UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

MEMORANDUM TO: Noel Dudley, Senior Staff Engineer
ACRS/ACNW

FROM: Dr. F. Peter Ford, Chairman
Materials and Metallurgy Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS MATERIALS AND METALLURGY SUBCOMMITTEE MEETING CONCERNING THE PRESSURIZED THERMAL SHOCK (PTS)TECHNICAL BASIS REEVALUATION PROJECT, JANUARY 15-16, 2002 – ROCKVILLE, MARYLAND

I hereby certify that, to the best of my knowledge and belief, the minutes of the subject meeting issued on January 28, 2002, are an accurate record of the proceedings for the meeting.

Peter Ford

Dr. F. Peter Ford, Chairman
Materials and Metallurgy Subcommittee

February 7th 2002

Date

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

February 8 , 2002

MEMORANDUM TO: ACRS Members

Noel Dudley

FROM: Noel Dudley, Senior Staff Engineer
ACRS\ACNW

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON MATERIALS AND METALLURGY CONCERNING THE PRESSURIZED THERMAL SHOCK (PTS) TECHNICAL BASIS REEVALUATION PROJECT, JANUARY 15-16, 2002 – ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued on January 28, 2002, have been certified as the official record of the proceedings of that meeting. A copy of the certified minutes is attached.

Attachment: As stated

cc: Technical Support Branch
Operations Support Branch (3 copies)

cc via e-mail:
J. Larkins
S. Bahadur
ACRS Fellows and Technical Staff
E. Barnard

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D.C. 20555-0001

January 28, 2002

MEMORANDUM TO: Dr. F. Peter Ford, Chairman
Materials and Metallurgy Subcommittee

Noel Dudley

FROM: Noel Dudley, Senior Staff Engineer
ACRS/ACNW

SUBJECT: WORKING COPY OF THE MINUTES OF THE ACRS MATERIALS AND METALLURGY SUBCOMMITTEE MEETING CONCERNING THE PRESSURIZED THERMAL SHOCK (PTS) TECHNICAL BASIS REEVALUATION PROJECT, JANUARY 15-16, 2002 – ROCKVILLE, MARYLAND

A working copy of the minutes for the subject meeting is attached for your review. I would appreciate your review and comment as soon as possible. Copies are being sent to the Materials and Metallurgy Subcommittee members for information and/or review.

Attachment: As stated

cc: W. Shack
M. Bonaca

cc via E-Mail:
J. Larkins
S. Bahadur

CERTIFIED

ML 02 029 0003

Issued: 1/28/02
Certified: 2/7/02

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF SUBCOMMITTEE MEETING ON
MATERIALS AND METALLURGY
PTS SCREENING CRITERION REEVALUATION PROJECT INITIAL RESULTS
JANUARY 15-16, 2002
ROCKVILLE, MARYLAND

The ACRS Subcommittee on Materials and Metallurgy met on January 15-16, 2002, to hold discussions with representatives of the NRC staff concerning the initial results of the Pressurized Thermal Shock (PTS) Technical Basis Reevaluation Project's Fracture Analysis of Vessels – Oak Ridge (FAVOR) code. The entire meeting was open to public attendance. Mr. Noel Dudley was the cognizant ACRS staff engineer for this meeting. The meeting was convened at 8:30 a.m. and recessed at 4:30 p.m. on January 15, 2002, and reconvened at 8:30 a.m. and adjourned at 11:50 a.m. on January 16, 2002.

ATTENDEES

ACRS

P. Ford, Chairman
W. Shack, Vice Chairman
M. Bonaca, Member
N. Dudley, ACRS Staff

NRC REPRESENTATIVES

M. Mayfield, RES
J. Rosenthal, RES
E. Hackett, RES
S. Malik, RES
L. Abrams, RES
B. Arcieri, Information System Laboratory, Inc.
D. Bessette, RES
H. Woods, RES
D. Kolaczkowski, Sandia National Laboratory
T. Dickson, ORNL
A. Mosleh, University of Maryland
Y. Chang, University of Maryland

There were no written comments or requests for time to make oral statements received from members of the public. Approximately 12 members of the public attended the meeting. A list of meeting attendees is available in the ACRS office files.

INTRODUCTION

Dr. F. Peter Ford, Chairman of the Materials and Metallurgy Subcommittee, stated that the purpose of the meeting was to review the status of the PTS Technical Basis Reevaluation Project. He noted that the staff would present the initial results of the reactor vessel failure frequency of Oconee Unit 1 as calculated by the FAVOR code. Dr. Ford summarized ACRS past reviews of the Reevaluation Project and associated SECY papers. He called on Mr. Michael Mayfield, RES, to introduce the presentation.

STATUS OF THE PTS TECHNICAL BASIS REEVALUATION PROJECT

Mr. Michael Mayfield, RES, thanked the Subcommittee for the time and effort it had expended in reviewing the Reevaluation Project. He explained that the staff, at the request of Dr. George Apostolakis, ACRS, was prepared to provide a detailed example of how the FAVOR code models and treats uncertainties when determining reactor vessel fracture and failure frequencies.

Dr. Edwin Hackett, RES, presented the status of the Reevaluation Project including the approach developed to assess the PTS risk; the inputs and models developed by the probabilistic risk assessment (PRA), thermal-hydraulics, and probabilistic fracture mechanics (PFM) working groups; and recent accomplishments. See attached viewgraph (VG) - 4. He explained that further work remains to be done in finishing the internal event analysis of four plants, completing external event risk contributions, and integrating the results in risk criteria. Dr. Hackett stated that based on the results of the FAVOR code, the risk of through wall cracking in the Oconee Unit 1pressure vessel would be 4 orders of magnitude less than the regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Report for Pressurized Water Reactors," acceptance value of 5×10^{-6} per year even after 60 years of extended operation. screening criterion after 40 years of operation. See attached VG-6. He also summarized the changes between the FAVOR code analysis and the1980's analysis that resulted in this increased margin from the screening criterion. See attached VG-7.

The Subcommittee members and the staff discussed the following:

- the need to consider the effects of a containment in future analyses,
- dealing with plants that are approaching the screening criterion,
- calculating the reference transition temperature evaluated at the end of life fluence (RT_{PTS}) at the end of an additional 20 year license renewal period,
- identifying uncertainties in the FAVOR code results,
- crediting operator actions in PTS scenarios, and
- determining the appropriated level of statistical confidence to use in the FAVOR code.

COMPUTATIONAL MODELS AND UNCERTAINTY

Dr. Mark Kirk, RES, presented the guidelines for the Reevaluation Project and the intended materials screening criteria. He explained the interaction and integration among the probabilistic risk assessment (PRA), the thermal-hydraulic, and probabilistic fracture mechanics (PFM) analyses. See attached VG-11. He described the concepts of model development and uncertainty treatment, and how the staff implemented these concepts during the Reevaluation Project.

The Subcommittee members and the staff discussed the data used in developing the models and the feedback loops between the three analyses. They also discussed the lessons learned concerning managing the extensive Reevaluation Project, which included input from several RES branches, national laboratories, NRC contractors, licensees, and the Nuclear Energy Institute.

PRA: Mr. Alan Kolaczkowski, Sandia National Laboratory, described the conceptual model for the treatment of uncertainties through the three analyses. He presented the constraints and fundamental assumptions the staff used in its PRAs and human reliability assessments. Mr. Kolaczkowski provided an overview of scenario modeling, the iterative modeling process, and the treatment of uncertainties.

In response to a question by Dr. Ford, the staff explained that aleatory uncertainties are added in the last step of the process, while epistemic uncertainties are assigned to the input parameters and propagated through the process. The Subcommittee members and the staff discussed the following:

- credit given to operator actions for mitigating events,
- effect of plant differences on the probability that operators could mitigate events,
- estimates and uncertainties associated with human error,
- need for licensees to measure human reliability,
- types of scenarios that are screened out in the Reevaluation Project,
- intermediate break size loss-of-coolant accidents as dominate risk scenarios, and
- introduction of time dependency in the scenarios.

PFM: Dr. Kirk provided additional details concerning the probabilistic fracture mechanics analysis preformed by the FAVOR code. He explained how the results from the stress intensity factor model and the fracture toughness model are compared to determine the probabilities of crack initiation and through-wall cracking. See attached VG-62. Dr. Kirk presented the data that was used to develop the fracture toughness model and the embrittlement correlations to predict the fracture toughness of irradiation embrittled reactor vessel materials. He described the constraints and fundamental assumptions associated with these models. See attached VG-64. He explained the process for model building and uncertainty characterizations.

Dr. Kirk provided details of how the following models are used to derive the probability of crack initiation of the probability of thru-wall cracking:

- Initiation Fracture Toughness best estimate model: See attached VG-69.

- Transition Temperature Model: derives the reference transition temperature (RT_{NDT}) along with associated uncertainties. See attached VG-70.

- Initiation Fracture Toughness Model: derives the initiating fracture toughness (K_{Ic}) along with associated uncertainties. See attached VG-72.

- Irradiation Shift Model: derives toughness transition temperature (ΔT_0) along with the associated epistemic uncertainties. See attached VG-77.

- Arrest Fracture Toughness Model: derives the arrest fracture toughness (K_{Ia}) along with associated uncertainties. See attached VG-82.

Dr. Kirk described how the charpy V-notch energy test specimen transition temperature (ΔT_{30}), Master Curve toughness transition temperature shift (ΔT_0), and associated uncertainties are used to derive the irradiated shift for K_{Ic}. Similarly, the arrest temperature (ΔT_{ARREST}), along with associated uncertainties, is used to derive the irradiated shift for K_{Ia} .

Dr. Kirk described the assumptions and process used to build the flaw distribution model. He explained how the flaw model uses flaw data obtained from non-destructive testing, destructive testing, and expert judgement processes. The flow data was used to derive the density, size, orientation, and location of fabrication-induced flaws in welds, plates, and cladding materials in the reactor pressure vessels' beltline region along with the associated epistemic uncertainties in those flaw distributions.

The Subcommittee members and the staff discussed the following:

- peer reviews of the FAVOR code,
- validation of the reactor pressure vessel fluence at 40 and 60 years,
- grouping the aleatory and epistemic uncertainties instead of treating them separately,
- use of the Master Curve and temperatures to determine K_{Ic} ,
- difference between fitting data to RT_{NDT} or T_0 to determine K_{Ic} ,
- validation of the ΔT_{30} model when it is fitted to a small set of empirical data,
- validation of fluence across the reactor pressure vessel (RPV) wall,
- need to reevaluated the RPV wall 1/4 T flaw on the basis of the fluence attenuation,
- assumptions concerning the probability of detection of flaws, and
- the Monte Carlo sample size for flaw distributions.

THERMAL HYDRAULICS: Dr. Ali Mosleh, University of Maryland, presented the use of the thermal-hydraulic model results, with associated uncertainties, as inputs to the FAVOR code. He described the constraints and assumptions used in relationship to the RELAP5 code. He explained the development the RELAP 5 code and the thermal-hydraulic assessment process. He identified the equipment and systems modeled by the RELAP 5 code and noted that temperature and pressure were the important thermal-hydraulic parameters.

Dr. Mosleh described the Oconee PTS Event Classification Matrix, the development of PTS scenarios, and the derivation of the probability of occurrence for each PTS event. He noted that 94 percent of the total probability of PTS events is contributed by intermediate break size LOCA scenarios. Dr. Mosleh identified the sources of uncertainty in developing the scenarios and in the use of the RELAP 5 code. He described how these uncertainties are incorporated in the FAVOR code. He presented the sensitivity testing of various parameters and explained how the effect of multiple sources of uncertainty were combined.

The Subcommittee members and the staff discussed the binning process and using experimental data to validate the RELAP 5.

INITIAL RESULTS FOR THE OCONEE, UNIT 1, REACTOR PRESSURE VESSEL

Dr. Kirk presented the initial results from the FAVOR code evaluation of the Oconee Unit 1 reactor pressure vessel fatigue and failure frequencies. He identified the PRA improvements that have been made since a similar analysis was performed in the 1980s. He described the improved mapping of reactor vessel embrittlement and the crediting of operator actions. He identified the contribution of different classes of events to the total vessel failure frequency and made the following observations:

- Dominant scenarios are initiated by primary system LOCAs.
- Realistic accounting for operator actions significantly mitigates the influence of secondary system events and results in reduced vessel fatigue and failure frequencies.
- The time of primary safety relief valve closure has significant influence on event severity.

RELAP 5 ASSESSMENT

In response to Subcommittee members' questions related to the validation and verification of the RELAP 5 code, Mr. David Bessette, RES, described the assessment of the code during the staff review of the AP600 design. He explained that the AP600 design is more challenging to simulate than existing plant designs because of the passive safety systems. During the review of the AP600, the staff evaluated the RELAP 5 code against experiential data from the ROSA facility. Some of the Oconee PTS accidents scenarios were included in this evaluation. Mr. Bessette presented a comparison of the RELAP 5 code predictions and the ROSA experimental data for temperature and pressure transients initiated by a LOCA. He identified the uncertainties in the results of RELAP 5 code runs for PTS scenarios and explained that some of the uncertainties have been addressed by sensitivity studies.

Mr. Jack Rosenthal, RES, presented additional staff activities that were used to validate the RELAP 5 code. He noted that the ACRS Thermal-Hydraulic Phenomena Subcommittee toured the Oregon State University (OSU) APEX facilities and reviewed the basis for using the one dimensional version of the RELAP code. He stated that the staff plans to issue an overview document that would identified the reports and papers associated with activities to validate the RELAP 5 code.

The Subcommittee members, though discussions with the staff, verified the following:

- Some Oconee PTS scenarios were evaluated during the AP600 review.
- The staff performed break flow sensitivities.
- OSU APEX facility test results were used to validate the RELAP 5 code.
- There are no major uncertainties associated with the results of the RELAP 5 code.
- The RELAP 5 code reasonably predicted the transients of actual operating events.

The Subcommittee members questioned whether the staff had completed sensitivity studies of how uncertainties in the RELAP 5 code affect the FAVOR code results.

CREDITING OPERATOR ACTIONS

Dr. Alan Kolaczkowski presented the steps in the emergency operating procedures (EOPs) that the operators would follow in response to various scenarios that could lead to a PTS event. He explained that, on the basis of operator training and the EOPs, there is high confidence that operators will isolate a faulted steam generator. In addition, Mr. Kolaczkowski described how an expert elicitation assessment provided histograms for the probability of operator errors.

EXAMPLE PROBLEM

Dr. Mosleh presented an example of how the vessel failure frequency was calculated for the dominant transient in the Oconee analysis. He explained how the transient was derived from the PRA event characterization and binning process. He described how the thermal-hydraulic estimates of pressure and temperature over time were developed and how the FAVOR code calculated the reactor vessel fracture and failure frequencies. During the presentation, Dr. Mosleh identified the uncertainties in the different variables and explained how these uncertainties were quantified or how engineering judgement was used to determined the treatment of the uncertainties.

The Subcommittee members and the staff discussed the following:

- whether the Monte Carlo process would identify the tails of PTS event distributions,
- whether the assumed flaw size was realistic,
- fluence values assumed at a crack tip,
- use of the maximum value for the conditional probability of thru-wall cracking instead of a cumulative probability,
- assumptions concerning how a crack becomes thru-wall, and
- the numerical difference between the probability of crack initiation and the probability of thru-wall cracking.

SUBCOMMITTEE COMMENTS, CONCERNS, AND RECOMMENDATIONS

Dr. William Shack stated that the PTS Reevaluation Program was impressive.

Dr. Mario Bonaca stated that the initial results of the Reevaluation Project provide convincing evidence that PTS events happen less frequently than the staff previously predicted and that operators perform better than the staff had previously assumed.

Dr. Peter Ford opined that given the importance of PTS events, a confidence level greater than 95 percent should be considered. He stated that the Reevaluation Project was impressive and created unique program and managerial challenges. Given that there is no experimental data against which to validate the derived vessel failure frequency, he questioned the verification and validation of the RELAP 5 code and interactive models such as peer reviews.

STAFF AND INDUSTRY COMMITMENTS

None.

SUBCOMMITTEE DECISIONS

The Subcommittee requested that the staff present the following information at the February 7, 2002 ACRS meeting session on the Reevaluation Project.

- Overview of the PTS Technical Basis Reevaluation Project.
- Initial results for the Oconee Unit 1 reactor pressure vessel failure frequency.
- Example of how the vessel fracture and failure frequencies are calculated.

FOLLOW-UP ACTIONS

None

PRESENTATION SLIDES AND HANDOUTS PROVIDED DURING THE MEETING

The presentation slides and handouts used during the meeting are available in the ACRS office files or as attachments to the transcript.

BACKGROUND MATERIAL PROVIDED TO THE SUBCOMMITTEE:

- Letter dated October 12, 2000, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Pressurized Thermal Shock Technical Basis Reevaluation Project.
- SECY-00-0140, "Reevaluation of the Pressurized Thermal Shock Rule (10 CFR 50.61) Screening Criterion," dated June 23, 2000.
- SECY-01-0045, "Status Report - Reevaluation of the Technical Basis for the Pressurized Thermal Shock Rule (10 CFR 50.61)" dated March 16, 2001.
- SECY-01-0185, "Status Report - Reevaluation of the Technical Basis for the Pressurized Thermal Shock Rule (10 CFR 50.61)" dated October 5, 2001.

- Siu, N., NRC, "Uncertainty Analysis and Pressurized Thermal Shock: An Opinion; White Paper Last Revised September 3, 1999."
- Kirk, M., NRC, and William, P., ORNL, "Recommended Method to Account for Uncertainty in the Fracture Characterization Used to Re-Evaluate the Pressurized Thermal Shock(PTS) Screening Criteria," revised draft dated October 3, 2001. [**Internal Use Only**]
- Williams, P.T., and Dickson, T.L., ORNL, NUREG/CR-xxxx, ORNL/TM-2001-xx, "Fracture Analysis of Vessels - Oak Ridge FAVOR, v01.0, Computer Code: Theory and implementation of Algorithms, Methods, and Correlations," revised draft dated October 15, 2001. [**Internal Use Only**]
- Dickson, T.L., and Williams, P.T., ORNL, NUREG/CR-xxxx, ORNL/TM-2001-55, "Fracture Analysis of Vessels - Oak Ridge FAVOR, v01.0: Computer Code: User's Guide," revised draft dated October 10, 2001. [**Internal Use Only**]

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NOTE: Additional details of this meeting can be obtained from a transcript of this meeting available in the NRC Public Document Room, One White Flint North, 11555 Rockville Pike, Rockville, MD, (301) 415-7000, downloading or viewing on the Internet at "http://www.nrc.gov/ACRSACNW," or can be purchased from Neal R. Gross and Co., 1323 Rhode Island Avenue, NW, Washington, D.C. 20005, (202) 234-4433 (Voice), 387-7330 (Fax), e-mail: nrgross@nealgross.com.

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE MATERIALS AND METALLURGY SUBCOMMITTEE
PTS SCREENING CRITERION REEVALUATION PROJECT INITIAL RESULTS
JANUARY 15, 2002
ROCKVILLE, MARYLAND

- AGENDA -

	TOPIC	**PRESENTER**	**TIME**
I.	Opening Remarks	P. Ford, ACRS	8:30-8:35 a.m.
II.	Status of Pressurized Thermal Shock (PTS) Technical Basis Reevaluation Project	M. MAYFIELD, RES E. Hackett, RES	8:35-~~9:00~~ 9:10 a.m.
	A. Probabilistic Risk Assessment (PRA) Group	RES	
	B. Thermal Hydraulics (T/H) Group	RES	
	C. Probabilistic Fracture Mechanics (PFM)Group	RES	
III.	Oconee Results	RES A. MOSLEH, UMD	4:00-4:30 pm ~~9:00-10:15 a.m.~~
	A. Dominant transients		
	B. Predicted vessel failures		
	C. Relation to existing screening criteria		
	- **BREAK** -		10:15-10:30 a.m.
IV.	Modeling Process	RES A. KOLACZKOWSKI M. KIRK, RES A. MOSLEH, UMD	9:10 - 10:15 a.m ~~10:30-12:00 noon~~
	A. Derivation of new screening criteria		
	B. 1999 White paper		
	C. Constraints, models, and uncertainties for PRA, T/H, and PFM		
	MODELING PROCESS (CONTINUED)		10:30 - 12:00 NOON
	- **LUNCH** -		12:00-1:00 p.m.
V.	Modeling Process (Continued)	RES	1:00-2:15 p.m.
	- **BREAK** -		2:15-2:30 p.m.
VI.	Modeling Process (Continued)	RES	2:30-~~3:30~~ 4:00 p.m.
VII.	~~Example Problem~~	~~RES~~	~~3:30-5:00 p.m.~~
	~~A. Definition of event sequences~~		
	~~B. Decision for binning sequences~~		
VIII.	Recess	P. Ford, ACRS	~~5:00~~ 4:30 p.m.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE MATERIALS AND METALLURGY SUBCOMMITTEE
PTS SCREENING CRITERION REEVALUATION PROJECT INITIAL RESULTS
JANUARY 16, 2002
ROCKVILLE, MARYLAND

- AGENDA -

	TOPIC	PRESENTER	TIME
IX.	Opening Remarks	P. Ford, ACRS	8:30-8:35 a.m.
	RELAP 5 verification	D. Bessette, RES	8:35 - 8:55
	OPERATOR ACTIONS	A. Kolaczkowski,	8:55 - 9:45
X.	Example Problem (Continued) A. Definition of event sequences B. Decision for binning sequences C. Selection of one sequence to represent a bin D. Definition of initiating event frequencies E. T/H characterization of sequence F. PFM analysis of the sequence G. Combination of inputs to get vessel failure frequency	RES A. Mosleh, UMD	~~8:35~~ 9:45-10:00 a.m.
	- BREAK -		10:00-10:15 a.m.
XI.	Example Problem (Continued)	RES	10:15-11:~~30~~05 a.m.
XII.	Discussion	P. Ford, ACRS	~~11:30-12:00 noon~~ 11:05 - 11:30 a.m.
XIII.	Adjournment	P. Ford	~~12:00 noon~~ 11:30 a.m.

NOTE:

Presentation time should not exceed 50 percent of the total time allotted for specific item. The remaining 50 percent of the time is reserved for discussion.

Number of copies of the presentation materials to be provided to the ACRS - 25.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE MATERIALS AND METALLURGY SUBCOMMITTEE
PTS SCREENING CRITERION REEVALUATION PROJECT INITIAL RESULTS
JANUARY 15, 2002
ROCKVILLE, MARYLAND

- AGENDA -

	TOPIC	PRESENTER	TIME
I.	Opening Remarks	P. Ford, ACRS	8:30-8:35 a.m.
II.	Status of Pressurized Thermal Shock (PTS) Technical Basis Reevaluation Project	E. Hackett, RES	8:35-9:00 a.m.
	A. Probabilistic Risk Assessment (PRA) Group	RES	
	B. Thermal Hydraulics (T/H) Group	RES	
	C. Probabilistic Fracture Mechanics (PFM)Group	RES	
III.	Oconee Results	RES	9:00-10:15 a.m.
	A. Dominant transients		
	B. Predicted vessel failures		
	C. Relation to existing screening criteria		
	- BREAK -		10:15-10:30 a.m.
IV.	Modeling Process	RES	10:30-12:00 noon
	A. Derivation of new screening criteria		
	B. 1999 White paper		
	C. Constraints, models, and uncertainties for PRA, T/H, and PFM		
	- LUNCH -		12:00-1:00 p.m.
V.	Modeling Process (Continued)	RES	1:00-2:15 p.m.
	- BREAK -		2:15-2:30 p.m.
VI.	Modeling Process (Continued)	RES	2:30-3:30 p.m.
VII.	Example Problem	RES	3:30-5:00 p.m.
	A. Definition of event sequences		
	B. Decision for binning sequences		
VIII.	Recess	P. Ford, ACRS	5:00 p.m.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE MATERIALS AND METALLURGY SUBCOMMITTEE
PTS SCREENING CRITERION REEVALUATION PROJECT INITIAL RESULTS
JANUARY 16, 2002
ROCKVILLE, MARYLAND

- AGENDA -

	TOPIC	PRESENTER	TIME
IX.	Opening Remarks	P. Ford, ACRS	8:30-8:35 a.m.
X.	Example Problem (Continued)	RES	8:35-10:00 a.m.
	A. Definition of event sequences B. Decision for binning sequences C. Selection of one sequence to represent a bin D. Definition of initiating event frequencies E. T/H characterization of sequence F. PFM analysis of the sequence G. Combination of inputs to get vessel failure frequency		
	- BREAK -		10:00-10:15 a.m.
XI.	Example Problem (Continued)	RES	10:15-11:30 a.m.
XII.	Discussion	P. Ford, ACRS	11:30-12:00 noon
XIII.	Adjournment	P. Ford	12:00 noon

NOTE:

Presentation time should not exceed 50 percent of the total time allotted for specific item. The remaining 50 percent of the time is reserved for discussion.

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ACNW meeting notices, meeting transcripts, and letter reports are now available for downloading or viewing on the internet at *http://www.nrc.gov/ACRSACNW.*

Videoteleconferencing service is available for observing open sessions of ACNW meetings. Those wishing to use this service for observing ACNW meetings should contact Mr. Theron Brown, ACNW Audiovisual Technician (301/415–8066), between 7:30 a.m. and 3:45 p.m. EST at least 10 days before the meeting to ensure the availability of this service.

Individuals or organizations requesting this service will be responsible for telephone line charges and for providing the equipment and facilities that they use to establish the videoteleconferencing link. The availability of videoteleconferencing services is not guaranteed.

The ACNW meeting dates for Calendar Year 2002 are provided below:

ACNW Meeting No. and Meeting Date:
131st (Rockville, MD)—January 8–10 2002
132nd (Rockville, MD)—February 7, 2002
133rd (Rockville, MD)—March 19–21, 2002
134th (Rockville, MD)—April 16–18, 2002
135th (Las Vegas, NV—tentative)—May 21–23, 2002
136th (Rockville, MD)—June 18–20, 2002
137th (Rockville, MD)—July 23–25, 2002
August 2002—No Meeting
138th (Rockville, MD)—September 24–26, 2002
139th (Rockville, MD)—October 22–24, 2002
140th (Rockville, MD)—November 19–21, 2002
December 2002—No meeting

Dated: December 13, 2001.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 01–31213 Filed 12–18–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Meeting of the ACRS Subcommittee on Materials and Metallurgy; Notice of Meeting

The ACRS Subcommittee on Materials and Metallurgy will hold a meeting on January 15–16, 2002, Room T–2B3, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance.

The agenda for the subject meeting shall be as follows:

Tuesday, January 15, 2002—8:30 a.m. until the conclusion of business
Wednesday, January 16, 2002—8:30 a.m. until 12:00 Noon

The Subcommittee will review the preliminary results of the Fracture Analysis of Vessels: Oak Ridge (FAVOR) code calculation associated with the Reevaluation of the Technical Basis for the Pressurized Thermal Shock Rule Screening Criterion Project. The purpose of this meeting is to gather information, analyze relevant issues and facts, and formulate proposed positions and actions, as appropriate, for deliberation by the full Committee.

Oral statements may be presented by members of the public with the concurrence of the Subcommittee Chairman; written statements will be accepted and made available to the Committee. Electronic recordings will be permitted only during those portions of the meeting that are open to the public, and questions may be asked only by members of the Subcommittee, its consultants, and staff. Persons desiring to make oral statements should notify the cognizant ACRS staff engineer named below five days prior to the meeting, if possible, so that appropriate arrangements can be made.

During the initial portion of the meeting, the Subcommittee, along with any of its consultants who may be present, may exchange preliminary views regarding matters to be considered during the balance of the meeting.

The Subcommittee will then hear presentations by and hold discussions with representatives of the NRC staff, and other interested persons regarding this review.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, and the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor, can be obtained by contacting the cognizant ACRS staff engineer, Mr. Noel F. Dudley (telephone 301/415–6888) between 7 a.m. and 3:45 p.m. (EST). Persons planning to attend this meeting are urged to contact the above named individual one or two working days prior to the meeting to be advised of any potential changes to the agenda, etc., that may have occurred.

Dated: December 12, 2001.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 01–31214 Filed 12–18–01; 8:45 am]

BILLING CODE 7590–01–P

NUCLEAR REGULATORY COMMISSION

Privacy Act of 1974; New System of Records

AGENCY: Nuclear Regulatory Commission.

ACTION: Notice of New System of Records.

SUMMARY: The Nuclear Regulatory Commission (NRC) is providing notice of the establishment of a new system of records, NRC–12, Child Care Tuition Assistance Program Records.

EFFECTIVE DATE: The new system of records will become effective without further notice on January 28, 2002 unless comments received on or before that date cause a contrary decision.

ADDRESSES: Send comments to: Secretary, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. Attention: Rulemakings and Adjudications staff. Hand deliver comments to 11555 Rockville Pike, Rockville, Maryland, between 7:30 a.m. and 4:15 p.m. on Federal workdays. Copies of comments received may be examined at either the NRC Public Document Room, One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, or the NRC's Agencywide Documents Access and Management System (ADAMS). Comments are also available at the NRC's rulemaking Web site at *http://ruleforum.llnl.gov.* This site also enables you to submit comments. Comments may be uploaded as files (any format), if your Web browser supports that function. For information about the interactive rulemaking Web site, contact Ms. Carol Gallagher, 301–415–5905; e-mail: cag@nrc.gov.

FOR FURTHER INFORMATION CONTACT: Sandra S. Northern, Privacy Program Officer, FOIA/Privacy Act Team, Web, Publishing, and Distribution Services Division, Office of the Chief Information Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001, telephone: 301–415–6879; e-mail: *ssn@nrc.gov.*

SUPPLEMENTARY INFORMATION: The establishment of this new system of records, NRC–12, Child Care Tuition Assistance Program Records, will allow the NRC to collect and maintain family income data from NRC employees for the purpose of determining their eligibility for child care subsidies, and the amounts of the subsidies. It will also maintain information from the employee's child care provider(s) for verification purposes, e.g., that the provider is licensed. Data will be

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

SUBCOMMITTEE MEETING ON MATERIALS AND METALLURGY

JANUARY 15-~~17~~, 2002
Date

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NAME	AFFILIATION
Yung-Hsien Chang	UMD
TED MEYER	WESTINGHOUSE ELECTRIC CO.
BARRY SLOANE	WESTINGHOUSE ELECTRIC
Heshan Gunawardane	University of Maryland
Eric Frantz	Westinghouse Electric
Bruce A. Bishop	Westinghouse Electric
TERRY L. DICKSON	OAK RIDGE NATIONAL LABORATORY
Ken Yoon	Framatome ANP
Marjorie Natishan	PEAI
Robert Beaton	ISL, INC.
W Arcieri	ISL, Inc
Don Fletcher	ISL, Inc
Bob Hardies	CCNPPI
Gary Platt	NMC - Palisades
Ron Gamble	Sartrex
Ali Mosleh	Univ. of Maryland
FRED SIMONEN	PPNL (TEL-COM)
STEVE	PNNL (TEL-COM)

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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JANUARY 15-[redacted], 2002
Date

NRC STAFF SIGN IN FOR ACRS MEETING

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NAME	NRC ORGANIZATION
Mark Kirk	RES/DET/MEB
Ed Hackett	RES/DET
Hugh Woods	RES/DRAA
SHAH MALIK	RES/DET
Matthew A. Mitchell	NRR/DE/EMCB
Shoji Takeyama	NRR/RLEP
Scott Newberry	RES DRAA.
Nilesh Chokshi	RES/DET/MEB
Jack Rosenthal	RES/SMSAB
Cayetano Santos	RES/DET/MEB
James Davis	RES/DET/MEB
Owen Gormley	RES/DET/ERAB
BARRY ELLIOT	NRR/DE/EMCB
Yuri Orechwa	NRR/DSSA/SRXB
Carol Moyer	RES/DET/MEB
Debbie Jackson	RES/DET/MEB

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Gary F Pratt	NMC - Palisades
Ken Yoon	Framatome ANP
~~Nitesh~~	
Bill Arcieri	ISL
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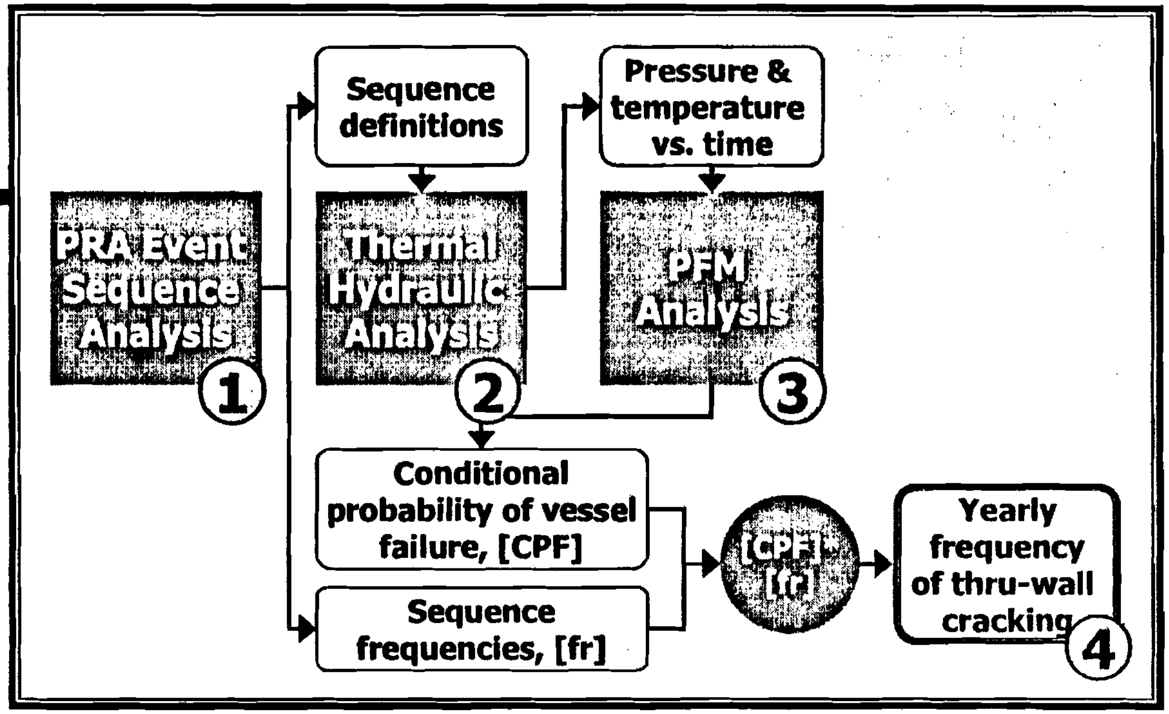
Cayetano Santos Jr.	RES/DET/MEB
Nilesh Chokshi	RES/DET/MEB
ED HACKETT	RES/DET/MEB
Mark Kirk	RES/DET/MEB
Debbie Jackson	RES/DET/MEB

ATTACHMENTS

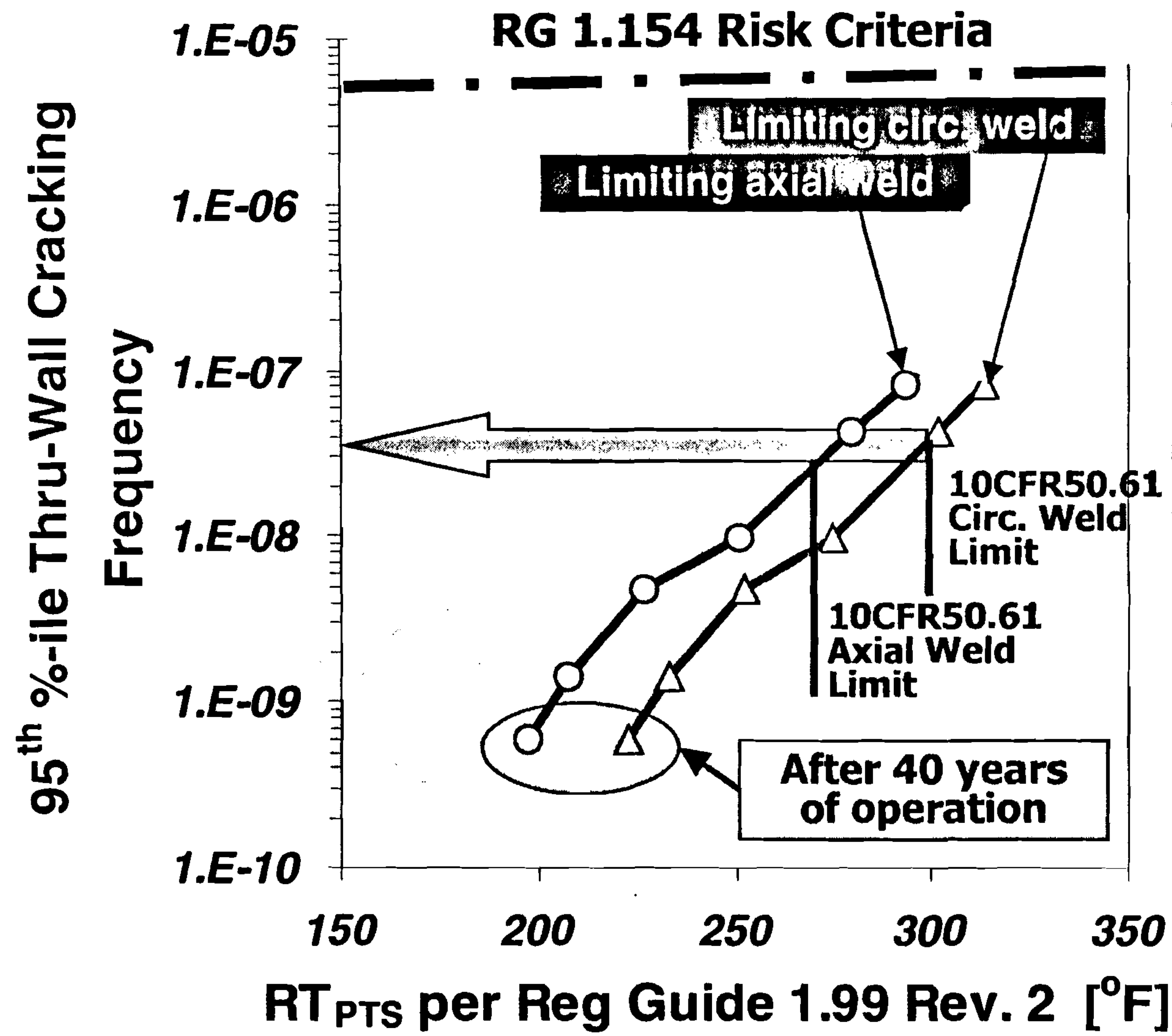
Project Status

- Approach developed to assess the PTS risk
- Involves inputs from and models developed in three different technical areas
  - Probabilistic Risk Assessment (PRA)
  - Thermal Hydraulics (TH)
  - Probabilistic Fracture Mechanics (PFM)
- Recent accomplishments
  - October 01: PFM Code (FAVOR) V01.0 released to public
  - Estimating the risk of vessel failure for 4 plants



Plant	1 PRA	2 TH	3 PFM	4 TWCF
Oconee	draft	draft	draft	draft
Palisades	Licensee revising	1st cut	1st cut	1st cut
Beaver	NRC building	1st cut	1st cut	1st cut
Calvert	begun	1st cut	--	--

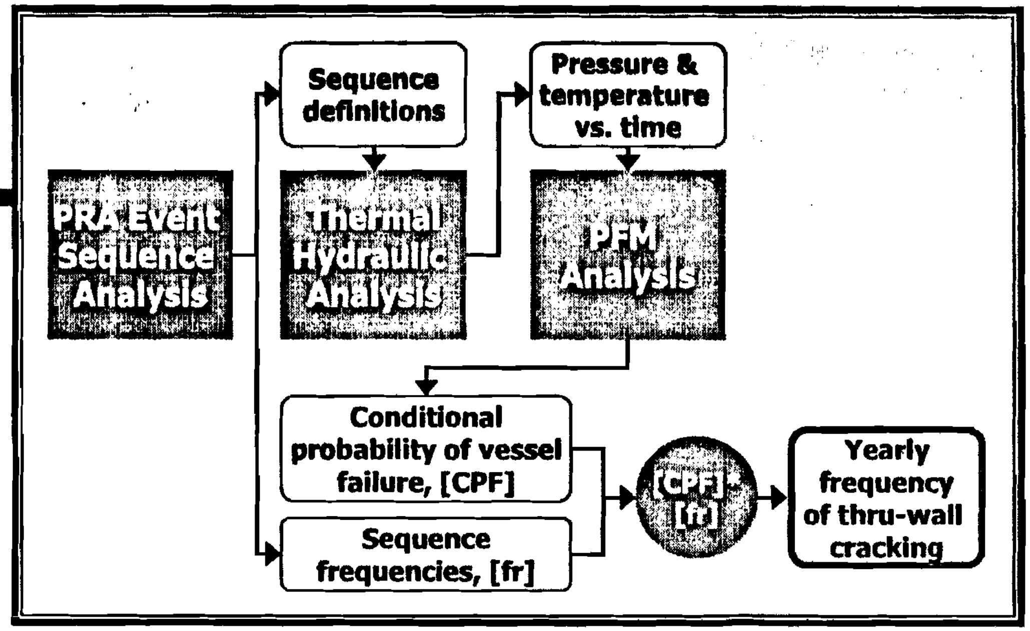
Preview of Results for Oconee 1



Relative to the RG 1.154 risk criteria, current results are

- ✓ ≈ 4 orders of magnitude lower after 40 years of operation
- ✓ ≈ 2 orders of magnitude lower at the current screening limits

