FUTURE DIRECTIONS

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OVERVIEW OF TOPICS

- SAFETY GOALS
- U.S. SEVERE ACCIDENT POLICY
- CODE DEVELOPMENT
- RESEARCH
- ANALYSES AND OPERATOR ACTION
- ACCIDENT MANAGEMENT
- LINKING PRA TO DETERMINISTIC ANALYSES

PRINCIPLE CONSIDERATIONS IN RISK GOALS

- SMALL INCREMENT IN EXISTING BACKGROUND OF RISKS
- PROTECT BOTH:
  - GENERAL POPULATION
  - NEARBY RESIDENTS
  - PLANT INVESTMENT
- CONSISTENT WITH RISK OF OTHER MAN-CAUSED ACTIVITIES
- COST-BENEFIT CONSIDERATION AFTER MINIMUM SAFETY LEVELS ACHIEVED
- GOAL PROPORTIONAL TO BENEFITS TO BE GAINED
- RECOGNITION OF INVOLUNTARY VS VOLUNTARY RISK AVERSION

NUCLEAR REGULATORY COMMISSION FOR TRAIL USE

PRIMARY GOALS

- PROMPT FATALITY
  - $<$0.1% OF BACKGROUND ACCIDENT RISK (5×10^{-4}/PERSON-YR) AVERAGED OVER FIRST MILE OFFSITE
- LATENT CANCER FATALITY RISK
  - $<$0.1% OF BACKGROUND CANCER RISK (2×10^{-3}/PERSON-YR) AVERAGED OVER 10 MILES OFFSITE

