
14"
6"
AO
FROM MISC OFF GAS
FROM AIR EJECTOR OFF GAS FILTERS
DILUTION FANS
STANDBY GAS TREATMENT SYSTEM
CPS 18" HE

nutech



FROM OFF GAS
FROM AIR EJECTOR OFF GAS FILTERS
DILUTION FANS
AO
AO
STANDBY GAS TREATMENT SYSTEM
AO
AO
CPS 18" IIE
nutech

CONTAINMENT VENTING

CONCEPT 5 - HARD PIPE AROUND STANDBY GAS TREATMENT SYSTEM

OBJECTIVE:

- ASSURE CONTAINMENT INTEGRITY AND PROVIDE FILTERED RELEASE FOR POST-CORE-MELT VENTING

EVALUATION:

- FULL FLOW VENTING UP STACK PROVIDES MAXIMUM DILUTION
- REDUCED FLOW VENTING THROUGH SBGTS POSSIBLE FOR ADDITIONAL FILTERING
- RELATIVELY SHORT VENT RUN REQUIRED FROM TORUS TO STACK PIPING OUTLET

COST: $1 - $2 MILLION

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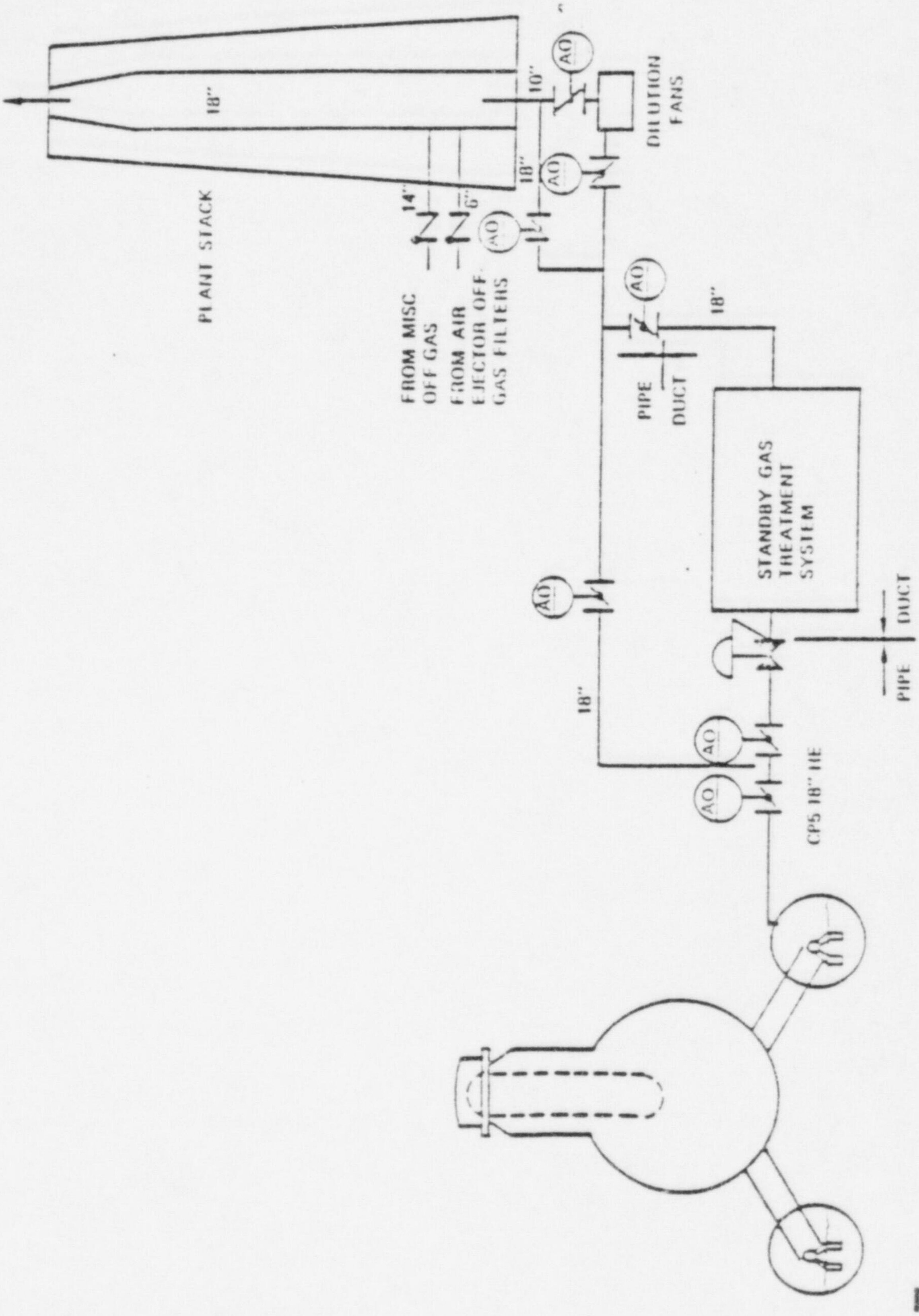


PLANT STACK
18"
10"
14"
6"
FROM MISC OFF GAS
FROM AIR EJECTOR OFF GAS FILTERS
AO
DILUTION FANS
18"
PIPE
DUCT
STANDBY GAS TREATMENT SYSTEM
18"
CPS 18" HE

nutech

CONTAINMENT VENTING

CONCEPT 4 - ALTERNATE VENT AT TOP OF REACTOR BUILDING

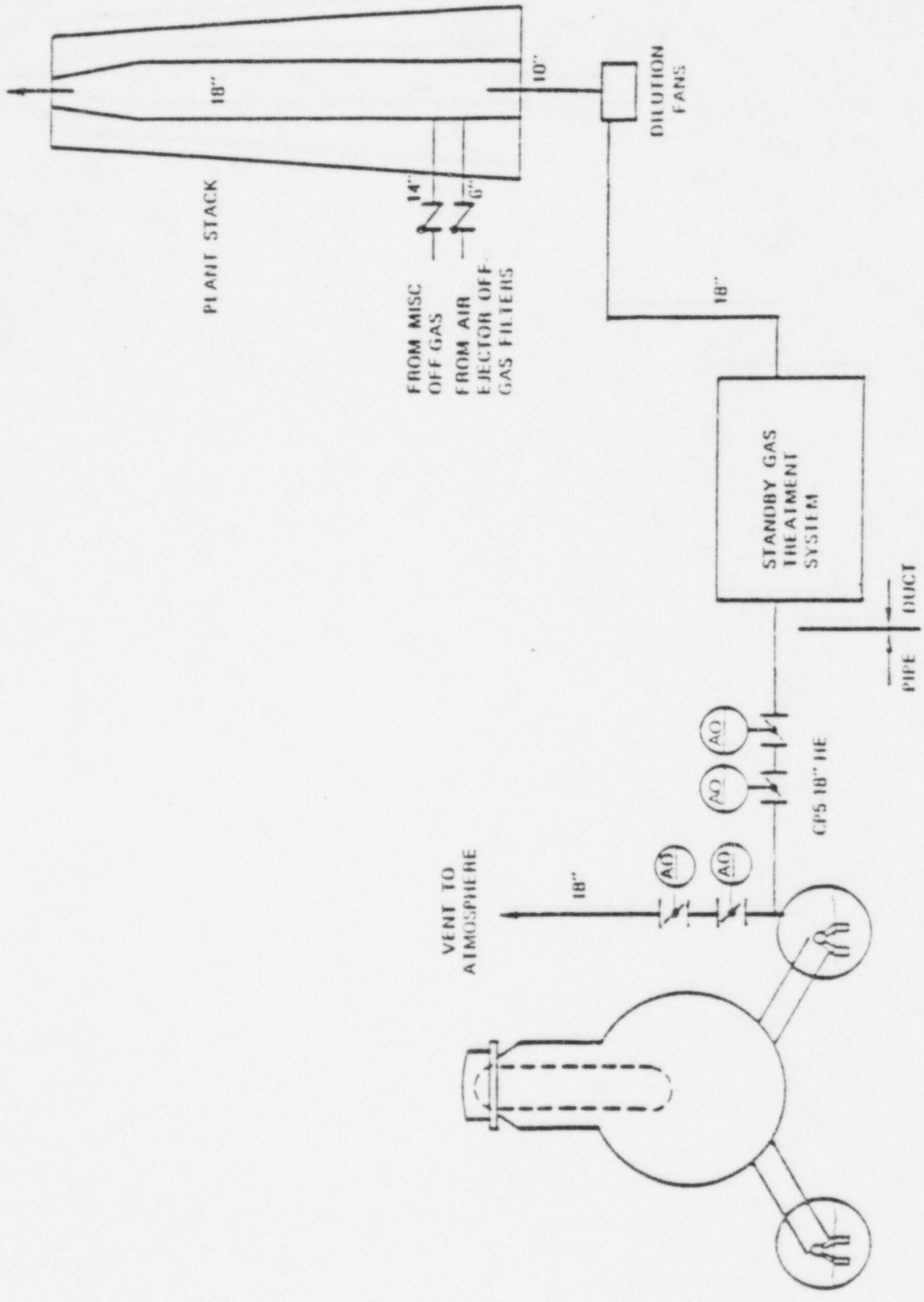
OBJECTIVE:

- ASSURE CONTAINMENT INTEGRITY AND PROVIDE ALTERNATE FULL FLOW VENT CAPABILITY

EVALUATION:

- HIGHER VENTING FLOW RATES ATTAINABLE, IF REQUIRED
- ATMOSPHERIC DILUTION NOT AS GOOD AS VENTING THROUGH STACK

COST: $1 - $2 MILLION


PLANT STACK
18"
10"
DILUTION FANS
14"
6"
FROM MISC OFF GAS
FROM AIR EJECTOR OFF GAS FILTERS
18"
STANDBY GAS TREATMENT SYSTEM
PIPE
DUCT
AO
AO
AO
AO
CPS 18" HE
VENT TO ATMOSPHERE
18"

nutech

CONTAINMENT VENTING

OTHER CONSIDERATIONS

- CONTAINMENT VENTING POTENTIALLY PURGES NITROGEN FROM CONTAINMENT

- FOLLOWING COOL-DOWN AND VACUUM RELIEF, AIR MAY BE DRAWN INTO CONTAINMENT

  - COULD LEAD TO COMBUSTIBLE MIXTURE

- VENTING ACTION PRESSURES AND POST-ACCIDENT PROCEDURES COULD PROBABLY ELIMINATE CONCERN

CONTAINMENT SPRAY

CONCEPT 1 - CONNECT TO NEAREST HOSE STATION IN REACTOR BUILDING

OBJECTIVE:

- PROVIDE ALTERNATE SOURCE OF WATER FOR CONTAINMENT PRESSURE CONTROL, COOLING, AND AEROSOL SCRUBBING

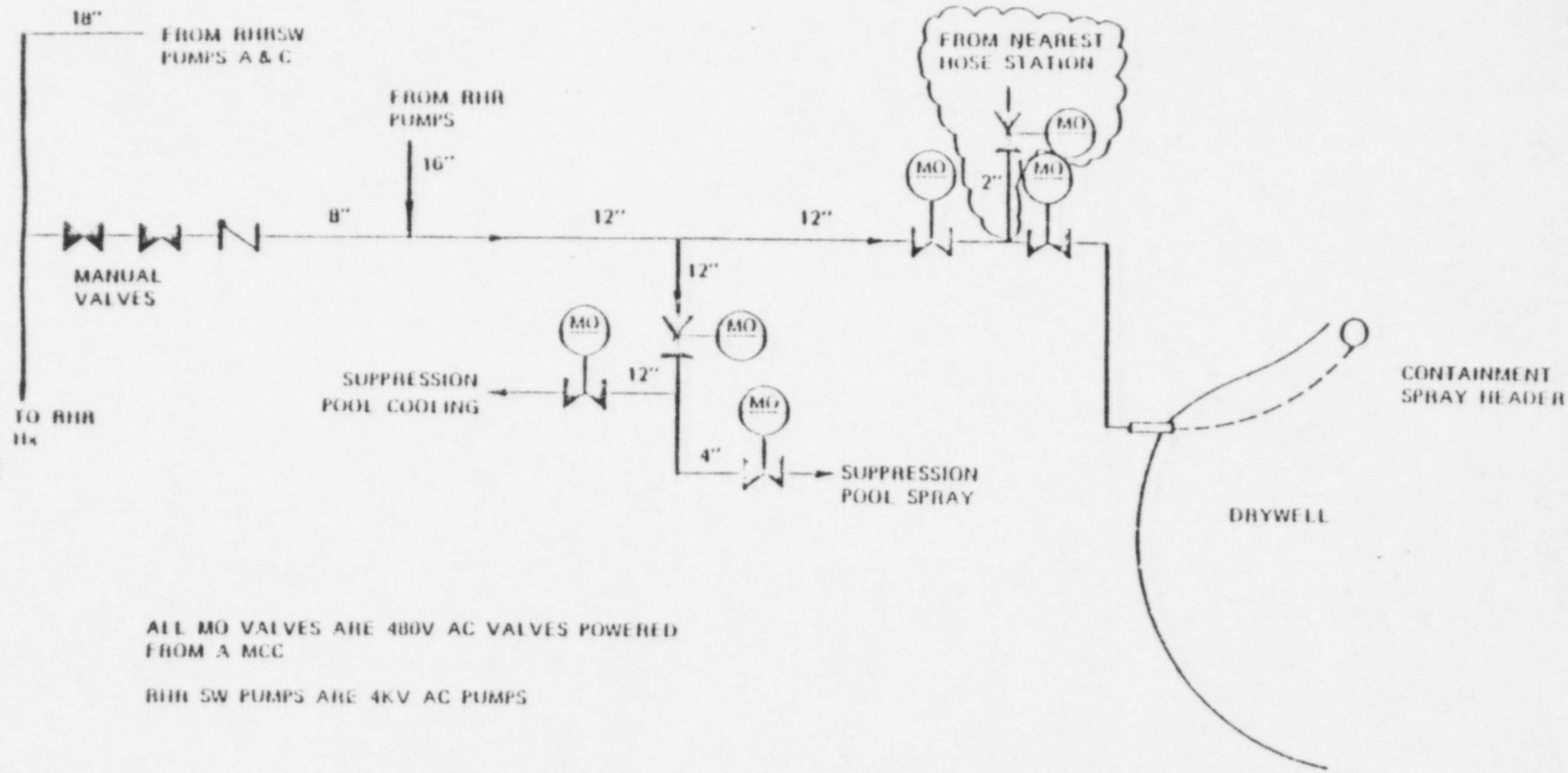
EVALUATION:

- ACCESS TO REACTOR BUILDING FOLLOWING CORE MELT ACCIDENT PROBABLY NOT POSSIBLE BECAUSE OF RADIATION LEVELS, SO FIRE HOSE CONNECTION UNACCEPTABLE

- AMOUNT OF WATER PROBABLY NOT SUFFICIENT - CLOSEST MAIN IS 2-INCH

- MAJOR COSTS ASSOCIATED WITH STATION BLACKOUT CONDITION

COST: $500,000



ALL MO VALVES ARE 480V AC VALVES POWERED FROM A MCC

RHR SW PUMPS ARE 4KV AC PUMPS

CONTAINMENT SPRAY

CONCEPT 2 - CROSS-CONNECT OUTLET OF DIESEL FIRE PUMP TO CROSS-TIE OF RHR SERVICE WATER SYSTEM TO RHR SYSTEM

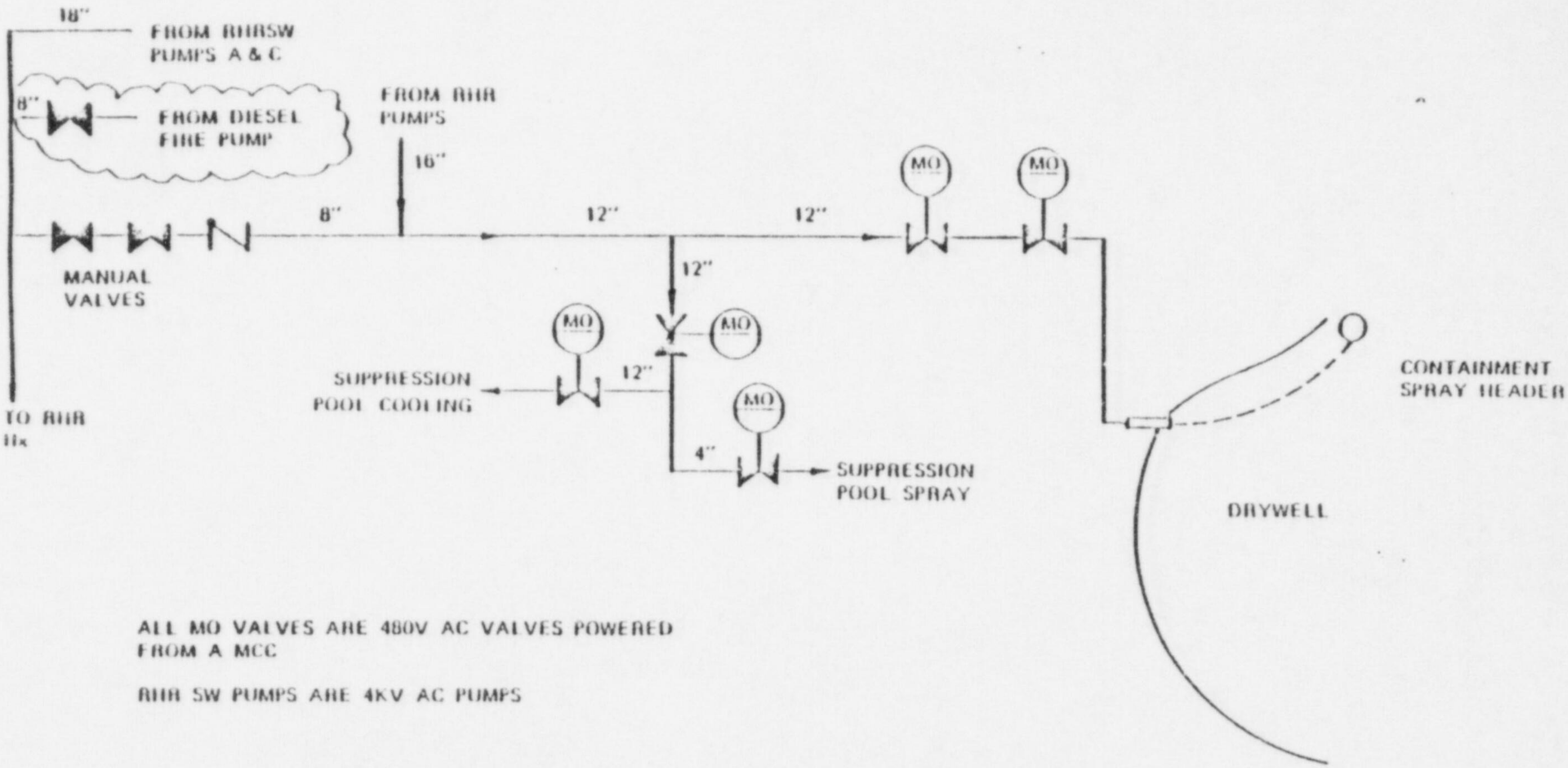
OBJECTIVE:

- PROVIDE FULL FLOW FROM FIRE PUMP TO DRYWELL SPRAY HEADER

EVALUATION:

- MANUAL ACTUATION OF VALVE (OR FIRE HOSE INTERCONNECTION) POSSIBLE SINCE LOCATED IN INTAKE STRUCTURE BUILDING
- APPROXIMATELY 1200 GPM FLOW AVAILABLE
- ADEQUACY OF SPRAY NOZZLES UNKNOWN
- ADEQUACY OF PUMP TO DELIVER FLOW AT ELEVATED PRESSURES REQUIRES FURTHER EVALUATION

COST: $500,000 - $750,000



ALL MO VALVES ARE 480V AC VALVES POWERED
FROM A MCC

RHR SW PUMPS ARE 4KV AC PUMPS

CONTAINMENT SPRAY

CONCEPT 3 - PROVIDE NEW INDEPENDENT DIESEL GENERATOR TO OPERATE RHR SERVICE WATER PUMPS

OBJECTIVE:

- PROVIDE MAXIMUM FLOW TO FLOOD DRYWELL AND WETWELL THROUGH SPRAY HEADER

EVALUATION:

- INTERCONNECTIONS EXIST TO FLOOD CONTAINMENT THROUGH SPRAYS AND SUPPRESSION POOL COOLING DISCHARGE LINE

- DC OPERATED VALVES REQUIRED

- CREDIT FOR DIESEL MAY NOT BE GRANTED UNDER STATION BLACKOUT RULES

COST: $3 - $6 MILLION

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CONTAINMENT SPRAY

CONCEPT 4 - NEW DIESEL GENERATOR AND PUMP CAPABLE OF HIGH PRESSURE INJECTION

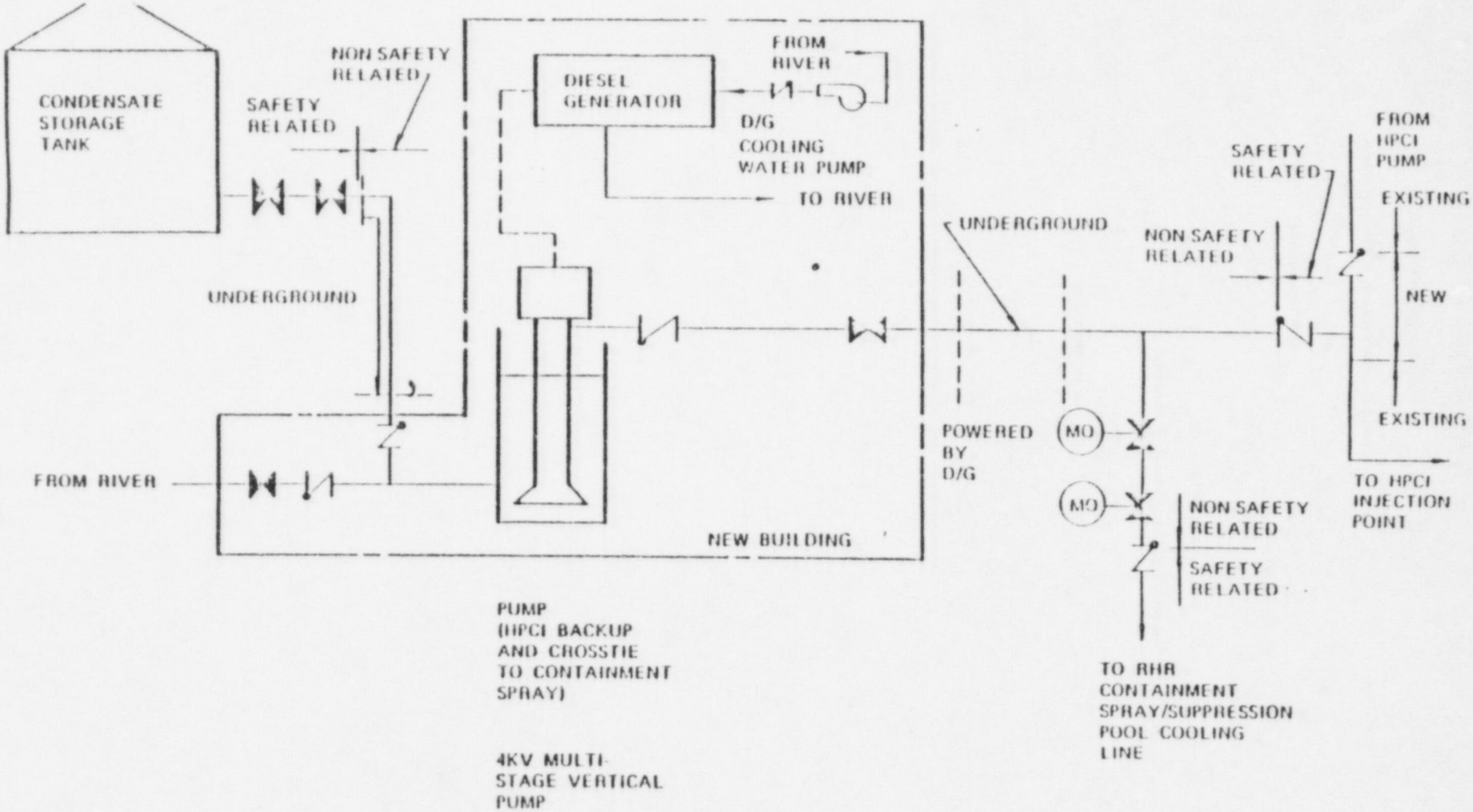
OBJECTIVE:

- PROVIDE ALTERNATE SOURCE OF WATER FOR SPRAYS AND/OR HIGH PRESSURE COOLANT INJECTION

EVALUATION:

- GOES BEYOND REQUIREMENTS FOR CONTAINMENT SPRAY
- PROVIDES ALTERNATE SOURCE OF HIGH PRESSURE COOLANT FOR MAINTAINING RPV WATER LEVEL

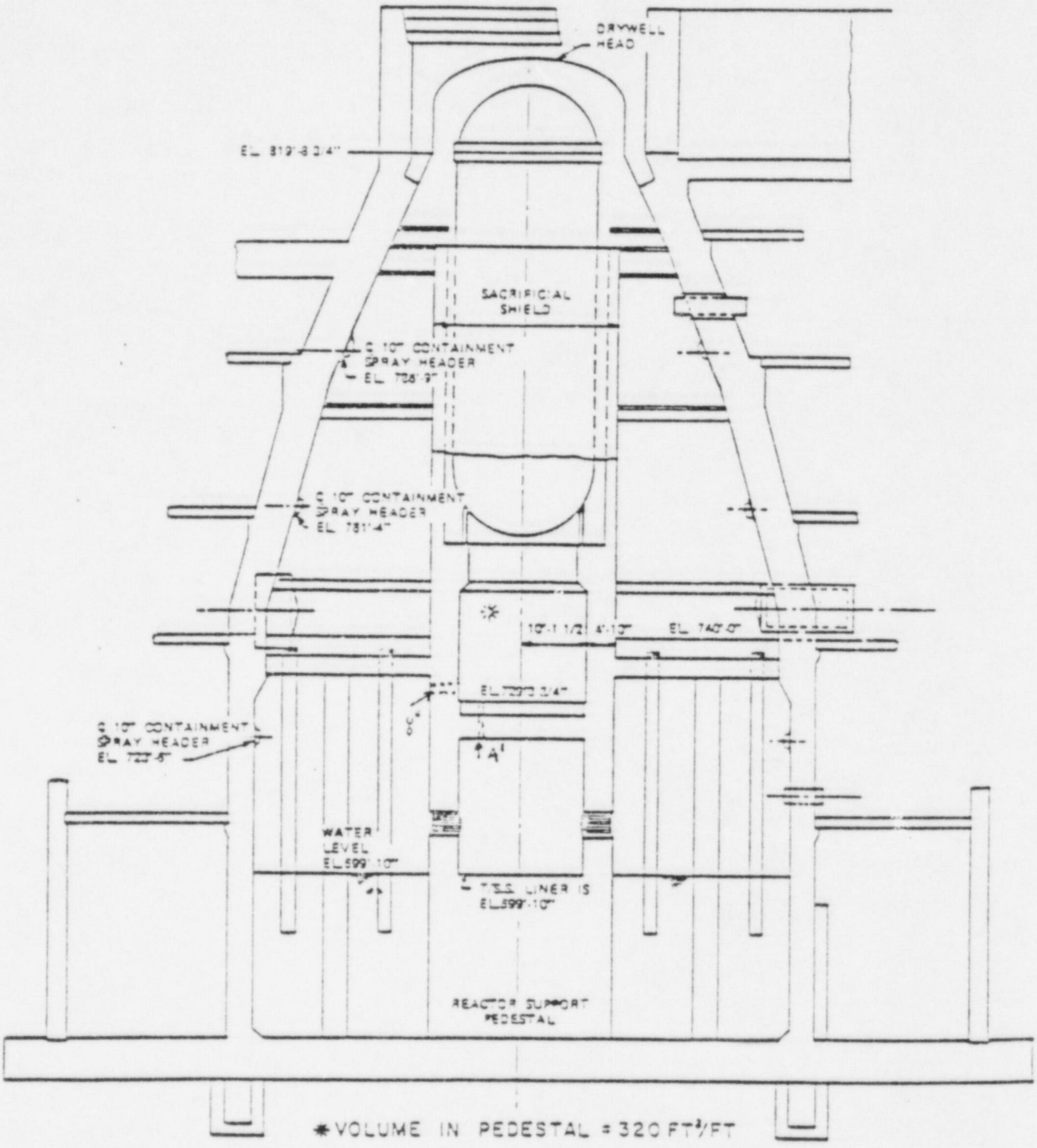
COST: $20 - $25 MILLION



CONDENSATE STORAGE TANK
SAFETY RELATED
NON SAFETY RELATED
UNDERGROUND
DIESEL GENERATOR
FROM RIVER
D/G COOLING WATER PUMP
TO RIVER
FROM RIVER
NEW BUILDING
PUMP (HPCI BACKUP AND CROSSTIE TO CONTAINMENT SPRAY)
4KV MULTI-STAGE VERTICAL PUMP
UNDERGROUND
POWERED BY D/G
MO
MO
NON SAFETY RELATED
SAFETY RELATED
TO RHR CONTAINMENT SPRAY/SUPPRESSION POOL COOLING LINE
NON SAFETY RELATED
SAFETY RELATED
FROM HPCI PUMP
EXISTING
NEW
EXISTING
TO HPCI INJECTION POINT

LASALLE CONTAINMENT EVALUATION

- CORE DEBRIS PATHS
- CONTAINMENT VENTING
- ALTERNATE SPRAY SYSTEM WATER SUPPLY



*VOLUME IN PEDESTAL = 320 FT³/FT

LA SALLE CONTAINMENT

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CORE DEBRIS PATH EVALUATION

OBJECTIVE:

- ASSURE THAT CORE DEBRIS DOES NOT DIRECTLY DAMAGE CONTAINMENT PRESSURE BOUNDARY

EVALUATION:

- EXISTING AREA BELOW SUMP MORE THAN ADEQUATE TO CONTAIN POTENTIAL CORE MELT
- PROBABLE EARLY FAILURE OF DRYWELL-TO-WETWELL FLOOR BENEATH RPV
- SECONDARY HOLD-UP OF MOLTEN CORE WITHIN PEDESTAL, BUT IN WETWELL AIRSPACE
- SUPPRESSION POOL FILTERING ABILITY LOST
- ANOTHER CONCERN IS LOSS OF PEDESTAL STRUCTURAL SUPPORT FOR DRYWELL EQUIPMENT AND STRUCTURES

CORE DEBRIS PATH EVALUATION

CONCEPT 1 - RAISE WETWELL PEDESTAL CONCRETE

OBJECTIVE:

- ASSURE QUENCHING OF CORE DEBRIS AFTER FAILURE OF DRYWELL/WETWELL FLOOR UNDER RPV

EVALUATION:

- RAISED CONCRETE PEDESTAL ASSURES THAT MOLTEN CORE FALLS INTO SUPPRESSION POOL

COST: $500,000 - $1,000,000

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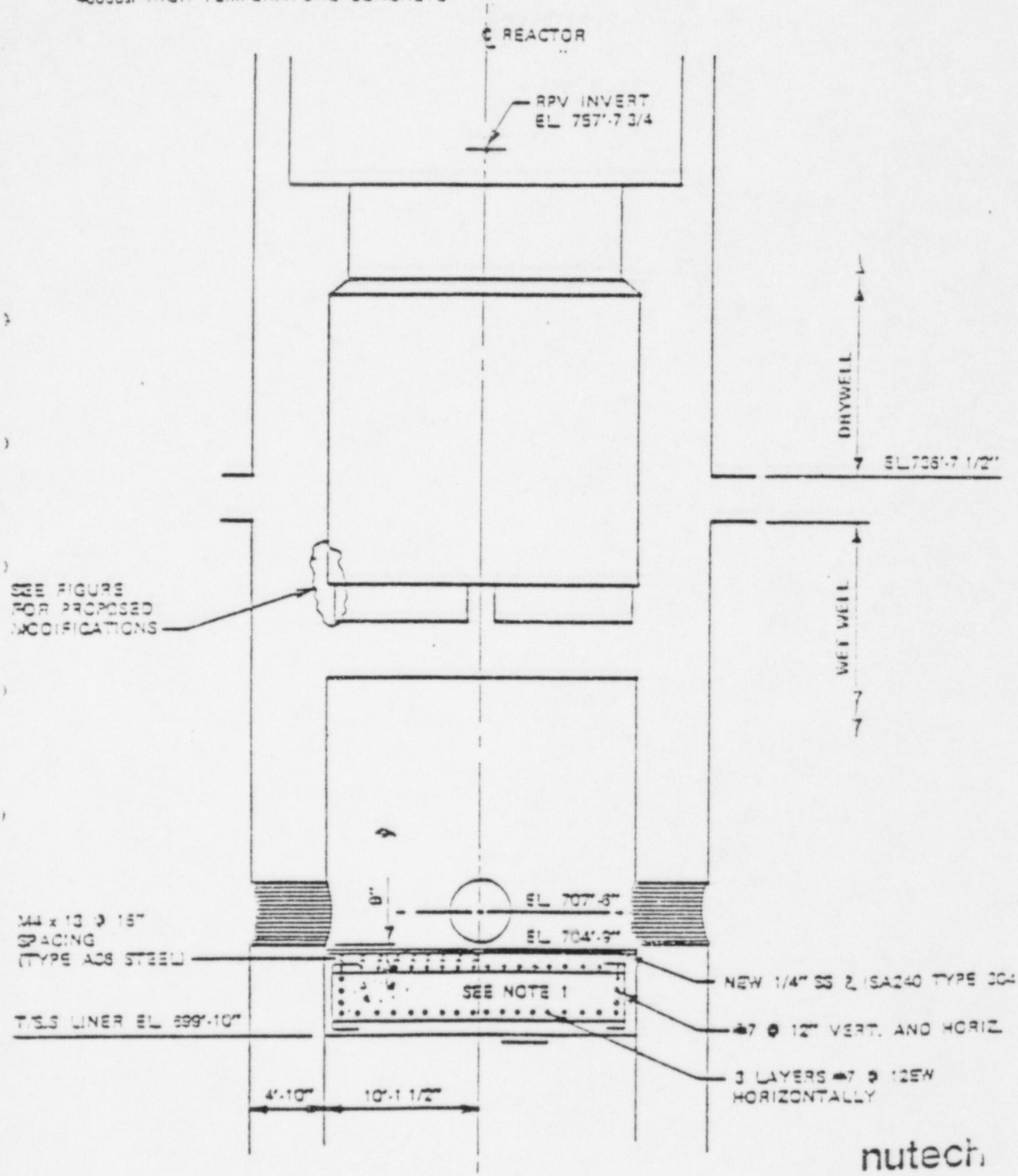
(1) PLUG TO BE POURED USING f'_c = 4000psi HIGH TEMPERATURE CONCRETE



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PROPOSED MODIFICATIONS TO RPV PEDESTAL

CORE DEBRIS PATH EVALUATION

CONCEPT 2 - SUPPRESSION POOL FLOODING OF CENTRAL PEDESTAL AREA

OBJECTIVE:

- PROTECT PEDESTAL FROM DAMAGE BY MOLTEN CORE MATERIALS AFTER FAILURE OF DRYWELL FLOOR

EVALUATION:

- RAISING SUPPRESSION POOL LEVEL COULD POTENTIALLY PREVENT PEDESTAL DAMAGE, AND WOULD RESTORE FILTERING BY SUPPRESSION POOL AND COOLING OF MOLTEN CORE MATERIALS

- SPRAY SYSTEM OPTIONS MAY BE CAPABLE OF PROVIDING ADEQUATE FLOW

COST: (SEE FOLLOWING DISCUSSION ON SPRAY SYSTEMS)

CONTAINMENT VENTING

CONCEPT 1 - VENT TO SECONDARY CONTAINMENT

OBJECTIVE:

- PROVIDE ABILITY TO VENT CONTAINMENT PRESSURE TO ASSURE CONTAINMENT INTEGRITY

EVALUATION:

- SECONDARY CONTAINMENT VENTING POSSIBLE THROUGH CONTAINMENT PURGE VALVES

- SAME ARGUMENTS AGAINST CONCEPT AS FOR MONTICELLO

COST: NOT EVALUATED

CONTAINMENT VENTING

CONCEPT 2 - VENT THROUGH STANDBY GAS TREATMENT SYSTEM
OR CONTAINMENT PURGE SYSTEM

OBJECT:

- ASSURE CONTAINMENT INTEGRITY AND PROVIDE FILTERED VENT PATH

EVALUATION:

- SYSTEMS NOT CAPABLE OF FULL-VENTING FLOW RATES

COST: NOT EVALUATED

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CONTAINMENT VENTING

CONCEPT 3 - BYPASS AROUND CONTAINMENT PURGE SYSTEM

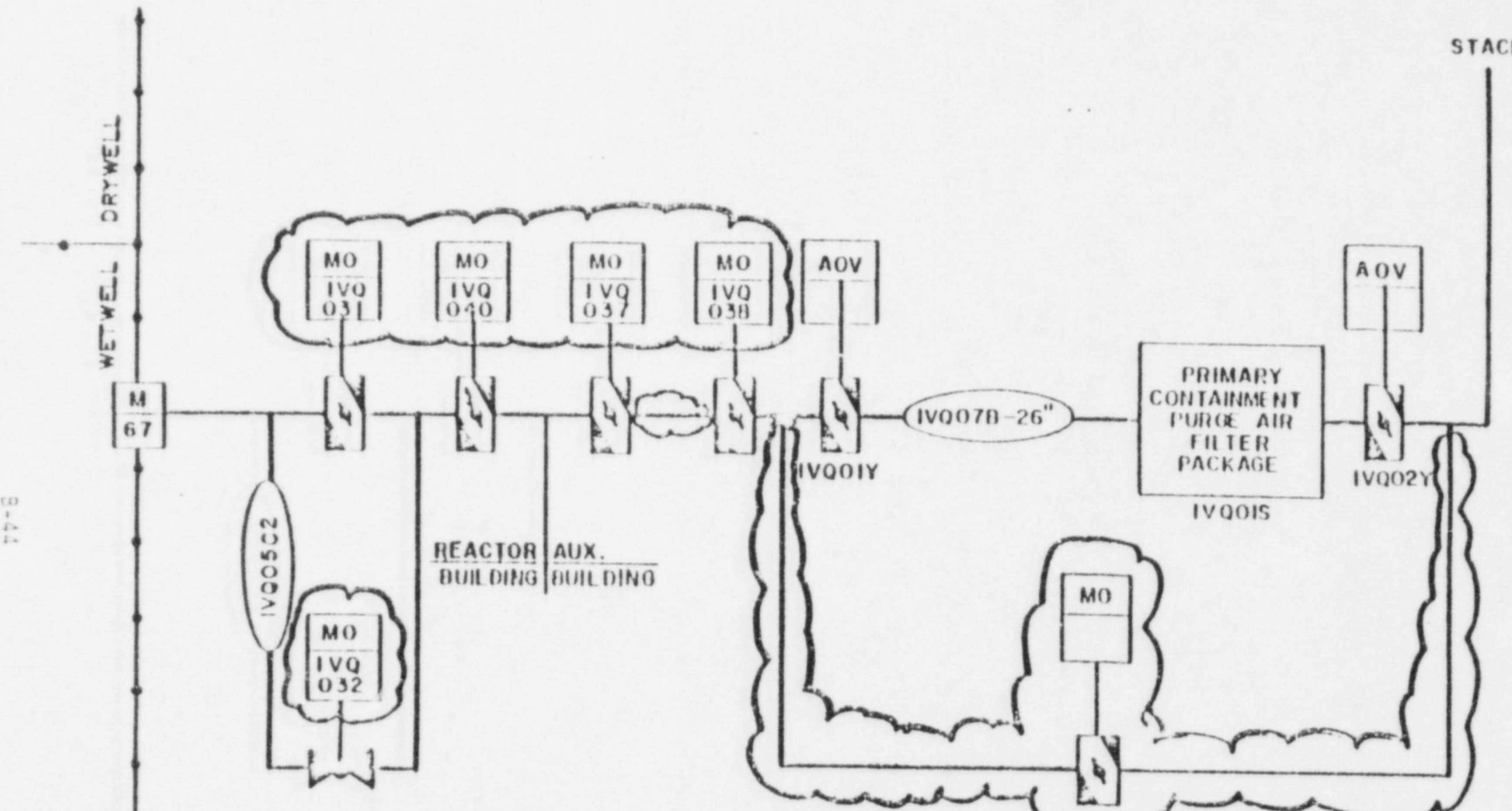
CONCEPT:

- ASSURE CONTAINMENT INTEGRITY, PROVIDE FULL-FLOW VENTING CAPABILITY AND PROVIDE LOW-FLOW FILTERED VENTING

EVALUATION:

- BYPASS AROUND CONTAINMENT PURGE SYSTEM PROVIDED FOR HIGH FLOW-RATE VENTING
- CONTAINMENT PURGE SYSTEM UTILIZED FOR LOW-FLOW VENTING
- STANDBY GAS TREATMENT SYSTEM ALSO AVAILABLE, IF POWER AVAILABLE, FOR POST ACCIDENT CLEANUP

COST: $1 - $2 MILLION



CONTAINMENT VENTING FLOW PATH

CONTAINMENT SPRAY

CONCEPT 1 - FIRE HOSE CONNECTION IN REACTOR BUILDING

OBJECTIVE:

- PROVIDE ALTERNATE WATER SOURCE FOR DRYWELL AND WETWELL SPRAY FOR PRESSURE CONTROL, COOLING, AND AEROSOL SCRUBBING

EVALUATION:

- WATER SOURCE AVAILABLE FROM 6-INCH MAIN IN REACTOR BUILDING
- ACCESS TO REACTOR BUILDING QUESTIONABLE
- BIG COST IS IN CONVERTING AC MOTOR OPERATED VALVES TO DC MOTORS

COST: $500,000 - $1,000,000

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CONTAINMENT SPRAY

CONCEPT 2 - INTERCONNECTION OF FIRE MAIN TO CONDENSATE SUPPRESSION CHAMBER FILL LINE

OBJECTIVE:

- PROVIDE ALTERNATE WATER SOURCE TO SPRAYS, WITH INTERCONNECTION OUTSIDE OF REACTOR BUILDING

EVALUATION:

- CROSS TIE LOCATED IN AUXILIARY BUILDING
- SPRAY FLOW EXCEEDING 4000 GPM AVAILABLE (2 PUMPS)
- SAME CONCERNS ON SPRAY EFFECTIVENESS AS FOR MONTICELLO
- MAJOR COST IS WITH CONVERTING VALVE MOTOR OPERATORS AND POTENTIALLY RESIZING BATTERIES

COST: $500,000 - 1,000,000

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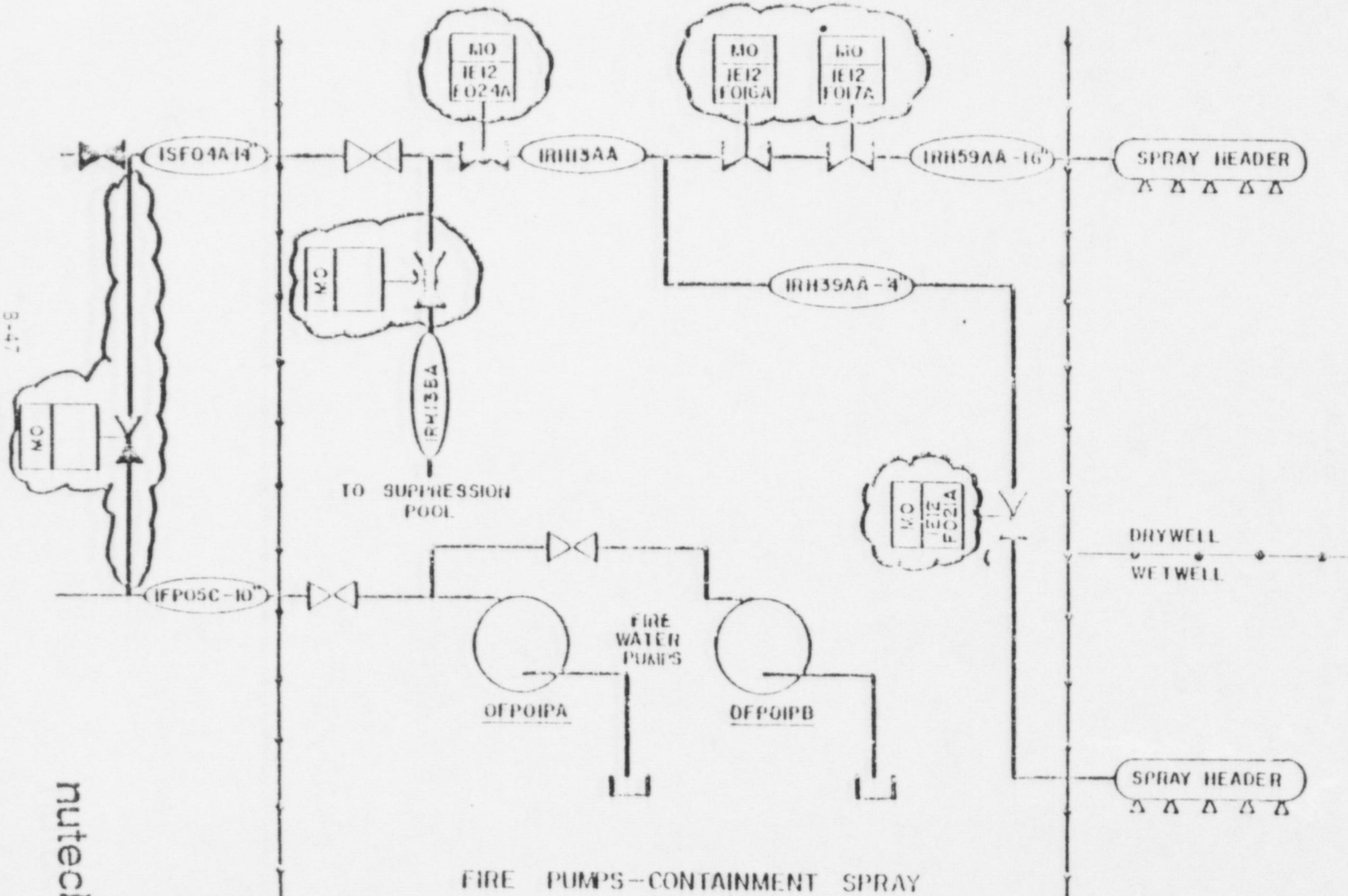

FIRE PUMPS—CONTAINMENT SPRAY FLOW PATH

nutech

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

October 3, 1986

MEMORANDUM FOR: James G. Partlow, Director
Division of Inspection Programs
Office of Inspection and Enforcement

FROM: Robert M. Bernero, Director
Division of BWR Licensing
Office of Nuclear Reactor Regulation

SUBJECT: PETITION OF WILLIAM R. G LDEN AND OTHERS FOR A SHOW CAUSE ORDER REGARDING PILGRIM STATION

As requested by Phil McKee by telephone, we have prepared the enclosed draft response to the part of a show cause petition by William R. Golden and others, dated July 15, 1986, which asserts inherent design deficiencies in the Pilgrim Station containment structure.

In the event we can be of further assistance in responding to the petition, please contact Gus Lainas (X29680), Jerry Hulman (X27941) or J. Kudrick (X27626) of my staff.

Robert M. Bernero, Director
Division of BWR Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: J. Taylor
P. McKee
G. Klingler
G. Lainas
L. Hulman
J. Kudrick
J. Zwolinski
P. Leech

FOIA-87-10

K/14

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DRAFT RESPONSE TO 2.206 PILGRIM PETITION

Containment Structures

The petition asserts that the General Electric Mark I pressure-suppression system employed by the Pilgrim reactor contains inherent design flaws which raise questions about its ability to withstand accidents. Generally, the concerns are related to:

(1) Design issues raised by Dr. S. H. Hanauer in the early 1970's, and

(2) The Pilgrim containment capability for severe accidents.

The following discussion addresses each of these concerns:

HANAUER ISSUES - The petitioner's concerns are based on AEC memoranda dated prior to 1978. More specifically, the key reference is a memorandum written by Dr. S. H. Hanauer in 1972 which raised several questions on the viability of pressure suppression containment concepts. The majority of those concerns have been either directly or indirectly quoted in the petition. They include references to inherent design flaws, lack of adequate testing, steam bypass susceptibility, volume limitations causing overcrowding, and a stated concern about the overall viability of the Mark I design.

At the time these issues were raised, the then AEC staff set about evaluating each concern to ensure that adequate safety margins were being maintained on existing plants. Additionally, the AEC also had under investigation various technical matters that related to the safety of pressure suppression type containments for light water cooled reactor plants. Based on its review of these efforts, the staff concluded that proper consideration had been given to each of the technical concerns. This staff evaluation was documented in NUREG-0474, dated July 1978: "A Technical Update on Pressure Suppression Type Containments in use in U.S. Light Water Reactor Nuclear Power Plants." As Enclosure A to this NUREG, a summary of NRC staff actions related to the technical issues identified by Dr. Hanauer's memorandum of September 20, 1972 was presented.

A review of the issues raised in the portion of the petition which are based on correspondence dated 1978 or earlier indicates that all of these issues have been addressed in NUREG-0474. The conclusions remain valid today and they are equally applicable to the Pilgrim Nuclear Power Station.

The petition also references statements from NUREG-0474 which relate to differences between expected experimental results and actual test results. Specific comments in the petition were made indicating that surprises repeatedly occurred during the course of the various, then ongoing, test programs. The statements extracted from NUREG-0474 were made during 1978 when many of the test programs were in their early stages.

Since the issuance of NUREG-0474, the generic test programs related to the Mark I containment design and the NRC assessment of the tests have been completed. These tests were a small portion of an extensive program which was conducted by the BWR Mark I owners group. The staff evaluation of this effort was reported in NUREG-0661. In addition, a plant-specific analysis was performed for the Pilgrim Nuclear Power Station based on the final test results. Based on this analysis, design changes were proposed to restore the intended safety margins. These changes have been both reviewed and approved by the staff, and modifications have been implemented at Pilgrim. As a result, the Pilgrim containment has been demonstrated to be capable of accommodating design basis accidents with adequate margin.

The petition refers to another of Dr. Hanauer's concerns related to the safety disadvantages of pressure suppression containments. This issue related to bypass paths in BWR pressure suppression containments, and was designated as Generic Issue 61, "SRV Line Break Inside the Wet Well Airspace of Mark I and II Containments." The staff evaluation of this issue has recently been completed and the results were presented in NUREG/CR-4594, "Estimated Safety Significance of Generic Issue 61." Based on these results, the staff concluded that no new requirements were justified, and no further study of this safety issue was warranted based on an overall risk assessment.

