U. S. NUCLEAR REGULATORY COMMISSION

REGION I

Meeting No. 50-293/86-30

Docket No. 50-293

License No. DPR-35

Licensee: Boston Edison Company M/C Nuclear
800 Boylston Street
Boston, Massachusetts

Facility Name: Pilgrim Nuclear Power Station

Meeting At: USNRC, Region I, King of Prussia, PA

Meeting Conducted: July 30, 1986

Prepared by: Jack Strosnider for M. McBride, Senior Resident Inspector — 8/19/86 Date

Approved by: Jack Strosnider J. Strosnider, Chief, Projects Section 1B, DRP — 8/19/86 Date

Meeting Summary: A management meeting was held at NRC Region I on July 30, 1986 to discuss plant status, SALP improvement status, overtime control, startup checklist, Mark I Containment enhancement and other issues. The meeting lasted about two hours. Members of the news media attended the meeting.

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PDR FOIA
TYE87-010 PDR

DETAILS

1.0 Purpose

The management conference was held at the request of NRC Region I. The licensee was asked to discuss recent program improvements and other topics of interest at the plant. The licensee announced prior to the meeting, that the current outage would be extended until early 1987.

2.0 Meeting Summary

Dr. T. Murley, NRC Region I Administrator, opened the meeting with a brief discussion of recent regulatory actions, including the Augmented Inspection Team, Confirmatory Action Letter (CAL) 86-10, and the agreement to seek Region I approval prior to starting the plant. Mr. A. L. Oxsen, Vice President of Nuclear Operations, opened Boston Edison's presentation by discussing recent licensee program changes, including a management realignment under an Executive Vice President and the decision to enter into refueling outage number seven. Mr. Oxsen indicated that Boston Edison will complete fire protection modifications required by 10 CFR 50, Appendix F during this outage. Boston Edison also plans to work on further improvements in management control processes and implement certain Mark I containment enhancements during the outage.

a. Plant Status

Current outage work was discussed, including inspections of the residual heat removal (RHR) pump impeller wear rings and ongoing containment leakage testing. No visual indications have been detected in the wear rings of the two RHR pumps that had been disassembled. The remaining two RHR pumps will be inspected in the near future. The licensee discussed bearing wear that had been detected in one of the pump motors and indicated that the pump impeller wear rings will be tested for hardness and dye penetrant checked.

The status of local leak rate testing (LLRT) was reviewed. Problems with the leakage test procedure for the feedwater check valves that may have contributed to a falsely high as-found leakage rate were discussed. The licensee stated that the LLRT procedure would be modified prior to as-left testing. The licensee's plans for integrated leakage testing were also discussed.

b. SALP Improvement Status

Improvements in management programs that relate to SALP-identified weaknesses were discussed, including the work control process at the station, the development of performance indicators, communication of problems within the licensee organization, and staff accountability. The licensee stated that full-time planners would be assigned to

each management section head, to help track and prioritize section work. Periodic morning meetings between the Plant Manager, QA, and the section heads were also described.

The licensee's preparations for this year's emergency planning exercise were discussed. The licensee indicated that several practice drills were planned.

c. Overtime Control During Startup

The projected number of licensed operators available for the plant start up in 1987 were discussed. The licensee indicated that five shifts will be manned and that an additional senior licensed operator will be added to each watch. Boston Edison agreed to discuss planned overtime controls during the next plant startup in more detail during the next NRC management meeting.

d. Startup Check List

The preparation of a detailed check list of items that should be completed prior to a plant startup was discussed. The licensee indicated that the startup will last several weeks and will contain hold points at particular power levels. Regional management indicated that the checklist and the startup itself will be reviewed in considerable detail and that Regional Administrative approval will be required prior to proceding past certain critical hold points to be identified later. The Regional Administrator indicated that he will not grant approval for a restart until after the NRC has reviewed the licensee's startup plan.

e. Mark I Containment Enhancements

The licensee discussed possible improvements for the Mark I containment at Pilgrim. The changes will probably be made to enhance such features as containment oxygen control, containment venting, and containment spray. The need to consider these changes carefully and avoid inadvertently degrading other aspects of containment was emphasized by regional management. The licensee agreed and indicated that the enhancements would receive a thorough review. The licensee also indicated that they would be working closely with Vermont Yankee, another plant that is reviewing the Mark I containment design.

f. Other Topics

Other discussion topics included maintenance planning, the maintenance backlog, once per cycle surveillance test scheduling, outage activities, and security program problems.

The licensee indicated that efforts were underway to reduce the maintenance backlog and implement a maintenance planning group. The licensee does not expect the planning group to be fully developed until the end of the current outage (although planning procedures should be in place by this fall). The licensee stated that the number of outstanding maintenance requests has been decreasing, although the number increased during the recent Boston Edison strike. The licensee indicated that the number of open maintenance requests by itself was not a good indicator of the plant physical condition. On another maintenance item, the licensee indicated that the deficiency stickers for the control rod position indication system in the control room could not be cleared without extensive under vessel work. The status of these repairs was questioned during a recent visit to the site by an NRC commissioner.

The licensee briefly discussed some of the recent surveillance test scheduling problems, and indicated that the testing frequency for once-per-cycle tests would be clarified with NRR. The NRC regional staff discussed the importance of carefully reviewing the surveillance frequency for each type of once-per-cycle test and noted that certain safety-related electrical equipment, required to be calibrated once-per-cycle, had recently been found out of calibration.