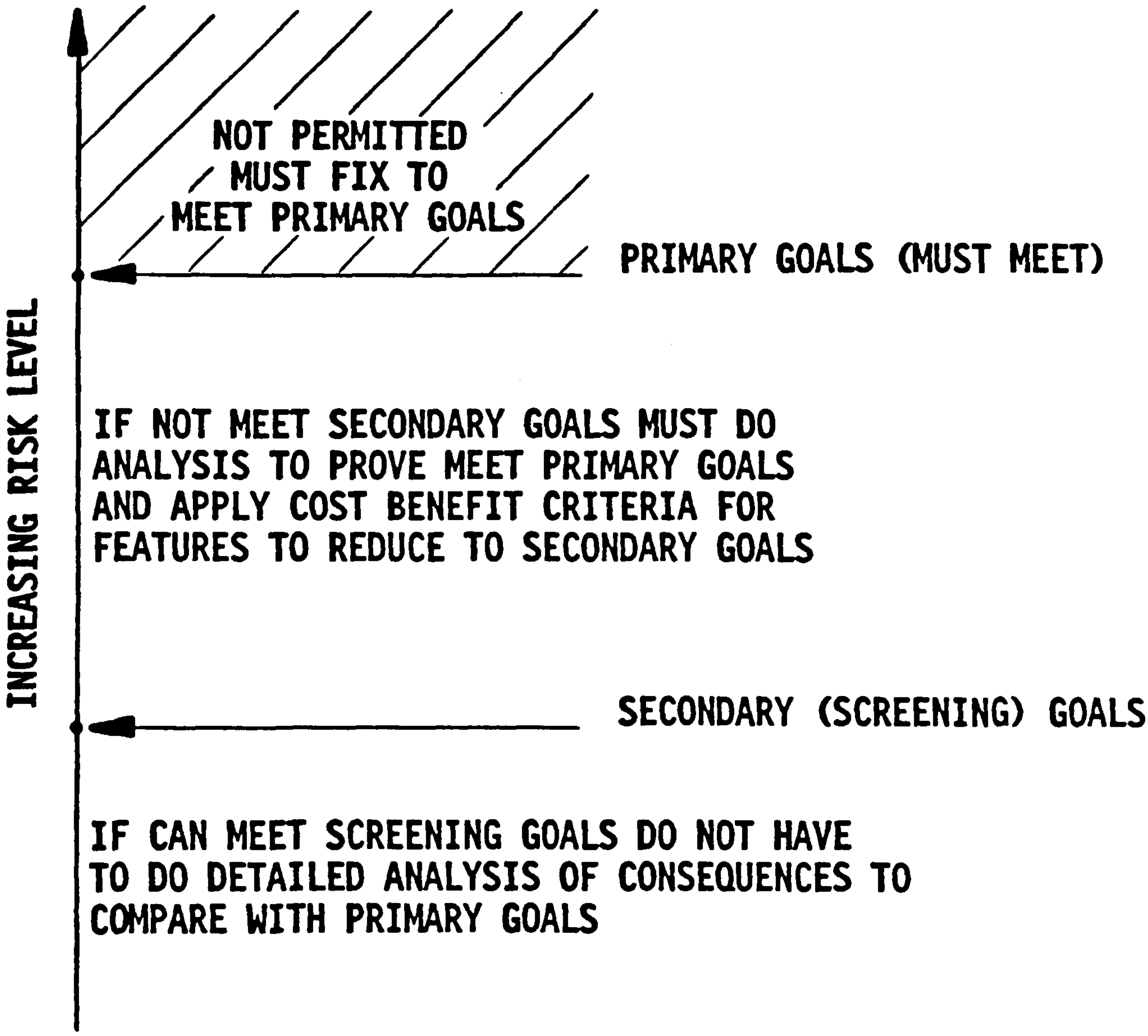
SECONDARY – SCREENING GOALS

- PLANT PERFORMANCE
  - $<10^{-4}$/YR COREMELT
- CONTAINMENT PERFORMANCE
  - $<$10-6/YR LARGE RELEASE

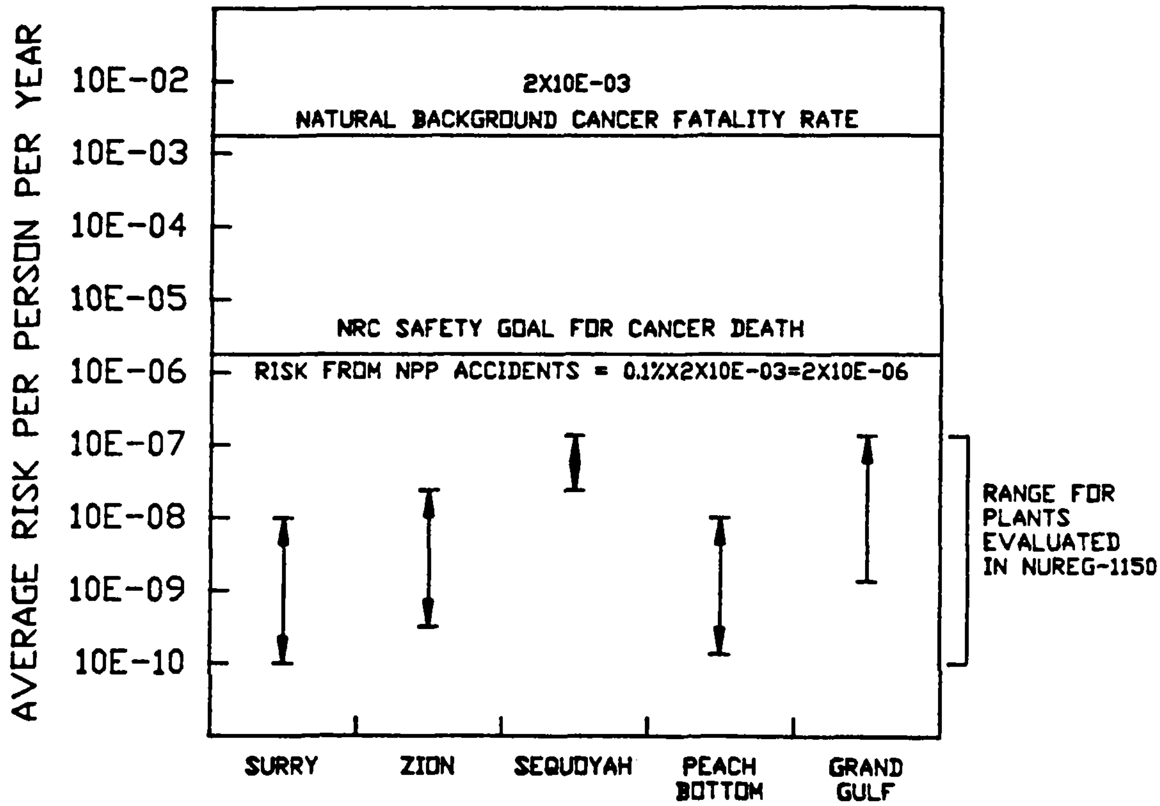
COST BENEFIT CRITERION

- SHOULD SPEND UP TO $1000/MAN-REM AVERTED (EQUIVALENT TO $10 MILLION PER LIFE SAVED)