In summary, the petitioners have asserted that the pressure suppression containment design is flawed and they have questioned the viability of this containment type. In response to these assertions we have shown that many of the specific concerns had been previously and satisfactorily addressed in NUREG-0474. For those concerns identified since issuance of NUREG-0474, we have shown that a generic program was conducted to determine the loads under investigation. Additionally, Pilgrim has implemented design changes based on the program results to fully establish design margins.

PILGRIM CONTAINMENT CAPABILITY FOR SEVERE ACCIDENTS - Assertions by the petitioners concerning the performance of the General Electric pressure-suppression containment were raised from the viewpoint of containment failure in the event of a severe accident. The petitioners assert that there is a tendency to underestimate the probability of various types of accidents, citing among other things, the recent very severe accident at Chernobyl in the Soviet Union. The petitioners also conclude that there is a high probability that Pilgrim's Mark I containment structure will not withstand various severe accident scenarios.

Although no PRA is available explicitly for the Pilgrim plant, other plant-specific PRA's and assessments of accident sequences at specific plants offer insights relative to Pilgrim. First, the probability of accidents that could challenge a Mark I containment have been found to be quite low, involving a multiplicity of failures before proceeding to a core-melt. Second, containment analyses and plant specific assessments at other BWRs indicate a range of containment failure probability estimates, assuming occurrence of a core melt accident.

A conditional Mark I containment failure probability was quoted as .9 by the petitioners as a basis for concern. This estimate derives from the 1975 Reactor Safety Study, WASH-1400/NUREG-75/014, Table 5-3. The value reflects the assumptions and findings for the surrogate BWR plant, Peach Bottom (a two unit plant also with Mark I containments) as it existed at the time of the study.

Probabilistic risk assessments (PRAs) have been found useful in licensing and other regulatory activities in identifying vulnerabilities. Such assessments have been extensively reviewed for several plants. The numerical results of these assessments contain considerable uncertainties and it is not possible to conclude with certainty that they are either excessively conservative or non-conservative. On balance, the Commission in August, 1985 concluded in its Severe Accident Policy Statement[1] that, based on the information available, the severe accident risks at all U.S. commercial light water reactors were sufficiently low that no immediate action was required but that investigations of potential plant vulnerabilities would be undertaken.

The petitioners present no new information on the Pilgrim containment that was not known and resolved to the staff's satisfaction prior to issuance of the Commission's policy statement, other than a reference to the Chernobyl accident and an inaccurate comparison of rated containment pressures. (The 57 pounds per square inch quoted for Chernobyl does not apply to the building structure that houses the reactor core.) Information on the Chernobyl accident has not demonstrated to date a defect in the Pilgrim design. The staff is continuing to evaluate the Chernobyl event relative to U.S. plants and expects to issue its report on this matter in a few months. The petitioners do reference factors "not taken into account" in PRAs from a Union of Concerned Scientists'

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1 "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," F.R.50,32138-32151, August 8, 1985.

January, 1986 paper by S. Sholly and G. Thompson that was submitted as comment on NUREG-0956, Draft for Comment. The factors identified include aging of structures, technical specification violations and temporary exemptions thereto, construction defects and weaknesses, partial failure sequences, and external events. (Many PRAs do take external events into account.) Petitioners have not, however, presented evidence to demonstrate that these factors constitute a sufficient contribution to public risk to warrant a show cause order for Pilgrim.

As part of the implementation phase of the Commission's Severe Accident Policy Statement, the NRC staff has identified a set of elements for particular consideration that have the potential for substantial improvement in the mitigation capability of Mark I containments. These are based upon extensive research and analyses that provide improved understanding of the physical and chemical phenomena associated with severe core melt accidents. These elements were identified to the owners of boiling water reactors at a public meeting on June 16, 1986 and include (1) hydrogen control, (2) drywell sprays, (3) wetwell venting, (4) core debris control, and (5) development of related emergency procedures and operator training. The common thrust of these elements is to reduce substantially the likelihood of bypassing the large volume of water in the suppression pool which can act to remove large quantities of fission products that would be released from the reactor in a core melt accident.

At a second public meeting on September 11, 1986 with the BWR owners, the latter presented a report on their "Evaluation of BWR Accident Mitigation Capability Relative to Proposed NRC Changes", dated August 1986. This technical dialogue between the staff and the owners is continuing and is planned to result in a set of proposed new requirements on the Mark I containments, including Pilgrim, to be issued in December 1986 for comment in the form of a draft generic letter.

Boston Edison is an active participant with the other BWR owners in this evaluation effort. In addition, and specifically with respect to the Pilgrim Station, Boston Edison announced on July 25, 1986, that it is proceeding to implement certain modifications to the containment. This action will keep the plant shutdown until early 1987.

At a meeting with NRC on September 9, 1986, Boston Edison stated that it plans to provide the capability to connect the fire water system to the RHR system. This would provide an additional source of water for the drywell sprays. Provision will also be made for connecting fire truck pumping capability to the RHR system. The installation of a third diesel generator, which would be particularly helpful during station blackout conditions, is under serious consideration. Also, being considered is a reduction of the 24-hour period at the end of reactor operation when the containment is not presently required to be fully inerted. Procedure changes and training appropriate to all such plant modifications will be implemented before plant startup. The NRC staff will monitor these modifications to assure that they do represent safety enhancements and have no adverse safety impact on existing systems.

Petitioners also requested that the NRC require Boston Edison to submit a feasibility study on all possible structural modifications prior to NRC approval of specific modification proposals. At the present time neither Boston Edison nor the staff nor the petitioners have identified any structural modifications to the Pilgrim containment that would be warranted by severe accident considerations. This request is, therefore, denied.

CONCLUSION - The petitioners' assertions with respect to inherent design flaws in the pressure suppression system utilized at the Pilgrim plant have been previously reviewed and conclusions reached within the context of NRC regulations and guidance. Modifications to re-establish containment design margins have been implemented. Evaluations of the Mark I containment with respect to severe accidents are continuing: (1) through the implementation of the Commission Policy Statement on Severe Accidents; (2) through NRC staff and industry initiatives to improve containment performance for all BWR's; (3) through an NRC appraisal of the Chernobyl accident; and, (4) through the licensee's initiative. In no case has sufficient evidence been presented that would indicate that the Pilgrim plant should not operate while resolution and risk reduction improvements are considered. That is, there is not sufficient evidence of either design flaws at Pilgrim, or high risk, which warrants a show cause order for the plant to remain closed or to suspend the operating license.

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

October 24, 1986

Mr. J. Gary Weigand, President
and Chief Executive Officer
Vermont Yankee Nuclear Power Corporation
RD 5, Box 169
Ferry Road
Brattleboro, VT 05301

Dear Mr. Weigand:

SUBJECT: VERMONT YANKEE CONTAINMENT STUDY

This is in response to your letter dated September 2, 1986, enclosing the Vermont Yankee Containment Study for our review. We acknowledge the extensive effort in generating the report in a short interval of 60 days.

We have reviewed the report in two parts. The first part relates to the review of your comparison of the Vermont Yankee design features to those of the reference plant in WASH-1400 and calculation of a Vermont Yankee specific containment conditional failure probability (CCFP). The second part deals with our review of your response to five generic NRC staff concerns related to Mark I containments for Boiling Water Reactors (BWRs).

Your approach was to quantify the Vermont Yankee CCFP using the Peach Bottom analysis as a surrogate and modifying the accident sequence frequencies to reflect the specific design features and 14 year operating data base of Vermont Yankee, supplemented by industry experience where Vermont Yankee data was not available. The range of plant specific conditions were determined by phenomenological analysis using Industry Degraded Core Rulemaking (IDCOR) group Modular Accident Analysis Program (MAAP), deterministic structural capability calculations made for Browns Ferry and the Peach Bottom plants, and your staff's engineering judgement. Using the above approach, you have provided a best estimate CCFP value for Vermont Yankee containment of 7% for all sequences where containment may fail within 24 hours and where radionuclide releases include all noble gases and greater than 0.1% Iodines. You have, however, not provided an analysis of uncertainties in your methodology, in the phenomenological analysis using MAAP, and your engineering judgments.

Based on the staff's experience with other BWR Probabilistic Risk Assessment (PRAs) we believe that the CCFP of 7% may be fairly representative but is quite uncertain. It is our judgement that the CCFP based on your estimating techniques would have associated uncertainties as discussed in Enclosure 1. We also believe that your evaluation has provided the staff with sufficient insights to conclude that the CCFP for Vermont Yankee may be lower than the 90% estimated from results of WASH-1400. Our assessment of the uncertainties leads us to believe that the Vermont Yankee CCFP is probably less than 50%. That conclusion, that the CCFP is probably less than 50% and may be fairly estimated to be about 10%, brings out the very reason for the five generic NRC staff concerns related to Mark I containments. We believe that greater

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assurance of containment performance in the face of core melt is desirable and achievable. Therefore, our attention now is shifted to the second part of your study.

Our review of the second part of your report dealing with containment performance enhancements has resulted in some preliminary conclusions and several questions. We find that the kind of enhancements proposed in your study are consistent with the type of improvements considered by the staff. You have indicated that further analysis should be performed to test the feasibility and effectiveness of the concepts discussed in your study. We agree.

Enclosure 1 is our preliminary assessment of your study. Our evaluation proposes further studies regarding combustible gas control and core debris barriers. We request that you reevaluate these two issues in the light of the staff comments. Enclosure 2 lists detailed staff questions. We request your response to the staff questions and comments by November 17, 1986.

Sincerely,

Robert M. Bernero, Director
Division of BWR Licensing

cc: See next page

Mr. R. W. Capstick
Vermont Yankee Nuclear Power Corporation

Vermont Yankee Nuclear Power Station

cc:

Mr. J. G. Weigand
President & Chief Executive Officer
Vermont Yankee Nuclear Power Corp.
R. D. 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

Mr. Donald Hunter, Vice President
Vermont Yankee Nuclear Power Corp.
1671 Worcester Road
Framingham, Massachusetts 01701

New England Coalition on Nuclear Pollution
Hill and Dale Farm
R. D. 2, Box 223
Putney, Vermont 05346

Mr. Walter Zaluzny
Chairman, Board of Selectman
Post Office Box 116
Vernon, Vermont 05345

Mr. J. P. Pelletier, Plant Manager
Vermont Yankee Nuclear Power Corp.
Post Office Box 157
Vernon, Vermont 05354

Mr. Raymond N. McCandless
Vermont Division of Occupational & Radiological Health
Administration Building
10 Baldwin Street
Montpelier, Vermont 05602

Honorable John J. Easton
Attorney General
State of Vermont
109 State Street
Montpelier, Vermont 05602

John A. Ritscher, Esquire
Ropes & Gray
225 Franklin Street
Boston, Massachusetts 02110

Mr. W. P. Murphy, Vice President & Manager of Operations
Vermont Yankee Nuclear Power Corp.
R. D. 5, Box 169
Ferry Road
Brattleboro, Vermont 05301

Mr. Gerald Tarrant, Commissioner
Vermont Department of Public Service
120 State Street
Montpelier, Vermont 05602

Public Service Board
State of Vermont
120 State Street
Montpelier, Vermont 05602

Vermont Yankee Decommissioning Alliance
Box 53
Montpelier, Vermont 05602-0053

Resident Inspector
U. S. Nuclear Regulatory Commission
Post Office Box 176
Vernon, Vermont 05354

Vermont Public Interest Research Group, Inc.
43 State Street
Montpelier, Vermont 05602

Regional Administrator, Region I
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

ENCLOSURE 1

PRELIMINARY EVALUATION OF CONTAINMENT SAFETY STUDY
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
FOR THE VERMONT YANKEE NUCLEAR POWER STATION
VERMONT YANKEE NUCLEAR POWER CORPORATION
DOCKET NO. 50-271

1.0 INTRODUCTION

By letter dated June 30, 1986 from J. G. Weigand to H. Denton, the licensee, Vermont Yankee Nuclear Power Corporation, committed to perform a containment safety study. By letter dated September 2, 1986 from J. G. Weigard to H. Denton, the licensee transmitted the study for staff review. On September 12, 1986 representatives of the licensee met with the staff, summarized the study, and answered questions. On September 28 and 29, 1986, staff representatives visited the plant and the neighboring Vernon Hydro Station to review plant features and portions of the study.

The purpose of this evaluation is to provide the staff's preliminary comments on the licensee's study. Final comments will be based on the licensee response to staff's questions and comments. A regulatory position on generic implementation of improvements must await review of backfitting implications under the requirements of 10 CFR 50.109. The following evaluation should not be considered a Nuclear Regulatory Commission position on the issue of Vermont Yankee Nuclear Power Station containment safety.

2.0 EVALUATION

The licensee's study basically consists of two parts. The first part is an evaluation of containment failure probabilities in severe accidents. The

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second part is an evaluation of potential improvements in five areas (combustible gas control, drywell spray capability, containment venting, core debris control, and training and procedures).

2.1 Containment Failure Probability Evaluation

The licensee evaluated the design and operation of Vermont Yankee Nuclear Power Station (VYNPS) with respect to core melt probabilities and containment system response. The evaluation used detailed assessments for a similar reactor, adjusted for specific VYNPS features. The principal conclusion by the licensee was that a best estimate of the conditional probability of containment failure was .07. That is, if a core melt accident were to occur at VYNS, there is a 7 percent chance that the containment systems will fail and result in a large release of fission products.

The bases for the licensee's estimate include a number of assumptions, evaluations and judgements. One important judgement pertains to the use of results from an industry developed computer code called Modular Accident Analysis Program (MAAP). The staff has not reviewed this code, at present has no plans to do so, but will assess the results and conclusions from its applications to VYNS. The accuracy and application of the code is a subject of considerable debate however. Therefore, uncertainty must be attached to the use of the code at this time.

A number of important conclusions have been drawn by the licensee based on the containment safety study. The more important ones are:

° A conditional containment failure probability of .07.

° A maximum containment system failure pressure of about 135 psia.

° The best estimate core melt probability was $3x10^{-5}$ per reactor year (one chance in 30,000 per year of reactor operation).

° One accident class (loss of cooling capability with the pressure vessel at high pressure) dominates the core melt probability (about 40 percent).

° Although station blackout and ATWS are significant contributors to the core melt probability at Vermont Yankee, as for other plants, the largest contributor is loss of makeup with the vessel at high pressure.

The staff's experience with core melt and containment failure estimates indicates that large uncertainties exist in these estimates. In addition, a high pressure core melt entails greater uncertainty regarding containment response than prevails with low pressure core melt. Pending receipt and analysis of the response to staff questions, the staff concludes that the licensee's estimates appear optimistic considering the uncertainties inherent in failure rate data, modeling of systems, human responses, accident initiator identification, as well as the physical processes that follow degraded core conditions.

Vermont Yankee has four SRVs which serve as the automatic depressurization system (ADS), compared to eleven SRVs at Peach Bottom, five of which serve as the ADS. Further, at Vermont Yankee the four SRVs are three-stage Target

Rock valves which have relatively high failure rates based on reported failures. When the the failure experience of SRVs is considered, accident sequences TPUV, TPQUV, T_EPQX, T_EPQUV, T_MQUV and TPW need to be added to the dominant accident sequences identified by the licensee. As a result, the total core damage frequency of $3x10^{-5}$/R-yr could be increased by about a factor of 2. When SRV failures (spurious opening and stuck open relief valves) related accident sequences are considered, the release type for Class II accidents (i.e., loss of containment heat removal and subsequent loss of makeup) may change. This is due to possible early failure of the RHR capability after loss of NPSH as a result of stuck open relief valves and subsequent pool heatup. As a result, the containment conditional failure probability for Vermont Yankee could exceed 7% for all sequences and result in containment failure within 24 hours.

The frequency of Loss Of Offsite Power (LOOP) at Vermont Yankee is estimated to be 0.07/yr. When compared with 0.22/yr for the national average grid, the Vermont Yankee value is deemed too low based on the conventionally used Bayesian estimation technique. The failure rate of the nearby hydropower plant is estimated to be higher than 0.07/yr. In the Bayesian estimate, the plant data are used as inputs modifying the a priori distribution, i.e., the national average LOOP frequency.

Additionally, the estimated 6-8 hour battery capacity at Vermont Yankee was based on 2175 ampere-hours rated capacity discharged at a constant load of 1.75 volts/cell, with the load estimated to be 27 amperes. This does not include the excessive loads likely to occur during the startup of motors, and the abnormally high ambient temperatures likely during the accident. Thus, a more likely estimate for the battery capacity is on the order of 3-4 hours when all those factors are considered. This will, in turn, affect the core damage frequency and the conditional containment failure probability of the complete station blackout sequence. If there is additional battery capacity not included in the Vermont Yankee Containment Safety Study, the Licensee should state so for clarification.

Other important areas of considerable uncertainty are the manner in which the vessel may fail (and the amount of resulting steel in the core debris), the effectiveness of the reactor building for fission product attenuation, iodine may not be a good surrogate for risk estimation (refractory products from core concrete interaction may be better), and the containment loadings and response following vessel head failure.

Based on the above, we conclude the CCFP for VYNP is probably less than 50%.

2.2 Current NRC Concerns For U.S. Containments

Severe accidents dominate the risk to the public associated with nuclear power plants. A fundamental objective of the Commission's Severe Accident Policy is to take all reasonable steps to reduce the chances of occurrence of a severe accident, and to assure the capability to mitigate the consequences of such an accident should one occur. The Reactor Safety Study report issued in 1975 found that for BWRs the probabilities of accidents resulting in core melt were low, but the containment performance following a severe accident was poor and tended to offset the benefits of low BWR core melt probabilities. Subsequent actions resulting from the TMI Action Plan have led to several plant modifications and required improvements in plant procedures to further reduce the likelihood and consequences of severe accidents. In concert with the Commission's policy to further reduce the chances of occurrence of severe accidents and to mitigate their consequences, an industry initiative is underway to develop a methodology for Individual Plant Evaluations (IPEs) for use in the search for risk outliers. The resulting approach will be applied on a plant specific basis. The staff may find that, while the IPE approach may satisfactorily address system reliability and containment performance for each plant specifically, the process will necessarily be a slow one. The staff has, therefore, identified several potential deterministic containment enhancements which lend themselves to generic implementation and have the potential to significantly mitigate the consequences of most severe accidents. The generic approach has the advantage of expeditious implementation on all plants and will be responsive to Commission's policy regarding mitigation of the consequences of severe accidents.

Based on the insights gained from PRAs, the staff has identified the following five areas of potential BWR containment enhancements.

1. Provisions should be made for reliable operation of drywell containment sprays for a broad spectrum of accident sequences, including blackout sequences. The reliability of containment sprays should be enhanced by providing independent water and power sources. Backup water sources and pumps, hose connection and use of fire mains should be considered. The provisions to be implemented should minimize

occupational exposures that could result from manual actuation, and procedures should be explicitly developed and expeditiously implemented as part of the BWR Owners Group development of the Emergency Procedures Guidelines.

2. Provisions should be made for symptomatic response and reliable actuation of containment wetwell purge and vent valves. They should open and close under accident conditions as a means to assure that the beyond-design-basis events do not lead to overpressure failure of the containment and the selected vent paths. They should provide a path for releases which will maximize the use of the suppression pool as a condensing and filtering medium.
3. Emergency procedures and training should be reviewed and modified as necessary to assure that operators are able to recognize severe accident conditions and use plant equipment to best advantage under such conditions. Revision 4 of the BWR Emergency Procedures guidelines should be implemented promptly following the staff's review and approval.
4. Paths for core debris travel should be evaluated for conditions representative of a large scale core melt. Where the expected path of debris travel indicates a substantial likelihood of loss of the suppression pool as a release filtering or debris quenching medium in BWR containments, the torus room under the suppression pool should contain a 3 foot high barrier to trap water and core debris.
5. Combustible gas control provisions should provide substantial assurance that containment failure due to hydrogen combustion is not likely in the

potential severe accident sequences, including blackout sequences. The period of operations while containments are deinerted while at power, particularly during potential preshutdown conditions, should be minimized by reducing the present Technical Specifications permitted value of 24 hours.

The licensee has evaluated the five areas as appropriate to Vermont Yankee and has found that with the exception of combustible gas control and debris control enhancements, the proposed enhancements could be beneficial for Vermont Yankee severe accident mitigation capability. The staff recommends that the licensee should reconsider the combustible gas control and debris control issues in the light of the following staff evaluation. The licensee should also complete the further studies proposed in its containment study report, propose specific modifications that it plans to undertake, and provide a tentative schedule.

2.2.1 Combustible Gas Control

Vermont Yankee is allowed by its Technical Specifications to be deinerted for 24 hours at startup and 24 hours prior to shutdown. The staff has proposed the approach of minimizing the deinerting time, thus reducing the vulnerability of hydrogen combustion. A 12 hour deinerted period may be sufficient. Based on discussions with the license during a September 11, 1986 site visit, the staff understands that it takes about 8 hours to deinert the containment. Therefore, it appears feasible to reduce the deinerting period in the Technical Specifications.

In the Vermont Yankee containment study the licensee has concluded based on operating experience that deinerted power operation is about 1.1 percent of the total operating time, and concludes that such a low percentage will not significantly impact safety.

Since the licensee was satisfied that its design objectives have been met, any impacts associated with shorter deinerting times were not identified. This issue should be explored further in order to address the staff's goal relating to increased assurance that the presence of a combustible mixture inside primary containment is reduced to as low as practicable levels. To this end, the study should include an evaluation of past practice to determine the impacts to plant operation due to a shorter allowable deinerting period.

2.2.2 Containment Sprays

The licensee has identified and implemented means to improve the reliability/availability of drywell sprays through the use of the following installed features:

a) redundant sources of water to the containment sprays; namely, the torus, the ultimate heat sink, and the cooling tower basin.

b) redundant pumping capability for the containment sprays; namely, the RHR pumps and the SWS pumps.

c) the capability to operate the containment sprays during loss of a.c. power by using d.c. operated valves and a diesel driven fire pump.

d) the capability to use the diesel driven fire pump to pump water through the RPV and into the containment during severe accidents that violate the RPV integrity. This mode could be used to cool the core debris and avoid over depressurizing the containment. Additionally the licensee indicated the possibility of automating the valves in the Reactor Building that are required to line up the diesel fire pump with the containment and RPV sprays.

The licensee has concluded that it is practical to use the containment sprays to control containment pressure and temperature and cool core debris during a severe accident.

The staff concurs with the licensee that using containment sprays after a severe accident can assist in the control of containment pressure and temperature, scrub the containment atmosphere and cool the core debris. However, in order to accomplish these objectives sufficient spray flow must be maintained. The diesel driven fire pump may have insufficient flow capacity to produce a spray that is effective for pressure and temperature control, scrubbing, and debris cooling. Insufficient flow to the sprays can result in water droplets that are too large to scrub or control pressure and temperature, and spray coverage may be insufficient to cover the core debris. Additional alternate water supplies (and/or spray nozzles) to the containment sprays should be considered since the diesel driven fire pump cannot supply sufficient water to ensure fission product attenuation and heat removal. The IDCOR/BWROG study (Evaluation of proposed BWR Accident Mitigation Capacity Relative to Proposed NRC Change) of the use of sprays is in conformance with the staff objectives. Further, IDCOR/BWROG estimated that only 250 gpm is necessary to cool the core debris to prevent it from damaging the containment. Although 250 gpm may be sufficient to remove decay heat from the core debris when the spray is ideally distributed, a larger flow of water from the sprays may be required to establish a good spray distribution for fission product removal and to remove the latent heat from the melted core and debris.

The licensee's proposed usage of the containment sprays is a reliable and cost effective means for post severe accident control of containment pressure and temperature, scrubbing of the containment atmosphere, and cooling the core debris, and is consistent with the staff's views. However, in order to be assured of sufficient flow for severe accidents further study is recommended to better understand spray performance at derated flow conditions.

2.2.3 Venting

With regard to the venting issue, the licensee discussed two of the most significant accident scenarios; station blackout and ATWS. The report concluded that containment venting should only be considered as a last resort. The staff agrees that venting the noble gas activity is a procedure of last resort. The licensee also concluded that venting may be of little value in preventing core damage for an ATWS. In the case of station blackout, the licensee states that opening a wetwell vent is difficult and the potential benefits are limited. Therefore, the licensee concluded that containment venting is not considered to be practical.

Even though the report concluded unfavorably with regard to venting, it identified six possible vent paths from the existing piping arrangements. After evaluating each path, a "12 inch vent path" was proposed as being the most desirable. This proposed path bypasses the Standby Gas Treatment System filters. The modification would require that approximately 15 feet of 12-inch pipe be added and that one new motor-operated isolation value he added to the line. The proposed three-inch Atmospheric Control System line includes the Standby Gas Treatment System. For this pathway, additional vendor analysis and testing would be required for the reliability of the valves. Both paths would utilize the vent stack. The final selection was left open pending further study, which will include consideration of competing safety requirements.

None of the potential vent paths identified by the licensee has been shown capable of opening and closing at the pressures, temperatures, radiation, and steam environments associated with severe accidents. The issues of a.c. independent power sources for the needed valves have not been evaluated by the licensee.

The staff agrees that further study should be performed. In the study, consideration should be given to more recent investigations such as the Evaluation of proposed BWR Accident Mitigation Capacity Relative to Proposed NRC Change, submitted by IDCOR/BWROG, dated August 1986.

In particular, the availability of adequate independent power under accident condition should be considered (station blackout). The Vermont Yankee study did not consider independent power sources because an earlier study (Harrington, R.M. and Hodge, S.A.; "Containment Venting as a Severe Accident Mitigation Technique for BWR Plants with Mark I Containment"; June 26, 1986) concluded that wetwell venting during Station Blackout has limited beneficial potential for BWR Mark I containment plants. This limited benefit may be significantly understated because of key assumptions in their analysis and, in addition, it should be weighed against the IDCOR finding that station blackout is a significant contributor to core melt probability. The study should also include an engineering evaluation that compares venting requirements with venting capabilities. The report stated that the need for substantial venting capability may arise during an ATWS scenario in which the energy in the form of steam that must be removed from the containment would be in the range of approximately 10% to 40% of rated reactor power. However, it is not clear if the vent paths noted earlier have this capability. The IDCOR study indicated that vent sizes for power levels of 20 to 30 percent may require vents of 26 to 34 inches in diameter, whereas the sizes required to remove decay heat after 10 minutes decay for other scenarios would range from 4 to 6 inches.

Another aspect of venting that can have a significant effect on accident consequences (onsite as well as offsite), and that needs further consideration, is the consequences of elevated releases.

If the vent path includes the plant stack, the elevated release during venting can prevent the contamination of onsite buildings and equipment. As noted in the licensee's study, some vent paths have the potential for making portions of the plant inaccessible for accident control and mitigation functions. Specifically, some of the vent paths have the potential for releases into the Reactor and Control Buildings. One advantage of an elevated release through the plant stack is that it maintains operator accessibility to vital plant areas. Elevated releases can also reduce substantially offsite dose consequences in the vicinity of the plant. Reduction factors of 10 or more are conceivable within two or three miles. Hence, the licensee should evaluate and quantify the specific benefits that can be achieved with respect to offsite dose consequences by venting through the plant stack. Specifically, the licensee should consider site specific meteorology and topography in the evaluation of venting through the plant stack. The offsite atmosphere transport should extend to distances where the effects of stack height become negligible.

2.2.4 Core Debris Barriers

The use of core debris barriers to prevent molten core debris from penetrating the steel containment shell or torus vent pipes has been investigated by the licensee. During initial construction the bottom of the steel drywell at Vermont Yankee was backfilled with concrete to El. 238'. Should core debris melt through the reactor it will fall on this concrete in a 17' -2" diameter enclosed area (231.5 ft^2) known as the Subpile Room. The only paths available for core debris to exit the Subpile Room are through its only doorway, or through either the three-inch steel floor drain or the three-inch steel equipment drain. The Subpile Room floor and equipment drain lines are imbedded in the drywell floor concrete and are routed to sumps inside the drywell. There is a minimum of approximately 1.5 feet of concrete underneath the sumps and piping before reaching the steel drywell shell. The Subpile Room doorway is a curbless opening in the portion of the biological shield wall which supports the reactor skirt.

The licensee states that should core debris melt through the reactor onto the circular Subpile Room floor and stay molten with enough vertical head available to cause it to flow, it could pass out of the Subpile Room though its only doorway, or through either the three-inch steel floor drain or the three-inch steel equipment drain. The Subpile Room floor and equipment drain lines are imbedded in the drywell floor concrete and are routed to sumps inside the drywell. If the entire volume core debris is available for spreading, a maximum layer thickness of approximately 1.1 inches would result. Since the vent line is approximately 1 foot above the drywell floor, it is not expected that the molten core debris could enter the torus through the vent pipes.

The licensee concludes that Vermont Yankee analysis indicates that a scenario in which sufficient core debris melts and propagates to the containment boundary does not appear feasible. Vermont Yankee has a similar size drywell and reactor vessel as the Peach Bottom plant analyzed in WASH-1400, yet has less than half the number of fuel assemblies and control rods.

Significant uncertainty exists in core and steel debris volume and transport analyses. The Vermont Yankee analysis indicates that core debris will not reach the drywell shell. Vermont Yankee already has diverse capability to spray water into the containment. The licensee states that additional physical barriers may be counter-productive as they may prevent containment spray from cooling the debris while it is confined in the Subpile Room. The Vermont Yankee analysis shows core debris control is closely coupled to containment spray capability.

Our preliminary assessment is that full core debris barriers as considered by the licensee may not be cost effective for Vermont Yankee. However, the licensee should examine localized barriers protecting the downcomer opposite the doorway of the Subpile room and two adjacent downcomers (one on each side). Such a modification may not be costly, and will provide added assurance that core debris travel will not cause suppression pool failure or bypass.

2.2.5 Severe Accident Procedures

Vermont Yankee's present symptom-oriented Emergency Operating Procedures (EPGs) are based upon the latest approved version of the BWROG EPG's (Revision 3). However, containment venting and reactor power control using water level for ATWS conditions were not included in the Vermont Yankee Emergency Procedures.

The licensee has indicated a willingness to consider updating the emergency operating procedures to include the EPG Rev. 4 guidance on venting and ATWS power control. The licensee should implement the operating procedures currently being considered by industry in EPG Rev. 4, following NRC staff reviews. Any future addenda which may result from the current program of enhancing the containments capability to mitigate the consequences of severe accidents.

2.2.6 Other Improvements

Other potential containment performance improvements should also be considered. For example, the licensee should consider supplemental power supplies for improved automatic depressurization system operation for station blackout response. This is particularly important since loss of makeup at high pressure appears to be the largest contributor to core melt.

3.0 CONCLUSIONS

The staff concludes that the Vermont Yankee containment safety study has provided evidence that the containment is more capable of performing its function during severe accidents than previous assessments of Mark I type containments would indicate. That is, given a core melt accidents, a preliminary evaluation indicates a less than 50 percent likelihood of containment failure. The staff also concludes that the likelihood and consequences of such a failure may be reduced substantially with modest improvements as discussed above. The staff, therefore, recommends that the licensee perform the feasibility studies indicated by the study, and those proposed in this evaluation.

ENCLOSURE 2

VERMONT YANKEE CONTAINMENT STUDY

REQUEST FOR ADDITIONAL INFORMATION

1. What is your estimate of the overall uncertainty of conditional containment failure probability and its basis?

2. What is the affect on core damage frequency when accident sequences TPUV, TPQUV, T_EPQX, T_EPQUV, T_M QUV, TPW are included in the dominant accident sequences based on reduced battery life, and the number and type of SRVs compared to Peach Bottom, and on the CCFP?

3. Given that the national average value for frequency of loss of offsite power is on the order of 0.22/yr justify, on the basis of the Bayesian estimate that the frequency of loss of offsite power at Vermont Yankee is 0.07/yr.

4. Verify that the total battery capacity available at Vermont Yankee is greater than 2175 ampere-hours, and that it could be maintained at a voltage greater than 1.75 volts/cell in high ambient temperature during the accident for 6-8 hours.

5. How often are the RHR/RHRSW interconnecting valves actuated to assure that the valves work properly?

6. How often are the interconnecting valves between the RHRSW and the fire protection system (fire pumps) actuated to assure that the valve works properly?

7. How readily can the MSIVs be reopened following closure at operating conditions? What interlocks must be bypassed and how complicated are the procedures (e.g., must the differential pressure across the MSIVs be reduced for the valves to be re-opened)?

8. It is not clear how the CCFPs given on page 74 of the report were obtained. Please explain.

9. What SLCS modifications are proposed for V.Y.? Page 86 discusses the advantages of two different possible modifications, but gives no commitment to either.

10. Identify the testing and maintenance requirements you use for the diesel driven fire pump. Do these requirements conform to those contained in the National Fire Codes? Also identify any reliability information for the system such as outages and failures to start on demand. What outage time limitations do you use for the system while at power?

11. Identify the scope of modifications required to the spray system, or increases in the pumping capacity, to assure a uniformly distributed spray with proper droplet size (as opposed to a dribble) if the diesel fire pump were used in a core melt event. Approximately what would the costs be of such modifications?

12. Can portable AC generators be used effectively to power vital valves and/or small pumps for station blackout accidents? If so, what modifications would be required, and what would be their approximate costs?

13. In section 2.2.1 you conclude that the containment can be "expected to withstand pressures approximately two times design prior to failure." Provide the bases for your conclusion.

14. In section 2.2.10.3, isn't the water supply from at least a portion of the cooling towers also available?

15. The use of the Vernon Hydroelectric Station is referenced in Section 2.2.11.1, and discussed in more detail in Section 4.4.2.3. Reliability estimates are presented on page 62. Please provide the basis for the reliability estimates with reference to both the historical operation of Vernon Hydro, and the transmission line and substations to Vermont Yankee.

16. The Nitrogen Containment Atmosphere Dilution (NCAD) system is referenced on pages 25 and 115. What maintenance and surveillance procedures are used to ensure operability?

17. In Section 4.1.4 MARCH/RMA and MAAP code package for Vermont Yankee is referenced. Were calculations made for Vermont Yankee, or were the results of computation for other reactors evaluated for the Vermont Yankee design? What calculations were made?

18. Invessel and exvessel steam explosions were not considered credible (pg. 55, 1st para.) based upon research. Identify the research that forms the basis for this conclusion.

19. As we understand tables 4.7 through 4.10, the designations E, L, and NCL refer to early (E) or late (L) containment failure estimates, and NCL refers to no containment failure. The second designators H, M and L refer to high, medium and low releases, respectively. To what extent can mitigation through manual actions in the time available, and in the temperature and radiation environments associated with such accident types be expected to be successful for early failures; for late failures? Specifically, for the combustible gas control, spray, and venting evaluations, what do you judge the effectiveness of the existing plant and procedures to be versus the possible improvements for early and late sequences?

20. NPSH during spraying is identified as a concern on page 117. To what extent will further investigation be undertaken to determine whether NPSH is an issue? Verify that procedures exist for the operator to lineup ECCS water sources outside the containment in the event NPSH reouirements are not met. If analysis indicates it is an issue, what do you propose be done to eliminate or reduce the level of concern?

21. Venting is considered for the station blackout sequences only. Please discuss your rationale for not considering other events when venting may be beneficial.

22. Since there is a substantial difference between the heights of the VYNPS plant stack (318 feet) and the reference plant stack (500 feet), indicate how this was taken into account in the comparison of the two plants.

23. Evaluate the differences in offsite dose consequences due to venting at ground level versus through the stack. Using site specific meteorology and topography, provide an estimate of the offsite dose differences between the two types of release as a function of distance from the site.

24. On page 125, rapid containment depressurization which could fail the drywell is offered as an uncertainty relative to containment venting. What analysis and/or tests are being conducted to reduce this uncertainty? If no analysis or tests are contemplated, what actions are proposed to minimize the uncertainty?

25. Remote manual valve operation is discussed in Section 5.3.5.1.1., primarily with respect to station blackout. To what extent can the remote vent valve and any spray valve alignment be counted on for the other classes of sequences you assessed? That is, if remote manual operation is not available, would the local environment the operators would encounter allow successful local operation?

26. Severe accident venting discussed in Section 5.4 does not include an evaluation of the reliability of the ADS system. Given the types of severe accident challenges you have described, provide your estimates of the reliability of ADS valves; i.e., their potential as a suppression pool bypass path. Can battery packs or portable generators be used to assure high reliability? If so, at what approximate cost?

27. What is the approximate cost for improving the valving for the diesel fire pump?

28. For the improvement options you have evaluated, what maintenance and surveillance guidelines would you propose to use?

29. To what extent do you consider the option of drywell flooding to be effective? If effective, would you include the option in future revisions to your emergency operating procedures.

30. It is estimated that the maximum debris layer thickness on the drywell floor would be approximately 1.1 inches (page 136). Provide the bases for such a conclusion.

31. What is the thickness of the vent duct between torus and drywell? (Fig 1, Page F-6).

32. It is stated that "additional physical barriers are believed to be counter-productive as they may prevent containment spray from cooling the debris while it is confined in the Subpile Room." Please elaborate how such a barrier would prevent the spray from effectively cooling the debris. (p 137).

33. It is indicated that, with the exception of the one-inch nitrogen CAD line and the six-inch nitrogen purge line, the pipe lines associated with four potential vent paths are likely to fail. Provide background information which led to this conclusion (p 131).

34. It is implied that a layer of debris (1.1 inch thick) would not penetrate the drywell steel shell and enter the torus (p 136). If such is the conclusion, please discuss why such a burn through is unlikely while core debris is attacking the dry well floor. There is a gap between drywell steel shell and concrete shield. If the molten core were to burn through

the steel shell at the indicated corium elevation, what would prevent the fission gas from entering the reactor building since the concrete shield outside the drywell shell is not designed as a pressure boundary?

35. It is stated that 135 psia is a reasonable value for the VY Containment failure pressure (page F-5)
    a) What is the uncertainty range associated with this value?
    b) What would be a change in core melt and conditional containment failure probabilities associated with the uncertainty?
    c) Provide references for the Ames and Sandia calculations mentioned in the Appendix F. (page F-1)

36. Your evaluation of deinerting indicates a relatively few hours of power operation while the containment is deinerted (i.e., about 1% in the Run Mode). For each such instance, please identify the following:
    a) the number of hours deinerted;
    b) the purpose for the deinerted condition, and whether it was successful; before shutdown was required by your technical specification, and
    c) the power level and its corresponding reactor pressure at which containment entry and exit were made.

    Please indicate the impacts you would expect for a deinerting technical specification to either 16 hours, or 12 hours.

37. Please provide your estimates of pressure and temperature as a function of time for the accident sequences you analyzed for CCFP estimates.

38. It is not clear from your evaluation why the probability estimates of early failures with higher releases are lower than for Class IV events. Please explain if venting of ATWS sequences before core melting was assumed?

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

0 8 DEC 1986

Docket No.: 50-443

Mr. Robert J. Harrison
President & Chief Executive Officer
Public Service Company of New Hampshire
Post Office Box 330
Manchester, New Hampshire 03105

Dear Mr. Harrison:

SUBJECT: TRANSMISSION OF BROOKHAVEN NATIONAL LABORATORY DRAFT REPORT OF THE SEABROOK EMERGENCY PLANNING SENSITIVITY STUDY

Brookhaven National Laboratory has submitted its draft report to the NRC documenting the findings from its review of the Seabrook Station Emergency Planning Sensitivity Study and supporting documents. The NRC is making the report available at this time for your information and use. You should note that the NRC staff process of review and comment on the Brookhaven draft has just begun, and the Brookhaven final report will not be produced until the authors have had the opportunity to assess formal NRC staff comments. Therefore, the report is subject to change.

Victor Nerses

Victor Nerses, Project Manager
PWR Project Directorate No. 5
Division of PWR Licensing-A

Enclosure:
Draft Technical Evaluation of the
EPZ Sensitivity Study for Seabrook
Technical Report A-3852

FOIA-87-10

K/20

8612170289XA

4pp

Mr. Robert J. Harrison
Public Service Company of New Hampshire

Seabrook Nuclear Power Station

cc:
Thomas Dignan, Esq.
John A. Ritscher, Esq.
Ropes and Gray
225 Franklin Street
Boston, Massachusetts 02110

Mr. Bruce B. Beckley, Project Manager
Public Service Company of New Hampshire
Post Office Box 330
Manchester, New Hampshire 03105

Dr. Mauray Tye, President
Sun Valley Association
209 Summer Street
Haverhill, Massachusetts 01839

Robert A. Backus, Esq.
O'Neil, Backus and Spielman
116 Lowell Street
Manchester, New Hampshire 03105

William S. Jordan, III
Diane Curran
Harmon, Weiss & Jordan
20001 S Street, NW
Suite 430
Washington, D.C. 20009

Mr. Philip Ahrens, Esq.
Assistant Attorney General
State House, Station #6
Augusta, Maine 04333

Mr. Warren Hall
Public Service Company of
New Hampshire
Post Office Box 330
Seabrook, New Hampshire 03874

Seacoast Anti-Pollution League
Ms. Jane Doughty
5 Market Street
Portsmouth, New Hampshire 03801

Mr. Diana P. Randall
70 Collins Street
Seabrook, New Hampshire 03874

Richard Hampe, Esq.
New Hampshire Civil Defense Agency
107 Pleasant Street
Concord, New Hampshire 03301

E. Tupper Kinder, Esq.
G. Dana Bisbee, Esq.
Assistant Attorney General
Office of Attorney General
208 State Hosue Annex
Concord, New Hampshire 03301

Resident Inspector
Seabrook Nuclear Power Station
c/o US Nuclear Regulatory Commission
Post Office Box 700
Seabrook, New Hampshire 03874

Mr. John DeVincentis, Director
Engineering and Licensing
Yankee Atomic Electric Company
1671 Worchester Road
Framingham, Massachusetts 01701

Mr. A. M. Ebner, Project Manager
United Engineers & Constructors
30 South 17th Street
Post Office Box 8223
Philadelphia, Pennsylvania 19101

Carol S. Sneider, Esq.
Office of the Assistant Attorney General
Environmental Protection Division
One Ashburton Place
Boston, Massachusetts 02108

D. Pierre G. Cameron, Jr., Esq.
General Counsel
Public Service Company of New Hampshire
Post Office Box 330
Manchester, New Hampshire 03105

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

cc:
Mr. Calvin A. Canney, City Manager
City Hall
126 Daniel Street
Portsmouth, New Hampshire 03801

Ms. Letty Hett
Town of Brentwood
RFD Dalton Road
Brentwood, New Hampshire 03833

Ms. Roberta C. Pevear
Town of Hampton Falls, New Hampshire
Drinkwater Road
Hampton Falls, New Hampshire 03844

Ms. Sandra Gavutis
Town of Kensington, New Hampshire
RDF 1
East Kingston, New Hampshire 03827

Chairman, Board of Selectmen
RFD 2
South Hampton, New Hampshire 03827

Mr. Angie Machiros, Chairman
Board of Selectmen
for the Town of Newbury
Newbury, Massachusetts 01950

Ms. Cashman, Chairman
Board of Selectmen
Town of Amesbury
Town Hall
Amesbury, Massachusetts 01913

Honorable Peter J. Matthews
Mayor, City of Newburyport
Office of the Mayor
City Hall
Newburyport, Massachusetts 01950

Mr. Donald E. Chick, Town Manager
Town of Exeter
10 Front Street
Exeter, New Hampshire 03823

Mr. Richard Strome, Director
State Civil Defense Agency
State Office Park South
107 Pleasant Street
Concord, New Hampshire 03301

Mr. Alfred V. Sargent,
Chairman
Board of Selectmen
Town of Salisbury, MA 01950

Senator Gordon J. Humphrey
ATTN: Tom Burack
U.S. Senate
Washington, D.C. 20510

Mr. Owen B. Durgin, Chairman
Durham Board of Selectmen
Town of Durham
Durham, New Hampshire 03824

Charles Cross, Esq.
Shaines, Mardrigan and
McEaschern
25 Maplewood Avenue
Post Office Box 366
Portsmouth, New Hampshire 03801

Mr. Guy Chichester, Chaiman
Rye Nuclear Intervention
Committee
c/o Rye Town Hall
10 Central Road
Rye, New Hampshire 03870

Jane Spector
Federal Energy Regulatory
Commission
825 North Capital Street, NE
Room 8105
Washington, D. C. 20426

Mr. R. Sweeney
New Hampshire Yankee Division
Public Service of New Hampshire
Company
7910 Woodmont Avenue
Bethesda, Maryland 20814

Mr. William B. Derrickson
Senior Vice President
Public Service Company of
New Hampshire
Post Office Box 700, Route 1
Seabrook, New Hampshire 03874

Mr. Robert J. Harrison
Public Service Company of New Hampshire

Seabrook Nuclear Power Station
(other)

cc:
Governor of New Hampshire
State House
Concord, New Hampshire 03301

Massachusetts Department of
Environmental Quality Engineering
100 Cambridge Street
Boston, Massachusetts 02108

State Planning Officer
Executive Department
State of Main
189 State Street
Augusta, Maine 04330

Chairman
New Hampshire Public Utilities
Commission
100 Cambridge Street
Boston, Massachusetts 02202

Federal Energy Regulatory Commission
825 North Capitol Street, NE
Washington, D.C. 20426

Office of the First Selectman
Town of Seabrook
Seabrook, New Hampshire 03874

Chairman
Vermont Public Service Board
120 State
State Office Building
Montpelier, Vermont 05502

Public Service Board
State of Vermont
120 State Street
Montpelier, Vermont 05602

Director, Criteria and Standards (SSERS)
(ANR-460)
Office of Radiation Programs
U.S. Environmental Protection Agency
Washington, D.C. 20460

Director, Eastern Environmental
Radiation Facility (SSERS)
U.S. Environmental Protection Agency
Post Office Box 3009

EIS Review Coordinator
Environmental Protection Agency
Region I
JFK Federal Building
Boston, Massachusetts 02203

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BNL
AUI

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.

Upton, Long Island, New York 11973

Department of Nuclear Energy

(516) 282
FTS 666 2630

December 5, 1986

Mr. G. Bagchi
Engineering Branch
Division of PWR Licensing - A
U.S. Nuclear Regulatory Commission
Mail Stop 316
Nicholson Lane Buildig
5650 Nicholson Lane
Rockville, MD 20852

Dear Mr. Bagchi:

Re: FIN A-3852

Please find enclosed two draft copies of a BNL technical report for NRC review, entitled "Technical Evaluation of the EPZ Sensitivity Study for Seabrook." Note that this draft is currently incomplete and a revised draft will be sent in the near future.

If you have any questions on the draft, please call us.

Very truly yours,

W T Pratt   C Hofmayer csc

W. T. Pratt
Safety & Risk
Evaluation Division

and

C. Hofmayer
Structural Analysis
Division

WTP/CH:csc

Enclosure

cc: W. Y. Kato (w/o enclosure)

8612170301XA 1p

TECHNICAL REPORT
A-3852

TECHNICAL EVALUATION OF THE EPZ SENSITIVITY STUDY FOR SEABROOK

DRAFT

December 5, 1986

Prepared by

Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

Prepared for

U.S. Nuclear Regulatory Commission
Washington, DC 20555
Under Contract No. DE-AC02-76CH00016
FIN A-3852

8612170316 XA

LIST OF CONTRIBUTORS

Contributor	Affiliation
K. Bandyopadhyay	Structures & Components Evaluation Group/SAD
P. Bezler	Civil & Structural Mechanics Group/SAD
G. Bozoki	Risk Evaluation Group/SRED
T-L. Chu	Risk Evaluation Group/SRED
M. Chun	Accident Analysis Group/SRED
C. Hofmayer	Structures & Components Evaluation Group/SAD
M. Khatib-Rahbar	Accident Analysis Group/SRED
B. Luckas	Engineering Analysis & Human Factors Group/ETD
J. Pires	Structures & Components Evaluation Group/SAD
W. T. Pratt	SRED
A. Tingle	Accident Analysis Group/SRED
R. Youngblood	Facilities Risk Analysis Group
P. C. Wang	Civil & Structural Mechanics Group/SAD

SRED = Safety and Risk Evaluation Division
ETD = Engineering Technology Division
SAD = Structural Analysis Division

ABSTRACT

A technical evaluation of the Seabrook Station Emergency Planning Sensitivity Study (PLG-0465) has been performed. This was an evaluation which focused on those areas found to be the most influential in calculating the Seabrook risk estimates. The approach taken by Brookhaven National Laboratory (BNL) was to perform sensitivity studies to assess the impact on the results in PLG-0465 of the BNL evaluation of these areas.

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PREFACE

This report describes a technical evaluation of the Seabrook Station Emergency Planning Sensitivity Study (PLG-0465). The main objective of this technical evaluation is to assist the NRC in its evaluation of the validity of the conclusions presented in PLG-0465. This is therefore a focused review by Brookhaven National Laboratory (BNL) of those areas identified in PLG-0465 as being the most influential in calculating the Seabrook risk estimates. However, regardless of the conclusions of this focused review, BNL cannot attest to the validity of the overall risk profiles presented in PLG-0465. This follows from the observation that the risk estimates in PLG-0465 rely heavily on Seabrook Station Risk Management and Emergency Planning Study (RMEPS) (PLG-0432), which in turn relies on the Seabrook Station Probabilistic Safety Assessment (SSPSA). Unfortunately, the risk profiles in the SSPSA have not been independently reassessed, requantified, and validated, by the NRC staff or their contractors. Similarly, within the scope of the review, BNL has also not validated the accident sequence probability estimates in the SSPSA. Therefore, because these estimates form the foundation for the updated risk estimates in the RMEPS and ultimately in PLG-0465, BNL has not, and cannot, verify the total risk estimates in PLG-0465. This includes the predicted dose versus distance curves. The current review should therefore be regarded as an evaluation of selected issues related to the potential for a large early release of radioactivity at the Seabrook Station and not a reassessment or validation of the total risk profile.

ACKNOWLEDGEMENTS

The authors wish to thank Dr. R. Bari in the Department of Nuclear Energy at Brookhaven National Laboratory for many discussions, comments, and suggestions related to this program.

This work was performed for the Division of PWR Licensing - A, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. The authors wish to acknowledge G. Bagchi, S. Long, and W. Lyon for their support and guidance throughout the course of this program.

Lastly, the authors acknowledge the efforts of C. Conrad, A. Costini, E. Gilbert, and D. Votruba in preparing this document for publication.

SUMMARY

This report describes a technical evaluation of the Seabrook Station Emergency Planning Sensitivity Study (PLG-0465). The results of the PLG-0465 Sensitivity Study are reproduced in Figures S.1-S.3. The principal conclusion of PLG-0465 was that an emergency planning zone (EPZ) at the Seabrook Station of 1 mile radius or less is more justified in terms of its risk management effectiveness than the current 10-mile EPZ was justified by the results of NUREG-0396. This conclusion was made in PLG-0465 without accounting for any new insights about source terms since WASH-1400. The conclusion was based on the following observations:

- The individual risk of early fatalities in the population within 1 mile of the site boundary with no immediate protective actions is less than the NRC safety goal (refer to Figure S.1). This individual risk is substantially less when a 1-mile evacuation is assumed.

- The risk of early fatalities with a 1-mile evacuation is comparable to the WASH-1400 results, which assumed a 25-mile evacuation (refer to Figure S.2). The Seabrook Station results for a 2-mile evacuation are substantially less than those for WASH-1400.