Summary of Changes (*vs.* 1980s Analysis)



- **PRA**
  - **Use of latest PRA/HRA data**
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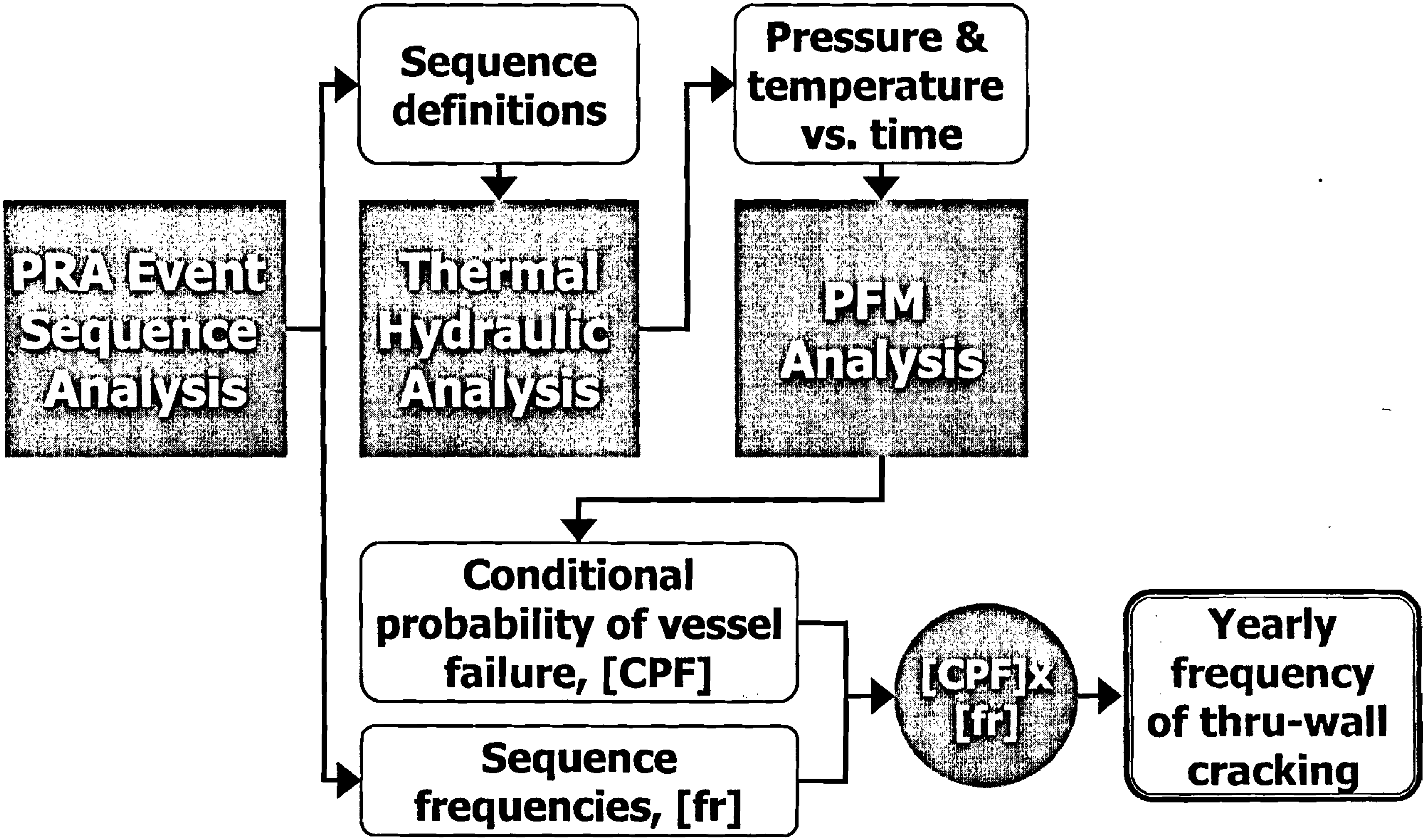
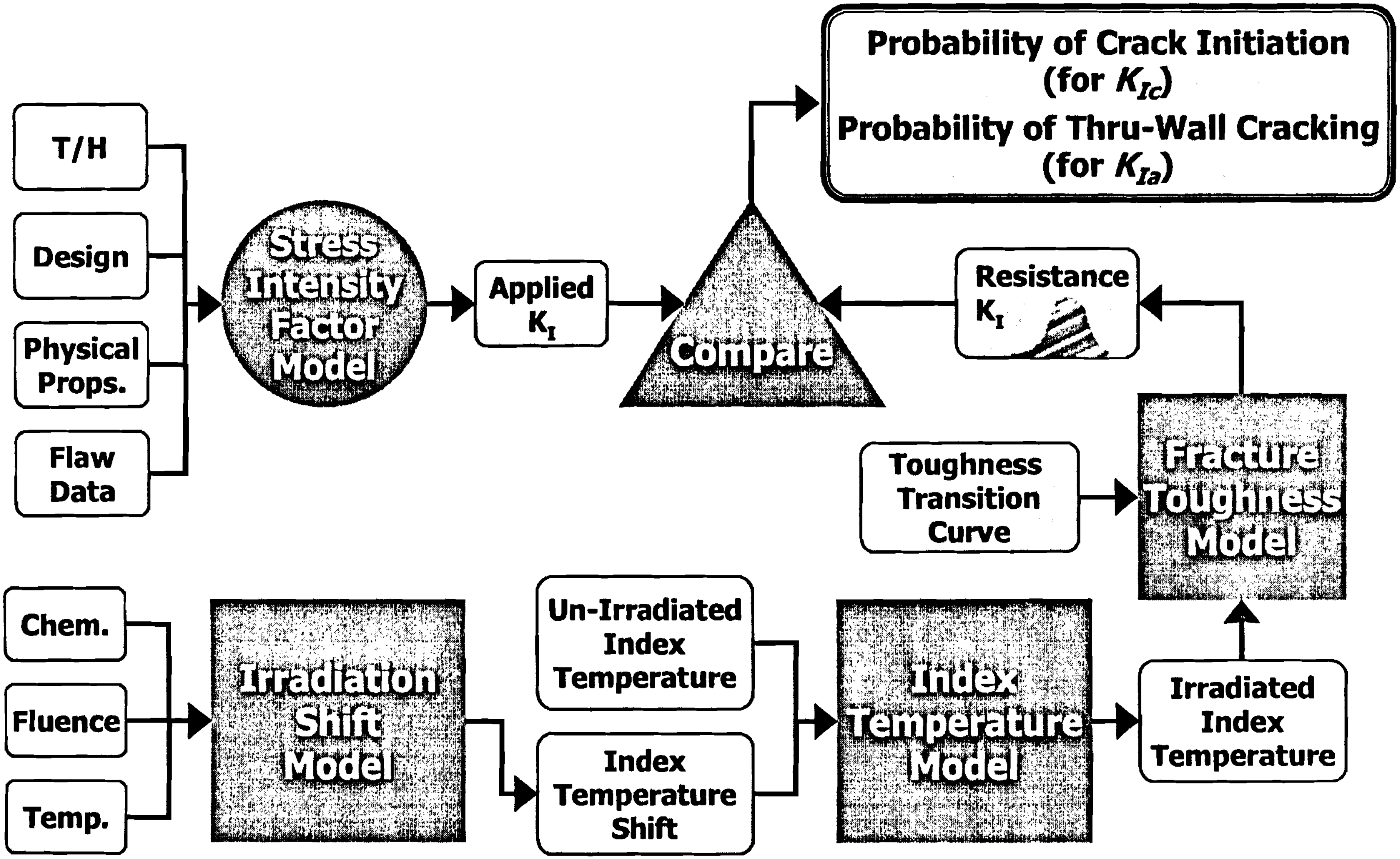
PRA, T/H, PFM Interaction & Integration



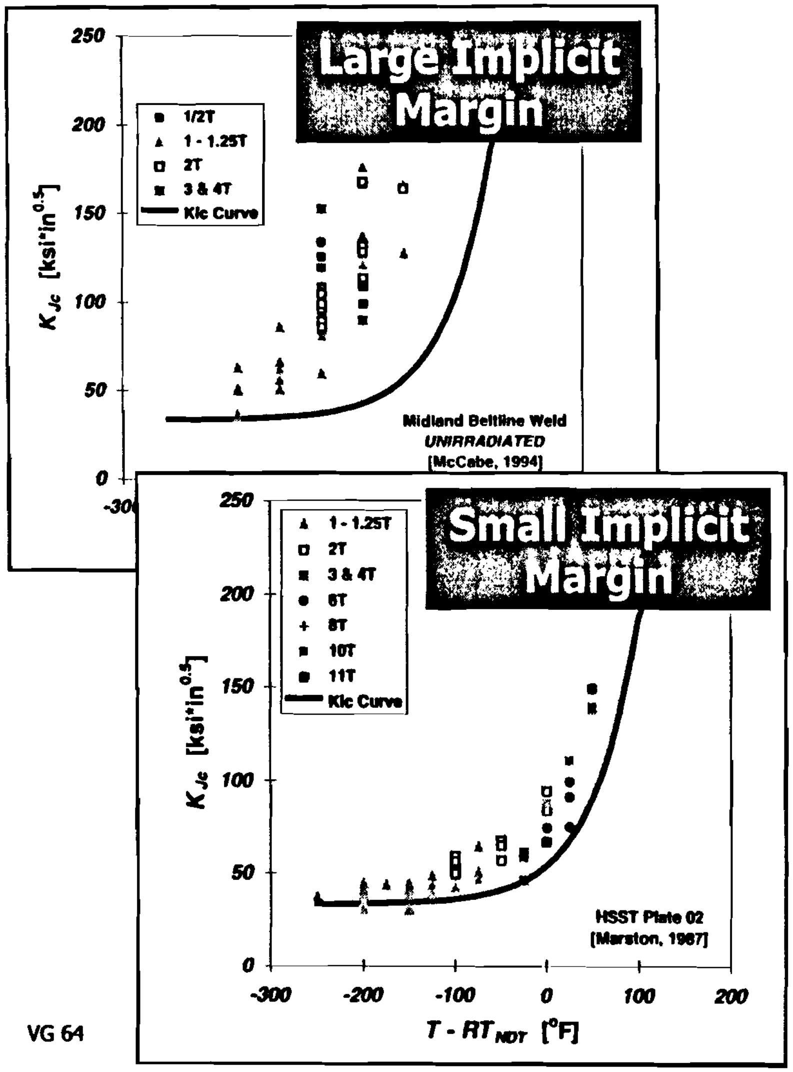
Probabilistic Fracture Mechanics Expanded



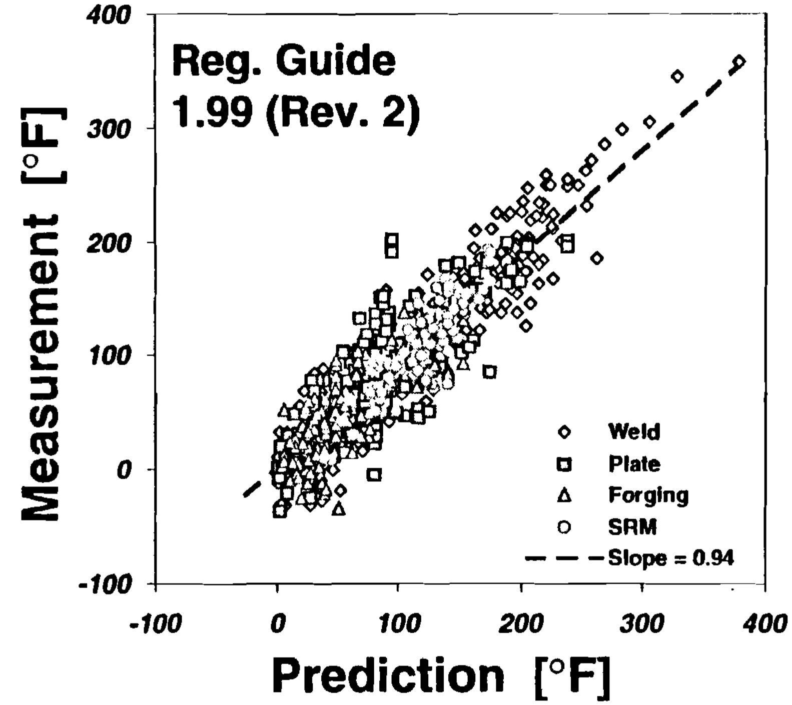
VG 62

Current Toughness & Embrittlement Models

Toughness Model



Embrittlement Model



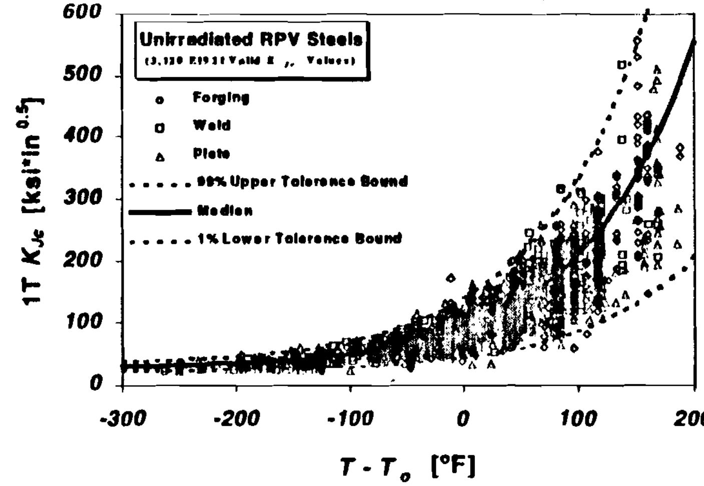
$$\Delta T_{30} = (CF)(\phi t)^{(0.28-0.1\log(\phi t))}$$

- ✓ **Non-toughness data *assumed to* represent toughness data**
  - → **Uncertainties treated *implicitly***
  - → **Uncertainties types *mixed***

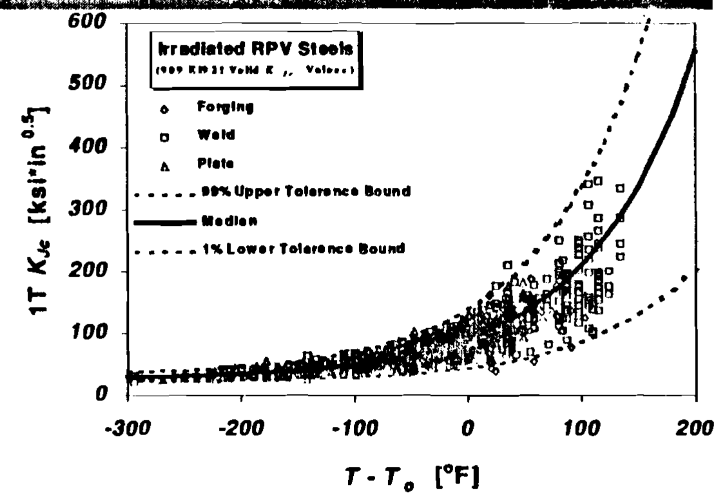
Initiation Fracture Toughness
(Best Estimate Model)

- **Physical understanding suggests**
  - **Common T-dependence**
  - **Common scatter**
  - **Irradiation produces shift only**
- **<u>Best Estimate: The Master Curve method</u> (with T_o transition temperature)**
  - **Physically based**
  - **Empirically validated**
  - **Weakest link statistics account explicitly for cleavage process**
  - **T_o defined consistently for all steels**
    - ✓ **Temperature at 100 MPa√m**
    - ✓ **Corresponds to the position of the data instead of a representation of data**

Un-Irradiated



Irradiated

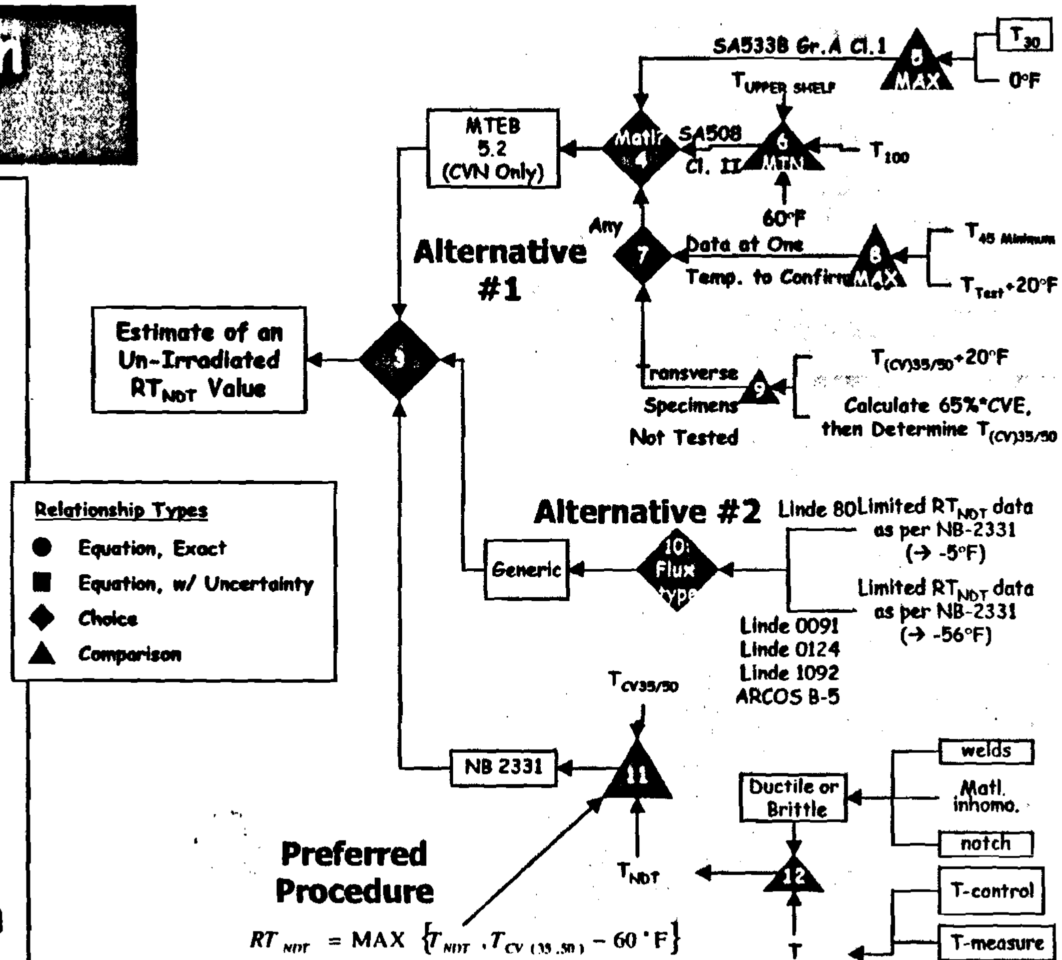


Current Transition Temperature Model: RT_{NDT}

Uncertainty Identification and Classification

Epistemic

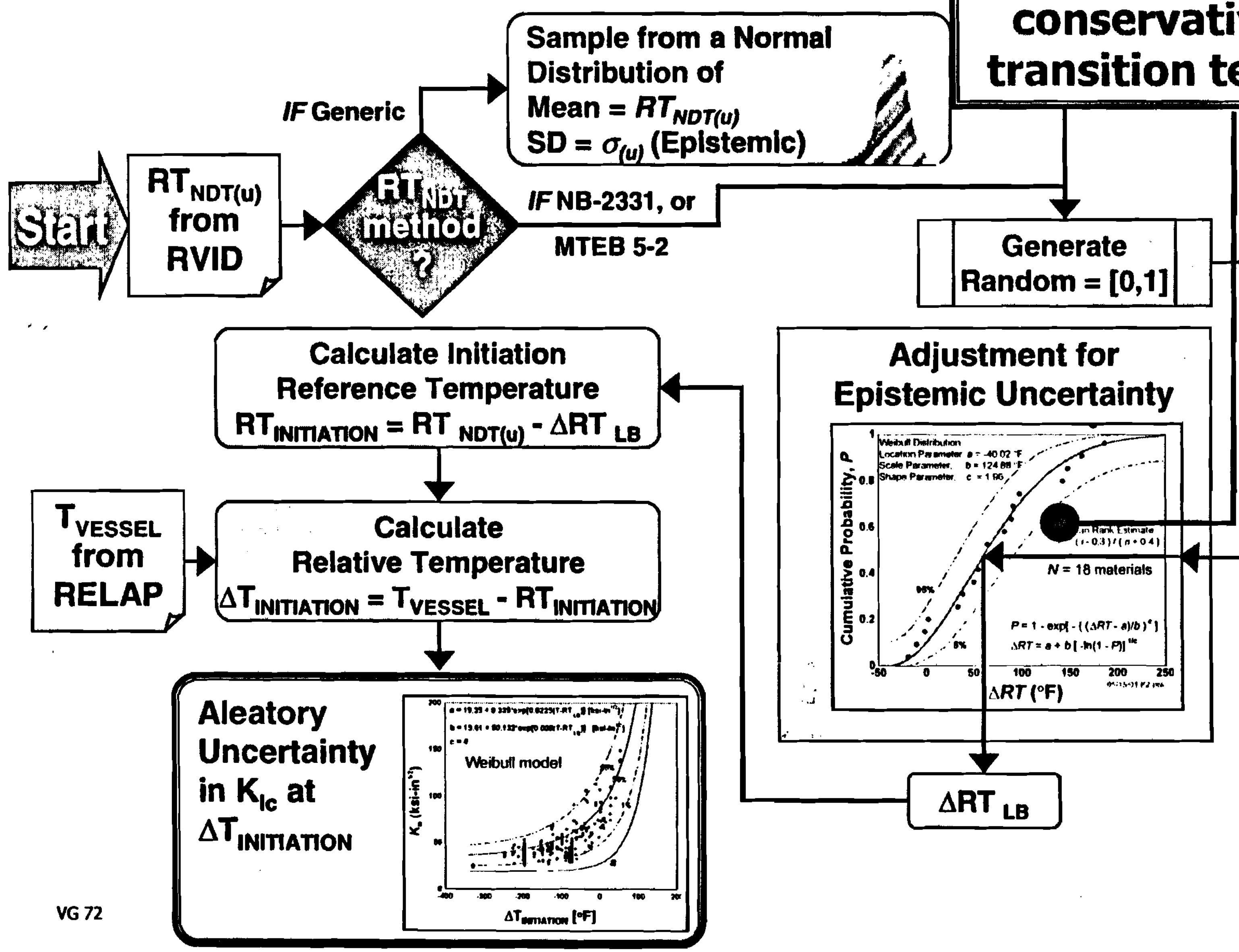
- **Conservative bias of NB-2331**
- **The myriad of methods and transition temperatures used to define RT_{NDT}**
- **The limited data sets used to define Generic RT_{NDT} values and to assess the appropriateness of the various transition temperatures used in RT_{NDT} definitions**
- **The lack of prescription in the test methods**
- **Neither *NDT* nor *CVN* are material properties**
- **Material variability**



RT_{NDT} uncertainty should be modeled as epistemic

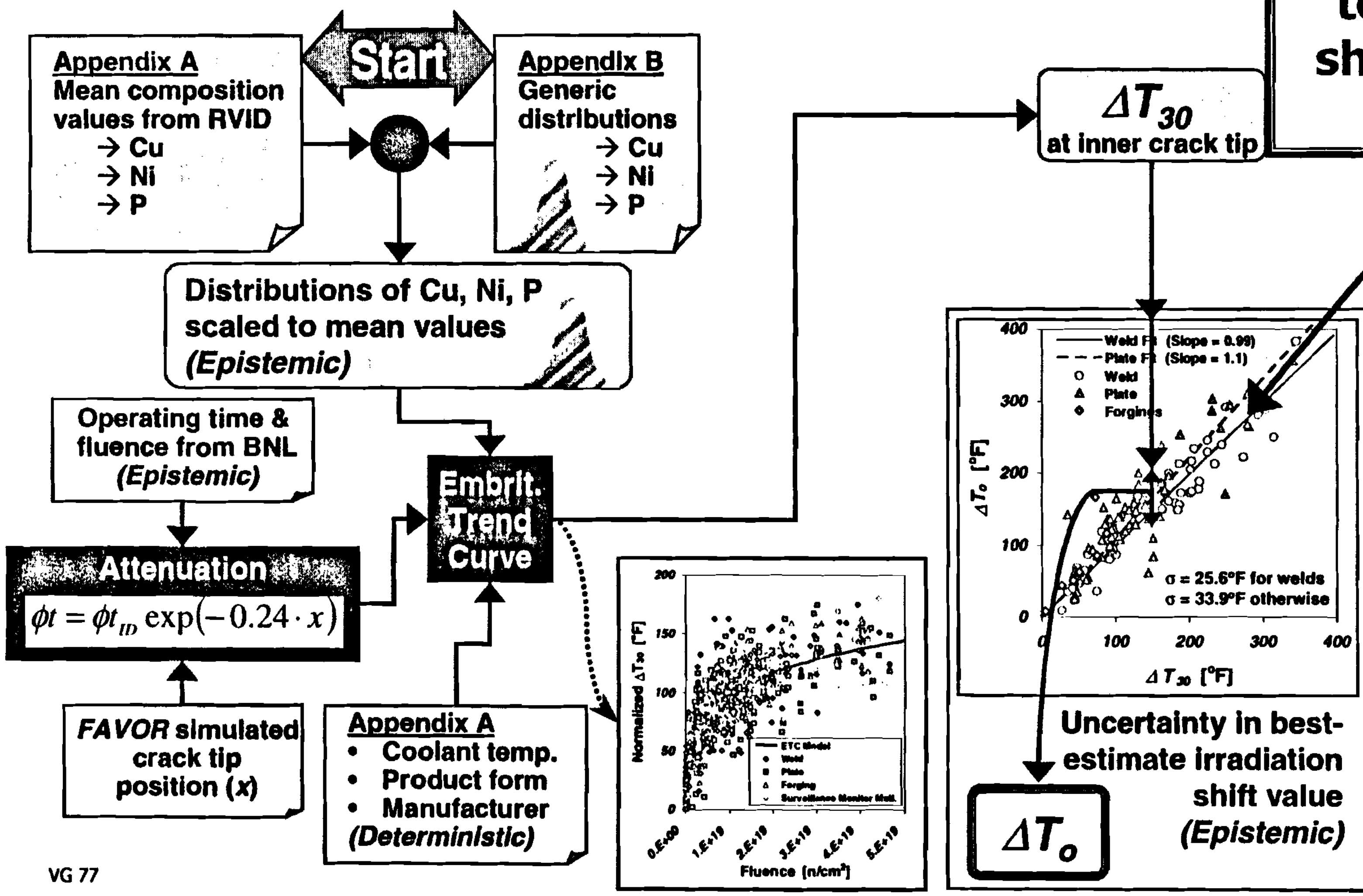
FAVOR Model of Initiation Fracture Toughness

Previous (circa-1980s) PTS analyses assumed a significant conservative bias on transition temperature



FAVOR Model of Irradiation Shift

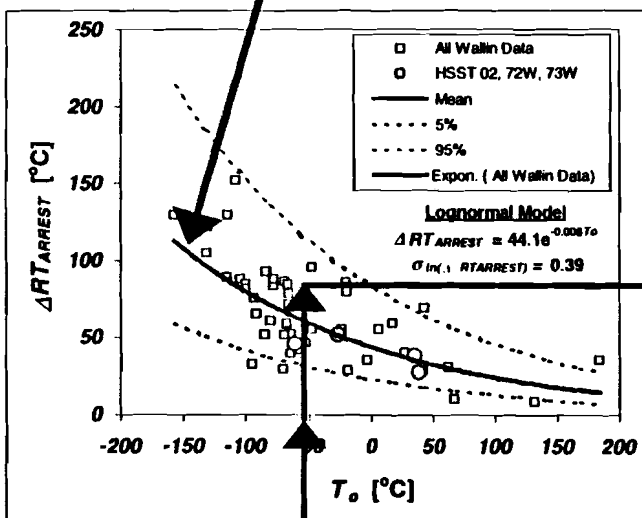
Previous (circa-1980s) PTS analyses assumed that CVN and toughness shift are the same



FAVOR Model of Arrest Fracture Toughness

Previous (circa-1980s) PTS analyses assumed a non-conservative fixed $K_{Ic} \rightarrow K_{Ia}$ shift

Determine Shift Between Initiation and Arrest Curves
(Epistemic Uncertainty)



$T_o \equiv RT_{INITIATION} - 14.4°F$ (from initiation model)

Start

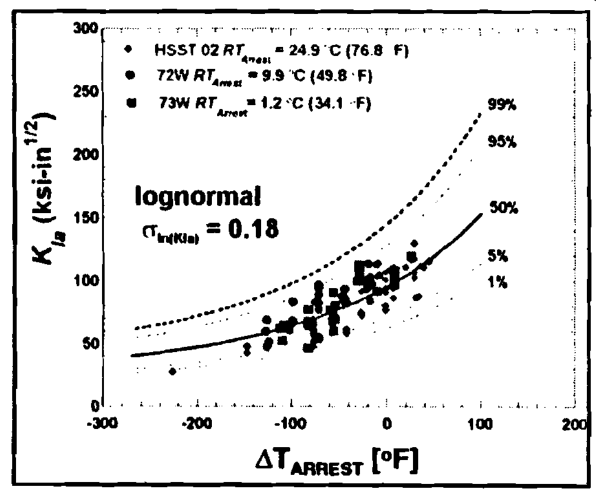
ΔRT_{ARREST}

Calculate New Reference Temperature
$RT_{ARREST} = RT_{INITIATION} + \Delta RT_{ARREST}$

Calculate Relative Temperature
$\Delta T_{ARREST} = T_{VESSEL} - RT_{ARREST}$

T_{VESSEL} from RELAP

Aleatory Uncertainty in K_{Ia} at ΔT_{ARREST}



Pressurized Thermal Shock Rule (10CFR50.61) Re-Evaluation
Model Definition and Uncertainty Treatment

Mark Kirk, Shah Malik, Terry Dickson, Ed Hackett

Probabilistic Fracture Mechanics

Roy Woods, Alan Kolaczkowski

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Thermal Hydraulics

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Meeting Objectives

- **Provide a status report on the PTS re-evaluation project**
- **Describe the modeling process and uncertainty quantification**
- **Discuss current results and insights from analysis of Oconee**
- **Provide one detailed example of the modeling and uncertainty process**

Project Status

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- **Approach developed to assess the PTS risk**
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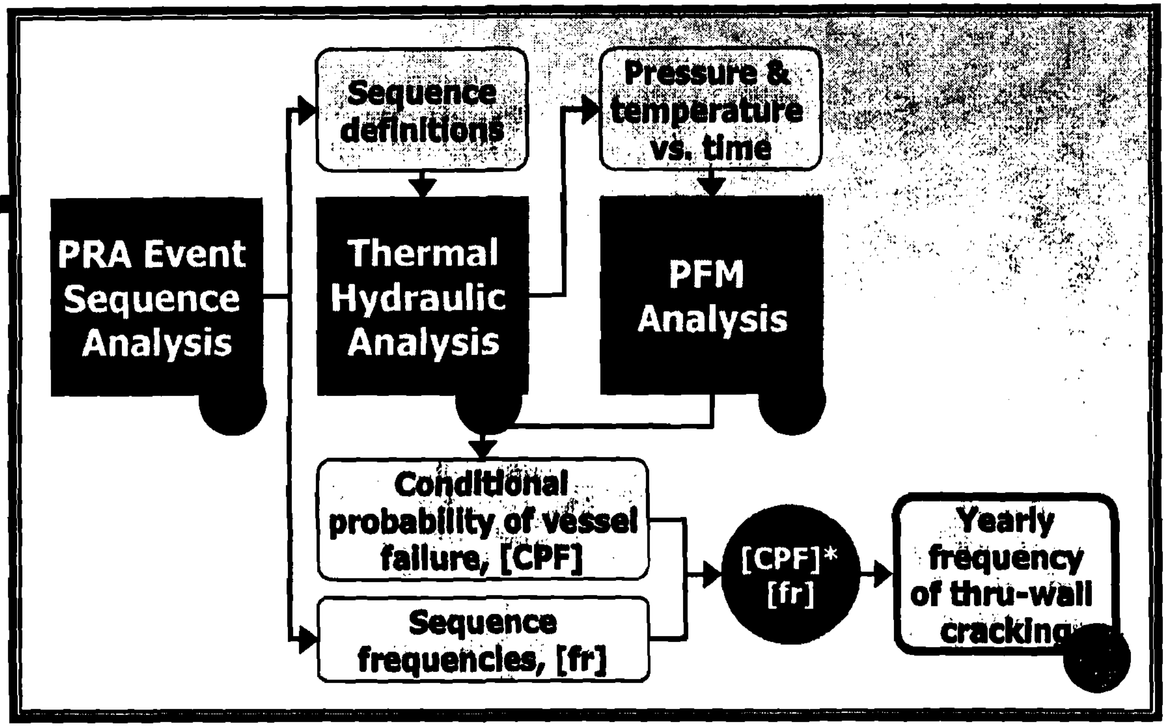

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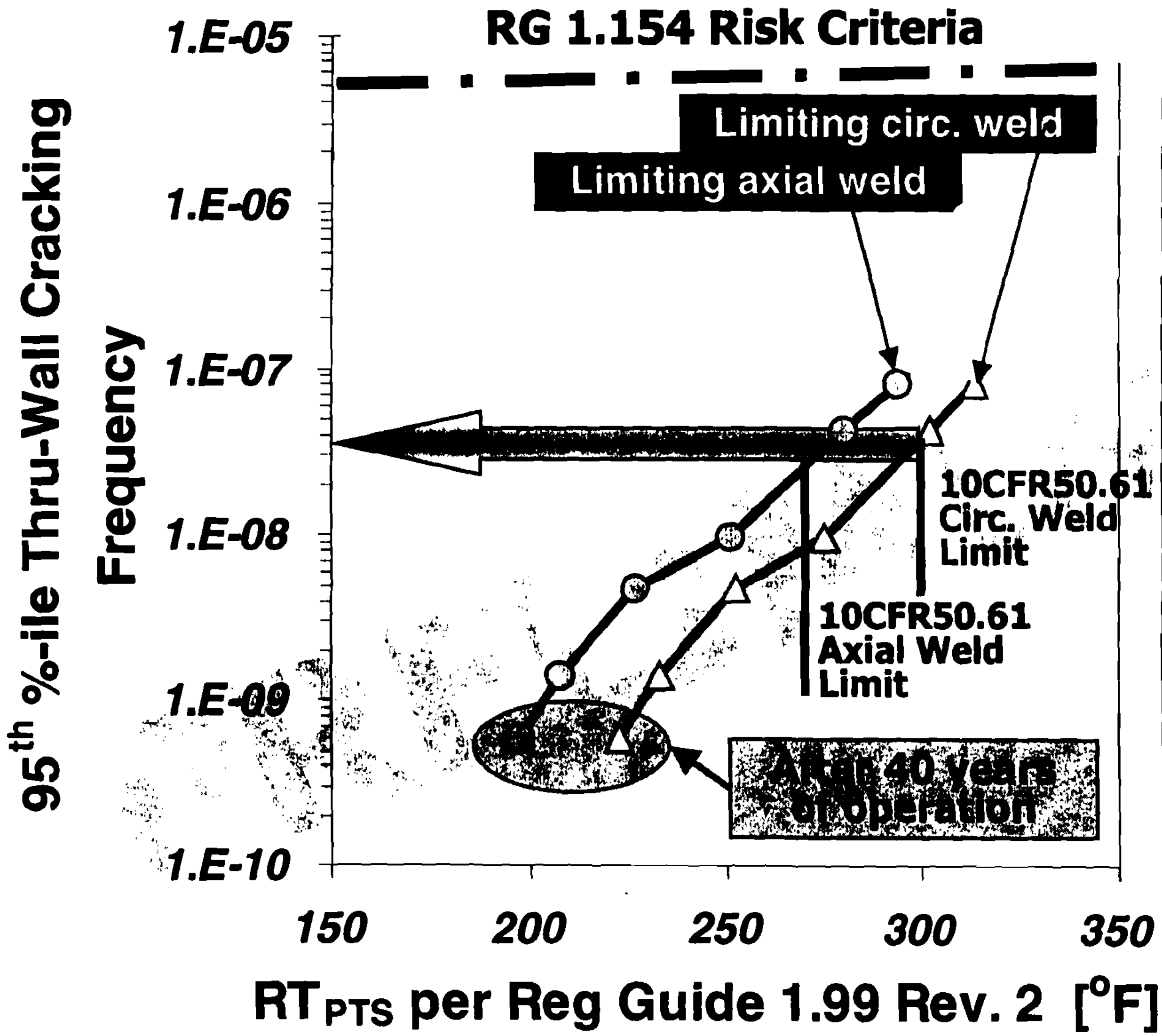
Work Remaining

- Q/A, finish internal events analysis for the Oconee, Palisades, Beaver, and Calvert
- External event risk contribution
- Integration of results (risk criteria)

Preview of Results for Oconee 1



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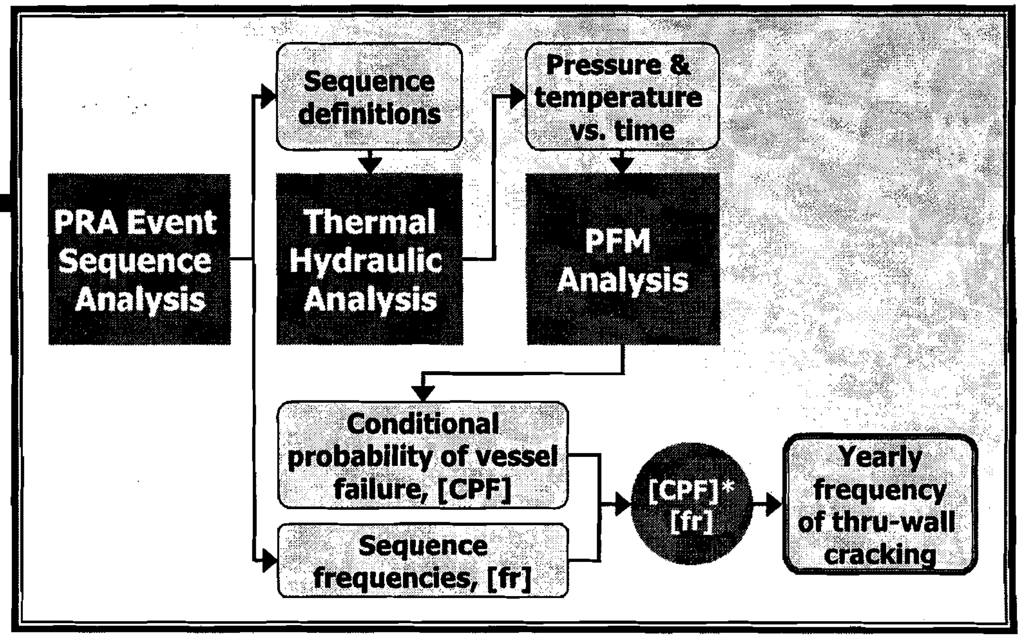

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Computational Models & Uncertainty