GENERAL NATURE OF HIERARCHY OF PROPOSED RISK GOALS



EXAMPLE OF SAFETY GOAL COMPARISON



COMPARISON OF ESTIMATED CANCER FATALITY RISKS TO SAFETY GOAL

EUROPEAN SAFETY GOALS

- BEING DEVELOPED SEPARATELY BY EACH COUNTRY --- NO EEC PROGRAM

- EXAMPLES

  ITALY – NEW PLANTS

  CORE MELT FREQUENCY $<10^{-5}$ PER YEAR
  RELEASE $>0.1\%$ OF CORE $<10^{-6}$ PER YEAR

O.K. – NEW PLANTS

  CORE MELT FREQUENCY $<10^{-5}$ PER YEAR

DESIGN OBJECTIVES FOR NEW REACTORS
U.S.A. INDUSTRY

- MEAN FREQUENCY OF OCCURRENCE FOR OFFSITE DOSES >25 REM WHOLE BODY OF $<10^{-6}$ PER YEAR

- MEAN FREQUENCY OF CORE DAMAGE OF $<10^{-5}$ PER YEAR

SEVERE ACCIDENT POLICY STATEMENT

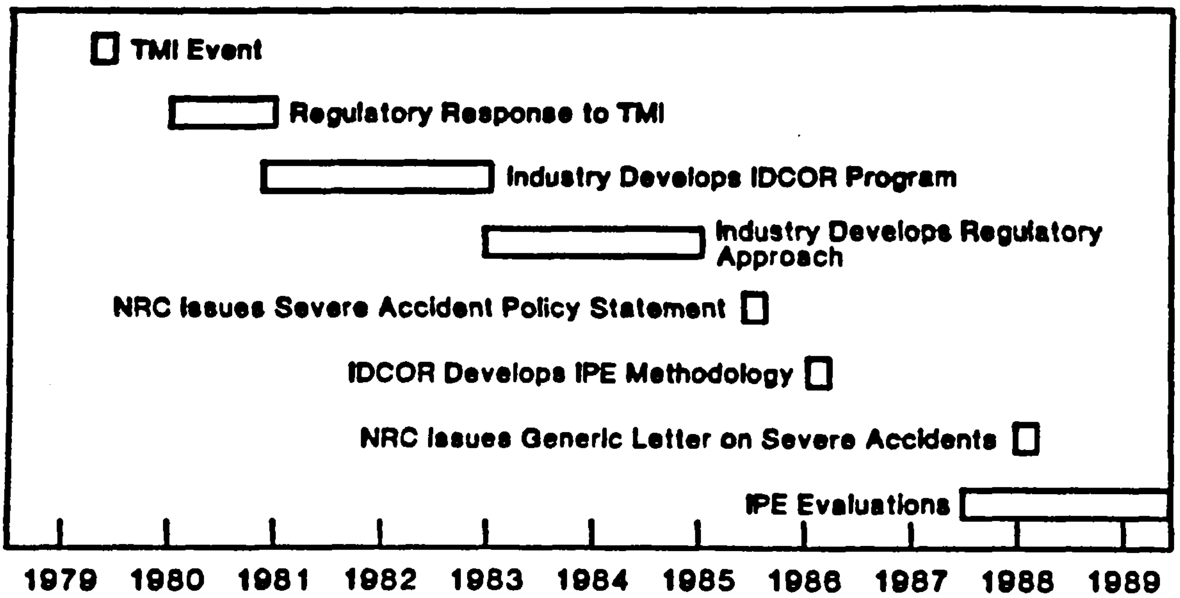
- NRC CONCLUDED THAT ON THE BASIS OF CURRENT INFORMATION EXISTING PLANTS POSE NO UNDUE RISK TO THE PUBLIC HEALTH AND SAFETY

- NO IMMEDIATE ACTION WAS TAKEN BY NRC ON GENERIC RULEMAKING OR OTHER REGULATORY CHANGES FOR EXISTING PLANTS

- EACH NUCLEAR UTILITY IS REQUIRED TO PERFORM A LIMITED SCOPE OF INDIVIDUAL PLANT EVALUATION (IPE)

SEVERE ACCIDENT POLICY STATEMENT

"A Decade of Evolution"



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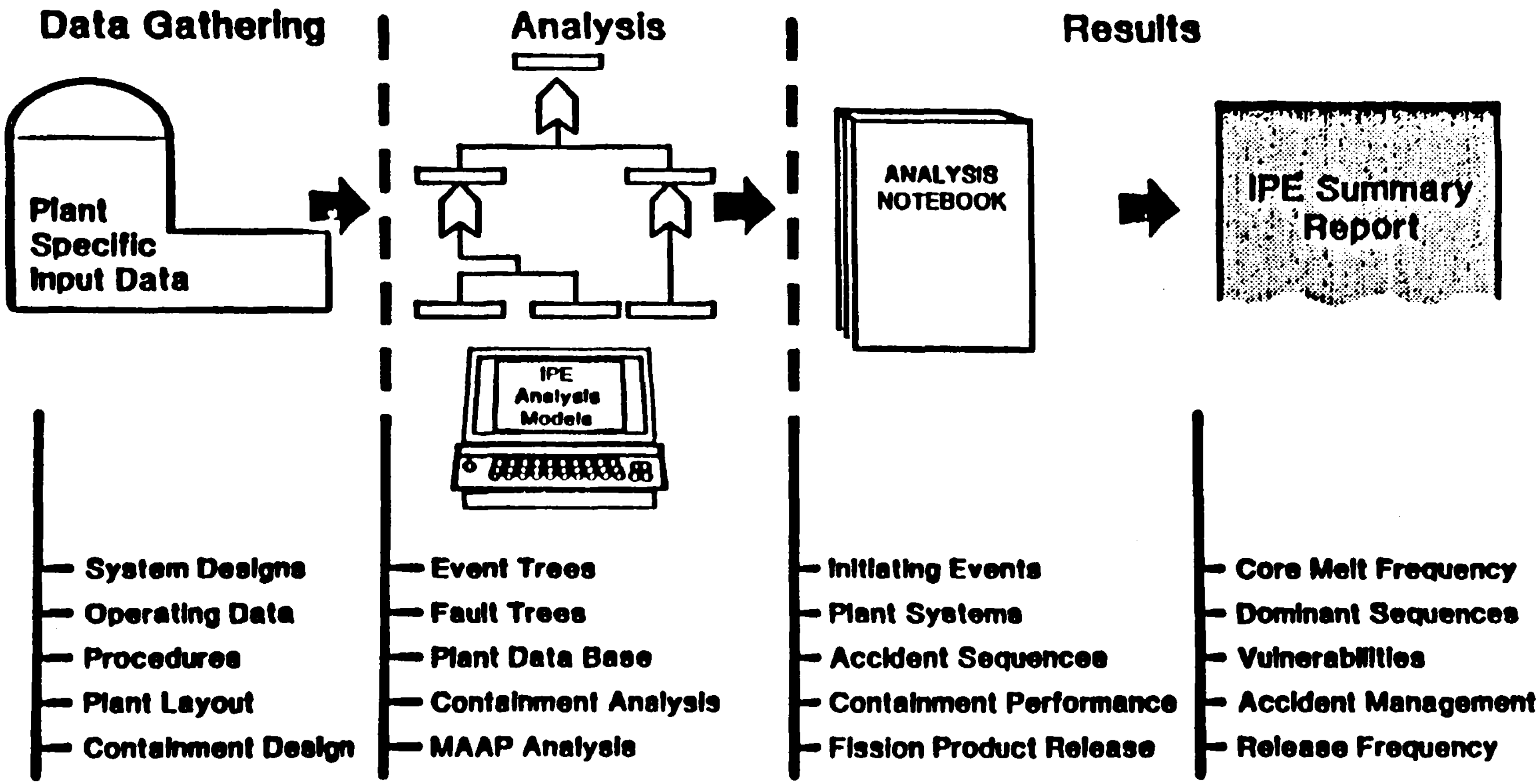
ELEMENTS OF AN IPE