- The risk of radiological exposures for 1, 5, 50, and 200-rem whole body doses with no immediate protective actions is less at 1 mile than the corresponding NUREG-0396 results at 10 miles (refer to Figure S.3).

The Seabrook study (PLG-0465) identified the following three areas as being the most influential in calculating the Seabrook risk estimates:

- The effectiveness of the Seabrook Station primary containment to either remain intact or to maintain its fission product retention capability for periods much longer than required for even delayed, ad hoc protective actions.

- A more realistic assessment of the strength and failure modes of the Seabrook containment than was possible within the state-of-the-art of PRA when the RSS was completed.

- A more realistic treatment of the initiation and progression of interfacing systems LOCA sequences.

At the request of the NRC, the BNL technical evaluation focused on the areas that were identified in PLG-0465. The approach taken by BNL was to perform sensitivity studies to assess the impact on the results in PLG-0465 of the BNL review of these areas. The BNL sensitivity studies used the conditional risk indices provided in PLG-0465 (and supporting documentation) to assess how changes in the probability of accident sequences and containment failure modes would change the risk estimates in PLG-0465. The sensitivity studies calculated revised 200 rem-dose versus distance curves and the mean absolute risk of early fatalities and total cancer fatalities. The results of the BNL sensitivity studies are given in Table S.1 and Figure S.4. The dose vs distance curves in Figures S.3 and S.4 can of course be directly compared;

however, the BNL mean absolute risk numbers in Table S.1 cannot be directly compared (without additional calculation) with the information from PLG-0465 in Figures S.1 or S.2. The mean absolute risk number in Table S.1 would have to be converted into individual risk of fatalities in the population within 1 mile of the site boundary before direct comparison within information in Figure S.1 could be made. However, the information provided in Table S.1 is a useful indication of how the PLG-0465 risk levels in Figures S.1 and S.2 would change if recalculated using the assumptions of the sensitivity studies.

In addition to the areas identified in PLG-0465, other areas were identified as potentially important to risk at Seabrook. In particular, the applicant was requested to provide information on the risk associated with accidents during shutdown. The results of the applicant's assessment of such accidents are also given in Table S.1 and in Figures S.5 and S.6. BNL was not able to assess the frequency of these events because there remain fundamental questions regarding modeling of these scenarios. In addition, the potential for induced steam generator tube rupture (SGTR) for accidents in which the primary system is at high pressure was identified as a topic for review. This topic was reviewed in detail by the NRC staff and is the subject of continuing NRC and industry research activities. In an effort to assess the potential influence of SGTR, a simple sensitivity study was performed at BNL and the results are also given in Table S.1 and Figure S.4.

In Table S.1 and Figure S.7 the effect on risk of combining all the sensitivity is presented. This calculation should not be interpreted as a reassessment of the overall risk profiles for Seabrook because there was not a systematic attempt by BNL to obtain completeness. It is simply intended to indicate how the results of the various sensitivity studies could influence the risk estimates in PLG-0465. The method used to combine the effect of all of the studies is not rigorous and could lead to inconsistencies. In addition, it is not normal practice in probabilistic risk assessments to combine bounding sensitivity studies. The results in Table S.1 and Figure S.7 should therefore be recognized for what they are, namely, a series of sensitivity studies and not be interpreted as a statement of the overall risk at Seabrook.

The results in Table S.1 and Figures S.4-S.7 are useful to focus on those areas of the BNL review that appear to have the greatest impact on the conclusions in PLG-0465. The results indicate, given the extent of this focused review, that the conservative assumptions regarding accidents during shutdown and induced SGTR have the most impact on the dose vs distance and risk estimates in PLG-0465. However, the more optimistic assumptions regarding these events have minor impact on the PLG-0465 results. These are areas of considerable uncertainty, which at the present time do not allow a better definition of the risk estimates than given by the ranges in Table S.1 and Figures S.4-S.7

The conclusion of this focused review is that the risk estimates quoted from PLG-0465 at the beginning of this summary do appear to be influenced by the various sensitivity studies performed at BNL. The individual risk of early fatalities with no immediate protective actions as predicted in PLG-0465 are only slightly below the proposed NRC safety goals as shown in Figure S.1. Therefore, from an inspection of Table S.1 it is clear that some of the sensitivity studies will result in risk estimates (if no immediate protective actions are taken) that approach and may exceed the ~~proposed~~ NR Safety goal. However, with a 1 mile evacuation none of the selected sensitivity studies exceeded the safety goal. In addition, the statement in PLG-0465 that "there is no significant frequency of exceeding 200 rem beyond 1.5 miles in the Seabrook sensitivity results" is not confirmed by some of the sensitivity study results presented in Figures S.4-S.7.

Whether or not the results of the BNL (and applicant) sensitivity studies and the fact that BNL cannot verify the total risk estimates at Seabrook challenge the principal conclusion of PLG-0465, namely "that an emergency planning zone (EPZ) at the Seabrook Station of 1 mile radius or less is more justified in terms of its risk management effectiveness than the current 10-mile EPZ was justified by the results of NUREG-0396" is a matter of NRC policy and beyond the scope of the BNL review.

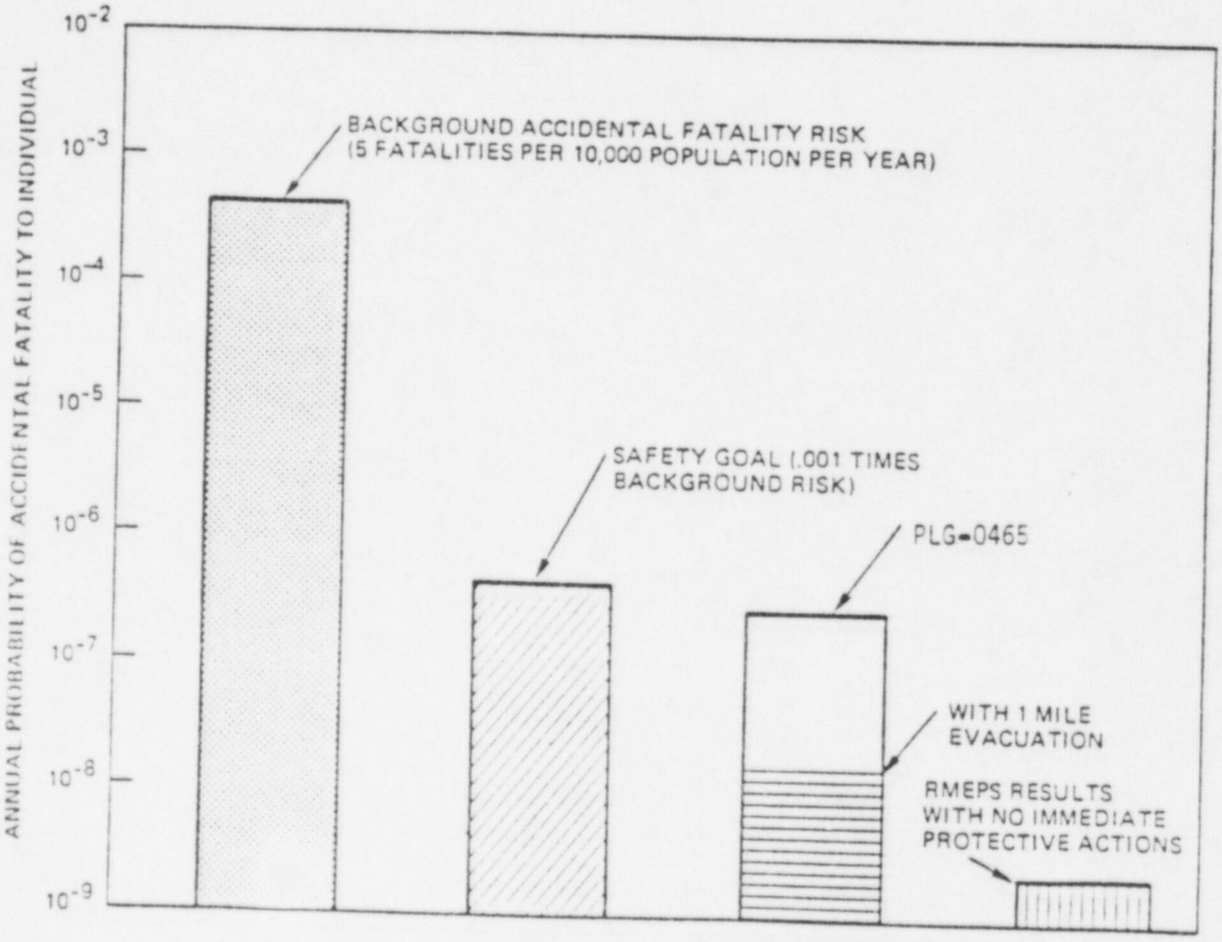


Figure S.1 Comparison of Seabrook Station sensitivity results using WASH-1400 source term methodology with background, safety goal individual and RMEPS risk levels.
(Reproduced from PLG-0465, April 1986)

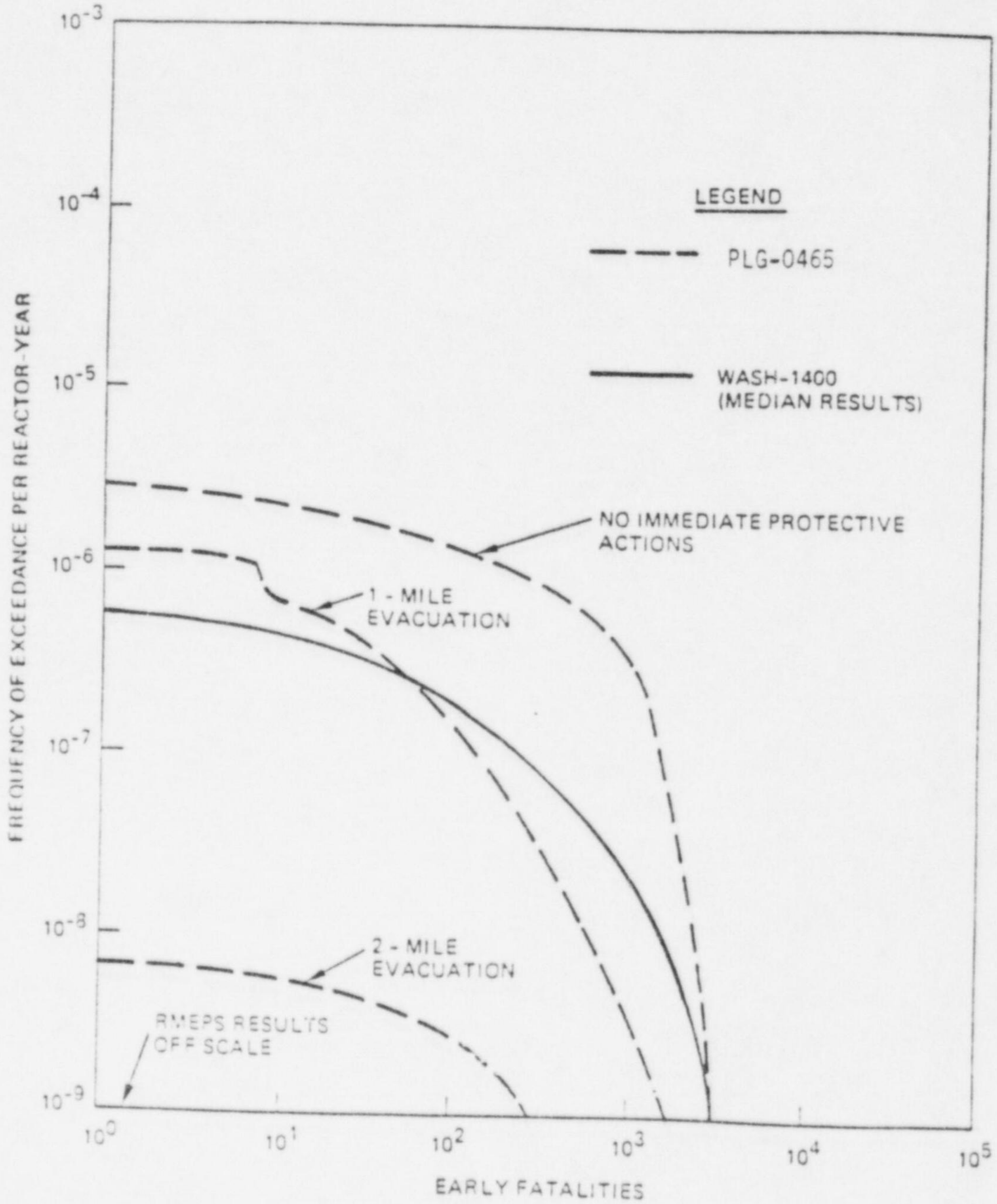


Figure S.2 Comparison of median risk of early fatalities at Seabrook station for different emergency planning options. (Reproduced from PLG-0465, April 1986)

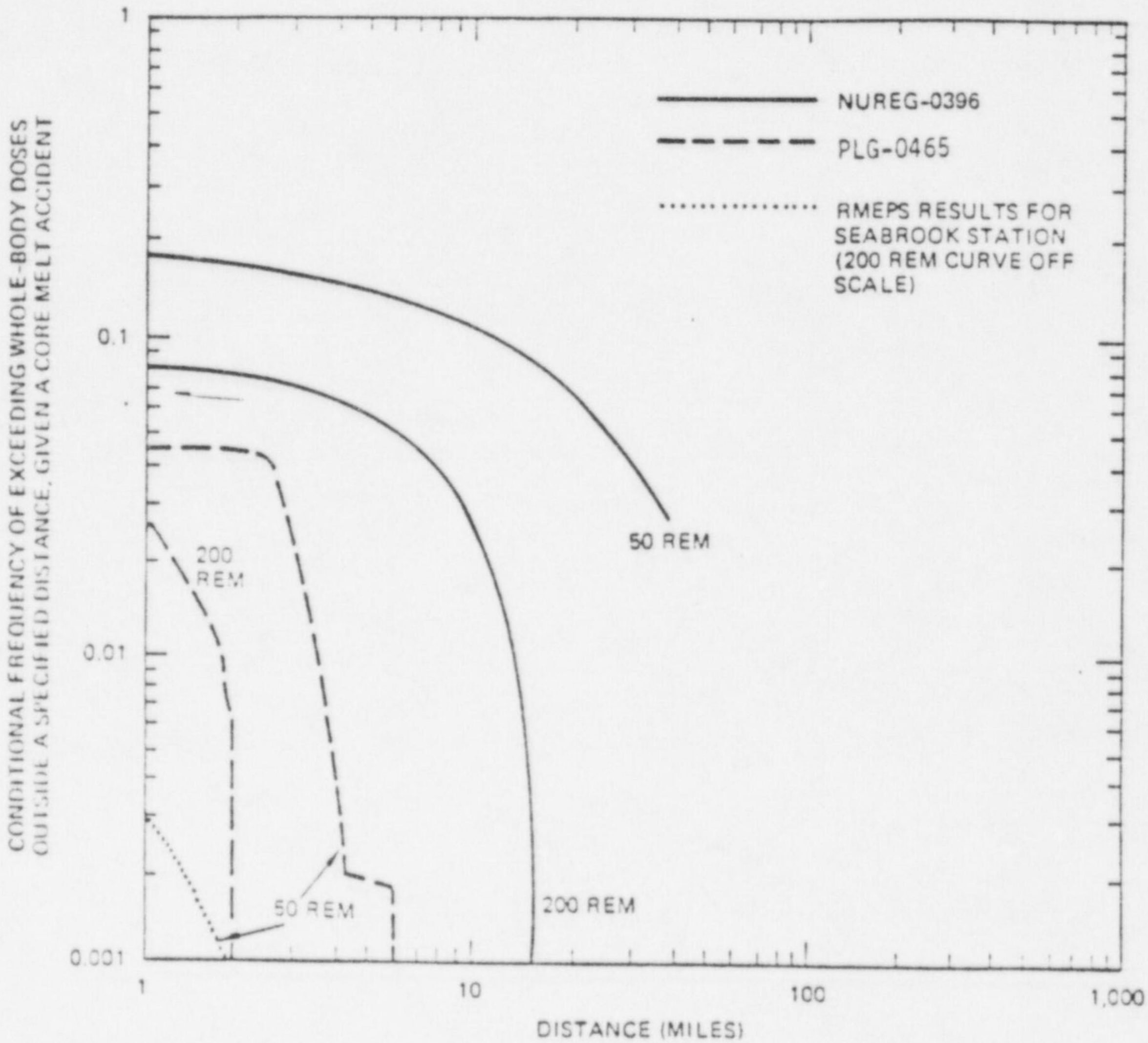


Figure S.3 Comparison of Seabrook Station results in this study and RMEPS with NUREG-0396 - 200-rem and 50-rem whole body dose plots for no immediate protective actions. (Reproduced from PLG-0465, April 1986)

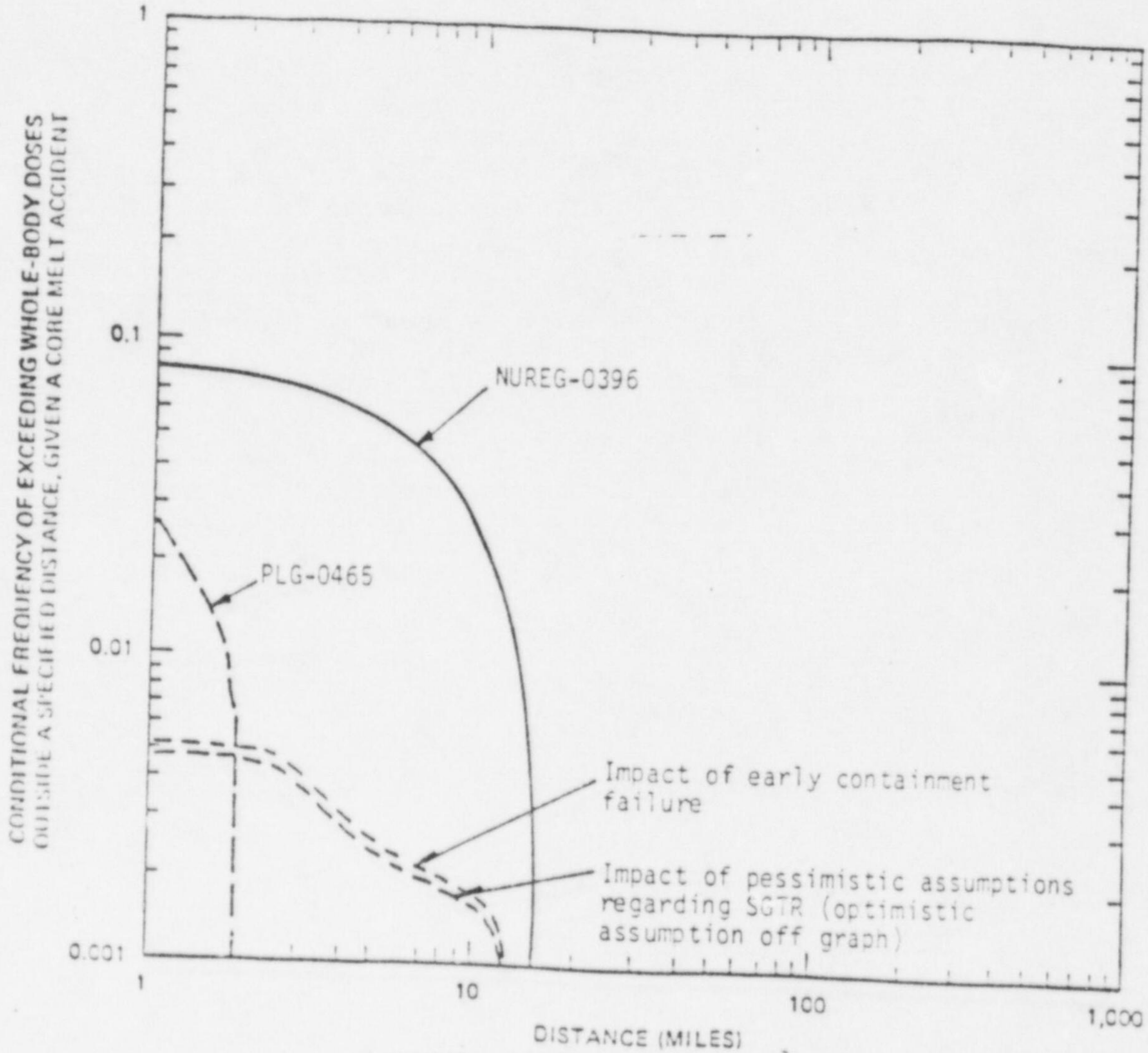


Figure S.4 Comparison of BNL sensitivity studies with PLG-0465 and NUREG-0396. (200-rem plots with no immediate protective actions)

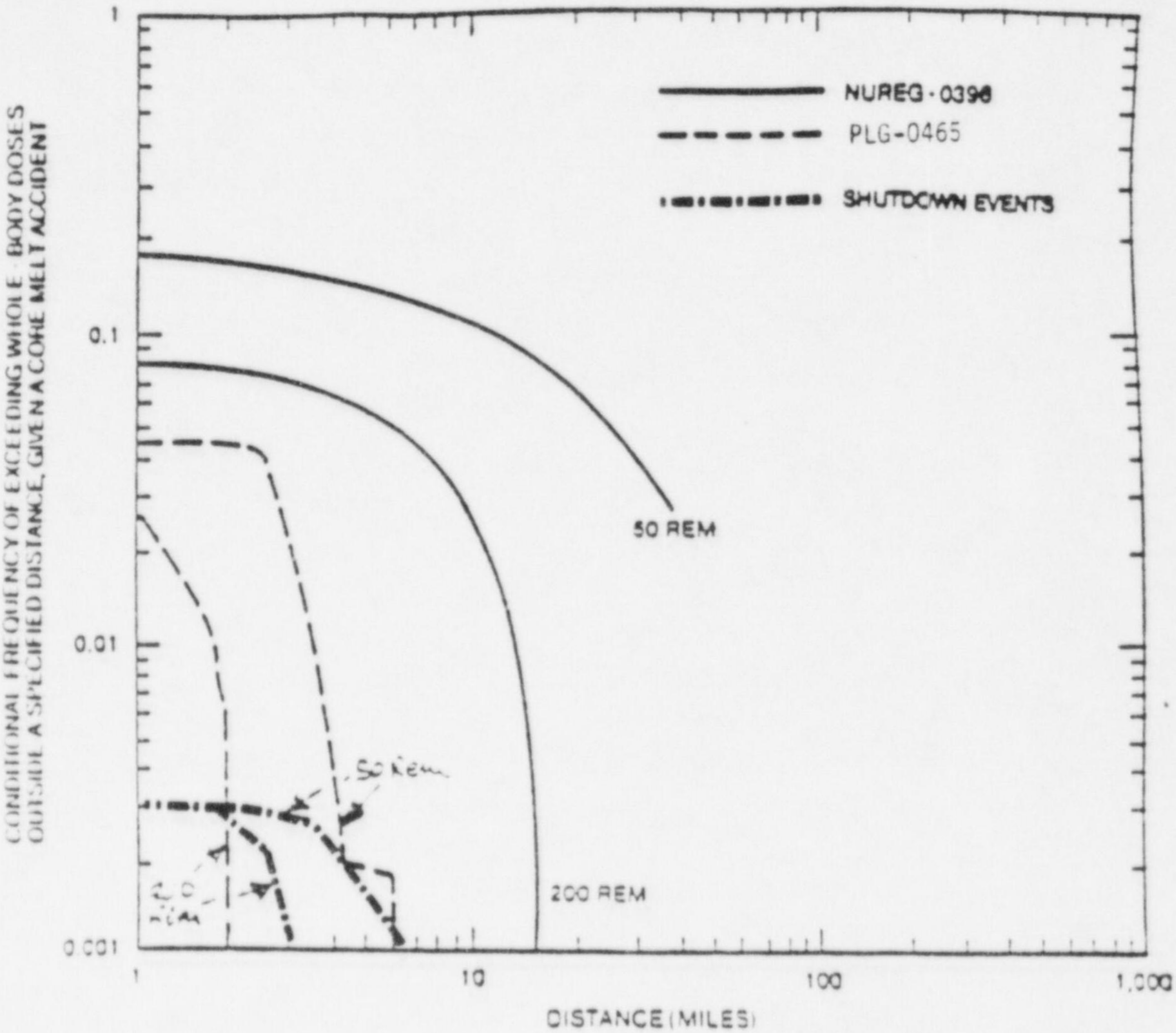


Figure S.5 Comparison of 200 rem and 50 rem dose versus distance curves with contributions from shutdown events. (Calculations performed by applicant)

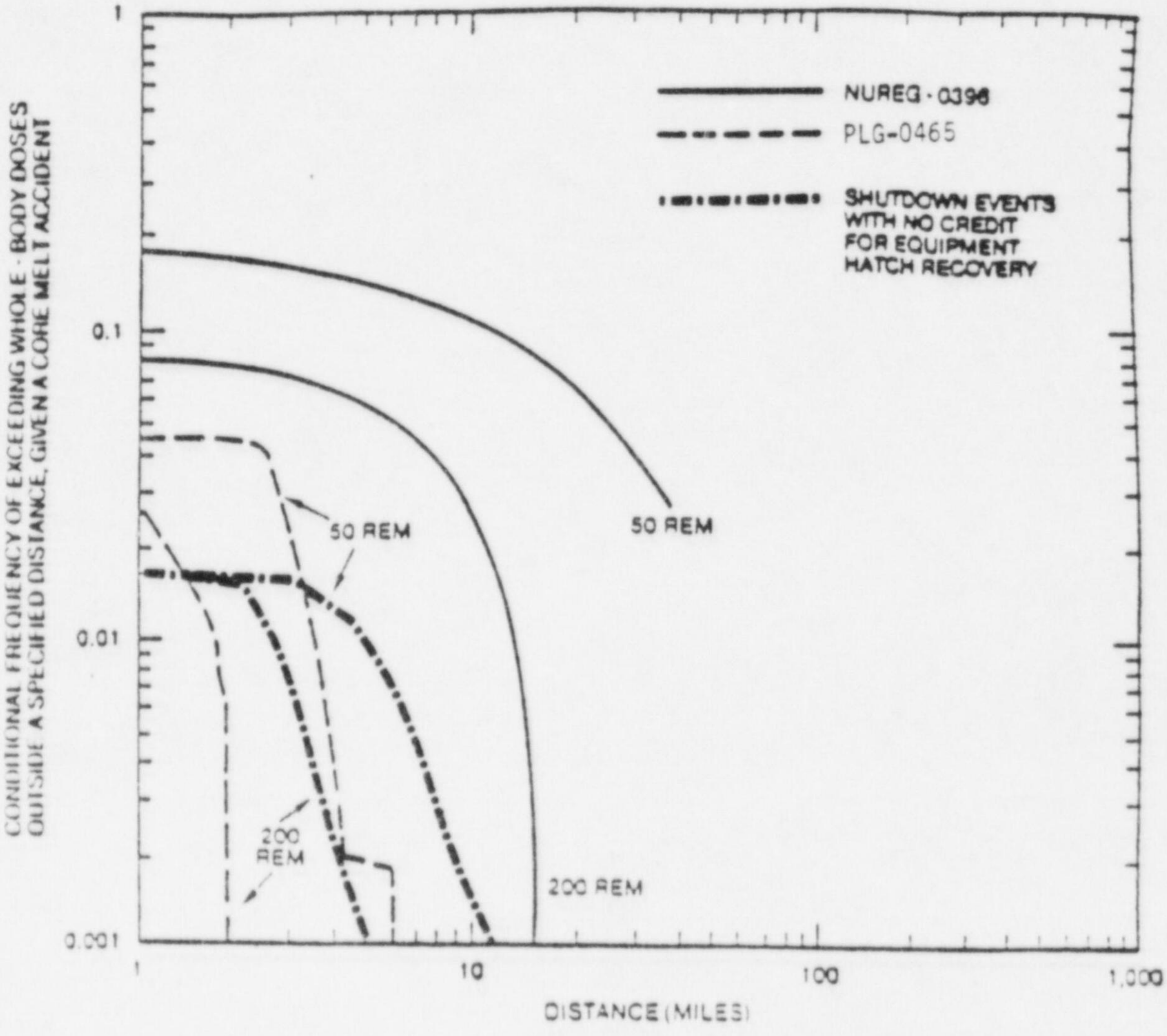


Figure 5.6 Comparison of 200 rem and 50 rem dose versus distance curves for conservative assumption of no credit for operator recovery of open equipment hatch. (Calculations performed by applicant)

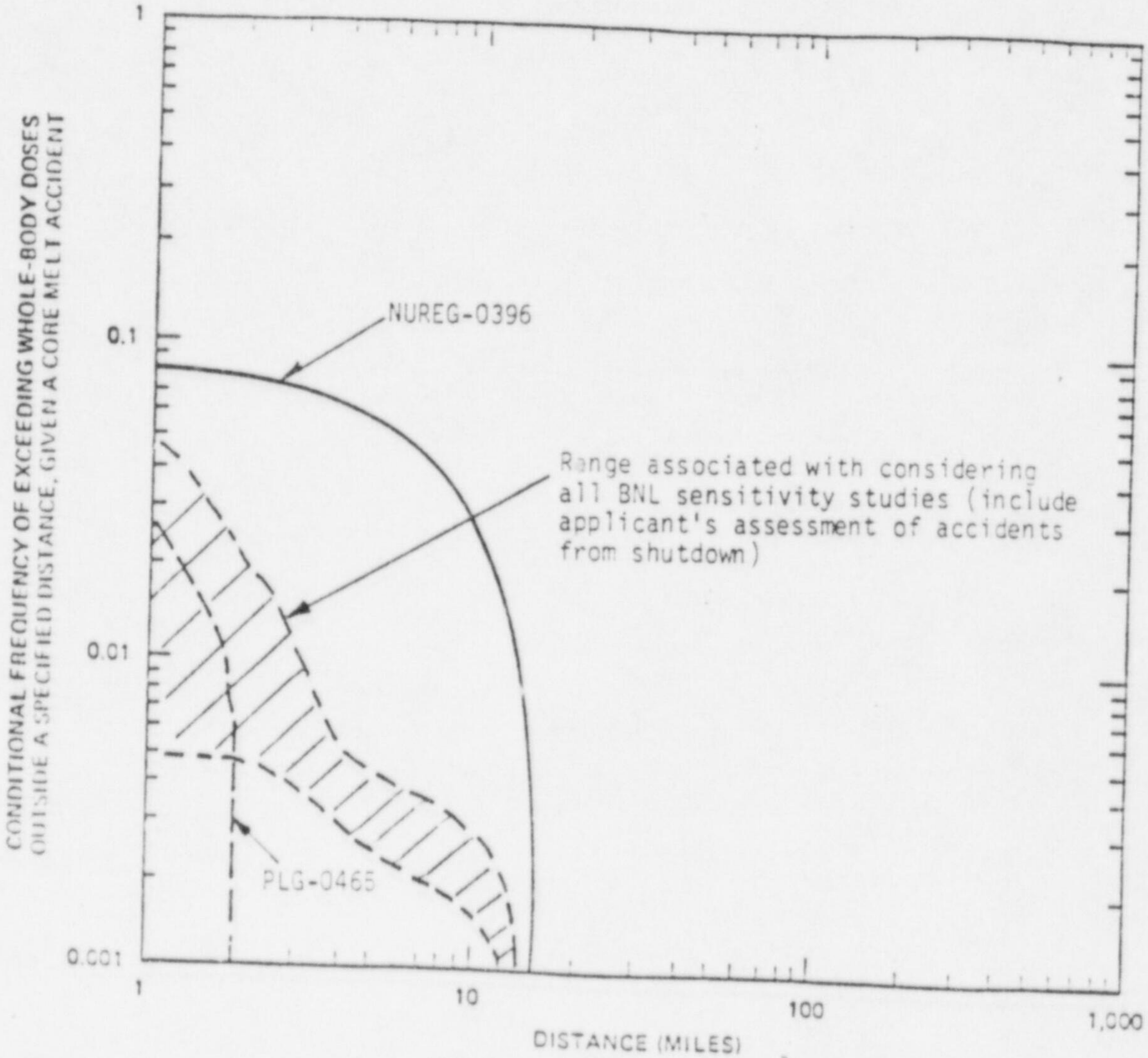


Figure S.7 Comparison of all BNL sensitivity studies with PLG-0465 and NUREG-0396 (200-rem plots with no immediate protective actions).

Table S.1 Impact of BNL Sensitivity Studies on PLG-0465 Risk Estimates

Sensitivity Study		Absolute Risk Per Reactor Year: Early Fatalities, No Evac.	Early Fatalities, One Mile Evac.	Early Fatalities, Two Mile Evac.	Total Cancers, No. Evac.	Total Cancers, One Mile Evac.	Total Cancers, Two Mile Evac.
Original PLG-0465 Results		2.7(-3)*	3.6(-4)	2.4(-5)	1.5(-2)	1.4(-2)	9.2(-3)
Revised Frequency for Inter-facing System LOCAs		2.8(-3)	4.6(-4)	1.2(-4)	1.5(-2)	1.4(-2)	9.3(-3)
Steam Generator Tube Ruptured	**	4.5(-3)	2.2(-3)	1.8(-3)	1.7(-2)	1.6(-2)	1.1(-2)
	***	2.8(-3)	4.2(-4)	8.4(-4)	1.5(-2)	1.4(-2)	9.3(-3)
Containment Loads and Performance		4.7(-3)	2.4(-3)	2.0(-3)	1.8(-2)	1.7(-2)	1.2(-2)
Accidents from Shut-down****	**	6.2(-3)	2.9(-3)	3.0(-4)	2.1(-2)	2.0(-2)	1.4(-2)
	***	2.7(-3)	3.8(-4)	2.8(-5)	1.5(-2)	1.4(-2)	9.2(-3)
Impact of all Issues	**	1.0(-2)	7.0(-3)	4.2(-3)	2.6(-2)	2.5(-2)	1.9(-2)
	***	4.9(-3)	2.6(-3)	2.2(-3)	1.8(-2)	1.7(-2)	1.2(-2)

*2.7(-3) = 2.7×10^{-3}
**Pessimistic assumptions.
***Optimistic assumptions
****Calculated by the Applicant, not confirmed by BNL.

1. INTRODUCTION

1.1 Background

The Seabrook Station Probabilistic Safety Assessment (SSPSA)[1] was completed by Pickard, Lowe, and Garrick (PLG) Inc., for the Public Service Company of New Hampshire and Yankee Atomic Electric Company in December 1983 and submitted to the Nuclear Regulatory Commission (NRC). The NRC staff and its supporting contractor initiated an in-depth review of sections of the SSPSA related to determining those accident sequences that could lead to core damage. However, this effort was terminated prior to completion of the review.[2] A separate contract was placed with Brookhaven National Laboratory (BNL) to perform a very limited[3] review of those portions of the SSPSA related to core meltdown phenomenology, containment response, and radiological source terms. The BNL review[3] did not include an assessment of the physical strength of the Seabrook Containment.

The key results of the SSPSA[1] are given below:

- The mean and median values of the uncertainty distribution for core melt frequency were found to be 2.3×10^{-4} and 1.9×10^{-4} events per reactor-year, respectively.

- Both the societal and individual risk provisions of the NRC safety goals were met by wide margins; hence, the risk to public health and safety was estimated to be extremely small.

- Different risk factors were found to have different key contributors. Interfacing systems LOCA events and, to a lesser extent, seismic-induced transient events were the principal contributors to early health risk. The contributors to core melt frequency and latent health risk were made up of a large group of initiators, including loss of offsite power, transient events, fires, and seismic events.

- The dominant contributors to core melt frequency were support system faults, external events, and internal hazards that affected both the core cooling and containment heat removal systems. As a result, a major fraction of the core melt frequency, 73%, was associated with sequences in which long-term containment overpressurization was indicated, while only 1% was associated with early containment failure.

- In contrast with previous containment analysis, the timing of containment overpressurization in the above sequences was found to be measured in units of days rather than hours.

A major result of the SSPSA was that interfacing system LOCA events were the principal contributors to early health risk. The results of the SSPSA were updated in the Seabrook Station Risk Management and Emergency Planning Study (RMEPS), PLG-0432" to account for new insights regarding radioactive release source terms and the progression of sequences involving loss of coolant events that bypass the containment.

The purpose of the RMEPS was to present the results of a technical evaluation of emergency planning options and other risk management actions that were under consideration for the Seabrook Station. The principal focus of the study was the evaluation of the impact of various protective actions such as evacuation and sheltering to various radial distances from the plant site.

A second report related to emergency planning was published (Seabrook Station Emergency Planning Sensitivity Study, PLG-0465) which determined the radius of the EPZ that could be justified without consideration of any advances regarding the source term methodology since the completion of the Reactor Safety Study in 1975. It is this second study that is the focus of the current BNL review.

1.2 Scope and Focus of the Review

The objective of this technical evaluation report is to assist the NRC in evaluating the robustness of the applicant's conclusions regarding the Emergency Planning Sensitivity Study for Seabrook (PLG-0465). The focus of the BNL review reflects those areas of PLG-0465 (and the supporting document PLG-0432) where major risk reductions (when compared with the results of the SSPSA) were calculated. Thus, our review assessed the physical strength of the Seabrook containment and the magnitude of the challenges to it. In addition, the potential for bypassing the containment via interfacing system LOCAs or by loss of containment isolation was assessed.

An important comparison in PLG-0465 relates to a comparison of the dose and distance risk curves developed for Seabrook with similar curves in NUREG-0396. Thus, the technical bases for the risk curves in NUREG-0396 will be reviewed and compared with the bases for the curves presented in PLG-0465 to determine the appropriateness of the comparison. Finally, the site consequence modeling in PLG-0465 will be evaluated and compared with current NRC consequence models.

1.3 Organization of the Report

The previous section identified the focus of our review of PLG-0465 and indicated the limitations of the effort. The report is organized to address each of the areas discussed in Section 1.2. Initially, in Section 2, those portions of PLG-0465 and the RMEPS (PLG-0432) related to system failure are reviewed to determine the appropriateness of the accident sequence probabilities.

Section 3 reviews the ability of the Seabrook Station primary containment to withstand the very severe pressure/temperature loads associated with core meltdown accidents. This is a very important review because the applicant considers that the Seabrook containment has a significantly greater capability for containing core meltdown accidents than a number of other large dry containments that have been reviewed by the NRC staff over the last several years.

The sensitivity of the conclusions in PLG-0432 to uncertainties in containment loads (pressure/temperatures histories) and containment performance (based on the review in Section 3) is explored in Section 4. The source terms used in PLG-0432, which were based on RSS methodology, are reviewed in

Section 5. Finally, in Section 6, the site consequence model and the risk calculations presented in PLG-0432 are reviewed.

1.4 References

1. SSPSA.
2. LLNL Review Draft.
3. NUREG/CR-4540.

2. SYSTEM EVALUATION

In this section, those positions of PLG-0465[1] and the RMEPS[2] (PLG--0432) related to system failure are reviewed to determine the appro-priateness of the accident sequence probabilities. This review focused on the reduced frequency of the interfacing system LOCA claimed in PLG-0432. This section also addresses other sequences that might lead to early containment bypass and hence the potential for a large early release of radioactivity. This is a particularly important topic for review because the applicant's risk estimates are based upon the conclusion that essentially all accidents that may lead to a large early release of radioactivity have a very low probability (less than one chance in ten million per year of reactor operation).

At NRC direction, since the present concern is with the frequency of a substantial release, this section addresses several selected areas which are already known to bear directly on this question. The strength of containment is addressed in a later section; here, selected modes of containment bypass are discussed. These include classical interfacing systems LOCA, failure of containment isolation, and breakdown of steam generator tube integrity during a severe accident.

The study generally creates a strong impression of completeness. However, the present decision logic imposes unusual demands on the analysis. When one is quantifying a multiple passive failure event whose frequency and consequences are both clearly dominated by events whose absolute likelihood has been established experientially, some latitude can be tolerated in the quantification, and neglect of the less likely contributors can be justified. Here, the decision process requires that certain categories of events be shown to be essentially incredible on an absolute scale; there can be no dominating events for them to hide behind.

Several types of questions arise:

1) whether the basic assumptions of the submittal are adequate, and whether the analysis confronts all the important issues

2) whether, given the basic assumptions, the plant-specific modeling is self-consistent and complete within the range of issues addressed by the modeling
3) whether the scenarios actually modelled are properly quantified (including common cause considerations).

An example of a Type 1 completeness question will be discussed here in Section 2.3, namely, whether a high-pressure melt scenario can lead to steam generator tube degradation and concomitant containment bypass. This is a fairly generic question, and one that is not answered within the material submitted to date. This question is presently being studied by the utility and its consultants.

An example of a Type 2 completeness question is whether the check valve between residual heat removal (RHR) suction and the refueling water storage tank (RWST) is likely to fail in scenarios wherein the RHR system is overpressurized. The study raises and discusses this question, but without clearly establishing why the check valve survival probability is high enough to warrant not exploring such scenarios. We believe that this information will be forthcoming, but while this particular question may be laid to rest, it is emblematic of a family of such questions which cannot be exhaustively tallied within the scope of this limited review, but which bears on the submittal's conclusions.

Finally, a particular instance of a type 3 question will be discussed below in Section 2.1, where the frequency of multiple check valve failure will be addressed.

2.1 Interfacing System LOCA

2.1.1 General

According to the Seabrook Station Emergency Planning Sensitivity Study,[1] one of the principal contributors dominating early health risk--and one which has been subjected to extensive reanalysis since the SSPSA--is an Interfacing Systems LOCA that bypasses containment. From all the potential pathways through which an Interfacing Systems LOCA (ISL) may occur, the study identified six lines as possible initiators for ISLs:

- Four lines in the cold leg safety injection (Low Pressure Injection [LPI]/Residual Heat Removal [RHR] Loop Return lines)

- Two lines in the suction side of the RHR system.

The corresponding initiator frequencies were determined as:

Cold Leg Safety Injection Path (VI):	4.5×10^{-6} event/reactor year
RHR Suction Side Path (VS):	3.3×10^{-6} event/reactor year

which resulted in the following core damage frequency:

$CD_{ISL} = 4.4\times10^{-8}$ event/reactor year.

These frequencies were obtained as a result of an enhanced and innovative analysis which involved new treatments of various aspects of the accident. The new treatments are listed below:

- More complete modeling of valve failure modes
- New data on check valve failures versus leak size
- More realistic treatment of dynamic pressure pulse
- Explicit modeling of RHR relief valves
- Quantification of RHR piping fragilities to overpressure
- Modeling of RHR pump seal leakage
- Operator actions to prevent melt considered
- Thermal hydraulic and source term factors modeled using MAAP[4]
- Uncertainties quantified.

BNL has performed a limited review of the new analysis, compiled a number of questions and observations, and performed a new evaluation of the initiator frequencies. The objective of this chapter is to provide the results of this limited review.

2.1.2 Other ISL Paths

BNL performed a cursory line survey for potential containment bypassing pathways for ISL. The survey identified some pathways (e.g., RHR lines to the RCS hot legs, letdown line, excess letdown line, etc.) which were ignored in the Seabrook study. Since BNL believes there were good reasons to ignore them, these pathways were not analyzed. However, BNL believes that all the potential paths should have been considered and the reasons for the rejection of each path, with the probability of the path being open, should have been discussed.

2.1.3 Initiator Frequencies

The determination of the initiator frequencies is one of the most important parts of the Seabrook EPZ Study.[1] It depends essentially on the correct estimate of the frequencies of relevant failure modes of valves in various interfacing lines. These valve failure modes are:

- Disc rupture or gross leakage of series valves (check valves) in the LPI lines
- Disc rupture or gross leakage of MOVs, failure of stem mounted limit switches, and disc failing open when indicating closed, in the RHR suction lines.

The approach applied for modeling of initiator frequencies in the Seabrook study is based on two "innovative" steps:

a) Separation of the gross (reverse) leakage of check valves failure mode into "gross reverse leakage" and "failure to reseat on demand" failure modes, which were treated together in earlier data bases. This step required, of course, a "remodeling" of the appropriate Boolean equations.
b) An analysis of data on check valve leakage frequency versus leak rate for check valves of the RCS/ECCS system boundary. This step resulted in applying a reduced check valve leakage failure frequency in the quantification of the initiator models.

In the process of surveying data of the Nuclear Power[6] Experience (NPE) data base, no disc rupture events were identified by PLG for check valves and MOVs. The maximum observed leak rate was 200 gpm. Leak rates were estimated based on other available evidence: the rate of boron concentration change in the accumulators, pressure reduction, and similarity to other occurrences for which the leak rates were known. To estimate the total check valve hours, the information provided in NUREG/CR-1363[5] on the number of valves in the ECCS in various PWRs was used. PLG's total number of check valve-hours was approximately $1.0x10^8$. To estimate the frequency of check valve failure to reseat on demand, two types of data were used: estimates from generic sources of failure data, and experiential data from eight U.S. nuclear plants for which PLG performed plant specific PRAs.

2.1.3.1 Check valve failure frequencies

Since the check valve failure frequencies play a crucial role in the ISL analysis, BNL performed a somewhat more detailed review of that part of the Seabrook study. As a consequence of the review process the following observations are made:

a) PLG selected a particular subset of those events listed in the NPE data base, namely, events involving check valves at the RCS-ECCS interface.
b) To estimate the total number of check valve hours, it used the total population of check valves in the ECCS instead of the particular subset of check valves at the interfaces. This resulted in substantial overestimation of check valve hours.
c) The correct exposure time for check valve failures is not merely the time when the plant is operating. For example, check valves in the RHR are almost continuously exposed to potentially degrading conditions (during cold shutdowns, as well). A correction factor for pressure exposure of interfacing lines should be considered separately, in calculating the initiator frequencies.
d) When estimating leak rates for accumulator check valves from accumulator inleakages, it must be recognized that the deduced leak rates relate to two check valves in series, rather than lealage through a single check valve.

e) The leak failure frequencies versus leak rate curve presented in the study (reproduced in Figure 2.1) is only a first approximation for a more precise leak failure frequency versus relative leak rate curve. In particular, this curve pooled data involving a variety of check valve sizes. A more sophisticated treatment would require knowledge of the size population of check valves at the interfacing pathways.

f) The largest leak rate in Figure 2.1 is of the order of 200 gpm, whereas the arena of interest ranges to 65,000 gpm. The "linear" extrapolation to higher rates is not necessarily justified. If the shape of the distribution is Pareto, the linear extrapolation is in order. However, if it follows a Rayleigh distribution, the extrapolation is not correct (but conservative). Seabrook-specific considerations (valve sizes, designs) are not made in the analysis.

g) The initiator models implicitly assume that the leak tests of the valves "discover" all failures and valves behave as new after each test. The study does not describe the relevant test processes and the expected "real" efficiency of these tests.

h) The report does not consider common cause failures. Such failures indeed happen due to boron deposition, improper maintenance such as installation of improper components (gaskets, seats, or valve disks) which may fail almost immediately or at a later time.

In order to estimate quantitatively the consequences of some of the above mentioned deficiencies, BNL performed a more detailed reevaluation of relevant check valve failure data. The process was facilitated by the availability of relevant failure events selected for an independent study of ISL at PWRs, which is presently ongoing at BNL for the NRC.

Tables 2.1 and 2.2 present failure events for High Pressure/Low Pressure isolation check valves selected by BNL. These tables contain more relevant events than are listed in Table 3.8 of PLG-0432.

Table 2.1 contains events for the valve "leakage" failure mode. Table 2.2 presents the events for the "valve failure to close" failure mode. Table 2.1 shows also data on the estimated leak flow rates. These latter data are obtained essentially with the same method as those of Table 3.8 of PLG-0432.

Using the failure events related to the accumulator check valves only, BNL attempted to determine the leakage failure rate for a "clean" subset of check valves. The appropriate check valve population was relatively easy to determine for all the PWRs in the U.S. (see Table 2.3). The total time from start of commercial operation of the individual plants was used as "time of exposure" for this clean subset of check valves, since, e.g., water with boric acid degrades these valves. The total number of check valve-hours obtained is 2.34×10^7.

The frequencies of accumulator check valve leakage events for various leak rate ranges are given in Table 2.4. The corresponding frequency exceedance/hr values are plotted against the check valve leak rates in Figure 2.1. For comparison, Figure 2.1 shows also the PLG data. The shape of the curve is almost identical with that of PLG, but shifted higher, with almost one order of magnitude, due to the higher number of events selected and the more precise value for check valve-hours.

It is appropriate to mention here several precautions concerning the leakage failure characteristics derived from accumulator check valve failure events.

- In the NPE data base the majority of interfacing check valve leakage events involve accumulator valves. Although this seeming bias could arise from the extra monitoring of the accumulator, it could also reflect a particularly severe environment acting on the valves. If the latter is true, then leakage exceedance frequency data (ordinates in Figure 2.1) may lead to overestimates of the frequency for <u>other</u> interfacing check valves.
- The leak flow rate data ("leak sizes"; abscissas in Figure 2.1) represent lower limits for these quantities, because leakage flow rates estimated from accumulator inleakages involve, in most of the cases, leakage through <u>two</u> check valves in series, where the less-leaking valve dominates (the other valve may be even wide open).
- As a result of these factors, a more realistic leakage failure exceedance frequency/hr versus leak rate curve for non accumulator interfacing paths may be somewhat lower in frequency at low leak rates, but

might fall off more slowly with increasing leak rate than do the curves in Figure 2.1.

- This more realistic curve yet has to be determined.

Since there are no more accurate data available, BNL recalculated the initiator frequencies by using the data obtained for accumulator check valves. Since the purpose of this calculation is to contrast the result with that of the PLG analysis, the same extrapolation and calculational techniques are used as those of PLG.

2.1.3.2 Cold leg safety injection path frequency

This section presents a revised estimate of VI, the frequency of interfacing LOCA through the injection lines. The calculation presented below is intended to follow the PLG analysis step by step, except that the check valve failure statistics have been modified as indicated above. Subsequently, these modified initiator frequencies will be propagated through to illustrate new plant damage state frequencies. However, it should be noted that one problem noted above is not addressed by this modification: the fact that much check valve leakage experience actually corresponds to leakage through multiple check valves. For example, referring to Figure 2.1, if we take 10^{-8} per hour as the frequency of 1800 gpm leakage through multiple check valves, we obtain an annual frequency on the order of 10^{-4} per year (per line) for these rather sizable events. At 10^{-7} per hour for 150 gpm leakage, we would obtain 10^{-3} per year, which is near the threshold of observability. We believe that this treatment is simplistic and conservative, and accordingly have not propagated these numbers; but on the other hand, a proper treatment would ultimately have to confront the multiple leakage aspect of the accumulator experience, which neither PLG's analysis, nor the one presented below, accomplishes. The following is, then, a recalculation of the PLG result using PLG methods but modifying the single check valve failure rate as previously discussed.