Overall Modeling / Uncertainty Process

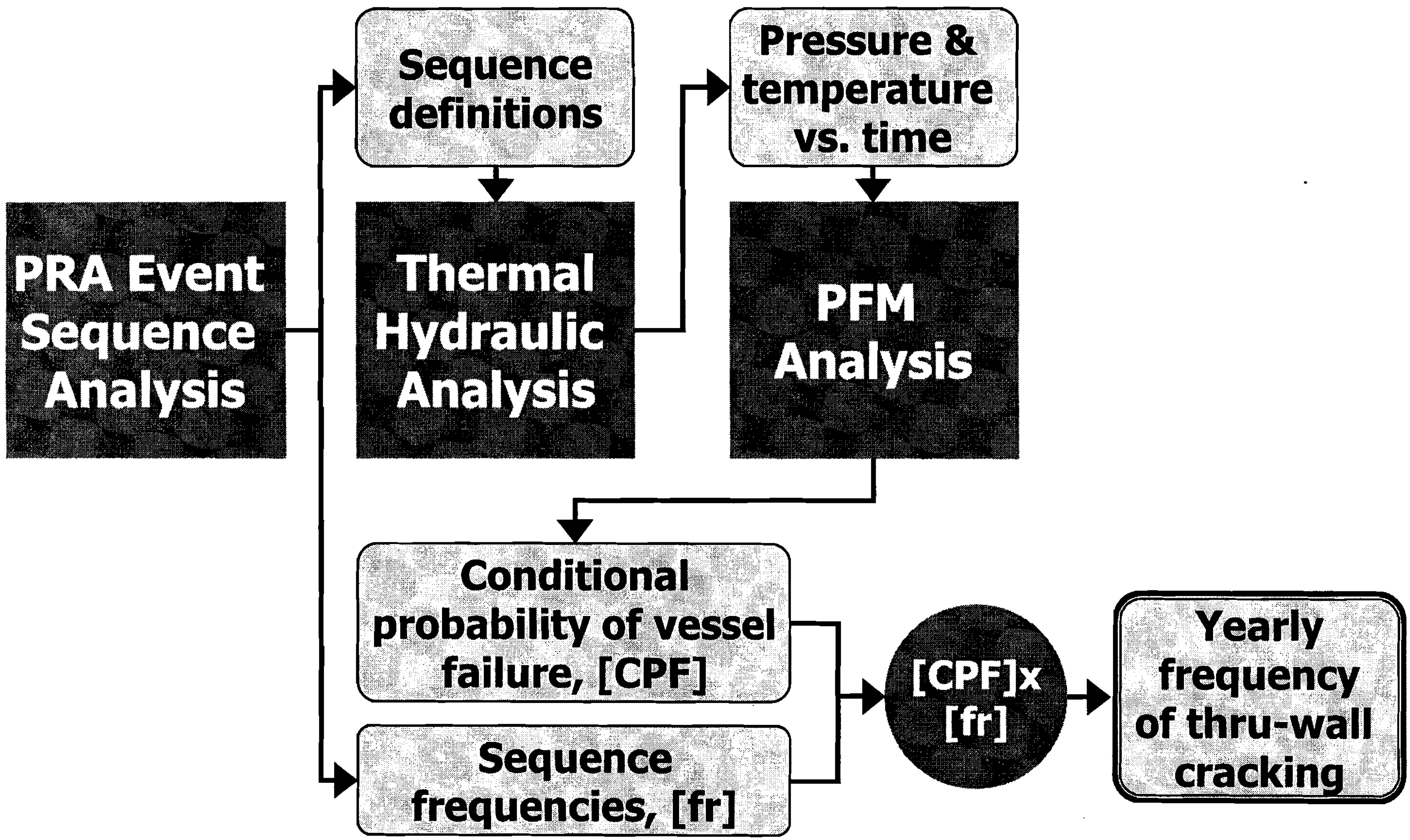
Topics Discussed

- **Guidelines for project, and intended material screening criteria**
- **Interaction / integration of PRA, T/H, PFM analyses**
- **A concept for model development and uncertainty treatment was established in 1999. Today we focus on model development / uncertainty treatment procedures as implemented in this project**
  - **PRA**
  - **T/H**
  - **PFM**

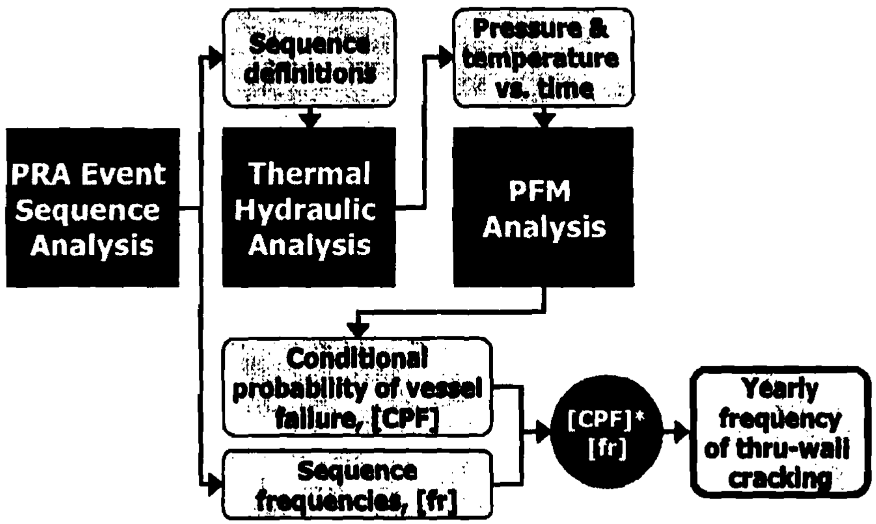
Guiding Principles & Intended Materials-based Screening Criteria

- **The methodology used in the PTS re-evaluation project requires an**
  - **Explicit treatment of uncertainties across technical disciplines**
  - **Uncertainty classification & separation**
    - ✓ **Aleatory**
    - ✓ **Epistemic**
  - **Uncertainty quantification**
- **<u>Intent</u>: No new material measurements needed to assess vessel integrity**

PRA, T/H, PFM Interaction & Integration



Implemented Model Development / Uncertainty Treatment



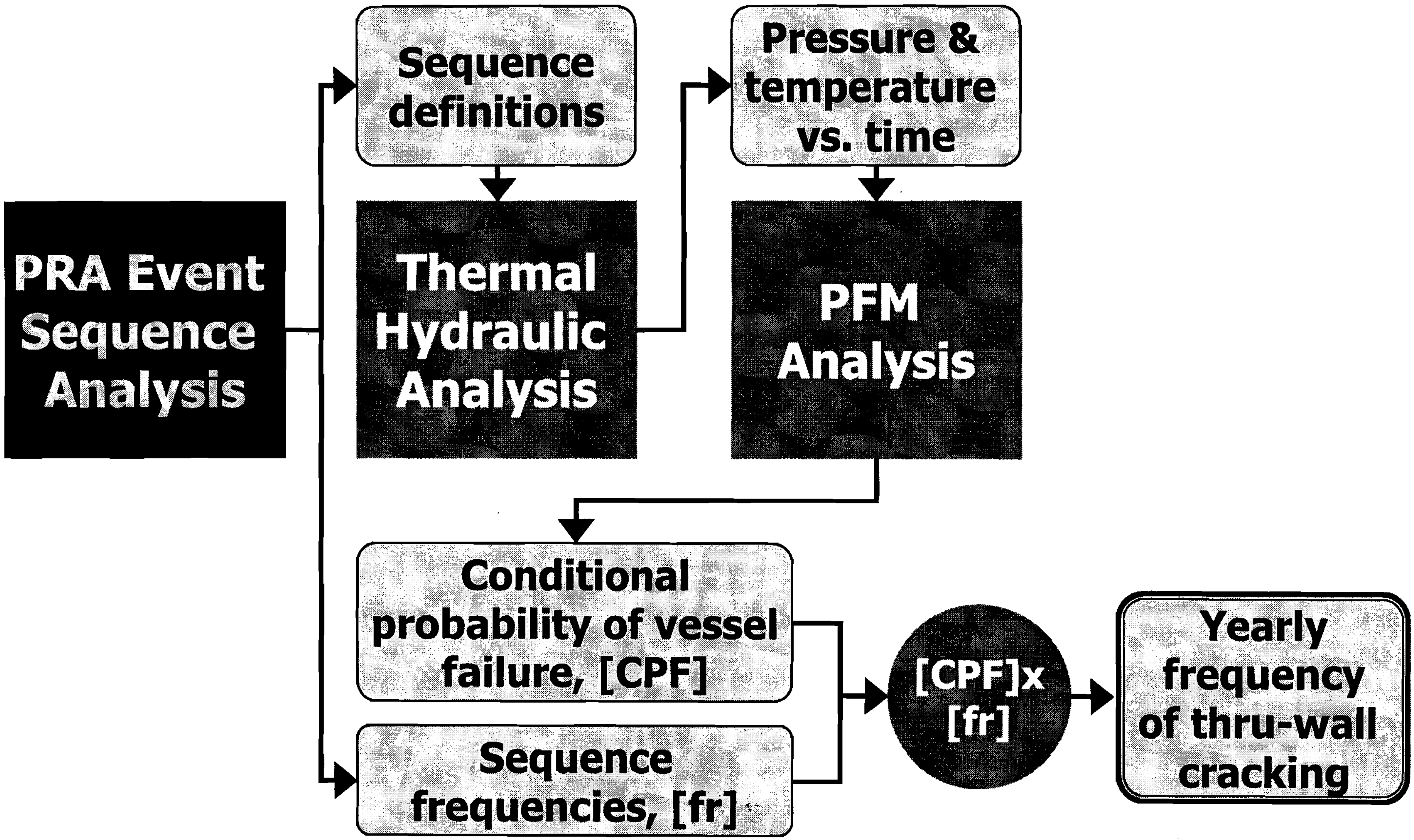
3 Main Elements

- PRA event sequence analysis
- Thermal hydraulic analysis
- Probabilistic fracture mechanics analysis

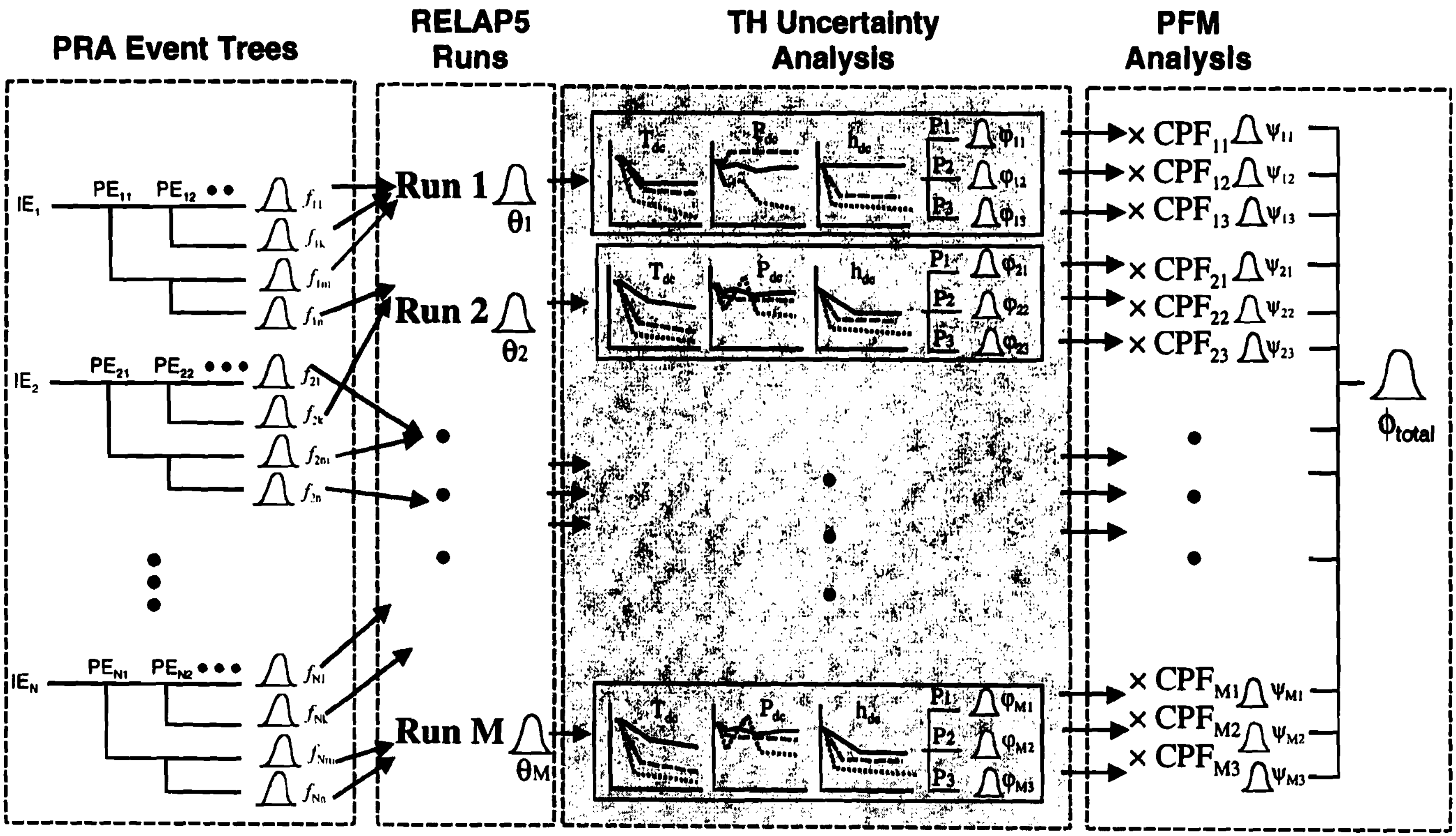
For each element

- Constraints imposed on the element, and/or fundamental assumptions
- Components of the element
- Process used for model building
- Uncertainty treatment
- Significant changes since 1980s evaluation

PRA in the Overall Process



Conceptual Model



PRA / HRA

(Constraints & Fundamental Assumptions)

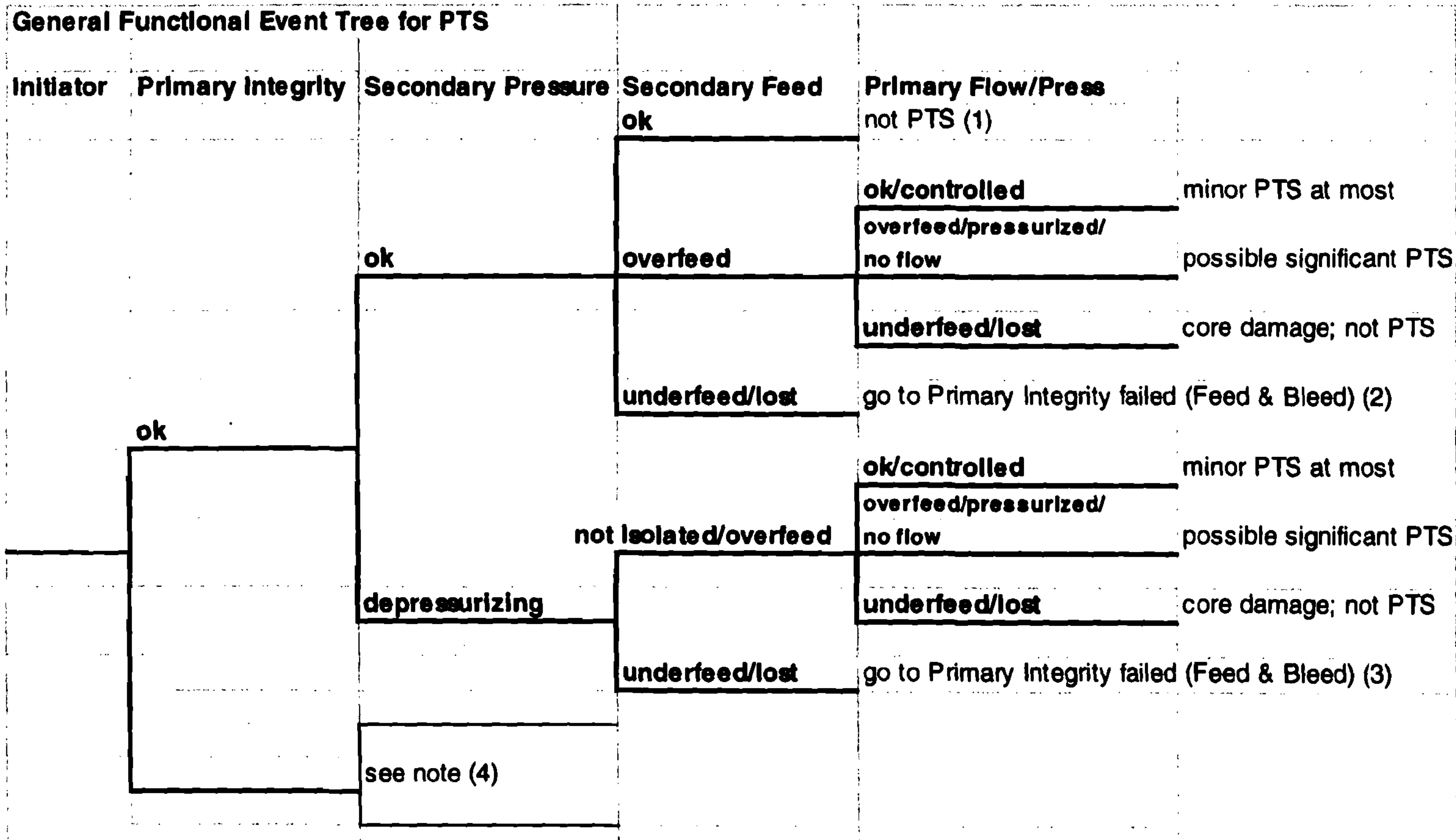
Limitations

- Typical PRA limitations
- Screened scenarios based on T-H and frequency (i.e., not passed on to PFM)
- External events being evaluated

Considered

- Both full power and hot zero power initial conditions
- Timing of events generally early in scenario, though did consider, "late" failures and recoveries
- Considered both errors of omission & acts of commission
- Four functions of interest

Overview of Scenario Modeling



(1) not considered a PTS concern regardless of primary flow/pressure

(2) loss of feed to both SGs; procedures call for Feed & Bleed which is equivalent to entering tree at Primary Integrity "failed"

(3) like (2) above except secondary depressurization has further lowered RCS temp

(4) logic is identical to rest of tree above except choices also exist for Primary Flow/Pressure even for Secondary Pressure and Feed "ok" state and PTS effects are generally potentially greater for all scenarios

Iterative Modeling Process

Collect Information

- Past events
- Former PTS studies
- Current Oconee PRA
- FSAR sections
- P&IDs/Electricals
- Emergency/Abnormal procedures
- PTS training material
- Observed Simulations
- Interactions w/Licensee ***(periodic)***

Identify the Following:

- Initiators
- Functions
- Relevant equipment
- Relevant crew actions
- Dependencies
- Important timing

Build PRA-PTS Model (SAPHIRE)

- Large event trees/small fault trees
- All sequences (no prior screening)
- Integrated equipment/operator events
- Discrete times for operator actions
- Early/late events & failure/recovery of equipment
- Data from Oconee-specific and industry experience (Bayesian update)
- Initial Human Error Probabilities (HEPs)

Perform Preliminary Quantification

- No frequency screening
- Plant "states"(not T-H bins)

Perform Review

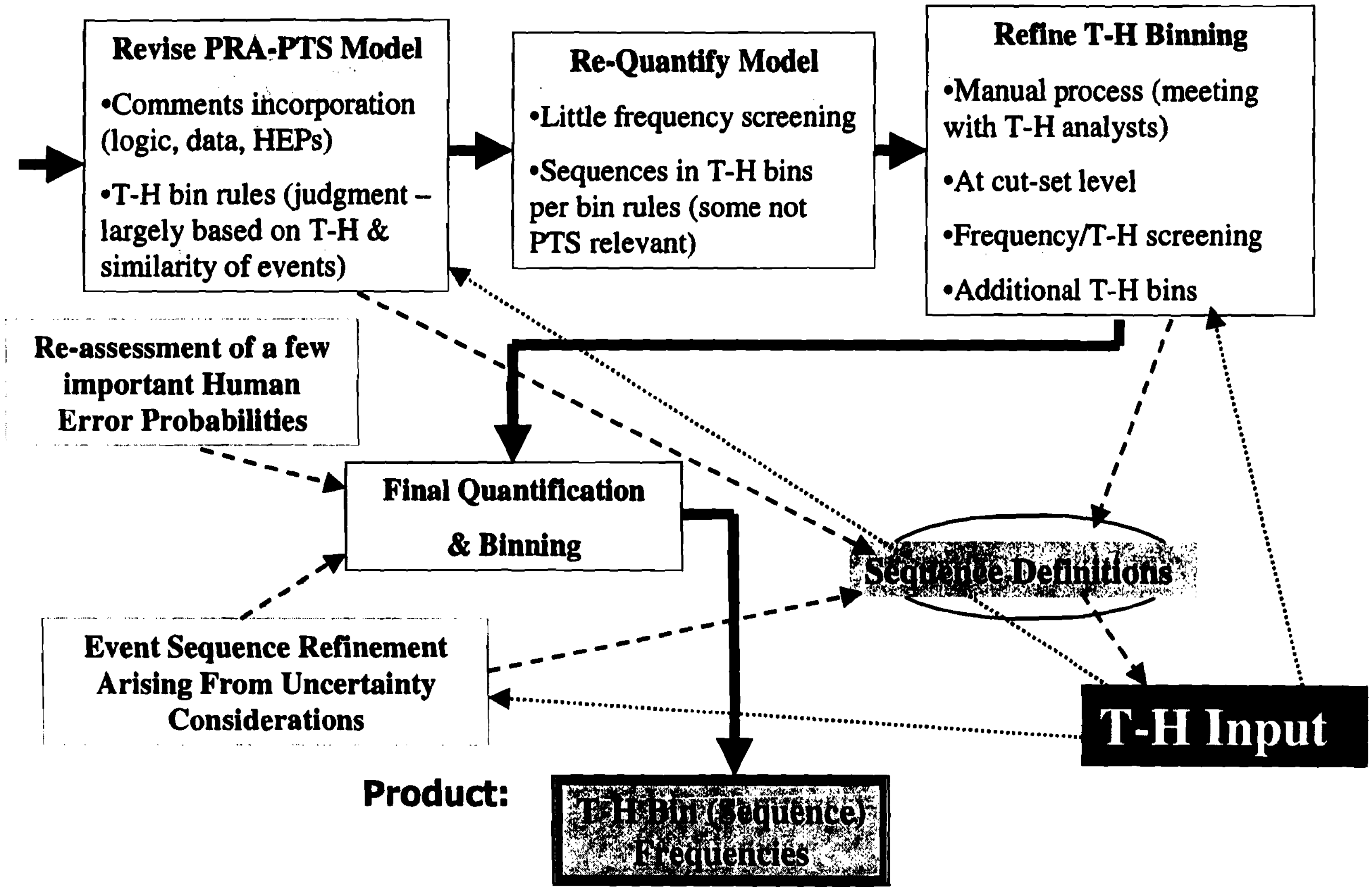
- Internal
- Licensee (visit)
- Comments

Cont'd

Sequence Definitions

T-H Input

Iterative Modeling Process (cont.)



PRA / HRA

Initiators Modeled*

- LOCAs: Small, Medium, Large
- Transients
  - Rx-Turb Trip
  - 2 Loss of Bus
  - Loss of Instrument Air
  - Loss of Main Condenser/ Main Feedwater
  - Loss of Offsite Power (including station blackout)
- Other
  - Steam Gen Tube Rupture
  - Steam Line Break: Small, Large

* for both full power & hot zero power

Equipment Modeled

- **Primary Integrity**
  PORV & block valve, SRVs, RCS as break source, consideration of pressurizer spray/heaters
- **Secondary Pressure**
  Steam Lines as break source, TBVs & associated block valves, MSSRVs, consideration of Turbine Stop/Control valves (ADVs not used at Oconee)
- **Secondary Feed**
  Main Feed, Emergency Feed, Condensate
- **Primary Flow / Pressure**
  Rx Coolant Pumps, HPI/charging, consideration of Core Flood Tanks/Low Pressure Injection, vent valves

& actuation/control (including Integrated Control System) & support systems

Operator Action Considerations

Primary Integrity Control

- ✓ Operator fails to isolate an isolable LOCA
- ✓ Operator induces a LOCA

Secondary Pressure Control

- ✓ Operator fails to isolate
- ✓ Operator isolates when not needed
- ✓ Operator isolates wrong path/SG
- ✓ Operator creates an excess steam demand

Secondary Feed (& T) Control

- ✓ Operator fails to stop/throttle or properly align feed
- ✓ Operator feeds wrong (affected) SG
- ✓ Operator stops/throttles feed when inappropriate

Primary Flow/Pressure Control

- ✓ Operator does not properly throttle injection
- ✓ Operator trips RCPs when not supposed to &/or fails to restore them
- ✓ Operator fails to trip RCPs
- ✓ Operator does not inject enough when required (heading for core damage rather than a PTS concern)

PRA / HRA

(Uncertainty Treatment)

- **What is the PRA quantifying?**
  - **Frequencies of a wide range of representative plant responses to plant upsets (i.e., scenarios), each described by a set of T-H curves, as a result of mitigating equipment successes and failures as well as operator actions, that result in various degrees of overcooling of the internal reactor vessel downcomer wall.**
- **Sources of uncertainty:**
  - **Modeling of the representative plant scenarios**
  - **The frequency of each modeled scenario**

PRA / HRA

(Modeling of Representative Scenarios)

- **Each scenario is a collection of events**
- **Explicit modeling of event timing for operator actions; e.g., failure to take an action in multiple discrete times (by 10 min, by 20 min...)**
- **Dominant model uncertainties were quantified (e.g., timing of SRV reclosure)**
- **Minor model uncertainties were not quantified**

PRA / HRA

(Modeling the Frequency of Each Scenario)

- **Each scenario is the interaction of what is treated as random events:**
  - **Initiating event**
  - **Series of mitigating equipment successes/failures**
  - **Operator actions**
- **So, the occurrence of each scenario is random**

$$\mathrm{Freqy}_{scenario} = \mathrm{Freqy}_{Init\ Event} \times \mathrm{Prob}_{Equip\ Response} \times \mathrm{Prob}_{Oper\ Actions}$$

each with *epistemic* uncertainties described by a distribution

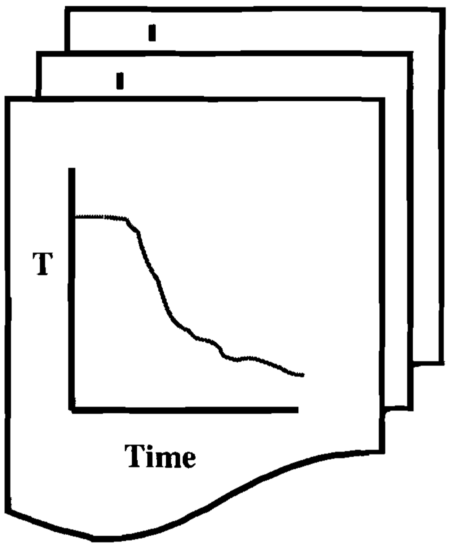
- **The various scenarios frequencies characterize the *aleatory* uncertainties associated with the occurrence of a PTS challenge**
- **Latin hypercube sampling techniques are used to propagate the *epistemic* uncertainties to generate a probability distribution for each scenario frequency**

PRA / HRA *(Significant Changes from Circa-1980s PTS Analyses)*

- **Slight expansion of possible scenarios & initiators**
- **Slight expansion of support systems as initiators and as dependencies**
- **Latest initiating event frequencies, equipment failure probabilities, common cause failure evaluations...**
- **Detailed HRA**
  - **Scenario context-based (considered variability in each context)**
  - **Includes observations from simulator exercises at Oconee**
  - **Latest procedures/training**
  - **More discrete time considerations for actions (less 'gross binning')**
  - **Includes NRC contractor & Licensee judgment**
  - **Consideration of acts of commission that would exacerbate cooling**
- **More sequence/T-H bins (less 'gross binning')**
- **Detailed uncertainty analysis**

PRA / HRA Results

Set of T-H Curves for Bin (Scenario)



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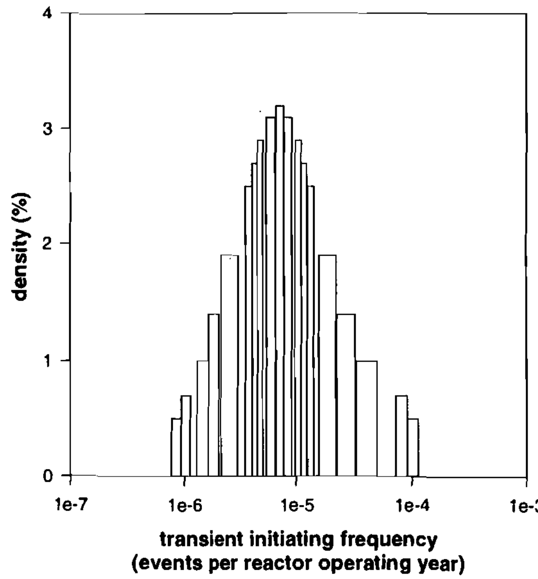


PFM & PRA Integration Tasks

Bin (Scenario) Frequency

(per reactor-year)

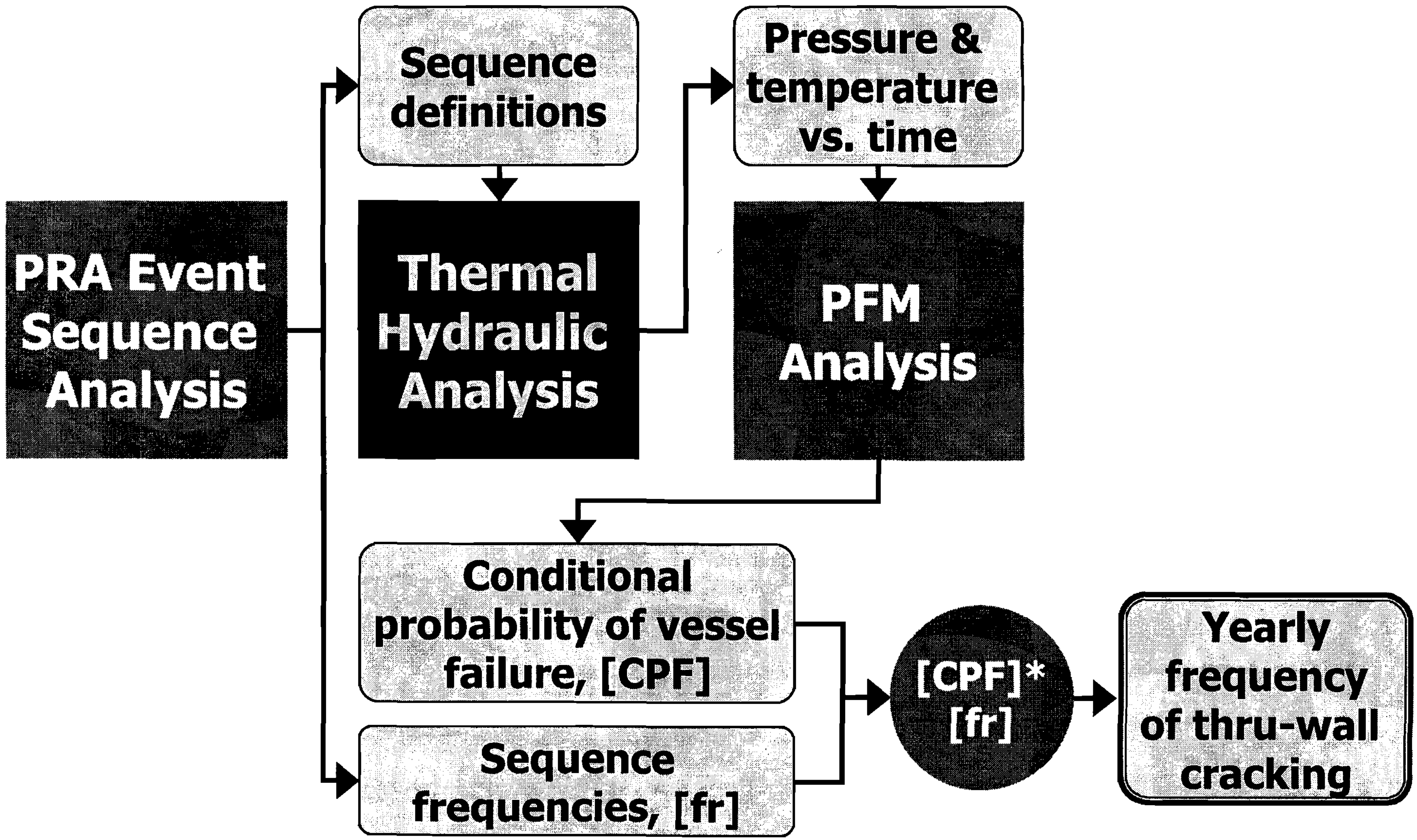
Sampling performed to quantify the *epistemic* uncertainty in the bin frequency

- **Histogram: 19 Quantile Levels (0.5% - 99.5%)**
- **95% confidence interval on each Quantile Value – Lower & Upper Bounds**

PRA Summary

- Modeled relevant initiators, functions, and equipment
- Modeled key operator actions
- Treatment of uncertainty:
  - *Important* sequence modeling uncertainties analyzed
  - Described each sequence frequency uncertainty with a histogram

T/H and Related Uncertainties in the Overall Process



Overview of TH Presentation

- Constraints and assumptions
- RELAP model
- Top-down method of defining plant states with PTS potential and important parameters
- Method of identifying needed RELAP runs
- Identification of dominant sources and types of uncertainty
- Identification of TH uncertainty runs, and corresponding frequency distributions

TH Constraints & Assumptions

- **Large number of actual TH sequences (1E4) need to be reduced to a manageable number of runs (1E2)**
- **Due to complexity of TH model (non-linearity), simplified screening criteria needed to focus uncertainty analyses performed using a detailed model**
- **RELAP5 provides an appropriate model for this RPV analysis**

RELAP5 Model Description

- **Started with Oconee model developed by the INEEL for the early 1980's PTS evaluation**
- **Changes in Setpoints to Current Plant Values**
- **RWST Water Temperature Changed to 70°F (90° F originally used)**
- **Control Models Added to Simulate Operator Actions**
  - **RCP trip on 0.5°F subcooling**
  - **HPI throttling - used combination of RCS temperature, Pzr level as throttling criteria depending on PRA transient definition**

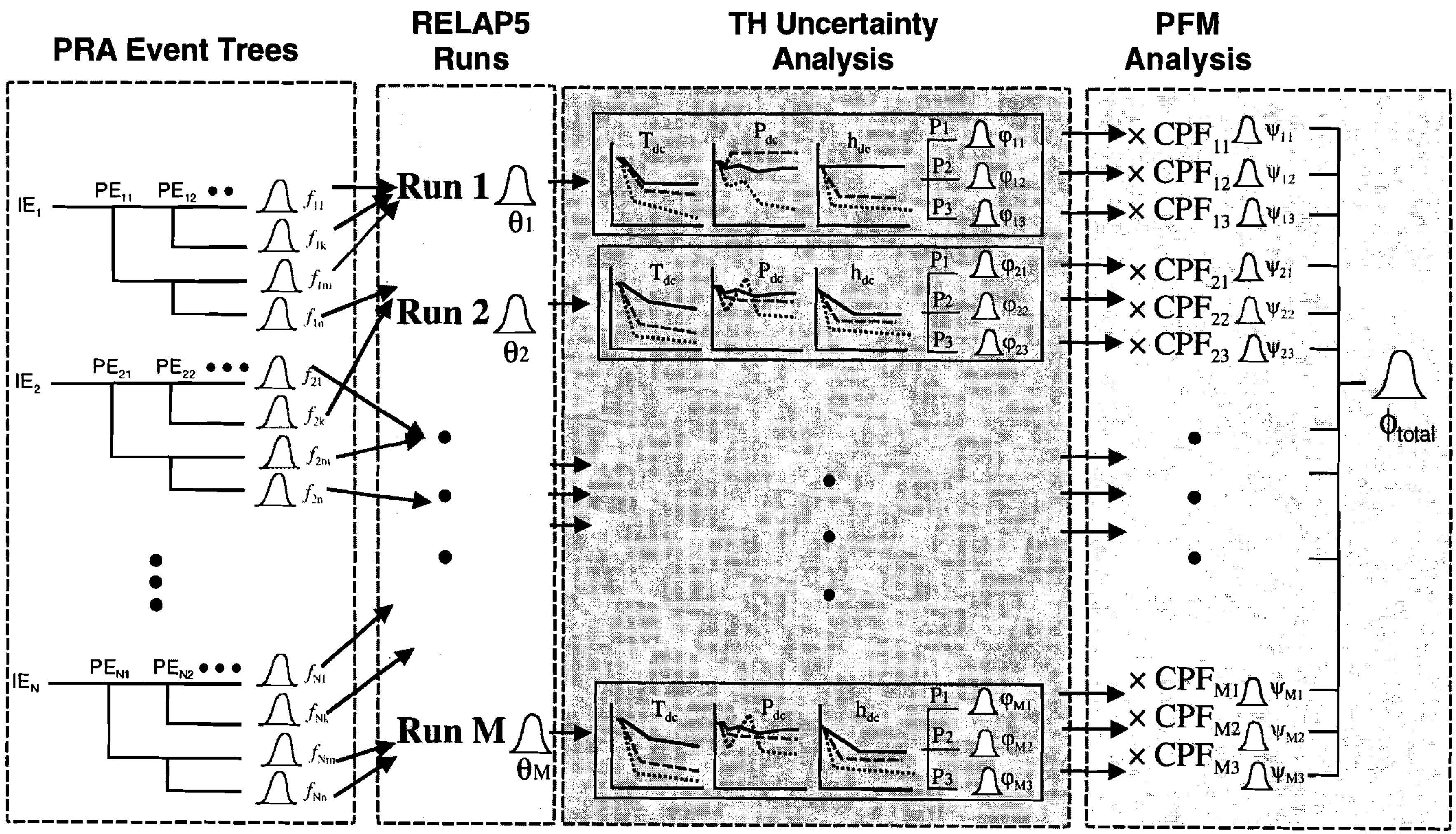
RELAP5 Model Description

- **Various Model Corrections.**
  - **Example - modified level control model so that turbine-driven EFW flow to each SG is controlled independently to correct problem with overfilling of "intact" SG when the level in the other SG was low.**
- **Added two-dimensional downcomer model**