Security staffing problems and the amount of inoperable security equipment (requiring compensatory measures) was discussed. Boston Edison indicated that staffing levels for the contractor security force would be increased and overtime closely monitored by Boston Edison management in the future. Boston Edison agreed to furnish the Pilgrim Senior Resident Inspector with the expected completion date for engineering evaluations of the scope of security equipment improvements. Subsequently, the licensee indicated that the evaluations would be completed by August 18, 1986.

Boston Edison stated that a major decontamination effort would be made in the plant during the current refueling outage. The licensee expects to decontaminate areas with safety-related equipment in the reactor building quadrant rooms so that these areas can be toured without protective clothing.

3.0 Concluding Statements

At the end of the meeting, Dr. Murley requested that Boston Edison be prepared to discuss the following issues during the next management meeting:

-- Overtime control during the plant startup
-- The expected scope of startup testing
-- Containment leakage test results and fixes
-- Mark I containment enhancements
-- Updated outage schedule

Dr Murley also indicated that the NRC will need to review Boston Edison's start up plans and hold points.

ATTACHMENT TO MEETING REPORT

50-293/86-30

MEETING ATTENDEES

Boston Edison Company (BECo)

J. Lydon	Chief Operating Officer
E. Howard	Vice President - Engineering & QA
A. Oxsen	Vice President - Nuclear Operations
A. Pedersen	Plant Manager
K. Roberts	Director of Outage Management
R. Swanson	Nuclear Engineering Manager
S. Hudson	Operations Section Head
C. Gustin	Manager, Public Information

U. S. Nuclear Regulatory Commission

T. Murley	Regional Administrator
R. Starostecki	Director, Division of Reactor Projects
S. Ebneter	Director, Division of Reactor Safety
H. Kister	Chief, Reactor Projects Branch #1
L. Bettenhausen	Chief, Operations Branch
J. Durr	Chief, Engineering Branch
J. Strosnider	Chief, Reactor Projects Section 1B
M. Shanbaky	Chief, Facilities Radiation Protection Section
M. McBride	Senior Resident Inspector, Pilgrim
G. Nejfelt	Resident Inspector, Hatch
G. Smith	Safeguards Specialist
P. Leech	Project Manager, NRR

Press

J. Gehorgam	Associated Press
J. Impemba	Boston Herald

VY wants relief from [illegible]
other commitments to
do the work on this topic
Will work w/Rooney + Muller
to identify which items.

Presented by
Warren Murphy
Rich Laudwick
Mgr ops Support
Dr. ? Scholls ? YA

VERMONT YANKEE

CONTAINMENT SAFETY STUDY

Rasmussen has
reviewed - thinks
its a good try.

NRC review
to be done by Oct
end. (include questio
Want VY responses
early (mid NOV) so we can publish
for comment proposed
generic req'mts for mark I Cont
in December 86.
Want to compare VY design
etc to the proposed req'mts
per Bernero.

FOIA-87-010
F/4.

ATTENDEES
VERMONT YANKEE CONTAINMENT SAFETY STUDY
SEPTEMBER 11, 1986 MEETING

NAME	ORGANIZATION	TITLE
D. R. MULLER	NRR/DBL/PD#2	PROJECT DIRECT
V L ROONEY	NRR/DBL/PD#2	PROJECT MANAGER
Ed Wenzinger Sr.	NRC Region 1	Branch Chief
Jerry Hulman	NRR/DBL/PSB	Branch Chief
D.B. Vassallo	NRR/DBL/FOB	Branch Chief
R W Houston	NRR/DBL	Dep. Dir.
Mohan C Thadani	NRR/DBL/PD#2	Project Manager
DAVID SHUM	NRR/DBL/FOB	Reactor Sys. Engineer
GUS LAINAS	NRR/DBL	ASST. DIR.
FAROUK ELTAWILA	NRR/DSRO/RIB	Sr. Reactor Systems En
Wayne Hodges	NRR/DBL/RSB	Branch Chief
R J LODWICK	Vermont Yankee	Operations Support M
JAY K. THAYER	YANKEE ATOMIC	ENGINEERING MG
S. P. SCHULTZ	YANKEE ATOMIC	MGR. NUCLEAR EVAL
Warren P Murphy	Vermont Yankee	Vice President - Operat
J. Goeby	Washington Reporting Service	Reporter
[illegible]	[illegible]	Nuclear [illegible]
T.H. Landers	New York Power Authority	Sr. Nuc Operations Engine
J. A. GRAY, JR.	NEW YORK POWER AUTHORITY	DIRECTOR NUCLEAR LICENSING - B
L. S. GIFFORD	GE (BETHESDA MD)	REG PGMS MGR
S. D. Floyd	Carolina Power & Light Company	Principal Licensing Engi
Kevin Holtzclaw	General Electric	Principal Engineer

N. W. EDWARDS	NUTECH	SR. VICE PRESIDENT
TERRY PICKENS	NORTHERN STATES POWER	CHAIRMAN - BWR OWNERS GR
Michael J May	TVA - BFN	Mgr Lic BFN
RICHARD BACHMANN	NRC/OGC	ATTORNEY
Stephanie Markiewicz	Massachusetts Governors Washington Office	Policy Analyst
KAZIMIERA S CAMPE	NRR/DBL/PSB	SECTION LEADER
Jacqueline Cooter	VPIRG	
Ellyn R. Weiss	Harmon Weiss Union of Concerned Scientists	Nuclear Project Coordi... Atty
Stephanie Murphy	Nucl. Information + Resource Service	research
Eve Fotopoulos	SERCH Licensing, Bechtel	engineer
SANG B. KIM	NRR/DBL/PSB	STAFF
Franklin Coffman	N.R.C.	S/L
Allen Notafrancesco	NRR/DBL/PSB	Mech. Eng.
C. L. REID	BECHTEL	PROJECT ENGINEER
O.M. SCOTT	SOUTHERN CO. SERVICES	ENGINEER
JACK KUDRICK	NRR/DBL/PSB	SECTION LEADER
Phil Paul		
Gerald Tenant		