From Figure 2.1, the median frequency of a single check valve failure resulting in leakage, that exceeds the capacity of one charging pump (i.e., 150 gpm) is about 1.1×10^{-7} per hour. Assuming a lognormal distribution for this frequency and a range factor of 10 (which may be too conservative for this increased statistic) yield:

Parameter	Frequency (events per reactor year)
95th percentile	9.6×10^{-3}
Mean	2.6×10^{-3}
Median	9.6×10^{-4}
5th percentile	9.6×10^{-5}

Similarly, the median frequency of exceeding 1800 gpm is 1.4×10^{-8} per hour. Assuming a lognormal distribution with a range factor of 14 yields:

Parameter	Frequency (events per reactor year)
95th percentile	1.7×10^{-3}
Mean	4.4×10^{-4}
Median	1.2×10^{-4}
5th percentile	8.8×10^{-6}

The frequency of "fail to operate on demand" for check valve, λ_d is taken to be the same value as that used by PLG.

$$\lambda_d = 2.7 \times 10^{-4} \text{ (mean value).}$$

By using Formula 3.14 of PLG-0432, the estimated mean frequency of failure of two series injection check valves, that produces leakage to the RHR system in excess of 150 gpm is:

$$4.90 \times 10^{-5} \text{ events/reactor-year.}$$

Since there are four injection paths, the mean value for Cold Leg Safety Injection Path,

$$VI = 1.96 \times 10^{-4} \text{ events/reactor-year.}$$

Top event, LR in the injection path event tree represents the fraction of the initiating event frequency, VI, in which the leakage not only exceeds 150 gpm, but also exceeds 1800 gpm. The product of LR and VI thus represents the frequency of pressure challenges to the RHR system due to failure of both check valves in the four injection paths. Based on the above values, LR has a mean value of .058.

2.1.3.3 RHR suction side frequency

For an ISL to occur in the RHR hot leg suction path, failure of two series MOVs must occur. In the PLG-model for this path, the failure involves:

a) independent failures of both MOV valves, causing excessive leakage; or
b) independent failure of one of the valves and a demand failure of the second valve, or
c) "valve fail open while indicating closed" failure for the first value and excessive leakage failure of the second valve.

In the PLG treatment, the frequency of MOV valve disc leakage and failure upon demand (due to a sudden pressure loading) were assumed to be identical to that for the check valves. For the frequency of failure of an MOV to close on demand but indicate closed, a mean value of λ_d = 1.1×10^{-4} failure/demand was used in the PLG treatment.

Applying the same approach as PLG (Formula 3.15 of PLG-0432) with the newly determined check valve leakage frequency, BNL recalculated the total (2 lines) suction side ISL frequency, VS. The new mean frequency for the RHR suction side path is:

$$VS = 1.44\times10^{-4} \text{ events/reactor-year.}$$

The split fraction, LR, for the fraction of VS in which the leakage past the series MOVs is greater than the capacity of the relief valves, is practically the same as in the case of injection lines: .058.

For these check valve leak rates, the PLG procedure of using the check valve leakage failure rates as "conservative" estimates for the leakage failure rates of the MOVs is probably too conservative, and appropriate MOV leakage failure frequencies should be used.

In the case of a reanalysis of the initiator frequency, VS, there are several BNL observations to be taken into account in a new suction side ISL model:

a) Inadvertent opening of the two MOVs due to common cause failures such as improper maintenance, malfunction of the interlock system, design error, improper tests, or testing operations.
b) Failure of the stem or other internal connections in valves equipped with limit switches or failure of a limit switch (including improper maintenance such as reversing indication).
c) It is difficult to see why only two MOVs have limit switches, instead of four.
d) It would be very useful to describe the valve inspections that are promised each time the plant goes to cold shutdown or is refueled. For example, at a plant recently investigated by NRC, Region 1, everything was tested thoroughly, but the relays for the MOVs were not inspected.
e) Considerations should be given to operating procedures and the likelihood that the procedures will not be followed, and interlock behavior.

2.1.4 Operator Actions

The ability of the Seabrook operators to diagnose, respond to, and mitigate a Reactor Coolant System (RCS) to Residual Heat Removal (RHR) Interfacing Systems LOCA will be reviewed in this subsection. Appropriate operator actions can mitigate the consequences of the ISL sequence that result in leakage outside containment when the capacity of the RHR pump suction relief valves is exceeded and subsequent failure of the RHR pump seals results. As discussed by PLG-0432, the success of these mitigative actions is dependent on the ability of the Seabrook operating staff based on their training and emergency procedures, to correctly diagnose a LOCA outside containment. The correct diagnosis may be hampered by operator confusion between symptoms associated with those LOCAs inside containment which fill and pressurize the Pressurizer Relief Tank (PRT) by pressurizer relief or safety valve discharge flow and those associated with a RCS-RHR Interfacing Systems LOCA outside containment which also fills and pressurizes the PRT via the RHR suction relief valve discharge flow.

The following is the result of a brief BNL evaluation of operator diagnosis and actions to mitigate the consequences of RHR pump seal failure Interfacing Systems LOCA sequences at Seabrook and its assessment in PLG-0432,[2] Section 3.1.4.3 entitled "Operator Actions and Emergency Procedures."

This evaluation was preceded by an independent and fairly extensive familiarization preparation with the Seabrook procedures as they relate to the ISL to be studied. This preparation was followed by observation of a series of Seabrook Simulator demonstrated accident sequences which illustrated the distinguishing characteristics of the LOCA outside containment and the responses expected of the Seabrook operators. The BNL evaluation was performed by a former Senior Licensed Operator and Westinghouse Reactor Plant simulator Certified Engineer. A more detailed and complete evaluation of operator response would require a comprehensive Human Reliability Analysis (HRA) such as Team Enhanced Evaluation Method (TEEM)[6] by a knowledgeable team of specialistis providing expertise in PWR operations, PWR systems engineering and human reliability. This team would develop a detailed task sequence analysis of the Seabrook operating staff performing the detailed tasks required to mitigate these sequences and analyze the associated human reliability of the staff response using the analysis.

There are three sets of operator tasks identified by PLG-0432[2] which are to be important to the mitigation of the sequences by the Seabrook operating staff (each with a unique Operator-Action Sequence identification number in parenthesis), namely:

(01) Diagnose the RHR system LOCA
(02) Isolate the RHR system LOCA
(03) Provide makeup to the RWST.

To successfully accomplish these tasks, the operating staff must follow the appropriate parts of the following Seabrook procedures which are applicable to the RHR system LOCA event.

- Procedure E-0 (Reactor Trip or Safety Injection), Rev. 00, dated 05/16/86.
- Procedure ECA-1.2 (LOCA Outside Containment), Rev. 00, dated 05/16/86. This procedure provides guidance on isolating the rupture.
- Procedure ECA-1.1 (Loss of Emergency Coolant Recirculation--ECR), Rev. 00, dated 05/16/86. This procedure provides guidance for supplying adequate ECCS flow and plant stabilization.
- Procedure E-1 (Loss of Reactor or Secondary Coolant), Rev. 00, dated 05/16/86. This procedure provides guidance for long-term cooling and stabilization.

Plea:e note that ECA-1.2, Rev 00 needs to be revised to ensure that valves RH-'21 and -V22, the RHR pump hot leg injection cross connection valves are closed prior to trying to identify and isolate a break in the low pressure systems. This need was identified by a detailed review of the above four procedures.

The quantification of the three operator tasks identified by the Operator Action Sequence identification numbers 01, 02, and 03 above have been provided in PLG-0432,[1] Table 3-10. According to the accompanying discussion in Section 3.1.4.3, "These operator actions include the hardware contribution, where applicable, and are based on enhanced procedures and instrumentation in order to aid the operators in their diagnosis of the event." For each of the three operator tasks, a "base," human error rate with a "mean" value of 0.005 has been identi- fied as "OP".

This singular human reliability analysis number is identified in PLG-0432[2] as ". . . recommended in Table 20-6 of NUREG/CR-1278[7] . . . for following a procedure under abnormal conditions. This human error rate is interpreted to have a mean value of 0.005 and to be represented by a lognormal distribution range factor of 10." Therefore, t e only part of the three sets of operator tasks 01, 02, and 03 which changes their quantifications values is the hardware contribution. The human reliability quantification contribution of each of three operator tasks use the same estimated Human Error Reliability (HEP) value of 0.005 with an error factor of 10. Each HEP is based on NUREG/CR-1278,[7] Table 20-6 (entitled "Estimated Human Error Probability (HEP) related to failure of administrative control"), Item (4) HEP entitled "Use written operations procedures under abnormal operating conditions." Therefore, no differentiation is made to distinguish quantitatively among operator actions related to "diagnose," to "isolate," and to "provide."

The October 15, 1986 demonstration at the Seabrook Simulator with the abovementioned Seabrook abnormal/emergency related procedures provided (in the absence of a detailed TEEM[6] equivalent human reliability analysis performed on an actual Seabrook licensed operator shift) a reasonable assurance that a licensed Seabrook crew would adequately perform the necessary actions within the time required on the simulator. This assurance is heightened especially since the Seabrook Training Center has recently instituted, in October 1986, a training module entitled "RHR Interface LOCA/Student Handout," as part of its Requalification Training Program. The inclusion of such a module will reinforce the importance of the RCS-RHR Interfacing Systems LOCA. Please note that this module confirms the need to revise Seabrook Procedure No. ECA-1.2, Revision No. 00, dated 5/16/86 entitled "LOCA Outside Containment" to close (or verify closed) valves RH-V21 and -V22, the RHR pump hot leg injection cross connection valves to identify and isolate a break in the low pressure systems.

Nevertheless, there were a number of concerns raised during a plant walkthrough on the same date as the simulator demonstration which the Seabrook Simulator cannot adequately answer. These concerns include the following:

a) Ability of RHR pump leakage to be detected in the control room - concern lies with vault compartmentation design with the Equipment Vault sump not receiving leakage promptly thereby delaying level detection input in the control room.

b) Ability of RHR pump relief discharge into the PRT to be distinguishable in the control room from the pressurizer relief and safety valve discharge - concern with the latter relief and safety valve discharge tailpipe temperatures.

In summary, the operator action analysis performed in PLG-0432,[1] Section 3.1.4.3 appears to be superficial at best. The use of one single HEP value from one table of NUREG-1278[7] is an example of a lack of detailed and insufficient task analysis in assessing human performance appropriately. A detailed TEEM[6] equivalent human reliability analysis is a far more appropriate and rigorous approach to assessing Seabrook operator actions during a RCS-RHR Interfacing systems LOCA. Nevertheless, the simulator demonstration emphasizing human reliability and task sequence timing, plus the new training module reinforcing a commitment to train the operator, provide a reasonable assurance that a licensed and trained Seabrook crew would adequately perform the necessary actions within the time frame required.

2.1.5 Break Location

The "weakest link" of the RHR pressure boundary when subjected to accidental pressurization was identified by the applicant to be the RHR pump seals. A tabular listing of failure probabilities at 2250 psia showing pump seal failure probabilities ranging to 0.5 while metallic failure probabilities (piping, valves, and tubing) were 0.006 seems to support this observation.

The estimates of metallic component failure probabilities were based on:

a) accidental pressurization peak pressure limited to the initial RCS pressure of 2250 psia.
b) a probability of failure at the yield strength of the material to be 0.01 and the probability of failure at the ultimate strength of the material to be 0.99.
c) the characterization of the overpressurization event as a quasi static process.
d) the statement that at 2250 psia, the stresses in the limiting RHP piping are only approaching yield stresses and the heat exchanger tube and other mechanical components are at a small fraction of their respective yield stresses.

The characterization of the overpressurization event as a quasi static process with a limiting peak pressure equal to the initial RCS pressure of 2250 is based on IDCOR evaluations which have not been reviewed by BNL. The assignment of 1% and 99% failure probabilities to the yield and ultimate strengths of the material respectively is acceptable since a failure at yield is considered unlikely while a failure at ultimate is considered very likely. The statement concerning the safety margins inherent in the RHR piping and metallic components and the basis for their calculation, however, must be substantiated. Further, the influence of life or time dependent effects on these safety margins must also be considered. Of particular concern in this regard is the capacity of the potentially corrosion degraded or embrittled heat exchanger tubes to withstand any dynamic loads associated with the overpressurization event.

The following request for further information has been made. "In the description of RHR Pressure Boundary Failure Modes, it is stated that the maximum value of stresses due to pressurization to 2250 psia in the limiting RHR piping are approaching the yield stress and the stresses in the other metallic components are at a small fraction of their respective yield stresses. Describe the analyses conducted to support this conclusion and provide a summary of the pertinent results. In addition, clarify whether the pressure loading has been applied as a dynamic pulse coupled with corrosion degradation effects (such as heat exchanger tube embrittlement). If these effects have been considered, describe the analyses and the dynamic loads. If not, provide the basis for not considering these effects."

The applicant's response to this request will be reviewed and the results of our evaluation discussed in a future supplement to this report.

2.1.6 Event Tree Quantification

This section summarizes the effect of observations made by BNL in previous sections to the event tree quantification.

From the above observations, it is obvious that the main problem in the quantification of various ISL scenarios are related to the determination of the initiator frequencies. The other observations and questions expose mainly the overall uncertainty of the real frequencies of these accident scenarios.

The effect of the change in the initiator frequencies to the plant damage states can be demonstrated if the new initiator frequencies, VI and VS, given in Sections 2.1.3.2 and 2.1.3.3 are propagated through the corresponding event trees.

Table 2.5 presents the results of the BNL requantification. The table and its notation is essentially the same as Table 3-14 of the Seabrook EPZ Study. For convenience, in Table 2.5, the meaning of some plant damage states are repeated.

From the table, the new value of the total core damage contribution due to ISL can be determined (the sum of PDS states 8C through 1FV). This is:

$$CD_{ISL} = 1.37X10^{-6} \text{ event/reactor year.}$$

The value obtained is much higher than the updated value (see Section 2.1.1) of the Seabrook EPZ Study. It is much closer to the result of an earlier assessment given in the SSPSA, which is,

$$CD_{ISL} = 1.8x10^{-6} \text{ event/reactor year.}$$

2.2 Accidents During Shutdown and Refueling Conditions

The Seabrook Emergency Planning Study[2] concentrated on accidents that would occur during power operation, and did not assess the risk during non-power operation. Table 2.6 defines 6 modes of plant operations as were defined in standard technical specifications. As far as early releases are concerned, there are some potentially significant contributors from operation in modes 4, 5, and 6. Typically, technical specifications do not address the status of containment isolation in mode 5, and require isolation in mode 6 only during periods of fuel handling. Consequently, it is possible to have a core melt accident with the containment wide open.

NSAC-84[8] is the only study that was performed to assess the core damage frequency due to accidents during non-power operation at PWR. It is an innovative and detailed study for the Zion plant, using the plant-specific procedures and experience. Three types of initiating events were considered: loss of cooling, low temperature overpressurization, and loss of coolant. NSAC-84 results show that the dominant core damage sequences are due to loss of RHR system and human errors. The contribution of LOCA to core damage frequency during shutdown and refueling is approximately $2x10^{-6}$/year. The contribution of low temperature overpressurization is assessed to be less than 10^{-10}/year. The total core damage frequency during shutdown or refueling was assessed to be $1.8x10^{-5}$/year which is comparable to the frequency of core damage during power operations, i.e., $5.7x10^{-5}$/year. It was stated in the executive summary of NSAC-84 that "with the uncertainties involved, the risk of fuel damage during some period of a shutdown may be as great as the risk at power."

BNL performed a limited review of NSAC-84. Based on the limited information documented in NSAC-84 and limited information for Seabrook, BNL believes that the dominant sequences identified in NSAC-84 may also apply to Seabrook, while the frequencies of the accident sequences need to be reassessed. The NSAC-84 analysis of low temperature overpressurization may be too optimistic. Events such as those of Turkey Point-4[9] indicate that the frequency with which a rapid pressurization occurs with the RHR system isolated and the PORVs unavailable is higher than 10^{-3} per year. The operators have only a few minutes to respond to the event before the pressure reaches the setpoint for the safety valves. Therefore, the human error probability is not going to be very small. If the operators fail to terminate the overpressurization, the primary system pressure will reach the setpoint for the safety valves. At this pressure, the vessel rupture probability may be of the order of 10^{-3}.[10,11]

The following sections discuss the possible initiating events in more detail. Operational experiences and causes of failures are provided for each type of initiating events. Some scenarios that may lead to core damage are provided based on reports in the related area. Related safety issues are also provided.

2.2.1 Loss of Decay Heat Removal

The RHR system is designed to remove decay heat from the primary coolant system during modes 4, 5, and 6. Most of the time when the system is operating, only one train is actually running; the other train is either on standby or unavailable due to test or maintenance. If the operating train fails to continue running, the standby train will not start automatically. Therefore, loss of the operating train leads to loss of the system and operator actions will be required to restore it. The Office of Analysis and Evaluation of Operational Data (AEOD) identified and analyzed 130 loss-of-DHR events at PWRs during approximately 500 reactor years of operations.[12] According to this experience base, the frequency of loss-of-RHR is estimated to be 0.25 per reactor year. Table 2.7 lists the categories of the 130 events. It can be seen that automatic closure of suction valves and inadequate RCS inventory are the two dominant causes of loss of DHR. Automatic closures of suction valves were caused by spurious high pressure signals, loss of instrument bus, and human errors in calibration of pressure transmitters. Inadequate RCS inventory was caused by human errors and inadequate vessel indications during drained-down operations. It was estimated[12] that approximately two-thirds of the events were human error related.

Upon loss of DHR, operators may be able to restore the failed train by reopening the spuriously closed suction valve, or starting the standby train. Alternative methods for decay heat removal include use of steam generators and use of charging pumps. Typically several hours are available before core uncovery occurs. Therefore, the most important thing is that the operators must be able to recognize the loss of DHR. In the 130 loss-of-DHR events identified in the AEOD study, the operators responded in a timely fashion, such that no serious damage resulted. However, the duration of loss of DHR in some cases exceeded one hour. If a loss of DHR occurs 36 hours after a reactor trip with the plant in a partially drained condition, the onset of core uncovery may be 1 hour after the loss-ofDHR.[12]

The dominant core damage sequences in NSAC-84 represent scenarios in which decay heat removal is lost and the operators fail to determine that action to restore cooling is required. For example, the accident sequence with the highest frequency is a sequence in which the RHR suction valve is inadvertently closed, the operator fails to trip the RHR pumps, and also fails to determine that action to restore cooling is required. Its frequency is estimated to be 4.3×10^{-6} per reactor year. Such scenarios can be postulated for Seabrook. However, quantitative assessment of the accident scenarios must take into consideration the plant-specific information. For example, operator performance is strongly affected by the indications or alarms available in the control room. Zion does not have any alarm in the control room for inadvertent closure of the RHR suction valves, while the Seabrook control room has an audible alarm on the video alarm system if the RHR pump is running with the suction valve closed. Another difference between Zion and Seabrook is that Zion has a single drop line and Seabrook has double drop lines. This is not expected to be a significant difference because the auto closure logic at Seabrook will isolate both suction lines when a spurious signal is generated, i.e., a single pressure transmitter provides input to the interlock logic of the two inner isolation valves, and a separate pressure transmitter provides input to the interlock of both outer isolation valves. BNL performed a LER search for loss-of-DHR events due to spurious closure of suction valves at plants with double drop lines. Seven events were found in approximately 40 reactor years. This indicates that the frequency of spurious closure of suction valves at plants with double drop lines is not lower than the average frequency for all plants.

In response to the NRC request for additional information, the Public Service of New Hampshire (PSNH) provided a shutdown risk analysis for Seabrook[13] utilizing the results of NSAC-84. Two differences between Seabrook and Zion were accounted for, viz., the number of hot leg suction lines, and the support system interfaces with the RHR system. In the analysis, credit for the additional suction line at Seabrook was taken, based on the statement "For spurious valve closure to cause a loss of RHR cooling at Seabrook station, it is necessary to postulate either a common cause event involving one valve in each suction path, or a coincidence of a single valve closure and maintenance being performed on the other RHR train." This reduced the frequency of loss of RHR by a multiplicative factor of 0.145 and the core damage frequency by approximately a factor of 2. From the information available to BNL, it is not clear that this reduction is justified. It is true that a single train of the RHR system is adequate for decay heat removal. However, the standby train is not normally operating and will not start automatically when the operating train becomes unavailable. The analysis in reference 13 assumes perfect automatic start signal for the standby train or perfect operator response to the loss of the operating train; however, it has previously been shown that operator error is important in these sequences, and neglect of it here is inappropriate.

The analysis of the support systems in reference 13 has not been reviewed by BNL. As was stated in the analysis, the differences in the support system interfaces with the RHR system are unfavorable for Seabrook.

Two issues are related to the availability of RHR system, i.e., unresolved safety issues A-45 and generic issue 99. A-45 addresses the adequacy of decay heat removal systems in existing light water reactor nuclear power plants. Generic issue 99 addresses the RHR suction line interlocks on PWRs. BNL is currently involved in a project to investigate methods to improve the reliability of RHR systems during shutdown or refueling. The results of the project will be used towards resolution of generic issue 99.

2.2.2 Low Temperature Overpressurization

Low temperature overpressurization may occur during shutdown as a result of unanticipated addition of mass to the reactor coolant system, for example, inadvertent actuation of safety injection pumps, or imbalance of letdown and charging flows. Imbalance of letdown and charging flow may be caused by spurious isolation of the RHR system, thus, violation of the letdown flow, and loss of instrument air that causes the letdown flow control valve to close and the charging line flow control valve to open. To protect the Seabrook plant against such scenarios, a low temperature overpressurization protection system is activated when the primary system is cooled down after a reactor trip. The system monitors the primary system pressure and temperature and actuates a main control board alarm when the pressure reaches a pre-determined fraction of the allowable pressure, and on a further increase in measured pressure, transmits an actuation signal to the PORVs and the PORV isolation valves. Also, the safety injection pumps and one or more of the charging pumps are made inoperable during initial cooldown. In addition to the PORVs, the relief valves in the RHR system may be available to relieve the pressure. Each RHR suction line has a relief valve with 900 gpm capacity at 450 psig, and each RHR discharge line has a relief valve with 20 gpm capacity at 600 psig. However, these relief valves may be made ineffective if the RHR suction valves close automatically when the setpoint of 600 psig is reached, as was the case in the Turkey Point-4 events. Actually, the Seabrook Technical Specifications only require either both PORVs or both RHR relief valves to be available.

Two generic issues are related to the subject of low temperature overpressurization, generic issues 94 and 70. Generic issue 94 considers additional low-temperature-overpressurization protection for light water reactors. It has a "high" priority ranking.[11] Enclosure 1 to reference 11 is the prioritization evaluation for the issue. It was stated in the evaluation that before 1979 30 events in PWRs were reported where the pressure/temperature of the reactor coolant system violating Technical Specifications. After 1979, following changes to operating procedures and the implementation of overpressurization mitigation systems, there have been two reported events of overpressure excursion events, i.e., Turkey Point-4 events. Based on the operational experience and the use of the Vessel Integrity Simulation Analysis (VISA) code,[14] the prioritization evaluation estimated that the core damage frequency due to vessel rupture in a low-temperature-overpressurization event at Oconee 3 to be 4.5×10^{-6} per reactor year. Generic issue 70 considers the reliability of PORV and its block valves. BNL is currently investigating the issue, and a draft report of the work is upcoming.

2.2.3 Loss of Coolant Accidents

NSAC-52[15] reviewed operating experience within 5 calendar years up to the end of 1981, and identified 10 loss of coolant events at PWRs. They were caused by the following causes:

1. Inadvertent manual initiation of RHRS supplied containment spray.
2. Inadvertent loss of inventory to the containment building sump and/or automatic initiation of recirculation mode of low pressure safety injection.
3. Inadvertent loss of inventory via the RHRS relief valves.

4. Inadvertent loss of inventory via mispositioned crossconnect or drain valves.
5. RHRS valve packing gland removal during plant pressurization, dislodging the valve packing and gland.
6. Gross valve packing leak.

As for the loss-of-cooling initiating event, LOCA during shutdown or refueling requires operator response to terminate the inventory loss and to provide inventory make up. The NSAC-84 analysis for Zion assessed the core damage frequency due to a LOCA at shutdown or fueling to be approximately $2x10^{-6}$ per reactor year. The dominant scenario is that the operator fails to close the RHR return valve to the RWST after draining the cavity, on reestablishing RHR flow, a LOCA via the RWST vent outside the plant occurs, and the operator fails to respond to it.

2.3 Induced Steam Generator Tube Rupture (SGTR)

For accidents in which the primary system is at high pressure during core uncovery and melting, it is possible that large natural circulation flow patterns could develop within the primary system. These flow patterns could in turn heat up regions of the primary system remote from the reactor core. As the primary system heats-up, it is possible that parts of the pressure boundary could degrade. Of particular concern is the possibility of degrading the steam generator tubes such that the primary system could become open to the secondary system. If the secondary system were in turn open to atmosphere, then a direct path could exist between the primary system and the atmosphere, which bypasses containment.

This is a very important topic for review because it could potentially lead to a relatively large early release of radioactivity, and the applicant considers it to be very unlikely. The topic was not included as part of the work scope for the current BNL review. However, the topic was reviewed[16] in detail by the NRC staff and is the subject of continuing NRC and industry research activities.

Scoping studies were performed to assess the impact of induced steam generator tube rupture on risk at Seabrook. First, the frequency of accidents in which the primary system would be at high pressure had to be determined. The applicant estimated[17] the frequency of high pressure sequences in which a SGTR might have an effect to be 4×10^{-5} per reactor year. This estimate was considered reasonable in the NRC review[16] and therefore used as the basis for the BNL scoping study.

Given that core meltdown occurs with the primary system at high pressure, the probability that the steam generator tubes will fail had then to be determined. In addition, it is also possible (provided methods are available) for the operators to depressurize the primary system prior to induced failure of the SGT. The probability of successful depressurization had also to be determined.

Estimating the probabilities of the above events is subject to significant uncertainty. However, the Severe Accident Risk Reduction Program at SNL attempted to quantify these probabilities by use of expert judgment. The probabilities were developed specifically for the Surry plant and reported in Appendix B of SAND86-0119.[18] The experts concluded that there was a conditional probability of 0.8 for successful depressurization of the primary system. In addition, they felt that the probability of an induced steam generator tube rupture might be between 0.01 and 0.1 (for both small and large tube ruptures) conditional on no depressurization. These estimates are reasonably consistent with an earlier NRC memorandum[19] on this subject, which suggested a conditional probability of about 0.01 to 0.3 for SGTR given a high pressure core meltdown. It was therefore decided to use 0.2 as the conditional probability of failure to depressurize and a range of 0.01 to 0.3 for SGTR to assess the impact of this phenomenon on risk at Seabrook. The results are summarized in Section 2.5.

2.4 Containment Isolation Failure

To be provided.

2.5 Summary

In the introduction to this section, several categories of questions were listed; the topics treated here are each particular instances of those general categories. It must be emphasized that there has been no top-down review of the process of establishing the frequency of any release category; the topics addressed here were chosen essentially a priori. For some topics, sensitivity studies have been performed using the applicant's conditional risk indices to show how the dose vs distance and risk profiles might change as a result of the concerns raised in this section. The results are summarized below.

Interfacing System LOCA

The contribution to release categories S1W and S7W from interfacing system LOCA (as given in reference 1) are reproduced below:

Source Term Category	Plant Damage State	Frequency
S1W	1FV	4.6×10^{-9}
S7W	1FPV, 7FPV	3.9×10^{-8}

The BNL requantification of the event tree for interfacing system LOCAs (in Section 2.1) suggested the revised frequencies in Table 2.4. These revised frequencies would imply the following contribution to source term categories S1W and S7W:

Source Term Category	Plant Damage State	Frequency
S1W	1FV	1.4×10^{-7}
S7W	1FPV, 7FPV	1.1×10^{-6}

The impact of the above changes on the risk estimates in PLG-0432 is given in Table 2.8. The revised frequencies for interfacing system LOCAs have (as expected) no impact on the risk of cancers (refer to Table 2.8) because this health effect is dominated by the higher frequency S2W category. The impact of the revised frequencies on early fatality risk is very small if no evacuation is assumed. However, as successively larger evacuation distances are assumed the revised frequencies have rather more effect. However, as the no evacuation assumption is already below the proposed safety goals, the changes in risk for the one and two mile evacuation assumptions are also within the ~~proposed~~ NRC safety goals. The above changes have no impact on the 200 rem dose vs distance curves in PLG-0432 (reproduced in Figure 2.2) because they influence conditional frequencies below 0.001.

Note that in Section 5.2 questions were raised regarding the credit to be taken for pool scrubbing of fission products for the S7W source term category. In the RSS[21], a decontamination factor of 100 was used for subcooled pool scrubbing rather than the factor of 1000 which was used in PLG-0465 to calculate the S7W release fractions. BNL has confirmed that the effect of using a factor of 100 rather than 1000 would not change the results of the BNL sensitivity study in Table 2.8. However, the statement made in Section 5.2 that neglecting pool scrubbing for all interfacing LOCA sequences would have no impact on risk is only true if the frequencies in PLG-0465 are used. If pool scrubbing was neglected and the revised BNL frequencies in Table 2.4 was used, then the risk results would be effected. However, BNL does not consider it appropriate to neglect pool scrubbing for that subset of interfacing system LOCA sequences where the break location is under water. Thus, BNL considers the sensitivity results presented in Table 2.8 to be appropriate even when our concerns in Section 5.2 are taken into acocunt.

Accidents During Shutdown

This topic was not originally addressed in PLG-0465 and a detailed assessment of such events is beyond the scope of the current BNL work on this project. However, the applicant was requested to provide information on the risk associated with accidents during shutdown. The results of the applicant's assessment of such accidents were presented in the form of sensitivity studies in reference 13. BNL is not presently in a position to assess the frequency of these events for Seabrook becuase, as explained in Section 2.2, there remain fundamental questions regarding the modeling of these scenarios. However, in spite of this, the applicant's results have been included in Table 2.8 and Figures 2.2 and 2.3 for comparison with the BNL sensitivity study results on other topics. It should be noted that the applicant considers their upper bound estimates in Table 2.8 and Figure 2.3 to be very conservative. In particular, in order to assess the impact of these events, they were included in source term categories derived for accidents from full power, which could lead to predictions of shorter times and larger quantities of fission product release than would be expected from accidents during shutdown.

As shown by the applicant in Table 2.8 and Figures 2.2 and 2.3, the influence on risk of accidents during shutdown can be significant. It should be noted that this topic is being addressed generically in other ongoing NRC work.

Induced Steam Generator Tube Rupture

In Section 2.3 a sensitivity study was suggested to assess the impact of induced SGTR on risk at Seabrook. The frequency of high pressure sequences taken together with the conditional probabilities of failure to depressurize and induced SGTR given in Section 2.3 give the following range of probabilities of induced SGTR:

$4.0 \times 10^{-5} \times 0.2 \times 0.3 = 2.4 \times 10^{-6}$ per reactor year
$4.0 \times 10^{-5} \times 0.2 \times 0.01 = 8.0 \times 10^{-8}$ per reactor year.

In order to estimate the impact of the above probabilities on risk an appropriate source term category had to be selected. It was decided to allocate SGTR events to release category S1W, which represents a large early bypass of the containment. It was felt that this was a conservative assumption because significant retention of the fission products in the secondary side could occur and this was not considered when calculating the S1W release fractions. The impact of adding the above frequencies to source term category S1W is illustrated in Table 2.8 and Figure 2.4.

The impact of the above changes relative to the risk of cancers given in PLG-0465 is again relatively small even when the higher probability of SGTR is assumed. However, the impact on early fatality risk of the highest probability of SGTR has rather more impact. The risk of early fatalities increases from 2.7(-3) to 4.5(-3) per reactor year assuming no evacuation and remains relatively high even with a two mile evacuation. The risk of early fatalities assuming no evacuation and the high SGTR probability is close to and may exceed the ~~proposed~~ NRC safety goals. It should be noted that the risk estimates in Table 2.8 are sensitivity studies and that the NRC staff believes[19] that the actual probability of a SGTR is closer to the lower estimate. However, given the uncertainty associated with predicting such events it is prudent to indicate the impact on risk of a range of assumptions regarding SGTR.

The impact on the 200 rem dose vs distance curves in PLG-0465 is shown in Figure 2.2. Only the more conservative probability for SGTR has an impact on the 200 rem dose vs distance curves and then only on conditional frequencies below 0.01. The statement in PLG-0432 that "the risk of radiological exposures for 200-rem whole body dose with no immediate protective actions is less at 1 mile than the corresponding NUREG-0396 results at 10 miles" is not affected by this particular sensitivity study related to the potential for SGTR.

Containment Isolation Failure

(to be provided)

2.6 References

1. PLG-0465.
2. Seabrook Station Risk Management and Emergency Planning Study, PLG-0432, December 1985.
3. SSPSA.
4. MAAP code.
5. NUREG/CR-1363.
6. Team Enhanced Evaluation Method (TEEM) - Checklists and Instructions, Informal Report BNL-38585, September 1986.
7. Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications, Final Report NUREG/CR-1278, August 1983.
8. Zion Nuclear Plant Residual Heat Removal PRA, NSAC-84, July 1985.
9. "Overpressurization of Reactor Coolant System," IE Information Notice No. 82-17, U.S. NRC, June 11, 1982.

10. T. J. Burns et al., "Pressurized Thermal Shock Evaluation of the Oconee-1 Nuclear Power Plant," NUREG/CR-3770, Draft, April 1984.

11. Memo from H. R. Denton, Director, Office of NRR, to R. M. Bernero, Director, Division of System Integration, on the Subject of Schedule for Resolving and Completing Generic Issue No. 94, July 23, 1985.

12. H. Ornstein, "Decay Heat Removal Problems at U.S. Pressurized Water Reactors," Office for Analysis and Evaluation of Operational Data, U.S. NRC, December 1985.

13. PSNH Letter (SBN-1225), dated October 31, 1986, "Response to Request for Additional Information (RAIs)," J. DeVincentis to S. M. Long

14. NUREG/CR-3384, "VISA--A Computer Code for Predicting the Probability of Reactor Vessel Failure," U.S. NRC, September 1983.

15. Residual Heat Removal Experience Review and Safety Analysis, NSAC-52, January 1983.

16. Warren Lyon's memo on SGTR.

17. PSNH Letter (SBN-1237), dated November 21, 1986, "Emergency Planning Sensitivity Study," J. DeVincentis to S. M. Long.

18. SAND86-0119.

19. NRC Memorandum, dated February 14, 1985, "Steam Generator Tube Response during Severe Accidents," B. W. Sheron to D. B. Liaw.

20. EGG-NTA7287.

21. RSS.

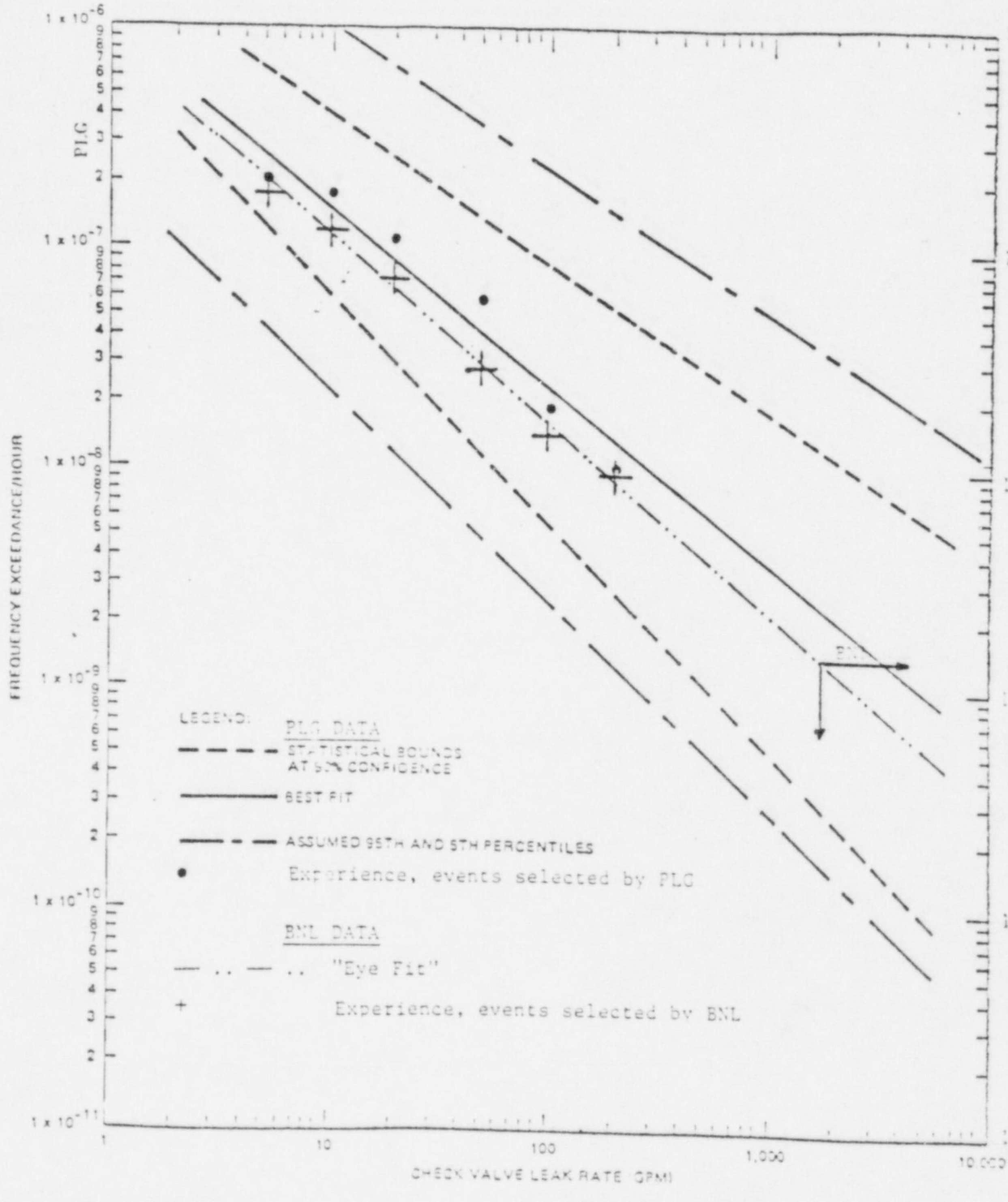


Figure 2.1 Frequency of accumulator check valve leakage events.

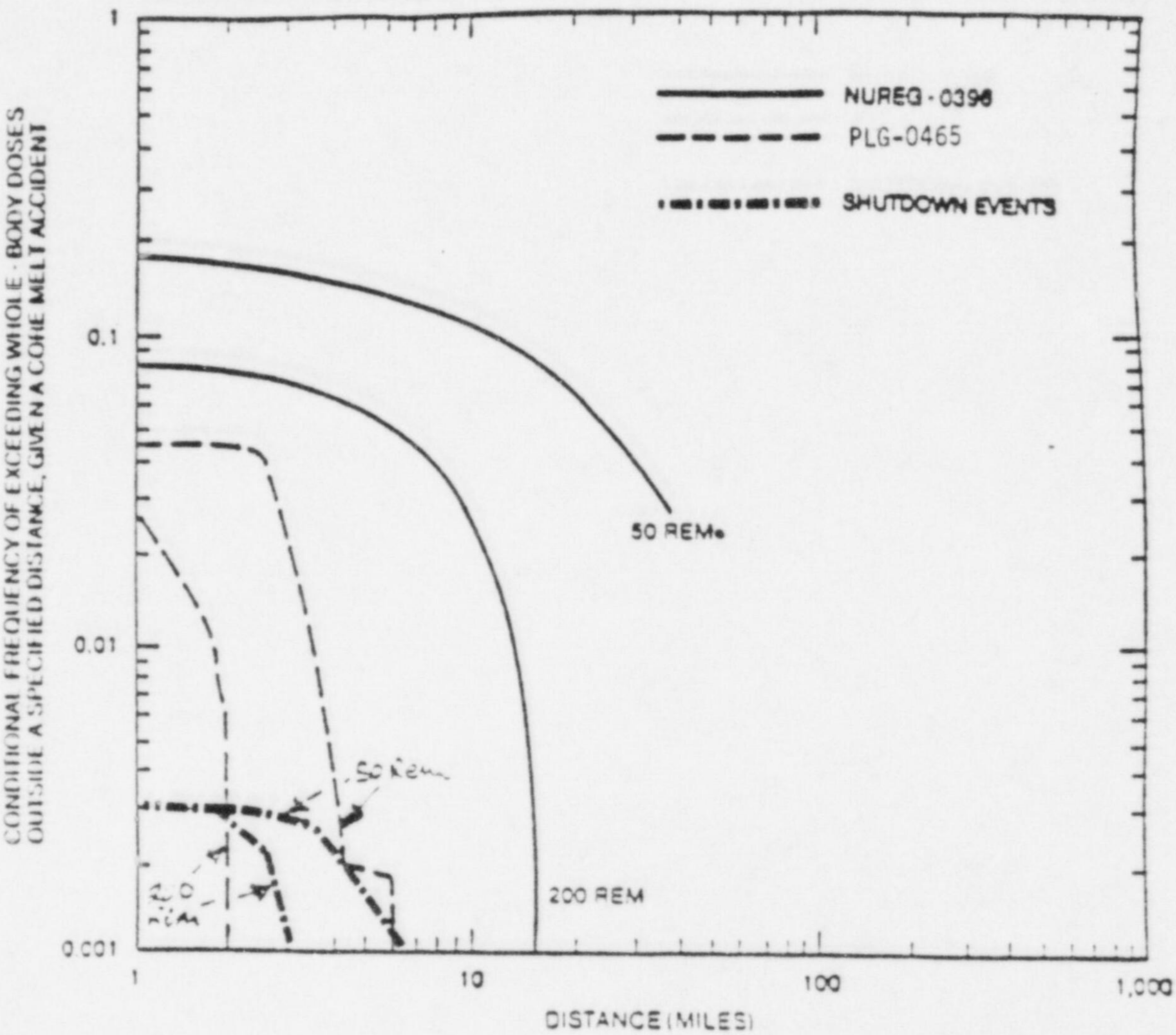


Figure 2.2 Comparison of 200 rem and 50 rem dose versus distance curves with contributions from shutdown events. (Calculations performed by applicant)

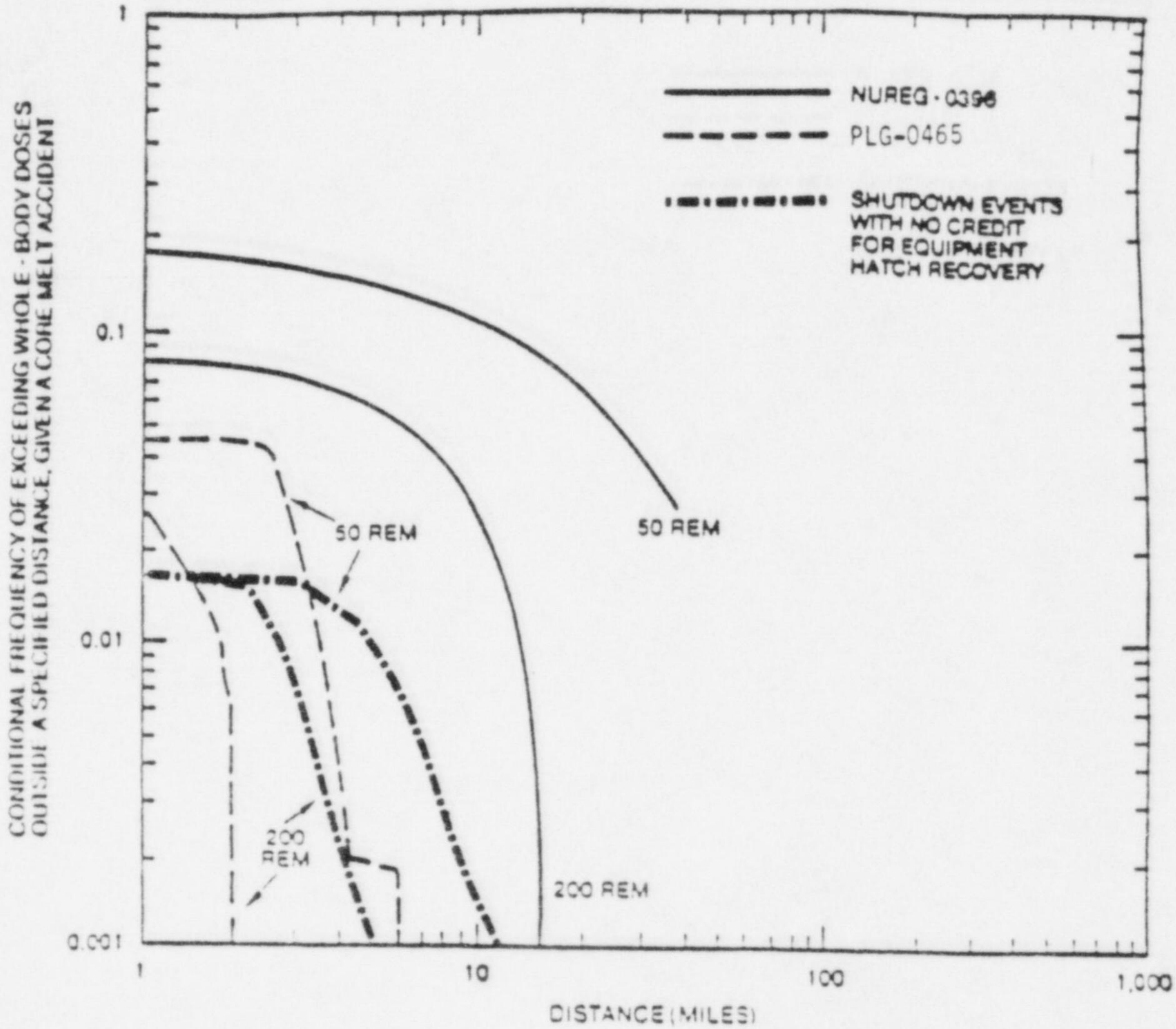


Figure 2.3 Comparison of 200 rem and 50 rem dose versus distance curves for conservative assumption of no credit for operator recovery of open equipment hatch. (Calculations performed by applicant)

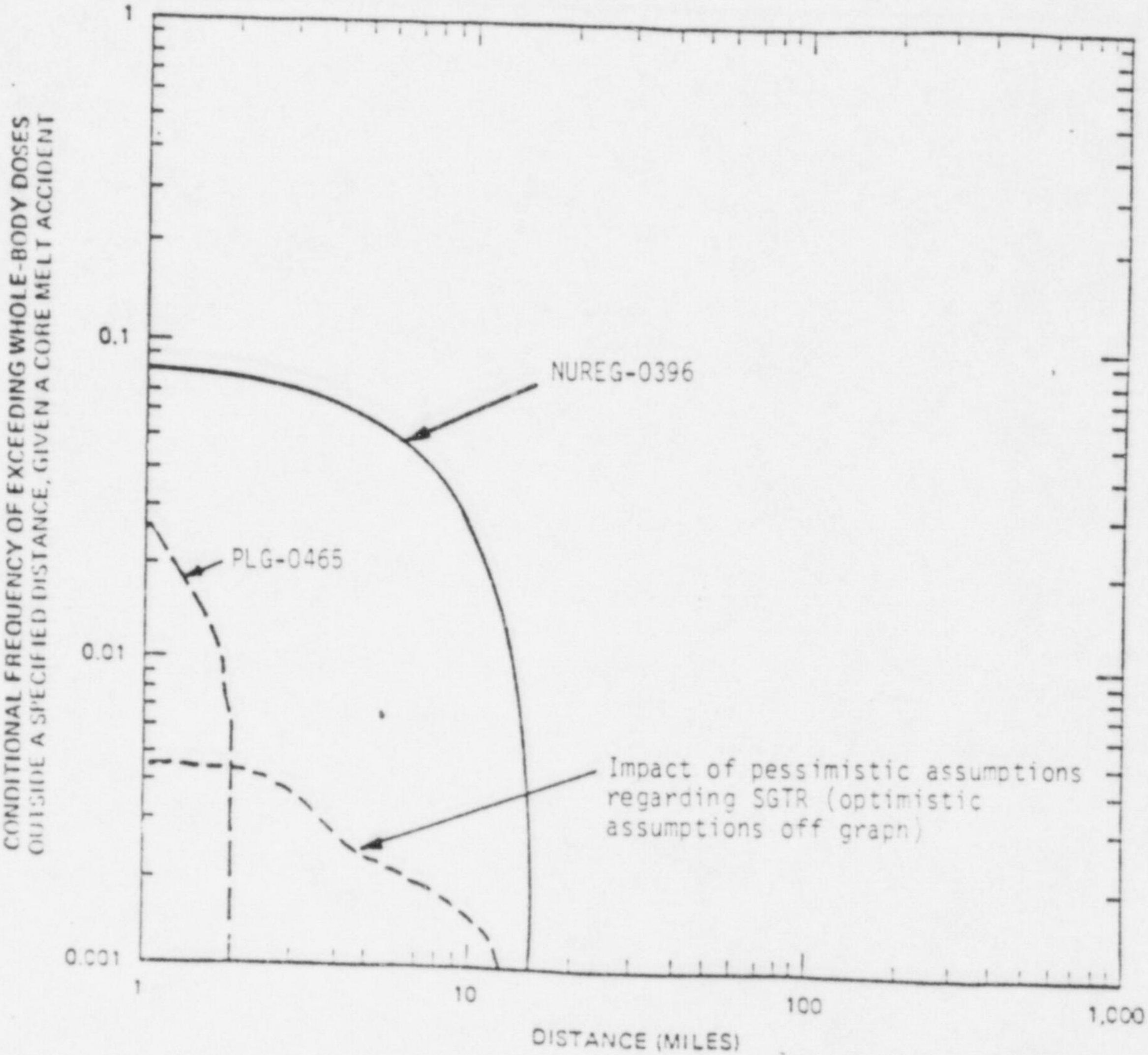


Figure 2.4 Comparison of BNL sensitivity study related to SGTR with PLG-0465 and NUREG-0396 (200 rem plots with no immediate protective actions).

