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ACRS Subcommittee Meeting Summary/Minutes
For Severe (Class 9) Accidents
December 19, 1986
Washington, DC

Purpose

The ACRS Subcommittee on Severe (Class 9) Accidents met on December 19, 1986 in Washington, DC. The purpose of this meeting was to discuss the Staff's (NRR) proposed generic letter for Individual Plant Examinations (IPEs) as part of the Implementation Plan for the Severe Accident Policy Statement in regard to the evaluation of existing nuclear power plants. Included in the discussion were: (1) Guidelines and Criteria for Five Reference Plants, and (2) IDCOR-Individual Plant Examination Methodology (IPEM). Copies of the agenda and selected slides from the presentations are attached. The meeting began at 8:30 a.m. and adjourned at 4:00 p.m., and was held entirely in open session. The principal attendees were as follows:

Attendees

ACRS

W. Kerr, Chairman
M. Carbon, Member
C. Mark, Member
P. Shewmon, Member
C. Wylie, Member
I. Catton, Consultant
M. Corradini, Consultant
P. Davis, Consultant
J. Lee, Consultant
D. Houston, Staff

NRC/NRR

T. Speis
Z. Rosztoczy
R. Landry
F. Coffman
F. Eltawila

BNL (NRR Consultant)

R. Bari
K. Perkins
W. Lucas
R. Fitzpatrick

IDCOR

J. Carter (ITC)
R. Henry (FAI)
M. Kenton (FAI)
J. Gabor (FAI)
K. Vavrek (W)
R. Brown (Delian)

DESIGNATED ORIGINAL

Certified By ____________

Discussion

In his opening comments, W. Kerr noted that the subject of the meeting was considered important and difficult. He indicated that perfection in the methodology would most likely not be achieved the first time around and that the Subcommittee had a considerable responsibility in the review of this matter.

T. Speis (NRR) introduced the topics for discussion with a brief overview of the NRR Implementation Plan for the Severe Accident Policy Statement. In his discussion of accomplishments to date, he indicated that 16 of the 18 NRC/IDCOR technical issues of concern had been resolved. He briefly reviewed the documentation status for the IPEs and the tentative schedule for the generic letter and the completion of individual plant analysis.

R. Landry (NRR) presented the background, structure and content of the proposed IPE generic letter. He emphasized that the letter would instruct utilities to consider only internal initiators. He discussed the scope of the examination and listed five acceptable methods for application in performing an IPE. These are as follows:

(1) IDCOR-IPEM as approved,
(2) Level II or III PRA with update,
(3) Level I PRA with IPEM source term,
(4) Simplified or Smart (Phase I) PRA as approved, or
(5) Other systematic evaluation method as approved.

The NRR review of the acceptable methods is tentatively to be completed by January 8, 1987. He discussed the tentative schedule for completion of the IPEs and the regional/quarterly interfaces with the utilities while performing the IPEs.

Z. Rosztoczy (NRR) discussed, in detail, the scope of the IPE. He listed four expected accomplishments:

(1) identification and assessment of potential severe accidents,
(2) plant improvements to prevent severe accidents,
(3) improved containment performance, and
(4) development and implementation of an accident management program.

He indicated that extremely unlikely events need not be considered, only those giving a core damage frequency (CDF) greater than $2x10^{-6}$ per year. Also, sequences with a contribution to CDF of greater than 5%, irrespective of the frequency limit, should be considered. He used the current analysis for the BWR Mark I reference plant in discussing how one would apply plant specific values in the performance of the IPE. For accident management, he discussed three areas: (1) approach and organization, (2) training and procedures, and (3) instrumentation and equipment. He indicated that existing Emergency Operating Procedures would be reviewed and revised as necessary for accident management.

The severe accident guidelines and criteria for the five reference plants were presented by three BNL personnel: R. Bari - outline and overview, R. Fitzpatrick - guidelines and criteria for RCS integrity, RCS heat removal, RPV depressurization, ATWS response, station blackout response and support system failures, and K. Perkins - guidelines and criteria for containment integrity and control of hydrogen burning. BNL has produced a series of reports for the five reference plants. The reports for BWR Mark Is and IIIs and PWR ice condenser containments had been provided to the Subcommittee. The report for BWR Mark IIs was now available and the one for PWR large dry containments would soon be available. The guidelines highlighted essential functions and the criteria were based on system availability, operating/emergency procedures or maintenance surveillances.

The IDCOR-IPEM presentation was given in three parts: (1) BWR-IPEM by R. Brown (Delian), (2) PWR-IPEM by K. Vavrek (W) and (3) Source Term Methodology by R. Henry (FAI). J. Carter (ITC) gave the introduction to the IDCOR activities. He indicated that the methodology is a screening methodology, not a PRA technique but based on PRA techniques. IDCOR believes existing plants are safe enough and is searching only for those cases of unusually high core damage frequency or unusually poor containment performance. The IDCOR presentation was planned to inform the Subcommittee of the revisions to the methodology that had been made since the previous presentation on this matter on September 24, 1986. He stated that IDCOR has some serious disagreements with the NRC/NRR efforts and feels that NRR is mixing other agenda or programs with the guidance in the Severe Accident Policy Statement.

R. Brown (Delian) presented an overview of the BWR-IPEM, some results obtained on various BWR reference plants and some responses to NRC/EPRI comments on the BWR-IPEM. Four plants were identified for the verification phase: Peach Bottom (Mark I), Susquehanna (Mark II), Shoreham (Mark II) and Grand Gulf (Mark III). The resources estimated to perform an individual plant analysis ranged from 24 to 48 man-months. In most cases, the core damage frequencies were higher by using the BWR-IPEM as compared to other PRA (ASEP) results. The dominant accident sequence differed in the two studies: station blackout and transient initiated sequences about equal for IDCOR, station blackout the dominant one for ASEP.

K. Vavrek (W) presented an overview of the PWR-IPEM and indicated that revisions to the methodology were only in the form of an expansion of the earlier model. He discussed the plant walk-through checklist and the control room/man-machine interfaces system interaction checklist. He mentioned that the IPEM document would be expanded in the following manner: (1) Section/Chapter 3.0 would be the User's Guide, (2) Appendix

C would address system interaction, and (3) Appendix (X) would address internal flooding.

R. Henry (FAI) reviewed the IPE source term methodology. As with the PWR-IPEM, significant revisions to the source term methodology were not apparent. He discussed the severe accident sequences considered for the Zion, Indian Point 2, Peach Bottom and Limerick analyses.

F. Coffman (NRR) presented the NRC/NRR comments on the IDCOR-IPEM. A major concern to NRR was that the IDCOR analysis failed to identify any vulnerabilities in the reference plants. He indicated that the IDCOR position was derived from a 1983 draft of the safety goals while the NRC position was based on the 1986 final version of the safety goals. He discussed the numerical differences in the two positions and indicated that IDCOR was less conservative than NRC. He further discussed the two positions in terms of consistency with generic resolutions, specifically in details regarding the use of the MAAP code and consideration of uncertainties. The IDCOR results appear to be insensitive to uncertainties, thus IDCOR feels that uncertainties can be ignored. He indicated that NRC was concerned about non-uniform results obtained with the IDCOR methodology when applied to similar plants by the same personnel. There were unexpected differences. NRC feels that the IDCOR-IPEM is not yet complete and in its present form, allows too many options. A schedule for the NRC evaluation of the IPEM was discussed. The present schedule calls for a final evaluation report on January 30, 1987, contingent upon receipt of three or four key IDCOR documents by early January: revised source term methodology, PWR-IPEM, Sequoyah IPE and Grand Gulf IPE. He concluded with a listing of documents that formed the basis for the IDCOR-IPEM review. At least three of these are forthcoming and have not been provided to the Subcommittee for review. Rosztoczy (NRR) requested another meeting on this matter later in January 1987 and W. Kerr indicated that while the request would be considered, it appeared to be premature based on the schedules for IDCOR documentation submittal and NRC review and approval.

During the meeting, Subcommittee members and consultants expressed concerns and opinions as follows:

(1) W. Kerr questioned the Staff about the application of the safety goal and their assignment of numerical values to it when the Commission had deliberately chosen not to put in specific values. He also expressed concerns about the selection process based on an aggregate probability of extremely unlikely events being less than E-6. If one summed a large number of sequences in the E-8 range, the limit (E-6) would be exceeded and all of the low probability sequences would have to be considered. He asked about the qualifications required of the staff that will perform the IPE study.

(2) P. Shewmon also questioned the selection process based on the aggregate probability. In regard to the evaluation of only internal initiators at this time, he expressed concerns about the resources required to perform the first IPE, to be followed in the near term with another IPE which considers external events as well. He thinks that direct containment heating (DCH) is incredible, and that the review of DCH needs to be performed by someone with an understanding of heat flow and failure mechanisms.

(3) M. Carbon questioned the goal of the proposed generic letter and indicated that for low probability sequences, he felt that the letter went beyond the safety goal.

(4) C. Mark asked if the IPE would provide assurance that the plant was built properly. He also questioned how operator performance was evaluated, e.g., a review of operating procedures, an interview of the operators or other.

(5) I. Catton asked how Bernero's proposed Mark I containment requirements are related to the Implementation Plan. He also stated that systems parameter display should be a line item on the PWR human factors checklist.

(6) M. Corradini questioned the Staff about comparison of the IPE results. There seems to be a disconnect between the BNL guidelines and criteria and the IPE results.

(7) P. Davis expressed a concern related to on-going programs at NRC, e.g., resolution of station blackout, decay heat removal and ATWS, and a possible conflict between new requirements from these programs and the guidelines and criteria. He also indicated that the IDCOR-IPEMs do not seem to be directed toward giving CDF and containment failure probability values in a manner desired by NRC. He doubts that NRC will get an approved IPEM from IDCOR. He predicts that the IPEM will need to be modified by NRC for acceptability.

(8) J. Lee questioned the Staff's reason for not providing a list of instrumentation and operating procedures for severe accident management. He asked if there were hardware modifications that could be made to relieve operator stress. Also, he asked if BNL had compared their guidelines and criteria with the IDCOR IPE for any reference plant. In regard to the systems interaction checklist for PWRs, he asked if or how the IPEM was set up to use the six decision factors.