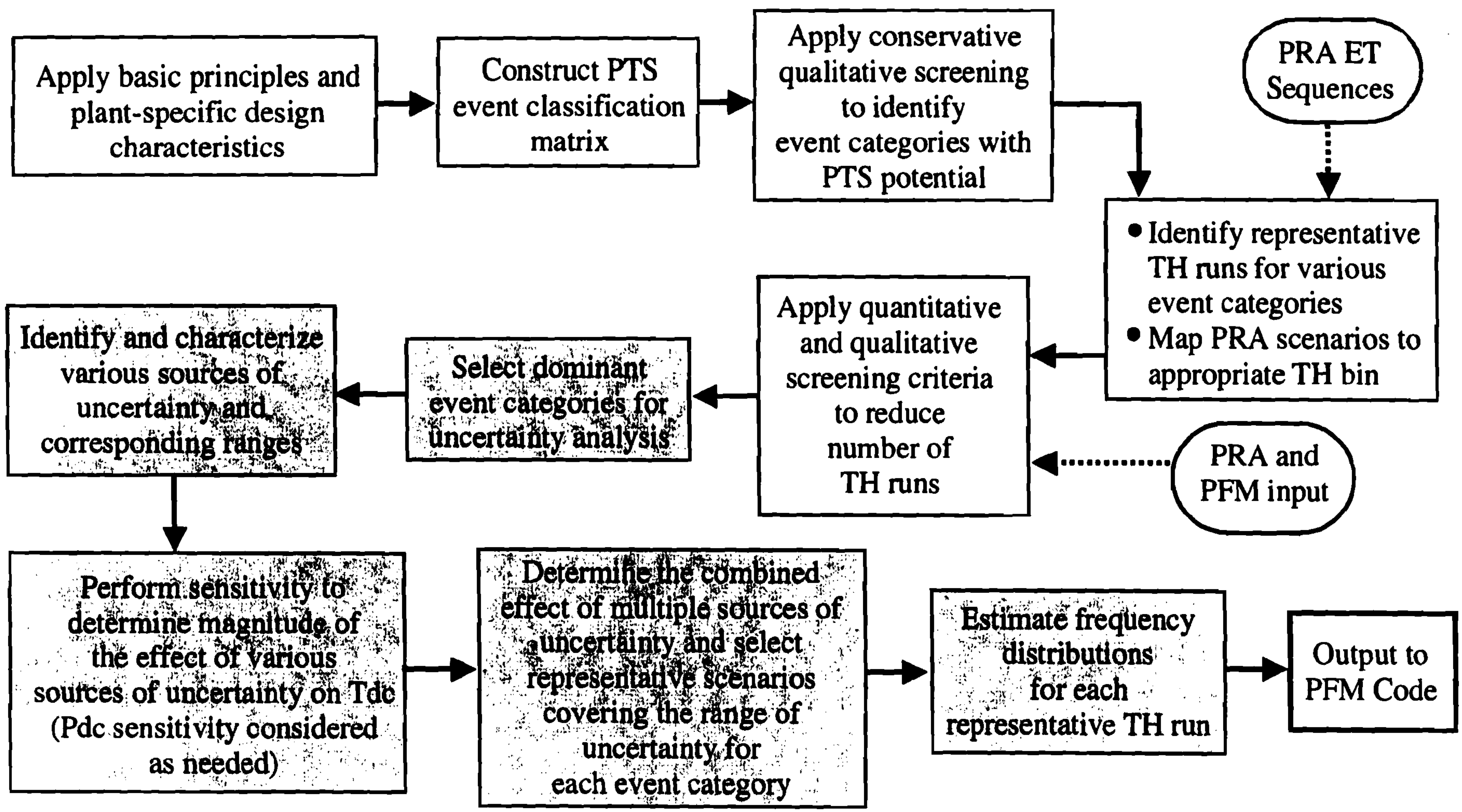
Current vs. 1980s Study

- **What is different this time?**
  - **Major changes in computing capabilities**
    - ✓ **300+ cases vs < 10 cases in the 1980's studies**
    - ✓ **Allowed performance of uncertainty analysis**
  - **Input preparation effort is about the same.**
    - ✓ **Still need to develop input data from plant information**
  - **Post processing capability greatly improved**
    - ✓ **xmgr5 and Automatic Validation Script allows easy generation of plots**
  - **Uncertainty evaluation included**
  - **Experiments (APEX-CE) demonstrate a 1D model is appropriate**
  - **2 fluid code vs. 5 equation**

Conceptual Model



TH Uncertainty Assessment Process



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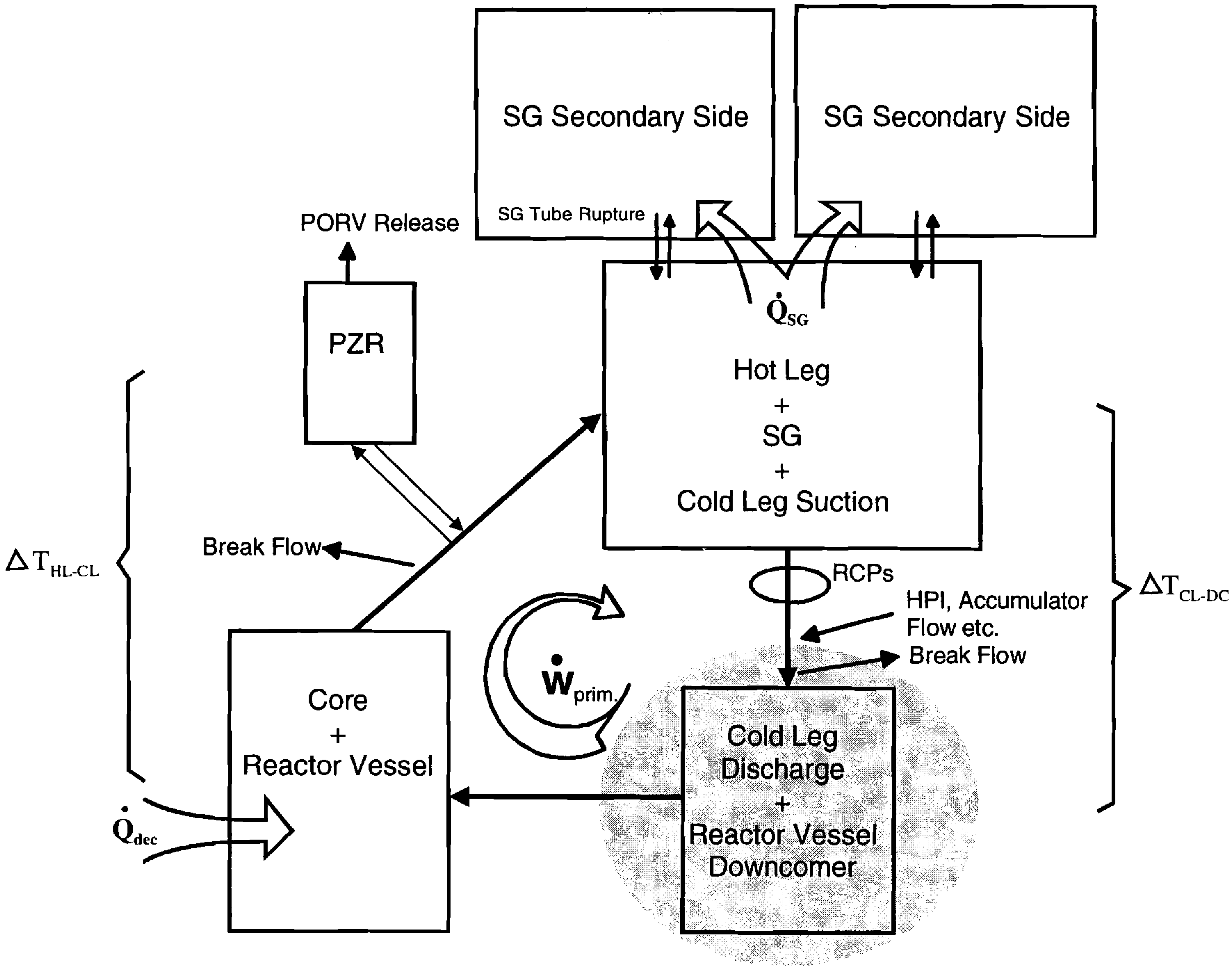
Simple TH Model to Identify PTS Controlling Parameters



Important TH Parameters

Temperature

- Heat Capacities
  - Primary and secondary system
- Heat Sources
  - Decay heat and RCPs
- Heat Sinks
  - Primary system breach, SGs, HPI, CFTs, and LPI
- RCS Coolant Flow Rate
  - RCPs state
- RPV Energy Distribution
  - Mixing of core water in downcomer phenomenon (RPV Vent Valves)
  - RCS flow interruption-and-resumption caused by vapor in candy cane
  - Boiling-condensation

Pressure

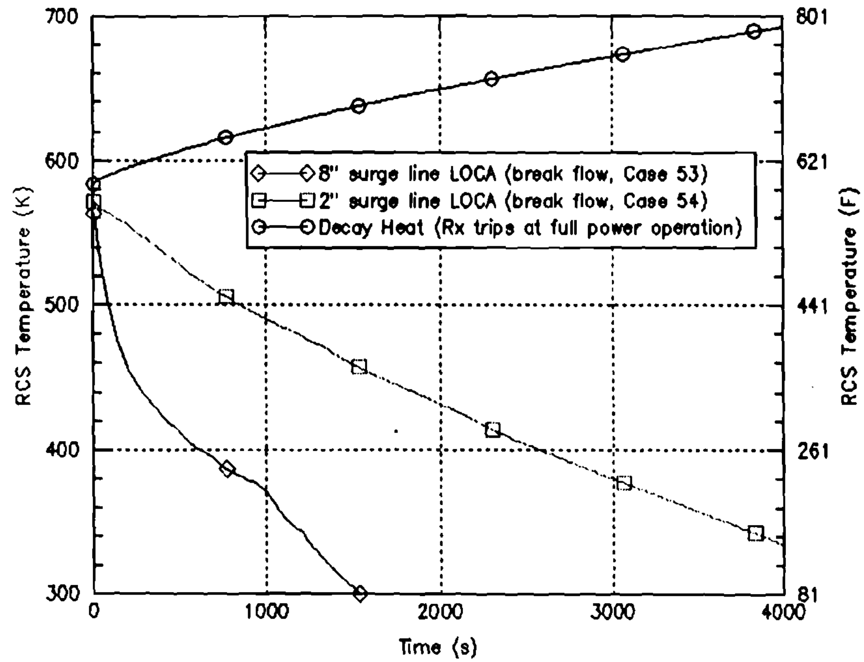
- RCS coolant mass change
  - Primary system breach
  - HPI
- RCS energy change
  - Heat sources
  - Heat sinks
- Short term rapid RCS steam condensation
  - Mixing of core water in downcomer phenomena
  - Boiling-condensation
  - PZR spray

Heat Capacity

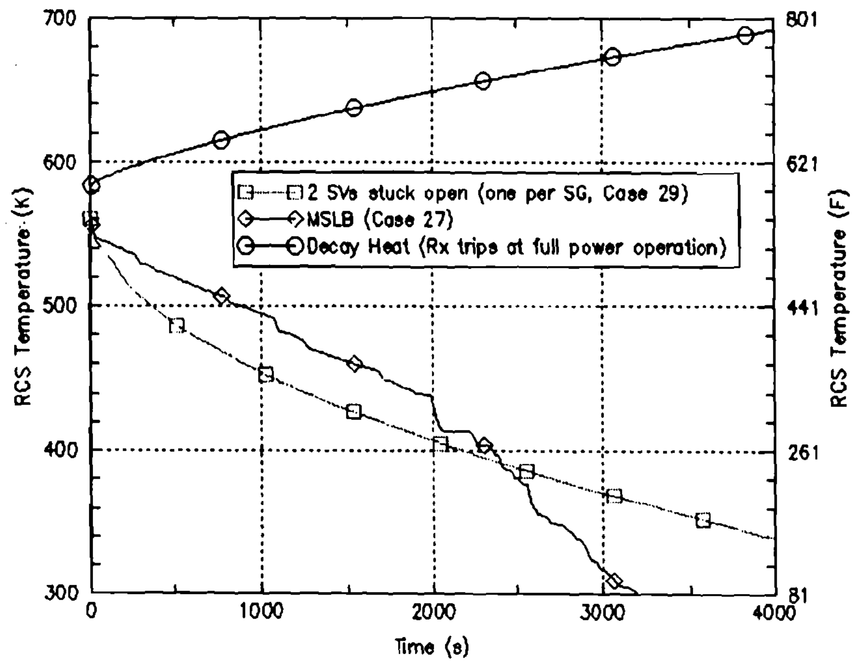
- **Large RCS heat capacity**
  - **requires large heat loss to decrease T_{dc} fast**
- **Heat capacities are significant**
  - **Primary system, between ~1000 and ~1700 MJ/K, depending on steam quality**
  - **Secondary system, between ~120 to ~280 MJ/K in nominal situation, depending on SG level**

Dominant Heat Source and Sinks

Primary System Breach (RCS cumulative enthalpy loss through the primary system break)



Secondary System Breach (RCS cumulative enthalpy loss through the two steam generators)

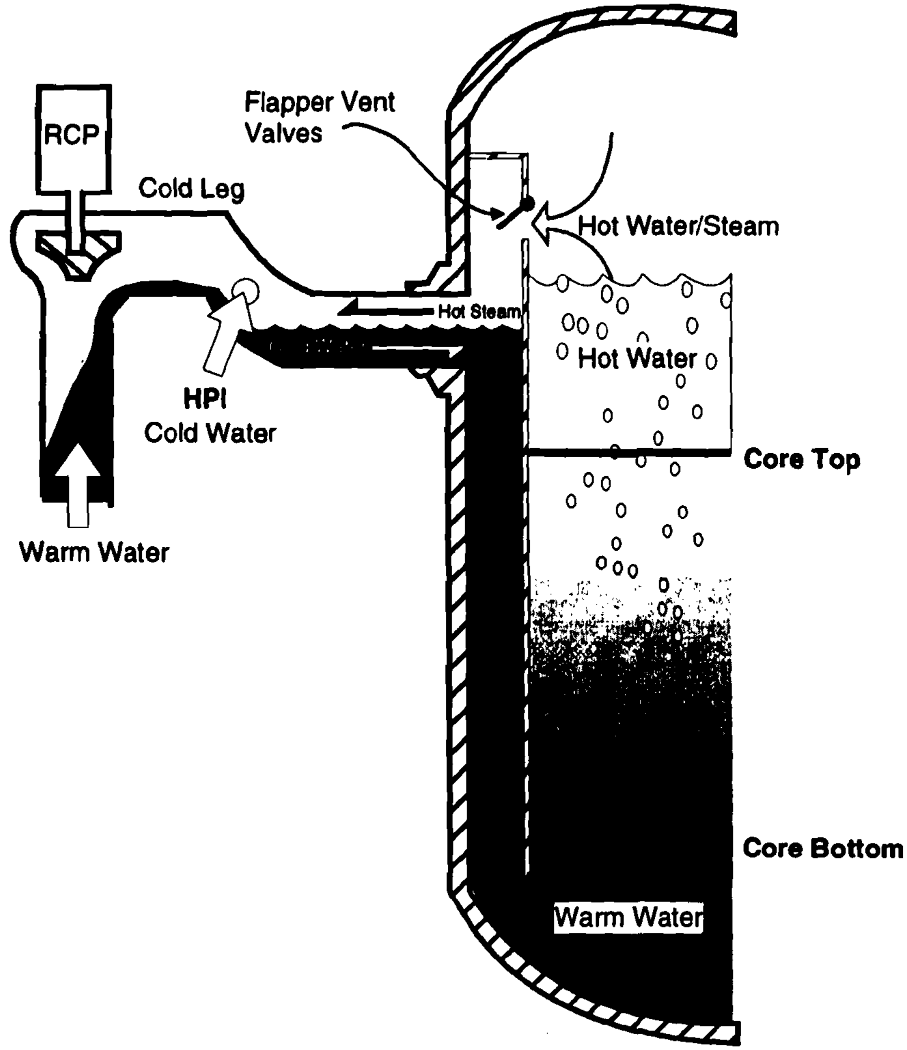


TH Characteristics of PTS

- **TH screening criteria**
  - **Requires rapid downcomer temperature decrease (Cooldown ramp > 100 °F/hr)**
    - ✓ **Only primary system breach and secondary side malfunction (breach or SGs Overfed) can satisfy this criterion alone**
  - **T_{dc} needs to be below ~400 °F**
  - **Transients not screened based on RCS pressure**
- **Observations from analysis**
  - **PTS is more sensitive to T_{dc} than P_{dc}**
    - ✓ **Primary parameter in uncertainty analysis is T_{dc}**
    - ✓ **P_{dc} variation contributing to PTS uncertainty considered only in scenarios involving RCS repressurization**
  - **Downcomer heat transfer coefficient variation has little contribution to PTS risk uncertainty**

Oconeee-Specific TH Characteristics

RPV vent valves:
provide a flow path for hot core steam/water entering the downcomer



Oconee PTS Event Classification Matrix

Primary Side State / Secondary Side State	Intact	Breached: Break Size <~ 1.5" Breach flow could be compensated by HPI	Breached: Break Size > ~1.5" Breach flow cannot be compensated by HPI
Nominal	Not PTS Concern		
One SG Breach			
Two SGs Breach			
SG(s) Overfed			
SG(s) Breach + SG(s) Overfed			

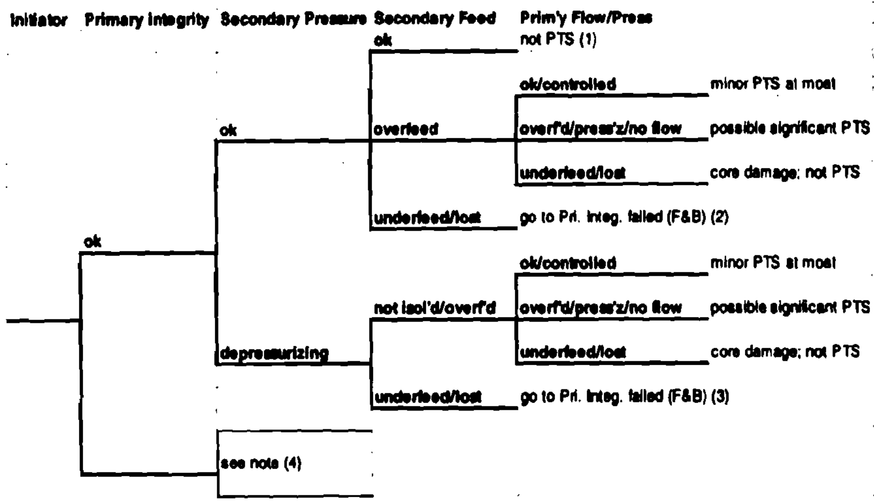
Each cell is further divided into four sub-cells

(1) HPI activated without being controlled
(2) HPI activated and controlled
(3) HPI is failed or not required
(4) HPI fails and is recovered

Note: For break sizes > 1.5-inch, RCPs are tripped, and Secondary and primary sides are decoupled

Convergence of Top-Down and Bottom-up PTS Event Classification Approaches

FIGURE 1. General Functional Event Tree for PTS



(1) not considered a PTS concern regardless of primary flow/pressure
(2) loss of feed to both SGs; procedures call for F&B which is equivalent to entering tree at Primary Integrity "failed"
(3) like (2) above except secondary depressurization has further lowered RCS temp
(4) logic is identical to rest of tree above except choices also exist for Primary Flow/Pressure even for Secondary Pressure and Feed "ok" state and PTS effects are generally potentially greater for all scenarios

Primary Side State / Secondary Side State	Intact	Breached: Break Size <~ 1.5" Breach flow could be compensated by HPI	Breached: Break Size > ~1.5" Breach flow cannot be compensated by HPI
Nominal	Not PTS concern		
One SG Breach			
Two SGs Breach			
SG(s) Overfed			
SG(s) Breach + SG(s) Overfed			

Each cell contains a number of TH runs, and each TH run represents many PRA sequences

(1) HPI activated without being controlled
(2) HPI activated and controlled
(3) HPI is failed or not required
(4) HPI fails and is recovered

Mapping PRA Event Sequences to TH Runs

- **Through an iterative process**
  - **Combined cells based on similarity of PTS-relevant TH behavior (e.g., net impact on T_{dc}) to limit the number of required TH runs**
  - **Identified new TH runs**
- **Mapped PRA event tree sequences (or groups of sequences) to TH runs**
- **Applied sequence frequency and engineering judgment to further screen out sequences with**
  - **Sequence frequency <~1E-8 per year, or**
  - **T_{dc} above 400F within first ~8000 sec and no cooldown ramp >100F/hr**
  - **Criteria later validated by PFM analysis**

Event Category Selected for TH Uncertainty Analysis

Primary Side State / Secondary Side State	Intact	Breached: Break Size <~ 1.5” Breach flow could be compensated by HPI	Breached: Break Size > ~1.5” Breach flow cannot be compensated by HPI
Nominal			6.1e-4
			8.9e-4
One SG Breach		2.7e-7	
	3.7e-6	6.3e-5	
Two SGs Breach		2.8e-7	
	1.7e-5		
		4.9e-8	3.1e-6
		3.3e-6	1.2e-6
SG(s) Overfed			
SG(s) Breach + SG(s) Overfed			
	1.3e-6		

(1) HPI activated without being controlled
(2) HPI activated and controlled
(3) HPI is failed or not required
(4) HPI fails and is recovered

After screening out non-PTS risk sequences, **94%** of the total sequence Frequencies fall in this cell;
TH uncertainty analysis focuses on this dominant cell

The cell is further divided into four categories for TH uncertainty analysis:

1. **PZR SRV stuck open and remains open with valve open area greater than 1.5-inch in diameter**
2. **PZR SRV stuck open and is reseated with valve open area greater than 1.5-inch in diameter**
3. **LOCA between ~ 1.5-inch and 4-inch in diameter**
4. **LOCA between 4-inch and 8-inch in diameter**

Uncertainty Sources

- **Model Uncertainty**
  - **Event Sequence Modeling and Mapping to TH Runs**
    - ✓ **Level of Details in Event Tree Models (e.g, explicit representation of component degraded states)** *[Treated by adding needed details]*
    - ✓ **Assignment of Event Tree Scenarios → TH Bins** *[not treated, believed to be small]*
    - ✓ **Assignment of Representative RELAP Runs → TH Bins** *[Treated explicitly]*
  - **Use of TH Code**
    - ✓ **RELAP5 Internal Modeling Uncertainties** *[important factors treated explicitly]*
    - ✓ **RELAP5 Input Deck Preparation(nodalization)** *[not treated , believed to be small]*
- **Parameter Uncertainty**
  - **All parameters associated with modeling steps, as well as those used within models**
  - **Important parameters treated explicitly**

Treatment of RELAP5 Related Uncertainties

- 1D volume-average calculations validated
  - Included experimental and CFD results
  - Oregon State APEX program
- Empirical correlations
  - Perform uncertainty analyses and sensitivity studies
  - Important correlations treated explicitly

Uncertainty Sources Treated and their Types

Aleatory	Parametric (Boundary Condition) Uncertainty
	➢ Primary side breach size ➢ Primary system breach location ➢ Decay heat ➢ Season ➢ HPI state ➢ HPI flow rate core flood tank pressure

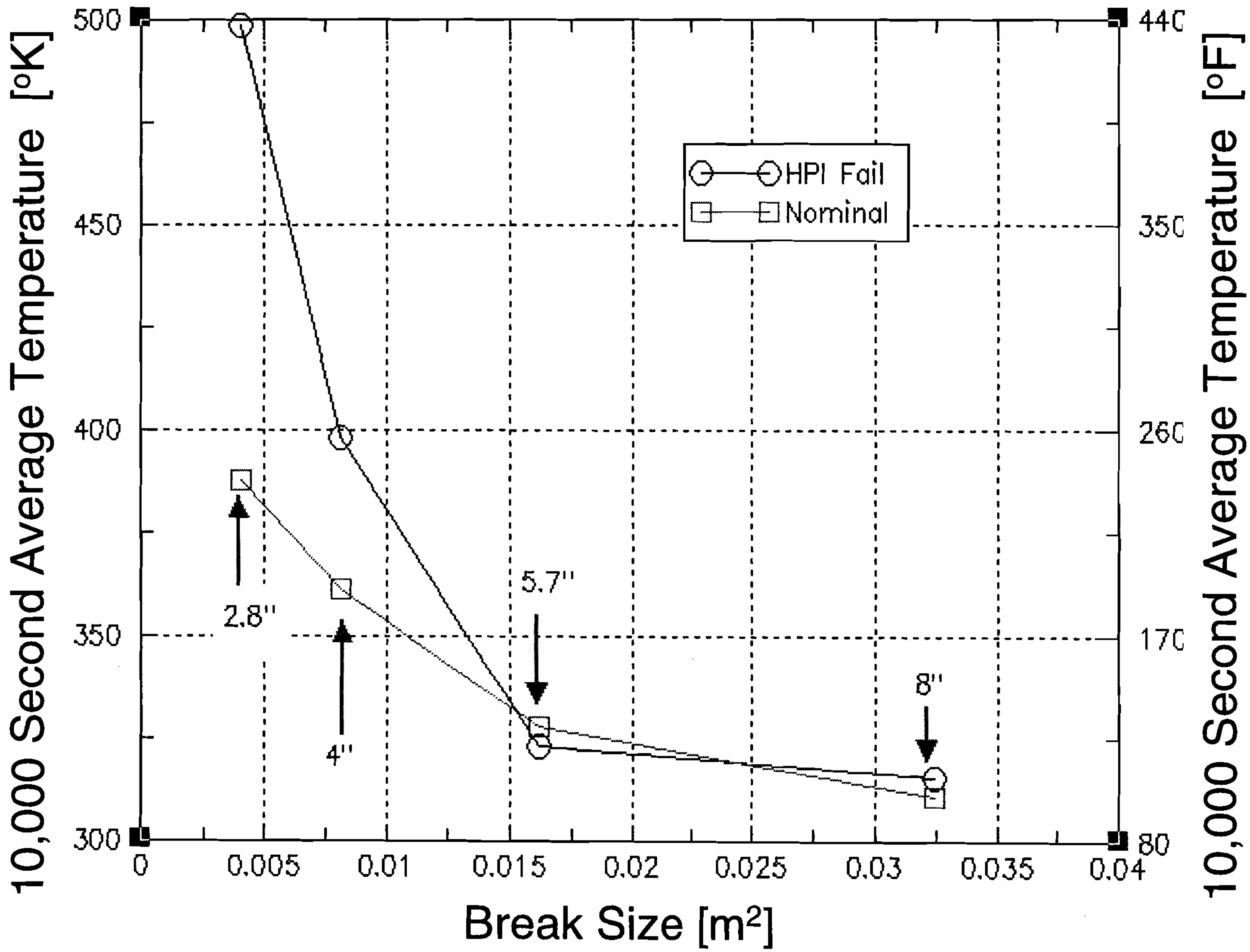
Epistemic	RELAP5 Code Model Uncertainty
	➢ RPV vent valves state ➢ Component heat transfer coefficient ➢ Flow resistance ➢ Break flow rate ➢ Numerical "mixing" (removed by conservatively using a high cold leg reverse flow resistance)

Sensitivity of Tdc to Various Uncertainty Sources

- **Purpose: Determine the individual impact of each factor on Tdc.**
  - **One-factor-at-a-time (1-FAT) method**
  - **Finite discrete probability distribution (DPD) for range of each variable (typically 3-point DPD)**
  - **Use of the average Tdc of the first 10,000 seconds as the measure of the effect**
- **Simultaneous effect of important factors considered subsequently**

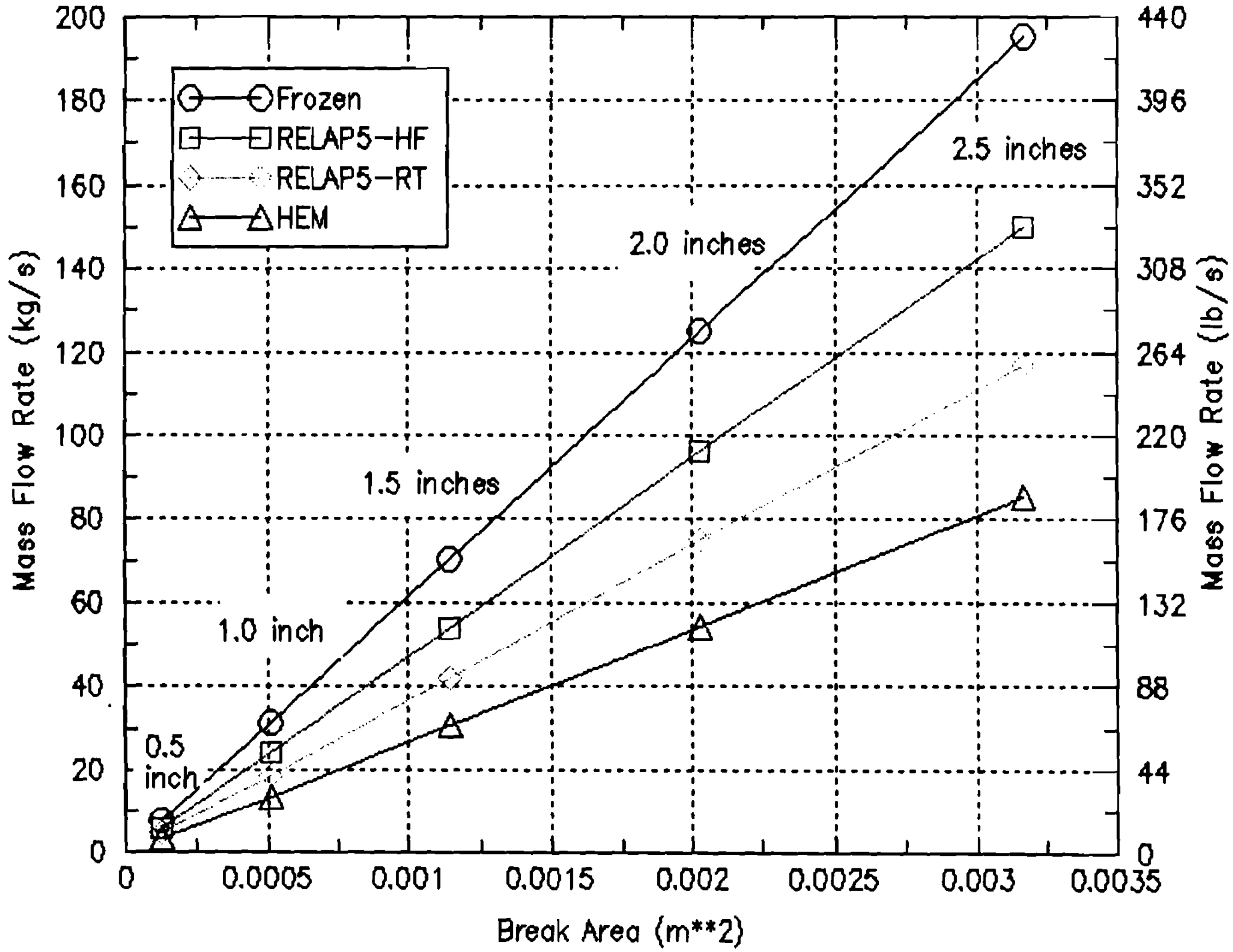
Examples of Sensitivity Study Results

(Impact of HPI State)



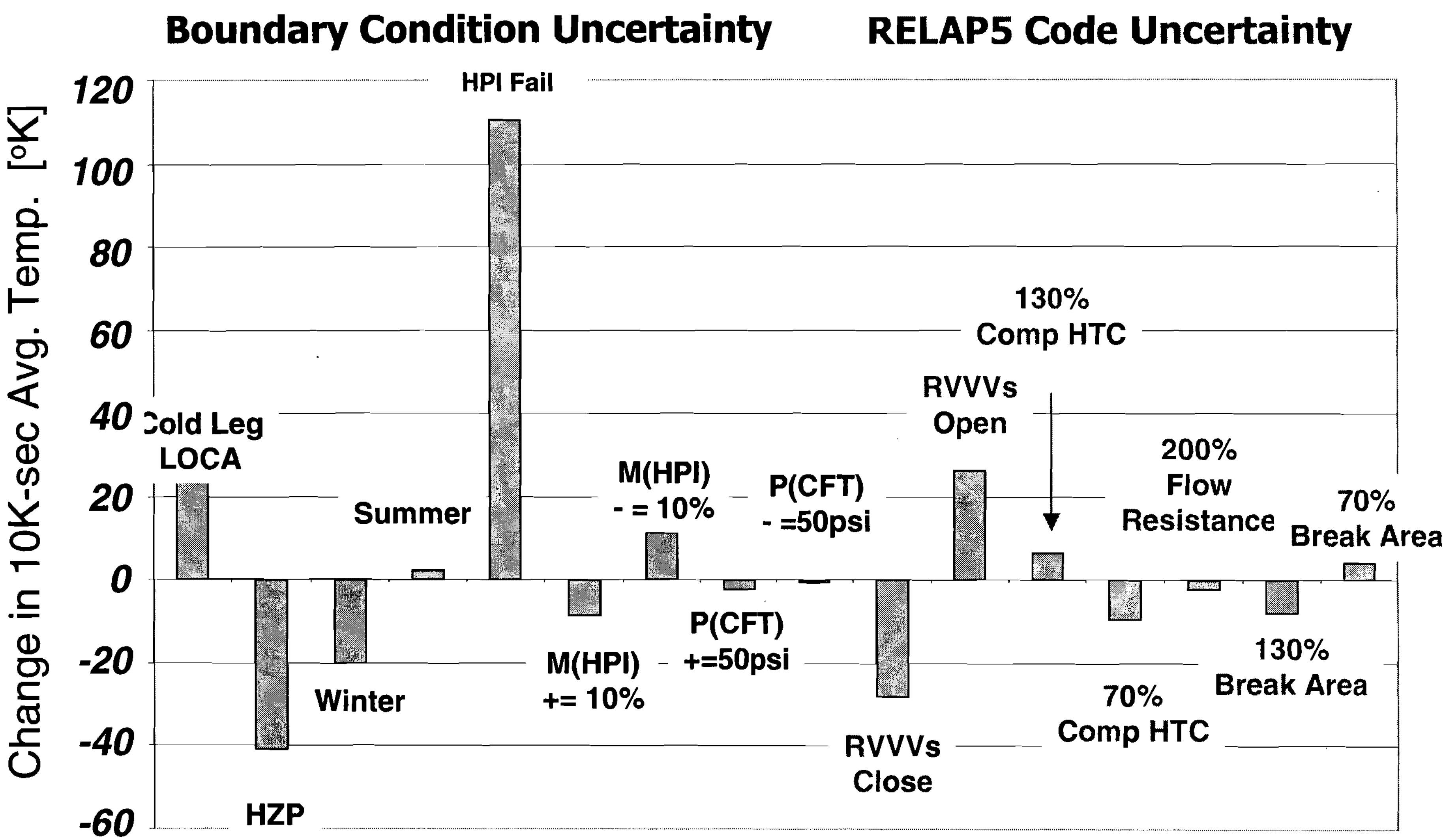
Treatment of Break Flow Rate Model Uncertainty

(Break Upstream Pressure: 7MPa/1028 psia)



Varied break area by +/- 30% to account for model differences

Impact of Various Sources of Uncertainty for Fixed Break Size (2.8-inch Surge Line LOCA)



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Combined Effect of Multiple Sources of Uncertainty (1/3)

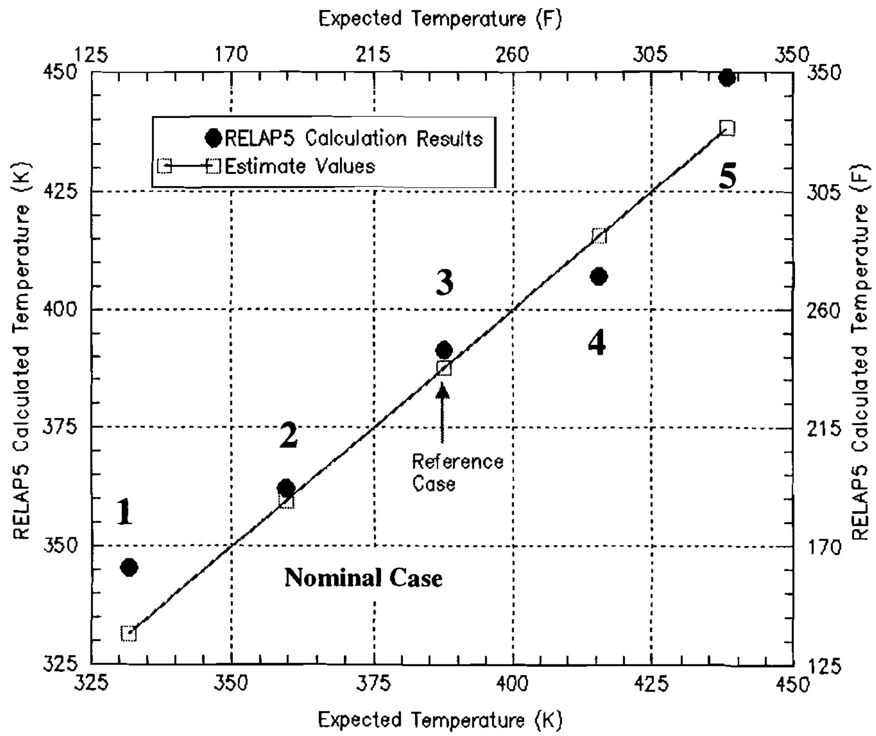
1. **Selected the most influential source of uncertainty**
2. **For a given value of the selected variable (in Step 1), varied other uncertainty variables shown to have significant impact in the sensitivity analysis**
3. **For the selected variables, considered all possible combinations**
4. **For each combination, calculated the net effect of Tdc using additive assumption**
5. **Corresponding probability was calculated as**

$$T = \sum \Delta T_i + T_{Nominal}$$

$$Prob(T) = \prod Prob(\Delta T_i)$$

Verification of Linear Additively Assumption

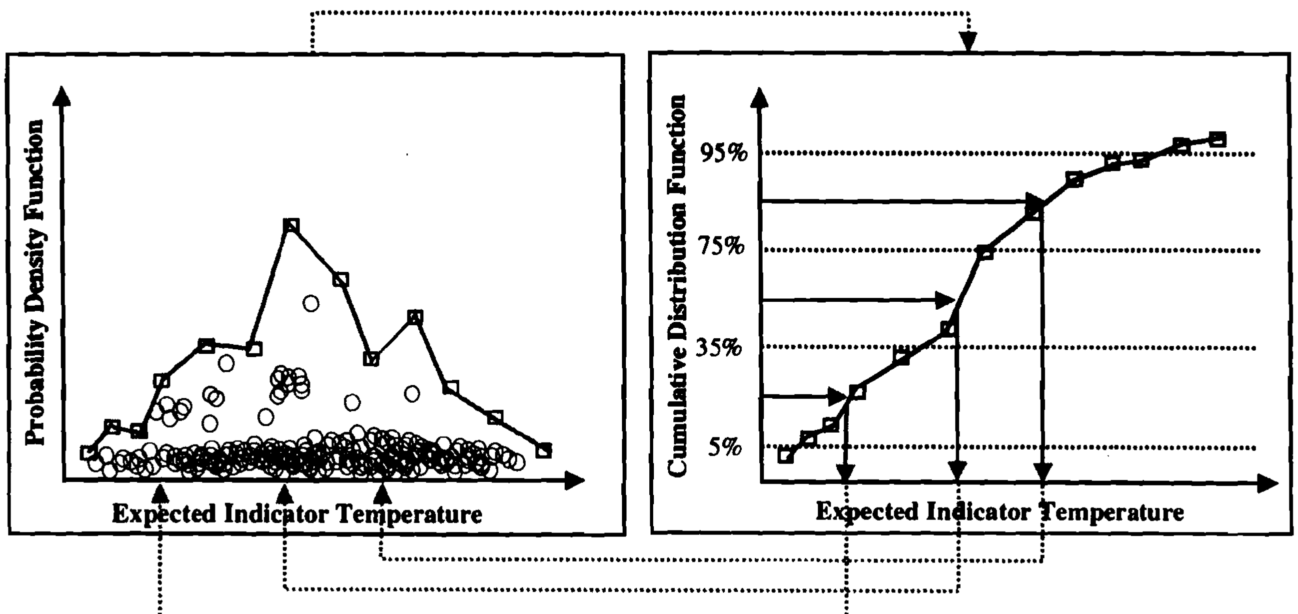
(2.8-Inch Surge Line LOCA)



#	Event Description	Expected Avg. Tdc (K)	RELAP5 Cal. Avg Tdc (K)
1.	Winter; p(CFT) + 50 psi; 70% A_{brk}; RVVVs Close; 70% HTC	332	345
2.	Summer; RVVVs Close; 200% flow resistance	360	362
3.	p(CFT) + 50 psi; 110% m(HPI); 70% A_{brk}; 130% HTC	387	391
4.	Summer; p(CFT) + 50 psi ; 90% m(HPI); 130% Abrk ; RVVVs fully Open; 200% flow resistance	415	406
5.	Summer; 90% m(HPI); 90% Abrk; RVVVs fully Open; 130% HTC	438	449

Combined Effect of Multiple Sources of Uncertainty (2/3)

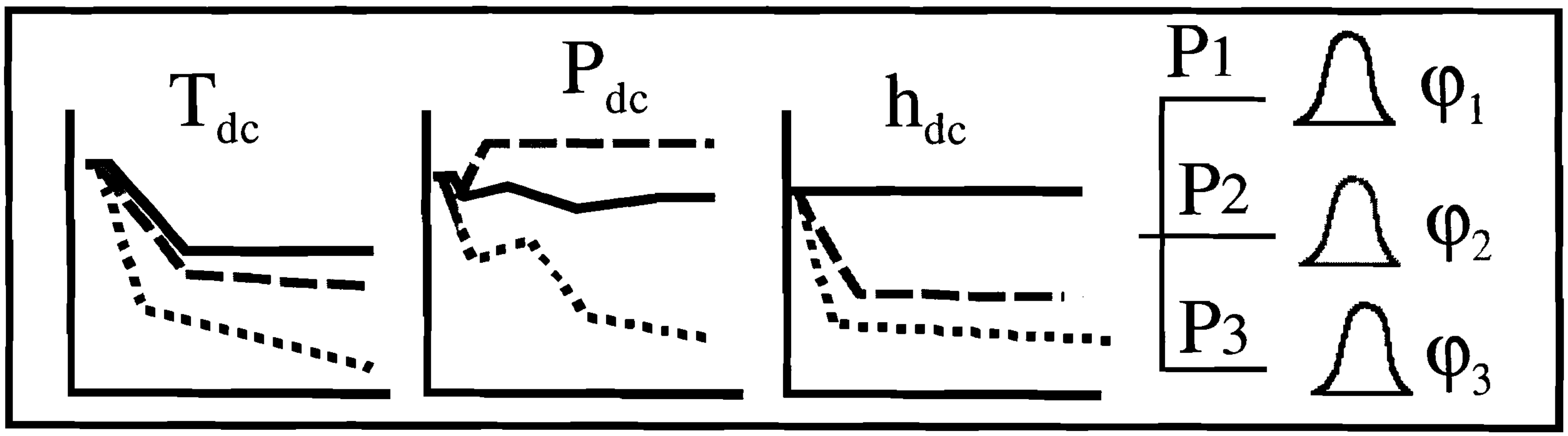
6. **Repeated Steps 2-5 for all other values of the selected source of uncertainty in Step 1**
7. **Plotted the CDF of the resulting Ave Tdc values for all combinations of variables**
8. **Discretized the CDF into a finite number of representative Ave Tdc (typically 3-5) and corresponding probability mass : {Tdc(i) ; p(i) }**