VERMONT YANKEE

CONTAINMENT SAFETY STUDY

I INTRODUCTION

- PERSPECTIVE
- SCOPE
- RESULTS SUMMARY

II MARK I CONTAINMENT DESIGN REVIEW

III SEVERE ACCIDENT ANALYSIS AND QUANTIFICATION

- BWR DOMINANT SEQUENCES
- CONTAINMENT CONDITIONAL FAILURE PROBABILITY
  - o METHODOLOGY
  - o ACCIDENT SEQUENCE QUANTIFICATION
  - o INITIATING EVENT FREQUENCIES
  - o CORE MELT FREQUENCIES
  - o CONTAINMENT EVENT TREES
  - o CONTAINMENT PERFORMANCE RESULTS

IV MARK I CONTAINMENT POLICY ISSUES

- o HYDROGEN CONTROL
- o DRYWELL SPRAY CAPABILITY
- o CONTAINMENT VENTING
- o CORE DEBRIS CONTROL
- o SEVERE ACCIDENT TRAINING PROCEDURE

V DECISION ANALYSIS FOR RECOMMENDATIONS

- o RECOMMENDATIONS AND INSIGHTS

PERSPECTIVE

- INITIATED - TO RESPOND TO QUOTED STATEMENT THAT MARK I CONTAINMENTS HAVE A 90% CONTAINMENT CONDITIONAL FAILURE PROBABILITY w/a ~~LOCA~~ Severe Accident

- AGREED UPON BY VERMONT YANKEE, THE NRC AND THE GOVERNOR OF THE STATE OF VERMONT ON JUNE 30, 1986

- PROGRESS
  - INITIATED JULY 1, 1986
  - INTERIM STATUS MEETING IN BETHESDA ON AUGUST 6, 1986
  - FINAL REPORT COMPLETE AND TRANSMITTED TO NRC AND STATE OF VERMONT ON SEPTEMBER 2, 1986

STUDY SCOPE

- MARK I CONTAINMENT DESIGN REVIEW

- QUANTIFY A VERMONT YANKEE SPECIFIC CONTAINMENT CONDITIONAL FAILURE PROBABILITY GIVEN A SCENARIO LEADING TO CORE MELT

- ADDRESS MARK I CONTAINMENT POLICY ISSUES
  - HYDROGEN CONTROL
  - DRYWELL SPRAY CAPABILITY
  - CONTAINMENT VENTING
  - CORE DEBRIS CONTROL
  - SEVERE ACCIDENT TRAINING AND PROCEDURES

STUDY RESULTS

- VERMONT YANKEE CONDITIONAL CONTAINMENT FAILURE PROBABILITY OF 7%

- CONTAINMENT CAPABILITY IMPROVEMENTS IDENTIFIED
  - PROCEDURE CHANGES/ENHANCEMENTS
  - ISSUES FOR FURTHER EVALUATION
  - INSIGHTS BASED ON SEVERE ACCIDENT SEQUENCES

MARK I CONTAINMENT DESIGN REVIEW

- SCOPE: COMPARE VERMONT YANKEE WITH THE MARK I REFERENCE PLANT TO IDENTIFY DESIGN AND OPERATIONAL FEATURE DIFFERENCES SIGNIFICANT TO CONTAINMENT PERFORMANCE DURING SEVERE ACCIDENT SEQUENCES