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<u>NOTE</u>: Additional meeting details can be obtained from a transcript of this meeting available in the NRC Public Document Room, 1717 H Street, N.W., Washington, D.C., or can be purchased from ACE-Federal Reporters, 444 North Capitol Street, Washington, DC 20001, (202) 347-3700.

REVISED: 12/11/86

ACRS Severe (Class 9) Accidents Subcommittee
December 19, 1986
Washington, DC

Individual Plant Examination Generic Letter

A.	Subcommittee Chairman's Remarks	W. Kerr	8:30am
B.	Introductory Remarks	T. Speis, NRR	8:40am
C.	Generic Letter		
	1. Structure and Content	R. Landry, NRR	8:55am
D.	Scope of the Individual Plant Examination Performed by Licensees	Z. Rosztoczy, NRR	9:35am
	*** Break ***		10:35 - 10:45am
E.	Guidelines and Criteria ° BWR Mark I ° BWR Mark II ° BWR Mark III ° PWR Large Dry ° PWR Ice Condenser	T. Pratt, BNL	10:45am
	*** Lunch ***		12:30 - 1:30pm
F.	IDCOR-IPEM Revisions and Responses to ACRS Comments	J. Carter, IDCOR	1:30pm
	° BWR Methodology	E. Burns, Delian	
	° PWR Methodology	M. Hitchler, W	
	*** Break ***		3:00 - 3:10pm
	° Source Term Methodology	R. Henry, FAI	
G.	NRC Comments on IDCOR-IPEM	F. Coffman, NRR	4:15pm
H.	Concluding Remarks	W. Kerr	4:30pm
	*** Adjourn ***		4:45pm

NRR STAFF PRESENTATION TO THE
ACRS

SUBJECT: INTRODUCTORY REMARKS

DATE: DECEMBER 19, 1986

PRESENTER: THEMIS P. SPEIS

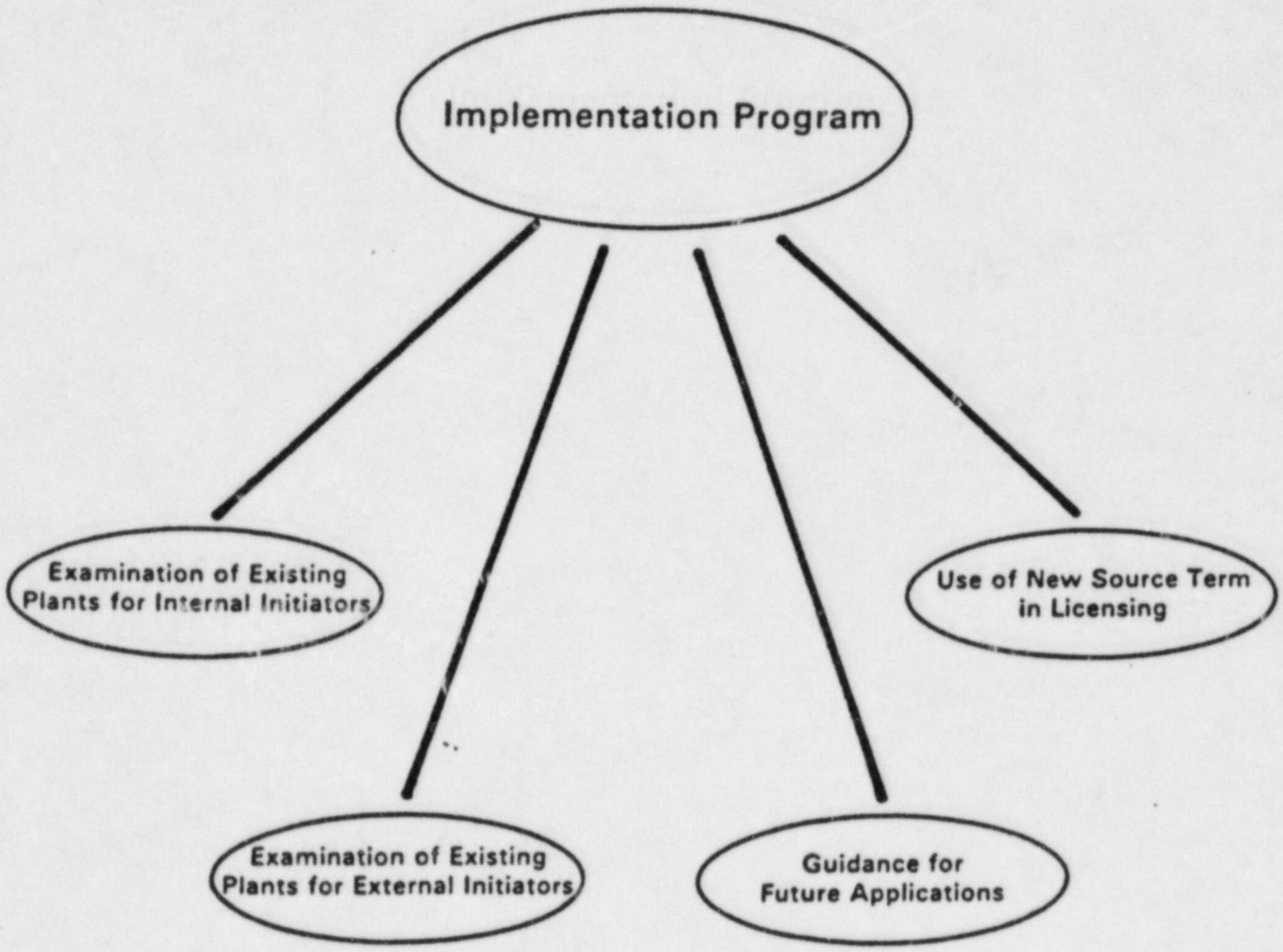
PRESENTER'S TITLE/BRANCH/DIV: DIRECTOR
DIVISION OF SAFETY REVIEW & OVERSIGHT
OFFICE OF NUCLEAR REACTOR REGULATION

PRESENTER'S NRC TEL. NO.: 492-7517

SUBCOMMITTEE: SEVERE ACCIDENT

DISCUSSION TOPICS

- GENERIC LETTER FOR EXISTING PLANTS
- SCOPE OF INDIVIDUAL PLANT EXAMINATIONS
- GUIDELINES AND CRITERIA
- IDCOR-IPEM REVISIONS AND RESOLUTION OF CONCERNS
- NRC COMMENTS ON THE IDCOR-IPEM



Implementation Program
Examination of Existing Plants for Internal Initiators
Examination of Existing Plants for External Initiators
Guidance for Future Applications
Use of New Source Term in Licensing

II. EXAMINATION OF EXISTING PLANTS FOR SEVERE ACCIDENT VULNERABILITIES

- ACCOMPLISHMENTS

- GENERIC LETTER

- SCOPE OF INDIVIDUAL PLANT EXAMINATION

- GUIDELINES AND CRITERIA

- INDIVIDUAL PLANT EVALUATION METHODOLOGY

ACCOMPLISHMENTS

- o BOTH IDCOR AND NRC UPDATED AVAILABLE PRA STUDIES FOR FOUR REFERENCE PLANTS.
- o NRC QUANTIFIED UNCERTAINTIES ASSOCIATED WITH THE RISK ASSESSMENTS.
- o NRC AND IDCOR IDENTIFIED 18 TECHNICAL ISSUES OF CONCERN. 16 OF THESE ISSUES HAVE BEEN RESOLVED.
- o APPROXIMATELY 20 IDCOR/NRC TECHNICAL EXCHANGE MEETINGS.
- o IDCOR DEVELOPED METHODOLOGY FOR SYSTEMATIC PLANT EXAMINATION.
- o NRC DEVELOPED GUIDELINES AND CRITERIA FOR THE PLANT EXAMINATIONS.
- o NRC DRAFTED A GENERIC LETTER WHICH WILL INITIATE PLANT EXAMINATION.

DOCUMENTATION STATUS

- RES 18/20 REPORTS RECEIVED
  5 IN FINAL FORM

- IDCOR IPEM 15 REPORTS RECEIVED
  13 IN DRAFT FORM
  TEST APPLICATION PLANT CASES RECEIVED

- ACRS 3 MEMOS DEALING WITH SEVERE ACCIDENT IMPLEMENTATION
  CONSULTANTS" REVIEW OF IDCOR-IPEM

SCHEDULE

o PROPOSED GENERIC LETTER AND ATTACHMENTS PREPARED BY	DEC. 86
o CRGR AND ACRS REVIEWS COMPLETED BY	FEB. 87
o COMMISSION MEETING	FEB. 87
o GENERIC LETTERS ISSUED IN	MARCH 87
o EXAMINATION OF INDIVIDUAL PLANTS	
- PLANTS WITH LEVEL II AND III PRAs, AND IPEM TEST APPLICATION PLANTS	6 TO 12 MONTHS
- REST OF THE PLANTS	12 TO 18 MONTHS

NRR STAFF PRESENTATION TO THE ACRS

SUBJECT: IPE GENERIC LETTER

DATE: DECEMBER 19, 1986

PRESENTER: RALPH LANDRY

PRESENTER'S TITLE/BRANCH/DIV: NUCLEAR ENGINEER
REGULATORY IMPROVEMENTS BRANCH
DIVISION OF SAFETY REVIEW & OVERSIGHT

PRESENTER'S NRC TEL. NO.: 492-4914

SUBCOMMITTEE: SEVERE ACCIDENT

GENERIC LETTER FOR INDIVIDUAL
PLANT EXAMINATIONS

BASIS: COMMISSION POLICY STATEMENT ON SEVERE REACTOR ACCIDENTS REGARDING FUTURE DESIGNS AND EXISTING PLANTS

"...STAFF WILL ISSUE GUIDANCE ON THE FORM, PURPOSE AND ROLE THAT PRAS ARE TO PLAY IN SEVERE ACCIDENT ANALYSIS AND DECISION MAKING FOR BOTH EXISTING AND FUTURE PLANT DESIGNS AND WHAT MINIMUM CRITERIA OF ADEQUACY PRAS SHOULD MEET."