Table 2.1 Summary of Operating Events, Emergency Core Cooling System, Isolation Check Valves, Leakage Failure Mode

Reference (NPE #)	Plant	Date	ECCS System	Event Description	Number of Check Valves Reported	Estimated Leak Rate (gpm)	Remarks
VII.A.13	Palisades	5/72	ACC	Leakage into SI tank. The internals of a check valve on the outlet of an SI tank was incorrectly assembled.	1	$\gamma<5$	Note 1
VII.A.25	Main Yankee	12/72	ACC	Leakage into SI tank. A small piece of weld slag had lodged under the seat of the outlet check valve allowing back leakage. Dilution: 1700 ppm (limit is 1720 ppm).	1	$\gamma<5$	Note 1
VII.A.32	Turkey Point	5/73	HPI	One of the three check valves in the SI lines developed a leakage of 1/3 gpm. Two other check valves showed only slight leakage. Failure of soft seats.	3	$\gamma\leq.33$	
VII.A.63	Ginna	9/74	ACC	Leakage of a check valve caused boron dilution in ACC. "A" (from 2250 ppm to 1617 ppm).	1	$\gamma<20$	Note 1
VIII.A.85	Surry 1	8/75	ACC	Check valve did not seat. ACC ("IC") level increased. Leakage rate: ≈6 gpm.	1	$\gamma<10$	Note 1
VII.A.126	Zion 2	10/75	ACC	Wrong size gasket installed in the check valve for ACC. "A". Leak rate: ≈.25 gpm.	1	$\gamma\leq.25$	Note 1
VII.A.105	Robinson 2	1/76	ACC	Accumulator ("B") inleakage through leaking outlet check valve.	1	$\gamma<20$	Note 1
V.A.122	Zion 1	6/76	ACC	Inleakage to ACC. "1D" from RCS.	1	$\gamma<20$	Note 1
VII.A.114	Surry 1	7/76	ACC	Two check valves in series (1-SI-128, 130) leaked causing boron dilution in ACC. "B".	2 in series	$\gamma<10$	Note 1
VII.A.120	Surry 2	8/76	ACC	Boron dilution (from 1950 ppm to 1893) in SI ACC. "C" caused by leaking check valves (2-SI-145, 147).	2 in series	$\gamma<10$	Notes 1 and 2
VII.A.225	Millstone 2	4/77	ACC	Inleakage of RC through outlet check valves to SI tank "4". Low boron concentration. Five occurrences in 1977.	2 in series	$\gamma<20$	Note 2
VII.A.175	San Onofre 1	5/78	LPI	Tilting disk check valve (first valve inside containment) failed to close with gravity installed in a vertical rather than a horizontal pipeline.	1	$\gamma<5$	
VIII.A.182	Calvert Cliffs 2	9/78	ACC	Outlet check valves for SI tanks 21B and 22B leaked. Boron concentration reduction from 1724 and 1731 ppm to 1652 and 1594 ppm in one month period.	2	$\gamma<10$ $\gamma<10$	Note 1

Table 2.1 Continued

Reference (NPE #)	Plant	Date	ECCS System	Event Description	Number of Check Valves Reported	Estimated Leak Rate (gpm)	Remarks
VII.A.262	Crystal River 3	7/80	ACC	Check valve CFV-79 to core flood tank failed. The isolation valve to the N_2 system was open for N_2 mixing. ≈600 gallon liquid entered the N_2 system and ≈20 gallons was released. The corresponding activity released estimated as 1.07 mCi.	1	$100 < \gamma < 200$	Notes 1 and 2
VII.A.273 IE Info. Notice 80-41	Davis Besse 1	10/80	ACC	RHR system isolation check valve CF-30 leaked back excessively. Valve disk and arm had separated from the valve body. Bolts and locking mechanism were missing. Core flood tank overpressurized.	1	$50 < \gamma < 100$	Note 1
VII.A.291	Surry 2	1/81	ACC	Accumulator ("C") boron diluted. Check valve (1-SI-144) leaked. Flushing system improperly set up, resulting in charging system pressure to exist on the downstream side of the check valve.	1	$\gamma < 10$	Note 1
VII.A.301	Palisades	3/81	ACC	Leakage of RC into the SI tank (T-823).	1	$\gamma < 5$	Notes 1 and 2
VII.A.306	McGuire 1	4/81	ACC	Accumulator "A" outlet check valves IN-159 and IN-160 were leaking. RCS pressure: 1800 psig. Acc. pressure: 425 psig. Water level above alarm setpoint.	2 in series	$\gamma < 10$	Note 1
VII.A.307	McGuire 1	4/81	ACC	Similar events with Accs. "C" and "D".	2x2 in series	$\gamma < 10$ $\gamma < 10$	Notes 1 and 2
VII.A.343	Point Beach 1	10/81	LPI	RCS/LPI isolation check valve (1-853C) leaks in excess of acceptance criteria (>6 gpm).	1	$\gamma < 10$	
VII.A.384	Calvert Cliffs 1 & 2	7/82	ACC	Acc. outlet check valve at Unit 1 leaked due to deterioration of the disk sealing o-ring. The o-ring material has been changed on all check valves of Unit 1 and 2 1/2 SI-215, 225, 235, and 245.	1	$\gamma \leq 200$	Note 1
VII.A.403	Surry 2	9/82	ACC	Acc. outlet check valve (2-SI-144) leaked RCS water into tank "C" during a pipe flush resulting in low boron concentration.	1	$\gamma < 20$	Notes 1 and 2
VII.A.396	Palisades	9-12/82	ACC	Minor leakage into SI tank (compounded by level indication failure) via check valve leakages.	1	$\gamma < 5$	Notes 1 and 2

Table 2.1 Continued

Reference (NPE #)	Plant	Date	ECCS System	Event Description	Number of Check Valves Reported	Estimated Leak Rate (gpm)	Remarks
VII.A.407	McGuire 1	5/83	ACC	RCS water inleakage through outlet check valves IN-170 and IN-171, resulting in low boron concentration in CLA "B".	2 in series	$20<\gamma<50$	Note 1
VII.A.437	Farley 2	9/83	HPI	SI check valve to loop 3 cold leg was excessively leaking, incomplete contact between the valve disk and seat.	1	$50<\gamma<100$	
LER 84-001	Oconee 1	3/84	ACC	Accumulator ("A") inleakage through leaking valves. Administrative deficiency, no management control over a known problem (since 8/83).	2 in series	$\gamma<5$	Note 1
V.F.0043 LER 84-012	Palisades	7/84	ACC	Accumulator inleakage through leaking check valves CK-3146 and CK-3116.	2	$\gamma<5$	Notes 1 and 2
VII.A.452	St. Lucie 2	12/84	ACC	Inleakage to SI tank. Seal plate cocked, valve seat compensating joint ball galled.	1	$20<\gamma<50$	Note 1
VII.A.456	Calvert Cliffs 1 & 2	1/85	ACC	Inleakage to safety injection tanks through check valve, o-ring material degradation (Unit 1 = 1.6 gpm, Unit 2 = 27.2 gpm).	2	$\gamma<5$ $20<\gamma<50$	Note 1
VII.A.457	McGuire 1	4/85	ACC	Low accumulator boron concentration.	1	$\gamma<5$	Note 1
LER 85-007	Palisades	6/85	ACC	Inleakage from the RCS via a check valve. Low level boron concentration.	1	$\gamma<5$	Note 1
VII.A.474	Palisades	11/85	ACC	Accumulator (SIT-82D) inleakage from RCS through a check valve, CK-3116. Boron dilution.	1	$\gamma<5$	Note 3

Note 1: Estimated leak rate is the resultant one through two check valves in series.
Note 2: Not listed in Table 3.8 of PLG-0432.
Note 3: The Palisades unit has a chronic accumulator inleakage problem.

Table 2.2 Summary of Operating Events, Emergency Core Cooling System, Isolation Check Valves, "Failure to Close" Failure Mode

Reference (NFE #)	Plant	Date	ECCS System	Event Description	Number of Check Valves Reported	Remarks
VII.A.270	Sequoyah 1	9/80	HPI	SI check valve 63-635 was found to be stuck open. It was caused by interference between the disk nut lockwire tack weld and the valve body.	1	Note 1
VII.A.285	Salem 1	1?/80	HPI	SI check valve failed to close during a test. It is an interface between RCS hot leg and SI pumps. Valve was found to be locked open due to boron solidification during the last refueling.	1	Note 1
VII.A.294	Oconee 1	2/81	LPI	Reactor vessel LPI loop "B" isolation valve (GCF-12) leaked excessively during LOCA leak test. The valve disk had become frozen at the pivot in a cocked position. Buildup of deposit in the gap between the hinge and disc knob caused the freezing.	1	Note 1
VII.A.302	Oconee 3	3/81	LPI	Similar to event at Unit 1 (valve involved is 3 CF-13).	1	Note 1
VII.A.310	McGuire 1	5/81	ACC	Leak test damaged acc. check valves - seat type changed.	2	Note 1
VII.A.311	McGuire 1	5/81	ACC	Acc. check valves failed.	2	Note 1
VII.A.315	Point Beach 1	7/81	LPI	RCS/LPI isolation check valves 1-853 C and D were found to be stuck in the full open position. High leakage rate.	2	Note 1
VII.A.392	ANO-2	10/82	HPI	SI isolation check valves 2 SI-13C and 2 SI-13B stuck in the open position during test requested by IE Notice 81-30. Disk stud protruded above nut, disk misaligned.	2	Note 1

Note 1: Not listed in Table 3.8 of PLG-0432.

Table 2.3 Accumulator Check Valve Exposure Data

Plant Name	Start of Commercial Operation	Number of Years	Number of Accumulator Check Valves	Total Number of Check Valve (10^5 Hours)
Arkanasas Nuclear One 1	December 1974	11.08	4	3.882
Crystal River 3	March 1977	8.83	4	3.094
Davis-Besse 1	November 1977	8.16	4	2.859
Oconee 1	July 1973	12.50	4	4.380
Oconee 2	March 1974	11.83	4	4.145
Oconee 3	December 1974	11.08	4	3.882
Rancho Seco	April 1975	10.75	4	3.767
Three Mile Island 1	September 1974	11.33	4	3.970
Three Mile Island 2	December 1978	7.08	4	2.481
Arkansas Nuclear One 2	March 1980	5.83	8	4.086
Calvert Cliffs 1	May 1975	10.67	8	7.478
Calvert Cliffs 2	April 1977	8.75	8	6.132
Fort Calhoun	September 1973	12.33	8	8.641
Millstone 2	December 1975	10.08	8	7.064
Maine Yankee	December 1972	13.08	6	6.875
Palisades	December 1971	14.08	8	9.867
St. Lucie 1	December 1976	7.08	8	6.363
Beaver Valley 1	April 1977	8.75	6	4.599
D. C. Cook 1	August 1975	10.42	8	7.302
D. C. Cook 2	July 1978	7.50	8	5.256
Indian Point 2	July 1974	11.50	8	8.059
Indian Point 3	August 1976	9.42	8	6.602
Joseph M. Farley 1	December 1977	8.08	6	4.247
Kewaunee	June 1974	11.58	4	4.058
North Anna 1	June 1978	7.58	6	3.984
Prairie Island 1	December 1973	12.08	4	4.233
Prairie Island 2	December 1974	11.08	4	3.882
Point Beach 1	December 1970	15.08	4	5.284
Point Beach 2	October 1972	13.25	4	4.643
R. E. Ginna 1	March 1970	15.83	4	5.547
H. B. Robinson 2	March 1971	14.83	6	7.795
Salem 1	June 1977	8.50	8	5.957
Surry 1	December 1972	13.08	6	6.875
Surry 2	May 1973	12.67	6	6.659
Trojan	May 1976	9.67	8	6.777
Turkey Point 3	December 1972	13.08	6	6.875
Turkey Point 4	September 1973	12.33	6	6.481
Yankee Rowe	June 1971	14.50	2	2.540
Zion 1	December 1973	12.08	8	8.466
Zion 2	September 1974	11.33	8	7.940
McGuire 1	December 1981	4.08	8	2.859
Sequoyah 1	July 1981	4.50	10	3.942
Sequoyah 2	June 1982	3.58	10	3.136
TOTAL				2.369(2)

Table 2.4 Statistical Data on Leakage Events of Check Valves to Accumulators

Leak Rate (gpm)	Number of Events	Frequency of Occurrence (per hour)	Frequency of Exceedance
5	11	4.64(-7)	1.48(-6)
10	9	3.80(-7)	1.01(-6)
20	9	3.80(-7)	6.33(-7)
50	3	1.27(-7)	2.53(-7)
100	1	4.22(-8)	1.27(-7)
200	2	8.44(-8)	8.44(-8)

Table 2.5 ISL Results Initially Assigned Plant Damage States

Plant Damage State	Frequency Contribution From VI	Frequency Contribution From VS	Total Frequency
LOCA	1.96-4	1.44-4	3.4-4
DLOC	1.2-5	0	1.2-5
DILOC	9.8-8	7.7-6	7.8-6
8C	2.1-8	0	2.1-8
7D	1.5-7	0	1.5-7
7FPV	7.4-8	1.7-7	2.4-7
1FPV	1.8-8	8.0-7	8.2-7
1FV	8.4-8	5.9-8	1.4-7
Totals	2.1-4	1.5-4	3.6-4

Note:
LOCA: denotes a PDS, which contains those sequences in which the RC leakage in both ISL pathways analyzed exceeds 150 gpm, but does not exceed the RHR system relief valve capacity. The sequences are essentially medium LOCAs.
DLOC: denotes a PDS, which contains sequences in which the ISL is terminated.
DILOCA: denotes a PDS, in which coolant makeup is being supplied to the core, but the ISL has not been terminated.
The other plant damage states are involving containment bypassing ISLs and core damage.

Table 2.6 Plant Operational Modes*

Operational Mode	Reactivity Condition, Keff	% of Rated Thermal Power**	Average Coolant Temperature
1. POWER OPERATION	> 0.99	> 5%	> (T_{DHR})°F
2. STARTUP	> 0.99	< 5%	> (T_{DHR})°F
3. HOT STANDBY	< 0.99	0	> (T_{DHR})°F
4. HOT SHUTDOWN	< 0.99	0	(T_{DHR})°F>T_{avg}>200°F
5. COLD SHUTDOWN	< 0.99	0	< 200°F
6. REFUELING***	< 0.95	0	< 140°F

T_{DHR} = temperature at which the DHR system is initiated (generally 280°F - 350°F

*As defined in B&W, CE, and W standard technical specifications. Note many plants do not use standard technical specifications.

**Excluding decay heat.

***Fuel in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

Table 2.7 Categories of 130 Reported Total DHR System Failures When Required to Operate (Loss of Function) at U.S. PWRs 1976-1983

	No. of Events	(% of Events)
Automation closure of suction/ isolation valves	37	(28.5)
<u>Loss of inventory</u>		
Inadequate RCS inventory resulting in loss of DHR pump suction	26	(20.0)
Loss of RCS inventory through DHR system necessitating shutdown of DHR system	10	(7.7)
<u>Component Failures</u>		
Shutdown or failure of DHR pump	21	(16.2)
Inability to open suction/ isolation valve	8	(6.1)
Others	28	(21.5)
Total	130	(100.0)

Table 2.8 Impact of BNL Sensitivity Studies on PLG-0465 Risk Estimates

Sensitivity Study		Absolute Risk Per Reactor Year					
		Early Fatalities			Total Cancers		
		No Evac.	One Mile Evac.	Two Mile Evac.	No. Evac.	One Mile Evac.	Two Mile Evac.
Original PLG-0465 Results		2.7(-3)*	3.6(-4)	2.4(-5)	1.5(-2)	1.4(-2)	9.2(-3)
Revised Frequency for Interfacing System LOCAs		2.8(-3)	4.6(-4)	1.2(-4)	1.5(-2)	1.4(-2)	9.3(-3)
Steam Generator Tube Ruptured	**	4.5(-3)	2.2(-3)	1.8(-3)	1.7(-2)	1.6(-2)	1.1(-2)
	***	2.8(-3)	4.2(-4)	8.4(-4)	1.5(-2)	1.4(-2)	9.3(-3)
Accident from Shutdown****	**	6.2(-3)	2.9(-3)	3.0(-4)	2.1(-2)	2.0(-2)	1.4(-2)
	***	2.7(-3)	3.8(-4)	2.8(-5)	1.5(-2)	1.4(-2)	9.2(-3)

*2.7(-3) = 2.7×10^{-3}
**Pessimistic assumptions.
***Optimistic assumptions.
****Calculated by the Applicant, not confirmed by BNL.

3. EVALUATION OF CONTAINMENT BEHAVIOR

3.1. Capacity at General Yield

Seabrook Containment Building

The Seabrook Station containment building (See Fig. 3.1) is a reinforced concrete structure consisting of a basemat, a cylindrical wall and a hemispherical dome.[1] The basemat is essentially a 10' thick circular slab which supports the cylinder and other internal structures. The cylinder has an internal diameter of 140', a height of 149' and a minimum wall thickness of 4'-6". The dome internal diameter is 69'-11 7/8" and the minimum wall thickness is 3'-6 7/8". In addition, the containment has a mild steel liner on the inside. The liner thickness is 1/4" at the base, 3/8" in the cylindrical wall, and 1/2" in the dome.

The containment is reinforced with ASTM A615 grade 60 reinforcing bars of various sizes, mainly #18, #14 and #11. The specified yield strength for the reinforcing bars is 60 ksi. Median yield stress and lognormal standard deviation obtained from the results of tensile tests are shown in Table 3.1.[2] For the ultimate strength of #18 reinforcing bars a mean value of 109 ksi and a CoV of 0.025 has been reported.[2] The liner steel conforms with ASME SA516 grade 60 for which the specified yield strength is 32 ksi. The mean yield stress was found to be 45.4 ksi with a CoV of 0.042.[2] At 271°F a mean yield stress of 40.5 ksi and a CoV of 0.065 are reported.[2] The mean ultimate strength at 271°F was estimated to be 59 ksi and with a CoV of 0.09.[2]

The primary membrane reinforcement in the cylindrical wall is divided into two equal groups placed near the inside and outside faces of the containment wall. Each group consists of two layers of hoop bars and one layer of meridional bars as shown in Figure 3.2. Since the cylinder basemat intersection is subjected to high bending moments and shear forces, secondary meridional reinforcement is placed in this region (See Figure 3.2). In addition, two layers of seismic rebars inclined 45° to the vertical axis are placed near the outside surface of the cylinder wall. Shear ties are also placed in the cylinder near the cylinder-basemat intersection (See Figure 3.2). Major reinforcement details for the containment wall are summarized in Table 3.2.

The dome reinforcement follows the cylinder reinforcement until 9.4° above the springline. Between 9.4° and 79.2° the hoop reinforcement is reduced to one #18 bar near each face. Above 79.2° the hoop reinforcement is terminated and the reinforcement pattern is orthogonal. The meridional cylinder reinforcement is continued to 60° above the springline with an increase in its density as the elevation increases. Above 60° every alternate meridional rebar is terminated, and they are bent such that the reinforcement pattern near the dome apex is orthogonal. Details of the dome reinforcement are shown in Table 3.3.

Concrete with two design strengths were used in the Seabrook containment building. A 4,000 psi design strength concrete was used for the basemat, for the cylinder near the intersection with the basemat, and for both the cylinder and dome near the dome-cylinder intersection. In the cylinder and dome portions where primary membrane behavior is expected a 3,000 psi design strength concrete was used.

The median and lognormal standard deviation for the 4,000 psi and 3,000 psi design strength have been reported for 28-day old cylinders and for aged concrete.[2] These quantities are given in Table 3.4 obtained from Ref, 2.

Seabrook Containment Model

A finite element model of the Seabrook containment was developed to be used with the computer code NFAP.[3] The model is shown in Figure 3.3 and is based on an axisymmetric idealization of the geometry, which is considered a good approximation for a structural failure evaluation under axisymmetric pressure loads. The containment finite element model consists of 408 eight-noded isoparametric elements and 1401 nodes. A set of nonlinear spring elements with a bilinear stress-strain law are used to model reinforcing details such as shear ties. In addition, a set of nonlinear spring elements with high compressive stiffness and zero tensile stiffness are placed under the basemat to model the foundation conditions.

Throughout the cylinder wall and dome the model has 8 layers of eight-noded elements across the wall thickness, as shown in Figure 3.3. Six layers of elements were used through the thickness of most of the basemat. The element layers and its properties were chosen to represent separately the liner, the plain concrete, and the reinforced concrete with hoop, meridional and diagonal rebars. Spacing and sizes of the layers have been chosen in order to model the actual rebar placements as close as possible. This is particularly pertinent at the cylinder-basemat intersection where high bending moments and shear forces will develop. In addition to these criteria, the modeling requirements commonly used with finite element analysis were also taken into consideration.

The inelastic behavior of the plain concrete is described by the Chen and Chen elastic-plastic-fracture model.[3] Material properties for this model were estimated from the aged (as built) concrete properties and are shown in Table 3.5. Post-cracking behavior of the concrete was modeled using a normal stiffness reduction factor α of 10^{-4}, and a shear stiffness reduction factor $\beta=0.5/(\varepsilon_1/\varepsilon_{to})$, where ε_1 is the principal normal strain normal to the crack and ε_{to} the tensile strain at crack initiation.[3] The shear stiffness reduction factor is limited to a value not less than 0.10 to account for the cummulative effect of interface shear transfer and dowel action. The normal stiffness reduction factor α reduces the normal stress diagonal element in the stress-strain matrix, and β reduces the shear stress diagonal element in the stress-strain matrix. The tension stiffening effect was modeled with a factor -0.1, which multiplies the concrete Young's modulus.

The elastoplastic behavior of the reinforcing bars and liner steel was modeled by a bilinear stress-strain curve and a Von Mises plasticity model with isotropic hardening. Since the #18 reinforcement bars provide most of the reinforcement, the mean material properties for these bars were used for all reinforcing bars. For the liner, the mean properties at 271°F were used. The plain concrete properties used are shown in Table 3.5.

Loads included in the analysis are the dead weight of the containment and internal pressure. The dead weight is applied to the containment in the first load step at the beginning of the analysis, while the pressure load is applied to the containment in small increments (5.0 psig) in order to detect the initiation of nonlinear concrete behavior and concrete cracking. Once the concrete cracks its stiffness in the direction normal to the crack plane is reduced by the factor α defined above, and the released stresses are redistributed to the reinforcing steel. As the pressure load is increased the next nonlinearity is the yielding of the liner steel. At this internal pressure the containment is cracked in both the cylinder and dome, and some flexural cracking has been initiated in the cylinder-basemat intersection region. At this stage the load increments are necessarily small (2.0 psig) in order to accurately predict the cracking pattern and stress redistribution. Thus, it becomes possible to detect if a failure mechanism may develop at the base of the cylinder before yielding of the primary membrane reinforcement.

Based on the results of the analysis described above, at a pressure of 154 psig yielding of the internal layer of hoop reinforcing is initiated. At a pressure of 157 psig the yielding reaches the external layer of hoop reinforcing and extends over a large portion of the cylinder wall. Up to this pressure the analysis does not predict a shear failure at the base. At pressures above 154 psig it is observed that as the rate of growth of the radial displacements with internal pressure increases, the shear force also increases at a higher rate. Thus it appears that a shear failure at the base may develop at a pressure slightly above 157 psig. Since general yielding has been reached the load increments necessary to continue predicting containment response must be decreased even further than described above (less than 0.5 psig). Consequently, it was decided not to continue the analysis any further.

3.2 Behavior at Large Deformation

As discussed in Section 3.1, the containment structure is predicted to reach a general yield state at a pressure of 157 psig, which confirms the estimate provided by SMA. As the pressure is increased above this level the containment structure will begin to undergo large de mations. SMA evaluated the behavior of the containment structure at such pressure levels[4] and the results of this evaluation are summarized in Appendix H.1 of the PSA[2].

The hand calculations performed by SMA and used for the probabilistic assessment primarily identify several possible weak places in the structure and determine the corresponding maximum pressure capacity in search of the controlling failure mode. The uncertainties in the results are estimated and identified as coefficients of variation (CoV) to account for both uncertainty and randomness of material behavior and lack of knowledge regarding the exact structural behavior. The break of the liner plate is defined as the failure mode. The capacity of the containment structure is computed in terms of the internal accident pressure it can withstand. Any leakage associated with the pressure level is estimated with a CoV.

Accident scenarios are postulated for both wet and dry containment conditions. The corresponding containment liner temperatures are 271°F for the wet case and 700°F for the dry case. The structural calculations are first performed for the wet case and then modified to reflect the reduced material strengths for the dry case. The various failure modes considered in this analysis are discussed and evaluated below. During the course of its review, BNL observed that the SMA calculations did not show any checker's signature. As a result, PSNH has committed to perform a complete and independent check for all containment strength calculations.

Membrane Failure

The cylindrical wall and the hemispherical dome are assumed axisymmetric and they take the pressure load by membrane action. Both the hoop and the meridional pressure capacities are determined based upon the ultimate strength of reinforcing steel bars, failure of which will lead to a gross containment failure. The median pressure capacities calculated by SMA at 271°F are as follows:

MODE	PRESSURE	CoV
cylinder, hoop tension	216 psig (governs)	.12
dome, hoop or meridional	223 psig	.12
cylinder, meridional tension	281 psig	.12

The govering hoop failure at 700°F corresponds to a median pressure of 198 psig. The above capacities are based on the assumption that the membrane forces are resisted by the reinforcing bars and the liner plate, and not by concrete.

Since the above pressure values correspond to the ultimate strength of reinforcing steel (109 ksi at 7.5% strain), the containment will undergo a great amount of expansion before failure. This is illustrated in Figure 3.4 which plots containment pressure vs. radial displacement of the containment wall as calculated by SMA. At 216 psig the radial displacement of the containment wall away from the base is in excess of 3.0 feet. SMA believes that at this pressure there is a 50 percent chance that the containment liner will remain intact and there will be no gross containment rupture. BNL believes that at these large containment deformations it is difficult to accurately predict the behavior of the containment and that containment liner failure is much more likely. It is also noted from Figure 3.4 that at a pressure of 216 psig the pressure-displacement curve is almost horizontal. Thus, any further pressure capability of the containment would have to be attributed to even greater material strength of the reinforcing steel. Although some reinforcing steel may have a greater strength, BNL believes that for the high strain levels being considered that further consideration must be given to instances of progressive failure of the reinforcing steel.

In the light of the above discussion, BNL considers the 216 psig pressure capacity predicted by SMA to be an upper bound failure pressure. BNL believes that a more suitable median failure pressure should correspond to the pressure level at which the reinforcing steel reaches 1 percent strain (175 psig for the Seabrook containment). Such a level recognizes the ability of the containment to withstand pressures beyond the general yield, but limits the amount of containment deformation to levels more commensurate with the current state of knowledge concerning containment performance. It is of interest to note that if one assumes that the 216 psig level (ultimate strength) represents a 95 percent probability of failure and the 150 psig level (yield strength) represents the 5 percent probability of failure, then the median failure pressure assuming a lognormal distribution is 180 psig. This pressure is more consistent with the pressure corresponding to the 1 percent strain level.

The median failure pressure corresponding to the 1 percent strain level for the dry condition is 168 psig as indicated by PSNH in the response to NRC question 20 (PSNH letter dated October 31, 1986).

Shear Failure of Wall at Base

The shear failure of the cylindrical wall is estimated by SMA at a median pressure value of 319 psig with a CoV of 0.29. This pressure value is determined based upon the yield strength of the reinforcing steel and on the assumption that the critical section will occur at a distance of 0.7 x effective wall thickness above the base. The shear failure corresponding to the ultimate strength of the reinforcing steel is estimated by SMA at a median pressure of 408 psig with a CoV of 0.3.

SMA assigned a large variability to this failure mode due to their uncertainty about the applicability of their elastic analysis when some yielding occurs. However, BNL feels that this failure mode is more critical than assumed by SMA. As discussed in Section 3.1, BNL investigated this mode of failure by means of a non-linear finite element analysis. It was confirmed that such a failure is not expected to occur for pressures up to 157 psig. However, BNL believes that a shear failure mode at the base may develop at a pressure slightly above 157 psig.

Flexural and Shear Failure of Base Slab

The flexural capacity of the base slab is determined by SMA based upon the yield line theory. The median basic capacity is estimated to be 168 psig. However, when the friction and mechanical locking between the base slab and the ring girder of the enclosure building are considered, the median overall capacity is estimated by SMA as 400 psig with a CoV of 0.25. Consequently, it is concluded that flexural failure of the base slab is not a controlling failure mode. The shear strength of the base slab is also calculated considering restraint from the ring girder of the enclosure building. The median pressure capacity is estimated by SMA as 323 psig with a CoV of 0.23.

BNL reviewed the SMA calculations concerning the shear and flexural failure modes of the base slab and agreed that these failure modes would not be controlling.

Containment Deformations

The deformation of the containment is calculated by SMA based upon the assumption that concrete will share tensile load with steel even at the ultimate strength of steel. This assumption presupposes a bond between concrete and the reinforcing bars up to the failure pressure except at the location of the cracks which are postulated to occur with a spacing of approximately 21 inches. The biaxial tension test results presented by Julien and Schultz[5] indicate that concrete will crack at an early steel stress level and the deformation at a high steel stress level is due to steel strain only. The hoop and the meridional reinforceing bars used in these tests were of same diameter as those for Seabrook, namely, No. 18 and No. 14. The concrete cracked at a steel stress of 9.4 ksi and the effect of concrete stiffness disappeared beyond a steel stress of approximately 25 ksi. Consequently, BNL is concerned that SMA may have underestimated the containment deformations corresponding to the containment pressure levels. Since the containment deformations can result in containment penetration failures, an underestimation of the deformations would result in higher predicted failure pressures for the penetrations.

In response to this concern PSNH provided a comparison of the containment pressure displacement curve with and without bond stress (RAI 32, PSNH letter dated November 7, 1986). This comparison is shown in Fig. 3.4. PSNH concluded that an assumption of no bond stress would have no effect on the conclusions of their studies. The effect that this assumption has on the reported capability pressures for critical containment penetrations is discussed in Section 3.3.

3.3 Capability of Penetrations

Many penetrations through the containment shell are provided. These include a few major penetrations such as the equipment hatch, personnel airlock and fuel transfer tube and numerous smaller penetrations accommodating system high energy piping, moderate energy piping, electrical, instrumentation and ventilation lines. All penetrations are anchored to sleeves which are embedded in the concrete containment wall. For the major penetrations, the containment wall is thickened into a hub around the penetration sleeve with the wall hoop and meridional reinforcing members directed past the opening in a continuous fashion and additional reinforcement provided as sleeve anchorage. For each high energy line, the penetration is a forged member, termed a flued head, which forms an integral part of the piping and the containment sleeve which is welded to the containment liner. For all other penetrations the closure is a flat plate welded to the containment sleeve and either welded or connected with a compression fitting to the penetrating element. These flat plateclosures accommodate either single or multiple penetrations.

To assess the capacity of large penetrations, SMA performed an evaluation of the equipment hatch. This hatch is the largest of the large penetrations and was considered to represent the bounding or most critical penetration in this category. In the evaluation, the capacity of the hatch anchorage was determined to be in excess of 300 psi. Possible failure of the liner at the hatch juncture due to sleeve-concrete separation was also evaluated and found to be improbable due to the low magnitude of the predicted liner strain. These evaluations are considered acceptable.

Although the capacity of the fuel transfer tube anchorage was established in the equipment hatch evaluation, the containment wall in the vicinity of this penetration is subject to punching shear failure since it makes hard contact with the fuel transfer building when the containment expands. Using a simple approximation to model the loading and relying primarily on doweling action of the containment reinforcement to resist the load, SMA determined a mean capacity of 320 psig in this failure mode. Acknowledging the approximate nature of this calculation, SMA assigned a large factor of uncertainty to the results. Probabilistic aspects notwithstanding the crude nature of this calculation warranted further verification of the results. Therefore, BNL performed additional calculations for this failure mode to form an independent assessment of the important force-displacement parameters.

In the BNL evaluations an approximate model of the system was again used but this model differed from that used by SMA. The results, although different from those developed by SMA, indicated that no gross deficiencies existed in the SMA calculations. Further, the estimate of the pressure at which contact is made by the containment shell against the fuel transfer building, a controlling parameter in the evaluations, is not subject to the large uncertainty associated with the force-displacement parameters mentioned above. Consequentially the SMA calculations although approximate in nature are considered sufficient to characterize the impact this failure mode has on containment integrity.

To assess the capacity of small pipe penetrations, SMA performed evaluations for three specific penetrations, X-26, X-28 and X-23. X-26 was stated to be a bounding or most critical example of a single pipe moderate energy penetration, X-8 a bounding case for a high energy penetration and X-23 a bounding case for a multiple pipe moderate energy penetration. For each case, simplistic inelastic analysis methods were used to estimate the forces developed at the pipe/penetration interface as a function of containment internal pressure. This data coupled with estimates of the penetration failure characteristics allowed the calculation of the probability of penetration failure as a function of containment pressure in each case. The median failure pressure and the associated median leak areas were 166 psig/0.5in^2, 180 psig/50in^2 and >216 psig/6in^2 for X-26, X-8 and X-23 respectively.

The discussions included in the SMA evaluations provided the basis for the SMA contentions that the penetrations evaluated were the bounding cases for the penetration types considered. Those discussions, however, did not adequately characterize all other penetrations. For this reason SMA was requested to compile a list of all penetrations, categorize them in accordance with design features and demonstrate that the performance of each is adequately represented and bounded by the sample of three evaluated. As a response SMA provided Table 3.6 characterizing all penetrations and the calculations considered to be pertinent for their qualification.

A review was made of the evaluations provided for the bounding cases. In each instance the structural aspects of the calculation seemed appropriate, with the exception noted below, but the assignment of leakage area was considered arbitrary. In addition, for each case, failure was induced by the displacement of the containment shell. Since the correspondence between this displacement and the containment pressure is dependent on the bonding assumption made for the containment reinforcement and since BNL has requested SMA to perform evaluations corresponding to a no bonding assumption (see discussion in Section 3.2) BNL elected to further assess the failure pressure and leakage area for the two penetrations X-8 and X-26. Penetration X-23 was not considered since it exhibited a high failure pressure.

For the high energy penetration, X-8, SMA estimated the median failure pressure to be 180 psig for the wet case with an associated median leakage area of 50 in^2 and a lognormal standard deviation of 0.5. The estimate of the median leakage area was based on an annular gap of 1/2 in, for the full circumference, at the containment sleeve. The estimate of the standard deviation was arbitrary. For the no bond case BNL estimates the median failure pressure to be 167 psig for the wet case and 152 psig for the dry case. BNL accepts the SMA estimate for the median leakage area but disagrees with the assumption regarding the standard deviation. In the absence of more explicit data concerning the behavior of penetration sleeves at failure, BNL believes that an upper bound for the leakage area approaching the total annulus between the pipe and containment sleeve should be considered. Based on these considerations failure of this penetration corresponds to a type B failure for the median leak and a type C failure for the upper bound leak (these agree with the SMA findings).

For the moderate energy penetration, X-26, SMA estimated the median failure pressure to be 166 psig for the wet case with an associated median leakage area of 0.5 in^2 and a lognormal standard deviation of 0.69. The estimate of the median leakage area was based on an annular crack of 0.06 in, the machined clearance between the pipe and the thru hole in the closure plate, extending over 60% of the circumference. The standard deviation was derived by considering 0.02 inches a miniumum crack width and full circumference cracks. For the no bond case it is estimated that the median failure pressure is 159 psig for the wet case and 147 psig for the dry case. Regarding leakage area, the estimate for the median leakage area is accepted but an assumption for the upper bound leakage area equivalent to that recommended for X-8 should be used. Specifically, consideration should be given to an upper bound for the leakage area approaching the total annulus between the pipe and the containment sleeve. Based on these considerations failure of this penetration corresponds to a type A failure for median leak and a type B failure for the upper bound leak (the SMA estimates yielded type A for both conditions).

As noted above, one deficiency was noted in the structural evaluations for the penetrations. In those evaluations only a simple concrete shear cone calculation for a generic case was provided to show that the penetration anchorage capacities were adequate. Owing to the highly cracked state of the containment wall at nigh containment overpressures the reliance on normal concrete action was questioned. SMA was requested to provide additional calculations to demonstrate that small diameter penetration sleeves do not punch through the containment wall under the worst pressure conditions assumed in the analysis. The applicant's response to this request is currently under review by BNL.

Another potential failure mode for the piping penetrations, is the failure of the pipe both inside and outside the containment. This failure mode was evaluated by SMA for the piping in the sample considered most prone to this failure, the piping passing through penetration X-8. The calculations indicated that the piping failure pressure exceeded the penetration failure pressure. Further given the high ductility of the piping, any failures of the piping would have gross distortion, crushing and section collapse associated with them limiting the size of the potential leakage areas. These evaluations seem appropriate for the piping considered.

Other piping penetrations involve the containment ventilation and air purge systems and the containment sump system. The containment ventilation lines have isolation valves both inside and outside of the containment. For these penetrations, the most likely mode of failure is considered to be deterioration of the valve sealant materials at elevated temperatures. In the event of the seal failure of the inner containment valve, the volume between the valves must fill and achieve an elevated temperature before failure at the outer isolation valve can occur. The elapse time for this failure mode is anticipated to be long as compared to other containment failure modes and is therefore considered of little consequence.

The sump system penetration is at elevation -31'6". A review of the drawings originally provided to BNL indicated that the penetration sleeve is welded to the liner at the inside of the containment and to the train A&B sump suction valve containment tank on the outside of the containment. As such the sump suction valve containment tank was considered to be a direct extension of the containment vessel and would have to have sufficient capacity to withstand the temperatures and pressures associated with containment overpressurization events. It appeared that SMA did not consider the tank in their evaluations and therefore, they were requested to perform an assessment of the capacity of the sump containment vessel to the accident conditions. In the response PSNH provided drawings that showed that the tank is isolated from the containment atmosphere by a welded plate closure between the penetration sleeve and the suction line piping. Because of its isolation the sump tank is not subject to accident conditions of pressure and temperature and no further evaluation of its capacity is required.

Another type of penetration is the electrical penetration assemblies. The applicant has indicated that they briefly reviewed these penetrations and that they were not a controlling mode of failure. These types of penetrations have been included in the ongoing SANDIA test program sponsored by the NRC. BNL is currently reviewing the available test data from this program to assess the extent to which the test results will support the applicant's conclusion.

3.4 Summary of Structural Findings

Based on its nonlinear finite element analysis of the Seabrook containment BNL concludes that shear failure at the base of the cylindrical wall is a critical failure mode but would not occur before reaching a pressure of 157 psig.

BNL agrees that the containment structure would reach a general yield state in the hoop reinforcing steel at a pressure of 157 psig and that it is appropriate to consider this pressure as a lower bound pressure for the hoop mode of failure. However, BNL believes that the median hoop failure pressure should correspond to the one percent strain level in the reinforcing steel, which is a pressure of 175 psig. The above pressures are for the wet containment conditions. For the dry containment conditions the corresponding median failure pressure is 158 psig and the lower bound pressure (general yield) is estimated to be 145 psig. This latter value is based on the reduction factor recommendation in Section 11.3.4.1 of PLG-0300.

With regard to containment penetrations, BNL believes that the failure pressures should be based on containment deformations assuming no bond strength between the reinforcing steel and concrete. Based on this assumption BNL estimates median failure pressures for the wet containment condition of 159 psig and 167 psig for penetrations X-26 and X-8. For penetration X-26, BNL agrees that a Type A (less than 6 square inches) leak path is appropriate for the median estimate; however a Type B (6 square inches to about 0.5 square foot) leak path should be considered as an upper bound estimate. For penetration X-8, BNL agrees that a Type B leak path is appropriate for the medium estimate; however, a Type C (greater than 0.5 square foot) should be considered as an upper bound estimate.

For the dry containment conditions, BNL estimated the median failure pressures for penetrations X-26 and X-8 to be 147 psig and 152 psig, respectively. These values are also based on the reduction factor recommended in Section 11.3.4.1 of PLG-0300.

Although BNL has performed some independent calculations to support its conclusions regarding the containment strength, it also relied on the results of calculations performed by PSNH and its contractors. Therefore, BNL recommends that a complete and independent check of all relevant containment strength calculations be performed by PSNH. PSNH committed to such a check in their letter to the NRC dated October 31, 1986.

3.5 References

1. Containment Design Report For Public Service Company of New Hampshire. Seabrook Station Unit Nos. 1 & 2, by United Engineering Constructors Inc., January, 1985.

2. Seabrook Station Probabilistic Risk Assessment, Pickard, Lowe and Garrick, Inc., PLG-0300, Appendix H.1, December, 1983.

3. Sharma, S., Wang, Y. K. and Reich, M., "Ultimate Pressure Capacity of Reinforced and Prestressed Concrete Containments", NUREG/CR-4149, May 1985.

4. Hand Calculations by Structural Mechanics Associates (SMA), Originated by RP, dated December 1982.

5. Julien, J.T. and Schultz, D.M., Tension Test of Concrete Containment Wall Elements, Transaction of the 7th International Conference on Structural Mechanics in Reactor Technology, Vol. J, pp. 237-244.

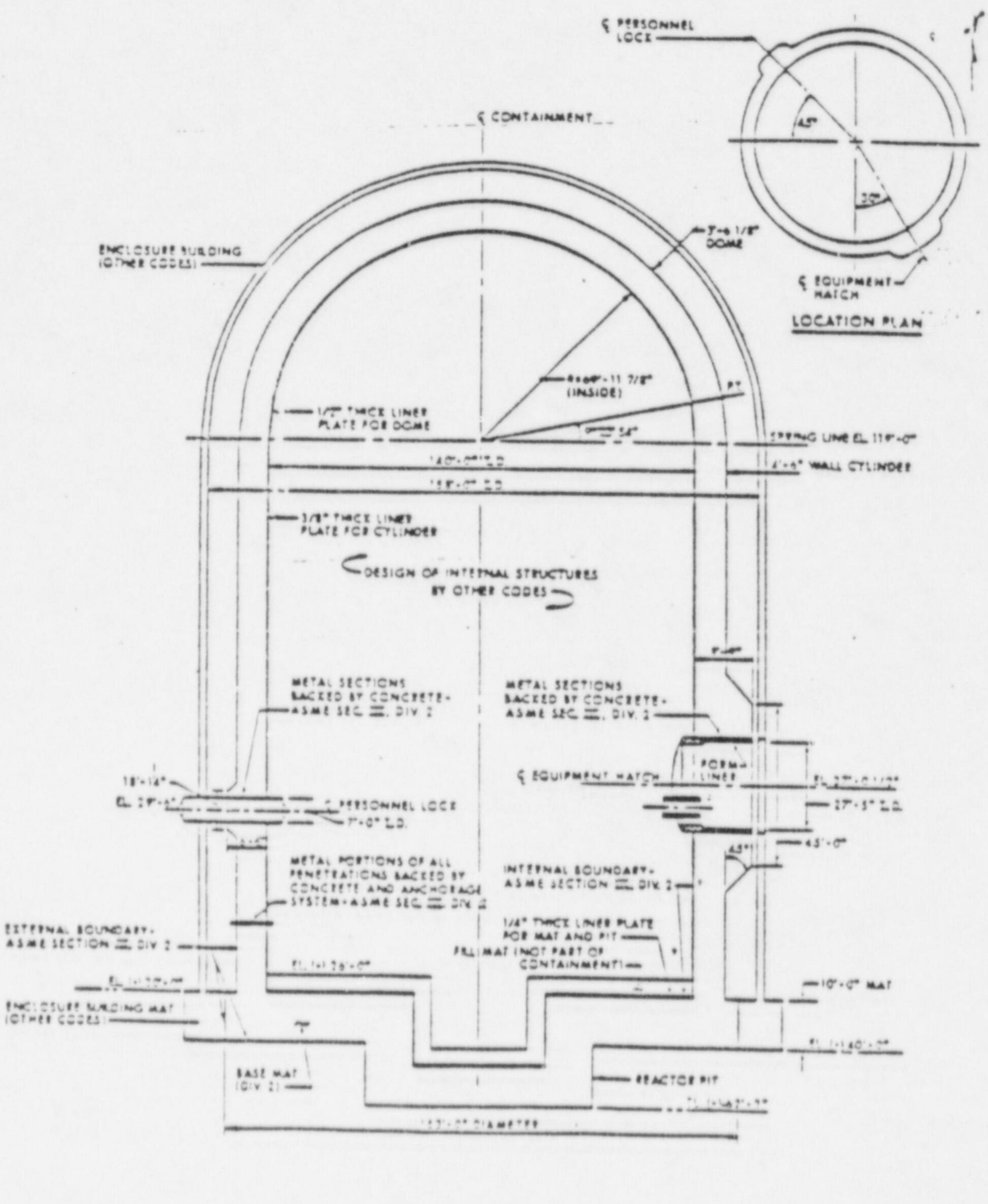


Fig. 3.1 Containment Building Cross-Section

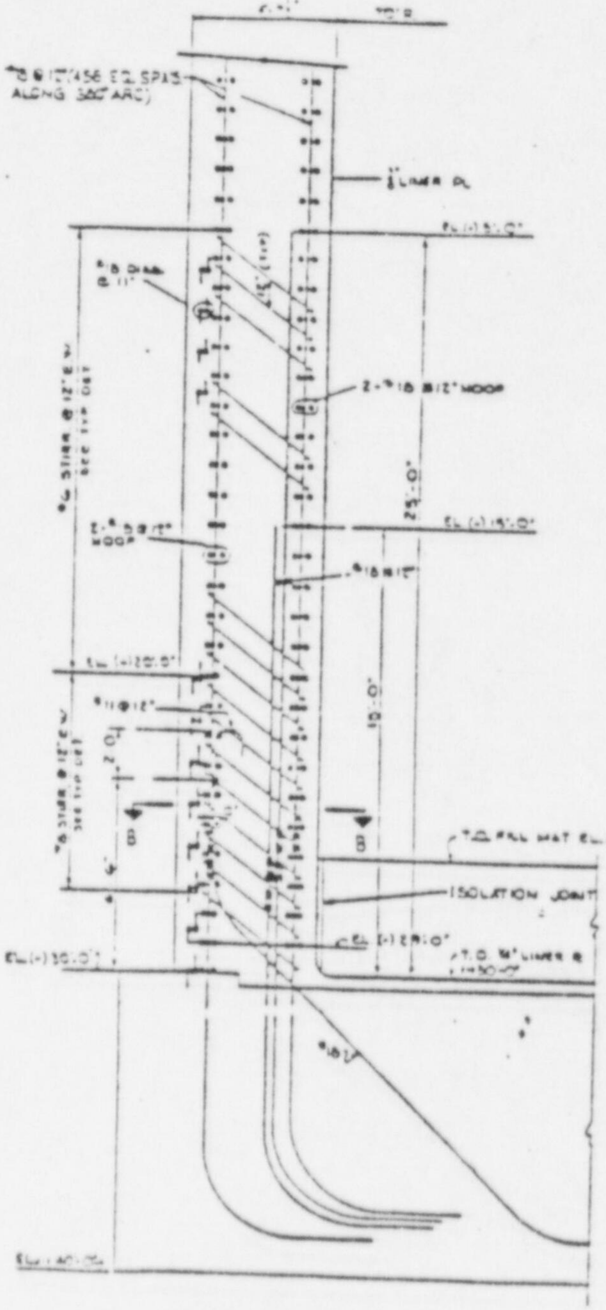


Fig. 3.2 Cylinder Reinforcement

1 LINER

2 & 5 PLAIN CONCRETE

3 & 7 HOOP REINFORCEMENT

4 & 6 MERIDIONAL REINFORCEMENT

8 DIAGONAL SEISMIC REINFORCEMENT

Fig. 3.3 Containment Finite Element Model (NFAP

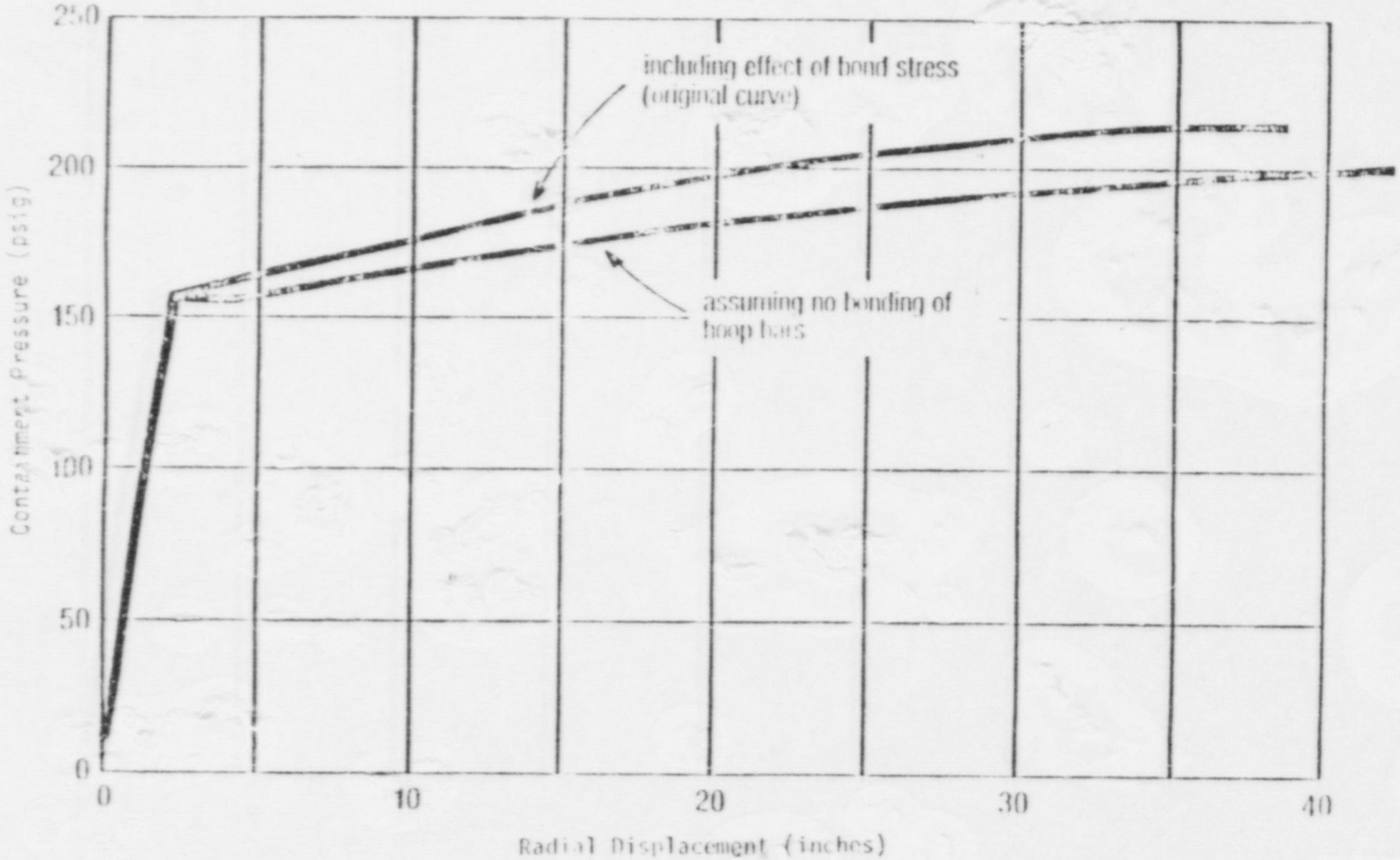


Figure 3.4 - Pressure-Radial Displacement Relation for Containment

TABLE 3.1. Statistics of Rebar Yield Strength for Various Sizes

Bar Size	Median Yield Stress (ksi)	Lognormal Standard Deviation
#4	70.5	.031
#5	67.9	.031
#6	67.9	.033
#7	70.0	.035
#8	69.6	.034
#9	70.3	.040
#10	70.9	.037
#11	70.6	.040
#14	73.6	.031
#18	72.3	.026
#4 - #11*	69.8	.040

*Sizes #4 to #11 taken as a group

Table 3.2 Reinforcement Details of the Containment Cylinder

Elevation	Hoop (Both Faces)	MERIDIONAL Primary (Both Faces)	MERIDIONAL Secondary (Inside Face)	Seismic Diagonal
-30.0' -15'.0	2-#18 @ 12"	#18 @ 12"	2-#18 @ 12"	#18 @ 11"
-15.0' - 5'.0	2-#18 @ 12"	#18 @ 12"	#18 @ 12"	#18 @ 11"
- 5.0' -80'.0	2-#18 @ 12"	#18 @ 12"	---	#18 @ 11"
80.0' 119'.0	2-#18 @ 12"	#18 @ 12"	---	#18 @ 22" #14 @ 22" (alternate)

Table 3.3 Reinforcement Details of the Containment Dome

Elevation	Hoop (Both Faces)	Meridional (Both Faces)	Seismic Diagonal
0°(S.L.) 9.4°	2-#18 @ 12"	#18 @ 12"	#18 @ 22"
9.4° 30°	#18 @ 12"	#18 @ 12"	#18 @ 22" #14 @ 22" (alternate)
30° 45°	#18 @ 12"	#18 @ 10.4"	#14 @ 19"
45° 60°	#18 @ 12"	#18 @ 12"	---
60° 79.2°	#18 @ 12"	#18 @ 12"	---
79.2° 90°	#18 @ 6.4"	#18 @ 6.4"	---

TABLE 3.4. Statistics of Concrete Compressive Strength

Concrete Type	28-Day Old Cylinders		Aged Concrete*	
	Median Strength (psi)	Logarithmic Standard Deviation	Median Strength (psi)	Logarithmic Standard Deviation
3000 psi Design Strength Concrete	4750	0.14	5700	0.17
4000 psi Design Strength Concrete for Containment	5450	0.10**	6540	0.14
4000 psi Design Strength Concrete for Tunnels	5780	0.096	6940	0.14
4000 psi Design Strength Concrete for Other Structures	5590	0.10	6710	0.14

* Median strength and logarithmic standard deviation are obtained by multiplying the 28 day strength by a random factor, which is assumed to be independent of the 28 day strength and has a median of 1.2 and a lognormal standard deviation of 0.10.