Combined Effect of Multiple Sources of Uncertainty (3/3)

9. **Selected matching TH runs for each Ave Tdc point, and performed TH runs to generate time traces of Tdc, P, and h**
10. **Calculated frequency uncertainty distribution for TH runs in Step 9 as**

$$\varphi_i = \theta \cdot p_i$$



List of Results Generated as Input to PFM Analysis (1/2)

Secondary Side State \ Primary Side State	Intact	Breached: Break Size <~ 1.5" Breach flow could be compensated by HPI	Breached: Break Size > ~1.5" Breach flow cannot be compensated by HPI
Nominal	**Not PTS Concern**		[8.3e-6] 55 (2.828" surge line) [4.9e-5] 141 (4" + 30% A_{brk} surge line) [4.3e-5] 142 (4" - 30% A_{brk} surge line) [4.9e-5] 143 (4" Cold Leg) [6.3e-5] 144 (2.828" surge line, A_{brk} -= 30%, VV Closed) [6.3e-5] 145 (1.5", A_{brk} += 30%, RCPs trip) [1.0e-4] 146 (PZR-SRV, 2.54", A_{brk} += 30% + Summer + VV Close) [9.0e-5] 147 (PZR-SRV, 2.54", summer) [1.0e-4] 148 (PZR SRV Stuck open area = 1.5", Comp HTC *= 130%, RCPs trip) [1.0e-5] 109(PZR-SRV reseated at 100 minutes) [1.0e-5] 149 (PZR SRV reseated at 50 minutes) [7.2e-6] 52 (5.656" surge line) [8.3e-6] 53 (8.0" surge line)
			[4.3e-4] 112 (PZR SRV SO, SRV reseated at 100 min, HPI throttled 1 min after 5F subcool and 100" PZR level) [1.5e-5] 113 (PZR SRV SO. SRV reseated at 100 min, HPI throttled 10 min after 5F subcool and 100" PZR level) [4.3e-4] 114 (PZR SRV SO. SRV reseated at 50 min. HPI throttled 1 min after 5F subcool and 100" PZR level) [1.6e-5] 115 (PZR SRV SO. SRV reseated at 50 min, HPI throttled 10 min after 5F subcool and 100" PZR level)
One SG Breach		[5.6e-8] 8 (1" surge line + 1 SG SV SO) [1.0e-7] 28 (F&B, 1SG SV SO) [1.1e-7] 30 (#28 + HZP)	
	[2.1e-6] 27(MSLB) [4.0e-7] 101(#27 + HZP) [1.2e-6] 37 (1 SG SV SO + HZP)	[4.8e-7] 12 (1" surge line, 1SG SV SO) [7.0e-7] 90 (2 SG SVs SO, HPI throttled @ 20 min after it can be throttled) [2.1e-7] 102 (#90 + HZP) [6.1e-5] 91 (SGA TR+ 1SGB SV SO and reseated @ 10 min after initiation + RCP tripped @ 1 min + HPI throttled @ 10 min after it can be throttled) [5.0e-8] 103 (#91 + HZP) [2.3e-7] 99 (MSLB + HPI throttled 20 min after it can be throttled) [2.3e-7] 100 (#99 + HZP)	

Uncertainty Cases

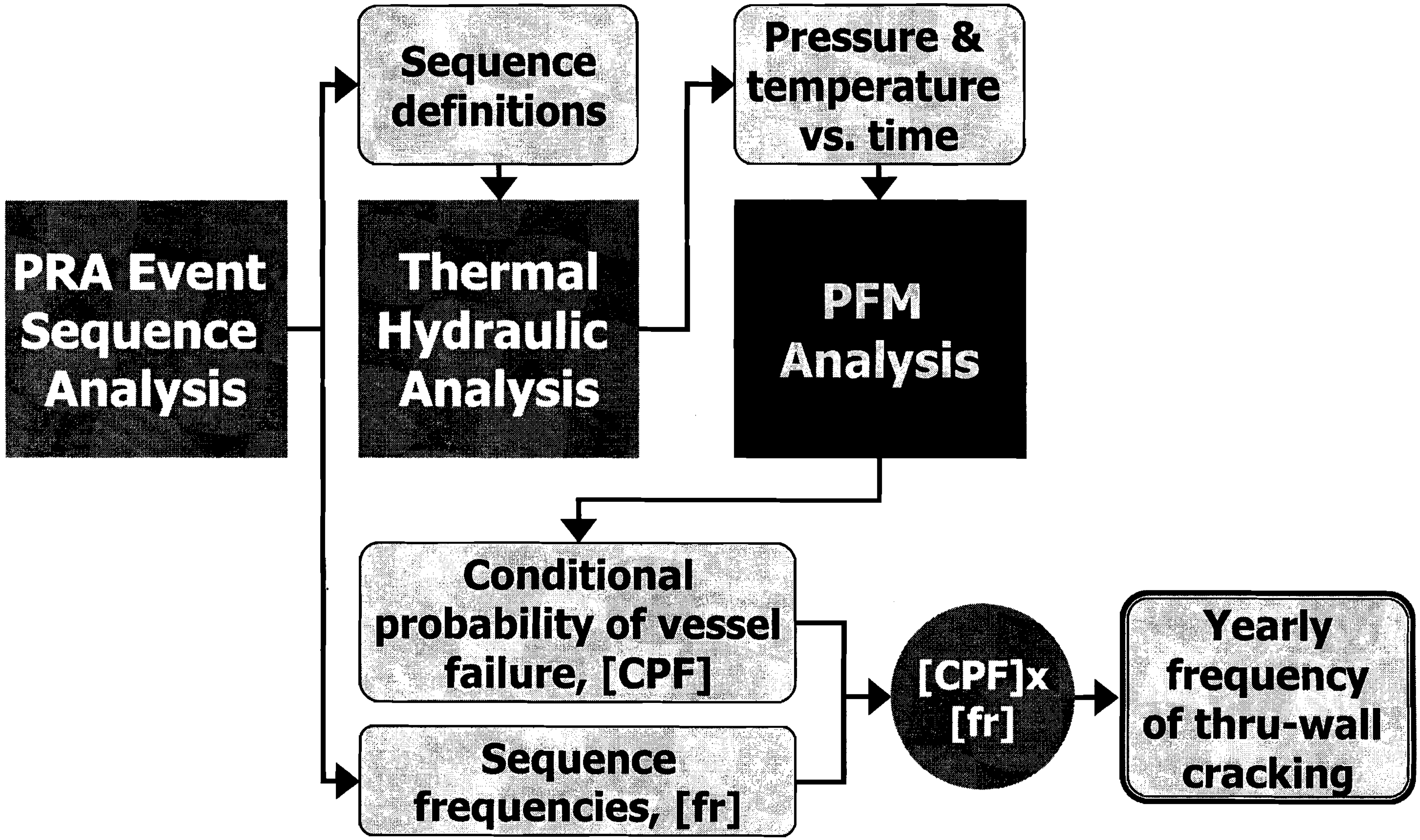
List of Results Generated as Input to PFM Analysis (2/2)

Secondary Side State \ Primary Side State	Intact	Breached: Break Size <~ 1.5" Breach flow could be compensated by HPI	Breached: Break Size > ~1.5" Breach flow cannot be compensated by HPI
Two SGs Breach		**[2.7e-7]** 29 (2 SG SVs SO) **[5.0e-9]** 31 (#29 + HZP)	
	[1.4e-5] 36 (2SVs SO) **[2.6e-6]** 38 (#36 + HZP)		
		[3.1e-8] 15 (1" + 4 TBVs fully SO + No HPI) **[1.8e-8]** 119 (#15 + HZP)	**[3.1e-6]** 110 (2" surge line, 4 TBVs opened @ 15 min)
		[2.7e-8] 44 (1" LOCA + HPI F&R @2250s, 4 TBVs fully open) **[1.3e-7]** 120 (#44 + HZP) **[3.1e-6]** 111 (1" + 4 TBVs Opened @ 15 min, HPI recovered when CFTs are 50% discharged, HPI throttled @ 50 min)	**[2.4e-7]** 116 (PZR SRV SO, HPI fail, 4 TBVs opened @ 15 min, HPI was recovered when CFT are 50% discharged; HPI was throttled @ 20 min after available) **[4.2e-8]** 125 (#116 + HZP) **[7.4e-7]** 117 (PZR SRV SO, HPI fail, 4 TBVs opened @ 15 min, HPI was recovered when CFT are 50% discharged; SRV reseated 5 min after HPI was recovered, HPI throttled 1 min after available). **[1.3e-7]** 126 (#117 + HZP)
SG(s) Overfeed			
SG(s) breach + SG(s) Overfed			
	[1.2e-6] 89 (F&B + 4 TBVs are opened and HPI is throttled after RCS pressure reaches 2275 psi) **[6.6e-8]** 98 (#89 + HZP)		

Thermal Hydraulics Summary

- **Advances in computing allow much more extensive analysis, including uncertainty evaluation**
- **Convergence of top-down and bottom-up approaches to arrive at uncertainty evaluation**
- **Dominant uncertainty sources identified**
- **Uncertainty of multiple sources combined effects quantified**

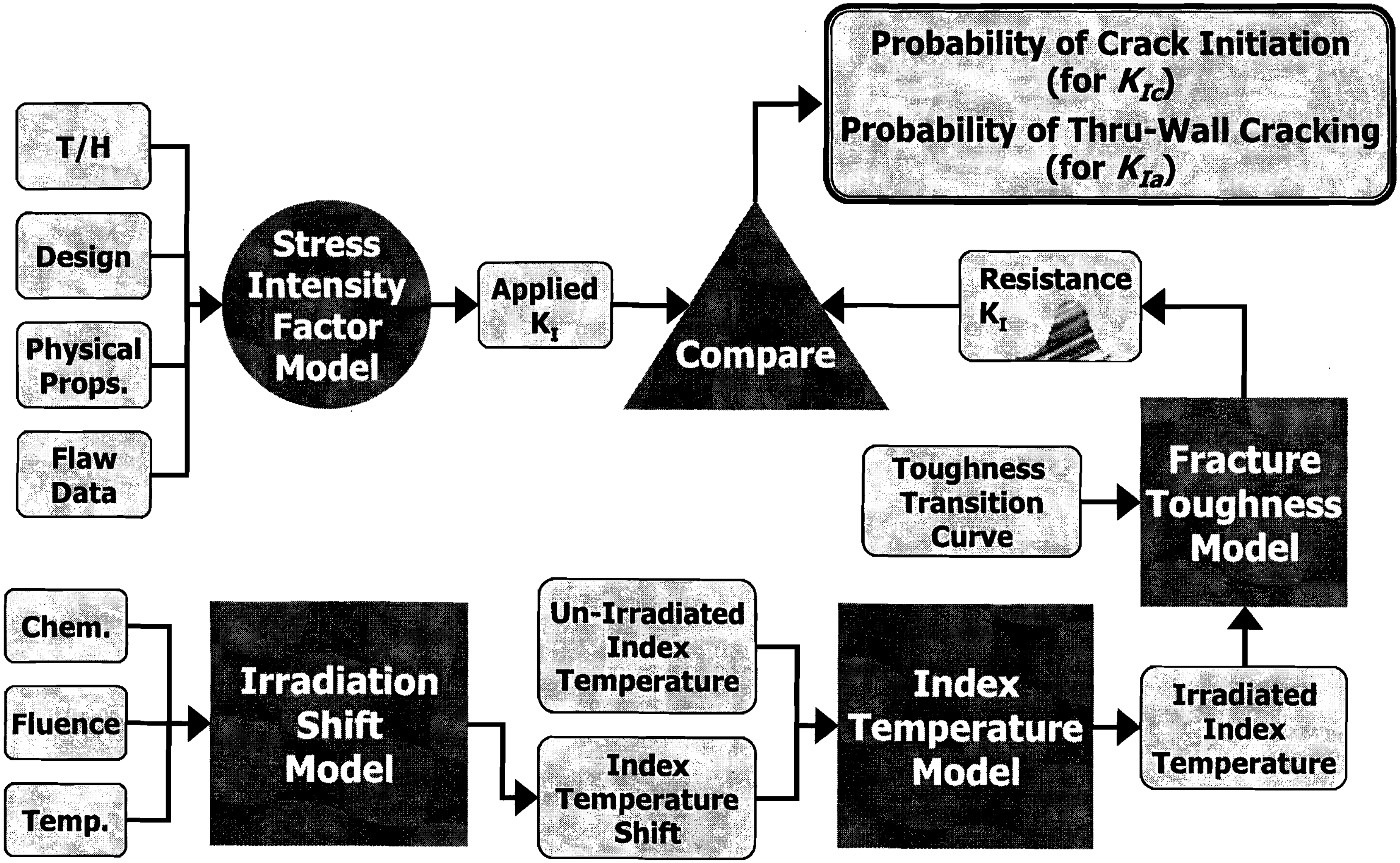
PFM in the Overall Process



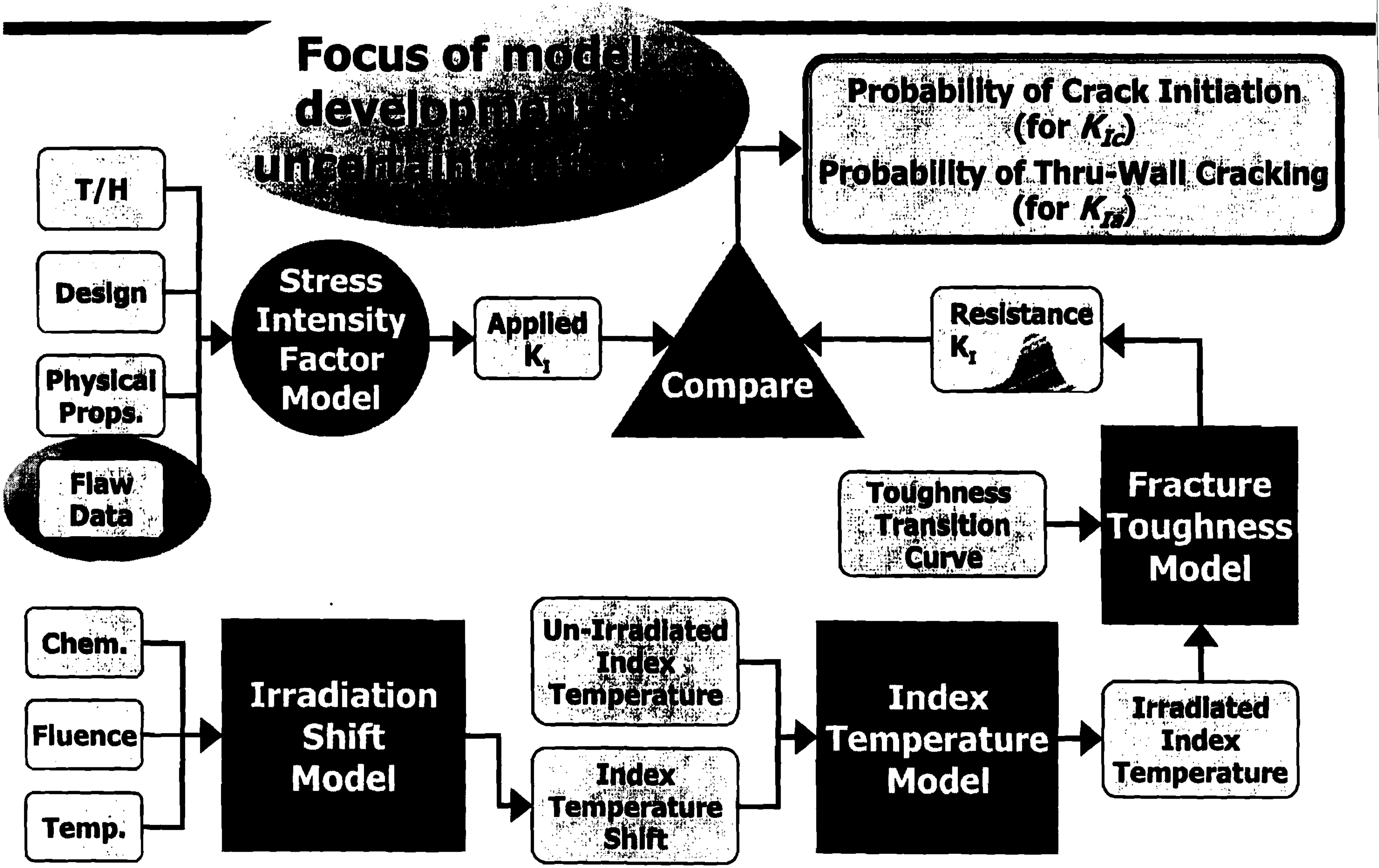
Probabilistic Fracture Mechanics Expanded

PFM
Analysis
(FAVOR)

Probabilistic Fracture Mechanics Expanded



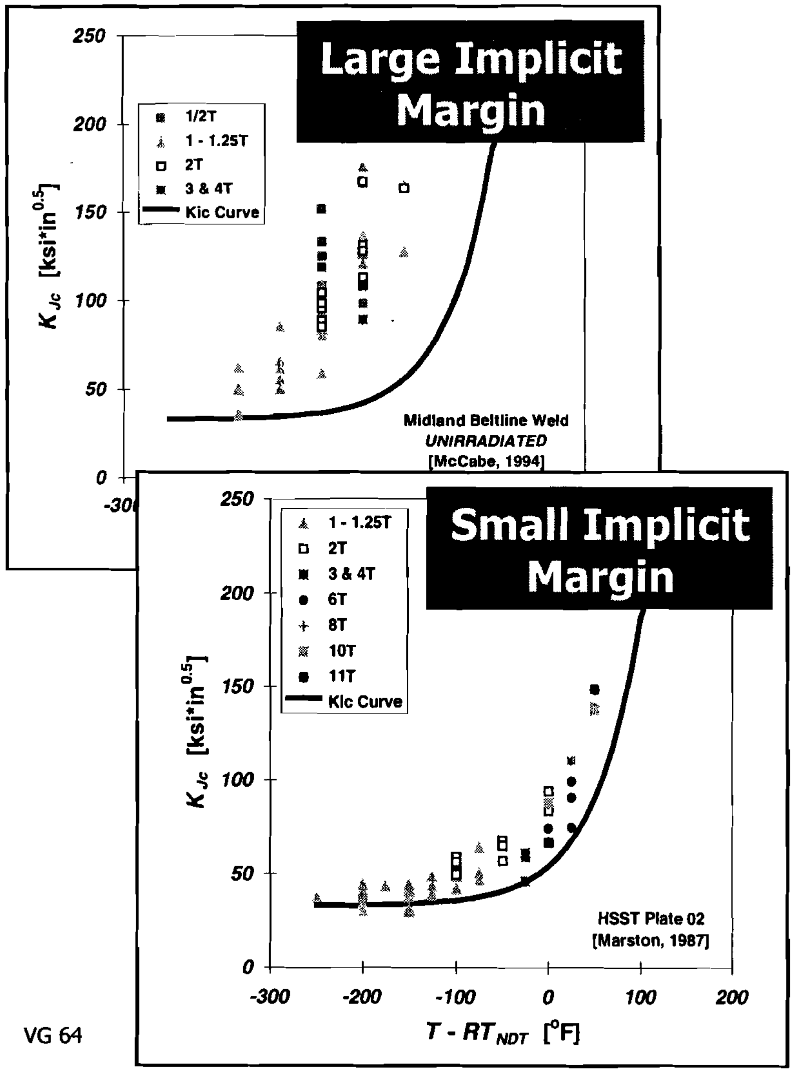
Probabilistic Fracture Mechanics Expanded



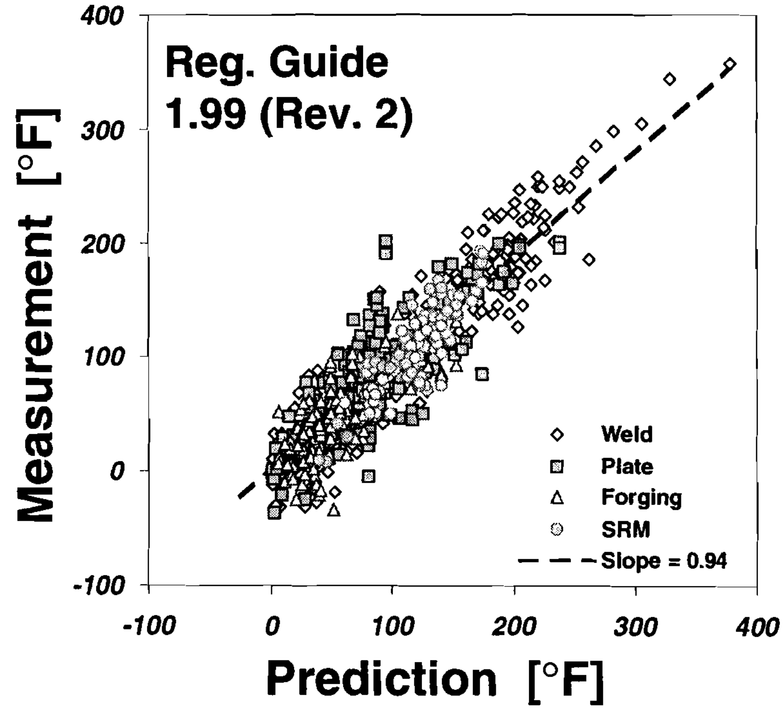
VG 63

Current Toughness & Embrittlement Models

Toughness Model



Embrittlement Model



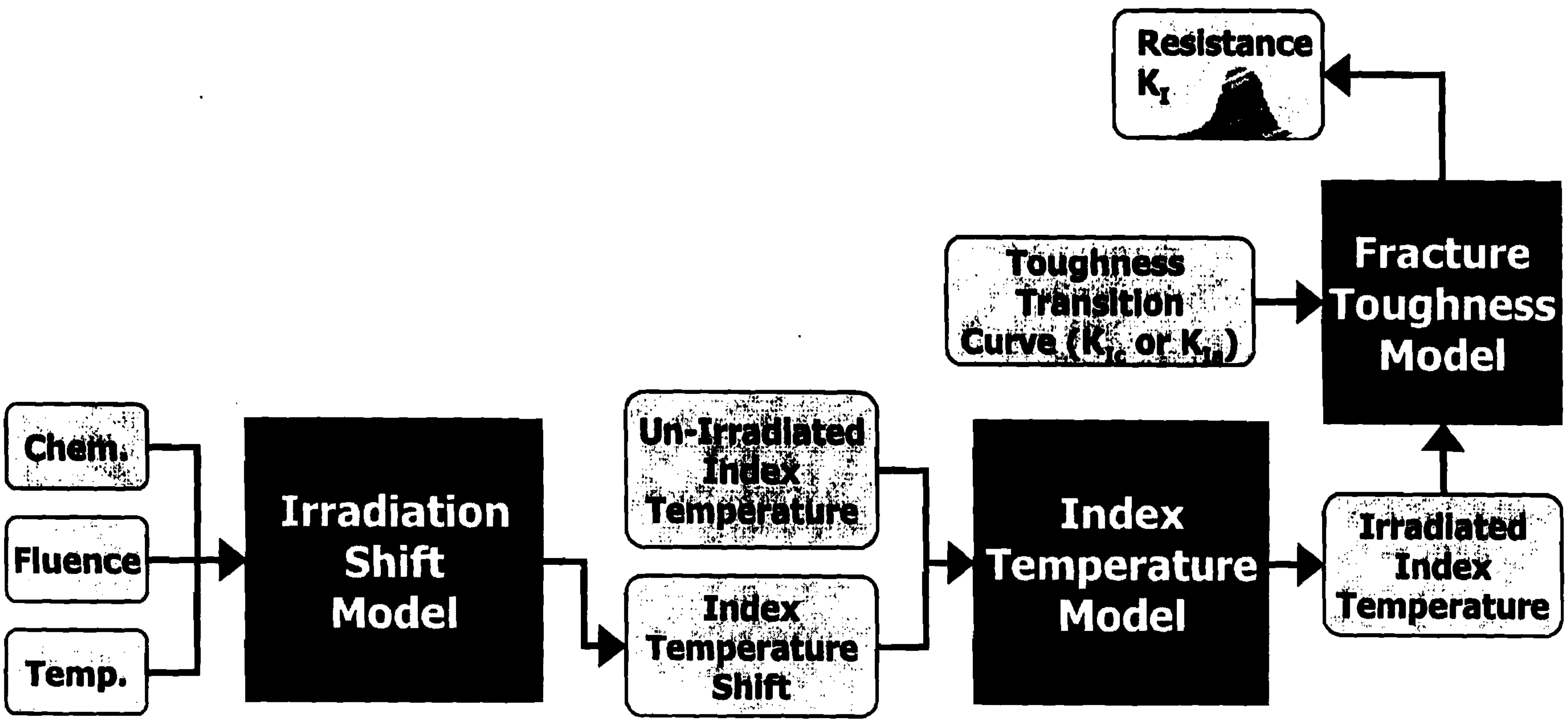
$$\Delta T_{30} = (CF)(\phi t)^{(0.28-0.11\log(\phi t))}$$

- ✓ **Non-toughness data *assumed to* represent toughness data**
  - → **Uncertainties treated *implicitly***
  - → **Uncertainties types *mixed***

Toughness & Embrittlement Models

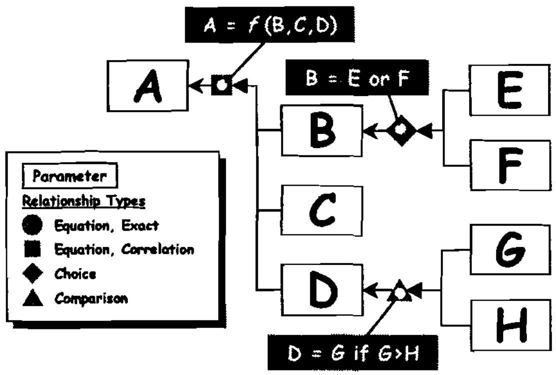
→ *Constraints & Fundamental Assumptions* ←

- **Retain LEFM basis for FAVOR**
- **No new material measurements required of licensees**



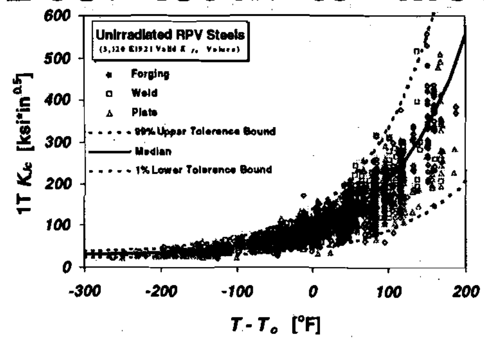
Process for Model Building & Uncertainty Characterization

Root Cause Diagrams



✓ Depict current process
✓ Identify uncertainties

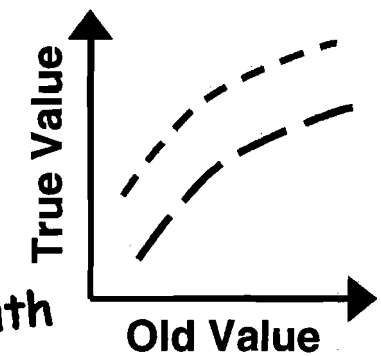
Physically Motivated "Best Estimate" Model



✓ Classify uncertainty type
- ➢ Aleatory
- ➢ Epistemic

A Means to Account for Uncertainty

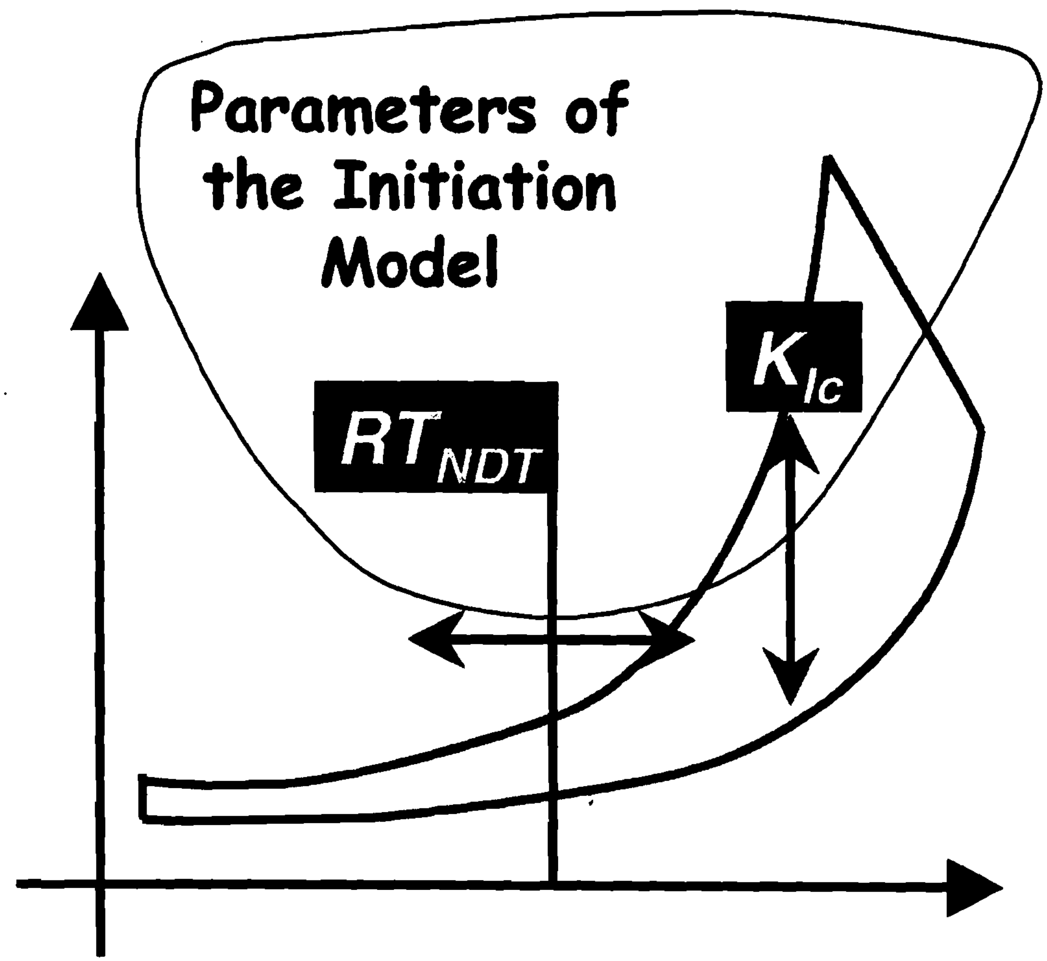
Fracture Toughness Data Is Truth



A recommended *FAVOR* procedure …

… is needed for RT_{NDT}, K_{Ic}, ΔT_{30}, K_{Ia}, ΔRT_{Arrest}

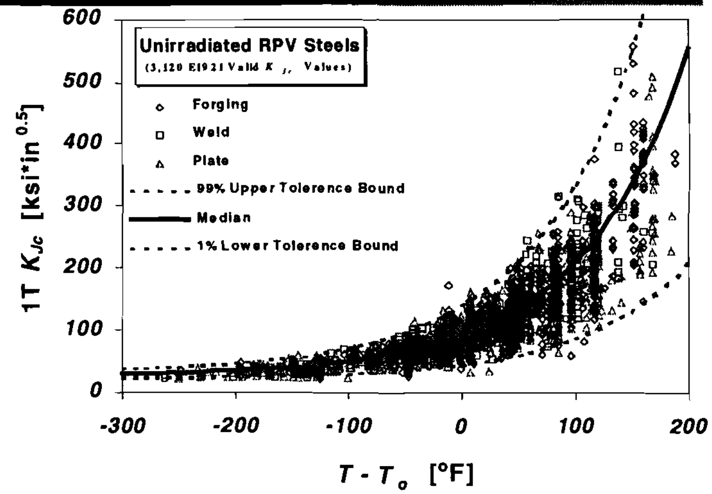
Initiation Fracture Toughness Model



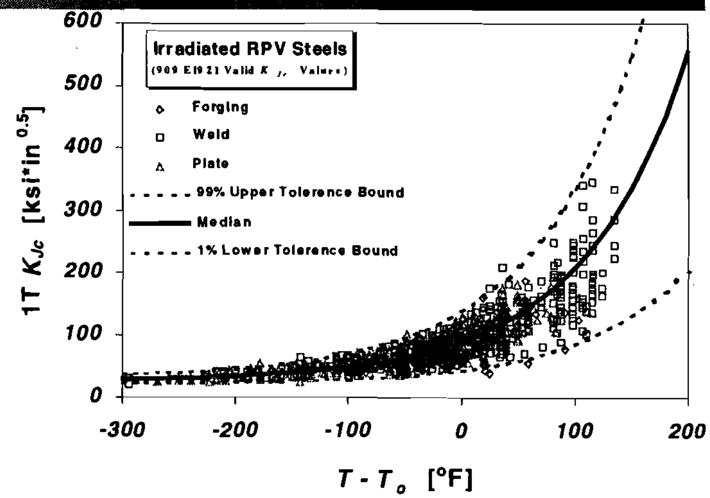
Initiation Fracture Toughness
(Best Estimate Model)

- **Physical understanding suggests**
  - **Common T-dependence**
  - **Common scatter**
  - **Irradiation produces shift only**
- **<u>Best Estimate</u>: The Master <u>Curve method</u> (with T_o transition temperature)**
  - **Physically based**
  - **Empirically validated**
  - **Weakest link statistics account explicitly for cleavage process**
  - **T_o defined consistently for all steels**
    - ✓ **Temperature at 100 MPa$\sqrt{m}$**
    - ✓ **Corresponds to the position of the data instead of a representation of data**

Un-Irradiated



Irradiated

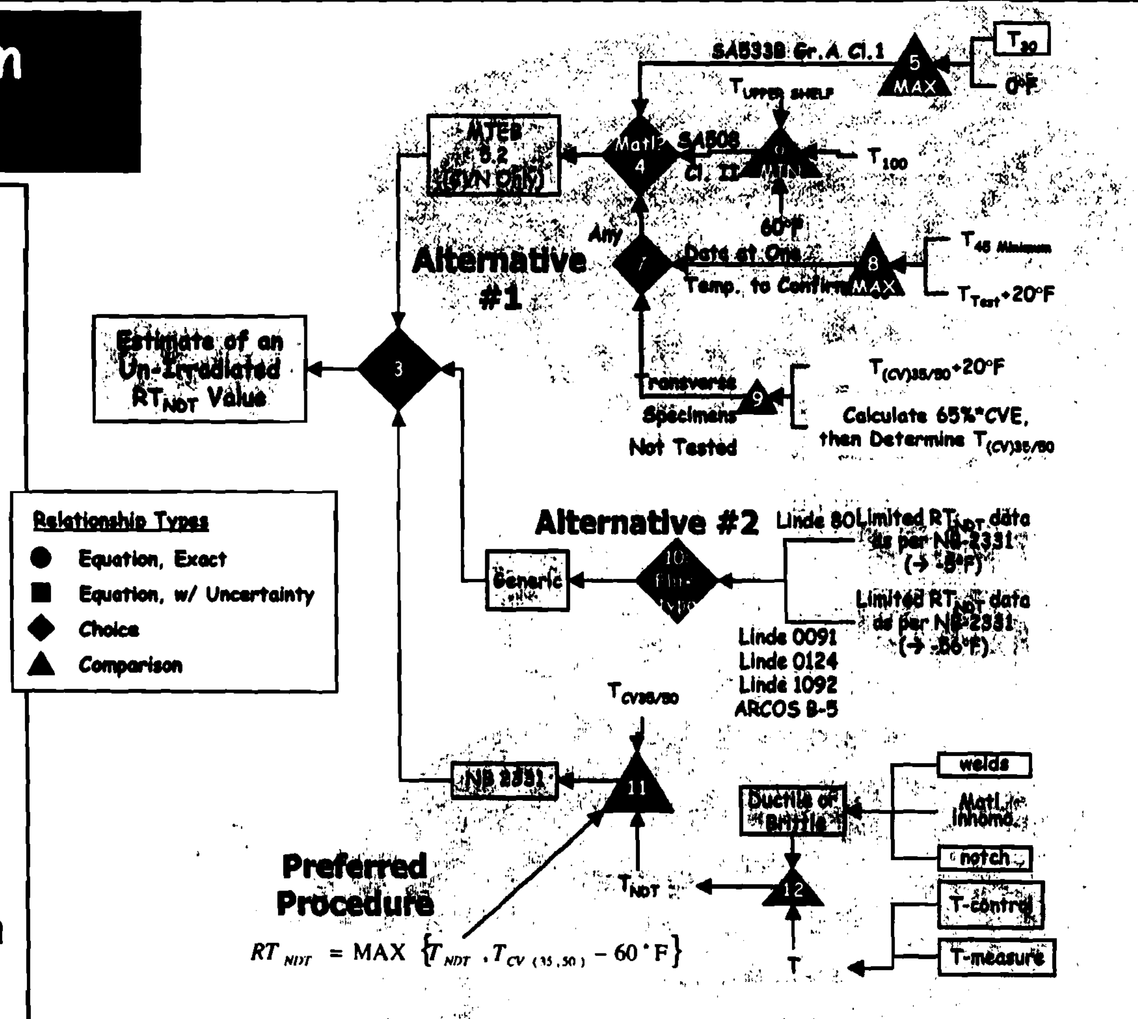


Current Transition Temperature Model: RT_{NDT}

Uncertainty Identification and Classification

Epistemic

- **Conservative bias of NB-2331**
- **The myriad of methods and transition temperatures used to define RT_{NDT}**
- **The limited data sets used to define Generic RT_{NDT} values and to assess the appropriateness of the various transition temperatures used in RT_{NDT} definitions**
- **The lack of prescription in the test methods**
- **Neither *NDT* nor *CVN* are material properties**
- **Material variability**