LIMITS OF THE GENERIC LETTER

- SYSTEMATIC EXAMINATION OF EXISTING PLANTS.
- INTERNAL INITIATORS

"...THE COMMISSION PLANS TO FORMULATE AN INTEGRATED SYSTEMATIC APPROACH TO AN EXAMINATION OF EACH NUCLEAR POWER PLANT NOW OPERATING OR UNDER CONSTRUCTION FOR POSSIBLE SIGNIFICANT RISK CONTRIBUTORS (SOMETIMES CALLED "OUTLIERS") THAT MIGHT BE PLANT SPECIFIC AND MIGHT BE MISSED ABSENT A SYSTEMATIC APPROACH"

SCOPE OF EXAMINATION

SCOPE OF IPE

- SYSTEMATIC EXAMINATION

- ASSESSMENT OF

  ACCIDENT PREVENTION - DESIGN AND OPERATION

  ACCIDENT MITIGATION - DESIGN AND EMERGENCY ACTIONS:

- RESULTS WILL BE MEASURED AGAINST

  GUIDELINES AND CRITERIA

  SAFETY GOAL POLICY STATEMENT

- VULNERABILITY

  THE FAILURE TO FULFILL ANY NECESSARY PREVENTIVE OR MITIGATIVE FUNCTION (HARDWARE, HUMAN ACTION OR PROCEDURE) SPECIFIED IN THE PROPOSED CRITERIA, OR EQUIVALENT CRITERIA

ACCEPTABLE METHODS

- IDCOR - IPEM AS APPROVED

- LEVEL II OR III PRA WITH UPDATE

- LEVEL I PRA TOGETHER WITH SOURCE TERM IPEM,
  OR EQUIVALENT

- SIMPLIFIED, OR PHASE I, PRA WITH NRC APPROVAL

- OTHER SYSTEMATIC EVALUATION METHOD WITH NRC APPROVAL

SCHEDULES

- EXPECTED SCHEDULES

  - PLANTS WITH LEVEL II OR III PRA AND IPEM TEST-APPLICATION PLANTS
    7 TO 14 MONTHS AFTER GENERIC LETTER

  - OTHER PLANTS
    14 TO 20 MONTHS AFTER GENERIC LETTER

- ACTUAL SCHEDULES

  - LICENSEES SUBMIT ACTUAL SCHEDULES
    60 DAYS AFTER GENERIC LETTER

- NRC-LICENSEE INTERFACES DURING PERFORMANCE OF IPE

  - SHORTLY AFTER ISSUANCE OF GENERIC LETTER REGIONAL MEETINGS WITH LICENSEES TO DISCUSS REQUEST

  - QUARTERLY MEETINGS WITH LICENSES IN BETHESDA TO DISCUSS QUESTIONS AND CLARIFICATIONS.

  - MEETINGS WITH UTILITIES OR GROUPS OF UTILITIES AS REQUIRED

NRR STAFF PRESENTATION TO THE ACRS

SUBJECT: SCOPE OF THE INDIVIDUAL PLANT EXAMINATION

DATE: DECEMBER 19, 1986

PRESENTER: ZOLTAN R. ROSZTOCZY

PRESENTER'S TITLE/BRANCH/DIV: CHIEF
REGULATORY IMPROVEMENTS BRANCH
DIVISION OF SAFETY REVIEW & OVERSIGHT

PRESENTER'S NRC TEL. NO.: 492-8016

SUBCOMMITTEE: CLASS 9 ACCIDENTS

INDIVIDUAL PLANT EXAMINATION (IPE)

PURPOSE:

- o DEVELOP AN UNDERSTANDING OF WHAT COULD POSSIBLY GO WRONG IN THE PLANT
- o BE PREPARED TO HANDLE THESE EVENTS

EXAMINATION PROCESS

- o A THOROUGH, SYSTEMATIC EXAMINATION OF PLANT DESIGN, OPERATION, MAINTENANCE AND EMERGENCY OPERATION
- o IDENTIFICATION OF DESIRED PLANT ATTRIBUTES AND ACCIDENT MANAGEMENT MEASURES
- o IDENTIFICATION OF POTENTIAL IMPROVEMENTS IN AREAS WHERE DESIRED FEATURES ARE NOT IN PLACE
- o DECISION ON POTENTIAL IMPROVEMENTS

CRITERIA TO IDENTIFY DESIRED ATTRIBUTES

- o SAFETY GOAL POLICY STATEMENT
- o GUIDELINES AND CRITERIA
- o LICENSEE'S PROPOSED CRITERIA
- o SCREENING CRITERIA

DECISION CRITERIA

- o LICENSEE'S JUDGEMENT
- o BACKFIT RULE

EXPECTED ACCOMPLISHMENTS

- o IDENTIFICATION AND ASSESSMENT OF POTENTIAL SEVERE ACCIDENTS
- o PLANT IMPROVEMENTS TO PREVENT SEVERE ACCIDENTS
- o IMPROVED CONTAINMENT PERFORMANCE
- o DEVELOPMENT AND IMPLEMENTATION OF AN ACCIDENT MANAGEMENT PROGRAM

IDENTIFICATION OF POTENTIAL SEVERE ACCIDENTS

SELECTION OF DOMINANT SEQUENCES

- DEFINITION OF A SEVERE ACCIDENT: AN ACCIDENT THAT RESULTS IN SEVERE CORE DAMAGE (SUBSTANTIAL CORE MELTING, SIGNIFICANT FRACTION OF FISSION PRODUCTS RELEASED FROM FUEL)

- EXTREMELY UNLIKELY EVENTS NEED NOT BE CONSIDERED (AGGREGATE PROBABILITY OF EXTREMELY UNLIKELY EVENTS SHOULD BE LESS THAN 10^{-6})

- IDENTIFY SEQUENCES THAT COULD LEAD TO SEVERE CORE DAMAGE

- REVIEW EACH SYSTEM RELATED SAFETY TO DETERMINE UNAVAILABILITY PERFORM WALKDOWNS AS NEEDED

- CALCULATE SEQUENCE FREQUENCY AND CORE DESIGN FREQUENCY

- PREDICT CONTAINMENT LOADINGS AND CONTAINMENT PERFORMANCE FOR EACH OF THE SEVERE CORE DAMAGE SEQUENCES

- USE SCREENING CRITERIA TO IDENTIFY DOMINANT SEVERE ACCIDENTS

- BASE ASSESSMENT OF PREVENTION, MITIGATION AND ACCIDENT MANAGEMENT ON DOMINANT SEQUENCES

NRC SCREENING CRITERIA FOR DOMINANT SEQUENCES

- CONSIDERATION OF CORE DAMAGE FREQUENCY
  - CDF $>$ 2.10^{-6} PER YEAR
  - CONTRIBUTION TO CDF IS GREATER THAN 5%
- CONSIDERATION OF CONTAINMENT PERFORMANCE
  - CONTAINMENT FAILS IN LESS THAN 1 DAY
  - CONTAINMENT BYPASS PROBABILITY IS GREATER THAN 10^{-7} PER YEAR
  - CONTAINMENT FAILS IN LESS THAN 10 DAYS AND RELEASE IS NOT FILTERED
- ENGINEERING JUDGEMENT
  - SEQUENCES IMPORTANT FOR PLANT DESIGN
  - SEQUENCES IMPORTANT FOR ACCIDENT MANAGEMENT

DOMINANT SEVERE ACCIDENT SEQUENCES

BWR MARK I REFERENCE PLANT

- STATION BLACKOUT

  CDF = ~ $8.1 \cdot 10^{-6}$ PER YEAR

  88% OF CDF

- ANTICIPATED TRANSIENTS WITHOUT SCRAM

  10% OF CDF

  ACCIDENT MANAGEMENT

- LOSS OF DECAY HEAT REMOVAL

  DOMINANT IN PREVIOUS STUDIES

  ACCIDENT MANAGEMENT

- LOSS OF HIGH PRESSURE INJECTION

  DOMINANT IN PREVIOUS STUDIES

  ACCIDENT MANAGEMENT

PLANT IMPROVEMENTS - ACCIDENT PREVENTION

- o PROBABILISTIC MEASURE OF ACCIDENT PREVENTION
- o DETERMINISTIC APPROACH TO ACCIDENT PREVENTION
- o CORE DAMAGE FREQUENCY - BWR MARK I REFERENCE PLANT
- o MAIN CONTRIBUTORS TO CDF - BWR MARK I REFERENCE PLANT
- o POTENTIAL PLANT IMPROVEMENTS - BWR MARK I REFERENCE PLANT

CORE DAMAGE FREQUENCY

BWR MARK I REFERENCE PLANT

- NRC ESTIMATE: 9.9×10^{-6} PER YEAR

- IDCOR ESTIMATE: 8.1×10^{-6} PER YEAR

- UNCERTAINTY BAND AS DETERMINED BY NRC

  2.9×10^{-5} TO 1.6×10^{-6} PER YEAR

PROBABILISTIC MEASURE OF ACCIDENT MITIGATION

- LARGE RELEASE FREQUENCY $> 10^{-6}$ PER YEAR. IMPROVEMENTS IN CONTAINMENT PERFORMANCE ARE DESIRABLE

- LARGE RELEASE FREQUENCY $< 10^{-6}$ PER YEAR. CONTAINMENT PERFORMANCE IS SATISFACTORY, CHECK MAJOR CONTRIBUTORS TO LARGE RELEASE

DETERMINISTIC CONTAINMENT PERFORMANCE MEASURES

- PREVENTION OF EARLY CONTAINMENT FAILURE IS HIGHLY DESIRABLE
- PREVENTION OF CONTAINMENT FAILURE IS DESIRABLE
- LIKELIHOOD OF MAJOR CONTAINMENT BYPASS EVENTS SHOULD BE KEPT AS LOW AS PRACTICAL
- RELEASE THROUGH CONTAINMENT LEAKAGE AND CONTAINMENT VENTING SHOULD NOT ENDANGER PUBLIC HEALTH AND SAFETY - PART 100 LIMITS

SEVERE ACCIDENT GUIDELINES AND CRITERIA

FOR THE FIVE REFERENCE PLANTS

DEPARTMENT OF NUCLEAR ENERGY

BROOKHAVEN NATIONAL LABORATORY

UPTON, NY 11973

PRESENTED AT ACRS SUBCOMMITTEE MEETING

ON CLASS 9 ACCIDENTS

DECEMBER 19, 1986

BROOKHAVEN NATIONAL LABORATORY

ASSOCIATED UNIVERSITIES, INC.