PHASE I: SYSTEM ANALYSIS

- DETERMINE EXPECTED FREQUENCY OF CORE DAMAGE

PHASE II: SOURCE TERM ANALYSIS

- DETERMINE CONTAINMENT PERFORMANCE AND FISSION PRODUCT RETENTION CAPABILITY


Data Gathering
Analysis
Results
Plant
Specific
Input Data
IPE
Analysis
Models
ANALYSIS
NOTEBOOK
IPE Summary
Report
System Designs
Operating Data
Procedures
Plant Layout
Containment Design
Event Trees
Fault Trees
Plant Data Base
Containment Analysis
MAAP Analysis
Initiating Events
Plant Systems
Accident Sequences
Containment Performance
Fission Product Release
Core Melt Frequency
Dominant Sequences
Vulnerabilities
Accident Management
Release Frequency

INDIVIDUAL PLANT EVALUATION REGULATORY STATUS

- NRC EXPECTED TO ISSUE SEVERE ACCIDENT POLICY STATEMENT GENERIC LETTER IN JULY 1988

- GENERIC LETTER WILL STATE:

(1) ALL UTILITIES MUST PERFORM LIMITED SCOPE PROBABILISTIC SAFETY STUDIES

(A) IPE METHODOLOGY
- STANDARDIZED, OPTIMIZED AND PREAPPROVED

(B) LEVEL 1 PRA STUDY PLUS CONTAINMENT EVALUATION

(C) OTHER PRA METHOD WITH NRC APPROVAL

(2) UTILITIES MUST SUBMIT IPE SCHEDULE TO THE NRC WITHIN 120 DAYS

(3) UTILITIES MUST COMPLETE IPE'S WITHIN 30 MONTHS

GENERIC LETTER APPENDICES

- GUIDANCE ON EXAMINATION OF CONTAINMENT SYSTEMS PERFORMANCE
  - A REPORT, "ASSESSMENT OF SEVERE ACCIDENT PREVENTION AND MITIGATION FEATURES: MARK I CONTAINMENT DESIGN" OR AN ANALOGOUS DOCUMENT APPROPRIATE TO THE MAJOR CONTAINMENT TYPES
- SEQUENCE SELECTION CRITERIA – TO SELECT ACCIDENT SEQUENCES SIGNIFICANT ENOUGH FOR FURTHER CONSIDERATION
- PLANT DESIGN GUIDELINES – TO ESTABLISH THRESHOLD RISK VALUES, BASED ON COMMISSION'S SAFETY GOAL POLICY, FOR USE IN FINAL EVALUATION OF IPE'S
- ELEMENTS OF AN ACCEPTABLE ACCIDENT MANAGEMENT PLAN
- DOCUMENTATION REQUIREMENTS
- DECAY HEAT REMOVAL VULNERABILITY INSIGHTS

NRC IPE REVIEW PROCESS

- 120 DAYS TO RESPOND TO IPE GENERIC LETTER WITH IMPLEMENTATION SCHEDULE

- WHATEVER OPTION IS SELECTED, NRC RESPONSE TO SUBMITTALS IS INTENDED TO BE RESPONSIBLY PROMPT

- WORKSHOPS AFTER ISSUANCE OF IPE GENERIC LETTER

- NRC WILL PREPARE AN IPE REVIEW DOCUMENT OF THE STAFF AND CONTRACTOR REVIEWERS TO INCLUDE:

  - ARCS OF REVIEW
  - DOCUMENTATION OF ADEQUACY OF IPE RESULTS
  - INTERPRETATION OF THE RESULTS
  - ACTION LEVELS
  - SAMPLE EVALUATIONS