** This number was estimated.

Table 3.6 Concrete Properties

MATERIAL PARAMETER	f'c=3000psi	f'c=4000psi
Young's Modulus	4340ksi	4650ksi
Poission's Ratio	0.19	0.19
Yield Strength in Uniaxial Tension	0.233ksi	0.262ksi
Yield Strength in Uniaxial Compression	2.46ksi	2.81ksi
Yield Strength in Biaxial Compression	2.85ksi	3.27ksi
Fracture Strength in Uniaxial Tension	0.54ksi	0.61ksi
Fracture Strength in Uniaxial Compression	5.7ksi	6.54ksi
Fracture Strain in Tension	0.00045	0.0005
Fracture Strain in Compression	0.005	0.005

TABLE 3.6

CHARACTERIZATION OF CONTAINMENT PENETRATIONS

Penetration Closure Type	Penetration Number	Penetration Specifically Analyzed	Qualification Method
I. Flued Head	X-1 to X-8, X-9 to X-15, X-63 to X-66	X-8 (18 inch, sch 100 Carbon Steel)	Report pages H.1-34 to H.1-50
II. Flat Plate Closure Thick Wall - Large Bore Piping	X-25, X-26, X-27	X-26 (4 inch, sch 160, stainless)	Pages H.1-39 to H.1-44
III. Flat Plate Closure Thin Wall - Large Bore and Small Bore Piping	X-16 thru X-24 X-28 thru X-34 X-39, 41, 42, 50, 60, 61, 67	X-23 (12 inch, sch 40, Carbon Steel)	Pages H.1-39 to H.1-43
IV. Flat Plate Closure Thin Wall Piping Multiple Penetration	X-35 thru X-38 X-40, X-43, X-47, X-48, X-49, X-50, X-52, X-57, X-71 thru X-76	X-71	Page H.1-37, H.1-39
V. Fuel Transfer Tube	X-62	X-62	Page H.1-50 to H.1-55

4. CONTAINMENT EVENT TREE REVIEW

In this section the ability of the Seabrook containment to contain severe accident loads is examined. Note that ways in which the containment might be bypassed or not isolated are discussed separately in Section 2. This section, therefore, specifically deals with ways in which severe accident loads might result in loss of integrity of the Seabrook containment. The section is divided into two major parts. Firstly, the impact of uncertainties in containment loads is examined. Secondly, the impact of the BNL review of the containment behavior (in Section 3) is assessed. Finally, the results of the section are summarized in Section 4.3.

The bottom line of the updated assessment of containment performance for Seabrook is given in Table 4.1, which was reproduced from PLG-0465.[1] The conditional probability of a gross early containment failure given a core melt accident was predicted to be 0.001 at Seabrook compared with 0.34 in the RSS.[2] The probability conditional on core melt of early failure in PLG-0465 is an order of magnitude lower than in the SSPSA.[3] This is largely due to the reduced frequency (relative to the value in the SSPSA) of interfacing system LOCAs, which are discussed separately in Section 2.

Note that the containment event tree quantification in the SSPSA[3] was reviewed at BNL in NUREG/CR-4540.[4] The BNL review was limited in scope and did not include at that time a detailed assessment of the Seabrook containment behavior. This has subsequently been performed as part of the present review, and it is documented in Section 3. However, the review of the SSPSA was sufficiently detailed to allow BNL to conclude:

"There is negligible probability of prompt containment failure. Failure during the first few hours after core melt is also unlikely and the timing of overpressure failure is very long compared to the RSS. Most core melt accidents would be effectively mitigated by containment spray operation."

The above conclusions were not based on Seabrook specific calculations performed at BNL but reflected our best judgment based on extensive reviews of other similar containment designs (in particular, our review[5] of the Zion Probabilistic Safety Study[6]). In this section, we critically review the above conclusions based on our current understanding of containment loads and performance during severe accidents.

4.1 Sensitivity to Containment Loads

During a core melt accident, there are several possible types of containment loads that could occur. Each are briefly discussed below:

H_2 combustion: During a core melt accident, significant quantities of H_2 and other combustible gases could be generated. If these combustible gases accumulated to large concentrations before igniting, the resulting deflagration could impose a high pressure/temperature load on the containment. The applicant presented information to indicate that such loads would not seriously challenge the Seabrook containment.

Steam/noncondensible gas partial pressures: Without the containment heat removal systems operating, steam and noncondensible gases generated during the core melt accident would cause the pressure in containment to increase. At the time of vessel failure, there is the potential for the hot core debris to contact water. This contact could result in rapid boiling of the water and a sharp pressure pulse in containment. Limiting calculations were performed to demonstrate that the pressure pulses resulting from simply boiling the water would not pose a threat to the Seabrook containment.

Direct contact of core debris: In some containment designs the containment boundary is directly accessible from the region below the reactor vessel. In these designs the core debris after it melts through the reactor vessel could contact the containment boundary. However, the Seabrook containment design is not susceptible to this mode of containment failure.

Steam explosions: When molten core materials fall into water, experiments indicate that the boiling can become explosive in nature. It has been postulated that these explosions could generate missiles which could directly fail the containment boundary. The potential for an invessel steam explosion to occur and generate a missile capable of failing a containment building was investigated by a group of experts, and the results were published in NUREG-1116.[7] The conclusion of this expert group was that such events have a relatively low probability. The results of this expert group are consistent with the applicant's submittals on Seabrook. The allocation of a very low probability (10^{-4} conditional on core melt) to this event is supported by the authors.

Direct containment heating: This is an area of significant phenomenological uncertainty related specifically to core meltdown with the primary system at high pressure. If molten core materials are ejected from the reactor vessel under pressure, experiments[8] at SNL have indicted that they form fine aerosols, which could be dispersed into the containment atmosphere and directly heat it. An additional concern is the oxidation of the metallic content of the core debris. These reactions are very exothermic and would add an additional heat load to the containment. The pressure rise in containment due to direct heating is directly proportional to the quantity of core debris ejected from the reactor vessel and to how much of this core debris is dispersed into the containment atmosphere. The applicant considered that this phenomena was not a concern at Seabrook because of the design configuration of the containment, which they felt would prevent dispersal of the core materials into the bulk of the containment atmosphere.

The combined probability (conditional on core melt) of the above phenomena resulting in early containment failure was determined by the applicant to be less than 10^{-4} for Seabrook. Within the scope of the present review, BNL has not performed Seabrook-specific containment loads. However, BNL has been involved in updating[9] the risk profile for the Zion plant for input to the NRC's "Reactor Risk Reference Document," NUREG-1150.[10]

The updating of risk for Zion was based on a methodology developed[11] as part of the Severe Accident risk Reduction Program (SAND86-0119) at SNL. This methodology used expert judgment in an attempt to estimate the uncertainty associated with some of the phenomena noted above. The methodology also addressed other areas of uncertainty such as accident sequence probabilities, source terms, and containment performance. The methodology was developed at

SNL specifically for the Surry plant but was extrapolated to Zion at BNL. The Zion plant is very similar to Seabrook in terms of the containment volume to reactor power ratio. Thus, extrapolation of the Zion loads to Seabrook would give some indication of the impact of applying this new methodology to Seabrook. It must be emphasized that this exercise should in no way be interpreted as a Seabrook-specific calculation because the composite containment failure probability distributions for the two plants appear to be quite different. It simply gives some indication of the sensitivity of the Seabrook results to the types of uncertainty discussed in NUREG-1150. It should also be noted that this work is preliminary and has not yet undergone full peer review outside of NRC and its contractors. It is, therefore, subject to revision.

However, application of the new methodology to the Zion plant gives a median probability of approximately 10^{-2} for an early containment failure conditional on core melt. The higher probability of early containment failure is due to more conservative assumptions about direct containment heating and H_2 combustion taken in combination with high steam/noncondensible gas partial pressures. These assumptions were considered by the applicant to be not applicable to the Seabrook containment. However, the applicant was requested at a meeting on November 12, 1986 to assess the impact of the NUREG-1150 assumptions made for Zion on the dose versus distance curves and the risk profiles in PLG-0465. In reference 12, the applicant provided a response to this information request. The conclusions given in reference 12 are quoted below:

(1) The NUREG-1150 assumptions with respect to early containment failure are not credible in the context of the Seabrook specific design and configuration.
(2) Postulating these assumptions in the sense of a sensitivity analysis would not have a large impact on the results presented in the WASH-1400 sensitivity study.
(3) All of the conclusions with regard to emergency planning would still be valid.

The calculations that form the basis for the above conclusions were also presented in reference 12. These calculations were reviewed at BNL and (given the assumptions imposed were found to be correct. The increase in risk estimates with and without evacuation were found to increase by relative small amounts and were still less than the ~~proposed~~ NRC safety goals for individual risk and societal risk. The impact on the dose versus distance curves was not presented in reference 12 but it was assumed to be small. The results in reference 12 are based on the applicant's assumption (refer to Figure 4.1) that for a conditional probability of early failure at Seabrook of 0.01 most of the failures will be type B leaks rather than gross type C failures. This implies that most of the early failures will result in an S6W release rather than the more severe S1W release. The appropriateness of this assumption was reviewed by BNL staff in Section [illegible]. The impact on risk of the BNL review is given in the next subsection.

4.2 Sensitivity to Containment Performance

The composite containment failure probability distributions for various types of containment failure are given in Figure 4.1, which was reproduced from reference 12. Two total failure probability distributions are given for "dry" and "wet" conditions. The "wet" failure distribution corresponds to accident sequences in which the reactor cavity is flooded, the core debris forms a coolable debris bed, and all of the decay heat goes to boiling water. This configuration is a relatively effective way of pressurizing containment (if the containment heat removal systems are not working). However, the temperatures in the containment are relatively benign and would be close to saturation. The "dry" failure distribution corresponds to accident sequences in which the reactor cavity is dry and extensive interactions occur between the core debris and concrete. Under these circumstances, hot noncondensible gases are generated during the core/concrete interactions. These gases result in slower containment pressurization than the "wet" condition but the hot gases heat the containment atmosphere to higher temperatures. Thus, for any given pressure level, the "dry" accidents would have higher atmospheric temperatures than the "wet" accidents, which in the applicant's analysis (refer to Figure 4.1) resulted in a higher probability of containment failure.

In order to assess the potential for early containment failure due to short duration pressure pulses, we considered it appropriate to use the "wet" probability distribution. This is simply because the containment structure would not have been exposed to the higher "dry" atmosphere temperatures prior to the pressure pulse occurring. The structures would therefore not have degraded as a result of high temperatures and the "wet" conditions are more appropriate.

If we, therefore, focus on the "wet" failure distribution (in Figure 4.1) as being more appropriate to determine the potential for early containment failure, then the relative contributions of gross (type C) failures and benign (type B) failures at a conditional probability of 0.01 were determined[12] by the applicant (refer to Figure 4.1). The impact of the above assumptions on the dose versus distance and risk profiles were presented in Section 4.1. In this section we assess how the BNL review of containment behavior (refer to Section 3) might impact the above results.

The median failure pressure based on hoop failure assuming "wet" conditions was estimated to be 175 psig in Section 3. In addition, the possibility of shear failure at the base of the containment at a pressure of 140 psig was also discussed in Section 3. However, the median probability of approximately 10^{-2} for a large early containment failure at Zion (refer to Section 4.1) was based on ~~median~~ a failure pressure of 134 psig. It is not therefore likely that the Seabrook specific failure pressures (estimated in Section 3) would significantly change the conditional probability of early failure at Seabrook. In fact, as the Seabrook failure distributions are higher than the Zion failure distribution, one would expect the conditional probability of early failure to decrease if everything else were the same between the two plants.

However, the Seabrook specific failure pressures discussed in Section 3 are lower than the distributions presented by the applicant in Figure 4.1 and closer to the Zion values. This, in turn, affects the assumption made by the applicant, in reference 12, related to the relative contributions of gross (type C) failures and benign (type B) failures at a conditional probability of 0.01. Therefore, because of the uncertainty associated with the relative contributions of the two failure modes, a limiting sensitivity calculation was performed to assess the impact of assuming that at a conditional probability of 0.01 all failures were gross (type C) failures. The results of the sensitivity are presented in Table 4.2 and Figure 4.2

4.3 Summary

As a sensitivity study, the impact on risk of more conservative assumptions with regard to containment loads and performance during severe accident conditions has been made. The results of the study are shown in Figure 4.2 and Table 4.2. The sensitivity study resulted in a higher conditional probability of early containment failure than considered credible in PLG-0465 (10^{-2} versus 10^{-4}). The higher probability was not based on Seabrook specific calculations but inferred from calculations performed at BNL for a nuclear plant (Zion) with a similar containment volume to reactor power ratio. It must therefore be emphasized that this sensitivity study is not Seabrook-specific and it was performed simply to assess the robustness of the PLG-0465 results to uncertainty in containment loads and performance.

The impact on risk of considering a conditioal probability of 10^{-2} was found to be not significant when the composite containment failure probability distributions developed in PLG-0465 (and reproduced in Figure 4.1) were used. This was because, at a probability of 10^{-2}, most of the containment failures would be leakage failures rather than gross failures. A leakage failure results in lower offsite health effects than a gross failure. However, if the more conservative BNL assumptions regarding containment performance (refer to Section 3) are combined with the higher probability of early containment failure then the dose vs distance and the risk profiles in PLG-0465 were affected. As a bounding sensitivity study, it was assumed that all failures at a conditional probability of 10^{-2} would be gross failures and the results are given in Table 4.2 and Figure 4.2. The impact on the dose vs distance curves is similar to the influence of the more pessimistic assumptions regarding induced steam generator tube rupture (refer to Section 2.3 and Figure 2.4). The similarity of the results is to be expected as, given the limitations of the RSS source term methodology, both types of events had to be conservatively binned into the same source term category (namely S1W). The impact on the risk of cancers and early fatalities of more conservative assumptions regarding containment loads and performance is given in Table 4.2

and again follows similar trends to the SGTR study (refer to Table 2.8). The impact on the risk of cancers is small but the early fatality risk increases by less than a factor of 2 with no immediate protective actions taken.

4.4 References

1. PLG-0465.
2. RSS.
3. SSPSA.
4. NUREG/CR-4540.
5. NUREG/CR-3300, Volume 2. "Review and Evaluation of the Zion Probabilistic Safety Study," July 1983.
6. ZPSS.
7. NUREG-1116.
8. SNL Direct Heating Experiments.
9. NUREG/CR-4551, BNL/NUREG-52029, Volume 5, Draft, September 1986.
10. NUREG-1150, to be published.
11. SAND86-0119.
12. PSNH Letter (SBN-1237), dated November 21, 1986, "Emergency Planning Sensitivity Study," J. DeVincentis to S. M. Long.

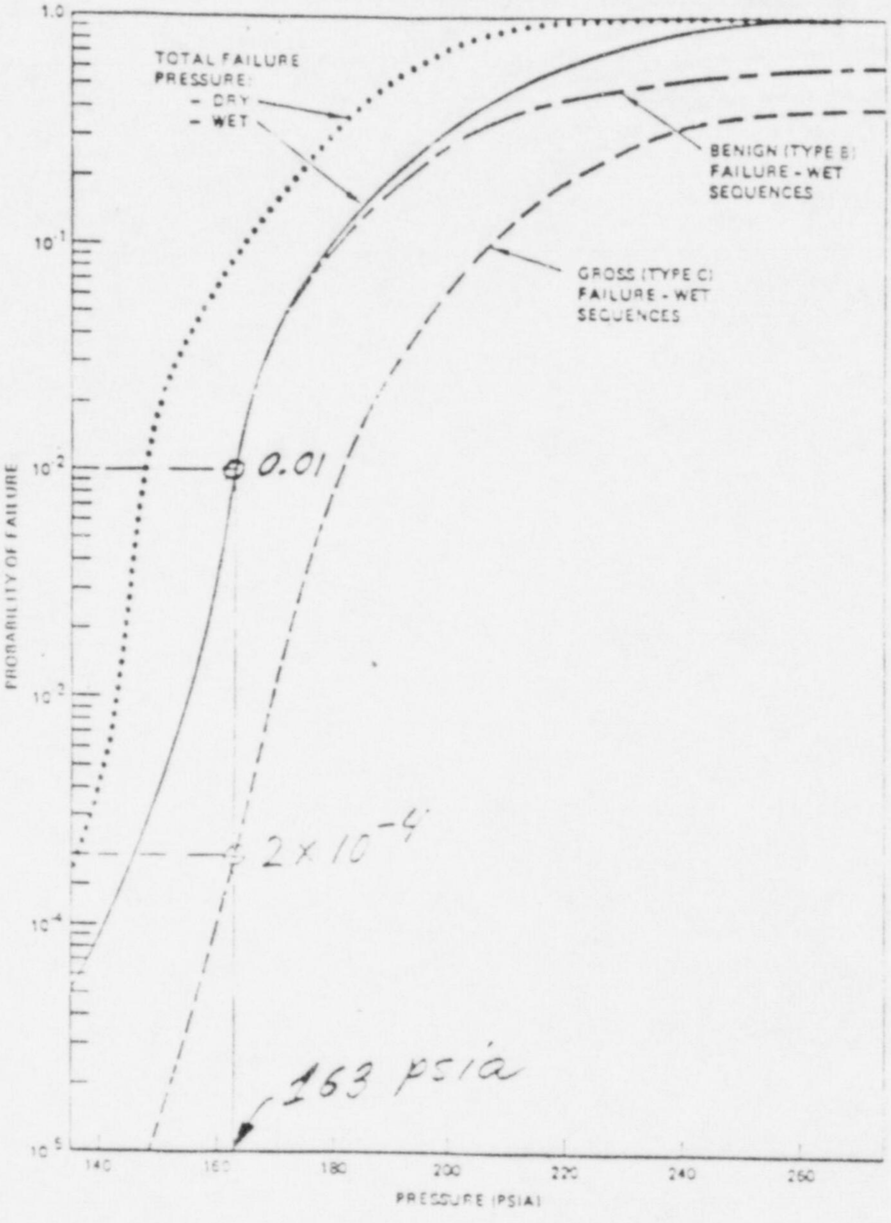


Figure 4.1 Composite containment failure probability distributions for benign failure, gross failure, and total failure. (Reproduced from Figure 11.3-14 of SEN:1037)

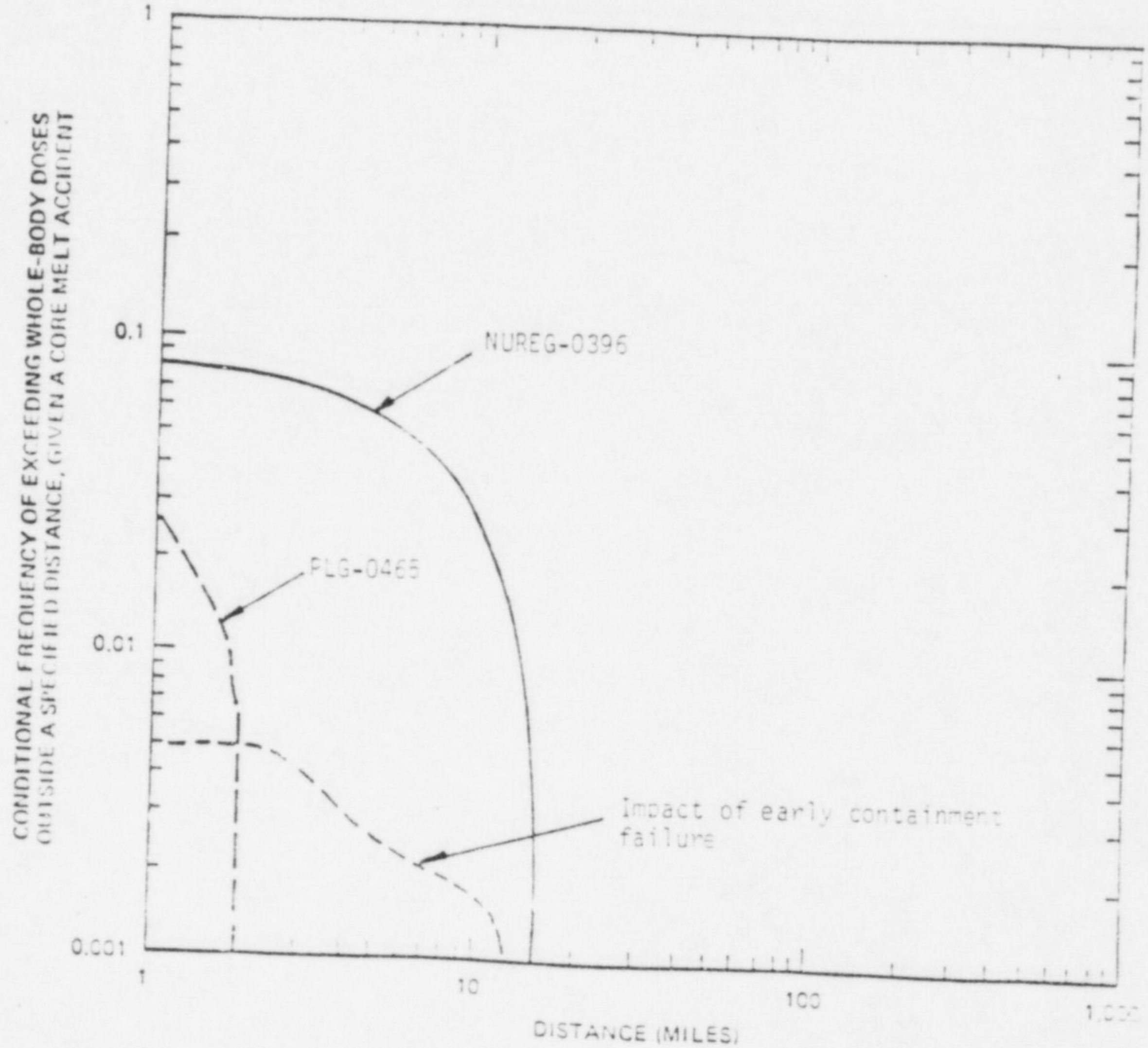


Figure 4.2 Comparison of BNL sensitivity study related to early containment failure with PLG-0465 and NUREG-0396 (200 rem plots with no immediate protective actions).

Table 4.[illegible] Comparison of Core Melt Frequencies and Distribution of Release Types (reproduced from Table 2.2 of PLG-046[illegible])

Risk Parameter	WASH-1400 PWR	SSPSA	Updated Results*
• Mean Core Melt Frequency (events per reactor-year)	9.9-5**	2.3-4	2.7-4
• Percent Contribution of Release Types			
- Gross, Early Containment Failure	34	1	0.1
- Gradual Containment Overpressurization or Melt-Through	66	73	60
- Containment Intact	0	26	40

*Based on RMEPS (PLG-[illegible]).
**Based on WASH-1400 uncertainty ranges.

NOTE: Exponential notation is indicated in abbreviated form; i.e., 9.9-5 = 9.9×10^{-5}.

Table 4.2 Impact of BNL Sensitivity Studies on PLG-0465 Risk Estimates

Sensitivity Study	Absolute Risk per Reactor Year					
	Early Fatalities			Total Cancers		
	No Evac.	One Mile Evac.	Two Mile Evac.	No Evac.	One Mile Evac.	Two Mile Evac.
Original PLG-0465 Results	2.7(-3)*	3.6(-4)	2.4(-5)	1.5(-2)	1.4(-2)	9.2(-3)
Containment Loads and Performance	4.7(-3)	2.4(-3)	2.0(-3)	1.8(-2)	1.7(-2)	1.2(-2)

*2.7(-3) = 2.7×10^{-3}

5. REVIEW OF SOURCE TERMS

In this section the fission product source terms developed for PLG-0465[1] are reviewed. The source terms used in PLG-0465[1] are reproduced in Table 5.1. The probabilities of each of the source terms are given in Table 5.2.

5.1 Fidelity to WASH-1400 Methodology

The fission product release fractions in Table 5.1 were determined by the applicant using RSS[2] methodology. These source terms are consistent with the point-estimate source terms used in the SSPSA.[3] The SSPSA source terms were reviewed by BNL in NUREG/CR-4540[4] and they were found to be reasonable given the limitations of the RSS methodology (principally the CORRAL[5] code).

When reviewing the PLG-0465 source terms, questions were raised and transmitted to the applicant. One question related to release category S2W (refer to Table 5.1). The fractional release of the noble gases for S2W was quoted as 0.123 whereas the release fractions for Cs and Te were quoted as 0.2 and 0.19 respectively. It appeared inconsistent to release more aerosols (Cs and Te) than noble gas, and the applicant was requested to either explain the predictions or provide revised source terms. In reference 6, the applicant provided a response to this question. Basically, the noble gas release in Table 4.3 (reproduced in Table 5.1 of this report) of PLG-0465 was a misprint. The noble gas release fraction for the S2W-3 release phase should have been 0.23 rather than 0.023. However, as the S2W-3 release phase occurs very late (approximately 20 hours) after reactor scram, the impact on risk of this larger noble gas release would be very small. Calculations performed at BNL have verified that the impact of increasing the noble gas release in the S2W-3 phase on both the dose versus distance curves and the risk profiles is negligible.

5.2 Credit for Scrubbing of Submerged Releases

Another question related to the credit assumed for the interfacing system LOCA source term in which the break location was assumed to be submerged under water (S7W in Table 5.1). In PLG-0465, the source term mitigation resulting from a subcooled 30 foot deep pool was modeled as a decontamination factor (DF) of 1000 for all release fractions except the noble gases. In the RSS a decontamination factor of 100 was used for fission product scrubbing in a subcooled pool. Thus, in order to be consistent with the RSS methodology it appeared that a lower DF should be used. However, if the pool were indeed subcooled, calculations at BNL indicated that using a DF of 100 rather than 1000 would have no impact on the risk calculations pre-sented in PLG-0465.

A more important question was whether or not the pool would be subcooled or saturated. In the RSS, no credit was given for fission product scrubbing in a saturated pool, and therefore, the applicant was requested to provide justification for the subcooled assumption. In reference 7, the applicant provided a response to this question. Arguments were provided to indicate that the pool would be at least 10oC subcooled and that this degree of subcooling together with the large pool depth was sufficient to justify a DF of 1000 rather than a DF of 100 as used in the RSS. However, the objective of the question was primarily to determine if the pool would be subcooled and based on the response, this appears to be the case. BNL had already concluded that changing the DF from 1000 to 100 would not change the dose versus distance nor the risk profiles. Finally, the conclusion (given in reference 7) that even if pool decontamination were completely ignored, that the dose versus distance and the risk profiles in PLG-0465 would not be significantly effected was examined at BNL. Calculations at BNL indicated that if the frequency of interfacing system LOCAs reported in reference 1 was used, then the conclusion was correct. The appropriateness of the interfacing system LOCA frequency was reviewed in Section 2.

5.3 Summary

In summary, the fission product source terms used in PLG-0465 appear in general to be consistent with the approaches used in the RSS. The misprint in Table 4.3 of PLG-0465 related to the fraction of noble gas release was found by the applicant (and confirmed by BNL) to have negligible impact on the risk profiles or the dose versus distance profiles reported in PLG-0465. Thus, the corrected noble gas-release fraction in the S2W release category would, by itself, not change the conclusions in PLG-0465. In addition, the argument presented by the applicant that the water in the RHR vault is sufficiently subcooled to warrant consideration of significant decontamination appears reasonable. Finally, the statement in the applicant's response that even if pool decontamination had been ignored, the risk profiles or dose versus distance profiles would not change significantly was confirmed at BNL. Note, however, that this conclusion is based on the frequency estimates for interfacing system LOCAs in PLG-0465. The influence of the BNL review of interfacing system LOCA events at Seabrook is given in Section 2.

5.4 References

1. PLG-0465.
2. RSS.
3. SSPSA.
4. NUREG/CR-4540.
5. Corral Code.
6. PSNH Letter (SBN-1234), dated November 17, 1986, "Response to Request for Additional Information (RAIs)," J. DeVincentis to S. M. Long.
7. PSNH Letter (SBN-1227), dated November 7, 1986, "Response to Request for Additional Information (RAIs)," J. DeVincentis to S. M. Long.

Table 5.1 Release Categories for Seabrook Station Based on WASH-1400 Source Term Methodology

Release Category	Release Time (hours)	Release Duration (hours)	Warning Time (hours)	Energy Release (MCA/S)	Release Fractions							
					XE	O.I.	I-2	CS	TE	BA	RU	LA
S1W	2.5	0.5	1.0	11.9	0.9	7-3	.7	.5	.3	.06	.02	4-3
S2W-1 S2W-2 S2W-3	4.8 6.8 19.8	2.0 4.0 18.0	0.5 2.5 15.5	0 0 0	.03 .07 .023	2.1-4 5.0-4 1.6-3	4.3-3 1.3-3 2.3-3	.023 .048 .126	4.2-3 .039 .117	2.8-3 5.5-3 .014	8.4-4 3.4-3 .011	8.4-5 5.2-4 1.9-3
TOTAL	4.8	24.0	0.5	0	.123	2.3-3	7.9-3	.20	.19	.022	.015	2.5-3
S3W	6.0	24.0	2.0	0	4.7-4	3.3-6	3.2-5	1.7-4	1.5-4	1.9-5	1.2-5	2.0-6
S6W-1 S6W-2 S6W-3	1.75 2.75 15.75	1.0 4.0 18.5	1.5 2.5 15.5	0 0 0	.15 .42 .32	1.1-3 2.9-3 2.2-3	.10 .07 .01	.11 .19 .13	.02 .063 .32	.014 .022 .011	4.1-3 .009 .020	4.1-4 .001 3.8-3
TOTAL	1.75	23.5	1.5	0	.9	6.2-3	.18	.43	.40	.047	.033	5.2-3
S7W	8.5	7.0	2.0	0	.9	7-6	7-4	5-4	3-4	6-5	2-5	4-6

NOTE: Exponential notation is indicated in abbreviated form; i.e., 7-3 = 7×10^{-3}.

Table 5.[illegible] Revised C-Matrix for New Source Term Categories

Plant Damage State (frequency)	Source Term Category					
	S1	S2	S3	S5	S6	S7
1F (2.0-8)					1.0 (2.0-8)	
1FV (4.6-9)	1.0 (4.6-9)					
1FP (1.4-6)		[illegible]				
1FPV (2.7-8)						1.0 (2.7-8)
2A (1.9-6)	3.4-5 (6.5-11)	1.4-[illegible] ([illegible])	1.0-2 (1.9-8)	0.99 (1.9-6)		
3D/7D (3.8-6)	2.0-5 (7.6-11)	[illegible]	0.95 (3.6-6)	0.05 (1.9-6)		
3F/7F (3.0-7)					1.0 (3.0-7)	
3FP/7FP (1.9-8)		[illegible]				
4A/8A (1.1-4)	3.1-6 (3.3-10)	[illegible]	5.0-3 (5.5-7)	0.995 (1.1-4)		
7FPV (1.2-8)						1.0 (1.2-8)
8D (1.0-4)	1.1-6 (1.1-10)	3.1-5 (3.2-9)	0.9999 (1.0-4)			
Total Frequency	5.2-9	2.0-5	1.4-4	1.1-4	3.2-7	3.9-8

NOTES:

1. Exponential notation is indicated in abbreviated form; i.e., 2.0-8 = 2.0 x 10^{-8}.
2. Numbers inside parentheses are unconditional frequencies (events per reactor year) based on mean values. Numbers not inside parentheses are conditional frequencies of source term categories, given the indicated plant damage state, also based on mean values. Median values of source term categories are presented in Section 3.

6. SITE CONSEQUENCE MODEL

6.1 NUREG-0396 Basis

NUREG-0396[1] introduced the concept of generic Emergency Planning Zones (EPZ) as a basis for the planning of response actions which would result in dose savings in the immediate vicinity of nuclear facilities in the event of a serious power reactor accident. The actions would be triggered if projected radiation doses to an individual would exceed Protective Action Guides (PAGs), as discussed and referenced in NUREG-0396, although ad hoc actions could be taken at any time. The PAGs are 1 to 5 rem whole body dose and 5 to 25 rem thyroid dose but are not intended to represent acceptable dose levels. Furthermore, protective actions may not assure that PAG levels can be prevented.

It was concluded in NUREG-0396 that the objective of emergency response plans should be to provide dose savings for a spectrum of accidents that could produce offsite doses in excess of the PAGs since no specific accident could be identified as the one for which to plan.

The most important guidance for emergency planners is the size of the EPZ. Based on factors that included risk, probability, and accident consequences, it was judged that a generic distance of about 10 miles was appropriate for core melt accidents. The less severe accidents would not have consequences in excess of the PAGs beyond this distance, whereas the more severe accidents would not in general cause early injuries or deaths beyond this distance. Hence, protective actions were judged to be most useful within this distance.

NUREG-0396 used the accident release categories of the Reactor Safety Study[2] (RSS) to compute the risk of exceeding various dose levels in the absence of protective actions for a spectrum of accidents. The RSS accidents and their median probabilities are given in Table 6.1. Using the original RSS consequence model (CRAC)[3] and the accidents PWR1 through PWR7, a 200 rem whole body risk curve was constructed, shown as the heavy line marked 0396 in Figure 6.1. This is the level at which serious injuries and some deaths can occur.

However, CRAC did not compute the 200 rem risk directly. The authors of NUREG-0396 had to interpolate to obtain the 200 rem curves. Interpolation was performed for each component accident to obtain the conditional risk given the accident. Then the conditional risk at various distances was multiplied by the probability shown in Table 6.1 and divided by the core melt probability of 5×10^{-5} per reactor year. The results for each were summed to give the overall risk of the accident spectrum. Each risk was computed using about 100 weather samples from a typical series of New York City hourly annual weather data with the assumption that people would follow normal activities for one day following arrival of the first plume to reach their location. That is, people would stay at their original location and receive groundshine doses for 24 hours.

BNL recomputed the 200 rem dose vs distance curve using an updated version of CRAC, called CRAC2,[4] and a more detailed grid. The conditional component risks were computed directly (no interpolation required) and the results for each of the WASH-1400 release categories are shown in Figure 6.1. PWR6 and PWR7 did not exceed 200 rem.

The overall 200 rem risk curve was then computed using a core melt probability of 6×10^{-5} per reactor year, which is the sum of the probabilities for PWR1 - PWR7 given in Table 6.1. The curve gives higher risk for 1-3 miles and lower risk for 4-10 miles than NUREG-0396, but the curve still drops sharply beyond about 10 miles. It should be noted that most of the core melt probability comes from PWR7 which does not contribute to the 200 rem curve. It can be concluded that the approximations used in NUREG-0396 are not substantially different from the more detailed calculations done by BNL using CRAC2.

6.2 Consequence Modeling

The applicant used the CRACIT[5] code for their consequence assessments in PLG-0465.[10] In this section CRACIT predictions are compared against consequences models currently being used by the NRC and their contractors, namely CRAC, CRAC2, and MACCS.[6] The factors involved in consequence modeling are discussed in Appendix 6 of the RSS. All codes compute early and delayed health effects from cloudshine, inhalation, and groundshine. The early health effects are based on data from the Marshall Islands, bomb tests, clinical data from radiation therapy, and lab animals (particularly for lung data). The three CRAC models (CRACIT, CRAC2, and CRAC) use a stepwise linear function with a threshold dose for early effects as discussed in the RSS. The MACCS code (recently developed at Sandia National Lab) uses a hazard function approach without a threshold as discussed in NUREG/CR-4214. The latent effects in the CRAC models are calculated from the BEIR-1 report[8] which uses Japanese data plus a modification to the linear dose response curve to account for reduced effectiveness at low doses. MACCS uses the BEIR-3 model[9] which is a linear quadratic dose response model with absolute risk of cancer for some organs such as bone marrow and relative risk for other organs, depending on population makeup. In addition, CRAC and CRAC2 allow only a one "puff" release of radioactivity whereas CRACIT and MACCS allow "multipuff" releases. There are also other differences in the codes, such as the shape of the plume, dry and wet deposition of particulates, weather sampling, resuspension, etc. which can account for differences of a factor of two in the results.

BNL used the MACCS code and the source terms defined in Table 5.1 to review the calculations presented by Seabrook in PLG-0465[10] using CRACIT. The comparisons are based on the 200 rem dose probability vs distance curves using the source terms and weather data supplied by Seabrook. In addition, BNL calculated the individual risk of exceeding 5, 25, and 300 rem to the thyroid and the individual risk of death as a function of distance. In all cases, it was assumed that the population was exposed to one day of groundshine following arrival of the first plume segment. They would also be exposed to other plumes that arrived at their location within the 24 hours.

6.2.1 Whole Body Dose Vs Distance

The MACCS code does not have an organ defined as "whole body" so red marrow was used as a substitute. In CRAC2 calculations it was found that the red marrow dose was about 30% higher than the whole body dose. Also, early health effects are sensitive to the red marrow dose. Thus, the red marrow dose is a good substitute.

MACCS does not directly calculate the risk of a dose vs distance, and it was necessary to define an appropriate risk function to obtain this curve. In the MACCS calculations, the risk of exceeding 200 rem to the red marrow was set to zero if a weather sample yields a mean dose less than 200 rem and one if a weather sample yields a mean dose greater than 200 rem. Thus the mean risk is dependent upon the annual weather data. The risks were calculated for the Seabrook release categories S1W (one puff), S2W (3 puffs), and S6W (3 puffs). The results are given in Figures 6.2, 6.3, and 6.4. The conditional probability of .001 risks extend to 25 miles for S1W, to 2 miles for S2W, and to 4 miles for S6W. These distances are somewhat less than those calculated by Seabrook.

6.2.2 Thyroid Dose Vs Distance

The thyroid doses were not presented by Seabrook in the reviewed report but were discussed at some length in NUREG-0396. Hence, BNL calculated the risk of exceeding the dose levels of 5, 25, and 300 rem to the thyroid, as was done in NUREG-0396 for Seabrook release categories S1W, S2W, and S6W. The results are given in Figures 6.2, 6.3, and 6.4. The results were truncated at 30 miles. The risk of exceeding 5 rem remained above 90% for all three release categories. The 300 rem curve shows a sharp drop at less than 10 miles for S2W and at about 15 miles for S6W. The curves were obtained by the same hazard function definition technique as discussed in Section 6.2.1.

6.2.3 Risk of Early Fatalities

MACCS uses the hazard function approach to calculate early fatalities as discussed in NUREG/CR-4214. First, the cumulative hazard is calculated as:

$$H = \ln(2)\,(D/D_{50})^{v} \qquad (6.1)$$

where D is the dose and D_{50} is the dose required for producing an effect in 50% of the exposed individuals, and v determines the steepness of the dose effect curve. The fatality risk is then given as:

$$Risk = 1 - e^{-(H_1 + H_2 + H_3 + H_4)} \qquad (6.2)$$

where H_1 is for red marrow, H_2 is for lungs, and H_3 and H_4 are for the lower large intestine and small intestine. The risk is assigned a threshold of .005.

In CRAC2 and CRACIT, the dose response is piece-wise linear due to irradiation of the bone marrow, lung and GI tract. The total risk is then:

$$R = R_1 + (1-R_1)R_2 + (1-R_1)(1-R_2)R_3 \tag{6.3}$$

where R_1, R_2, and R_3 are the risks to the three organs, respectively. MACCS gives somewhat higher risk, principally because the lung dose is now considered more effective in producing fatalities, and also because the hazard function gives some risk at lower doses.

The effect of the code differences is that MACCS predicts a higher probability of a small number of deaths while CRAC2 predicts a higher probability for large numbers of deaths. This is shown in Figure 6.8 from a comparison calculation performed by Sandia National Laboratory for a severe ground level release. This is for a uniform population distribution without evacuation. However, MACCS can predict substantially more early deaths when evacuation is modeled since the lung dose usually becomes dominant, but evacuation scenarios are not considered in this review.

BNL calculated the individual risk of fatalities versus distances for S1W, S2W, and S6W as shown in Figures 6.2, 6.3, and 6.4. It can be seen that the risk is similar to the 200 rem curve, but is not directly correlated because of the nonlinear interactions in the above equations.

6.3 Comparisons of Results

6.3.1 Results of Seabrook Study

BNL used as a basis for comparison the 200 rem risk of whole body dose as calculated by Seabrook using CRACIT and 200 rem red marrow risk using MACCS at BNL. This is the risk to a hypothetical individual located at a particular distance and actual population distribution is not considered. The BNL calculations were performed with meteorological data supplied by Seabrook. Projected numbers of fatalities were not computed since BNL didn't have the actual Seabrook population data.

The Seabrook calculations for accidents S1W, S2W, and S6W are shown in Figures 6.5, 6.6, and 6.7 with the corresponding BNL results superimposed. For S1W, MACCS predicts slightly higher risk at less than 8 miles and much lower risk beyond 12 miles. The differences may be partly explained by differences in weather sampling methods and plume rise formulations, since this is a high energy release. However, the differences in the tail of the risk curve is not considered by BNL to be significant considering the overall uncertainties in the calculations.

For S2W, the MACCS code again predicts higher risk in close and a somewhat sharper dropoff in the tail. However, the difference is that MACCS predicts a risk of .001 at 2 miles whereas the Seabrook results show this risk at 2.5 miles.

For S6W, the BNL results are close to those of Seabrook as in the case of S2W. MACCS predicts .001 risk at 4 miles whereas Seabrook predicts this risk of 200 rem at 6 miles.

In summary, BNL feels that the consequence modeling is fairly presented by Seabrook and that the relatively small differences computed by BNL are probably explained by [illegible].

6.4 Sensitivity Studies

Two categories of sensitivity studies have been performed as part of the BNL review. First, sensitivity calculations were performed to assess the affect of the duration of fission product release on the dose vs distance curves presented in PLG-0465. These sensitivity calculations are discussed in Section 6.4.1. Second, the impact on the dose vs distance and risk estimates in PLG-0465 of the various concerns raised by the BNL review was assessed in each section of this technical evaluation report. These revised risk estimates are summarized in Section 6.4.2.

6.4.1 Sensitivity of Results to Multipuff Release

BNL performed sensitivity calculations with regard to the multipuff releases and the release duration for S2W and S6W. The results are given in Figures 6.6 and 6.7, respectively using the one puff release categories defined by Seabrook. In both cases it was found that a one puff release increased the risk and also that a shorter duration of the release (0.5 hours) further increased the risk. For the one puff, 0.5 hour release S2W, the 200 rem .001 risk distance increased from 2 miles to 7 miles and the S6W distance increased from 4 miles to 15 miles. This demonstrates that a long release duration leads to a greater plume dilution and less risk at larger distances. Hence, in order to have confidence in the Seabrook calculations, one must have confidence that the releases will occur with rates and durations similar to those assumed by Seabrook.

6.4.2 Sensitivity of Results to BNL Review

The Seabrook study (PLG-0465) identified the following three areas as being the most influential in calculating the Seabrook risk estimates:

- The effectiveness of the Seabrook Station primary containment to either remain intact or to maintain its fission product retention capability for periods much longer than required for even delayed, ad hoc protective actions.

- A more realistic assessment of the strength and failure modes of the Seabrook containment than was possible within the state-of-the-art of PRA when the RSS was completed.

- A more realistic treatment of the initiation and progression of interfacing systems LOCA sequences.

Thus, the BNL technical evaluation focused on the areas that were identified in PLG-0465. The approach taken was to perform sensitivity studies to assess the impact on the results in PLG-0465 of the BNL review of these areas. The sensitivity studies used the conditional risk indices provided in

PLG-0465 (and supporting documentation) to assess how changes in the probability of accident sequences and containment failure modes would change the risk estimates in PLG-0465. The sensitivity studies calculated revised 200 rem-dose versus distance curves and the mean absolute risk of early fatalities and total cancer fatalities. The results of the sensitivity studies are given in Table 6.2 and Figure 6.9. The dose vs distance curves in Figure 6.9 can of course be directly compared with the dose vs distance curves presented in PLG-0465. However, the mean absolute risk numbers in Table 6.2 cannot be directly compared with the information on individual risk provided in PLG-0465. The mean absolute risk number in Table 6.2 would have to be converted into individual risk of fatlaities in the population within 1 mile of the site boundary before direct comparison within information in PLG-0465 could be made. However, the information provided in Table 6.2 is a useful indication of how the PLG-0465 risk levels in PLG-0465 would change if recalculated using the assumptions of the sensitivity studies.

In addition to the areas identified in PLG-0465, other areas were identified as potentially important to risk at Seabrook. In particular, the applicant was requested to provide information on the risk associated with accidents during shutdown. The results of the applicant's assessment of such accidents are also given in Table 6.2 and Figures 6.10 and 6.11. BNL was not able to assess the frequency of these events (refer to Section 2.2) because there remain fundamental questions regarding modeling of these scenarios. In addition, the potential for induced steam generator tube rupture (SGTR) for accidents in which the primary system is at high pressure was identified as a topic for review. This topic was reviewed in detail by the NRC staff and is the subject of continuing NRC and industry research activities. In an effort to assess the potential influence of SGTR, a simple sensitivity study was performed at BNL (refer to Section 2.3) and the results are also given in Table 6.2 and Figure 6.9.

In Table 6.2 and Figure 6.12 the effect on risk of combining all the sensitivity is presented. This calculation should not be interpreted as a reassessment of the overall risk profiles for Seabrook. It is simply intended to indicate how the results of the various sensitivity studies could influence the risk estimates in PLG-0465. The method used to combine the effect of all of the studies is not rigorous and could lead to inconsistencies. In addition, it is not normal practice in probabilistic risk assessments to combine bounding sensitivity studies. The results in Table 6.2 and Figure 6.12 should therefore be recognized for what they are, namely, a series of sensitivity studies and not be interpreted as a statement of the overall risk at Seabrook.

The results in Table 6.2 and Figures 6.9-6.12 are useful to focus on those areas of the BNL review that appear to have the greatest impact on the conclusions in PLG-0465. The results indicate, given the extent of this focused review, that the conservative assumptions regarding accidents during shutdown and induced SGTR have the most impact on the dose vs distance and risk estimates in PLG-0465. However, the more optimistic assumptions regarding these events have minor impact on the PLG-0465 results. These are areas of considerable uncertainty, which at the present time do not allow a better definition of the risk estimates than given by the ranges in Table 6.2 and Figures 6.9-6.12.

6.5 References

1. NUREG-0396.
2. RSS.
3. CRAC.
4. CRAC2.
5. CRACIT.
6. MACCS.
7. NUREG/CR-4214.
8. BEIR-1.
9. BEIR-3.
10. PLG-0465.

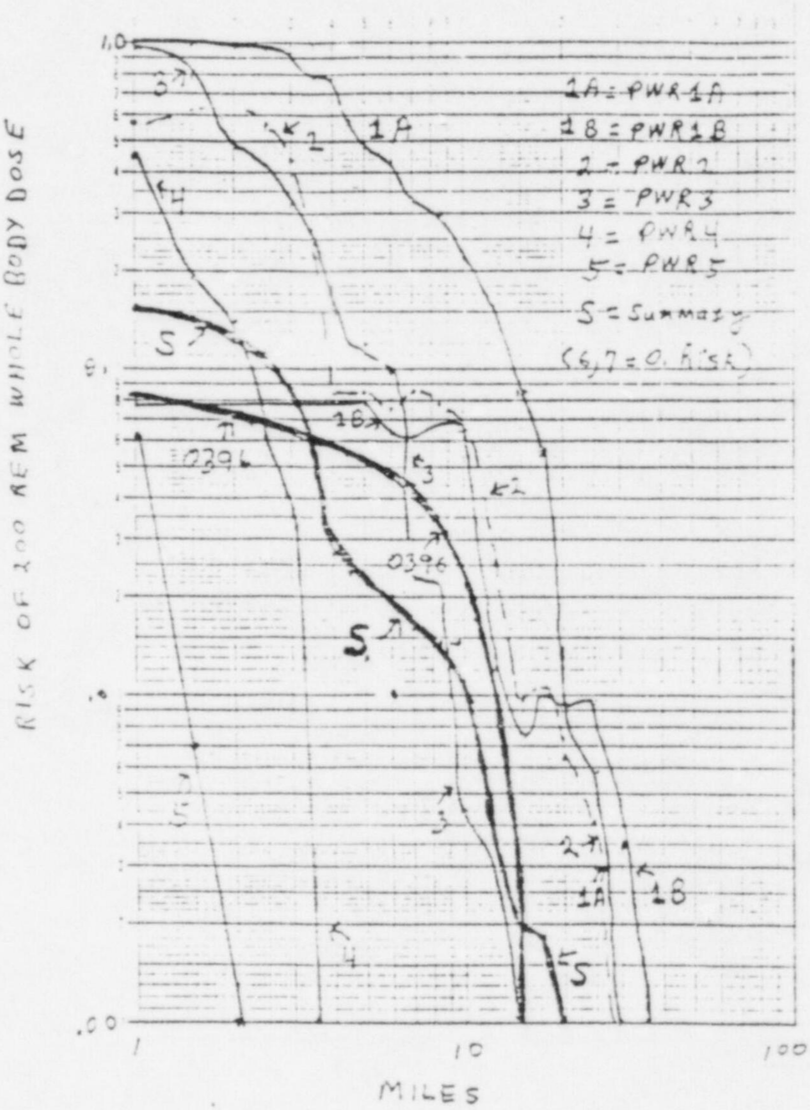


Figure 6.1 Components of NUREG-0396 curve as computed by BNL using CRAC2. The summary curve is normalized to $6x10^{-5}$ core melt probability. The result differs from NUREG-0396.