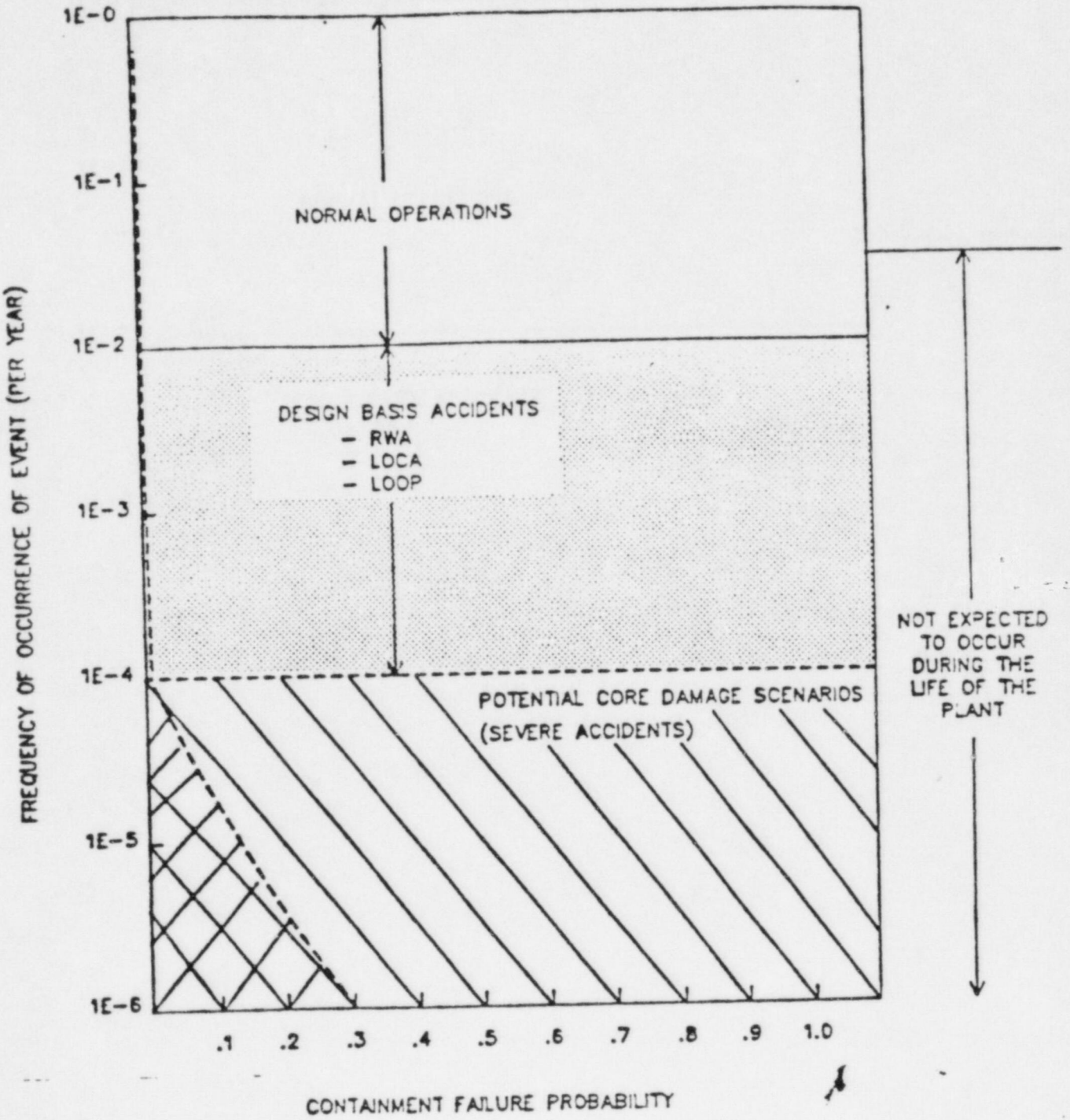
- RESULTS: 4 SIGNIFICANT DESIGN FEATURES WERE IDENTIFIED

  - RATIO OF CONTAINMENT SIZE TO REACTOR POWER sm plt large ratio 2x mo
  - ELECTRICAL PLANT DIVERSITY
    - DIESEL GENERATORS
    - TIE LINE TO ADJACENT HYDROELECTRIC STATIONS
  - CONNECTION FROM DIESEL FIRE PUMP TO LPCI/ CONTAINMENT SPRAY

CONTAINMENT FAILURE PROBABILITY EVALUATION

- BACKGROUND
  - WASH - 1400 (90%)
  - DEFINITION OF CCFP Cont. Cond. Fail. Prob.
- VERMONT YANKEE ANALYSES
  - APPROACH
  - ACCIDENT SEQUENCE IDENTIFICATION
  - ACCIDENT SEQUENCE QUANTIFICATION
  - CONTAINMENT EVENT TREE DEVELOPMENT
  - CET QUANTIFICATION
- SUMMARY OF RESULTS

Graphic Display of Containment Performance



DEFINITION OF "90%"

WASH-1400 ASSUMPTIONS

- ALL SEQUENCES (MELT/NON-MELT) ARE ASSOCIATED WITH CONTAINMENT FAILURE
- CORE MELT RELEASE CATEGORIES
  1. STEAM EXPLOSION
  2. OVERPRESSURE FAILURE DIRECT TO ATMOSPHERE
  3. OVERPRESSURE FAILURE THROUGH REACTOR BUILDING
  4. ISOLATION FAILURE
- RELEASE CATEGORIES 2 & 3 DOMINATE RISK
- SEQUENCES WHICH CONTRIBUTE COMPRISE 90%

  ATWS

  LOSS OF RESIDUAL CORE HEAT REMOVAL

CURRENT STUDY

- DIFFERENT DOMINANT SEQUENCES IDENTIFIED
- KNOWLEDGE OF CONTAINMENT FAILURE PHENOMENA AND CRITERIA HAS EVOLVED

CONTAINMENT CONDITIONAL FAILURE PROBABILITY

- CALCULATE THE CONTRIBUTION OF EACH ACCIDENT CLASS TO TOTAL LIKELIHOOD OF CORE MELT
- DETERMINE THE CONDITIONAL FAILURE PROBABILITY OF CONTAINMENT FOR EACH ACCIDENT CLASS
- CALCULATE THE WEIGHTED AVERAGE OF PRIMARY CONTAINMENT FAILURE PROBABILITY