- IPE REVIEW DOCUMENT WILL BE MADE AVAILABLE TO ALL UTILITIES SHORTLY AFTER THE GENERIC LETTER

CODE DEVELOPMENT

- MAAP SEVERE ACCIDENT CODE
  - EPRI ORGANIZED "MAAP USERS GROUP"
    - -- INTERACTION AMONGST MAAP USERS
    - -- CONTINUED DEVELOPMENT
  - W/FAI CONTINUED DEVELOPMENT
- NRC SEVERE ACCIDENT CODES
  - STCP
    - -- LIMITED FUTURE DEVELOPMENT
  - MELCOR/MELPROG
    - -- LARGE SCALE EFFORT FOR DEVELOPMENT/ VERIFICATION

RESEARCH AREAS

- CORE-CONCRETE INTERACTION
- FISSION PRODUCT PHENOMENA
- DIRECT CONTAINMENT HEATING
- FUEL COOLANT INTERACTIONS
- HYDROGEN BEHAVIOR
- ACCIDENT MITIGATION STRATEGIES

ANALYSES AND OPERATOR ACTIONS

- CURRENT SEVERE ACCIDENT ANALYSES ONLY MODEL OPERATOR ACTIONS WHICH CAN PREVENT CORE DAMAGE. EXAMPLES INCLUDE:

  - SWITCHOVER TO RECIRCULATION CORE COOLING
  - FEED AND BLEED CORE COOLING

- OTHER ACTIONS ARE SPECIFIED IN PLANT EMERGENCY PROCEDURES WHICH CAN IMPACT THE SEVERE ACCIDENT PROGRESSION

  - TIMING OF EVENTS
  - OCCURRENCE OF VARIOUS PHENOMENA

  THESE ACTIONS HAVE A HIGH PROBABILITY OF SUCCESSFUL IMPLEMENTATION DUE TO THE TIME AT WHICH THEY ARE PRESCRIBED. THE MAJOR ACTIONS FOR PWRS INCLUDE:

  - REACTOR COOLANT DEPRESSURIZATION USING PRESSURIZER PORVS AND
  - REACTOR COOLANT COOLDOWN AND DEPRESSURIZATION USING STEAM GENERATORS

Surry Dominant Accident Sequences

SEQUENCE	FREQUENCY	CONTRIBUTION TO CORE MELT
T1(SL)-D1CF1	6.6E-06	26.4 %
S3D1	2.6E-06	10.4 %
T4JQ-H1	1.9E-06	7.6%
T4HQ-H1	1.6E-06	6.4 %
T1(LT)D1CF1	1.3E-06	5.2 %
T1(ST)D1CF1	1.3E-06	5.2 %
T1LP	1.1E-06	4.4 %
TKRD4	1.1E-06	4.4 %
EVENT-V	9.0E-07	3.6 %
S2H1	8.9E-07	3.6 %
T4JQ-H2	8.1E-07	3.2 %

NOTE: Based on total core melt frequency of 2.5 E-04 given in Table IV.9-2 of NUREG/CR-4550, Vol. 3.

Surry Dominant Accident Sequence Operator Actions

(TOP 10 EVENT SEQUENCES, EXCLUDING EVENT -V)

Plant Damage State	Sequence	System Failures	Operator Actions
SNNN	T1(SL)-D1CF1	Offsite A.C.; Onsite A.C.; RCP Seals; Non-recovery of A.C. in 1/2 hr. after seal LOCA	None
TYYB	S3D1	RCP Seal LOCA; HPSI	None
SYYB	T4JQ-H1	Loss of 480 VAC bus PORV isolation;LPSI recirc.	None
SYYB	T4HQ-H1	Loss of 480 VAC bus PORV isolation;LPSI recirc.	None
TNNN	T1L(LT)D1CF1	Offsite A.C.; Onsite A.C.; Batteries at 4 hr.; Non-recovery of A.C. in 3 hr. after battery depletion	None
TNNN	T1L(ST)D1CF1	Offsite A.C.; Onsite A.C.; AFW; Non-recovery of A.C. within 1/2 hr.	None
TYYB	T1LP	Offsite A.C.; AFW; PORVs	None
TYYB	TKRD4	Transient; RPS; Manual Scram; HPSI-Emerg. Boration	Failure to manually scram; failure to perform emerg. boration
SYYB	S2H1	Small LOCA;LPSI recirc.	None
SYYB	T4JQ-H2	Loss of 480 VAC bus PORV isolation; HPSI recirc.	None

NOTE:
* Taken from Table V.1-2 of NUREG/CR-4550, Vol. 3
* Operator actions refer to successful or failed operator actions in the dominant sequence cutsets from Section V of NUREG/CR-4550, Vol. 3

STATUS OF IMPORTANT PARAMETERS
FOR THE CASE OF OPERATOR ACTIONS / NO OPERATOR ACTIONS

	RCS PRESSURE	RCS PRESSURE	SG LEVEL	SG PRESSURE	ACCUM. DISCHARGE
	@ CORE UNC.	@ VF	@ CORE UNC.	@ CORE UNC.	
T1(SL)-D1CF1	M/H	M/H	D/D	H/H	Prior to Unc.
S3D1	L/H	L/H	N/N	L/H	Prior to Unc.
T4JQ-H1	M/H	L/H	N/N	L/H	Prior to Unc.
T4HQ-H1	M/H	L/H	N/N	L/H	Prior to Unc.
T1(LT)D1CF1	H/H	H/H	D/D	H/H	Prior to Unc.
T1(ST)D1CF1	H/H	H/H	D/D	H/H	At VF
T1LP	H/H	L/H	D/D	H/H	At Core Heatup
TKRD4	H/H	H/H	N/N	H/H	At VF
S2H1	L/M	L/M	N/N	L/H	Prior to Unc.
T4JQ-H2	M/H	L/H	N/N	L/H	Prior to Unc.