OUTLINE

- BACKGROUND AND OBJECTIVES

- SUMMARY GUIDELINES AND CRITERIA FOR FIVE REFERENCE PLANTS:

  - PREVENTION

  - MITIGATION

ADDITIONAL INFORMATION:

- SPECIFIC GUIDELINES AND CRITERIA FOR EACH REFERENCE PLANT

GUIDELINES AND CRITERIA FOR THE INDIVIDUAL PLANT EXAMINATIONS

OBJECTIVES:

- TO IDENTIFY FEATURES OF THE PLANTS THAT INFLUENCE SEVERE ACCIDENTS AND TO PROVIDE FOR THE DETERMINISTIC AND VERIFIABLE BASES AGAINST WHICH TO JUDGE POTENTIAL VULNERABILITIES TO SEVERE ACCIDENTS BY PARTICULAR PLANT TYPES.

APPROACH:

- TO DEVELOP THE GUIDELINES AND CRITERIA FROM INSIGHTS DERIVED FROM PAST PRAs AND OTHER AVAILABLE SEVERE ACCIDENT INFORMATION

- TO MAINTAIN A BALANCE BETWEEN BOTH SEVERE ACCIDENT PREVENTION AND CONSEQUENCE MITIGATION WITH THE APPROPRIATE WEIGHT BEING GIVEN TO CONTAINMENT PERFORMANCE

EVALUATION PROCESS SCREENING

OBJECTIVE: TO SEPARATE THE POTENTIALLY IMPORTANT SEQUENCES FROM UNIMPORTANT SEQUENCES

CRITERIA:

1. SEQUENCE CDF GREATER THAN 1E-6/RY.

2. SEQUENCE CDF GREATER THAN 5% OF THE TOTAL CDF. (NOTE: INTERNAL EVENTS ONLY)

3. CONDITIONAL FAILURE PROBABILITY FOR CONTAINMENT WITHIN ONE DAY GIVEN VESSEL PENETRATION GREATER THAN 0.1.

4. PROBABILITY OF CONTAINMENT BYPASS GREATER THAN 1E-7/RY.

5. SEQUENCES JUDGED TO BE UNIQUELY IMPORTANT, E.G., VERY SEVERE CONSEQUENCES.

SUMMARY OF GUIDELINES AND CRITERIA

- GUIDELINES GROUPED ACCORDING TO FUNCTION

- DETERMINISTIC CRITERIA SUMMARIZED TO ADDRESS GENERAL AREAS OF APPLICATION

- PLANT TO PLANT COMPARISONS PROVIDED TO HIGHLIGHT SIMILARITIES AND DIFFERENCES

GUIDELINES AND CRITERIA

1. MAINTAIN RCS INTEGRITY

GUIDELINES	A. PREVENT OVERPRESSURE (LOW PRESSURE SYSTEMS)	B. PREVENT STEAM GENERATOR TUBE RUPTURE	C. PREVENT PUMP SEAL LOCA
RELATED CRITERIA	- TESTING AND MAINTENANCE - RELIEF CAPABILITY - OPERATOR TRAINING	- TECH. SPECS. - EMERGENCY PROCEDURES - OPERATOR TRAINING	- CCW/ESW AVAILABILITY - EMERGENCY PROCEDURES - TECH. SPECS. - SEAL INJECTION
PLANT APPLICATION			
BWR MARK I	YES	NOT APPLICABLE	NOT APPLICABLE
BWR MARK II	YES	NOT APPLICABLE	NOT APPLICABLE
BWR MARK III	YES	NOT APPLICABLE	NOT APPLICABLE
PWR ICE CONDENSER	YES	YES	YES
PWR LARGE DRY	YES	YES	YES

GUIDELINES AND CRITERIA

2. MAINTAIN RCS HEAT REMOVAL

GUIDELINES	A. AC INDEPENDENT INJECTION (STATION BLACKOUT MITIGATION)	B. HIGH PRESSURE INJECTION AVAILABILITY	C. ECCS EQUIPMENT FLOODING
RELATED CRITERIA	- EQUIPMENT AVAILABILITY - EMERGENCY PROCEDURES - OPERATOR TRAINING	- RECIRC. PROCS. - RECIRC. COOLING - CONTAINMENT HEAT REMOVAL	- SEPARATION - ELECTRICAL EVAL. - EMERGENCY PROCS.
PLANT APPLICATION			
MARK I	YES	N/A	(TO BE ADDED)
MARK II	YES	N/A	YES
MARK III	YES	N/A	NOT IDENTIFIED AS RISK SIGNIFICANT FOR OTHER PLANTS, BUT FLOODING
PWR ICE CONDENSER	NO (REFER TO 2E)	YES	
PWR LARGE DRY	NO (REFER TO 2E)	YES	LINE WILL BE ADDED.

GUIDELINES AND CRITERIA

2. MAINTAIN RCS HEAT REMOVAL (CONT'D)

GUIDELINES	D. LOW PRESSURE INJECTION AVAILABILITY	E. AUXILIARY FEEDWATER	F. FEED & BLEED COOLING
RELATED CRITERIA	- DEPRESSURIZATION - RECIRC. COOLING - EMERGENCY PROCEDURES - CONTAINMENT HEAT REMOVAL - VENTING	- DIVERSITY - REDUNDANCY - WATER SUPPLY - TRAINING - PROCEDURES	- EQUIP. AVAILABILITY - TRAINING - PROCEDURES - WATER SUPPLY
PLANT APPLICATION			
MARK I	YES	N/A	N/A
MARK II	YES	N/A	N/A
MARK III	YES	N/A	N/A
PWR ICE CONDENSER	N/A	YES	YES
PWR LARGE DRY	N/A	YES	YES

GUIDELINES AND CRITERIA

3. RPV DEPRESSURIZATION

GUIDELINES	A. AUTOMATIC ADS (ELIMINATE PRESSURE PERMISSIVE, ETC.)	B. SECONDARY BLOWDOWN
RELATED CRITERIA	- TECH. SPECS. - EMERGENCY PROCEDURES - TRAINING	- OPERATOR TRAINING - EMERGENCY PROCEDURES - EQUIP. AVAILABILITY
PLANT APPLICATION		
BWR MARK I	YES	NOT APPLICABLE
BWR MARK II	YES	NOT APPLICABLE
BWR MARK III	YES	NOT APPLICABLE
PWR ICE CONDENSER	NOT APPLICABLE	YES
PWR LARGE DRY	NOT APPLICABLE	YES

GUIDELINES AND CRITERIA

4. ATWS RESPONSE

GUIDELINES	A. OPERATOR RESPONSE AND EQUIPMENT
RELATED CRITERIA	- EMERGENCY PROCEDURES - OPERATOR TRAINING - ADS DEFEAT
PLANT APPLICATION	
BWR MARK I	YES
BWR MARK II	YES
BWR MARK III	YES
PWR ICE CONDENSER	YES (EXCEPT ADS)
PWR LARGE DRY	YES (EXCEPT ADS)

GUIDELINES AND CRITERIA

5. STATION BLACKOUT RESPONSE

GUIDELINES	A. OPERATOR RESPONSE AND EQUIPMENT	B. VENTING
RELATED CRITERIA	- EMERGENCY PROCEDURES - OPERATOR TRAINING - EQUIPMENT AVAILABILITY	- EMERGENCY PROCEDURES - OPERATOR TRAINING - EQUIPMENT CAPACITY - CONTROL LOCATION
PLANT APPLICATION		
BWR MARK I	YES	YES
BWR MARK II	YES	YES
BWR MARK III	YES	YES
PWR ICE CONDENSER	YES	STATION BLACKOUT NOT AS DOMINANT FOR PWRs. CONTAINMENT THREAT NOT AS RAPID
PWR LARGE DRY	YES	

GUIDELINES AND CRITERIA

6. EVALUATE SUPPORT SYSTEM FAILURES

GUIDELINES	A. EXAMINE SYSTEM INTERDEPENDENCIES
RELATED CRITERIA	- ANALYSIS OF SUPPORT SYSTEM FAILURE EFFECTS
PLANT APPLICATION	
BWR MARK I	YES
BWR MARK II	YES
BWR MARK III	YES
PWR ICE CONDENSER	YES
PWR LARGE DRY	YES

GUIDELINES AND CRITERIA

7. MAINTAIN CONTAINMENT INTEGRITY

GUIDELINES	A. VENTING	B. PREVENT POOL BYPASS	C. ASSESS DIRECT HEATING	D. CONTAINMENT SPRAY
RELATED CRITERIA	- EMERGENCY PROCS. - TRAINING - EQUIP. CAPACITY - EQUIP. FUNCTION	- DEBRIS CONTROL - CONTAINMENT ISOLATION - DOWNCOMERS	- CONTAINMENT CAPACITY - CAVITY GEOMETRY - CAVITY FLOODING	- DIVERSE POWER - EMERGENCY PROCS. - TRAINING - LONG TERM
PLANT APPLICATION				
BWR MARK I	YES	YES*	FOUND NOT IMPORTANT TO RISK	YES
BWR MARK II	YES	YES	" "	YES
BWR MARK III	YES	ONLY ADDRESSES CONTAINMENT ISO-LATION	" "	YES
PWR ICE CONDENSER	MAY BE NEEDED FOR LONG TERM CONTAIN-HEAT REMOVAL	CONTAINMENT ISO-LATION	YES	YES
PWR LARGE DRY	" "	CONTAINMENT ISO-LATION	YES	ADDITIONAL SPRAY CAPABILITY FOUND NOT TO BE IMPOR-TANT TO RISK.

*ADDRESSES DEBRIS CONTROL AND CONTAINMENT ISOLATION.

GUIDELINES AND CRITERIA

8. CONTROL HYDROGEN BURNING

GUIDELINES	A. PREVENT DEINERTING	B. IGNITER AVAILABILITY
RELATED CRITERIA	- EMERGENCY PROCEDURES - EQUIPMENT OPERABILITY	- DIVERSE POWER - EMERGENCY PROCEDURES
PLANT APPLICATION		
BWR MARK I	YES	NO (INERT CONTAINMENT)
BWR MARK II	YES	NO (INERT CONTAINMENT)
BWR MARK III	NOT APPLICABLE	YES
PWR ICE CONDENSER	NOT APPLICABLE	YES
PWR LARGE DRY	NOT APPLICABLE	IGNITERS ARE NOT IMPORTANT TO RISK UNLESS SPECIFIC PLANT IS VULNERABLE TO OVERPRESSURE FAILURE BY HYDROGEN BURNING (PENDING RESULTS OF GENERIC ISSUE 121).