RT_{NDT} uncertainty should be modeled as epistemic

Current Initiation Fracture Toughness Model: K_{Ic}

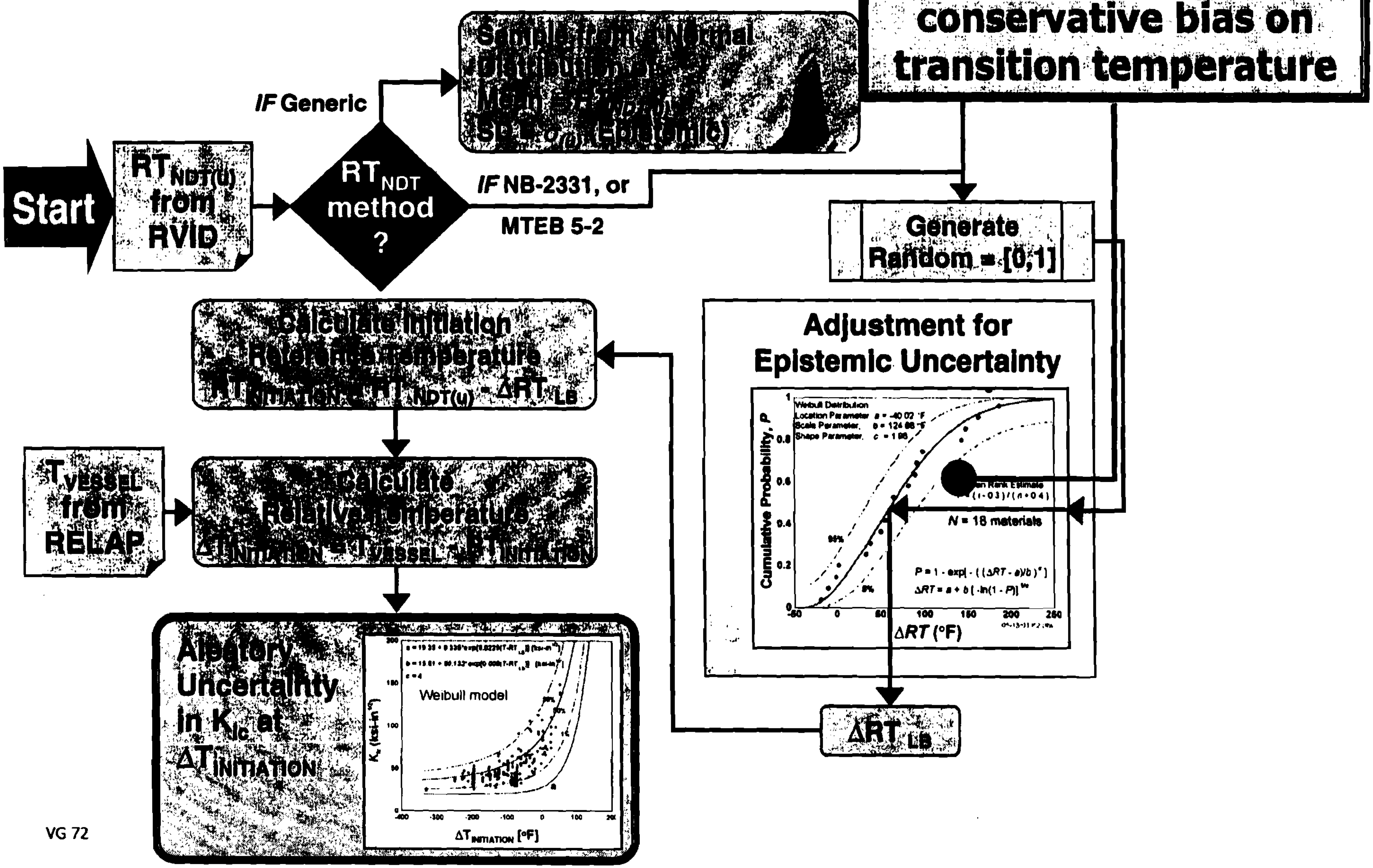
Uncertainty Identification and Classification

- **A physical understanding of the cleavage fracture process demonstrates that non-coherent particles (& other barriers to dislocation motion) are *alone* responsible for the scatter in K_{Ic}**
- **This physical understanding coupled with the ideas that:**
  - **K_{Ic} does not exist as a point property (associated length scale)**
  - **Both non-coherent crack initiating particles and postulated flaws are randomly distributed throughout the vessel**
  - **This distribution occurs over a size scale below that considered by a K_{Ic} toughness model**

... suggests that K_{Ic} uncertainty should be modeled as aleatory

FAVOR Model of Initiation Fracture Toughness

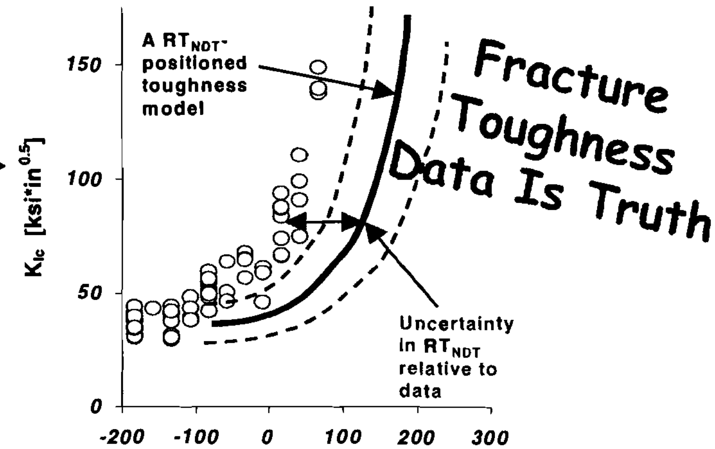
Previous (circa-1980s) PTS analyses assumed a significant conservative bias on transition temperature

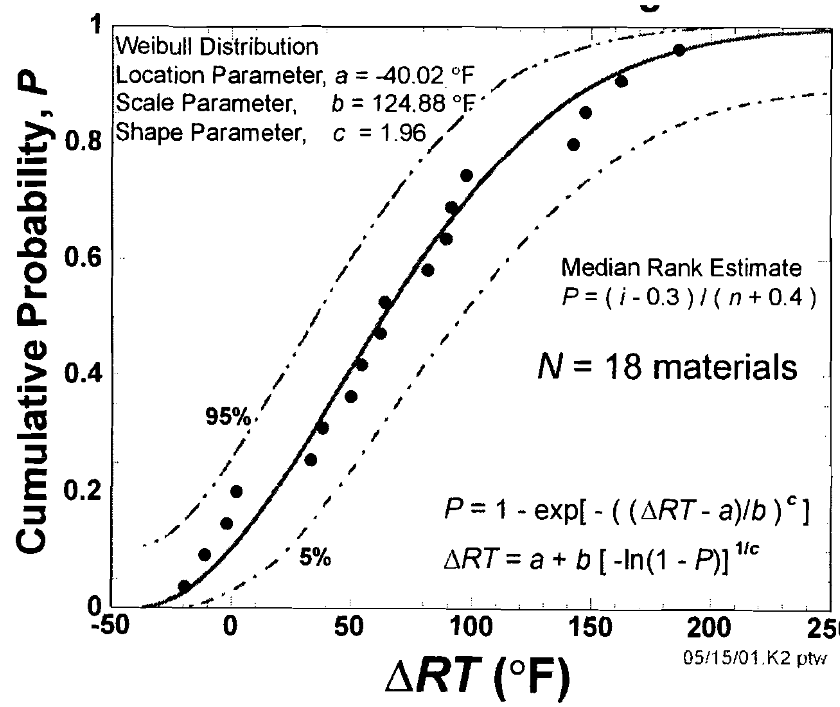


RT_{NDT} Bias Correction

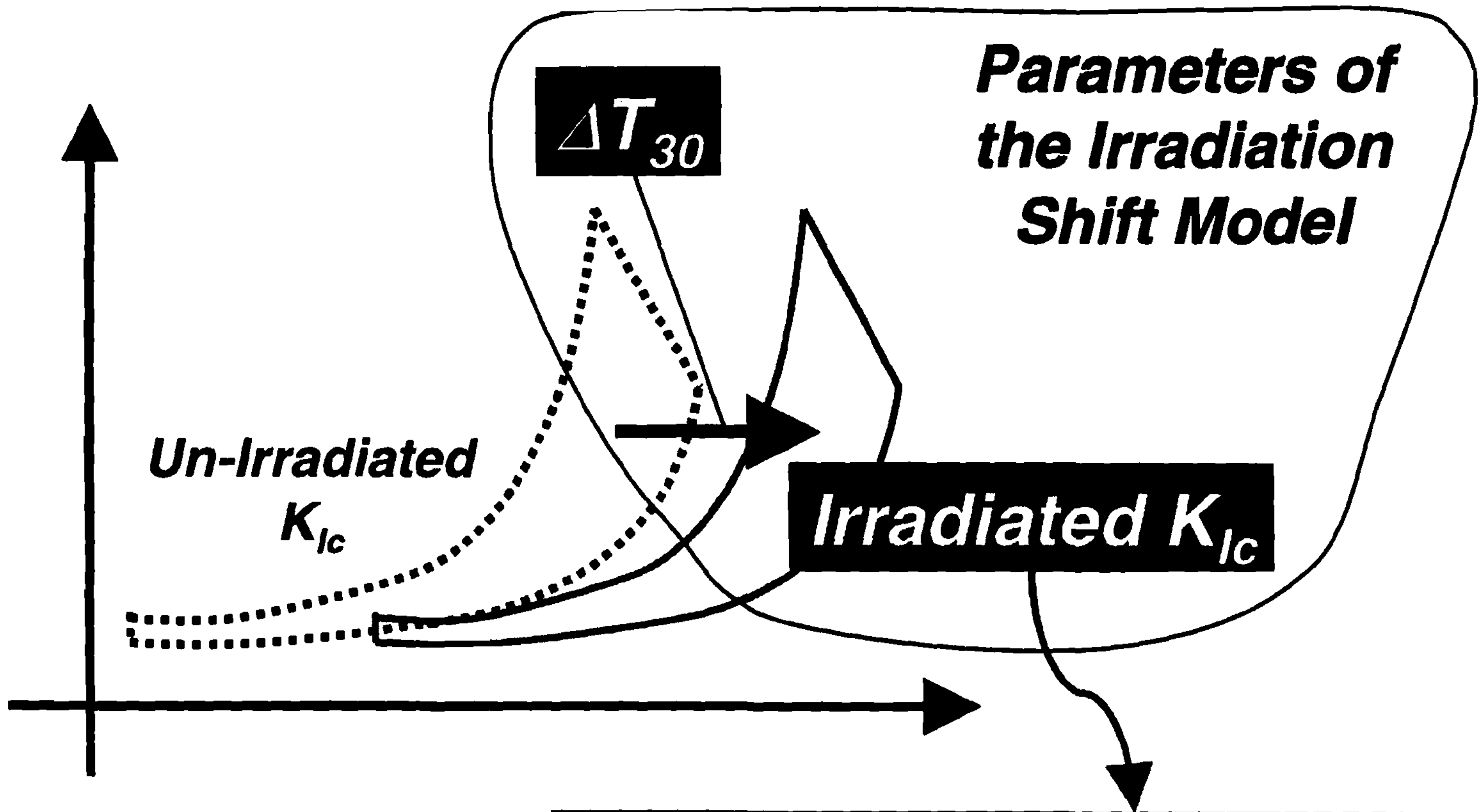
Uncertainty Quantification

- **Quantify how far off RT_{NDT} is from an accurate representation of real toughness data**
  - **Using a consistent representation of that data**
- **T_o best represents "true" fracture toughness transition data**
- **Adjustment based on CDF of $\Delta RT = RT_{NDT(u)} - T_o$**
- **ΔRT accounts for *all* known epistemic uncertainties**





Irradiation Shift Model



Temperature dependence and scatter already demonstrated to be the same as un-irradiated K_{Ic}

Current Model

Uncertainty is Epistemic

Matrix Hardening	Age Hardening

$$\Delta T_{30} = A\exp\left(\frac{19310}{T_{irr}+460}\right)(1+110P)(\phi t)^{0.4601} + B\left(1+2.40Ni^{1.250}\right)f(Cu)g(\phi t) + Bias$$

$$A = \begin{cases} 8.86\text{x}10^{-17}, & \text{welds} \\ 9.30\text{x}10^{-17}, & \text{forgings} \\ 12.7\text{x}10^{-17}, & \text{plates} \end{cases}$$

$$B = \begin{cases} 230, & \text{welds} \\ 132, & \text{forgings} \\ 206, & \text{plates in CE vessels} \\ 156, & \text{other plates} \end{cases}$$

$$f(Cu) = \begin{cases} 0, & Cu \le 0.072 \text{ wt\%} \\ (Cu-0.072)^{0.659}, & Cu > 0.072 \text{ wt\%} \end{cases}$$

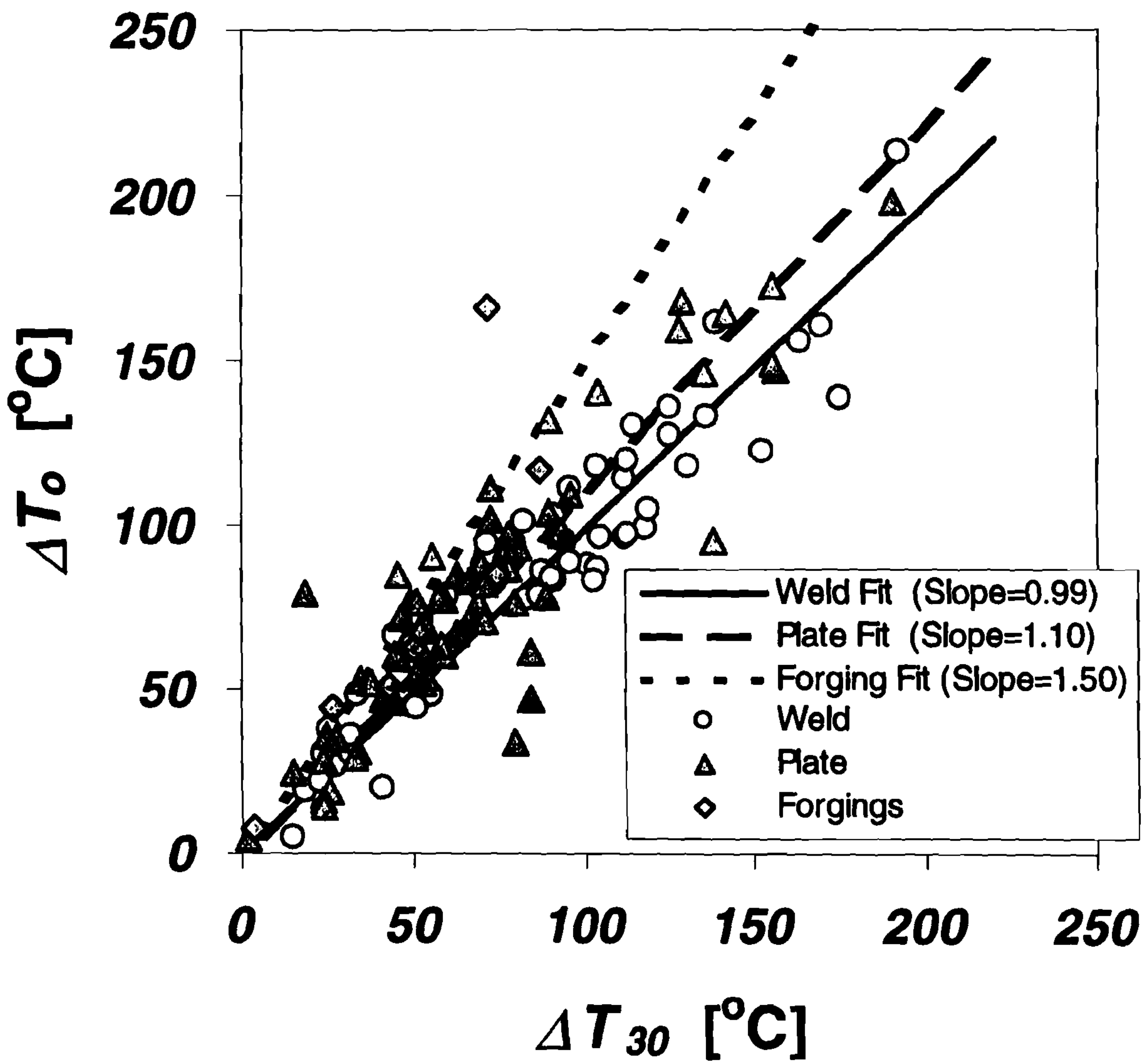
$$g(\phi t) = \frac{1}{2} + \frac{1}{2}\tanh\left[\frac{\log\left(\phi t + 4.579\text{x}10^{12}\, t_i\right) - 18.265}{0.713}\right]$$

$$Bias = \begin{cases} 0, & t_i < 97000 \text{ h} \\ 9.4, & t_i \ge 97000 \text{ h} \end{cases}$$

Does $\Delta T_{30} = \Delta T_o$?

subject to: $Cu_{max} = \begin{cases} 0.25, & \text{for welds with Linde 80 or Linde 0091 flux} \\ 0.305, & \text{for other welds} \end{cases}$

Relationship Between CVN Transition Temperature & Toughness Transition Temperature



VG 76

FAVOR Model of Irradiation Shift

Previous (circa-1980s) PTS analyses assumed that CVN and toughness shift are the same

Start

Appendix A
Mean composition values from RVID
→ Cu
→ Ni
→ P

Appendix B
Generic distributions
→ Cu
→ Ni
→ P

Distributions of Cu, Ni, P scaled to mean values *(Epistemic)*

Operating time & fluence from BNL *(Epistemic)*

Attenuation

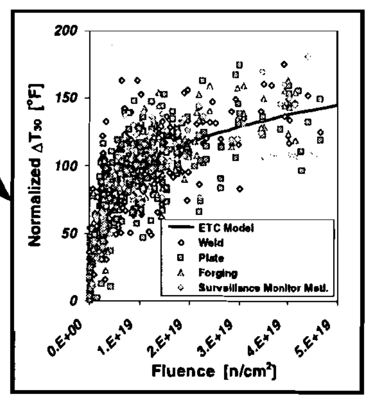
$$\phi t = \phi t_{ID} \exp(-0.24 \cdot x)$$

FAVOR simulated crack tip position (*x*)

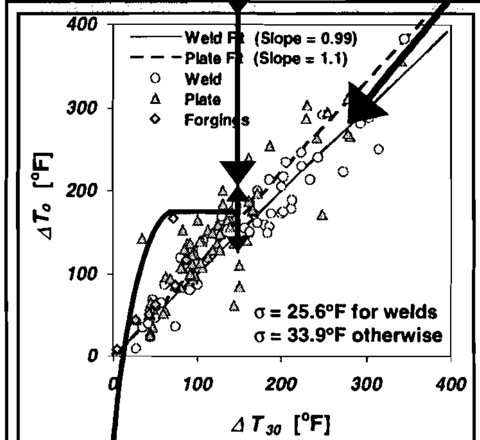
Embrit. Trend Curve

Appendix A
- Coolant temp.
- Product form
- Manufacturer

(Deterministic)



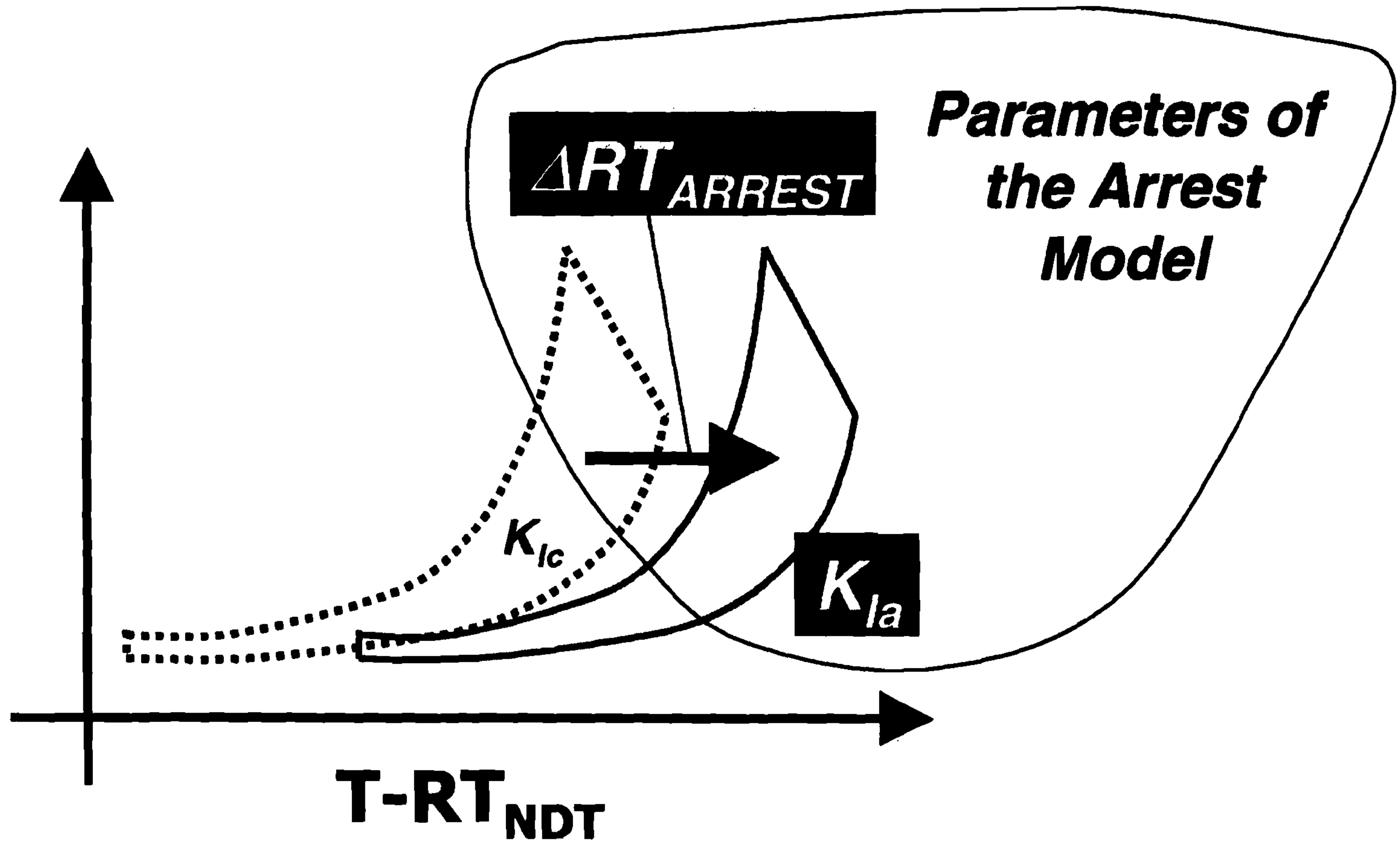
ΔT_{30} at inner crack tip



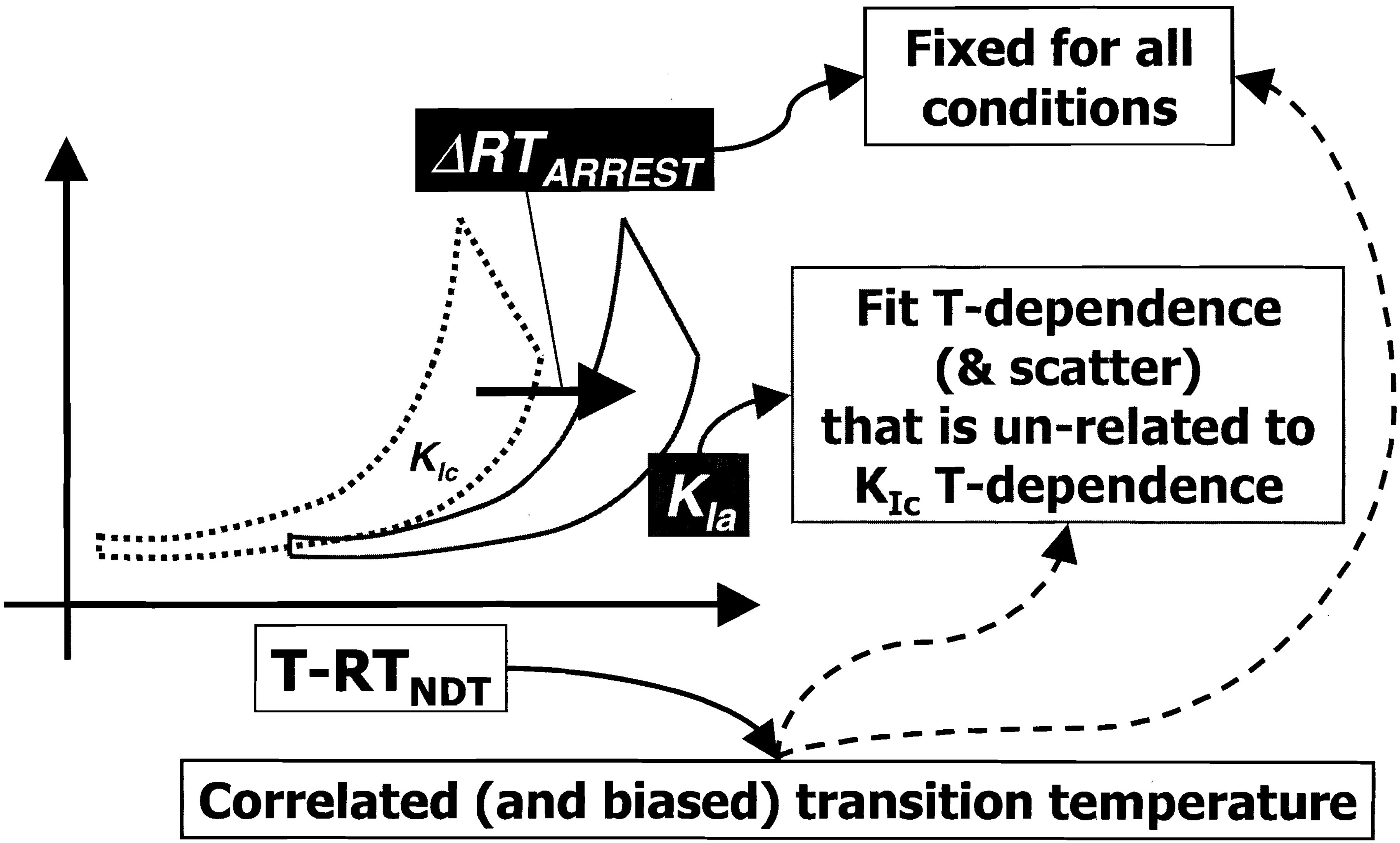
Uncertainty in best-estimate irradiation shift value *(Epistemic)*

ΔT_o

Arrest Fracture Toughness Model



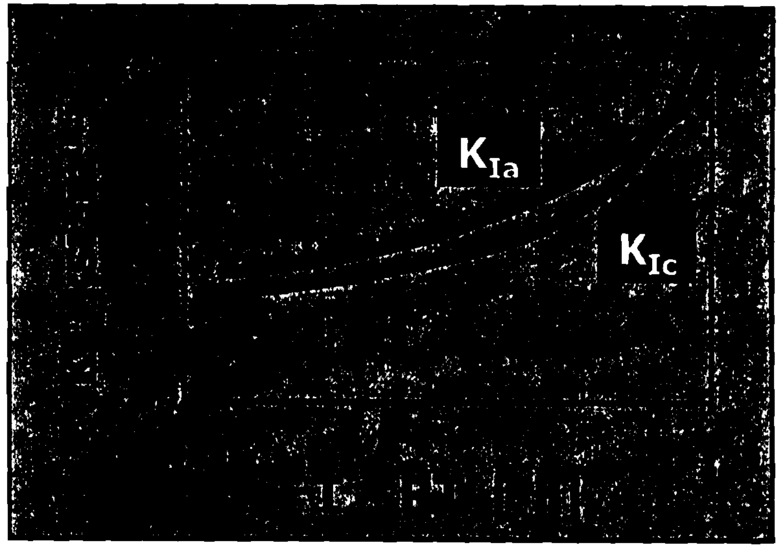
Current Arrest Fracture Toughness Model



Best Estimate Model for Crack Arrest Transition Temperature & Fracture Toughness

Physical Observation	→	Suggests	→	Data

Deterministic



μ-features of different spacings control initiation and arrest

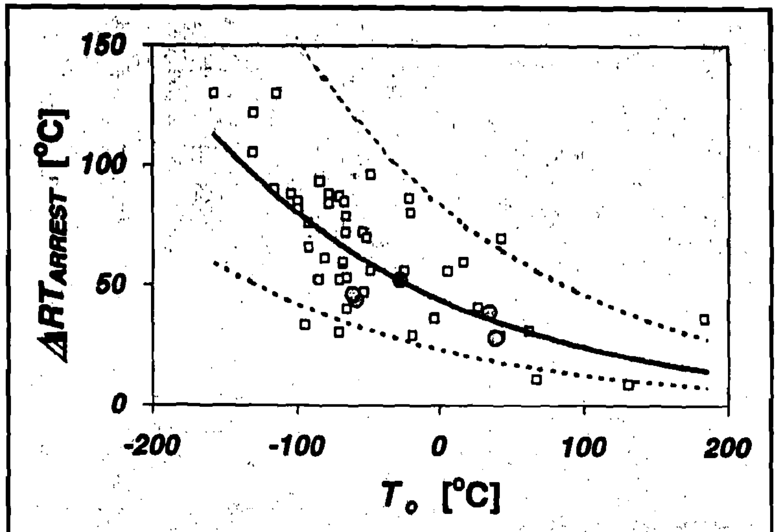
Aleatory

- K_{Ia} scatter < K_{Ic} scatter
- Size affect on K_{Ic} not K_{Ia}

A hardening curve universal to all steels

Epistemic

Initiation curve approaches arrest curve as material damage increases



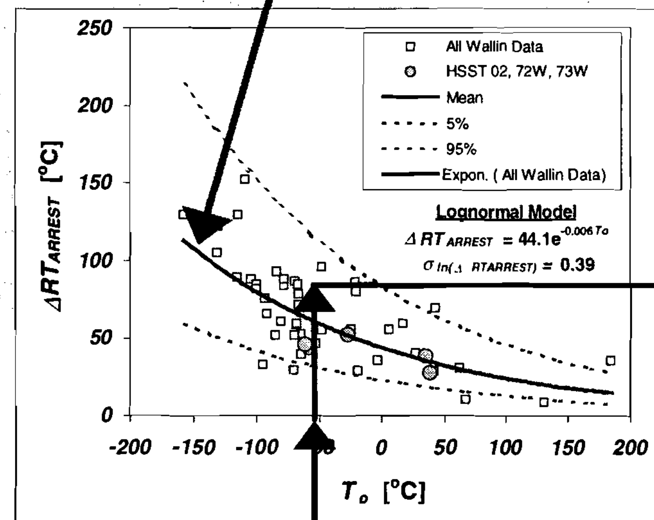
VG 80

FAVOR Model of Arrest Fracture Toughness

Previous (circa-1980s) PTS analyses assumed a non-conservative fixed $K_{Ic} \rightarrow K_{Ia}$ shift

Determine Shift Between Initiation and Arrest Curves

(Epistemic Uncertainty)



ΔRT_{ARREST}

Calculate New Reference Temperature

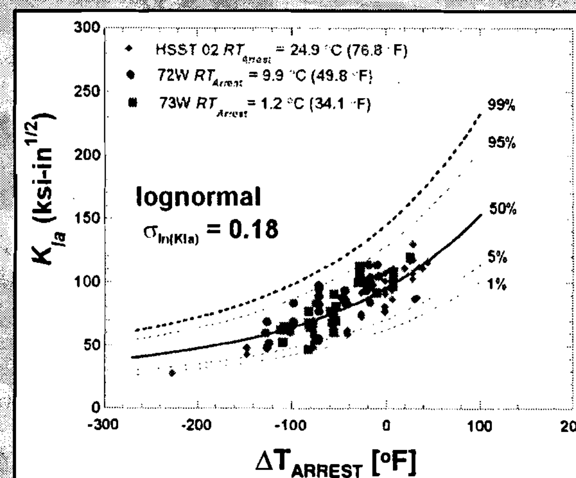
$RT_{ARREST} = RT_{INITIATION} + \Delta RT_{ARREST}$

Calculate Relative Temperature

$\Delta T_{ARREST} = T_{VESSEL} - RT_{ARREST}$

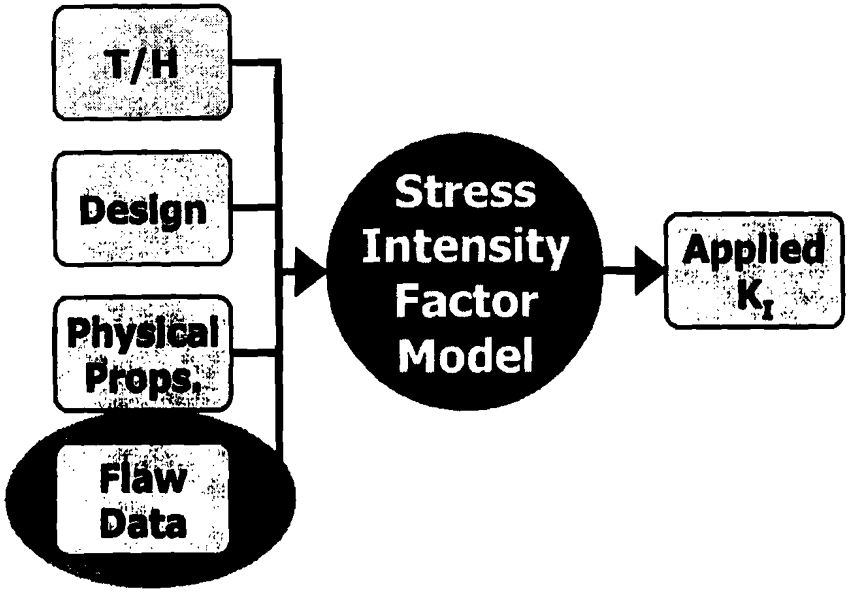
T_{VESSEL} from RELAP

Aleatory Uncertainty in K_{Ia} at ΔT_{ARREST}



$T_o \equiv RT_{INITIATION} - 14.4$°F (from initiation model)

Start



Flaw Model, Overview

- Sources of data
  - Experimental
    - ✓ Destructive
    - ✓ Non-destructive
  - PRODIGAL model
  - Expert elicitation

- Developed distributions of flaws in
  - Fabrication welds
  - Repair welds
  - Cladding welds
  - Plate materials
- Each distribution includes
  - Flaw density
  - Flaw size
  - Flaw orientation
  - Flaw location

Experimental Data Sources			
	Weld	Plate	Clad
PVRUF	☑	☑	☑
Shoreham	☑	☑	
Hope Creek		☑	
River Bend		☑	

Flaw Model

(Assumptions & Process for Model Building)

1. **Basic**
2. **Procedural**
3. **Based on observation and/or physical understanding**

<u>Basic Assumption</u> Except in limited cases, we employ no theoretical / physical model of flaw formation. Therefore

- Data is treated as the highest truth, and
- Expert opinions and models are relied on only when necessary. Consequently
- The inspected material must be assumed to represent adequately all material conditions of interest, so
  - Our most conservative data is assumed to represent the whole. Data is not avera

EPISTEMIC

CONSERVATIVE
(for the available data)

Flaw Model – Procedural Assumptions

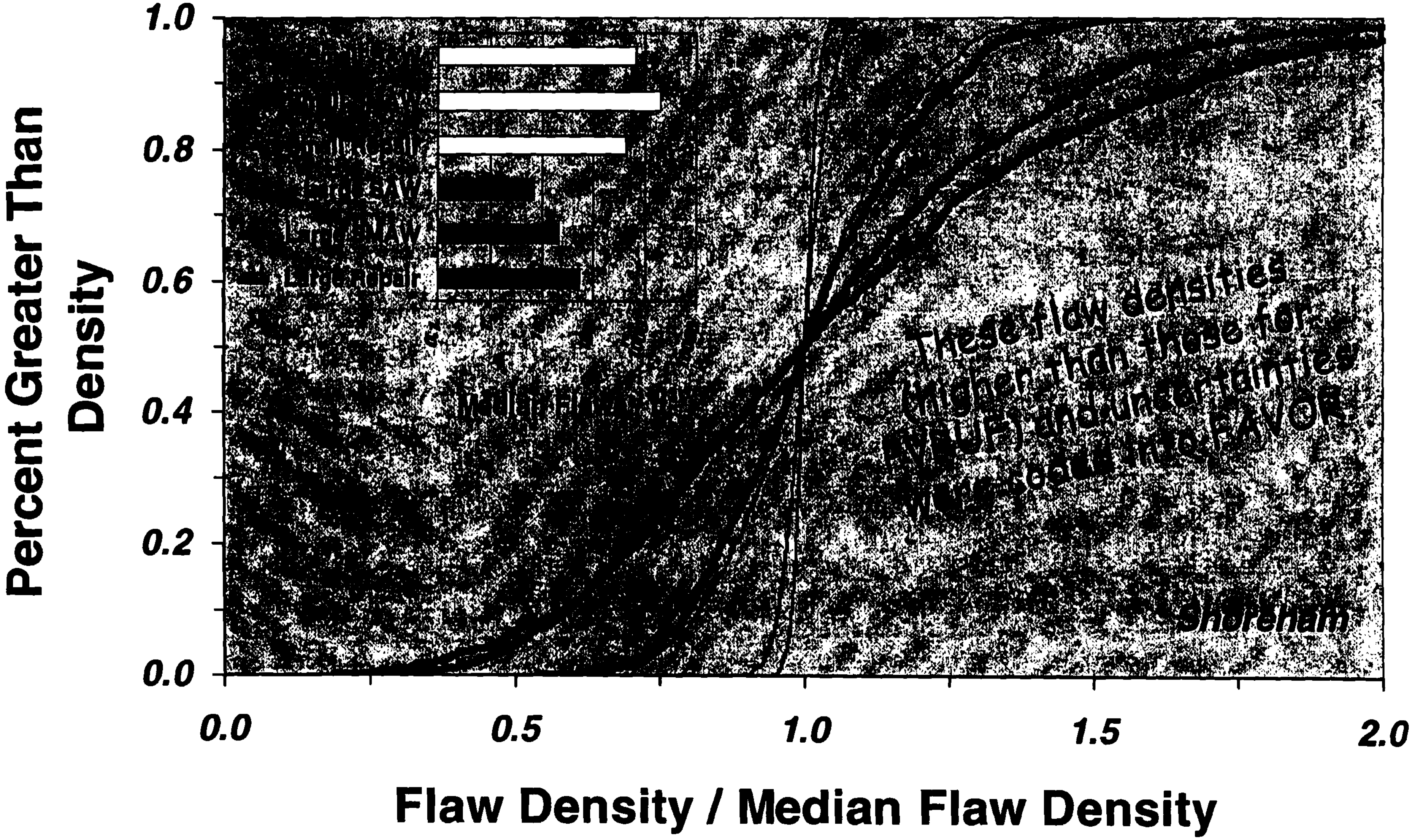
Assumption	Characterization	
Largest flaws main focus of destructive inspection (small flaws destroyed on sampling basis). Small flaw (NDE) data combined with larger flaw (DE) data in the final flaw distribution.	--	
All reported defects modeled as sharp cracks.	Conservative	Further refinement is beyond the scope of FAVOR.
Complex clusters of flaws idealized into single planar elliptical cracks.	Customary, but unknown	
Flaws measured in true size and shape, but in FAVOR assumed to lie only in the axial or circumferential direction.	Customary, but unknown	
All fusion line flaws assigned to the surrounding constituent (plate or weld) that is the most embrittled.	Conservative within the context of a bi-material model	
Weld flaw distribution based on rule of mixtures from different weld process constituents	Appropriate within the context of this model	

Flaw Model

(Assumptions based on observation or physical understanding)

Assumption	Characterization
Truncation limits established based on physical arguments	Conservative (but no effect)
All weld flaws assumed to exist on the fusion line because (a) 95% of all weld flaws were found there, and (b) the mechanisms that generate flaws suggest this is where the majority of flaws will be.	Appropriate (obs & phys)
All cladding flaws assumed to exist parallel to the welding direction (circumferential).	Appropriate (physical)
Distribution of clad flaws based on PRODIGAL model	Appropriate (observation)
Plate flaw densities: 1/10 of weld density for small flaws, 1/40 weld density for large flaws (based on expert elicitation)	Appropriate (observation)
50% of plate flaws assumed to be oriented axially, 50% circumferentially.	Appropriate within the context of this model (physical)