LEGEND:

High	High	Normal	High	Prior to core uncovery
Medium	Medium	Dry	Medium	At core heatup
Low	Low		Low	At reactor vessel failure

SEVERE ACCIDENT MANAGEMENT SCOPE

- ORGANIZATION FOR DECISION-MAKING
- INSTRUMENTATION REQUIRED FOR SEVERE ACCIDENT MANAGEMENT
- EMERGENCY OPERATING PROCEDURES FOR SEVERE ACCIDENT MITIGATION:
  - ONSITE DAMAGE
  - OFFSITE CONSEQUENCES
- TRAINING OF OPERATING STAFF

SEVERE ACCIDENT MANAGEMENT ORGANIZATION

- PRESENT UTILITY ORGANIZATION SPECIFIES RESPONSIBILITIES FOR:
  - PLANT CONTROL ROOM
  - EMERGENCY OPERATIONS FACILITY
  - TECHNICAL SUPPORT CENTER
  - UTILITY HEADQUARTERS

- PROPOSED MODIFICATIONS INCLUDE:
  - TSC STAFF GUIDANCE, TRAINING AND DRILLS
  - MONITOR STRATEGIES BY CONTROL ROOM
  - ANTICIPATE FURTHER DEGRADATION OF SYSTEMS
  - RE-ESTABLISH REDUNDANCY, DIVERSITY, INDEPENDENCE
  - IMPLEMENT SEVERE ACCIDENT STRATEGIES

SEVERE ACCIDENT MANAGEMENT INSTRUMENTATION

- PRESENT POST ACCIDENT MONITORING AND SAMPLING SYSTEM
  - APPLIES TO ACCIDENTS UP TO, BUT NOT INCLUDING, CORE DAMAGE EVENTS
  - REQUIRED TO BE ENVIRONMENTALLY QUALIFIED FOR APPLICABLE CONDITIONS

- SEVERE ACCIDENT INSTRUMENTATION REQUIRED IN ORDER TO:
  - MONITOR STATUS OF PLANT IN SEVERE ACCIDENT CONDITION
  - IMPLEMENT SEVERE ACCIDENT PROCEDURES OR MITIGATING ACTIONS

- INVESTIGATIONS REQUIRED TO:
  - DETERMINE LIMITED SET OF INSTRUMENTATION REQUIRED
  - DEFINE MINIMUM EQUIPMENT QUALIFICATION REQUIREMENTS

SEVERE ACCIDENT PROCEDURES

- PLANT OPERATING STAFF GUIDANCE FOR DEALING WITH A SEVERE ACCIDENT IN WHICH DAMAGE TO THE REACTOR CORE HAS OCCURRED.

- SEVERE ACCIDENT PROCEDURES ARE ALREADY BEING DEVELOPED IN EUROPE AND JAPAN

  - FRENCH U-PROCEDURES
  - WESTINGHOUSE PROJECT FOR RINGHALS
  - MITSUBISHI EFFORTS IN JAPAN

- U.S. ACTIVITIES

  - BWR OWNERS GROUP EVALUATION OF EOPS FOR SEVERE ACCIDENTS
  - BOSTON EDISON DEVELOPMENT OF TSC GUIDANCE DOCUMENT
  - SEABROOK INVESTIGATION OF STATION BLACKOUT