GUIDELINES AND CRITERIA

9. MAINTAIN CONTAINMENT HEAT REMOVAL

GUIDELINES	A. ALTERNATE RPV INJECTION/VENTING	B. ICE CONDENSER & FANS
RELATED CRITERIA	- DIVERSE POWER - LONG TERM WATER SUPPLY - EMERGENCY PROCEDURES - TRAINING	- EMERGENCY PROCEDURES - TRAINING
PLANT APPLICATION		
BWR MARK I	YES	N/A
BWR MARK II	YES	N/A
BWR MARK III	YES	N/A
PWR ICE CONDENSER	AC DEPENDENCE NOT DOMINANT CONTRIBUTOR. ALTERNATIVE APPROACH SUGGESTED: (AC INDEPENDENT FEEDWATER)	YES
PWR LARGE DRY		FAILURE OF CONTAINMENT FAN COOLERS NOT DOMINANT CONTRIBUTOR TO RISK

BWR

INDIVIDUAL PLANT

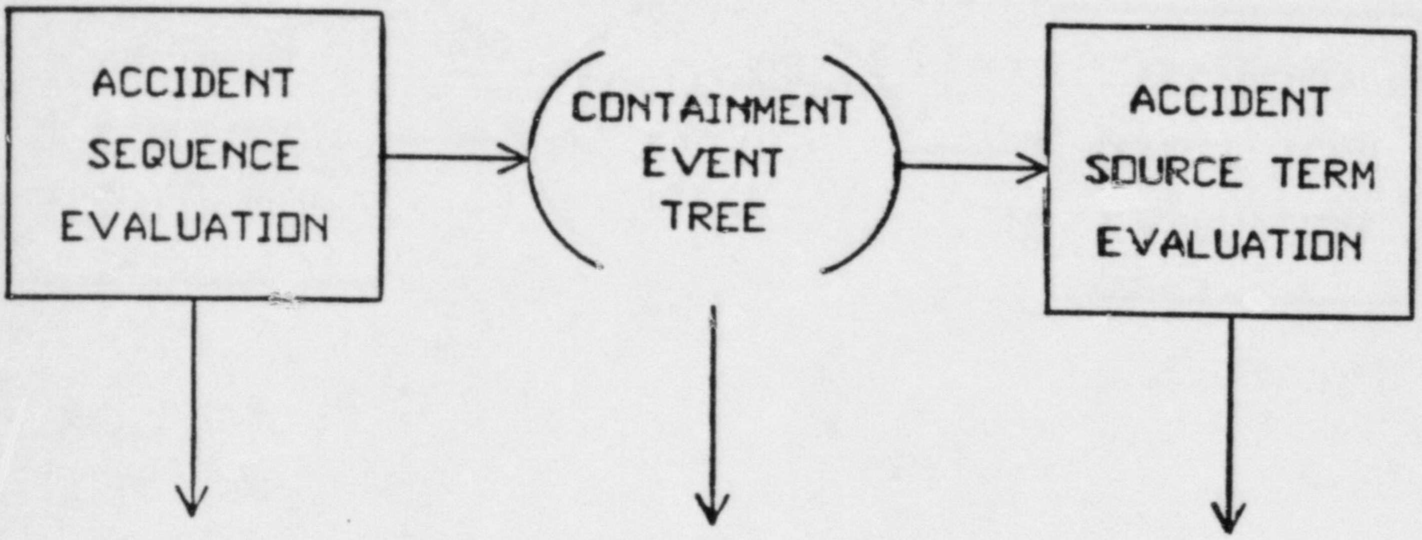
EVALUATION

METHODOLOGY

OBJECTIVES OF PRESENTATION:

- BRIEF REVIEW OF IPE
- DISCUSSION OF THE PHILOSOPHY OF THE TECHNIQUE
- KEY EVENTS IN IPE DEVELOPMENT PROCESS
- IPE APPLICATIONS RESULTS
- IDENTIFY COMMENTS AND RESOLUTION ON THE BWR IPE METHOD
- INSIGHTS FROM THE TEST PLANT APPLICATIONS
- IDENTIFY WHERE SUPPLEMENTAL PROGRAMS MAY ENHANCE THE METHODOLOGY
- SUMMARY

IDCOR
INDIVIDUAL PLANT EVALUATION METHODOLOGY



PLANT SPECIFIC EVALUATION TO IDENTIFY THAT THE PLANT RISK IS IN THE SAME RANGE AS THAT ESTIMATED IN THE IDCOR CONCLUSIONS.

SCOPE

- "INTERNAL" EVENTS:
  - TRANSIENTS, ATWS, LOCAs, RARE INITIATORS
  - INTERNAL FLOODS
  - INTERFACING LOCA
  - COMMON MODE FAILURES
  - SUPPORT SYSTEM DEPENDENCIES AND INITIATORS
- APPROXIMATE METHOD CAPABLE OF EXPANSION TO LEVEL 1 PRA
- DETAILED EVENT TREES
- ALL GE BWRs
- SELECT SUPPORT SYSTEMS ARE EXPLICITLY ADDRESSED:
  - ROOM COOLING
  - SERVICE WATER
  - AC POWER
  - DC POWER
  - INSTRUMENT AIR/N_2

  PROVISION IS MADE FOR ADDITIONAL SUPPORT SYSTEM DEPENDENCIES
- DEPENDENCIES ADDRESSED
  - FUNCTIONAL
  - HUMAN
  - INTERSYSTEM
- PLANT WALKDOWN
- SYSTEM NOTEBOOKS
- OPERATING EXPERIENCE DATA

PROCESS INCLUDES

- <u>SYSTEM NOTEBOOKS</u>: REFERENCE SOURCE OF INFORMATION ABOUT THE PLANT

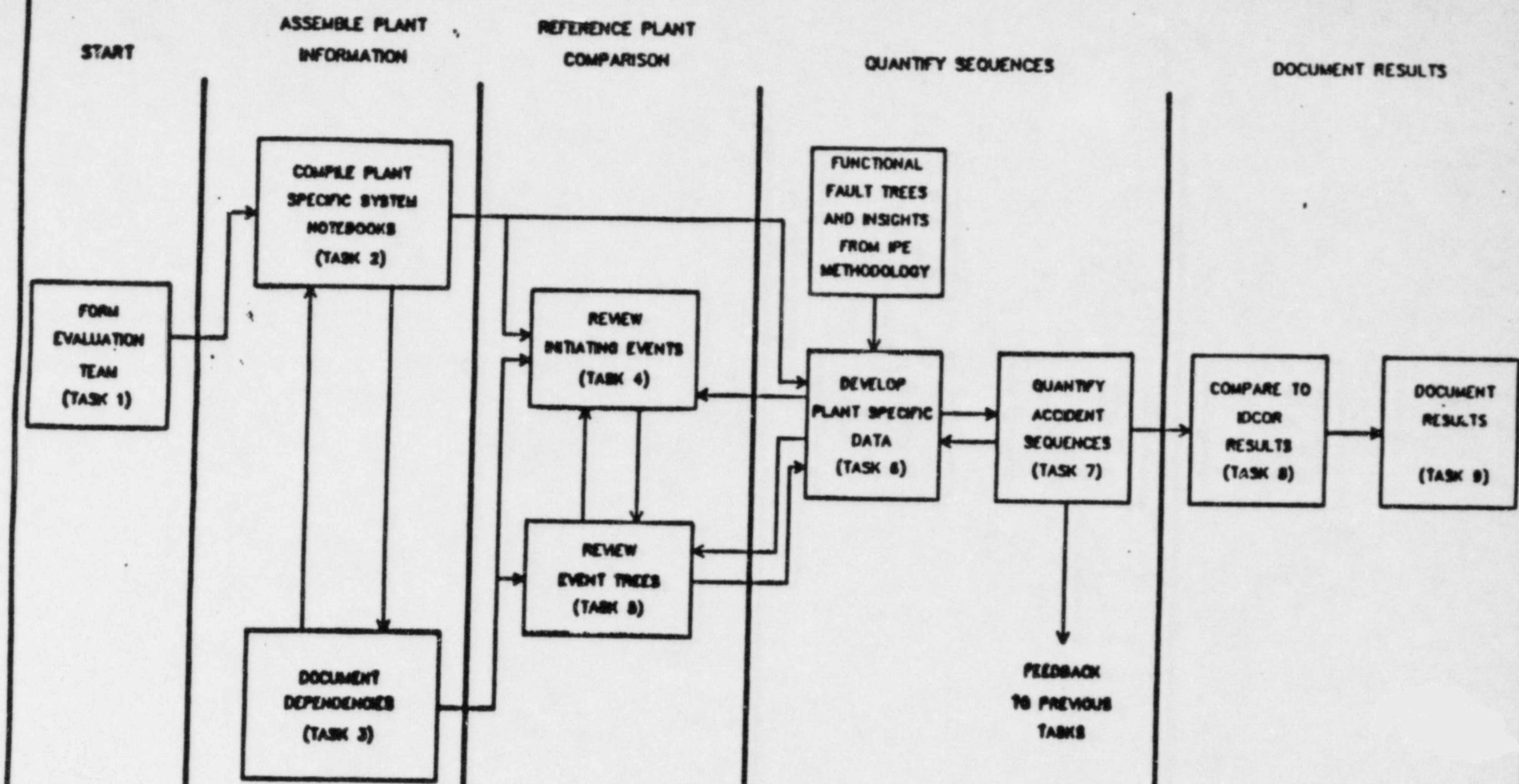
- <u>EVENT TREES</u>: FRAMEWORK FOR IDENTIFYING AND EVALUATING ACCIDENT SEQUENCES

- <u>FAULT TREES</u>: FOCAL POINT FOR PLANT SPECIFIC DESIGN, OPERATION, MAINTENANCE, AND TEST INFORMATION

- <u>SUPPORT SYSTEM DEPENDENCY MATRICES</u>: IDENTIFY AND MODEL IMPORTANT INTERACTIONS

- <u>AVAILABLE DATA</u>: QUANTIFY THE MODELS (GENERIC DATA, AND REFERENCE PLANT DATA IS PROVIDED IF PLANT SPECIFIC DATA IS NOT AVAILABLE)