CONDITIONAL CONTAINMENT FAILURE PROBABILITY

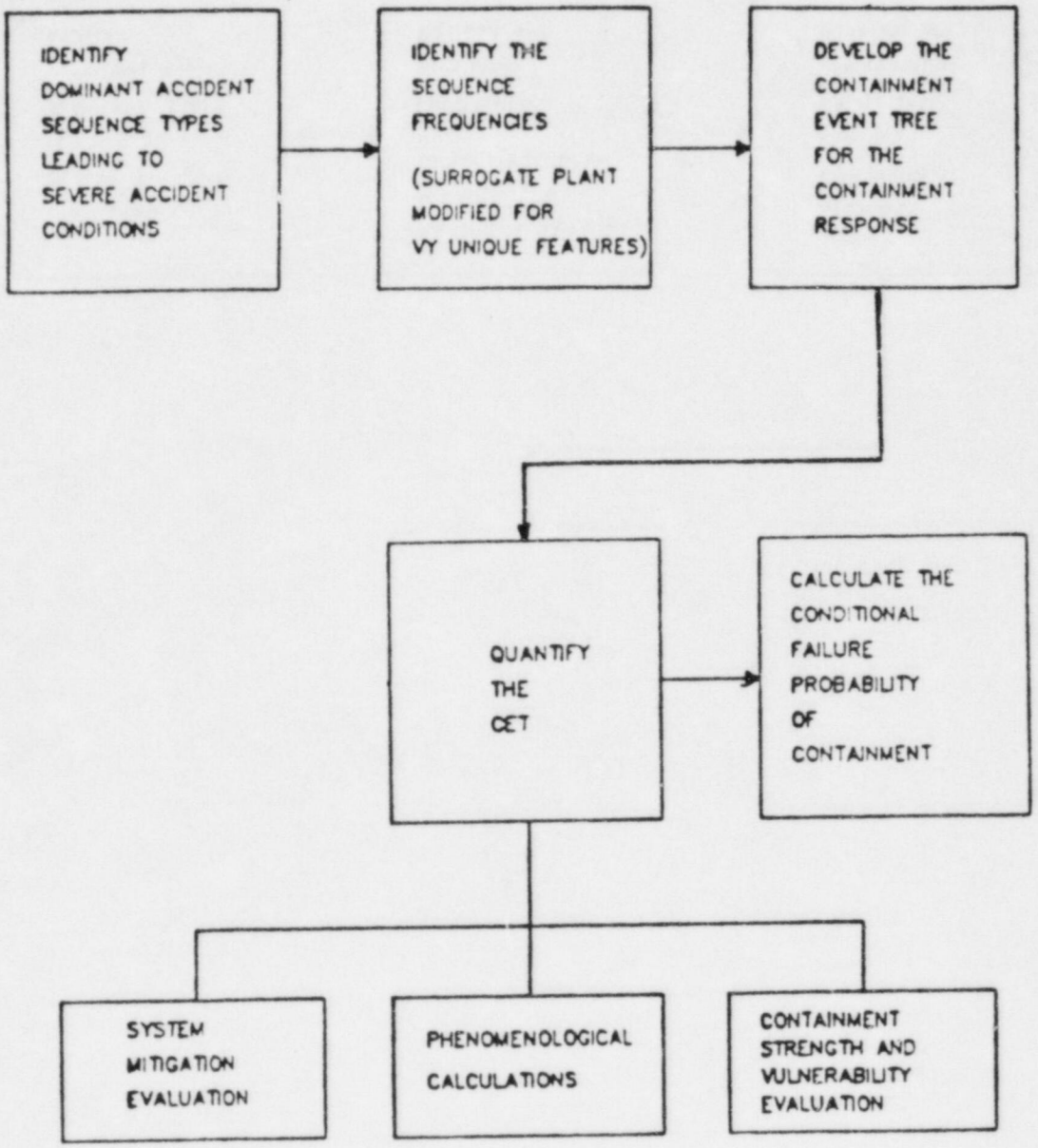
QUANTIFICATION

DOMINANT ACCIDENT SEQUENCE	FRACTIONAL CONTRIBUTION (CORE MELT)	CONTAINMENT FAILURE PROBABILITY	CONTRIBUTION TO WEIGHTED AVERAGE
LOSS OF COOLANT MAKEUP	A	A	A x A
LOSS OF CONTAINMENT HEAT REMOVAL	B	B	B x B
LOCA	C	C	C x C
ATWS	D	D	D x D
	1.0		Σ ( ) = CFFP

WHERE CCFP = CONDITIONAL CONTAINMENT FAILURE PROBABILITY

Internal Events On
excluded Seismic
Sabotage
for example.
(Phil Paull brought this up

CALCULATION PROCESS FLOW CHART



BWR DOMINANT ACCIDENT SEQUENCES

CLASS I LOSS OF COOLANT MAKEUP

IA HIGH PRESSURE

IB STATION BLACKOUT

IC ATWS

ID LOW PRESSURE

IE LOSS OF DC

CLASS II LOSS OF CONTAINMENT HEAT REMOVAL

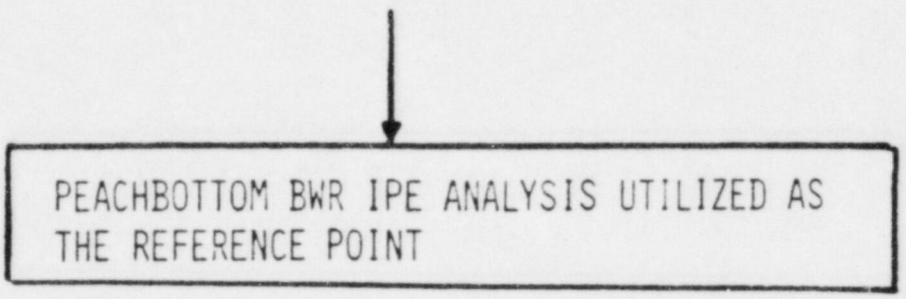
CLASS III LOCA

CLASS IV ATWS

DOMINANT ACCIDENT SEQUENCE QUANTIFICATION

ALTERNATIVES

- PLANT - SPECIFIC PRA
- VY INDIVIDUAL PLANT EVALUATION
- MODIFICATION OF REFERENCE BWR MARK I PRA TO ACCOUNT FOR KEY DIFFERENCES