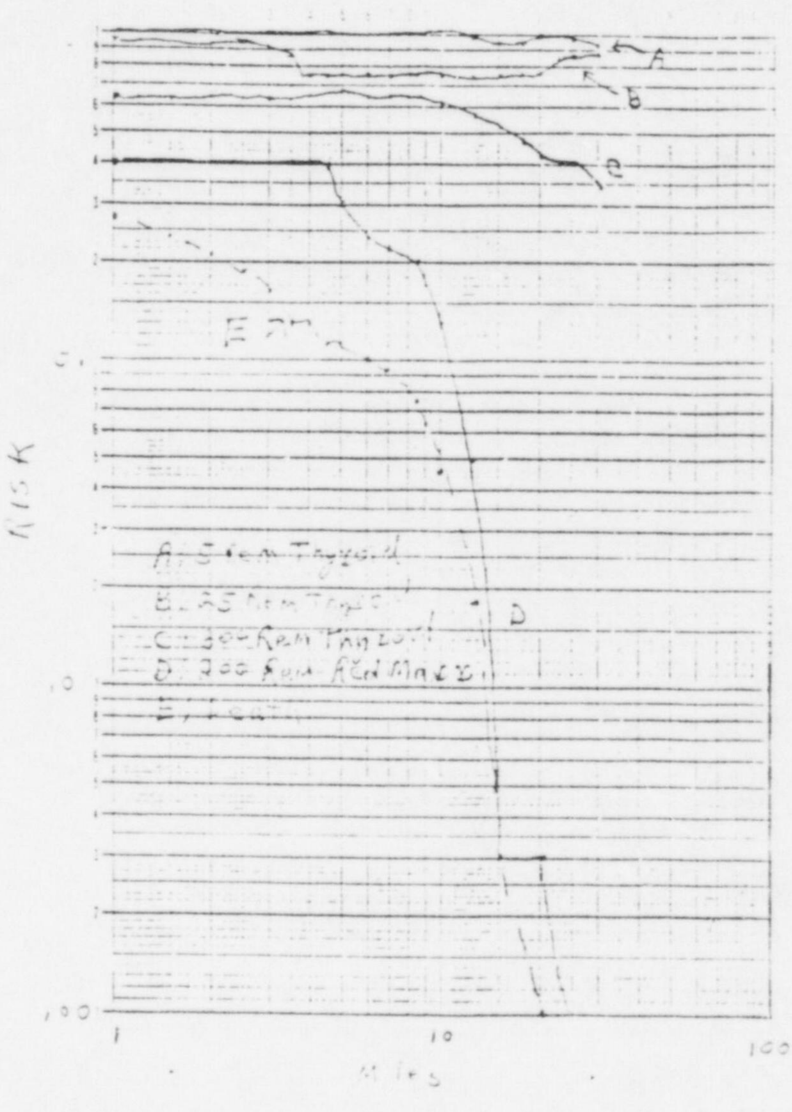


Figure 6.2 Risk of death or exceeding dose levels for SIW as calculated by BNL.

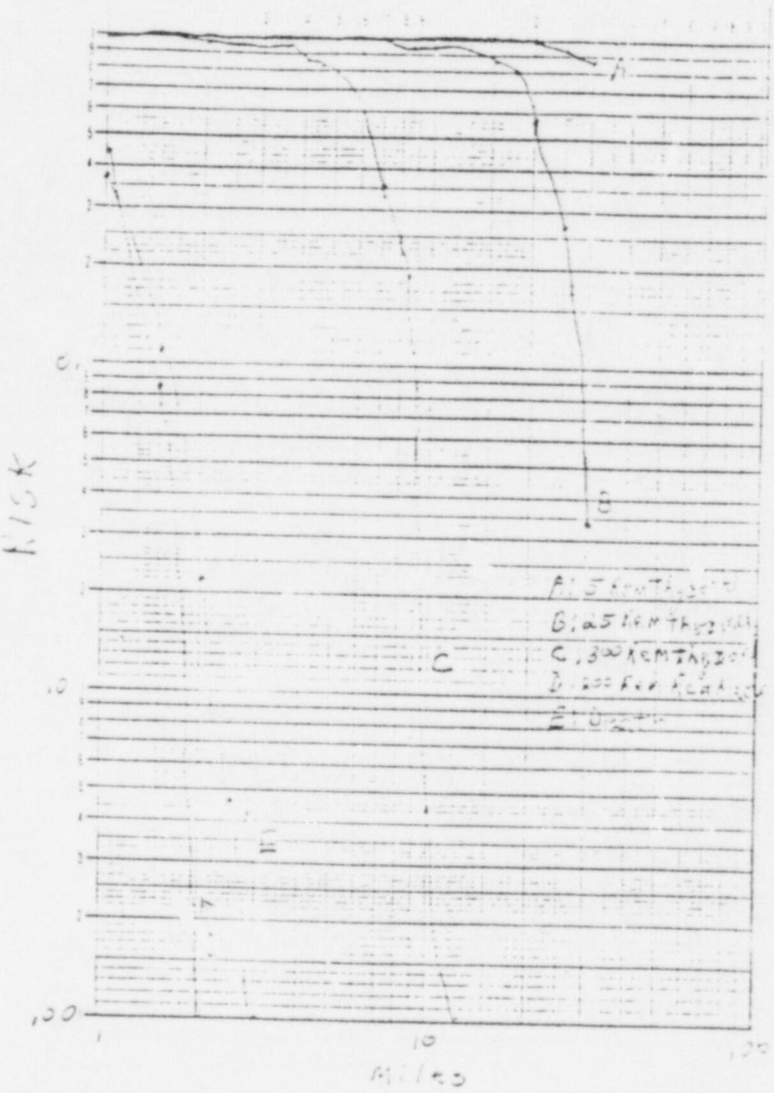


Figure 6.3 Risk of death or exceeding dose levels for S2W as calculated by BNL.

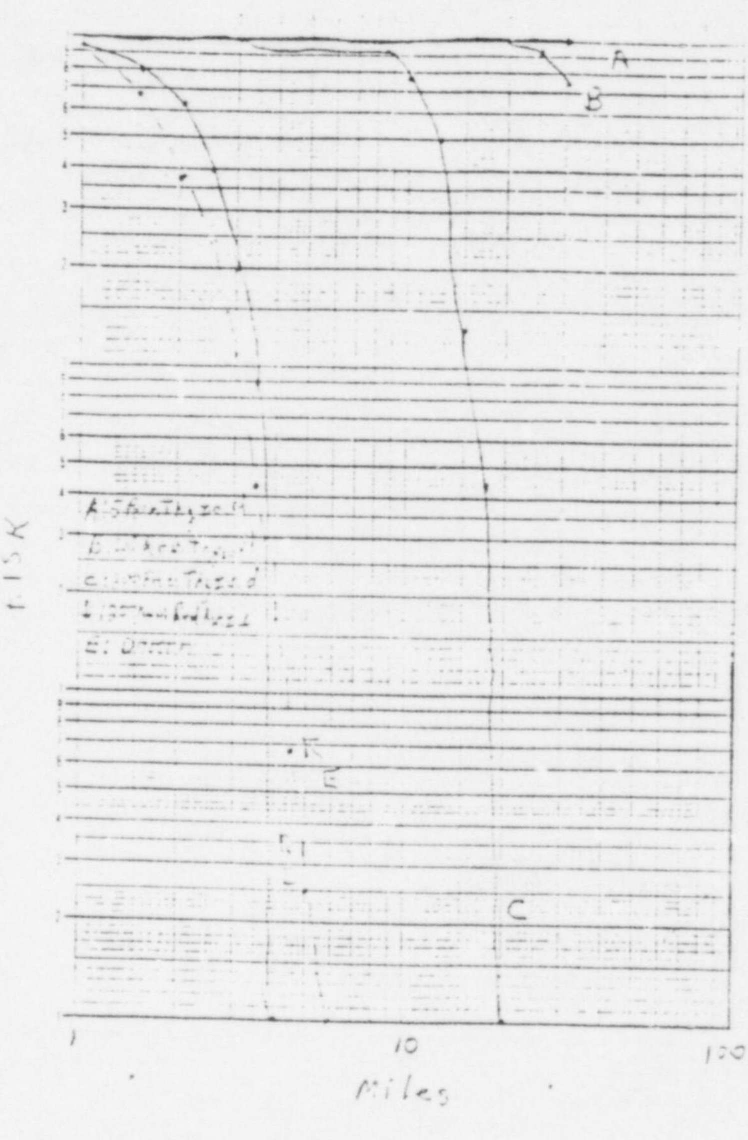


Figure 6.4 Risk of deaths exceeding dose levels for S6W as calculated by BNL.

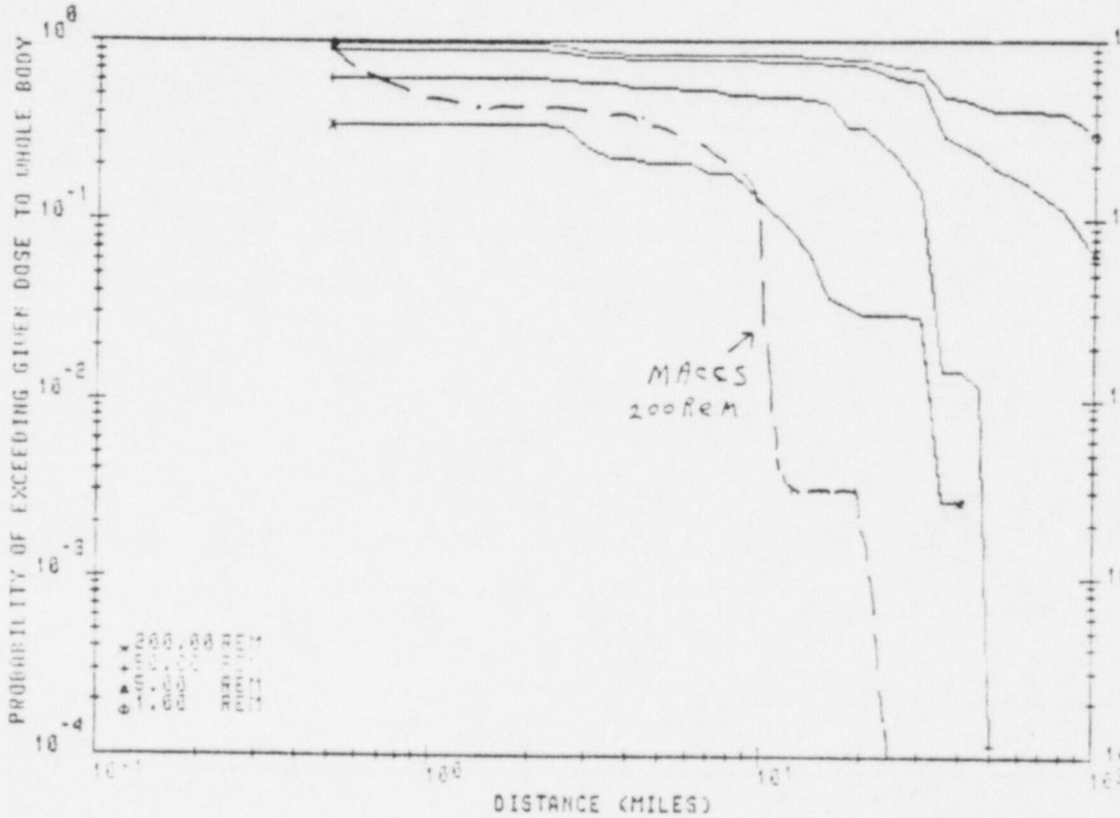


Figure 6.5 Dose versus distance curve for release category S1W from Seabrook for no immediate protective action with BNL results using MACCS superimposed.

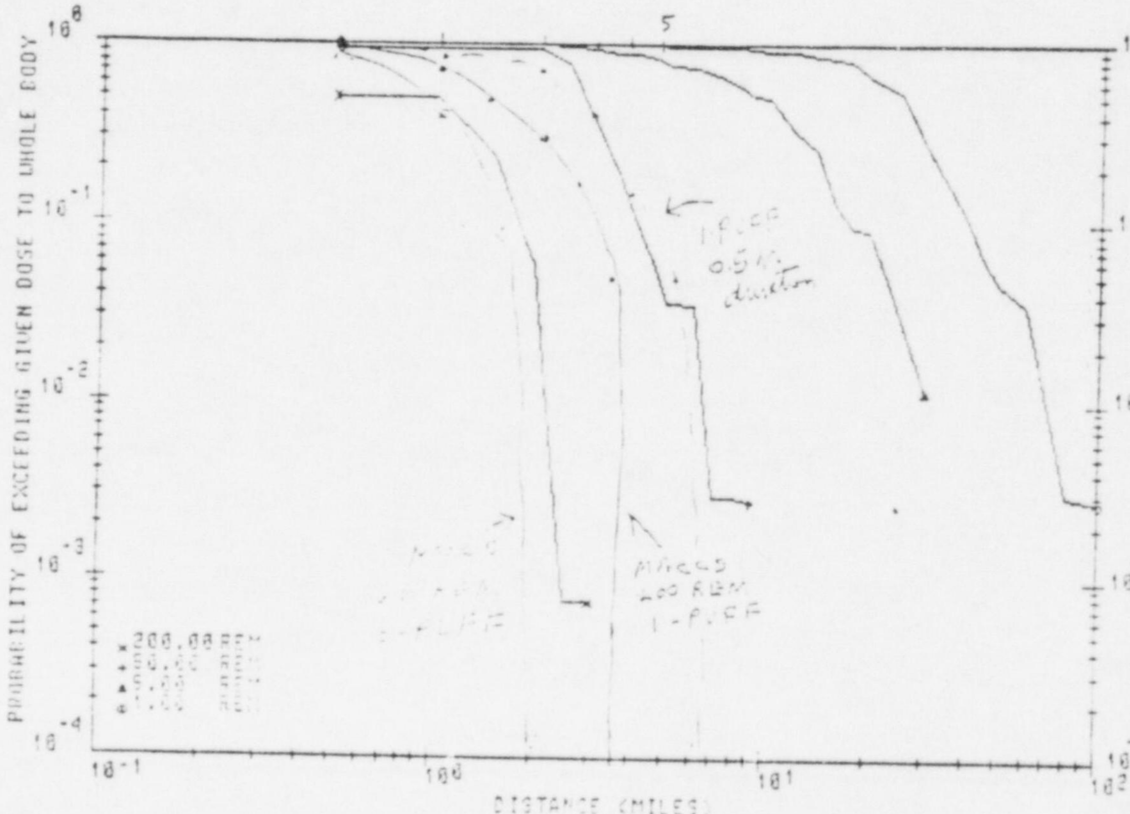


Figure 6.6 Dose versus distance curve for release category S2W from Seabrook for no immediate protective action with BNL results using MACCS superimposed.

Figure 6.7 Dose versus distance curve for release category S6_ from Seabrook for no immediate protective action with BNL results using MACCS superimposed.

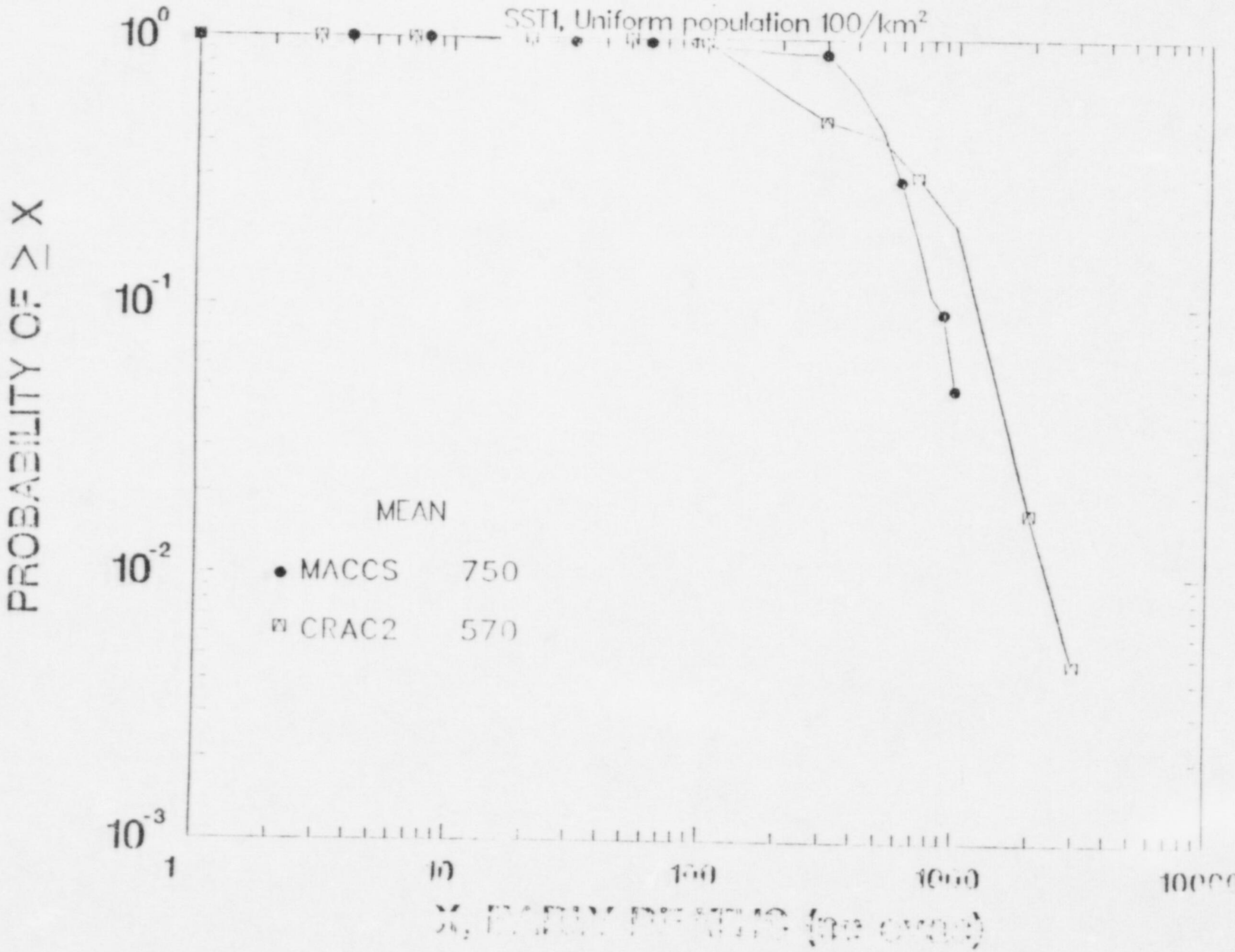


Figure 6.8 Comparison of MACCS to CRAC2 codes.
(from preliminary benchmark calculations performed by Sandia Labs.)

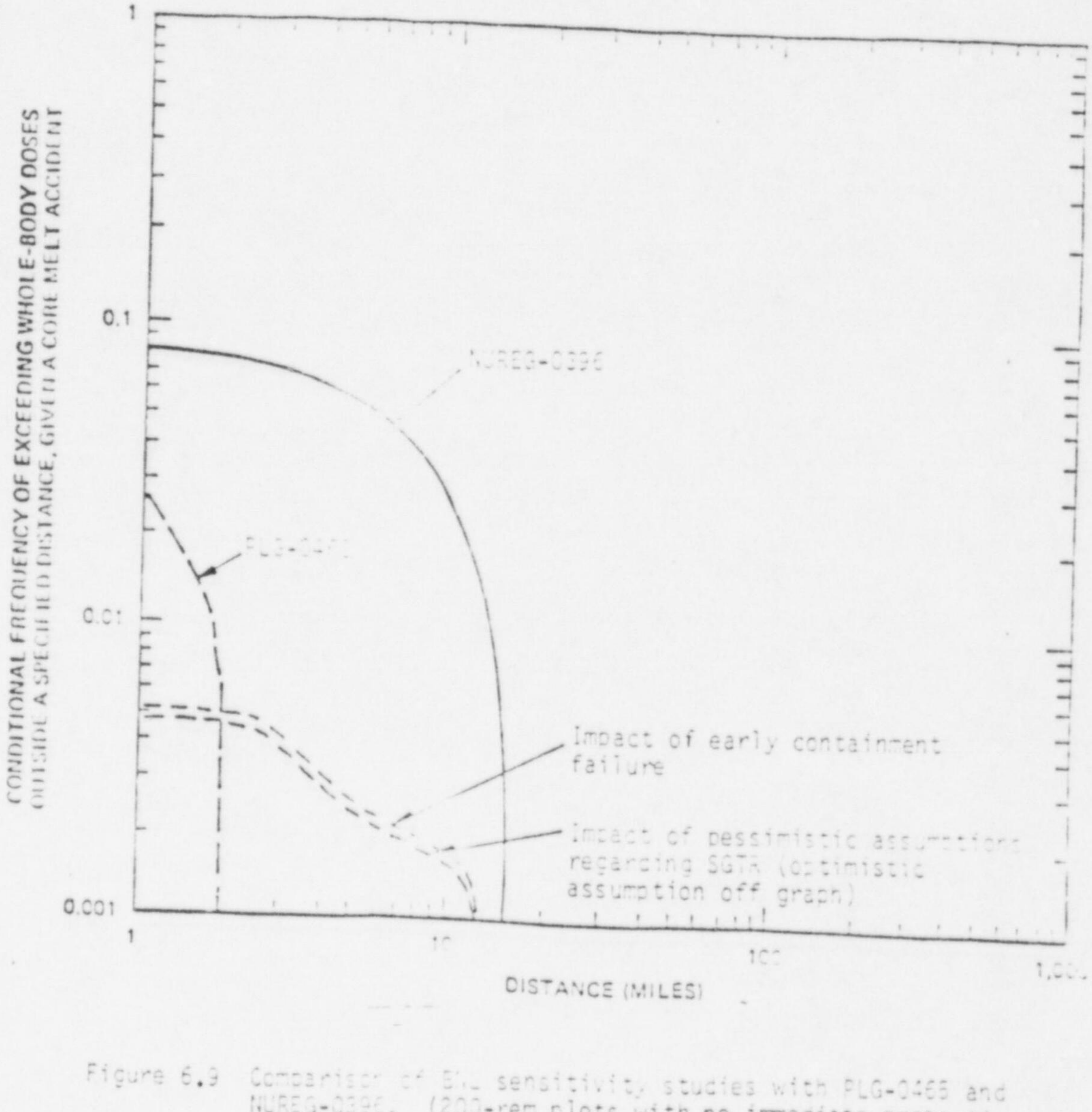


Figure 6.9 Comparison of BNL sensitivity studies with PLG-0465 and NUREG-0396. (200-rem plots with no immediate protective actions)

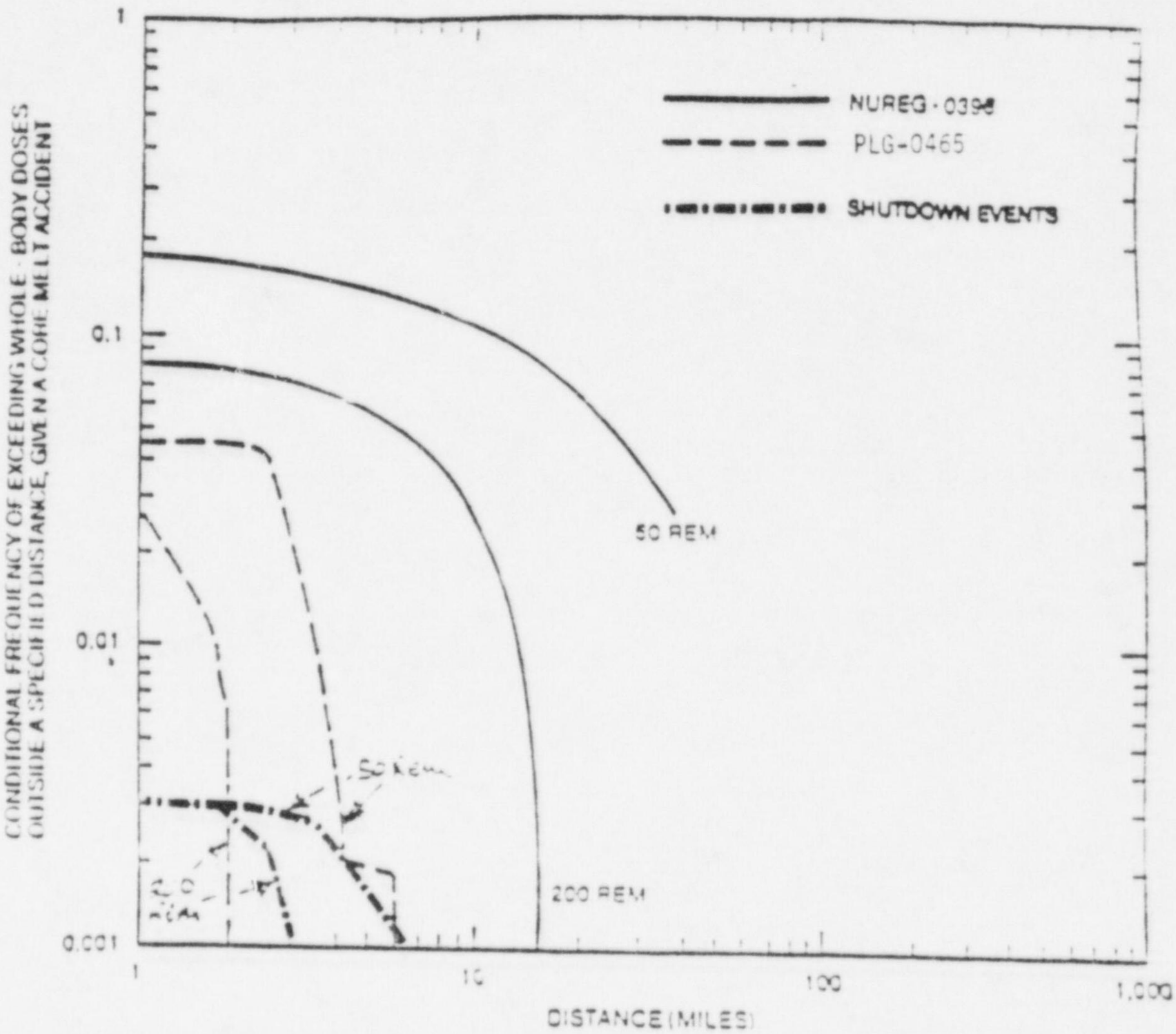


Figure 6.10 Comparison of 200 rem and 50 rem dose versus distance curves with contributions from shutdown events. (Calculations performed by applicant)

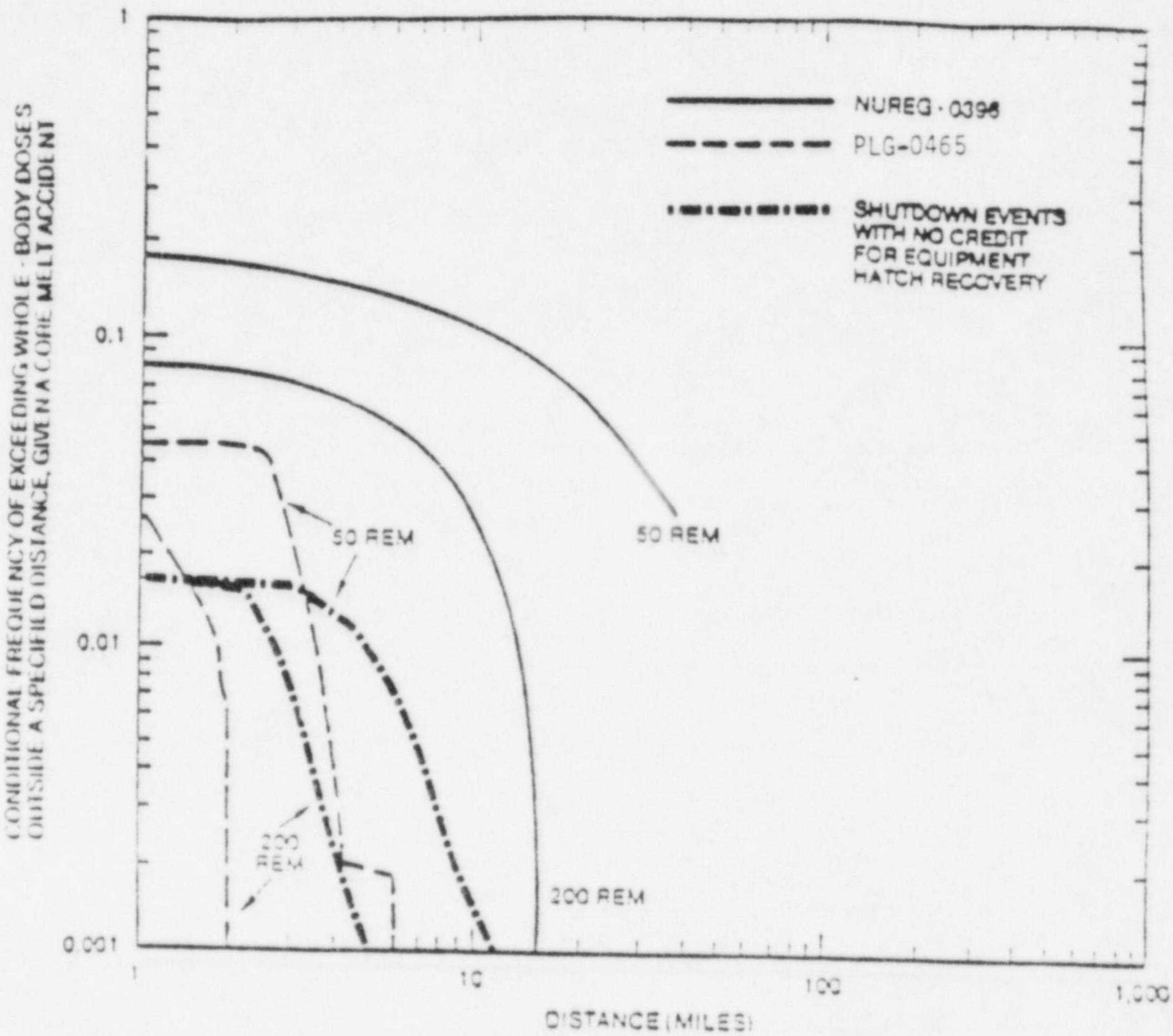


Figure 6.11 Comparison of 200 rem and 50 rem dose versus distance curves for conservative assumption of no credit for operator recovery of open equipment hatch. (Calculations performed by applicant)

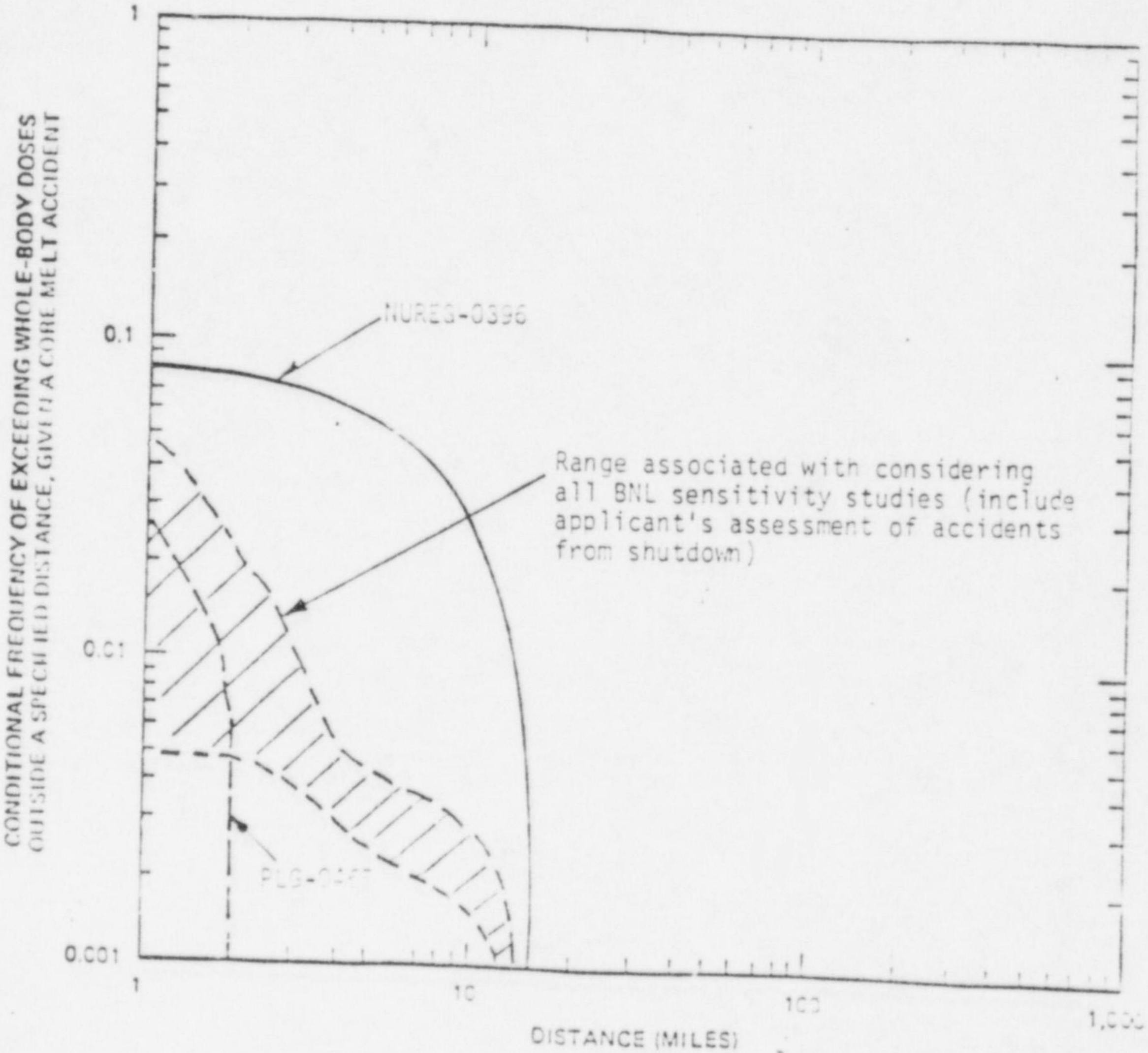


Figure 6.12 Comparison of all BNL sensitivity studies with PLG-0465 and NUREG-0396 (200-rem plots with no immediate protective actions).

Table 6.1 Summary of Release Categories Representing Hypothetical Accidents (From the RSS)

Release Category	Probability (per reactor-yr)	Time of Release (hr)	Duration of Release (hr)	Warning Time for Evacuation (hr)	Elevation of Release[g] (meters)	Energy Release (10^6 Btu/hr)	Fraction of Core Inventory Released[a]: Xe-Kr	Organic I[b]	I	Cs-Rb	Te-Sb	Ba-Sr	Ru[c]	La[d]
PWR 1	9×10^{-7}[e]	2.5	0.5	1.0	25	20 and 520[e]	0.9	6×10^{-3}	0.7	0.4	0.4	0.05	0.4	3×10^{-3}
PWR 2	8×10^{-6}	2.5	0.5	1.0	0	170	0.9	7×10^{-3}	0.7	0.5	0.3	0.06	0.02	4×10^{-3}
PWR 3	4×10^{-6}	5.0	1.5	2.0	0	6	0.8	6×10^{-3}	0.2	0.2	0.3	0.02	0.03	3×10^{-3}
PWR 4	5×10^{-7}	2.0	3.0	2.0	0	1	0.6	2×10^{-3}	0.09	0.04	0.03	5×10^{-3}	3×10^{-3}	4×10^{-4}
PWR 5	7×10^{-7}	2.0	4.0	1.0	0	0.3	0.3	2×10^{-3}	0.03	9×10^{-3}	5×10^{-3}	1×10^{-3}	6×10^{-4}	7×10^{-5}
PWR 6	6×10^{-6}	12.0	10.0	1.0	0	N/A	0.3	2×10^{-3}	8×10^{-4}	8×10^{-4}	1×10^{-3}	9×10^{-5}	7×10^{-5}	1×10^{-5}
PWR 7	4×10^{-5}	10.0	10.0	1.0	0	N/A	6×10^{-3}	2×10^{-5}	2×10^{-5}	1×10^{-5}	2×10^{-5}	1×10^{-6}	1×10^{-6}	2×10^{-7}
PWR 8	4×10^{-5}	0.5	0.5	N/A[f]	0	N/A	2×10^{-3}	5×10^{-6}	1×10^{-4}	5×10^{-4}	1×10^{-6}	1×10^{-8}	0	0
PWR 9	4×10^{-4}	0.5	0.5	N/A	0	N/A	3×10^{-6}	7×10^{-9}	1×10^{-7}	6×10^{-7}	1×10^{-9}	1×10^{-11}	0	0
BWR 1	1×10^{-6}	2.0	0.5	1.5	25	130	1.0	7×10^{-3}	0.40	0.40	0.70	0.05	0.5	5×10^{-3}
BWR 2	6×10^{-6}	30.0	3.0	2.0	0	30	1.0	7×10^{-3}	0.90	0.50	0.30	0.10	0.03	4×10^{-3}
BWR 3	2×10^{-5}	30.0	3.0	2.0	25	20	1.0	7×10^{-3}	0.10	0.10	0.30	0.01	0.02	4×10^{-3}
BWR 4	2×10^{-6}	5.0	2.0	2.0	25	N/A	0.6	7×10^{-4}	8×10^{-4}	5×10^{-3}	4×10^{-3}	6×10^{-4}	6×10^{-4}	1×10^{-4}
BWR 5	1×10^{-4}	3.5	5.0	N/A	150	N/A	5×10^{-4}	2×10^{-9}	6×10^{-11}	4×10^{-9}	8×10^{-12}	8×10^{-14}	0	0

(a) Background on the isotope groups and release mechanisms is presented in Appendix VII.
(b) Organic iodine is combined with elemental iodine in the calculations. Any error is negligible since its release fraction is relatively small for all large release categories.
(c) Includes Ru, Rh, Co, Mo, Tc.
(d) Includes Y, La, Zr, Nb, Ce, Pr, Nd, Np, Pu, Am, Cm.
(e) Accident sequences within PWR 1 category have two distinct energy releases that affect consequences. PWR 1 category is subdivided into PWR 1A with a probability of 4×10^{-7} per reactor year and 20×10^6 Btu/hr and PWR 1B with a probability of 5×10^{-7} per reactor year and 520×10^6 Btu/hr.
(f) Not applicable.
(g) A 10-meter elevation is used in place of zero representing the mid-point of a potential containment breach. Any impact on the results would be slight and conservative.

Table 6.2 Impact of BNL Sensitivity Studies on PLG-0465 Risk Estimates

Sensitivity Study		Absolute Risk Per Reactor Year: Early Fatalities: No Evac.	Early Fatalities: One Mile Evac.	Early Fatalities: Two Mile Evac.	Total Cancers: No. Evac.	Total Cancers: One Mile Evac.	Total Cancers: Two Mile Evac.
Original PLG-0465 Results		2.7(-3)*	3.6(-4)	2.4(-5)	1.5(-2)	1.4(-2)	9.2(-3)
Revised Frequency for Interfacing System LOCAs		2.8(-3)	4.6(-4)	1.2(-4)	1.5(-2)	1.4(-2)	9.3(-3)
Steam Generator Tube Ruptured	**	4.5(-3)	2.2(-3)	1.8(-3)	1.7(-2)	1.6(-2)	1.1(-2)
	***	2.8(-3)	4.2(-4)	8.4(-4)	1.5(-2)	1.4(-2)	9.3(-3)
Containment Loads and Performance		4.7(-3)	2.4(-3)	2.0(-3)	1.8(-2)	1.7(-2)	1.2(-2)
Accidents from Shutdown****	**	6.2(-3)	2.9(-3)	3.0(-4)	2.1(-2)	2.0(-2)	1.4(-2)
	***	2.7(-3)	3.6(-4)	2.8(-5)	1.5(-2)	1.4(-2)	9.2(-3)
Impact of all Issues	**	1.0(-2)	7.0(-3)	4.2(-3)	2.6(-2)	2.5(-2)	1.9(-2)
	***	4.9(-3)	2.6(-3)	2.2(-3)	1.8(-2)	1.7(-2)	1.2(-2)

*2.7(-3) = 2.7×10^{-3}

**Pessimistic assumptions.

***Optimistic assumptions

****Calculated by the Applicant, not confirmed by BNL.

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

DEC 08 1986

MEMORANDUM FOR: Vincent A. Noonan, Project Director
Project Directorate #5
Division of PWR Licensing-A

FROM: Charles E. Rossi, Assistant Director
Division of PWR Licensing-A

SUBJECT: STEAM GENERATOR TUBE RUPTURE DURING SEVERE ACCIDENTS AT SEABROOK STATION - DRAFT INTERIM REPORT

Plant Name: Seabrook Station, Unit 1
Docket Number: 5[illegible]
Resp. Directorate: PWR Directorate #5
Project Manager: Victor Nerses
Review Branch: Facilities Operations Branch, DPL-A
Review Status: Ongoing

An accident sequence with the potential to impact risk at pressurized water reactor plants under some conditions is the loss of steam generator tube integrity due to generation of high temperatures at high pressure during a core melt accident. The potential concern involves movement of high temperature fluid from the region of the melting reactor core into the steam generator tubes with a resultant overheating of the tubes which leads to steam generator tube rupture (SGTR). High pressure fluid containing radioactive material from the melting core would thereby be released to the secondary side of the steam generators, from where it could be released to the environment via the steam generator relief valves, thereby bypassing containment.

Public Service of New Hampshire (PSNH) has investigated the possibility of encountering conditions in the reactor coolant system under which SGTR can be of concern, and has determined the likelihood to be less than 4×10^{-5} per reactor year. This is sufficiently high, and the potential consequences of SGTR under severe accident conditions are sufficiently great under some circumstances, that PSNH initiated an investigation of this topic.

The enclosed report is a draft copy of an interim document prepared by Warren Lyon (FOB-A/DPL-A) which addresses the state of knowledge pertaining to Steam Generator Tube Rupture during postulated severe accidents, and the application of this knowledge to the Seabrook Station nuclear power plant. This report is prepared with the assumption that the work and assessment will continue. The report does not cover all material received from PSNH and its contractors, and PSNH and their contractors have offered to discuss the issue further. It also has not been subjected to comprehensive peer evaluation within the NRC.

Contact: W. Lyon
x28053

DRAFT

The major conclusions provided in the draft report are as follows:

1. Study of SGTR due to severe accident conditions is difficult due to the complexity of the phenomena and the developmental nature of analysis techniques.

2. Further work is necessary to conclude that SGTR is unlikely under conditions associated with a severe accident.

3. SGTR due to severe accident conditions can be shown not to be a problem if the reactor coolant system is depressurized.

Charles E. Rossi

Charles E. Rossi, Assistant Director
Division of PWR Licensing-A

Enclosure: As stated

cc: T. Novak
S. Long
F. Coffman
J. Han
V. Leung
V. Benaroya
S. Newberry

ENCLOSURE

SEABROOK STATION STEAM GENERATOR TUBE RESPONSE DURING SEVERE ACCIDENTS

DECEMBER 5, 1986

DRAFT

DRAFT

Foreword

This report addresses the state of knowledge pertaining to Steam Generator Tube Rupture during postulated severe accidents (approach to core melt and core melt), and the application of this knowledge to the Seabrook Station nuclear power plant. This is an interim report, prepared with the assumption that the work and assessment will continue. The report does not cover all material received from Public Service of New Hampshire (PSNH) and its contractors. It also represents the view of the author and has not been subjected to comprehensive peer evaluation within the NRC.

DRAFT

Nomenclature

AC	Alternating current
BNL	Brookhaven National Laboratory
EPRI	Electric Power Research Institute
ICC	Inadequate core cooling
KW	Kilowatt
LM	Larson Miller parameter
LOCA	Loss of coolant accident
MW	Megawatt
NRC	Nuclear Regulatory Commission
NSSS	Nuclear steam supply system
PDS	Plant damage state (See below)
PORV	Pressure operated relief valve
PRA	Probabilistic Risk Assessment
PSNH	Public Service of New Hampshire
RAI	Request for additional information
RCP	Reactor coolant pump
RCS	Reactor coolant system
RWST	Refueling water storage tank
SG	Steam generator
SGTR	Steam generator tube rupture

Plant damage states are used to classify conditions as follows:

1 Early core melt, low RCS pressure at time of reactor vessel failure, RWST injection not initiated

2 Early core melt, low RCS pressure at time of reactor vessel failure, RWST injection initiated

3 Early core melt, high RCS pressure at time of reactor vessel failure, RWST injection not initiated

4 Early core melt, high RCS pressure at time of reactor vessel failure, RWST injection initiated

5 Late core melt, low RCS pressure at time of reactor vessel failure, RWST injection not initiated

6 Late core melt, low RCS pressure at time of reactor vessel failure, RWST injection initiated

7 Late core melt, high RCS pressure at time of reactor vessel failure, RWST injection not initiated

8 Late core melt, high RCS pressure at time of reactor vessel failure, RWST injection initiated

9 Core melt with non-isolated SGTR

A Containment intact at start of core melt, containment heat and fission product removal available

B Containment intact at start of core melt, containment heat removal only available

C Containment intact at start of core melt, containment fission product removal only available

D Containment intact at start of core melt, none of the containment functions available

E Containment not intact at start of core melt, activity release filtered

F Containment not intact at start of core melt, containment opening larger than three inch diameter

FP Containment not intact at start of core melt, containment opening smaller than three inch diameter

FA Aircraft crash

1. OVERVIEW AND SUMMARY

The Public Service of New Hampshire (PSNH) has presented information to show that the Seabrook Station containment is one of the strongest of any nuclear power plant. It also contains one of the largest volumes. This combination leads to a conclusion that the containment has the capability to either significantly delay or prevent the release of large quantities of radioactive material during and following a severe (core damage or core melt) accident. Based on this premise, any significant risk associated with Seabrook Station would likely be found in accidents which bypass containment.

A number of potential bypass possibilities exist, some of which have traditionally been recognized in Probabilistic Safety Assessments (PRAs), and some of which have not. Historically, conservative assumptions have been applied to those cases which have been recognized, and the conservatism has been assumed to be sufficiently large that the unrecognized possibilities became insignificant since they were believed small in comparison.

PSNH and its contractors have provided a comprehensive PRA with additional follow up investigations in which a better representation of nuclear plant behavior has been attempted. Some conservative, and thereby misleading, representations have been removed. This approach to accident analysis leads to the possibility that when conservatisms have been removed, previously neglected bypass paths which were masked may now be found to contribute to risk. Recognizing this, the Staff and PSNH have explored containment bypass possibilities. One possibility, the topic of this report, and a potential issue that has been under investigation by industry and the Staff for several years, is the loss of steam generator tube integrity due to generation of high temperatures at high pressure during a core melt accident. The potential concern involves movement of high temperature fluid from the region of the melting reactor core into the steam generator tubes, with a resultant overheating of the tubes which leads to their rupture. High pressure fluid containing radioactive material from the melting core would thereby be released to the secondary side of the steam generators, from where it could be released to the environment via the steam generator relief valves, thereby bypassing containment.

For steam generator tube rupture (SGTR) to be a concern as addressed here, one must have a core damage (or melt) condition in progress with no water on the steam generator secondary side. The principal contributor to this condition is estimated to be a loss of all AC power concurrent with a loss of all turbine driven feedwater to the steam generators. PSNH has investigated the possibility of encountering conditions which can contribute to SGTR and has determined the likelihood to be less than 4×10^{-5} per reactor year. This is sufficiently high, and the potential consequences of SGTR under severe accident conditions are sufficiently great, that further investigation has been necessary. This investigation is ongoing. This report provides an interim assessment of the status of the investigation, as well as a projection of expected results.

Study of SGTR due to severe accident conditions is difficult. The phenomena are complex, and most analysis techniques used to investigate nuclear power plant behavior have utilized assumptions which are not applicable here. The principal complication is the multidimensional character of fluid behavior in the reactor coolant system. Suitable computer programs are just beginning to become available. Suitable experimental information is just being developed. Hence, pioneering work, such as provided by PSNH in investigation of this issue, can be expected to have weaknesses as well as strengths. We have found this expectation to be true.

The work reported by PSNH and its contractors is highly informative and addresses most aspects of the SGTR issue. It is based upon knowledge of what takes place within the Nuclear Steam Supply System (NSSS), upon a major computer program that is under development and is being verified (MAAP), and upon information derived from an experimental program at Westinghouse. The following is a summary of the reported information and our assessment:

1. Mathematical modeling. Expected phenomena, experimental information pertinent to the phenomena, and modeling assumptions have been addressed for each of the major components of the NSSS which are affected. Multidimensional fluid flow and energy transport have been established as dominant over most of the conditions of interest. We consider this area

to be in a preliminary stage of development, and there are some potential difficulties, which include:

a. Certain modeling assumptions are overly optimistic. An example is the assumption of complete mixing in the steam generator inlet plenum which tends to reduce the temperature of fluid entering the steam generator tubes. This assumption is not supported by the available experimental evidence, and the effects of the assumption are not balanced by identifiable pessimistic assumptions elsewhere in the analysis.

b. Experimental evidence is preliminary. The experimental facility at Westinghouse is providing information pertinent to this issue. However, testing has been limited to conditions which are only roughly scaled to NSSS representation. This is due to a logical progression in the test planning and facility development. Data from apparently well scaled test conditions are just becoming available. No other test facility addresses certain aspects of this issue.

c. The computer program used as the basis for much of the work has not been verified, nor is documentation available. We understand a verification program and an effort to provide documentation are underway. (PSNH contractors have offered to discuss this information with us. Our review has not progressed to the stage where we can make use this offer.) Although the phenomena we understand to be modeled by the code appear adequate for the purposes needed here, and the code results appear reasonable subject to our concerns as expressed elsewhere in this report, this is not sufficient information to accept the analysis results.

2. <u>Seabrook Station Representation.</u> The basic analyses and sensitivity studies have been based upon a plant configuration in which the NSSS state is assumed. Most of the assumed state conditions are reasonable. There are exceptions. For example, the steam generator secondary side is

assumed to be at a pressure corresponding to secondary side relief valve settings, and creep rupture of tubes is reported for this state. The resulting conclusions are similarly based upon this state. We believe there is sufficient likelihood the secondary side will be depressurized that this case should be considered. Depressurization would roughly double tube stress since the secondary side pressure would be decreased from roughly 1100 psi to atmospheric pressure while the RCS pressure remained at approximately 2300 psi.