- PROPOSED FUTURE REQUIREMENT IN THE U.S.A. AS PART OF IMPLEMENTATION OF SEVERE ACCIDENT POLICY

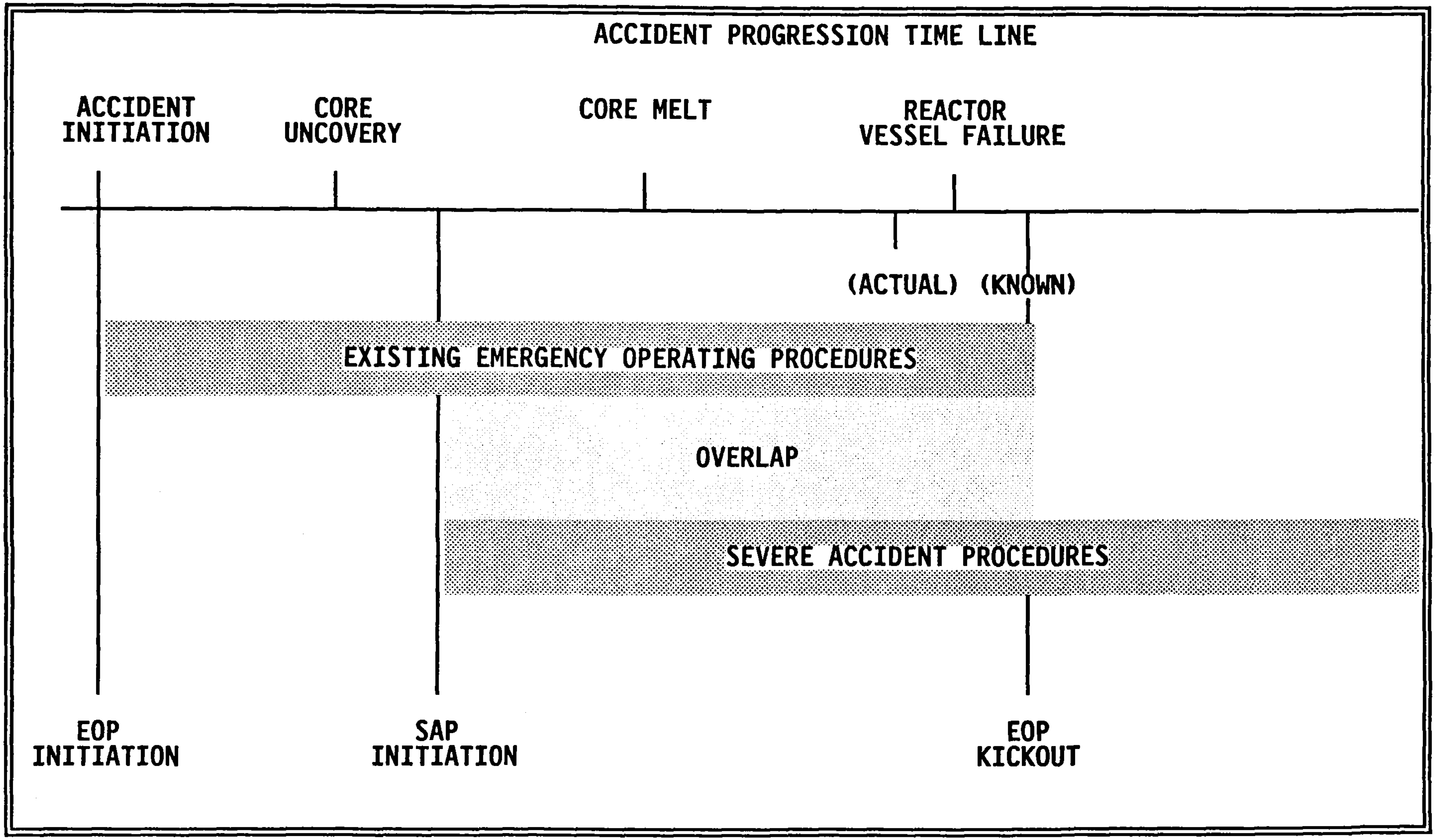
SEVERE ACCIDENT PROCEDURES

- THE INITIAL STAGES OF SEVERE ACCIDENTS ARE ADDRESSED BY THE PRESENT ERG/FRG FRAMEWORK
  - ERG/FRG COVER ACTIONS UP TO CORE DAMAGE
  - ERG/FRG STRUCTURE ASSUMES SOME SAFEGUARD SYSTEMS ARE AVAILABLE

- SEVERE ACCIDENT PROCEDURES WOULD CONTINUE BEYOND THE PRESENT ERG/FRG FRAMEWORK
  - INITIATING WOULD BE FROM PRESENT ERG/FRG
  - ACTION WOULD EXTEND APPLICABILITY TO ALL CREDIBLE POTENTIAL ACCIDENT OUTCOMES
  - ACTIONS WOULD COVER ALL POTENTIAL PLANT EQUIPMENT STATES
  - PROCEDURES WOULD BE FUNCTIONALLY ORIENTED (vs. EVENT ORIENTED)

- SEVERE ACCIDENT PROCEDURES IMPLEMENTATION RESPONSIBILITY (CONTROL ROOM vs. EOF vs. TSC) TO BE ADDRESSED:
  - INTERFACES
  - TECHNICAL EVALUATION REQUIRED
  - DOWN-SIDE RISKS

INTERFACE OF PRESENT EMERGENCY PROCEDURES
WITH SEVERE ACCIDENT PROCEDURES



OPERATING STAFF SEVERE ACCIDENT MANAGEMENT TRAINING

- PRESENT SIMULATORS ARE NOT CAPABLE OF REPRESENTING CORE DAMAGE ACCIDENTS

- OPTIONS FOR TRAINING ARE LIMITED

  - TRAINING SIMILAR TO PRESENT EMERGENCY DRILLS (PRE-ANALYZED SCENARIOS)

  - TRAINING WITH PC-BASED SIMULATOR SUCH AS MAAP-GRAAPH

  - TRAINING USING PRA RESULTS IN CLASSROOM SESSIONS

INTEGRATION OF SEVERE ACCIDENT AND DETERMINISTIC APPROACHES

- The accident at Three-Mile Island forced a re-evaluation of the "design basis"
  - Peak clad temperatures > 2200°F
  - Hydrogen generation > 1%
  - Coolable geometry was violated