INITIATING EVENT FREQUENCIES

Category	Types of Events	Number of Vermont Yankee Events in Data Base	Best Estimate Frequency (yr^{-1})
1	Transients Resulting in Reactor Scram with Bypass Potentially Available	20	1.62
2	Events Bounded by Loss of Feedwater	2	0.16
3	MSIV Closure	4	0.32
4	Events with Bypass Not Available	8	0.65
5	Inadvertent Opening of One or More SRVs	0	0.062
6	Loss of Off-site Power	1	0.071

DERIVATION OF VERMONT YANKEE SEVERE ACCIDENT SEQUENCE FREQUENCIES

SURROGATE PLANT DOMINANT SEQUENCES		VERMONT YANKEE UNIQUE FEATURES		VERMONT YANKEE DOMINANT ACCIDENT SEQUENCES AND FREQUENCIES	
DESIGNATOR	FREQUENCY (per Rx yr)	FEATURE	DELTA	DESIGNATOR	FREQUENCY (per RX yr)
Class IB				Class IB	
T_EQUV PHASE I 0-2 hrs	1E-6	-Initiator (x.044/.05)	X.35	T_EQUV PHASE I	3.5E-7
		-2 D/G (x2)			
		-Hydro Plant (:5 Phase I, 6.6 Phase II, 10 Phase III, IV)			
		-No. of Switchyards			
		-Underground transmission			
T_EQUV PHASE II 2-4	2E-6	-No Event Specific Station Blackout Procedures (Approx. same Phase II) Phase III, IV (X3)	X.52	T_EQUV PHASE II	1.0E-6
T_EQUV PHASE III 4-8	4E-6	-Battery Life (Approx. same) Phase II, III, IV	X1.1	T_EQUV PHASE III	4.4E-6
T_EQUV PHASE IV >8 hrs	1.6E-6	HPCI Potential Larger DC Power Reqm't (X2) Phase II, III	X.26	T_EQUV PHASE IV	4.2E-7
				TOTAL Class IB =	6.2E-6

SUMMARY OF CORE MELT FREQUENCY TYPES AT VERMONT YANKEE

Class	Description	Frequency (yr^{-1})	Fraction
IA	Loss of Makeup; RPV at high pressure	1.33E-5	.43
IB	Loss of Makeup: Station Blackout	6.2E-6	.20
IC	Loss of Makeup: ATWS	2.6E-6	.08
ID	Loss of Makeup: RPV at low pressure	3.9E-6	.12
IE	Loss of Makeup: Loss of DC buses	2.4E-7	.01
II	Loss of Containment Heat Removal	2.1E-6	.07
III	LOCAs	7.3E-7	.02
IV	ATWS	2.2E-6	.07
V	Interfacing LOCA	Negligible	Negligible
	TOTAL	3.1E-5	100%

CONTAINMENT EVENT TREE DEVELOPMENT

- CORE MELT ACCIDENT CLASS (ENTRY STATE)
- CONTAINMENT MITIGATING SYSTEM RESPONSE
- CONTAINMENT RESPONSE-FAILURE PATH & SIZE
- REACTOR/AUXILIARY BUILDING RESPONSE

CONTAINMENT EVENT TREE END STATES

TIMING

	T (BEYOND CORE MELT)
EARLY (E)	T < 10 HOURS
INTERMEDIATE (I)	10 < T < 24 HOURS
LATE (L)	T > 24 HOURS
NO CONTAINMENT FAILURE (NCF)	INTACT

RELEASE QUANTITIES

	I (EQUIVALENT IODINE) PLUS NOBLE GASES
HIGH (H)	I > 10%
MEDIUM (M)	0.1 < I < 10%
LOW (L)	I < 0.1%

FUNCTIONAL EVENTS: CONTAINMENT EVENT TREE

- CONTAINMENT ISOLATION AND INITIAL INTEGRITY
- ACTIVE MITIGATION: DRYWELL T/P CONTROL
- PASSIVE MITIGATION: CONTAINMENT OF DEBRIS
- CONTAINMENT HEAT REMOVAL
  - RHR
  - VENTING
- CONTAINMENT BREACH SIZE
  - LEAKAGE
  - OVERPRESSURE FAILURES
- CONTAINMENT BREACH LOCATION
  - DRYWELL
  - WETWELL AIRSPACE
- REACTOR BUILDING EFFECTIVENESS

CONTAINMENT EVENT TREES

TOP EVENTS

ACCIDENT CLASS ENTRY STATE	CONTAINMENT ISOLATION & INITIAL INTEGRITY	ACTIVE MITIGATION TEMPERATURE CONTROL (INJECTION)	PASSIVE MITIGATION	ACTIVE MITIGATION		CONTAINMENT INTEGRITY				REACTOR BUILDING
				CONTAINMENT HEAT REMOVAL	CONTAINMENT PRESSURE CONTROL (VENTING)	NO LEAKAGE	NO OVER-PRESSURE FAILURE	DRYWELL FAILURE	WETWELL AIRSPACE	
QM	CI	T	P	W	V	D	G	H	J	R
CLASS I + II										