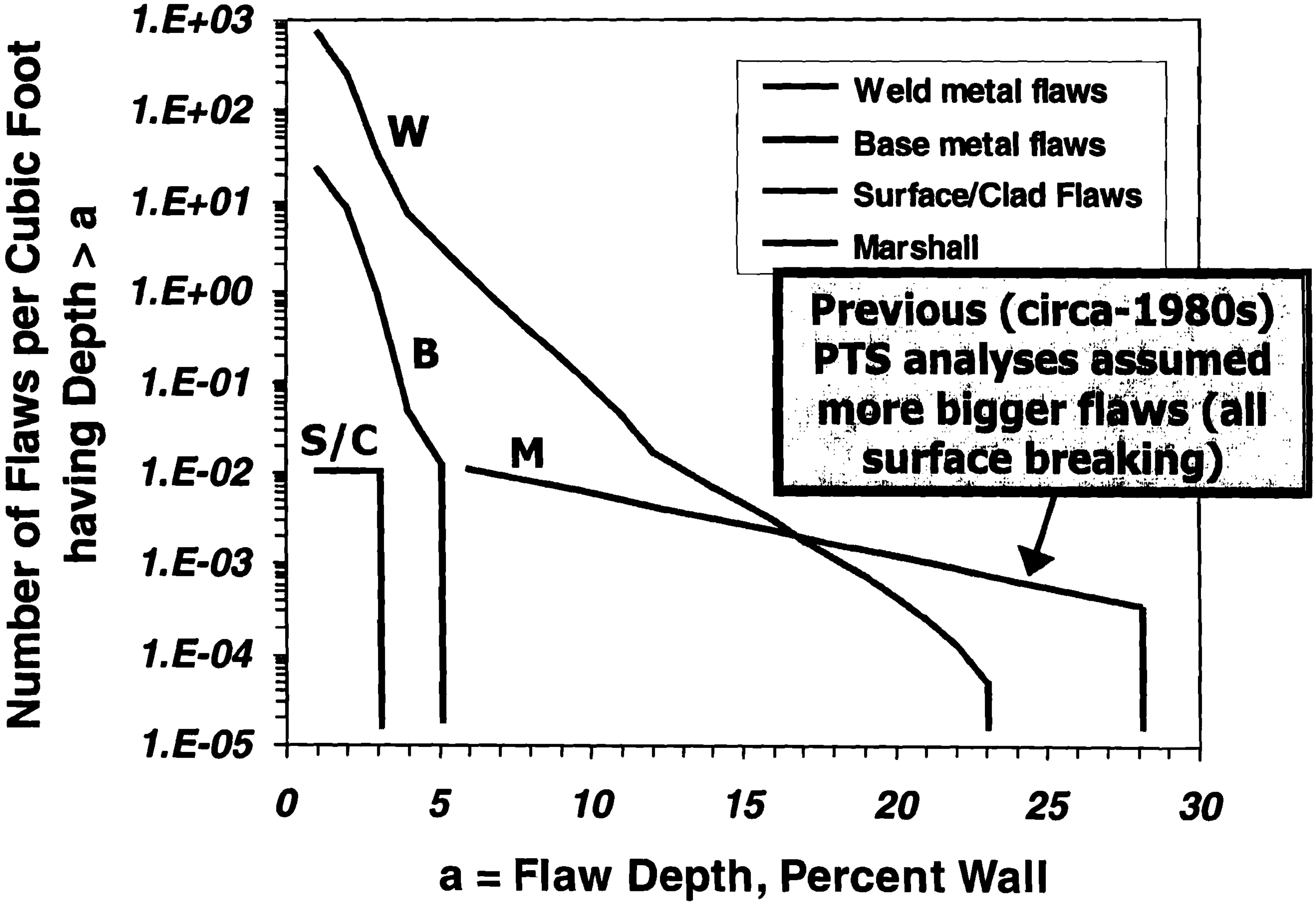
Shoreham Flaw Density



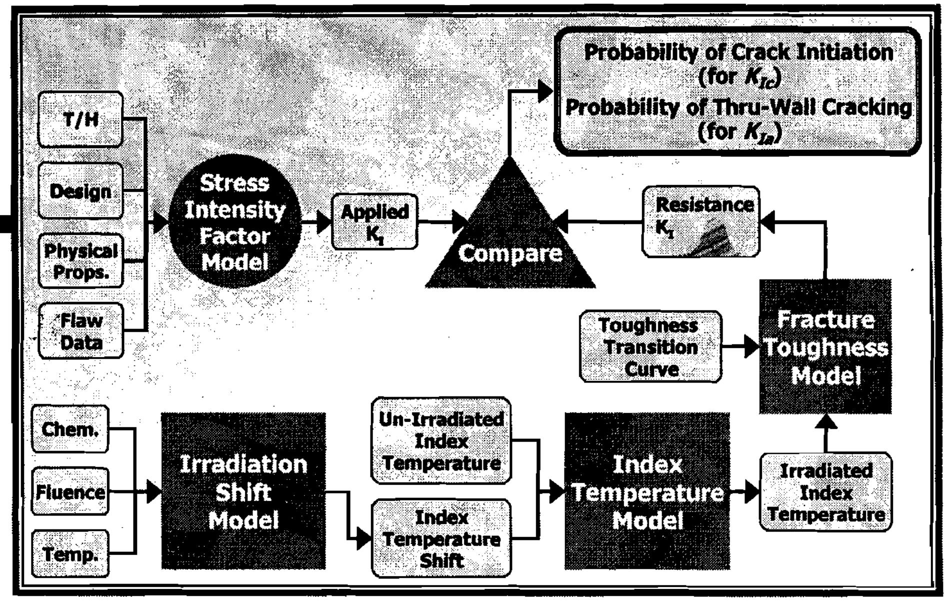
Flaw Model in FAVOR

- **Distribution used**
  - **Based on either rule of mixtures or bounding cases, as noted previously**
- **Treatment of uncertainty**
  - **Statistical uncertainty in data is the only uncertainty explicitly accounted for in the model**
  - **Uncertainty quantified by generating 1,000 different input files, randomly drawn from the distributions of possible flaw sizes and densities**
  - **Uncertainty modeled as *epistemic***

Flaw Model – Significant Changes



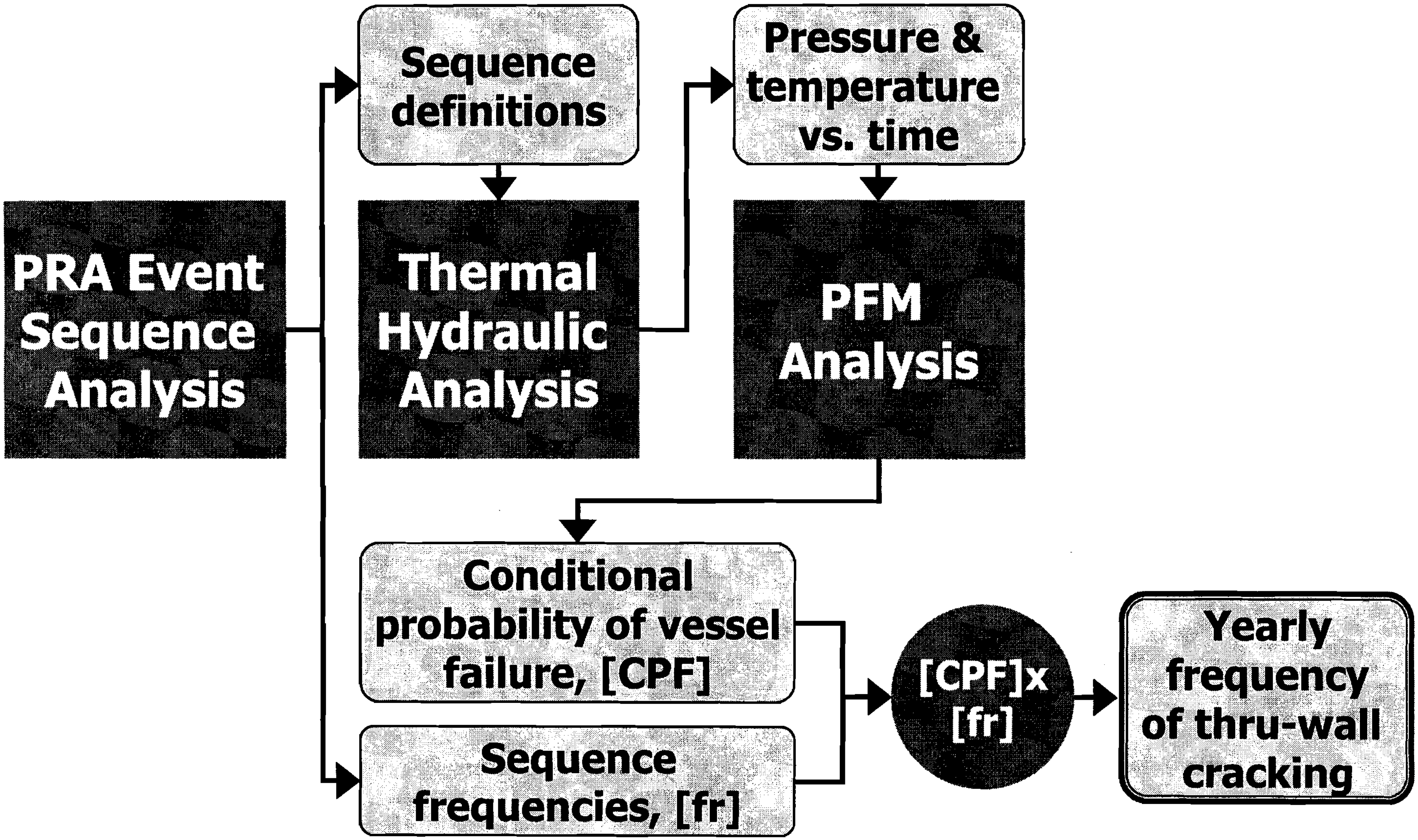
Probabilistic Fracture Mechanics Summary



- **Toughness**
  - **Referenced to toughness data & physical understanding**
    - ✓ **Significant conservative bias in un-irradiated index temperature removed**
    - ✓ **Non-conservatism in arrest model removed**
    - ✓ **Aleatory nature of toughness uncertainty quantified**
- **Embrittlement**
  - **Referenced to toughness data & physical understanding**
    - ✓ **Correlation with better empirical/physical basis**
    - ✓ **Slight biases in in CVN-based shift estimates removed**
- **Fluence**
  - **Spatial variation in fluence recognized, significant conservatism associated with max fluence assumption removed**
- **Flaws**
  - **Based on significantly more data than before**
  - **Most flaws now embedded rather than surface flaws**
  - **More flaws than before**

Oconee 1

Analysis Procedure



Oconee 1 Results

Vessel Specific Inputs

- T/H events (w/ event frequencies)
- Fluence map
- Material map
- Material embrittlement

Generic Inputs

- Flaw distribution
- Toughness distribution

Addressed Earlier

Outputs & Interpretations

- Description of dominant transients
- Insights from analysis
- Expression of results relative to existing screening criteria

PRA & T/H Inputs

- ≈ 150 total transients analyzed
  - ≈ 50 screened (eliminated by inspection)
  - ≈ 50 base case
  - ≈ 50 T/H sensitivity cases
- Initiating event frequencies
  - Range from 8E-9 to 3E-4
  - Reflect most recently available data and operator training procedures
  - Some IEFs considerably lower than in circa-1980s studies (e.g., MSLB dropped from an E-4 to an E-6 event)

PRA Improvements
(relative to Circa-1980s Analysis)

Residual Group

- The "everything else group" in the 1980s
  - Collection of all "small" frequency sequences (<E-6/yr)
  - Worst case CPF applied (5.4E-3)
  - Accounted for > ½ of all PTS risk
  - No human actions credited
- In the current study:
  - Latest frequencies/probabilities
  - More refined sequence grouping (no 1 catch-all group)
  - Human actions credited realistically
  - CPIs/CPFs assessed for each sequence group

Steam Generator Tube Ruptures

- Circa-1980s study
  - Likely SGTR sequences had low CPFs,
  - BUT binned less likely SGTRs in the "Residual Group", artificially elevating their significance
- Now
  - Small breaks → slow cooling rate (even w/ RCP shutdown)

Main Steam Line Break

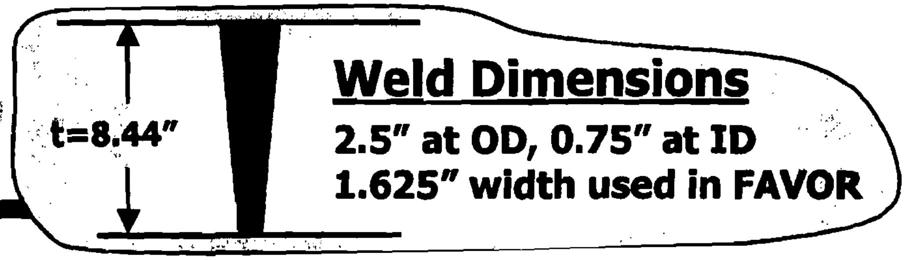
- In 1980s accounted for nearly all remaining PTS risk
- In the current study
  - Human credit for rapid isolation of feed, and for throttling HPI
    - ✓ Improved training/procedures
    - ✓ Almost no human credit in original Oconee study
    - ✓ This study-typical values
      - ➢ Isolate: E-2 by 10min
      - ➢ Throttle: E-1 by 10min

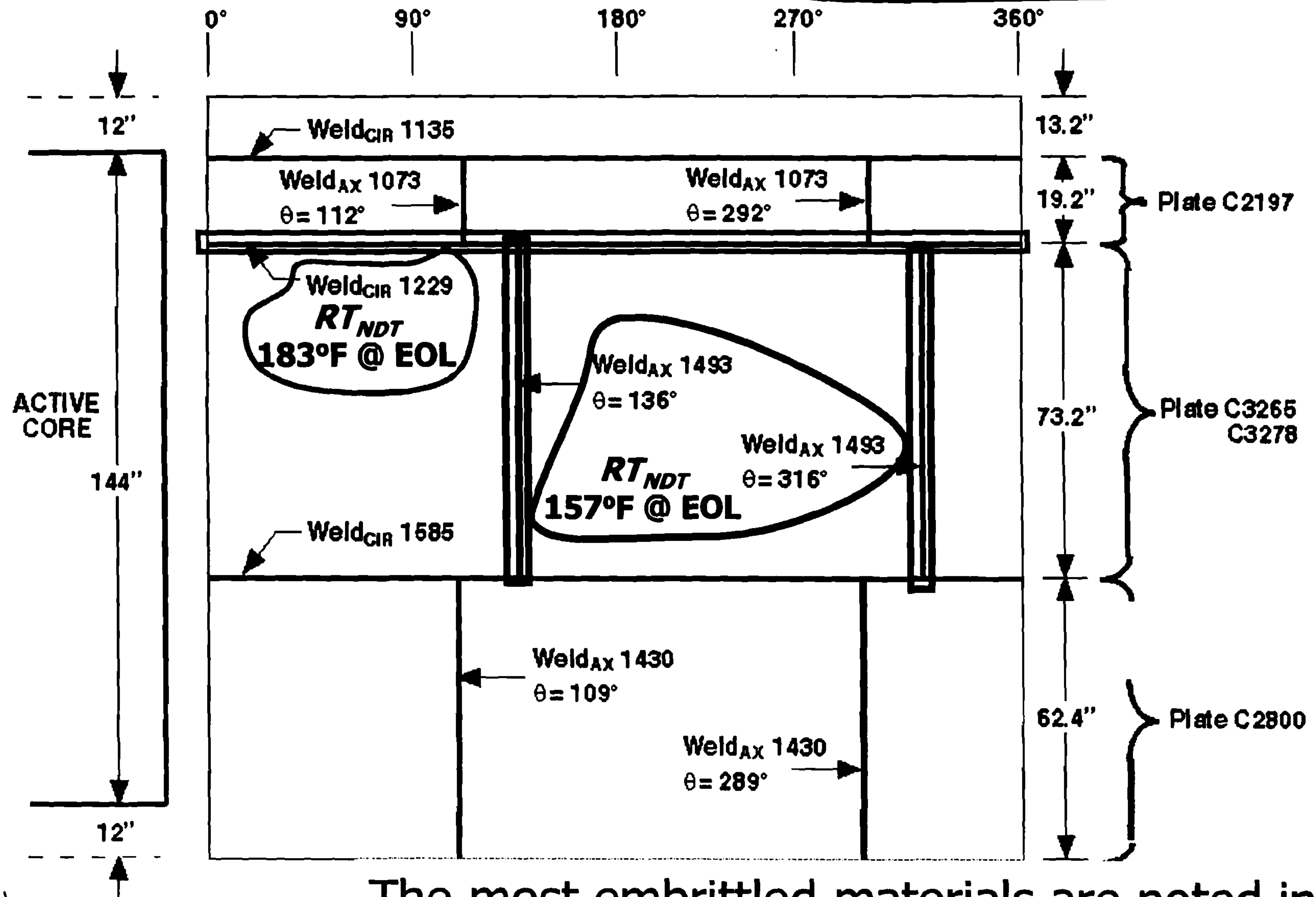
    Thus sequence frequencies low for severe cooldown
  - Successful actions mitigate potential for damage
    - ✓ Isolating feed limits the cooldown rate
    - ✓ Throttling HPI limits pressure
    - ✓ Thus, more likely scenarios have low CPI/CPF

PRA Improvements
(relative to Circa-1980s Analysis)

- **Why credit for operator actions during Main Steam Line Break?**
  - **Overcooling prevention & control are an integral part of Oconee crew training**
  - **Oconee operators are 'sensitive' to overcooling**
  - **Instrumentation is available & procedures are written to facilitate identification of an excessive steam demand**
  - **Procedural hierarchy promotes rapid response to such an event (isolate faulted steam generator)**
  - **Warnings to throttle HPI appear in numerous points throughout the procedures and it is a continuous action step**
  - **Simulator observations 'confirm' successful response is likely**

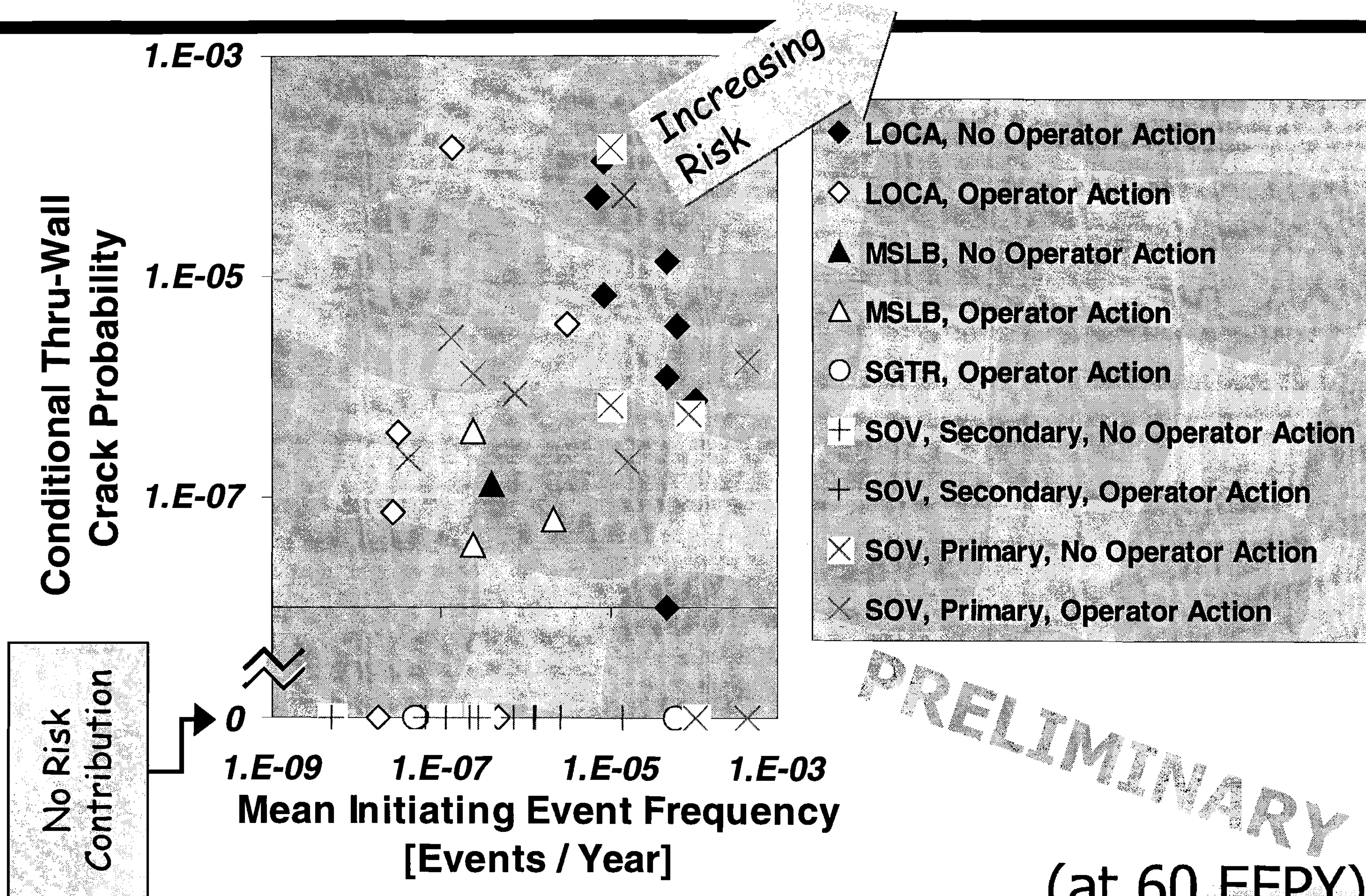
Oconee 1 Material Map





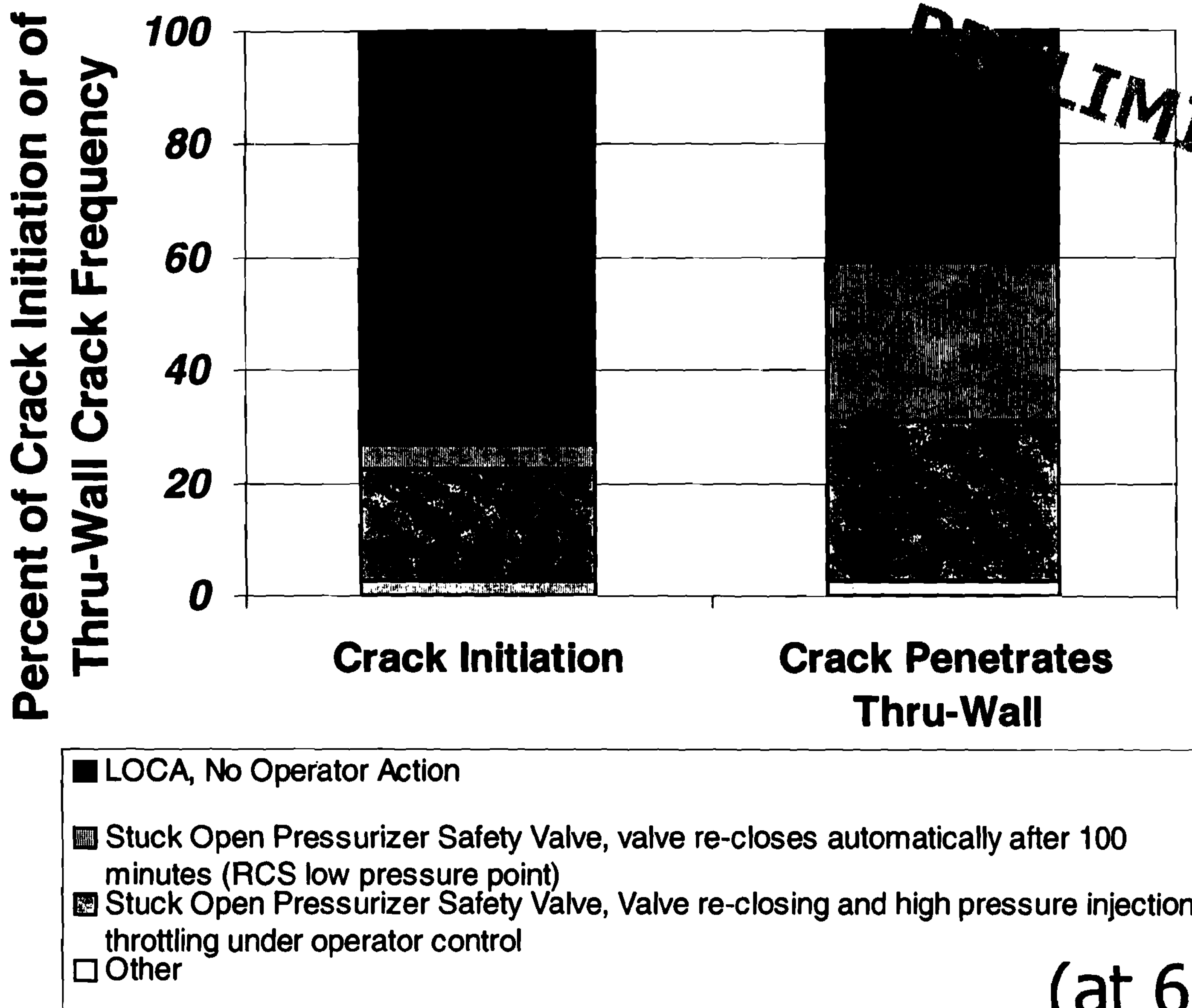
The most embrittled materials are noted in color.

Base Case Analysis Results for Oconee1



(at 60 EFPY)

The Contribution of Different Classes of T/H Events to Vessel Failure Frequency

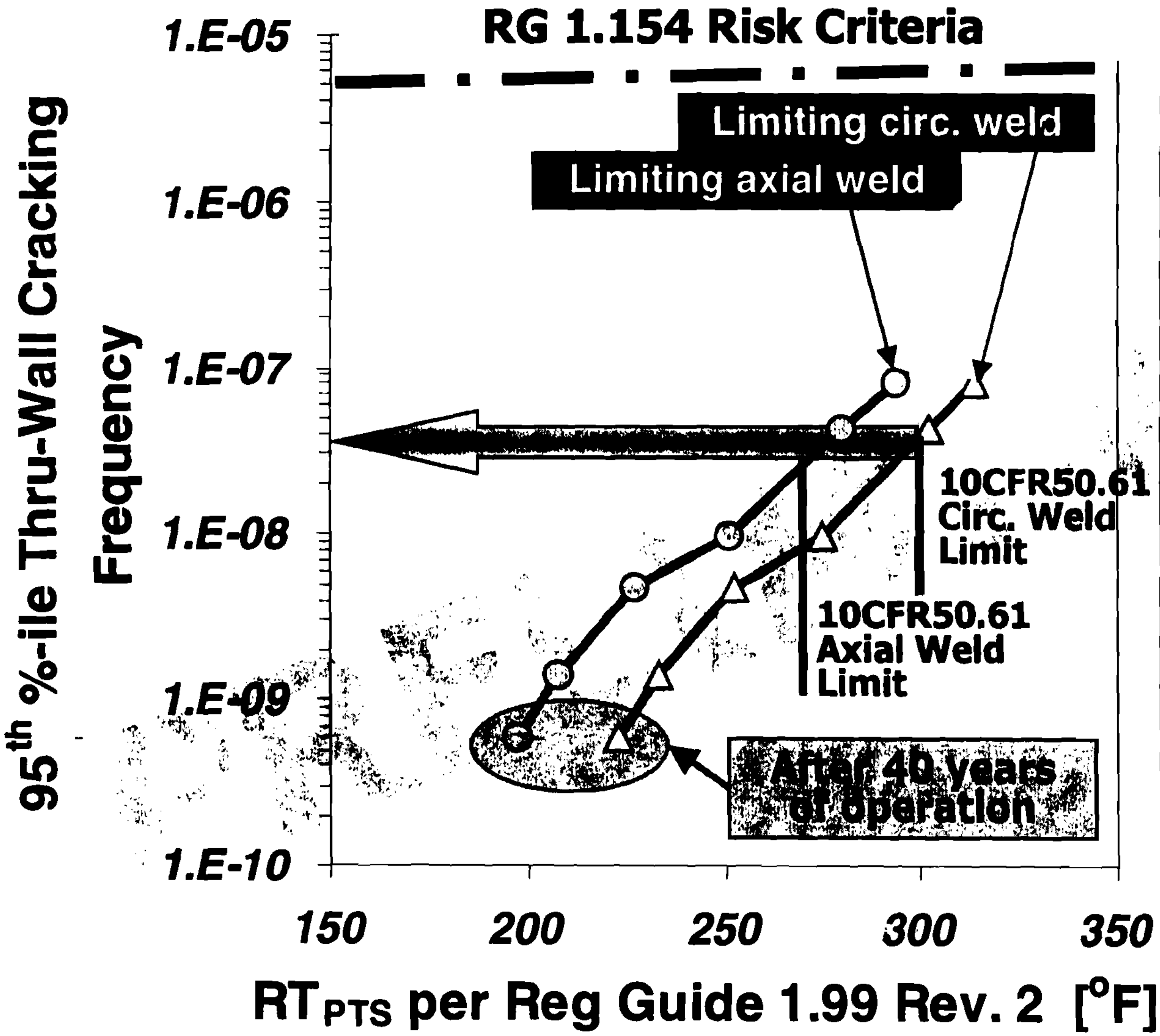


(at 60 EFPY)

Observations

- Dominant scenarios are all primary system LOCAs.
- Realistic accounting of operator action significantly mitigates the influence of secondary system events on total failure probability.
- Time of SRV closure (and thus re-pressurization) has significant influence on event severity. Consequently, operator action involving throttling of injection following closure (especially when closure is later in time) has a significant influence on these results.

Preview of Results for Oconee 1



Relative to the RG 1.154 risk criteria, current results are

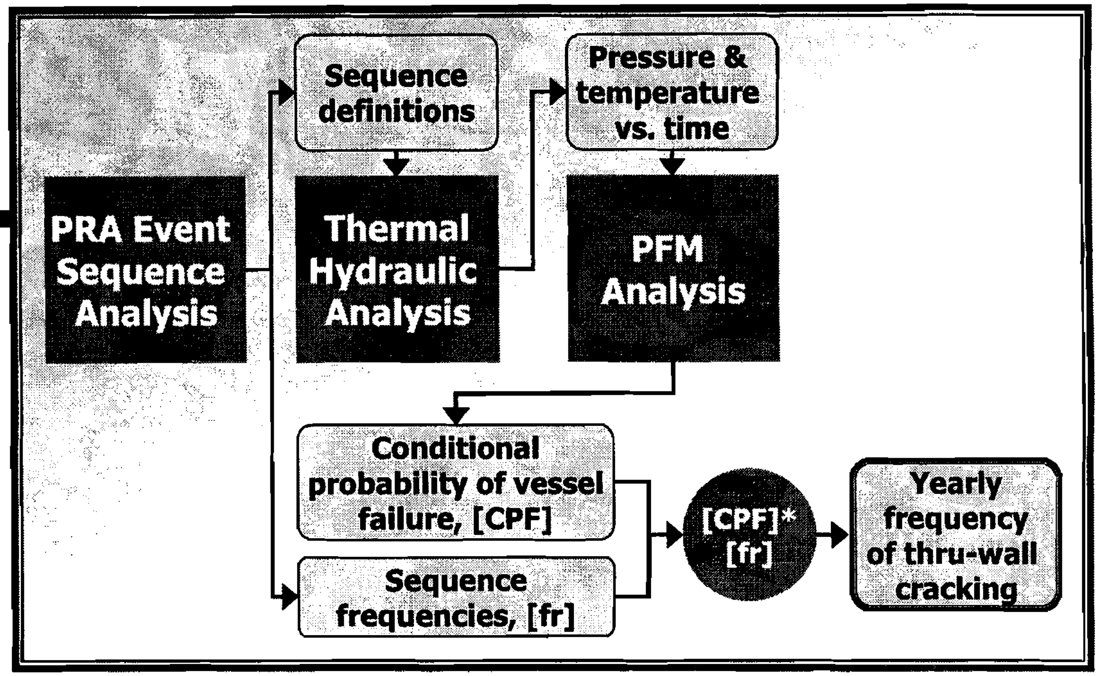
- ✓ ≈ 4 orders of magnitude lower after 40 years of operation
- ✓ ≈ 2 orders of magnitude lower at the current screening limits

Oconee Results Summary

- **Preliminary results for Oconee 1 look promising relative to the current risk criteria**
  - **Leads to perception that the risk of vessel failure is lower than we previously believed it to be**
  - **New risk goal remains to be established**
  - **Contribution of external events to overall risk**
- **Analyses of Palisades, Beaver Valley, and Calvert Cliffs are continuing**

Example

Structure of Example Discussion



- **We will follow the dominant transient in the Oconee analysis through this process, discussing for it the**
  - **PRA event characterization, and binning**
  - **TH estimation of pressure and temperature *vs.* time**
  - **PFM estimation of thru-wall cracking frequency**

  and emphasizing at each step
  - **Treatment of variables and models as uncertain, or not**
  - **How uncertainty is quantified**
  - **Engineering judgments made, and their basis**
- **This is an illustrative example only, *not* a comprehensive treatment**

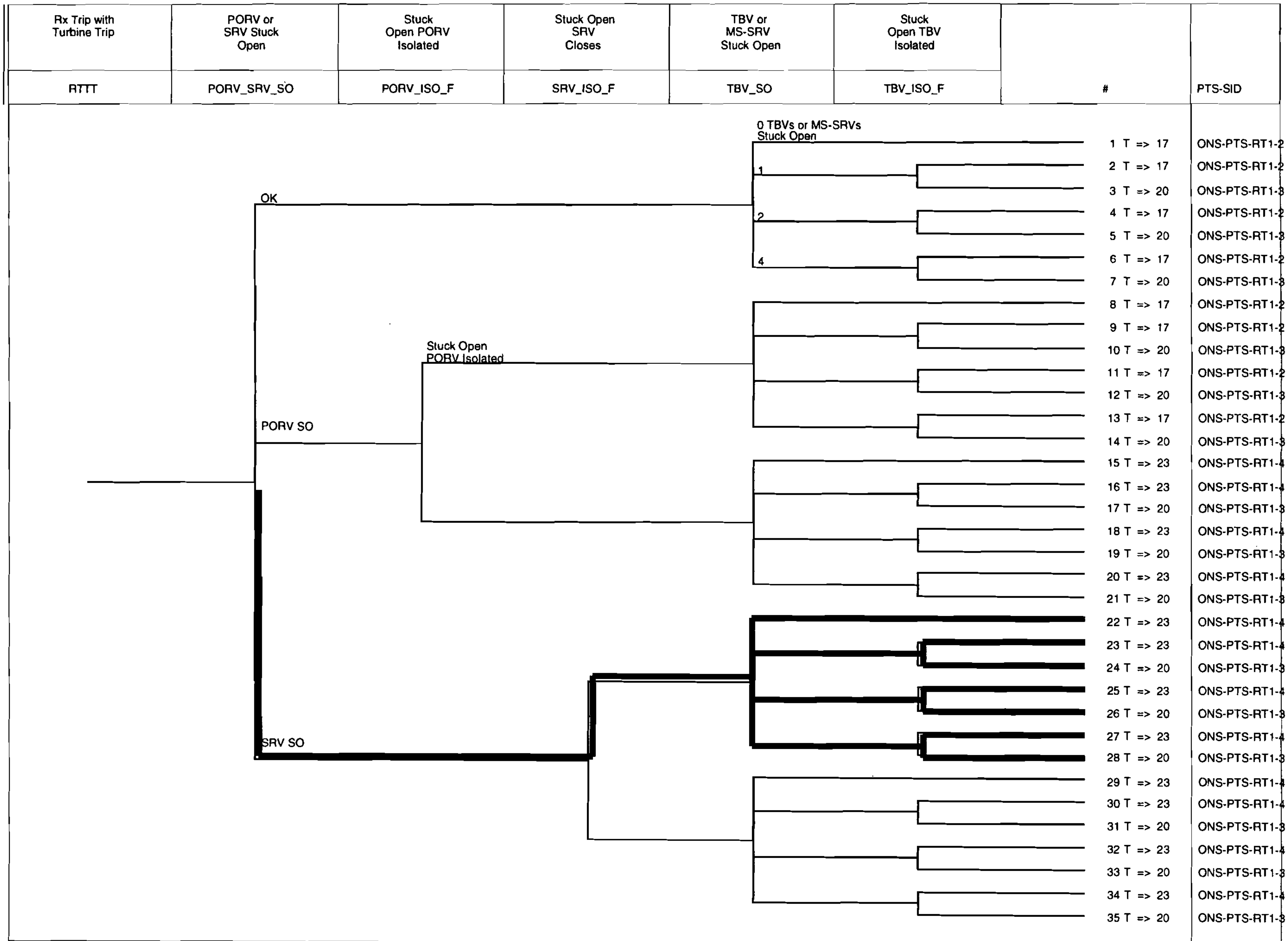
Example
(PRA)

EXAMPLE PROBLEM:
Scenario Bins 109, 112, 113

- **What are these scenario bins?**
  - **109: Stuck–open pressurizer SRV**
    - **SRV recloses at 100 min.**
    - **Operator fails to control repressurization**
  - **112: As above**
    - **Operator throttles HPI ~1 min after throttling criteria met**
  - **113: As above**
    - **Operator throttles HPI ~10 min after throttling criteria met**

At the beginning ... there was only bin 41

- **Bin 41:**
  - **Stuck–open pressurizer SRV**
  - **SRV recloses at 100 min.**
  - **No operator actions modeled**
- **Nearly all the initiators/event trees have many sequences that were originally placed into Bin 41 ... we'll show just one example**