3. Sensitivity Studies. PSNH and its contractors have performed a wide ranging sensitivity study as part of an assessment of the impact of various modeling assumptions and the state of the plant. Although this yields valuable information and insight, sensitivity studies should be approached with caution. They are only as good as the basic modeling. The impact of our difficulty with assumptions such as the behavior of the steam generator inlet plenum is not addressed in the sensitivity study, and could impact the results and conclusions.

4. Operator Actions. Plant response can be drastically altered by operator actions during a severe accident. SGTR is no exception. A number of operator responses have been discussed with PSNH. Although many of these were postulated actions, significant information has been developed from these postulations. Recognition that operator actions could depressurize the steam generator secondary side is one item raised during the review. Depressurization of the reactor coolant system via the pressurizer Pressure Operated Relief Valve (PORV) to avoid the SGTR problem is another.

We find that the topic of SGTR is in a developing state, with knowledge being rapidly accumulated. Further work is necessary to conclude that SGTR is unlikely under all conditions associated with a severe accident.

Existing knowledge can be used to support a conclusion that SGTR is not a problem if the RCS is depressurized. Consequently, reasonable assurance that progressions toward core melt would not occur at high RCS pressure, coupled with supporting evidence in regard to steam generator tube response, would alleviate our concern regarding SGTR under severe accident conditions. We have not conducted an evaluation of the trade-offs associated with such an approach, nor have we been provided with information that would either support or negate RCS depressurization under severe accident conditions. We have not provided a recommendation regarding whether RCS depressurization is attractive when all pertinent factors are considered.

Our judgement is that a carefully conducted thorough evaluation on the part of PSNH can establish that the likelihood that a SGTR will result due to overheating during severe accidents which initiate from power operation is sufficiently small that the risk associated with this event can be shown to be negligible. Our judgement is preliminary and has not been substantiated.

Substantiation of a judgement regarding SGTR under severe accident conditions originating from power operation with the RCS at high pressure can be obtained through a combination of analytic and experimental investigations. The ongoing test at Westinghouse in which reasonably close similitude is claimed between the test facility and appropriate parts of a Westinghouse four loop NSSS will provide key data which can be applied to assist in the development and confirmation of analysis techniques. Use of selected test data from other facilities and further examination of the analysis techniques, coupled with necessary changes when they are uncovered, should provide sufficient confirmation that reasonable reliance can be placed upon accident analyses pertinent to this issue. Suitable analyses can then provide a sufficient foundation to resolve this issue.

2. INTRODUCTION

The Public Service of New Hampshire (PSNH) reporting of Seabrook response to accident conditions in References 1 - 4 represents one of the most comprehensive investigations of nuclear power plant accidents in a specific plant that we have encountered. Some accidents which have a significant impact upon risk are treated more comprehensively than previously reported by any investigator. For example, References 3 and 4 describe an investigation of LOCA outside of containment that is more comprehensive than any we have reviewed. Many of the commonly used conservatisms, which distort the perception of accident impact, have been removed. What results is a serious attempt to better represent plant response to severe accident conditions, with particular attention to items which have previously been identified as having a serious impact upon risk. Paradoxically, as will be seen, this attempt to better represent plant behavior requires a more careful review of certain aspects of severe accidents than required for previously reported PRA investigations.

PSNH has presented information to show that Seabrook Station has one of the strongest containments of any nuclear power plant. It is also one of the largest with respect to containment volume. The combination of large volume and strength leads PSNH to a conclusion that the containment can mitigate virtually every severe accident and, at the worst, can significantly delay release of meaningful quantities of radioactive material during and following core melt accidents. Most core melt accidents can be contained within the Seabrook Station containment, and, if this is accomplished, little radioactive material will escape. The full mitigative capability of the Seabrook containment will be realized if there are no "holes" in the containment. Such holes can exist if any of the following occur:

1. Containment is not properly closed (isolated), such as can occur if containment ventilation is not properly closed upon receipt of a containment isolation signal,

2. A failure occurs which allows the containment atmosphere to escape, such as failure of a containment penetration due to a combination of high pressure and high temperature, or

3. A failure occurs which allows material to move directly from the Nuclear Steam Supply System (NSSS), principally the Reactor Coolant System (RCS), to the environment, such as occurs with the traditional "Event V" (Ref. 5), with LOCA outside containment leading to core melt and the release of radioactive material via the LOCA flow pathway.

Clearly, if PSNH conclusions regarding containment strength are verified, there will be little risk associated with accidents at Seabrook Station unless containment is bypassed. Therefore, core damage accidents with containment bypass deserve careful attention. PSNH has reported studying some bypass accidents in significant detail (Refs. 3, 4, 12, 17, and 18). Such studies have led them to conclude that certain bypass accidents at Seabrook, such as LOCA outside containment, engender significantly less risk than previously believed.

In PRA investigations, one may neglect some small contributors to risk since they are negligible in comparison to major contributors. If major contributors are found to be significantly smaller, then one must check the previously neglected contributors to assure they are still negligible or, conversely, they must be included in the contribution to risk if they are now significant. This is the situation that is typified by the Seabrook Station PRA investigation (Refs. 1-4). A major contributor to risk associated with containment bypass, Event "V", has been analyzed by PSNH, and they have concluded that it is not the significant contributor to risk that it was previously believed to be.

This situation has led us to ask "Are there any containment bypass risk contributors which have been missed or which require further consideration?" One potential area for bypass, as identified above, involves a path between the RCS and the environment. One way of searching for such paths is to ask "Are there any phenomena which may occur, and which have not been adequately addressed in past searches for accident possibilities?" We and PSNH, among others, have asked that question, and found that certain phenomena have been neglected in past PRA investigations because their contribution to risk is small in comparison to other contributors. The phenomena of potential significance involve multidimensional fluid behavior and fission product transport within the RCS during the approach to core melt and during the core melt process. Consideration of these phenomena has a significant impact upon RCS response, including potentially the location of RCS failure. There are many possible implications, including the possibility that the impact of RCS failure on containment may have been overestimated in past analyses. The implication of interest here is that failure to accurately model RCS fluid and fission product heating behavior might result in an RCS failure which bypasses containment. The only area discovered where this is of immediate concern involves the Steam Generator (SG) tubes. If these fail during a core melt accident while the RCS is at high pressure, there is a high potential of a major release via the SG relief valves or the SG Pressure Operated Relief Valves (PORVs), which vent directly to the environment.

The general concern addressed in this report is the rupture of SG multiple tubes in response to high temperature, which in turn is a result of core uncovery. This accident sequence should be of concern any time there is a core melt with the RCS at high pressure in combination with no water in the SG secondary sides. These conditions lead to a potential for natural circulation transport phenomena to significantly heat the tubes prior to breach of the reactor vessel. If this occurs, the resulting loss of tube strength could lead to tube rupture. If tube rupture occurs, and any of the secondary side valves are open, the secondary side is breached outside containment. Alternatively, if the RCS pressure is above the SG relief valve setpoints, containment is similarly bypassed. This has not been adequately investigated, and is not recognized as a release path in the early Pickard, Lowe and Garrick work on risk investigation at Seabrook Station (Refs. 1-4). It has been addressed in more recent work (Refs. 12, 17, and 18).

The concern was expressed as the rupture of multiple steam generator tubes. We do not believe single tube ruptures will occur under the severe accident conditions of interest. The reason for this is that if one tube ruptures, or even begins to leak significantly, this will induce flow of hot RCS fluid toward the leak. Therefore, the location of tube rupture will probably quickly become hotter. If high temperature is what led to the break, a higher temperature can only make it worse. Tubes in the vicinity of the break will be exposed to the high velocity break flow, in additional to high temperature, weakening them and, we believe, quickly leading to their failure. We believe this cascading effect would rapidly propagate to multiple tube rupture, stopping only when sufficient RCS depressurization has occurred that tubes are no longer stressed by a significant pressure differential across their walls.

Although this report is limited to SG tube rupture, there are other SG components which separate RCS fluid from the SG secondary side. These components, such as the SG tube sheet, must be investigated to achieve completeness in the investigation of containment bypass via the steam generator.

An initial consideration in investigation of the SG tube rupture issue is "What is the likelihood of attaining conditions where SG tube response could be of concern?" Principally, the conditions are loss of all SG feedwater with a simultaneous loss of RCS makeup capability; conditions which result, for example, from a loss of all AC electrical power with the simultaneous loss of the turbine driven auxiliary feedwater pump. PSNH estimated this condition to have a mean annual frequency of 4.5×10^{-5} per reactor year (Ref. 17), a value sufficiently high that tube response must be considered.

3. STEAM GENERATOR TUBE RUPTURE (SGTR) UNDER SEVERE ACCIDENT CONDITIONS

3.1. Description of Phenomena and Potential Concern.

The RCS is generally modeled with a one dimensional representation of fluid flow, and in some cases with parallel one dimensional modeling in regions such as the reactor vessel. This has been particularly true for PRAs, where to our knowledge, all have been based upon computer code analyses which incorporated single dimensional representations of fluid behavior within the RCS. Additionally, movement of the source of heat due to fission product migration is seldom modeled.

The possibility of RCS behavior being different from what is generally represented during severe accidents has been recognized for some time. Winters (Ref. 6) identified aspects of the problem in 1982 and Denny and Sehgal (Ref. 7) provided preliminary multidimensional analysis results in 1983. It was the subject of an NRC/Industry meeting (Ref. 8), and a formal request for work within NRC (Ref. 9), in 1984. Potential impact upon SGTR was estimated on a preliminary basis (Ref. 10), and experimental data were presented from an ongoing series of tests (Ref. 11), in 1985. Numerous analysis results have been published since the early publications of Denny and Sehgal which represent work sponsored by both industry and the NRC. However, there is no published analysis of overall NSSS response to a broad range of severe accident conditions which includes these phenomena, and which is based upon accident analysis methods which have been subjected to broad peer review and acceptance. This introduces a difficulty into review of SGTR during severe accidents with respect to the impact upon the Seabrook Station risk evaluation. As will be seen, sufficient work has been accomplished that what appear to be reasonable conclusions can be formulated, although confirmation will require additional effort. As will further be seen, there appear to be operational methods which can negate the problem, although the impact on other aspects of plant operation has not been evaluated.

The one dimensional analysis approach appears adequate for approximation of the time to core uncovery following accident initiation. Whether it is adequate to represent behavior following core uncovery depends on a number of factors, such as the conservatism associated with RCS modeling, system response to accident conditions, and the type of accident. In general, one should question the adequacy if two, and sometimes one, of the following conditions exists:

1. RCS pressure is in the vicinity of normal operating pressure

2. A liquid "plug" exists at RCS low points (the lower reactor vessel or the crossover legs between the SG exit and the Reactor Coolant Pump (RCP) suction connection), and the remainder of the RCS is filled with vapor or gas

3. Accident contributors generally found to be major contributors to risk have a lower probability of occurrence than found in most PRA investigations

In the case of Seabrook Station, all three conditions apply with respect to SG response. A number of conditions potentially lead to core melt with the RCS at high pressure (including, for example, a loss of all AC electrical power with loss of feedwater), such conditions are calculated to leave liquid plugs in RCS low points for Westinghouse designed NSSS's, and the Seabrook Station PRA work represents comprehensive modeling with removal of some unrealistic conservatisms.

The potential misrepresentation of system response of concern here stems from the fluid flow behavior inherent in one dimensional modeling. Such modeling typically represents flow through the reactor core as determined by the water boiloff rate from the lower core or lower plenum. This rate becomes small as the water level approaches the bottom of the core. Typical calculations (see historical references which were previously discussed) indicate that the flow rate due to natural convection which occurs in a multidimensional manner is of the order of ten or more times that of the flow due to boiloff. Hence, the calculations are typically based on a minor contributor to flow, and the major contributor is neglected.

The modeling difficulty also applies to upper plenum behavior. One dimensional modeling of any fluid (liquid, vapor, or gas) that passes through the core is typically assumed to flow through the upper plenum and out the hot leg.

This modeling is incorrect under severe accident conditions where a major portion of the core has been uncovered or the core is being vapor or gas cooled since strong recirculation patterns will develop which thermally link the core and upper plenum. At pressures in the range of 2250 psi, the linkage is strong, and some of the upper plenum component temperatures can be expected to closely follow core temperature during the early stages of the approach to core melt. The strength of the linkage diminishes with decreasing pressure. Information also exists which illustrates a decrease in linkage with increasing hydrogen concentration and core damage (although initial production of hydrogen may enhance circulation due to the buoyant gas "pushing" its way toward upper regions of the reactor vessel).

Similarly, correct consideration of the hot leg and steam generator behavior leads to calculation of significantly different behavior when contrasted to one dimensional modeling. Hot fluid, at a temperature far greater than predicted via a one dimensional model, will enter the upper portion of the hot legs from the reactor vessel, and flow toward the inlet plenum of the steam generators. Displaced colder fluid will return to the reactor vessel upper plenum along the bottom of the hot legs. Circulatory patterns will become established in the steam generator inlet plena in which some of the hot incoming fluid is mixed with plenum fluid. Fluid from the steam generator inlet plena will flow into some of the steam generator tubes in the nominal forward direction, displacing fluid in the steam generator outlet plena. This displaced fluid will flow through other tubes in a nominal reverse direction, reentering the steam generator inlet plena. (All of these flows have been observed experimentally as described in References 11, 13, and 14). This mechanism has the potential to transport hot fluid from the reactor vessel into the steam generator tubes during core heatup and melt, with the result of creating the potential of overheating the tubes if there is no water on the steam generator secondary side.

There are other possibilities which could challenge tube integrity as well. For example, many plant Inadequate Core Cooling (ICC) emergency procedures specify RCP operation if conditions exist which indicate an approach to core melt, and alternate mitigative measures have failed. Such a step could circulate hot fluid through the RCS, including the tubes. Although this may slightly extend the time to core melt, it may be an unattractive approach if it also introduces a high likelihood of loss of tube integrity. To our knowledge, these contrasting responses and the impact upon risk have not been studied. (Note the likelihood of encountering the situation is small.)

A final phenomenon that has received inadequate attention during conditions leading to core melt is fission product movement. Typical one dimension accident code calculations take such movement into account from the viewpoint of radiological hazard, but do not include the influence upon heat generation. Approximately a quarter of the heat producing radioisotopes probably has left the core under the conditions of interest, and substantial deposits can be expected in the upper plenum structure. This could have a significant influence upon thermal response, particularly if some of this material leaves the reactor vessel and enters the hot legs.

As will be seen in the following sections, PSNH has addressed many of these issues in the most comprehensive study of this problem that we have encountered.

3.2. Seabrook Station Steam Generator Integrity

3.2.1 Issues Addressed By PSNH

The PSNH has addressed many of the issues applicable to SG tube response to severe accident conditions (Refs. 12, 17, and 18). Analysis results were summarized which [illegible] intended to determine the thermal response of SG tubes under severe accident conditions. Basic analysis assumptions pertinent to the state of the plant were:

1. The steam generators must be dry to experience a significant thermal transient since, if the SG secondary side contains water, the tubes cannot overheat.

2. Station blackout conditions (Loss of all AC power) exist.

Analyses were conducted for the following:

1. Station blackout without operator actions or RCP seal LOCA

2. Station blackout with a 50 gpm RCP LOCA (each RCP) and no operator actions

3. Station blackout with operator actions

4. Uncertainty evaluation

Possible operator actions considered included:

1. Start steam turbine driven auxiliary feed water flow

2. Restore emergency AC power (diesels and/or switchgear)

3. Shed nonessential loads

4. Open RCS PORVs when core exit temperatures exceed 1200°F.

A number of other operator actions one might expect were discussed during a meeting with the PSNH at BNL on October 17, 1986, including:

5. SG blowdown and depressurization to enable filling the SGs by the condensate booster pumps or from fire water systems. (There are two diesel driven pumps and one electrically driven pump at Seabrook Station. The ability to use these for injection into the SGs has not been confirmed.)

6. RCP operation, a step that is not possible unless off site electrical power has been restored. (PSNH felt the likelihood was sufficiently low that there would be negligible effect on risk.)

3.2.7 Likelihood of Conditions Leading to Tube Failure

PSNH addressed the question of conditions necessary for SGTR in the response to the Staff Request for Additional Information (RAI) 47 (Ref.

[illegible]. In this response, PSNH stated the risk to be small for the following reasons:

1. The frequency of high pressure core melt with dry steam generators is very small.

2. Given the postulated occurrence of a high pressure core melt with dry steam generators, creep rupture of the SG tubes is not a credible failure mode.

3. A large number of tubes must fail to produce an early large containment bypass.

4. All three of the following must occur in order for there to be a containment bypass:

   a. Failure to recover water to the SG
   b. Failure to depressurize the RCS
   c. SG tube creep failure

3.2.3 PORV Considerations

PORV operation as identified in item 4, above, is not specifically contained in Seabrook Station emergency procedures, but is believed by PSNH to be a logical operator response as an attempt to depressurize and obtain water from the accumulators. (Operator monitoring of the temperatures is specifically identified in the procedures for loss of all AC power conditions.) In addition to potential core cooling via the accumulator water, opening the PORVs is claimed to have the following effects:

1. It reduces stresses in all primary system components

2. PORV flow overrides natural circulation such that high fluid temperatures are not attained in the SGs, including the tubes.

In response to a staff question, PSNH indicated that the likelihood of being able to open the PORVs under loss of AC and ICC conditions was high (See Section 2.2.9). They also indicated that one PORV was sufficient since its "worth" is about 50 MW of energy removal in the form of steam, and have presented blowdown rate information in Reference 18. (Note Seabrook is equipped with two PORVs.)

Although we consider the EPRI funded Westinghouse tests pertinent to this issue to be somewhat preliminary with respect to scaling to NSSS conditions, some interesting effects have been observed that are worth noting which pertain to PORV operation. These include:

1. Natural circulation flow restores itself readily to the pre-opening condition in the hot legs, core, and communication paths between the upper plenum and the upper head following PORV closure.

2. Heat transfer in steam generators between the primary and secondary side fluids increases 50% to 75% with periodic venting.

3. The core is little affected except for the boundary with the hot leg that connects to the pressurizer surge line.

Item 2 is of particular interest since it carries an implication that flow in the steam generator tubes is enhanced by PORV operation (as well as by opening and closing of RCS safety valves). Hence, if one visualizes opening and closing a pressurizer PORV when degraded conditions are well established with the steam generator secondary side depressurized, there may be a tendency to enhance flow of hot RCS fluid through the tubes, with the potential of causing tube rupture.

3.2.4 Loop Seals

Loss of RCS inventory under natural circulation conditions (RCPs not running) is expected to leave the RCS in a condition where water is trapped at low elevations. According to a number of preliminary analyses, these exist at the cross over leg between the SG exit and the RCP inlet, and in the lower region of the reactor pressure vessel. The absence of these water seals could significantly change circulatory conditions during ICC conditions, with the potential for changing SG tube response. Although we expect a careful examination of behavior in the Seabrook RCS would establish that the seals will remain under most boil down conditions, this expectation needs to be substantiated by suitable analyses which address the range of conditions which can exist during severe accidents.

Complete loss of the RCS liquid inventory with the RCPs running, followed by loss of the RCPs, could result in a homogeneous fluid condition in the RCS. Under this condition, fluid heated in the core would flow into the upper plenum, through the hot legs, the steam generators, the RCPs, and back into the reactor vessel and the core via the cold legs. Although multidimensional fluid flow conditions probably exist in the reactor vessel after RCPs are lost, one may estimate that thermal response is still reasonably realistic if modeling is restricted to one dimension provided the natural convection flow rates are high. For this case, existing analysis codes could be applied to roughly estimate steam generator tube response. If the response was not clear, then multidimensional analyses could be applied to estimate the influence. In such a case, uncertainty in the multidimensional analyses might not be of as great a concern as for the situation of multidimensional behavior dominating system response. However, nonexistence of the loop seal due to continuous RCP operation is an unlikely situation since the majority of conditions during which steam generator tube integrity is of concern will involve loss of off site AC power, and RCPs will be unavailable. To our knowledge, a complete, accurate, analysis of a four loop Westinghouse NSSS has not been performed for these conditions. In additional to an analysis approach, closure of consideration of this aspect of SG tube behavior could be obtained if the probability of occurrence of the RCS homogeneous fluid condition was established as negligibly small in contrast to other situations where SG tubes were shown to lose integrity, or if the risk associated with the condition was established as negligible when compared to other Seabrook Station risks.

A second situation involving free circulation in the RCS might be obtained if one considers the RCPs as being restarted in response to high core temperatures, as prescribed in the emergency procedures. For this case, sufficient head might be developed to clear the loop seals of water, and rehomogenize the RCS fluid, thereby generating the condition described in the previous paragraph. To our knowledge, rehomogenization under these conditions has not been established to occur at Seabrook. Insofar as SGTR at Seabrook is concerned, the issue can be dealt with as outlined in the previous paragraph.

A third situation of removal of loop seals also potentially exists during boil down of the RCS inventory. One may postulate that the ICC condition occurs with the loop seals in place, and that some other mechanism causes their disruption. This could occur if a sufficient pressure difference occurred across the seals that they were forced out of the low regions. Several analyses have been conducted which include consideration of this behavior, and none showed loss of the seals. One would expect that consideration of this condition could be closed if analyses applicable to Seabrook could reasonably establish that the seals remain.

A final condition can be visualized if one considers a LOCA to have occurred in the RCS. For example, a small cold leg LOCA (or an RCP seal LOCA) could be located between the two natural seal regions of the crossover leg and the reactor vessel lower plenum. Removal of RCS mass might occur under conditions such that the seal water was forced out of the RCS via the break. Elimination of consideration of this effect with respect to impact upon risk could be considered on the basis of a thermal-hydraulic investigation of RCS behavior, establishing that the potential impact on risk of the behavior is negligible in comparison to other established risk contributors, or both.

3.2.5 PSNH Modeling Considerations

The PSNH has reported application of the MAAP 3.0 code to investigation of natural circulation flow in Seabrook (Refs. 12 and 17). This code treats the major phenomena, including approximations of multidimensional flow and fission product (heating) movement, and is applied to the regions of the RCS which are affected by the SGTR issue.

Quasi-steady momentum balances and continuity equations are used to represent natural circulation flow, and the steam generator inlet plenum behavior is represented by quasi-steady mixing models. The modeling represents gas and wall temperatures using conventional lumped parameter models, with 15 gas control volumes and 17 two dimensional heat sinks. (Several volumes are subdivided into further volumes for some types of calculations. The core, for example, contains 70 nodes which comprise the core volume node.) The control volumes are based upon approximations of the flow patterns which were seen in the Westinghouse experiments on a scaled NSSS (Refs. [illegible], [illegible], and 14). This basis for definition of control volumes means that deviations from the assumed flow pattern and flow instabilities may not be represented in the model. Experimental evidence shows that there are asymmetric flow patterns, for example, which are not modeled, and which could lead to tube heating conditions which would not be calculated. Further, although instabilities have not been experimentally observed at the Westinghouse test facility, one must accept this evidence with care since testing with fluid conditions which closely simulate those expected in an NSSS are just being initiated.

Use of the lumped parameter model requires further discussion. Unlike computer codes such as COMMIX, which can determine flow patterns within certain bounds provided the configuration is properly modeled, a lumped parameter model is based more strongly upon a presupposed flow behavior. Although such representation can be valuable and accurate under certain conditions, such assumed behavior must be verified before it can be accepted. The preliminary Westinghouse experiments, as discussed briefly in the next section of this report, and some COMMIX and MELPROG calculations (Refs. 15 and 16), represent steps in this direction, but further evidence is necessary before we can accept the assumption as verified. (The experiments are somewhat preliminary, and the COMMIX and MELPROG calculations have not, to our knowledge, been carefully checked against experimental evidence.) We further note that, to our knowledge, there has been no independent study of the version of the MAAP code used for the analyses. At a minimum, we believe a reasonable knowledge of code modeling and logic, in addition to a verification program, are necessary for acceptance of the calculated results. (We note that EPRI has a MAAP verification program underway.)

One aspect of the modeling appears worthy of further consideration. The steam generator inlet and outlet plena are assumed to be completely mixed in the PSNH studies being reviewed here, and they are represented by single nodes with uniform properties. The Westinghouse facility test data indicate a partially stratified, partially mixed SG inlet plenum (Ref. 14), and modeling for the test facility is based upon a quasi-steady state model in which partial mixing is assumed at various (limited) locations between streams of different origins. Reference 14 describes the situation as follows:

> "The flow in from the hot leg rises rapidly in a plume in the inlet plenum and induces mixing. Some of the cold return flow from the tube bundle does avoid mixing, particularly near the divider which is furthest from the hot leg. Much of the cold return tubes' flow plunges through the hotter stratified fluid layer that spreads across the bottom of the tube sheet. The mixing flows could be observed from dye injection and from observation of light through the density gradients that resulted. Temperature measurements in the inlet plenum are indicative of mixing. The tubes carrying hot fluid from the inlet plenum were generally concentrated in the area above the hot leg entrance and scattered in the regions further away. Cold return tubes were also scattered and were found in the area above the hot leg inlet also."

Test facility modeling of the phenomena uses a six equation approximation which contains an experimentally determined mixing parameter.

We believe the assumption of complete mixing used for the PSNH investigations will reduce SG tube temperatures when contrasted to the experimentally identified situation. This modeling and its implications need further consideration. (This comment is repeated a number of times in the discussion of calculated NSSS response in the following sections of this report.)

3.2.6 Comparisons of Calculations to Experimental Data

Several comparisons between MAAP code calculations and experimental data have been briefly described by PSNH and its contractors to the BNL and NRC staffs (Refs. 12, 17, and 18). These are discussed below.

1. Core and upper plenum flow rates. The following comparison of experimental and calculated values was presented:

Test Condition	Experimental Flow Rate	Calculated Flow Rate
28 KW Water Test	0.54	0.50
0.9 KW SF_6 Test	0.016	0.017

2. Hot leg and steam generator natural circulation. Comparison of several parameters was provided:

Item	Experimental Value	Calculated Values for Indicated Number of Steam Generator Tubes Carrying Flow in the Out Direction		
		6	12	24
Heat Transfer Rate, KW	2.43	2.0	2.6	2.9
Entering Fluid, °C	30	30.7	29.2	28.4
Exiting Fluid, °C	19	24.2	21.7	18.8
Coolant, °C	10 - 11	9.4	11.2	12.8

where the entering fluid is flowing into the steam generator inlet plenum from the upper portion of the simulated hot leg, and the exiting fluid is flowing from the lower portion of the steam generator inlet plenum back toward the simulated reactor vessel along the bottom of the hot leg. The coolant temperature is that of the water leaving the secondary side of the simulated steam generator, and thus, can be related to the heat transfer rate from the primary to the secondary sides.

These results are clearly promising. Continuation of the comparisons with a wide range of experimental conditions in the same test facility, and with no changes in the modeling except for the change of experimental conditions and fluid properties, would be helpful in code verification. Extension of the same modeling approach to other experimental data (such as flow in ducts and components) would provide further confirmation. Completion of confirmation of modeling adequacy could typically include comparisons of existing data obtained in large facilities, selected contrasting of alternate calculational methods to portions of the code under consideration here, and establishment that scaling is adequately represented by the code.

3.2.7 Calculated Seabrook Thermal Response to Severe Accidents

Calculated behavior to selected accident conditions has been summarized by PSNH. Principal results and our comments are as follows:

1. Peak Steam Generator Temperature for Loss of AC Power and Loss of Feed Water Flow. The following temperatures and flow rates were calculated at the indicated condition:

Location	Temperature, °K	Flow Rate, kg/sec
Core (Peak)	1800	18 (recirculating between
Upper Plenum	1160	upper plenum and core)
Hot Leg	760 (wall)	2.4 (countercurrent)
SG Inlet Plenum	850	-
SG Tube	700 (wall maximum)	3.3 (total in each direction)
SG Outlet Plenum	640	-

PSNH indicated that the hottest core node would melt at about 30 seconds from the time of these values, and that the generated hydrogen and blockage due to relocated core material would cause natural circulation between the core and the upper plenum to almost stop. At this point, the upper plenum would begin to cool due to energy transfer to the hot legs.

Plys (Ref. 18) presents additional information which shows temperatures continue to increase after vessel blowdown, with the peak upper plenum temperature exceeding 1200°K for a short time. The tube temperature continues to increase for the time of the calculation (20,000 sec, with vessel rupture at 11,600 sec), reaching a maximum of about 1020°K. We would be interested in seeing plots of other parameters over the span of the calculations, including the hot leg and SG plena temperatures, to better understand the interactions and modeling.

In response to a question, PSNH indicated they had not performed a detailed analysis of reactor vessel hot leg nozzle thermal behavior, but felt a temperature of the order of 1000°K was necessary to cause failure. Discussion also identified that there was significant steam circulatory flow in the secondary side of the steam generator tubes, and that this steam, which was at a pressure corresponding to the steam generator safety valve settings, represented a significant heat sink. Further, it was an effective medium for transferring heat from hot tubes to colder tubes, thus tending to reduce the maximum tube temperature. This raises a question of what results would be obtained if the steam generators were depressurized to atmospheric pressure, thus maximizing pressure differential across the tubes and simultaneously removing a heat sink which could influence temperatures throughout the NSSS. (A sensitivity analysis was conducted in which this was one of the parameters.)

Information presented in Reference 12 and the above summary table shows fluid flow rates in the hot leg of roughly 2 kg/sec as contrasted with a rate above 3 kg/sec in the SG tubes for the time after effective boiloff of water from the core until melt through of the reactor vessel. Cooling via steam contained in the SG secondary side is thus an effective medium for cooling the SG inlet plenum. The total mixing assumption pertinent to fluid in the plenum is, in turn, effective in preventing hot fluid from reaching the tubes. This high tube flow rate is also effective in transferring heat from the reactor vessel to the SG secondary side, thus helping to limit fluid temperature in the hot legs as well.

We believe a study would be beneficial of behavior with the SG secondary side depressurized after SG dry out. Now there would be no heat sink on the secondary side, and tube flow rates may be lower due to less of a driving force for natural convection flow in the SG. Further, we would expect to see further stratification in both the hot leg and the SG inlet plenum (the latter not being allowed in the PSNH supported analyses due to the modeling assumption of complete mixing). We pose the question of whether temperatures may be significantly above what was calculated by PSNH and its contractors under these conditions.

2. Operator Induced Depressurization. This calculation was based on the assumption that the operator would open a PORV when the core exit thermocouples indicated 1200°F. The calculations indicated accumulator discharge approximately 1400 sec after opening the PORV, with the RCS depressurized prior to vessel failure. The accumulators were emptied at about 10,600 sec, and vessel failure occurred 2000 sec later. Accumulator water was found to cause a small additional amount of hydrogen production. Phenomena associated with depressurization and hydrogen decreased the effectiveness of heat transfer between the core and other regions of the NSSS. Steam generator inlet plenum temperature reached a peak of roughly 850°K during the depressurization, then cooled, and remained below 650°K for the remainder of the calculation (20,000 sec total calculation time, with PORV opening at approximately 8000 sec). Maximum tube temperature was about 650°K, and was reached at 20,000 sec, being identical to the inlet plenum temperature at that time. (Note RCS pressure is that of the containment following depressurization earlier in the calculation.)

We note that RCS pressure behavior (Ref. 18, Figure 4-4) is different for the base case and the PORV opening case prior to the time of opening of the PORV. We would like to discuss these differences for all parameters and we would like to understand the reasons they exist. (We note there is little difference in temperature over the range in question, and temperature is the important parameter for the SGTR issue.)

Volatile fission products represent about 20% of the decay heat, and the behavior of this energy source is calculated in the MAAP code. The calculations illustrated movement of the decay heat source. About 10% of the decay heat was associated with fission products which were in the upper plenum at the time of vessel failure. A small amount was in the hot legs, as was also the case for the pressurizer. The amount in the steam generator tubes was not significant. (Most of the CsI was in the upper plenum at the time of vessel failure, with about 10% of the CsI in the hot legs.)

3. Other Variations and Uncertainty. Several sensitivity calculations were performed to obtain a better understanding of behavior. These included:

a. Higher core melt temperature
b. RCP seal failure
c. SG secondary side blowdown
d. Core resistance variation
e. Reduced SG tube circulation
f. Core blockage changes.

These are discussed below.

a. Higher Core melt temperature. A case was run in which core melt temperature was assumed to be 3000°K as contrasted to the base case 2500°K. This was intended to delay the onset of core geometry degradation, which in turn provides more time to heat other portions of the RCS. The 500°K change in melt temperature was found to cause only a few degrees change in SG tube temperatures, which was attributed to the extremely rapid temperature increase rate in the core as melt temperature is approached, and a concomitant small increase in the time available for heat transport to the steam generators.

The model is based upon assumed symmetric behavior, whereas some asymmetries have been found experimentally. If these contributed to a preferential flow of hot fluid near one of the hot legs, that leg might transport hot fluid toward a steam generator and provide higher temperatures than determined in the calculation. This could increase the computed impact of the sensitivity calculation.

A second aspect of the modeling that would act to reduce the calculated impact of the sensitivity run is the assumption of mixing within the steam generator inlet plenum. We believe an assessment of this effect is needed, as previously identified.

b. RCP seal failure. RCP seal failure, if it were to occur, was felt to be a leak in the range of 50 gpm (water) per seal. This was modeled, with the break occurring in all four RCPs at 45 minutes after initiation of the accident. This was found to have an insignificant impact on the results (Refs. 12 and 18).

. PSNH also addressed preexisting leaks in SG tubes which are within technical specifications. These were stated to be small in comparison to the 50 gpm flow rate associated with seal leaks, and consequently were argued as being negligible (Ref. 17).

We believe the preexisting leak situation has a negligible impact on NSSS behavior as long as the leak remains small, but do not accept the argument advanced by PSNH as the reason. A comparison of the velocity associated with flow in a tube due to natural circulation with that associated with the leak, with establishing that the latter was negligible, would be more convincing. Similarly, a comparison of flow rate induced by the RCP seal rupture to that expected for natural convection flow would be helpful.

Provision of temperature information pertinent to fluid passing through the RCP seals would be helpful.

c. SG secondary side blowdown. Plys (Ref. 18) reports a calculation to investigate the effect of reduced cooling on the SG secondary side in which the steam generator PORVs are assumed to stick open, thus depleting the secondary side of a high pressure steam atmosphere. Drastic differences were discovered early in the accident due to cooling as the steam generators blew down. Sufficient cooling was provided that the pressurizer emptied due to primary fluid contraction. Reactor vessel failure occurred slightly earlier in this case as contrasted to the base case due to less heat removal from the primary system following removal of the secondary side heat sink. An initial peak in SG inlet plenum temperature of 860°K is identical to that of the base case, but occurs about 500 sec earlier. Following the initial peak, the plenum temperature behavior is similar to the base case, although displaced in time, but is 50 to 100°K higher over the remainder of the transient.

We suggest the calculation be conducted by assuming the PORV is stuck open after all water has been vaporized. This avoids the situation of overcooling associated with the early opening, and may be more compatible with some postulated operator actions associated with late attempts to deal with approaching core melt.

Again, we are concerned with the influence of assumed mixing in the steam generator inlet plenum and the impact upon calculated results.

d. Core resistance variation. Variation of the resistance of the core to flow was evaluated by lowering the axial and cross flow core friction factors in one calculation. This slightly increased heat transfer to the steam generators and correspondingly increased time to vessel failure. There was a slight tube temperature increase, but in general, the calculation showed little sensitivity of tube temperature to the change in core friction factors.

e. Reduced SG tube circulation. Selection of lower limit values of the number of steam generator tubes participating in flow from the inlet to the outlet plena was used for another sensitivity calculation.

This provided lower values of steam generator natural circulation flow relative to the hot leg natural circulation flow rate, and reduced cooling of the steam generator inlet plenum due to flow from the outlet plenum. Slightly less heat was removed from the reactor vessel due to the lowered flow rates, and vessel failure occurred slightly earlier. These changes were insignificant. However, the steam generator inlet plenum was found to be about 150°K higher than for the base case, reaching a temperature of 980°K for a short time. Steam generator tube temperature was relatively unaffected.

Comparison of inlet plenum and tube temperature transient behavior (References 17 and 18's Figures 4-11 and 4-12) appears to indicate a significant thermal inertial associated with the tubes, which do not increase in temperature to a significant degree in contrast to the temperature of the source fluid in the steam generator inlet plenum. We believe this needs further discussion. For example, what is the location of the tube temperature and does this location correspond to the highest tube temperature?

Again, as previously stated, the influence of the assumption of complete mixing in the steam generator inlet plenum will impact the results. A portion of the concern is that reduced flow rates may lead to greater stratification and less mixing in the SG plena, a phenomenon that is not modeled in the PSNH reported evaluations, and a phenomenon with the potential to increase tube temperatures over what was reported.

f. Core blockage. In this calculation, a delay of blockage in the core at the time of core melt to the time the node was completely filled with refrozen eutectic was assumed. This was done to continue core oxidation and core/upper plenum flow for a longer time. For this case, the maximum sustained SG inlet plenum temperature is roughly 1060°K, with a short time (less than 50 seconds) temperature "spike" to about 1120°K.

We again reiterate the concern with SG inlet plenum modeling and its impact upon the results.

c. Sensitivity Summary. An approximate comparison of the results of the sensitivity study is provided in Figure 1. The major early effect on increased tube temperature is due to changing the SG tube flow characteristics. Later, and with the greatest impact, is the effect of delaying formation of blockage in the core, which allows continued circulation of hot fluid through the core where the temperature is increased, as opposed to a drastic reduction in heat transport between the core and other RCS components when a core geometry change occurs.

4. Steam Generator Tube Strength. Plys, in Reference 18, Appendix B, addresses SG tube integrity. The presentation is based upon the SG secondary side pressure being at the SG safety or relief valve setpoints which, as previously discussed, may not be the case. We note that Plys identifies nominal hoop stresses of 9300 to 10000 psi for the assumed conditions. Hence, the case of the SG secondary being depressurized will result in a nominal hoop stress of roughly 19,000 psi. This stress, substituted into Reference 18's Figure B-6, results in a Larson Miller parameter of about 37. The Larson Miller parameter is defined as:

$$LM = T(20+\log t_r) \times 10^{-3}$$

where:

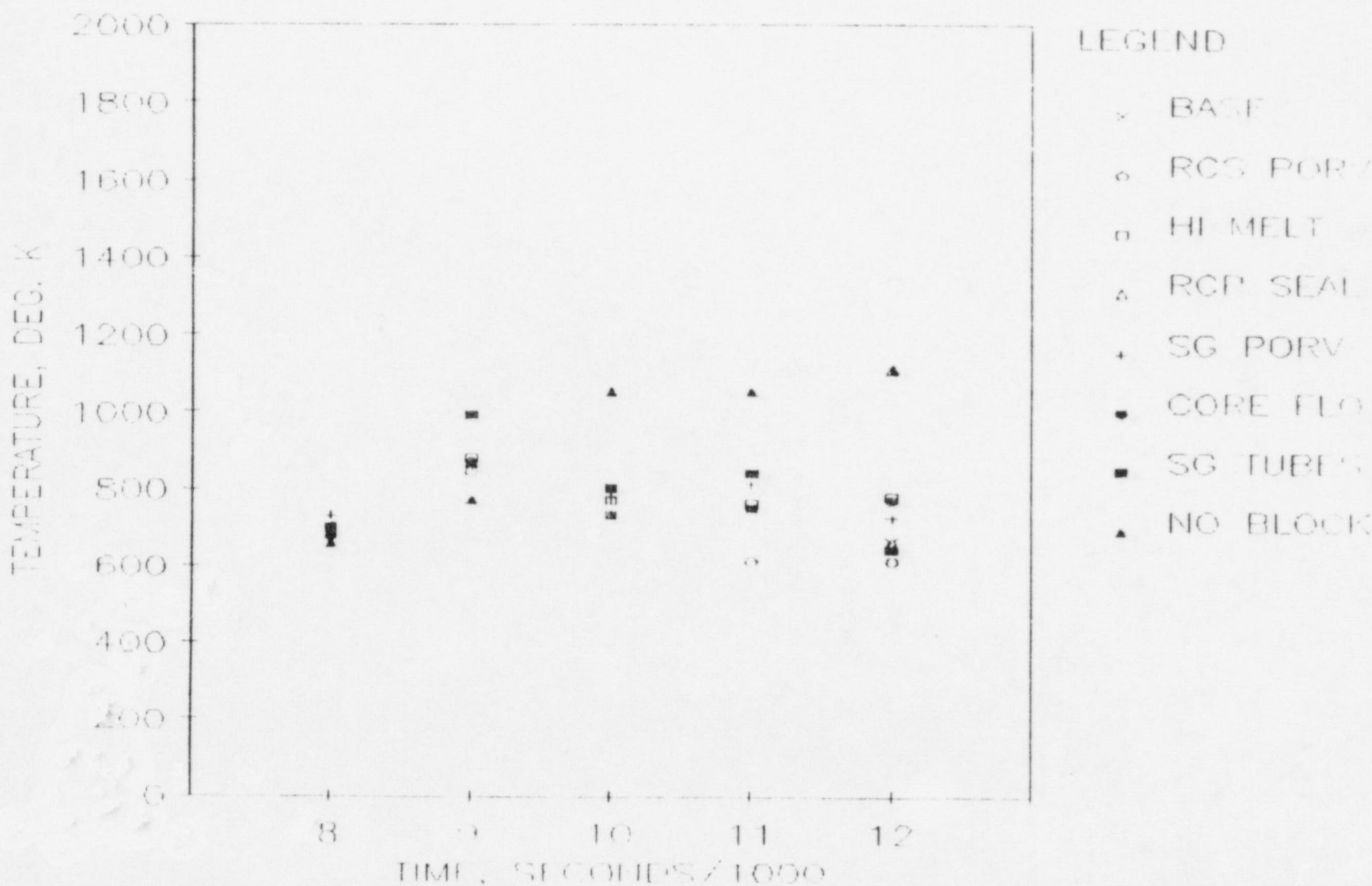
T = temperature, ^oR

t_r = time to rupture, hrs.

Substituting a temperature of 1090^{o}K (the value used by Plys to conclude the rupture time would be greater than 2.5 hrs) yields a time to rupture of about 5 minutes, a significant change from the Plys value.

Plys could have selected 1090^{o}K as conservative, with no need to consider an alternate since the no tube rupture position was supported by the

FIG. 1 STEAM GENERATOR INLET PLENUM TEMPERATURE COMPARISONS



result. If we recognize this possibility, and select a less conservative $1000^{\circ}K$, we find a rupture time of about 3.5 hours. These temperatures can be contrasted to the SG inlet plenum temperatures provided in Figure 1, with recognition that these are not tube temperatures, but also with the recognition that some of the parameters contributing to the temperatures remain to be evaluated.

Clearly, we are in a temperature region where relatively small changes have a significant impact upon creep rupture time. Equally clearly, tube stress could be roughly a factor of two higher than the value used to justify that tubes would not rupture. We conclude the picture is not as clear as presented in Reference 18, which presented a conclusion that tubes would not be ruptured.

3.2.8. Other Considerations

In Reference 17, PSNH stated that if one postulated creep rupture failure of steam generator tubes, the pressure inside the previously dried out and isolated steam generator secondary side would increase until the steam generator PORVs setpoint was reached, at which time the valves would lift and modulate until reactor vessel melt through and RCS depressurization into the containment. During the periods of SG PORV opening, there would be a high leak rate bypass condition directly from the RCS to outside the containment. They further stated that after vessel melt through, the leak rate out this path would be low and would correspond to any low pressure leakage through the reclosed PORV. They note this leak path could be enhanced if the SG safety valves also lift and fail to reseat properly; however, they believe it unlikely that the safety valve setpoint would be reached.

As previously discussed, we do not believe an individual tube would rupture, but instead believe there would be a massive failure in one steam generator. (Once the failure initiated, we would expect the RCS to depressurize rapidly, which would reduce stress on tubes in other steam generators.) It is difficult to postulate a PORV modulating this condition. It is further difficult to postulate the PORV or the safety valves would not be damaged when exposed to these conditions, and therefore their reclosing may be questionable. Finally, if the conditions which led to the accident sequence involve a loss of all AC power, which is one of the likely situations given a severe accident scenario, we pose the question of how long the PORVs can be expected to modulate pressure assuming they are not damaged by the fluid being modulated.

Plys (Ref. 18) has identified that the MAAP code does not model certain aspects of SG tube temperature, and a method of obtaining temperature was discussed. Aside from the impact of secondary side steam as a cooling medium, we are concerned about local heating due to small leaks. Such a leak could cause a small amount of hot fluid to pass through a localized area into the SG secondary side, with different heat transfer characteristics and tube temperatures than one would encounter with the treatment of overall inside to outside heat flow utilized by Plys in their estimation. Whether this is important to localized tube temperature over a sufficient area to be of concern should be addressed. (Note the effect could also be concentrated in an adjoining tube. This can be visualized by picturing a tube with a small hole which directs hot RCS fluid onto the secondary side surface of an adjoining tube, while the inside surface of that same tube is exposed to hot RCS fluid.)

3.3 Accident Likelihood

PSNH has estimated the mean annual frequency of accidents in which the core melts with the RCS at high pressure and the SGs dry as bounded by a value of 4.5×10^{-5} per reactor year (Ref. 17). This is composed of the following plant damage states:

DRAFT

Plant Damage State (PDS)	Mean Annual Frequency
3D	1.5×10^{-5}
3FP	8.9×10^{-6}
4A	1.4×10^{-5}
4C	1.7×10^{-7}
4D	2.8×10^{-6}
4E	2.2×10^{-11}
4FP	1.2×10^{-7}
8A	3.9×10^{-6}
Total	4.5×10^{-5}

The accident sequences which comprise the PDSs include transient and loss of off site power sequences with failure of all emergency feedwater, failure of feed and bleed with loss of all emergency feedwater, and transients without scram. PDS 8A consists of eight sequences which involve station blackout and emergency feed water failure with recovery of containment heat removal.

PSNH also addresses the potential impact of tube rupture on this information. They have assigned a high chance of no containment failure to PDSs A. PDSs C and D are considered as leading to a high likelihood of long term containment overpressure failure. PDSs FP are a high chance of small bypass, and PDS E is a high chance of large bypass. Hence, PDSs A, C, and D would be impacted by SGTR, and FP may represent some impact. Addition of the appropriate values indicates that the likelihood of being in a condition where SGTR could affect the results is about 4×10^{-5} (as contrasted to the assumption of no SGTR).

PSNH considers these values to be bounding because some of the values include states with water on the steam generator secondary side, for which SGTR is not a concern, certain operator recovery actions have been neglected, and RCS depressurizations prior to core melt have not been considered. As previously discussed, operator depressurization is one of the potential steps which one could consider to mitigate SGTR. PSNH estimates the frequence of operator failure to depressurize as less than 10^{-2} to 10^{-3} per demand, provided procedures are modified and adequate operator training is provided. These values lead to a conclusion that the frequency of obtaining conditions under which SGTR would be of concern can be reduced to of the order of 10^{-7} to 10^{-8} per reactor year.

Although these values appear reasonable, we note that the conditions which led to the factor of 10^{-2} to 10^{-3} reduction do not presently exist. We further would need substantiation for these values prior to acceptance.

Discussion is also provided concerning the likelihood of SGTR if exposed to high pressure core melt conditions (Ref. 17). PSNH points out that their calculations show SG tube temperatures that are roughly 200 to 300°F below what would be required for creep rupture, and this is identified as principally due to cooling by steam on the SG secondary side. Several things are necessary for acceptance of the tube temperature conclusions, including, as discussed elsewhere, substantiation of the calculational technique and investigation of the likelihood of the SG secondary side having a significant steam inventory (which also means having a significant pressure).

Finally, PSNH estimates a 9% chance that failure of SG tubes will not occur before reactor vessel melt through or piping nozzle failure. This value, combined with the prior estimates of frequencies, appears sufficient to establish that SGTR is not of concern as a significant contributor to risk. Therefore, one can reasonably anticipate that substantiation of the various items which led to the conclusion, as discussed in this communication, should provide substantiation of the above preliminary conclusion.

3.4 Additional Reviewer Observations

A number of observations and comments have been made in the previous discussion. We offer the following additional comments:

1. Much of the modeling utilized in the calculations has not been documented. We understand this is underway. Such documentation will be helpful in the continuation of the review.

2. The outside of the hot legs is assumed to be adiabatic. This probably introduces a small conservatism into the results with respect to hot leg temperature. The impact on other parameters is probably negligible. With respect to the hot legs, the parameter of interest may involve a relatively thin wall connecting pipe that is exposed to high fluid

temperature, and whose temperature will follow fluid temperature more closely than is the case with the relatively massive hot leg; or the vessel nozzle region of the hot leg, which will be more closely allied with fluid circulating rapidly within the upper plenum. Thermal response of these regions may be critical in determination of the failure point of the RCS pressure boundary.

3. Although the limited experimental evidence reveals some symmetry in flow behavior within the reactor vessel, there are also unsymmetrical flows and temperatures. We understand the MAAP calculations are based upon modeling the upper plenum fluid as a single volume. This appears to be a nonconservative approach.

4. STEAM GENERATOR TUBE RUPTURE CONCLUSIONS

The above discussed considerations lead us to the conclusion that this topic is in a developing state, with knowledge being rapidly accumulated. Insufficient information is presently available for one to conclude that SGTR cannot occur as a result of severe accident conditions.

Our judgement, at this juncture, is that a carefully conducted and thorough evaluation on the part of PSNH, that utilizes information which either exists or will be available within the near future, can establish that the likelihood is small that a SGTR will result due to overheating during severe accidents. Further, our judgement is that the risk associated with SGTR can be shown to be negligible for these conditions. Our judgement needs to be substantiated. We have encountered too many unanswered questions, unsubstantiated assumptions, and potential conditions which could lead to calculation of increased temperature to accept a conclusion that SGTR will not occur under circumstances such that the associated risk can be neglected. We note, as a qualifier to these conclusions, that our review is not complete, and, in addition, work is ongoing to provide further information.

Existing knowledge would support a conclusion that SGTR is not a problem if the RCS is depressurized. Consequently, reasonable assurance that progressions toward core melt would not occur at high RCS pressure, coupled with suitable technical backup for a conclusion that low pressure is not of concern, would eliminate our concern regarding SGTR under severe accident conditions. We have not conducted an evaluation of the trade-offs associated with such an approach, nor have we been provided with information that would either support or negate RCS depressurization under severe accident conditions. We have not provided a recommendation regarding whether RCS depressurization is attractive when all pertinent factors are considered.

Substantiation of a judgement that SGTR is not a concern under severe accident conditions with the RCS at high pressure can be obtained through a combination of analytic and experimental investigations. The ongoing test at Westinghouse in which reasonably close similitude is claimed between the test facility and appropriate parts of a Westinghouse four loop NSSS should provide key data which can be applied to assist in the confirmation of analysis techniques. Selected test data from other facilities and further examination of the analysis techniques, coupled with necessary changes when they are uncovered, should provide sufficient confirmation that reasonable reliance can be placed upon accident analyses pertinent to this issue. Application of a reliable analysis technique to issue investigation should then provide the necessary background to resolve this issue.

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