  All violated LWR design basis
- However
  - Core remained coolable
  - Containment remained intact
  - Core was put into safe shutdown mode

  Design was adequate to handle accidents beyond design basis
- However to the NRC/Industry, it was clear that accidents within the design basis could lead to severe accidents beyond the design basis
  - Needs were identified for severe accident analysis methods
  - For procedures/operations to prevent severe accidents
  - Severe accident sequence required study
- The design basis for the LWR was extended

LWR DESIGN BASIS

- The US LWR design basis has been broadened to include:
  - Hydrogen Rule (containment must withstand burning of H_2 from 75% Zirc/water Reaction)
  - Containment ultimate strength studies
  - Alternate methods of core coolability, feed and bleed, etc.
  - Sympomatic based emergency response guidelines
  - Offsite emergency planning requirements

- As a result
  - Individual plant evaluations are being performed (IP) to evaluate what design basis accidents could lead to severe core damage
  - Analytical methods (MAAP)/Compact) had been developed to evaluate LWR performance for severe accident sequences

TRANSIENTS WHICH ARE USED TO ESTABLISH FUNCTIONAL REQUIREMENTS FOR SAFETY GRADE EQUIPMENT

- LOSS OF FLOW
- BORON DILUTION
- CONTROL ROD EJECTION
- ROD WITHDRAWAL AT POWER
- DROPPED ROD
- STEAMLINE BREAK
- FEEDLINE BREAK
- LOSS OF NORMAL FEEDWATER
- STEAM GENERATOR TUBE RUPTURE

NON-LOCA TRANSIENTS

↓

DNB LIMIT
FUEL CENTERLINE MELT
RCS OVERPRESSURE
COOLABLE GEOMETRY

SMALL BREAK
LARGE BREAK LOCA

LOCA TRANSIENTS

↓

PEAK KW/M LIMIT

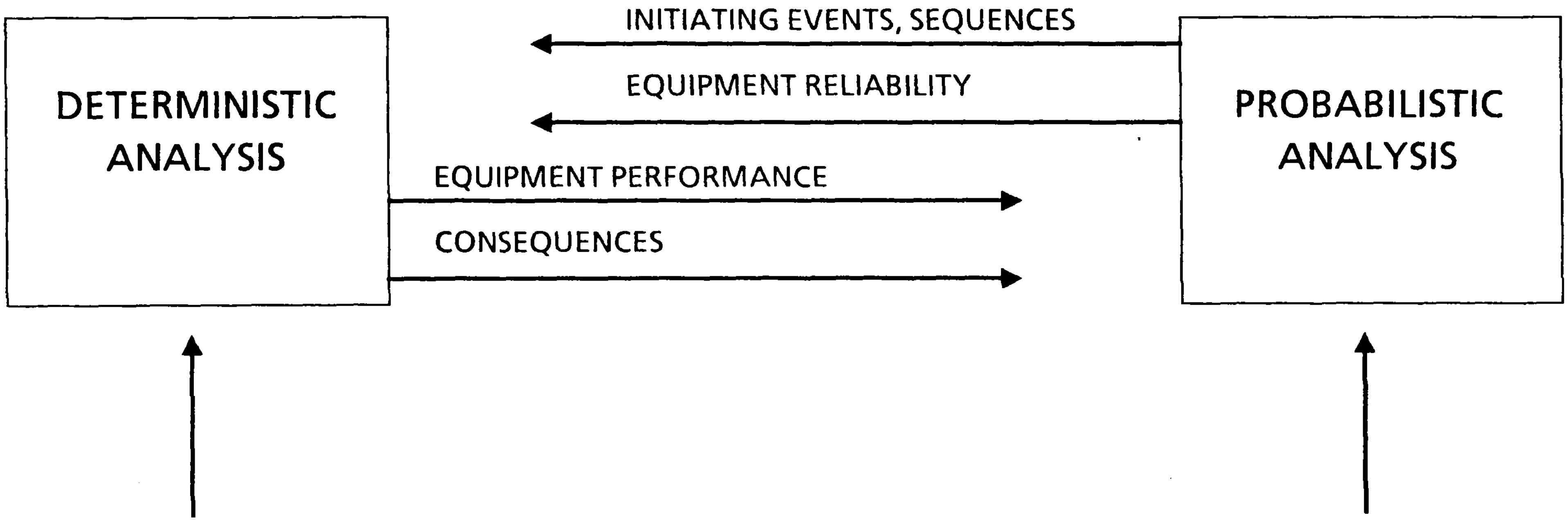
TOTAL LOSS OF ALL EQUIPMENT
LOSS OF ALL CORE COOLING

SEVERE FUEL DAMAGE,
DEGRADED CORE

↓

ULTIMATE CONTAINMENT
INTEGRITY

PLANT SAFETY EVALUATION/VERIFICATION



DETAILED THERMAL/HYDRAULIC ANALYSIS

- NON-LOCA
- LOCA
- SEVERE CORE DAMAGE

PLANT DESIGN CHARACTERISTICS

- EQUIPMENT RELIABILITY
- OPERATING PROCEDURES
- FAILURE PATHS