ACCIDENT CLASS ENTRY STATE	CONTAINMENT INTEGRITY					CONTINUED COOLANT MAKEUP TO RPV	DRYWELL CONTAINMENT ISOLATION & INITIAL INTEGRITY	ACTIVE MITIGATION TEMP. CONTROL (INJECTION)	PASSIVE MITIGATION	REACTOR BUILDING
	CONTAINMENT PRESSURE CONTROL (VENTING)	NO LEAKAGE	NO OVER-PRESSURE FAILURE	DRYWELL FAILURE	WETWELL AIRSPACE					
QM	V	(D)	(G)	(H)	(J)	C	CI	T	P	R
II IV										

CONTAINMENT EVENT TREE QUANTIFICATION BASES

- PAST PRA EXPERIENCE
- DETERMINISTIC EVALUATIONS (VY)
  - MARCH/RMA
  - MAAP *Modular Accident Analysis Program*
  - HAND CALCULATIONS
- IDCOR TECHNICAL REPORT INFORMATION
- ENGINEERING JUDGMENT

SUMMARY OF RELEASE FREQUENCY OF EACH RELEASE TYPE
FOR EACH ACCIDENT CLASS: BASE CASE
"BEST ESTIMATE" CASE

Accident Class Designator	Core Melt Frequency (yr^{-1})	Release Frequency (yr^{-1}) of Release Type					
		EH	EM	EL	LM	LL	NCF
IA	1.33E-05	7.38E-09	2.58E-08	1.25E-07	6.89E-08	1.81E-07	1.29E-05
IB	6.20E-06	7.07E-09	1.11E-07	5.07E-07	3.56E-07	9.03E-07	4.31E-06
IC	2.60E-06	5.27E-07	2.83E-07	1.61E-08	9.31E-09	2.44E-08	1.74E-06
ID	3.90E-06	3.32E-09	4.60E-08	2.86E-08	5.44E-07	1.54E-06	1.74E-06
II	2.10E-06	4.73E-10	5.39E-07	1.56E-06	-	-	-
III	7.30E-07	5.53E-10	9.06E-10	6.56E-09	4.55E-08	1.28E-07	5.49E-07
IV	2.20E-06	5.44E-09	6.55E-07	1.54E-06	-	-	-
Sum	3.10E-05	5.51E-07	1.66E-06	3.78E-06	1.02E-06	2.78E-06	2.12E-05
% of Total Core Melt Frequency	100%		7.1%	12%	3.3%	9.0%	68%

CONTAINMENT CONDITIONAL FAILURE PROBABILITY (%)

BASE AND SENSITIVITY CASE RESULTS

	CATAGORIZATION				
	EH/M	EL	LM	LL	NCF
BASE	7	12	4	9	68
S1	10	12	3	9	66
S2	3	9	4	10	74
S3	1	3	6	16	74

WHERE:

S1 - EARLY CONTAINMENT FAILURE INCREASE FOR BOTH HIGH AND LOW PRESSURE MELT DOWNS

S2 - INCREASED OPERATOR ACTION CREDIT FOR ATWS EVENTS

S3 - S2 PLUS CLASSIFICATION OF TW CLASS AS LATE VERSUS EARLY RELEASE

MARK I CONTAINMENT POLICY ISSUES

- HYDROGEN CONTROL
- DRYWELL SPRAY CAPABILITY
- CONTAINMENT VENTING
- CORE DEBRIS CONTROL
- SEVERE ACCIDENT TRAINING AND PROCEDURES

HYDROGEN CONTROL

- OBJECTIVE: PREVENT HYDROGEN COMBUSTION FOLLOWING SEVERE ACCIDENTS

- VY STATUS
  - CONTAINMENT INERTED
  - DE-INERT TIME CONTROLLED BY TECHNICAL SPECIFICATIONS
  - OPERATING HISTORY INDICATES DE-INERT TIME IS 1.1% OF TIME PLANT IS IN RUN MODE
  - VY RECENTLY CONVERTED TO N_2 CAD

- EVALUATION: CURRENT VY PERFORMANCE AND DESIGN MEETS OBJECTIVE

- RECOMMENDATIONS: NONE

DRYWELL SPRAY CAPABILITY

- OBJECTIVE: PROVIDE AC INDEPENDENT CAPABILITY TO SPRAY DRYWELL TO LOWER PRESSURE AND TEMPERATURE

- VY STATUS:
  - NORMAL CAPABILITY FROM RHR, BACKED UP BY SERVICE WATER (DEPENDENT ON AC POWER)
  - DIESEL DRIVEN FIRE PUMP CAPABLE OF BEING ALIGNED TO LPCI/ CONTAINMENT SPRAY (AC INDEPENDENT)
  - WATER SUPPLY FROM COOLING TOWER DEEP BASIN (AC DEPENDENT)

- EVALUATION: CURRENT CAPABILITY ACCEPTABLE, ENHANCEMENTS POSSIBLE TO INCREASE RELIABILITY OF SUPPLY FROM DIESEL FIRE PUMP

- RECOMMENDATIONS:
  - ENHANCE THE OPERATION OF THE DIESEL FIRE PUMP TO DRYWELL SPRAY FLOW PATH
  - IMPROVE ABILITY TO RESTORE AC POWER

CONTAINMENT VENTING

- OBJECTIVE: PREVENT UNCONTROLLED LOSS OF CONTAINMENT INTEGRITY

- VY STATUS:
  - ONLY VERY SMALL VENTING PATHWAY PROCEDURALIZED
  - STUDY IDENTIFIES 6 VENTING PATHWAYS
  - RISKS OF VENTING DISCUSSED

- EVALUATION: CAPABILITY FOR WETWELL VENTING, TO ACHIEVE FISSION PRODUCT SCRUBBING, SHOULD BE ENHANCED

- RECOMMENDATIONS:
  - ENGINEERING EVALUATION OF UPGRADING WETWELL VENTING TO BE INITIATED
  - VENTING GUIDELINE DOCUMENT PREPARED OUTLINING 32 POTENTIAL VENT PATHS, REQUIREMENTS FOR USE, ACCESS, ETC.