- <u>ENGINEERING INSIGHTS</u>: PROBE FOR PLANT UNIQUE FEATURES OR POTENTIALLY VULNERABLE AREAS



Implementation of IPE for Accident Sequence Quantification

INPUT TO CONTAINMENT EVENT TREE EVALUATION

ACCIDENT SEQUENCE END STATES

- CORE MELT
- SUPPORT SYSTEM CONFIGURATION
- CONTAINMENT STATUS
  - TEMPERATURE
  - PRESSURE
  - INTEGRITY
- RPV STATUS
  - REACTIVITY CONTROL
  - PRESSURE

PRELIMINARY RESULTS OF IPE VERIFICATION PHASE

COMMENTS	IPE UPDATES
1. NEED EXPLICIT DIRECTIONS ON MECHANICS OF PROCESS	1. PROVIDE: - ROAD MAP OF THE - PROVIDE NUMERICAL EXAMPLES (COMPLETE)
2. NEED PRA EXPERTISE ON TEAM	2. MODIFY RECOMMENDED TEAM MAKE UP TO IDENTIFY PRA PERSON, I.E. DELETE "OPTIONAL" (COMPLETE)
3. CLARIFY THE USE OF SOME QUANTITATIVE ESTIMATES	3. INCORPORATE (IN PROGRESS)
4. CLARIFY SELECTED ENGINEERING INSIGHTS TO MAKE THE ISSUES OBVIOUS AND THE ACTIONS CLEAR	4. UPDATE THE METHODOLOGY (COMPLETE)
5. PROVIDE EXAMPLES OF THE DEPENDENCY MATRICES	5. UPDATE THE METHODOLOGY (COMPLETE)
6. PROVIDE ADDITIONAL EXAMPLES OF SERVICE WATER SYSTEMS FOR QUANTIFICATION	6. UPDATE THE METHODOLOGY (COMPLETE)
7. CORRECT ERRORS	7. UPDATE THE METHODOLOGY (COMPLETE)
8. MAY REQUIRE MORE TIME THAN IDENTIFIED	8. INCLUDE DISCUSSION IN METHODOLOGY (COMPLETE)

RESOLUTION OF NRC COMMENTS

ON

BWR IPE

(SEPTEMBER 1986)

COMMENT	SYSTEMS ANALYSIS RESOLUTION
1. SAFETY GOAL CONSISTENCY	o NO COMMON MEASURE FOR COMPARISON o THRESHOLDS ACCOMPLISH SIMILAR OBJECTIVE
2. MATCHING CRITERIA FOR VARIOUS LEVELS	o IPE IDENTIFIES VULNERABILITIES AFFECTING OVERALL LEVEL OF SAFETY o ONLY FUNCTIONAL LEVEL MATCHING IS MEANINGFUL o INSIGHTS PROVIDED IN APPENDIX D
3. CHARACTERIZE VENTING FOR MARK I AND II	o INCLUDED IN APPENDIX D o ONGOING INDUSTRY PROGRAMS WILL PROVIDE ADDITIONAL GUIDANCE
4. EXAMPLES OF PLANT SPECIFIC VULNERABILITIES	o NO "OUTLIERS" IDENTIFIED o HIGH-LEVEL CHECKLIST o APPENDIX D

RESOLUTION OF NRC COMMENTS

ON

BWR IPE

(SEPTEMBER 1986)

COMMENT	SYSTEMS ANALYSIS RESOLUTION
5. EQUIPMENT SURVIVABILITY	o ESSENTIAL EQUIPMENT IDENTIFIED IN EVENT TREES AND APPENDIX D o GUIDANCE INCLUDED IN APPENDIX D
6. VISUAL INSPECTION PROCEDURES	o DISCUSSION ENHANCED
7. DOCUMENTATION REQUIREMENTS	o ADDITIONAL GUIDANCE PROVIDED TO SUPPORT INTERNAL UTILITY USE AND REVIEW
8. SOURCE TERM METHODOLOGY INTERFACE	o SEQUENCES BINNED o THRESHOLD o END STATE CONDITIONS o CONTAINMENT EVENT TREE

VERIFICATION PHASE

- o PURPOSE: TEST THE BWR IPE METHODOLOGY AND VERIFY ITS USABILITY

- o BWR PLANTS IDENTIFIED FOR VERIFICATION PHASE
  - PEACH BOTTOM (BWR/4) MARK I
  - SUSQUEHANNA (BWR/4) MARK II
  - SHOREHAM (BWR/4) MARK II
  - GRAND GULF (BWR/6) MARK III

- o RESULTS SUBMITTED TO NRC AS PACKAGE WITH THE UPDATED IPE METHOD IN MAY 1986 (GRAND GULF DECEMBER 1986)

BWR IPE

ACCIDENT SEQUENCE EVALUATION

PLANT	ESTIMATED MANPOWER (MM)		CALENDAR MONTHS
	UTILITY	CONSULTANT	
SHOREHAM	20	4	4
PEACH BOTTOM	16	12	8
SUSQUEHANNA	24	–	6
GRAND GULF	45*	3*	6

* GRAND GULF EFFORT INCLUDED TASKS NOT SPECIFIED IN IPEM.

BWR IPE

ACCIDENT SEQUENCE EVALUATION

PLANT	PRA AVAILABLE	CORE MELT FREQUENCY (PER YR.)	
		PRA	IPE RESULTS
SHOREHAM	YES	5E−5	8E−5
PEACH BOTTOM	YES (WASH 1400)	3E−5	4E−5
SUSQUEHANNA	YES	1E−5 (1)	3E−5

(1) NOT PUBLISHED

PEACH BOTTOM ATOMIC POWER STATION
CDF COMPARISON

SEQUENCE TYPE	CLASS	ASEP PROGRAM	IPE STUDY
TQUV & TQUX	IA & ID	6.8E-8/YR	1.4E-5/YR
TB	IB	8.7E-6/YR	8.6E-6/YR
TW	II	~1.0E-8/YR	7.4E-7/YR
AE & S_1E	III	~1.1E-7/YR	~1.5E-6/YR
TC	IV	1.0E-6/YR	4.1E-6/YR+
TOTAL		9.9E-6	~2.0E-5

\+ TC:

CLASS IC	5.9E-7/YR
CLASS III	1.1E-9/YR
CLASS IV	3.5E-6/YR
TOTAL	4.1E-6/YR

CORE DAMAGE FREQUENCY
DOMINANT SEQUENCE CONTRIBUTION
(IN PERCENT)

	ASEP PROGRAM+	PEACH BOTTOM IPE STUDY++
STATION BLACKOUT	88%	39%
ATWS	10%	19%
REMAINING SEQUENCES	2%	42%+++

\+ CDF = 9.9E-6/RX YR

++ CDF = 2.2E-5/RX YR

+++ MAJOR CONTRIBUTION FROM TRANSIENT INITIATED SEQUENCES

SUMMARY OF

THE METHOD

- DEVELOPED TO CALCULATE A REALISTIC PLANT SPECIFIC CORE MELT FREQUENCY
- BASED ON INSIGHTS FROM PAST PRAs AND IDCOR
- IS NOT A PRA
- IS USABLE FOR COMMUNICATION TO MANAGEMENT
- PROVIDE UTILITY A RISK PERSPECTIVE ON THEIR INVESTMENT
- PROVIDES REAL INSIGHTS INTO IMPROVEMENTS, HOWEVER SMALL, THAT CAN BE DONE NOW BY THE UTILITY TO MAKE THE PLANT SAFER
- IDENTIFY POTENTIAL OUTLIER
- QUANTIFY THE CORE MELT FREQUENCY BY TYPE OF SEQUENCE
- EXPANDABLE TO LEVEL 1 PRA
- IS EASILY UPDATED IF INFORMATION BECOMES AVAILABLE IN THE FUTURE, E.G. PLANT SPECIFIC DATA

PWR INDIVIDUAL PLANT EVALUATION STATUS

DECEMBER 18/19 1986

WESTINGHOUSE ELECTRIC CORPORATION

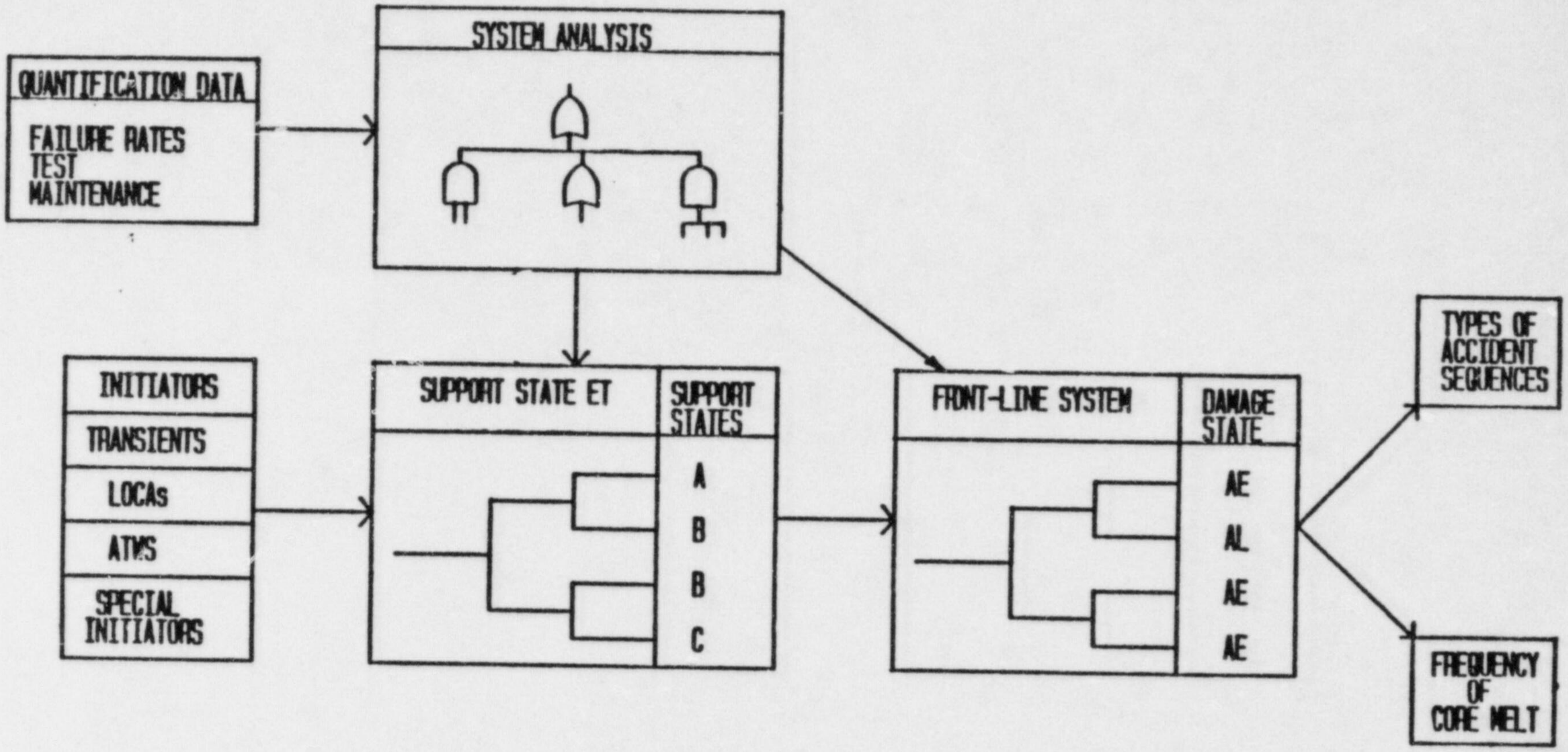
KEN J. VAVREK

(412) 374 4028

THE METHODOLOGY APPROACH INCLUDES THE FOLLOWING COMPONENTS:

- O INITIATING EVENTS
- O EVENT TREES/ACCIDENT SEQUENCES
- O SUCCESS CRITERIA
- O SUPPORT SYSTEMS
- O FAULT TREES/SYSTEMS ANALYSIS
- O FAILURE DATA

PLANT SPECIFIC IMPLEMENTATION FLOW DIAGRAM



IPE METHODOLOGY REVISIONS

AREAS OF NEW IPE DOCUMENTATION

- METHODS MANAGEMENT
- SYSTEM INTERACTION EVALUATION
- INTERNAL FLOODING EVALUATION

AREAS OF EXPANDED IPE DOCUMENTATION

- COMMON CAUSE FAILURE TREATMENT
- LOCA INITIATING EVENT FREQUENCIES
- EVENT TREE TEMPLATES
- ACCIDENT SEQUENCE NOTEBOOK

METHODS MANAGEMENT

COORDINATION

PROJECT LEADER & RESPONSIBILITIES

PLANNING

SELECTION OF PROJECT PARTICIPANTS

ALLOCATION OF RESOURCES PER TASK

SCHEDULING

SCHEDULING OF PROJECT TASKS

TECHNICAL QUALITY

CONSISTENT DOCUMENTATION OF IPE ANALYSIS

SYSTEM INTERACTION EVALUATION

PLANT WALKTHROUGH

ENVIRONMENTAL CONDUCTORS CHECKLIST

CONTROL ROOM MAN-MACHINE INTERFACES CHECKLIST

PLANT TALKTHROUGH

NORMAL, EMERGENCY, TEST & MAINTENANCE PROCEDURES CHECKLIST

INTERNAL FLOODING EVALUATION

QUALITATIVE EVALUATION

- DETERMINATION OF POTENTIAL FLOODING SOURCES
- COMPONENTS AFFECTED BY FLOODING SOURCES
- DESCRIPTION OF SOURCES & POSSIBLE MITIGATING ACTIONS
- MAPPING OF CONSEQUENCES OF FLOODING
- POTENTIAL PLANT AREAS IDENTIFIED

QUANTITATIVE EVALUATION

- INITIATING EVENT FREQUENCIES
- MITIGATING ACTION PROBABILITIES
- ACCIDENT SEQUENCE QUANTIFICATION
- IDENTIFICATION OF DOMINANT ACCIDENT SEQUENCES

TABLE D.4-4

INTERNAL FLOODING INITIATING EVENT DATA BASE

Component	Failure Mode	Failure Rate	Source
Air Operated Valve	Rupture	2.0E-7/hr	NUREG-1363
Manual Valve	Rupture	1.3E-8/hr	NUREG-1363
Motor-Operated Valve	Rupture	1.7E-7/hr	NUREG-1363
Check Valves	Rupture	5.2E-8/hr	NUREG-1363
Tank	Rupture	8.6E-10/hr	WASH-1400
Piping (> 3" Diameter)	Rupture	8.6E-10/section-hr	WASH-1400
(< 3" Diameter)		8.6E-9/section-hr	WASH-1400
Expansion Joints	Rupture	2.5E-4/expansion joint-year	Oconee 3 PRA

SOURCE TERM METHODOLOGY

R. E. Henry
J. R. Gabor
M. A. Kenton

Fauske & Associates, Inc.
Burr Ridge, Illinois

Presentation to:

ACRS Severe (Class 9)
Accident Subcommittee

December 19, 1986

Washington, D.C.

IPE SOURCE TERM METHODOLOGY

- Designed to search for potential outliers for containment behavior (fission product retention) under severe accident conditions.

- Focus is on major mechanisms for fission product retention.

  - Quenching of debris and containment heat removal.

    1. Containment sprays.

    2. Containment fan coolers where applicable.

  - Wetwell venting where applicable.

  - Deposition in containment.

  - Deposition in adjacent buildings.

IPE SOURCE TERM METHODOLOGY

- Uses streamlined containment event trees.
- Uses likelihood of occurrence for each decision point.

  High - Written procedures with equipment that can be implemented on a timely basis.

  Medium - Demonstrated capability with equipment that could be used on a timely basis.

  Low - No written procedures or no demonstrated capability.

IPE SOURCE TERM METHODOLOGY

- Identifies the same controlling features as full scope PRAs.
- IPE approximate source terms can be developed on a sequence specific basis.
- IPE approximate source terms are in agreement with those developed in full scope PRAs.
- IPE methodology is sufficient for searching for outlier conditions or configurations.

Table H.1

SEVERE ACCIDENT SEQUENCES CONSIDERED IN THE IDCOR ZION CONTAINMENT ANALYSES

Sequence Designation	Sequence Description	Environmental Source Term Calculated by the MAAP Code	IPE Approximate Source Term
Station Blackout With Recovery	Recovery of one vital bus at 2.5 hours.	No core damage.	Insignificant (containment heat removal available)
Station Blackout With Recovery	Station blackout with a seal LOCA with recovery of one vital bus at 1 hour.	No core damage.	Insignificant (containment heat removal available)
Station Blackout With Recovery	Station blackout with a seal LOCA with recovery of one vital bus at 2.5 hours.	Core damage without containment failure.	Insignificant (containment heat removal available)
Station Blackout With Recovery	Station blackout with a seal LOCA with recovery of one vital bus at 6 hours.	Core damage, vessel failure but no containment failure.	Insignificant (containment heat removal available)
Station Blackout With Recovery	Station blackout with a seal LOCA with recovery of one vital bus at 15 hours.	Core damage, vessel failure but no containment failure.	Insignificant (containment heat removal available)
Small LOCA With Recovery	Small LOCA with initial failure to achieve recirculation for injection. Recirculation capabilities recovered at 10 hours.	Core damage, vessel failure but no containment failure.	Insignificant (containment heat removal available)

Table H.1 (Continued)

SEVERE ACCIDENT SEQUENCES CONSIDERED IN THE IDCOR ZION CONTAINMENT ANALYSES

Sequence Designation	Sequence Description	Environmental Source Term Calculated by the MAAP Code	IPE Approximate Source Term
Large LOCA	A large break with only one charging pump operational and limited RWST refill capability.	No core damage.	Insignificant (containment heat removal available)
Station Blackout With a Seal LOCA	Loss of all AC power and auxiliary feedwater with a 50 gpm per pump seal LOCA at 45 minutes.	CsI - 0.002 Te - 2×10^{-5}	CsI < 0.01 Te - 0.004*
Station Blackout	Loss of all AC power and auxiliary feedwater without a seal LOCA.	CsI - 0.002 Te - 2×10^{-5}	CsI < 0.01 Te - 0.004*
Small LOCA	Small (2 inch) cold leg break, failure of ECCS recirculation but fan coolers and containment sprays are available.	Core damage, vessel failure but no containment failure.	Insignificant (containment heat removal available)
Large Break LOCA	Large cold leg break, failure of ECCS recirculation but fan coolers and containment sprays are available.	Core damage, vessel failure but no containment failure.	Insignificant (containment heat removal available)
V Sequence	Assumed failure of the isolation valve discs and consequential failure of the RHR pump seals.	Noble gases.	Noble gases.

*Sr and Ba assumed equal.

Table L-1

SEVERE ACCIDENT SEQUENCES CONSIDERED IN THE IDCOR PEACH BOTTOM CONTAINMENT ANALYSES

Sequence Designation	Sequence Description	Environmental Source Term Calculated by the MAAP Code	IPE Approximate Source Term
TW With Recovery	Loss of suppression pool cooling with refilling of the condensate storage tank and containment venting.	No core degradation.	Insignificant (containment heat removal available)
TC With Recovery	ATWS with injection flow throttled to TAF and containment	No core damage.	Insignificant (containment heat removal available)
Small LOCA With Failure of Injection	Small LOCA with failure of injection. Drywell sprays initiated at 22 hours. venting.	Core damage without containment failure.	Insignificant (containment heat removal available)
Transient With Loss of Low Pressure Injection and Containment Heat Removal With Recovery	Transient with loss of injection and containment heat removal with on-site power restored at 9 hours.	No core damage.	Insignificant (containment heat removal available)
Transient With Failure of Heat Removal	Transient with failure of containment heat removal and no recovery.	NG - 1.0 I - 0.2 Cs - 0.2	NG - 1.0 I - 0.1 Cs - 0.1
ATWS (TC) (Case 1)	ATWS with no operator actions taken.	NG - 1.0 I - 0.1 Cs - 0.1	NG - 1.0 I - 0.1 Cs - 0.1

Table L-1 (Continued)

SEVERE ACCIDENT SEQUENCES CONSIDERED IN THE IDCOR PEACH BOTTOM CONTAINMENT ANALYSES

Sequence Designation	Sequence Description	Environmental Source Term Calculated by the MAAP Code	IPE Approximate Source Term
ATWS (TC) (Case 2)	ATWS with wetwell venting at 115 psia.	NG - 1.0 I - 0.03 Cs - 0.03	NG - 1.0 I < 0.01 Cs < 0.01
ATWS (TC) (Case 3)	ATWS with refill of the CST.	NG - 1.0 I - 0.03 Cs - 0.03	NG - 1.0 I < 0.01 Cs < 0.01
ATWS (TC) (Case 4)	ATWS with wetwell venting and refill of the CST.	NG - 1.0 I - 6×10^{-4} Cs - 6×10^{-4}	NG - 1.0 I < 0.01 Cs < 0.01
Station Blackout	Station blackout without recovery.	NG - 1.0 I - 0.05 Cs - 0.05	NG - 1.0 I - 0.02 Cs - 0.02
Small LOCA Without Injection	Small LOCA with failure of all injection and no recovery.	NG - 1.0 I - 0.04 Cs - 0.04	NG - 1.0 I - 0.02 Cs - 0.02

NRR STAFF PRESENTATION TO THE
ACRS

SUBJECT: NRC COMMENTS ON IDCOR IPEM

DATE: DECEMBER 19, 1986

PRESENTER: FRANKLIN COFFMAN

PRESENTER'S TITLE/BRANCH/DIV: SECTION LEADER/
REGULATORY IMPROVEMENTS BRANCH/
DSRO

PRESENTER'S NRC TEL. NO.: 492-4609

SUBCOMMITTEE: SEVERE (CLASS 9) ACCIDENTS

EVALUATION STANDARDS FOR SEVERE ACCIDENT METHODS

1. CAPABILITY TO FIND VULNERABILITIES

2. CONSISTENCY WITH GENERIC RESOLUTIONS

3. SYSTEMATIC EXAMINATIONS
   A. INTEGRATE CONSIDERATIONS OF SAFETY CONCERNS
   B. BALANCE PREVENTIONS AND MITIGATIONS
   C. ACHIEVE UNIFORM EXAMINATIONS
   D. PROVIDE REASONABLE COMPLETENESS

4. LIMITATIONS AND CAUTIONS

5. COVERAGE OF GUIDELINES AND CRITERIA

6. DOCUMENTATION AND PRESENTATION

7. APPLICABILITY BETWEEN EXISTING PLANT AND REFERENCE PLANT

8. ROLE OF VISUAL INSPECTION

9. COVERAGE OF CURRENT INSIGHTS

1. CAPABILITY TO FIND VULNERABILITIES

A. IDCOR (POSITION DERIVED FROM 1983 DRAFT SAFETY GOALS)

OUTLIERS

1. CDF > 3E-4/RY (INTERNAL EVENTS)
2. SEQUENCE > 30 TIMES SIMILAR IDCOR SEQUENCE
3. "CHECKLIST OF EXAMPLES" (TABLE 3.2-3)
4. TWO NEGATIVE ANSWERS ON SIMPLIFIED EVENT TREE

POTENTIAL AREAS FOR IMPROVEMENT

1. KEY SYSTEMS UNAVAILABILITY ≥ 1E-2 FOR
   INTERNAL INITIATORS > 0.1/YR
2. SEQUENCE > 0.2 EQUIVALENT IODINE
   AT > 1E-5/RY

INTERNAL UTILITY RESTRICTIVE CRITERIA POSSIBLE

B. NRC

VULNERABILITIES (UNDER CONSIDERATION)

1. MEAN CDF   POSITION BEING DEVELOPED FROM '86 SAFETY GOAL
   (BOTH INTERNAL AND EXTERNAL)
2. MEAN PROBABILITY OF LARGE RELEASE ≥ 1E-6/RY

POTENTIAL VULNERABILITIES (UNDER CONSIDERATION)

1. SEQUENCE CDF > 2E-6/RY
2. SEQUENCE > 5% TOTAL CDF
3. PROB. OF CONT. FAIL. GIVEN CORE DAMAGE ≥ 0.01
4. CONTAINMENT BYPASS SEQUENCES > 1E-7/RY
5. UNIQUELY IMPORTANT SEQUENCES

C. TEST APPLICATIONS BY UTILITY AND IDCOR

1. NO "OUTLIERS" WERE DISCOVERED
2. MANY PLANT-SPECIFIC "INSIGHTS" WERE HIGHLIGHTED

2. CONSISTENCY WITH GENERIC RESOLUTIONS

A. IDCOR

USE OF MAAP CODE

1. REVIEW OF MAAP BEING DISCUSSED WITH NRC MANAGEMENT
2. NO REVISIONS TO MAAP PLANNED

CONSIDERATION OF UNCERTAINTIES

1. IDCOR IPEM RESULTS ARE INSENSITIVE TO UNCERTAINTIES
2. UNCERTAINTY STUDY UNDERWAY

EVENT "V" CHECKLIST

1. REVISIONS PLANNED

SIMPLIFIED CET REVISIONS

1. REVISIONS PLANNED

B. NRC

USE OF MAAP CODE

1. MAAP IS NOT A REVIEWED CODE
2. MAAP REVIEW IS SEPARABLE FROM APPROVAL OF IDCOR IPEM

CONSIDERATION OF UNCERTAINTIES

1. IPE CRITERIA MUST ADDRESS PHENOMENOLOGICAL UNCERTAINTIES

3. SYSTEMATIC EXAMINATIONS

A. IDCOR

ACHIEVE UNIFORM EXAMINATIONS

1. BWR AND PWR METHODS ARE EQUIVALENT

B. NRC

ACHIEVE UNIFORM EXAMINATIONS

1. TEST APPLICATIONS REPORTS ARE DIVERSIFIED
2. TEST APPLICATIONS APPEAR TO HAVE SIGNIFICANT VARIATIONS AMONG UTILITIES EVEN THOUGH IDCOR IPEM CONSULTANTS WERE USED.
3. NO TEST APPLICATION WITHOUT BEING SUPPLEMENTED BY PRA-TYPE ANALYSES
4. IDCOR IPEM STILL APPEARS TO BE UNDER DEVELOPMENT
5. IDCOR IPE METHODS MANAGEMENT ALLOWS MANY OPTIONS WITH THE POTENTIAL TO INTRODUCE DEVIATIONS
6. SOURCE TERM METHODS ARE SIGNIFICANTLY SIMPLER THAN ACCIDENT SEQUENCE METHODS

DOCUMENTS FOR REVIEW OF THE IDCOR IPEM:

1. DRAFT IDCOR PROGRAM REPORT, TECHNICAL REPORT 85.3-A1; PWR ACCIDENT SEQUENCE - INDIVIDUAL PLANT EVALUATION METHODOLOGY, APRIL 1986

2. DRAFT IDCOR PROGRAM REPORT, TECHNICAL REPORT 85.3-A2; PWR SOURCE TERM - INDIVIDUAL PLANT EVALUATION METHODOLOGY, APRIL 1986

3. DRAFT IDCOR PROGRAM REPORT, TECHNICAL REPORT FAI/85-58, APPROXIMATE SOURCE TERM METHODOLOGY FOR PRESSURIZED WATER REACTORS, FAUSKE & ASSOCIATES, DECEMBER 1986

4. DRAFT IDCOR PROGRAM REPORT, TECHNICAL REPORT 85.3-B1; BWR ACCIDENT SEQUENCE - INDIVIDUAL PLANT EVALUATION METHODOLOGY, APRIL 1986

5. DRAFT IDCOR PROGRAM REPORT, TECHNICAL REPORT 85.3-B2; BWR SOURCE TERM - INDIVIDUAL PLANT EVALUATION METHODOLOGY, APRIL 1986

6. DRAFT IDCOR PROGRAM REPORT, TECHNICAL REPORT FAI/86-1, APPROXIMATE SOURCE TERM METHODOLOGY FOR BOILING WATER REACTORS, FAUSKE & ASSOCIATES, DECEMBER 1986

7. DRAFT IDCOR PROGRAM REPORT, BWR IPE PLANT SPECIFIC ACCIDENT SEQUENCE EVALUATION METHODOLOGY, USER's GUIDE, Rev. 1, DEC 1986

8. DRAFT IDCOR PROGRAM REPORT, INDIVIDUAL PLANT EVALUATION, PEACH BOTTOM ATOMIC POWER STATION, PHILADELPHIA ELECTRIC COMPANY, MAY 1986

9. DRAFT IDCOR PROGRAM REPORT, INDIVIDUAL PLANT EVALUATION FOR SUSQUEHANNA STEAM ELECTRIC STATION, P.R. HILL, C.A. KUKIELKA, AND C.A. BOSCHETTI, SUBMITTED TO IDCOR APRIL 1986

10. DRAFT IDCOR PROGRAM REPORT, SHOREHAM NUCLEAR POWER STATION, IDCOR INDIVIDUAL PLANT EVALUATION, LONG ISLAND LIGHTING COMPANY, APRIL 1986

11. DRAFT IDCOR PROGRAM REPORT, INDIVIDUAL PLANT EVALUATION METHODOLOGY APPLIED TO THE OCONEE NUCLEAR GENERATING STATION, SUBMITTED TO: AIF/IDCOR PROGRAM, MAY 1986

12. DRAFT IDCOR PROGRAM REPORT, INDIVIDUAL PLANT EVALUATION METHODOLOGY APPLIED TO THE ZION NUCLEAR GENERATING STATION, SUBMITTED TO: AIF/IDCOR PROGRAM, FEBRUARY 1986

13. DRAFT IDCOR PROGRAM REPORT, IDCOR/IPE REPORT, CALVERT CLIFFS NUCLEAR POWER PLANT UNIT 1, SUBMITTED TO IT CORP BY BG&E WITH LETTER DATED OCTOBER 20, 1986.

14. LETTER FROM T. P. SPEIS, NRC, TO A. BUHL, IDCOR WITH PRELIMINARY EVALUATION OF THE IDCOR IPEM, SEPTEMBER 9, 1986

15. LETTER FROM A. BUHL, IDCOR, TO T.P. SPEIS, NRC SUBJECT: IDCOR RESPONSE TO NRC COMMENTS ON THE INDIVIDUAL PLANT EVALUATION METHODOLOGY, DEC. 1986

16. LETTER FROM J.W. HICKMAN, SNL, TO M.D. HOUSTON, ACRS STAFF, TRANSMITTING COMMENTS FROM A REVIEW OF THE IDOCR IPEM, SEPTEMBER 22, 1986

17. LETTER FROM A. BUHL, IDCOR, TO W. KERR, ACRS, SUBJECT: IDCOR RESPONSES TO SNL COMMENTS ON THE IDCOR IPEM, OCTOBER 30, 1986