CORE DEBRIS CONTROL

- OBJECTIVE: PRECLUDE CONTAINMENT FAILURE DUE TO CORIUM/DRYWELL SHELL CONTACT

- VY STATUS:
  - SMALL CORE VOLUME IN LARGE DRYWELL PROVIDES ASSURANCE CORIUM WILL NOT MIGRATE TO DRYWELL SHELL
  - DRYWELL SPRAYS WILL "REFREEZE" CORIUM

- EVALUATION: CURRENT VY CONFIGURATION MEETS OBJECTIVE

- RECOMMENDATIONS: NONE

SEVEERE ACCIDENT TRAINING AND PROCEDURES

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

- VY STATUS:
  - VY HAS IMPLEMENTED EOP's BASED ON REV. 3 OF THE EPG's

- EVALUATION: VY MEETS CURRENT GUIDANCE

- RECOMMENDATION:
  - IMPLEMENT REV. 4 OF EPG's WHEN AVAILABLE
  - CONSIDER REVISION TO EXISTING EOP's FOR:
    - o STATION BLACKOUT
    - o CONTAINMENT VENTING
    - o DRYWELL SPRAY USING DIESEL FIRE PUMP
    - o GUIDANCE TO SUPPLY CST VIA FIRE SYSTEM
    - o SAMPLE SUPPRESSION POOL BEFORE TRANSFER TO CST
    - o PROCEDURALIZE H_2/O_2 CONTROL
    - o COMPLETE DEVELOPMENT OF VY SPECIFIC LEVEL/POWER CONTROL

DECISION ANALYSIS FOR RECOMMENDATIONS

- TWO PART TEST
  - WHICH SEVERE ACCIDENTS WARRANT CONSIDERATION?
  - WHAT IS ACCEPTABLE CONTAINMENT PERFORMANCE?

- SEVERE ACCIDENT SEQUENCES
  - MOST CRITICAL TO CONTAINMENT PERFORMANCE ARE:
    STATION BLACKOUT
    ATWS WITH MSIV CLOSURE

- ACCEPTABLE CONTAINMENT PERFORMANCE
  - CONTAINMENT SHOULD REMAIN INTACT, WITHOUT EXCESSIVE LEAKAGE FOR 24 HOURS FOLLOWING EVENT INITIATION

- RECOMMENDATIONS REPRESENT THE CONCENSUS OF THE STUDY TEAM REGARDING PROCEDURAL AND DESIGN ENHANCEMENTS TO IMPROVE CONTAINMENT PERFORMANCE.
  - MAXIMIZE USE OF EXISTING PLANT FEATURES
  - AUGMENT EXISTING DESIGN WHERE CLEAR BENEFIT IS APPARENT

RECOMMENDATIONS

- PROCEDURE ADDITIONS/ENHANCEMENTS
- ISSUES FOR FURTHER EVALUATION
- INSIGHTS BASED ON SEVERE ACCIDENT SEQUENCES

PROCEDURE ADDITIONS/ENHANCEMENTS

- PROVIDE PROCEDURAL GUIDANCE FOR RESTORATION OF AC POWER FOLLOWING A STATION BLACKOUT

- PROVIDE A REPAIR PROCEDURE OUTLINING THE RESTORATION OF THE VERNON TIE LINE IF DAMAGE SHOULD OCCUR

- PROVIDE PROCEDURAL GUIDANCE FOR OPTIMIZATION OF DC POWER SOURCES FOLLOWING A STATION BLACKOUT

- PROVIDE PROCEDURAL GUIDANCE FOR USE OF THE SRV's FOR MANUAL DEPRESSURIZATION FOLLOWING A STATION BLACKOUT

- PROVIDE PROCEDURAL GUIDANCE TO ALIGN THE DIESEL FIRE PUMP FOLLOWING STATION BLACKOUT

- PROVIDE LEVEL/POWER GUIDANCE FOR ATWS

ISSUES FOR FURTHER EVALUATION

- UPGRADE EOP's TO REV. 4 OF EPG's
- UPGRADE SELECTED VALVE OPERATORS IN FLOW PATH FROM DIESEL FIRE PUMP TO LPCI/CONTAINMENT SPRAY
- ENHANCE RELIABILITY AND CAPABILITY OF CONTAINMENT WETWELL VENT

INSIGHTS BASED ON SEVERE ACCIDENT SEQUENCES

- PROVIDE ADDITIONAL EVALUATION, GUIDANCE AND OPERATOR TRAINING ON RESPONSE OF THE RHR PUMPS TO HIGH SUPPRESSION POOL TEMPERATURES

- IMPROVE RELIABILITY OF THE NITROGEN SUPPLY TO THE ADS VALVES

- EVALUATE DESIGN AND POST-ACCIDENT RELIABILITY OF THE SERVICE WATER SYSTEM

- EVALUATE STANDBY LIQUID CONTROL SYSTEM DESIGN AND RELIABILITY