Rx Trip with Turbine Trip
PORV or SRV Stuck Open
Stuck Open PORV Isolated
Stuck Open SRV Closes
TBV or MS-SRV Stuck Open
Stuck Open TBV Isolated
RTTT
PORV_SRV_SO
PORV_ISO_F
SRV_ISO_F
TBV_SO
TBV_ISO_F
#
PTS-SID
0 TBVs or MS-SRVs Stuck Open
1
2
4
OK
Stuck Open PORV Isolated
PORV SO
SRV SO
1 T => 17 ONS-PTS-RT1-2
2 T => 17 ONS-PTS-RT1-2
3 T => 20 ONS-PTS-RT1-3
4 T => 17 ONS-PTS-RT1-2
5 T => 20 ONS-PTS-RT1-3
6 T => 17 ONS-PTS-RT1-2
7 T => 20 ONS-PTS-RT1-3
8 T => 17 ONS-PTS-RT1-2
9 T => 17 ONS-PTS-RT1-2
10 T => 20 ONS-PTS-RT1-3
11 T => 17 ONS-PTS-RT1-2
12 T => 20 ONS-PTS-RT1-3
13 T => 17 ONS-PTS-RT1-2
14 T => 20 ONS-PTS-RT1-3
15 T => 23 ONS-PTS-RT1-4
16 T => 23 ONS-PTS-RT1-4
17 T => 20 ONS-PTS-RT1-3
18 T => 23 ONS-PTS-RT1-4
19 T => 20 ONS-PTS-RT1-3
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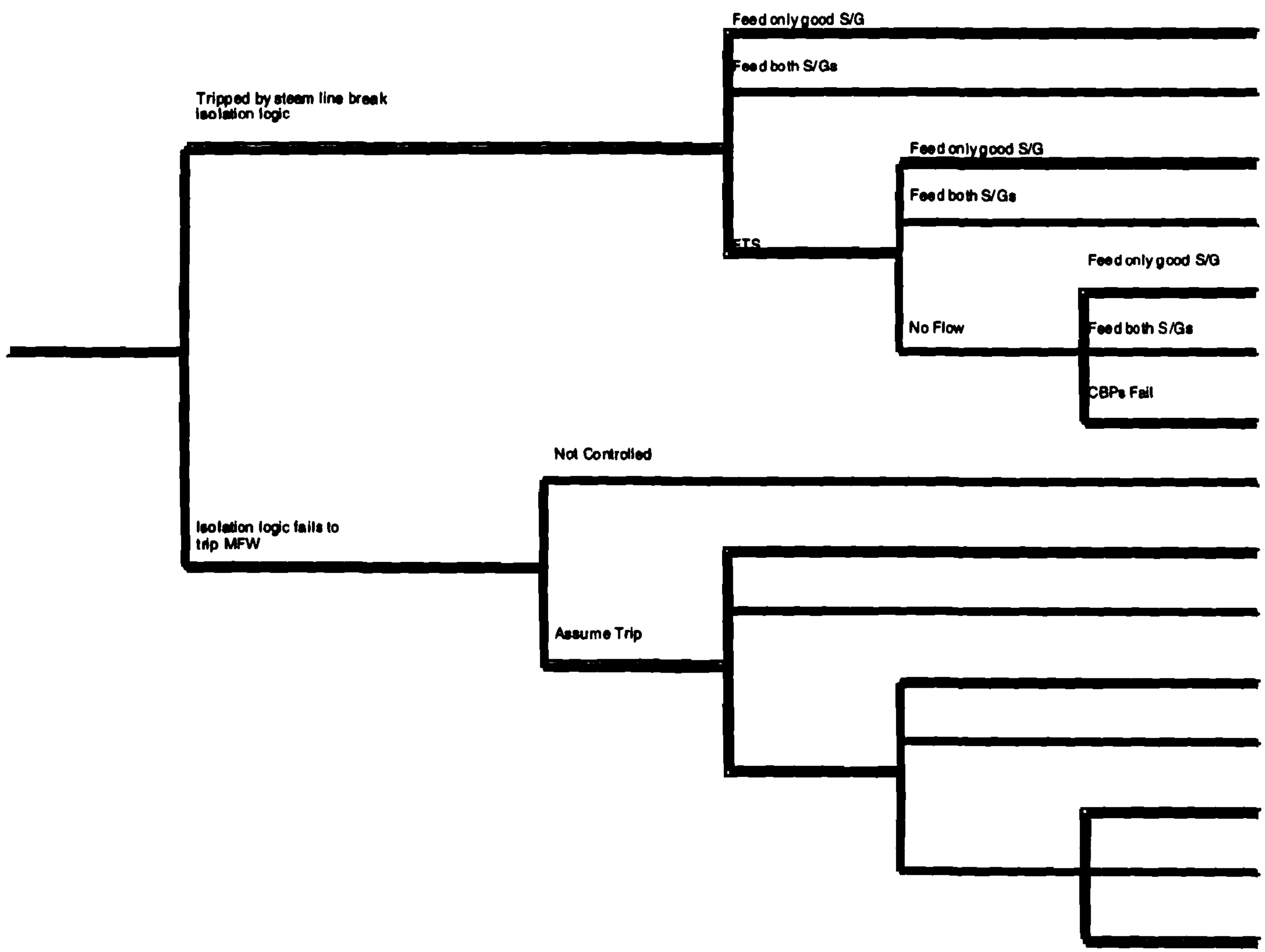
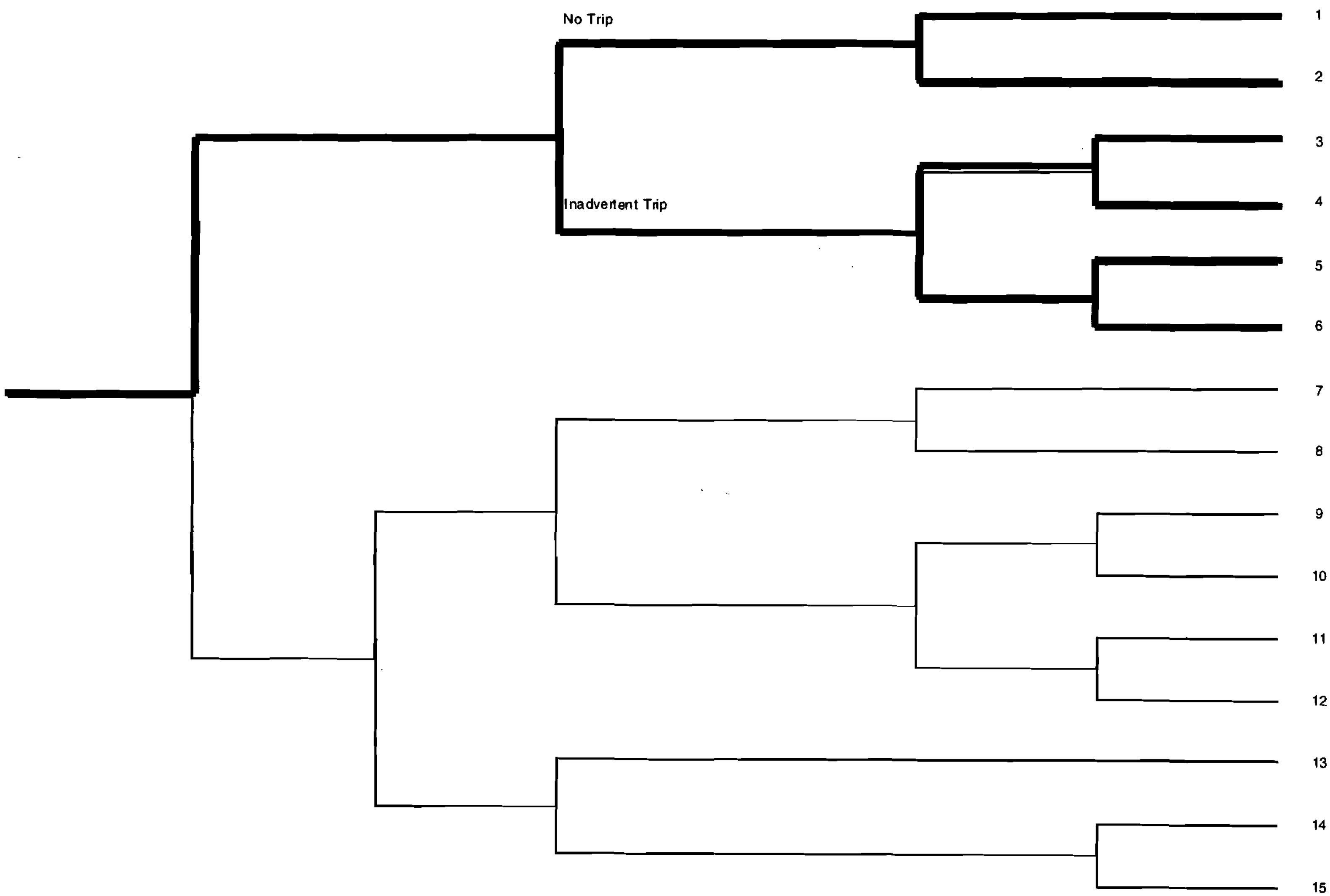
27 T => 23 ONS-PTS-RT1-4
28 T => 20 ONS-PTS-RT1-3
29 T => 23 ONS-PTS-RT1-4
30 T => 23 ONS-PTS-RT1-4
31 T => 20 ONS-PTS-RT1-3
32 T => 23 ONS-PTS-RT1-4
33 T => 20 ONS-PTS-RT1-3
34 T => 23 ONS-PTS-RT1-4
35 T => 20 ONS-PTS-RT1-3

Dummy Not used in Logic or Quantification	MFW Response to IE	MFW Fails to trip on S/G Hi-L	Fail to Recover From MFW Overfeed	EFW Response to MFW trip	Fail to Recover from EFW-FTS	Condensate Booster Pumps Fail	#	PTS-SID
DUMMY	MFW_F	MFW_TRIP_F	MFW_REC_F	EFW_F	EFW_REC_F	CBP_F		



#	PTS-SID
1 T => 21	ONS-PTS-RT1-3-T1
2 T => 21	ONS-PTS-RT1-3-T1
3 T => 21	ONS-PTS-RT1-3-T1
4 T => 21	ONS-PTS-RT1-3-T1
5 T => 21	ONS-PTS-RT1-3-T1
6 T => 21	ONS-PTS-RT1-3-T1
7 T => 22	ONS-PTS-RT1-3-T2
8 T => 21	ONS-PTS-RT1-3-T1
9 T => 21	ONS-PTS-RT1-3-T1
10 T => 21	ONS-PTS-RT1-3-T1
11 T => 21	ONS-PTS-RT1-3-T1
12 T => 21	ONS-PTS-RT1-3-T1
13 T => 21	ONS-PTS-RT1-3-T1
14 T => 21	ONS-PTS-RT1-3-T1
15 T => 22	ONS-PTS-RT1-3-T2

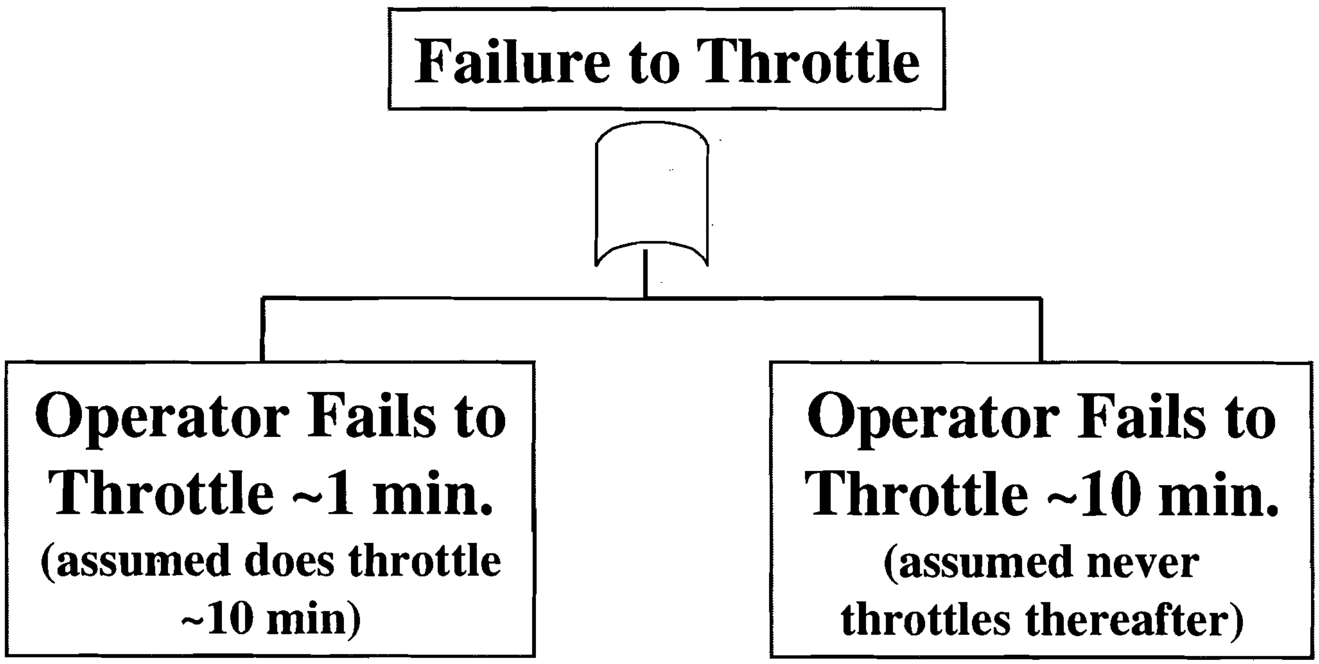
Dummy Not used in Logic or Quantification	HPI/F&B Response	HPI Recovered	RCP Trip (i.e., loss of RCS subcooling)	Secondary Cooling Recovered	Fail to Throttle HPI Flow	Fail to Restart RCPs	#	PTS-SID
DUMMY	HPI_STARTS_F	HPI_REC_F	RCP_TRIP	SEC_COOL_REC_F	HPI_THROTTLED_F	RCP_RESTART_F		



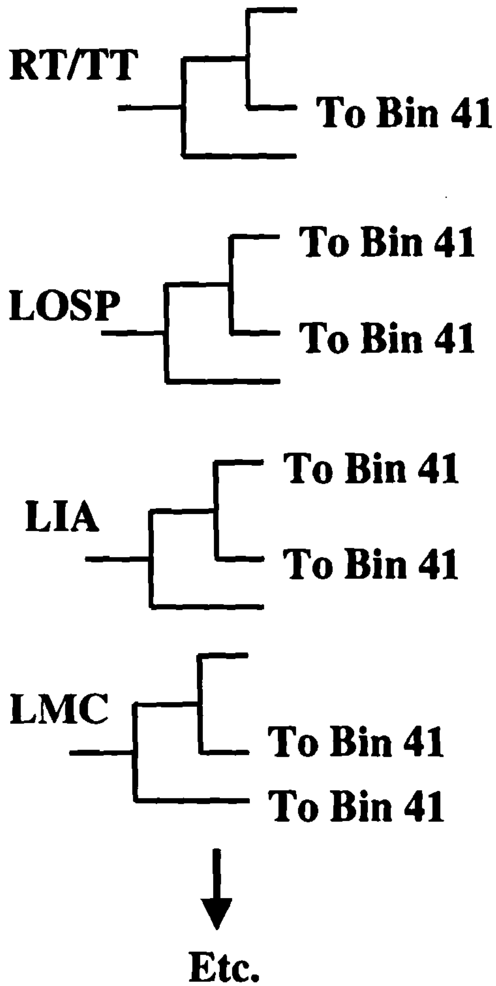
Note the following...

- Many sequences were originally binned into bin 41 including success as well as failure to throttle type sequences
- Various concurrent faults on the secondary side do not matter much to binning
  - Break (SRV open) is large enough that downcomer temperature response is largely driven by the primary (i.e., primary essentially "decouples" from the secondary)
  - Concurrent secondary faults make overall frequency quite low so don't matter much anyway
  - Conclusion reached by comparing a variety of T-H runs

'Failure to Throttle' event is handled with a fault tree of the form...



Solving the model for all the sequences binned to Bin 41...



We get sequences of the form...

I.E. * SRV S.O. * SRV Recloses * ....
***Operator Successfully Throttles (~<1 min)**

Create new Bin 83

I.E. * SRV S.O. * SRV Recloses * ....
***Operator Fails to Throttle ~1 min (does ~10min)**

Create new Bin 84

I.E. * SRV S.O. * SRV Recloses * ....
***Operator Fails to Throttle ~10 min (or thereafter)**

Stays in Bin 41

... & redistribute them as shown

Summary of Bin Frequencies

- **Bin 83 (successfully throttles ~1min)**
  1E-3/yr

- **Bin 84 (fails to throttle by 1 min; does ~10min)**
  4E-5/yr

- **Bin 41 (fails to throttle by 10min or thereafter)**
  3E-5/yr

Numerous Uncertainty Studies Performed for a SRV S.O.-Recloses

- **HPI flowrate**
- **Winter-Summer injection water temperatures**
- **Wall heat transfer rate**
- **Cold leg flow resistance...**

And found the following to dominate the uncertainty of the T-H response (besides when operator throttles):

- **Timing of the SRV reclosure**
- **Degree SRV is open**
- **Full vs. Hot Zero Power**
- **High Cold Leg reverse flow resistance**

Captured the uncertainty in these parameters by…

- Assigned a 50 –50 probability SRV recloses at either 50 min or 100 min
- Assumed a uniform distribution for the open area of the stuck-open SRV
  - Only 1.5" diameter to full open (1.8") provides considerable cooling
  - Hence probability SRV is stuck-open with an area that results in considerable cooling
  - = area of interest/total possible area = 0.3
- Multiplied bin frequencies by 0.5 x 0.3
- Accounted for full vs. hot zero power (HZP) by similar treatment of other bins for hot zero power conditions (bins 92,93,42). Added full & HZP (small) contribution.
- Probability of high cold leg reverse flow resistance = 1.0

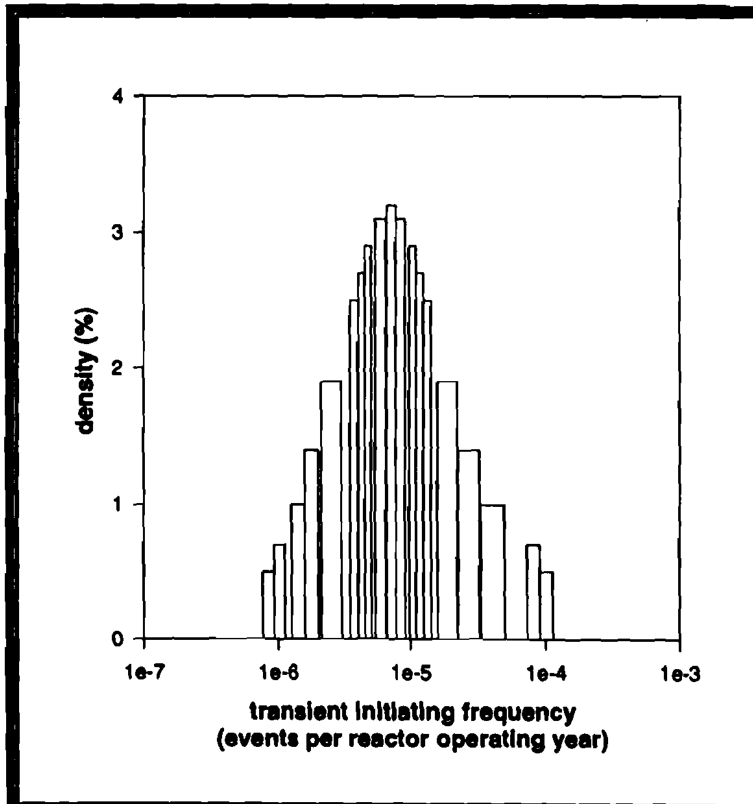
This resulted in the final bin frequencies

- **109: Stuck–open pressurizer SRV**
  - **SRV recloses at 100 min**
  - **Operator fails to control repressurization**

    1E-5/yr
- **112: As above**
  - **Operator throttles HPI ~1 min after throttling criteria met**

    4E-4/yr
- **113: As above**
  - **Operator throttles HPI ~10 min after throttling criteria met**

    1E-5/yr



Example (TH)

TH Results

- **300 transients run:**
  - **46 base cases included in the RPV failure frequency analysis**
  - **50 sensitivity cases included in the RPV failure frequency analysis**
  - **≈ 200 miscellaneous cases run to evaluate various aspects of plant response**

Transient 109 (& its Variants)

Case Number	Primary Side Failure	Operator Action
109	Stuck open pressurizer safety valve. Valve recloses at 6000 secs (RCS low pressure point).	None
112	Stuck open pressurizer safety valve. Valve recloses at 6000 secs.	After valve recloses, operator throttles HPI 1 minute after 5°F subcooling or 100" pressurizer level is reached (throttling criteria is 5°F subcooling and 100" pressurizer level)
113	Stuck open pressurizer safety valve. Valve recloses at 6000 secs.	After valve recloses, operator throttles HPI 10 minutes after 5°F subcooling or 100" pressurizer level is reached (throttling criteria is 5°F subcooling and 100" pressurizer level)

Event Category Selected for TH Uncertainty Analysis

Secondary Side State \ Primary Side State	Intact	Breached: Break Size <~ 1.5" Breach flow could be compensated by HPI	Breached: Break Size > ~1.5" Breach flow cannot be compensated by HPI
Nominal			6.1e-4
			8.9e-4
One SG Breach		2.7e-7	
	3.7e-6	6.3e-5	
Two SGs Breach		2.8e-7	
	1.7e-5		
		4.9e-8	3.1e-6
		3.3e-6	1.2e-6
SG(s) Overfed			
SG(s) Breach + SG(s) Overfed			
	1.3e-6		

(1) HPI activated without being controlled
(2) HPI activated and controlled
(3) HPI is failed or not required
(4) HPI fails and is recovered

After screening out non-PTS risk sequences, **94%** of the total sequence frequencies fall in this cell;
TH uncertainty analysis focuses on this dominant cell

The cell is further divided into four categories for TH uncertainty analysis:

1. **PZR SRV stuck open and remains open with valve open area greater than 1.5-inch in diameter**
2. **PZR SRV stuck open and is reseated with valve open area greater than 1.5-inch in diameter**
3. **LOCA between ~ 1.5-inch and 4-inch in diameter**
4. **LOCA between 4-inch and 8-inch in diameter**

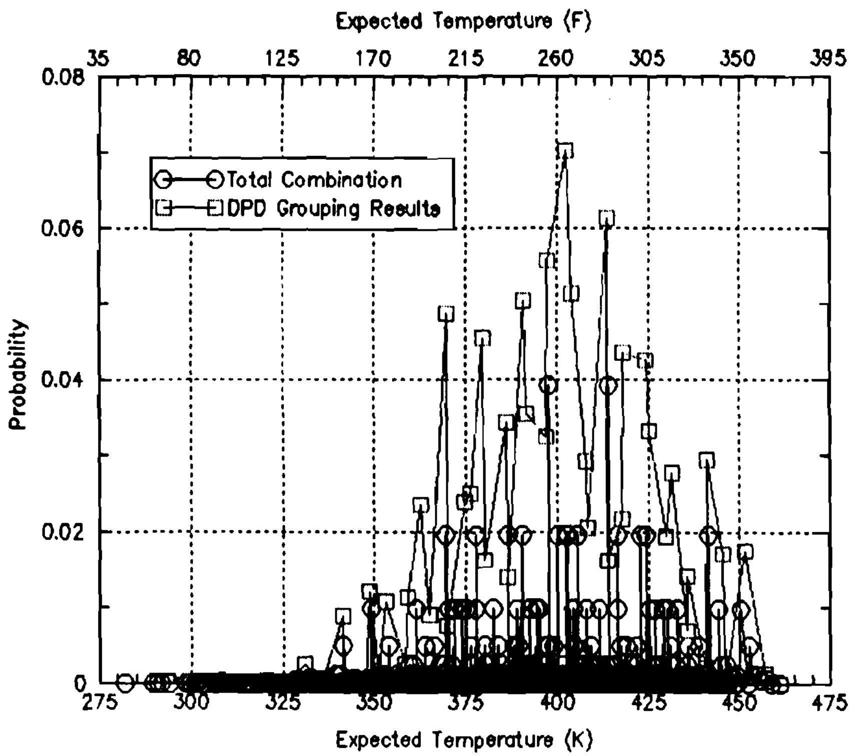
PZR SRV Stuck Open and Remains Open

Assessment of Discrete Probability Distribution of Various Variables

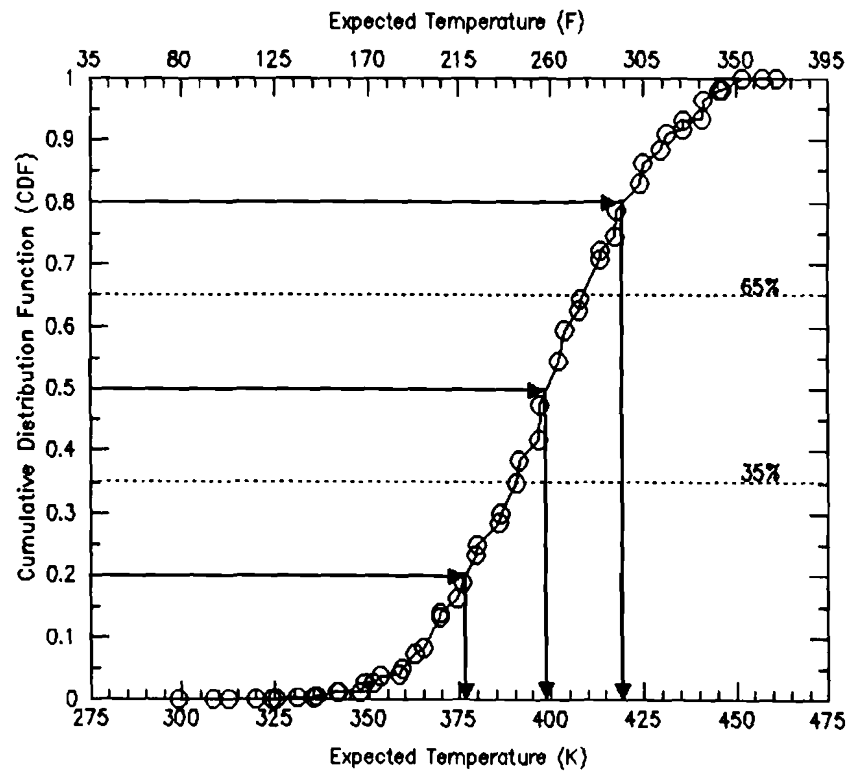
	Factors	Value 1 / Probability of Value 1	Value 2 / Probability of Value 2	Value 3 / Probability of Value 3
Parametric (Boundary Condition) Uncertainty	**Valve(s) Total Open Area**	1.5”	1.8” (Fully open)	--
		0.5	0.5	--
	Break Location	Not applicable	--	-
		--	-	
	Decay Heat	Nominal	HZP	--
		0.98	0.02	--
	Season	Winter	Spring/Fall	Summer
		0.25	0.5	0.25
	HPI State	Fail	Success	--
		0.0	1.0	--
	HPI Flow Rate	90%	Nominal	110%
		0.1	0.8	0.1
	CFTs pressure	Unimportant	--	--
		--	--	--
RELAP5 Code Model Uncertainty	**RVVVs state**	Fully close	Nominal	Fully open
		0.25	0.5	0.25
	Component Heat Transfer Rate	70%	Nominal	130%
		0.1	0.8	0.1
	Flow Resistance	Unimportant	--	--
		--	--	--
	Break Flow Rate (Break Area)	70%	Nominal	130%
		0.25	0.5	0.25
	Numerical “Mixing”	High CL rev. flow resistance	Nominal	
		1.0	0.0	

PZR SRV Stuck Open and Remains Open

PDF and CDF of Combined Effect of Multiple Uncertainty Variables



(992 combinations in total)



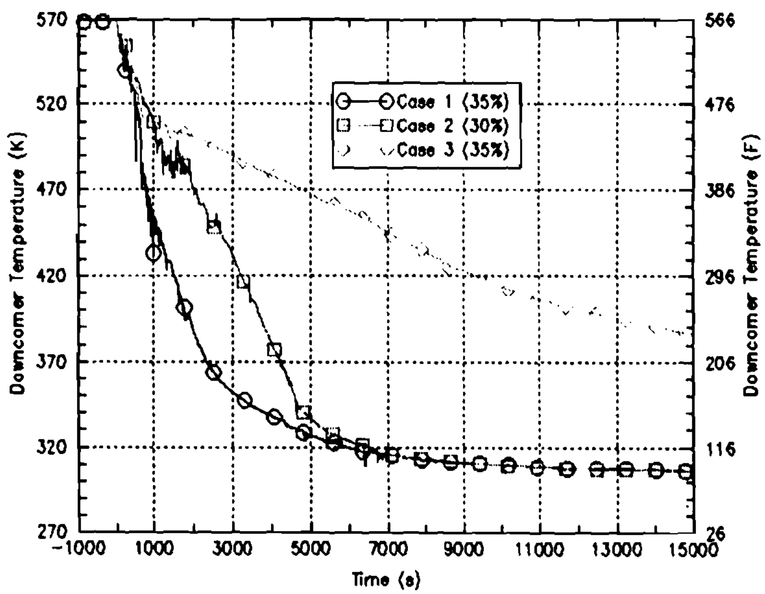
PZR SRV Stuck Open and Remains Open

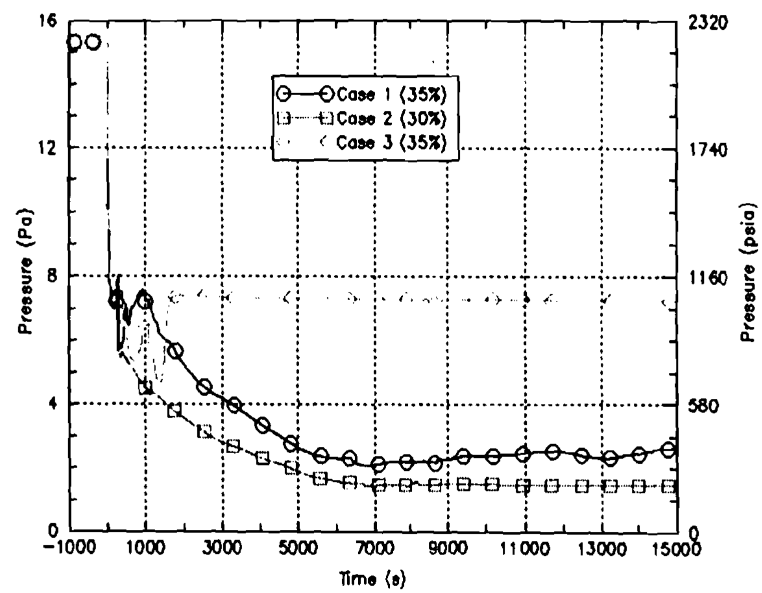
Selected TH Runs Covering Uncertainty

TH Run ID	Description	TH Case Probability Preliminary Mean Frequency
146	PZR SRV Stuck Open (fully open) with • Reduce 30% valve open area • Summer [T(HPI) = 85F, T(CFT) = 100F, and T(LPI) = 85F] • 3. RVVVs Closed • 4. High CL Rev. K	35% 2.9e-4 * 0.35 = 1.0e-4 per yr
147	PZR SRV Stuck Open (fully open) with • Summer [T(HPI) = 85F, T(CFT) = 100F, and T(LPI) = 85F] • 2. High CL Rev. K	30% 2.9e-4 * 0.30 = 9.0e-5 per yr
148	PZR SRV Stuck Open with • 1.5-inch valve open area • 130 % Component heat transfer coefficient • RCPs trip • 4. High CL Rev. K	35% 2.9e-4 * 0.35 = 1.0e-4 per yr

PZR SRV Stuck Open and Remains Open

Tdc and Pdc Trends for TH Runs Covering Uncertainty





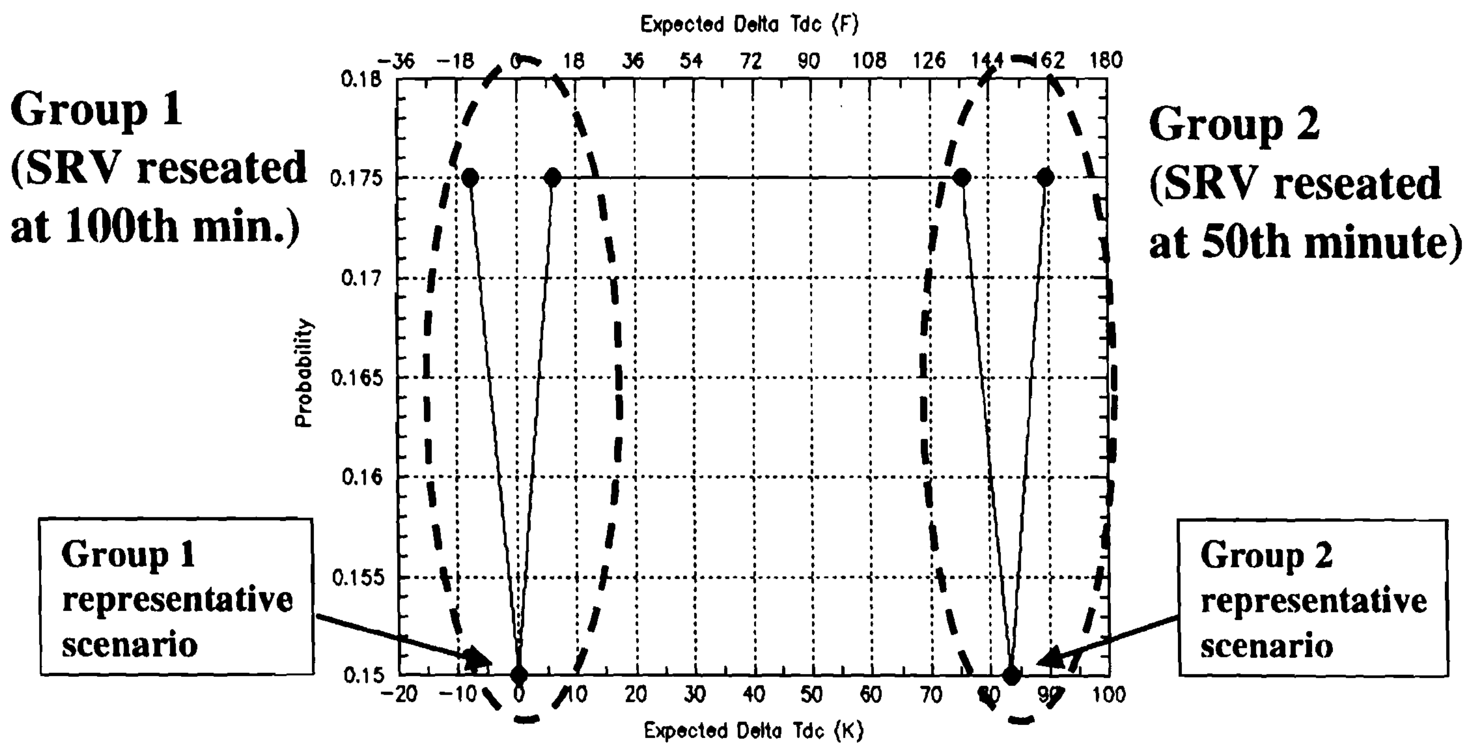
PZR SRV Reseats after being Stuck Open

Tdc and Pdc Representative Scenarios

- **Involves RCS repressurization**
  - **Considered combinations of T_{dc} and P_{dc} representative scenarios**
- **T_{dc} representative scenarios combined:**
  - **Representative scenarios in the event category SRVs Stuck Open and Remain Open (3 representatives)**
  - **SRV Reseating time (50 and 100 minutes) (2 representatives)**
- **P_{dc} representative scenarios include:**
  - **Different HPI throttling times (1 minute, 10 minutes, and not throttled) (3 representatives)**

PZR SRV Reseats after being Stuck Open

Tdc Representative Scenario Selection



- We considered six and selected two representative scenarios
- Timing of the SRV reseating dominates the uncertainty

PZR SRV Reseats after being Stuck Open

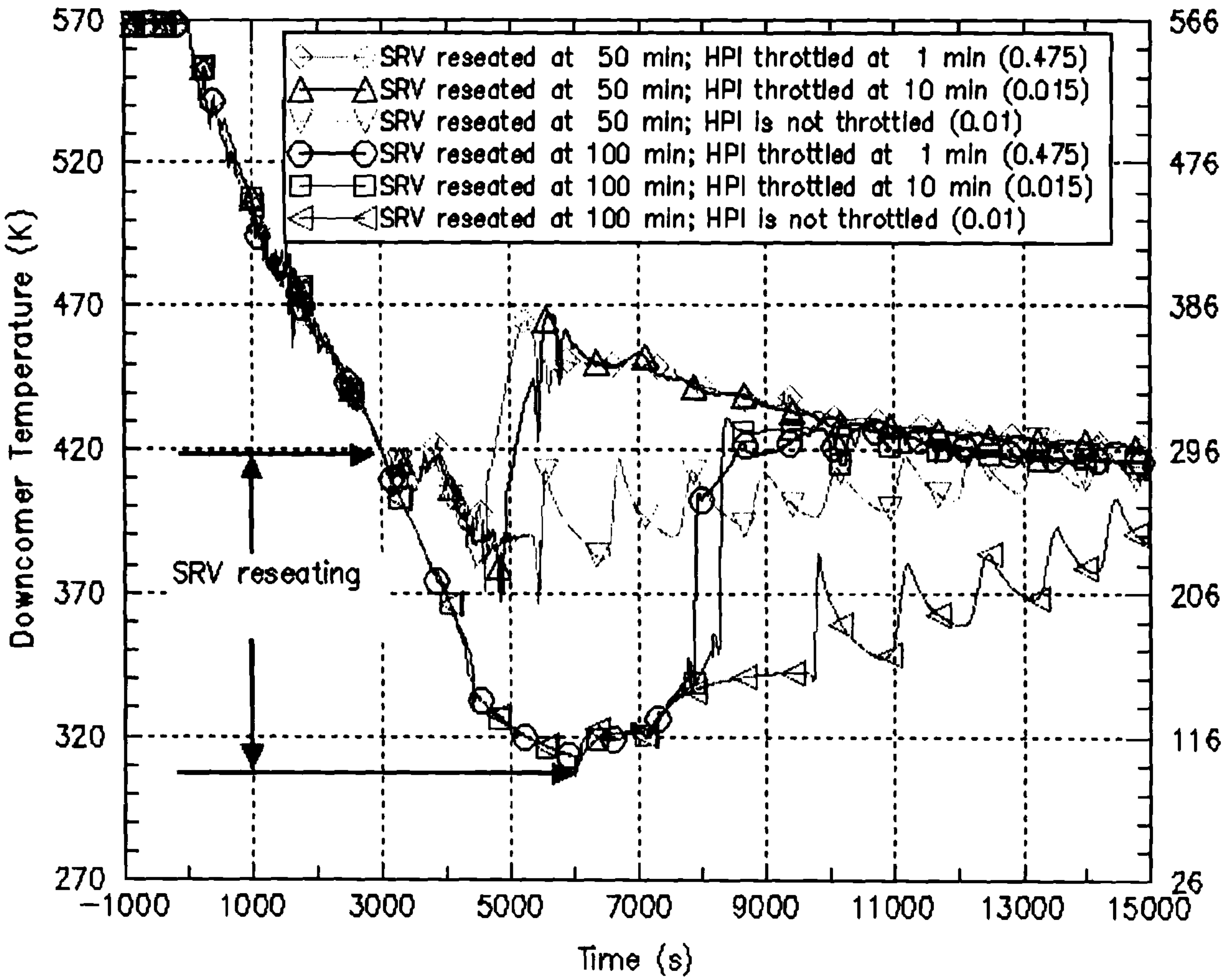
Representative Scenarios and Corresponding Probabilities

TH Run ID	Descriptions	TH Case Probability Preliminary Frequency
112	50th percentile + SRV reseated at 100 minutes + HPI Throttled at 1 minute after it could be throttled	0.475 4.3e-4 = 0.475 × 9.1e-4 /yr
113	50th percentile + SRV reseated at 100 minutes + HPI Throttled at 10 minute after it could be throttled	0.015 1.6e-5 = 0.015 × 9.1e-4 /yr
109	50th percentile + SRV reseated at 100 minutes + HPI is not throttled	0.01 1.0e-5 = 0.01 × 9.1e-4 /yr
114	50th percentile + SRV reseated at 50 minutes + HPI Throttled at 1 minute after it could be throttled	0.475 4.3e-4 = 0.475× 9.1e-4 /yr
115	50th percentile + SRV reseated at 50 minutes + HPI Throttled at 10 minute after it could be throttled	0.015 1.6e-5 = 0.015 × 9.1e-4 /yr
149	50th percentile + SRV reseated at 50 minutes + HPI is not throttled	0.01 1.0e-5 = 0.01 × 9.1e-4 /yr

- **T_{dc} representative scenarios:**
  - **PZR SRV reseated at the 50th minute (0.5)**
  - **PZR SRV reseated at the 100th minute (0.5)**
- **P_{dc} representative scenarios:**
  - **HPI throttled at 1 minute after it can be throttled (0.95)**
  - **HPI throttled at 10 minutes after it can be throttled (0.03)**
  - **HPI is not throttled (0.02)**

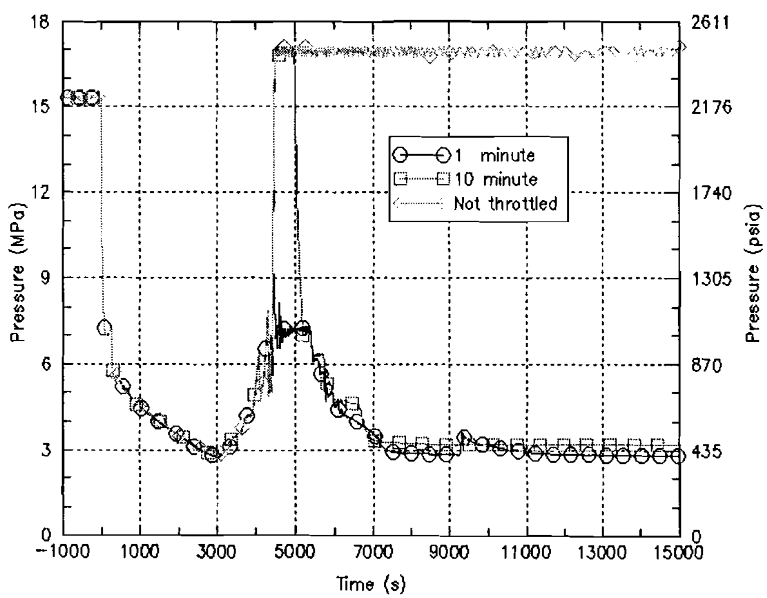
PZR SRV Reseats after being Stuck Open

T_{dc} Trends of Representative Cases

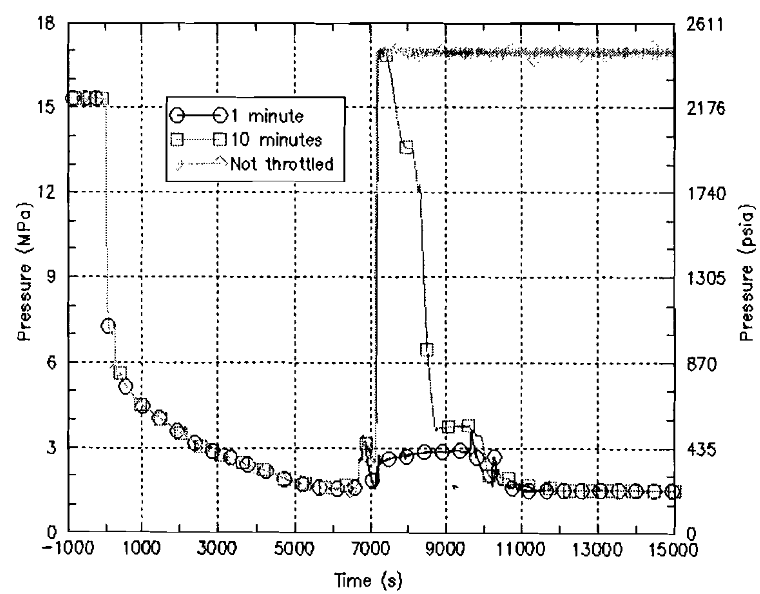


PZR SRV Reseats after being Stuck Open

Pdc Trends of Representative Cases

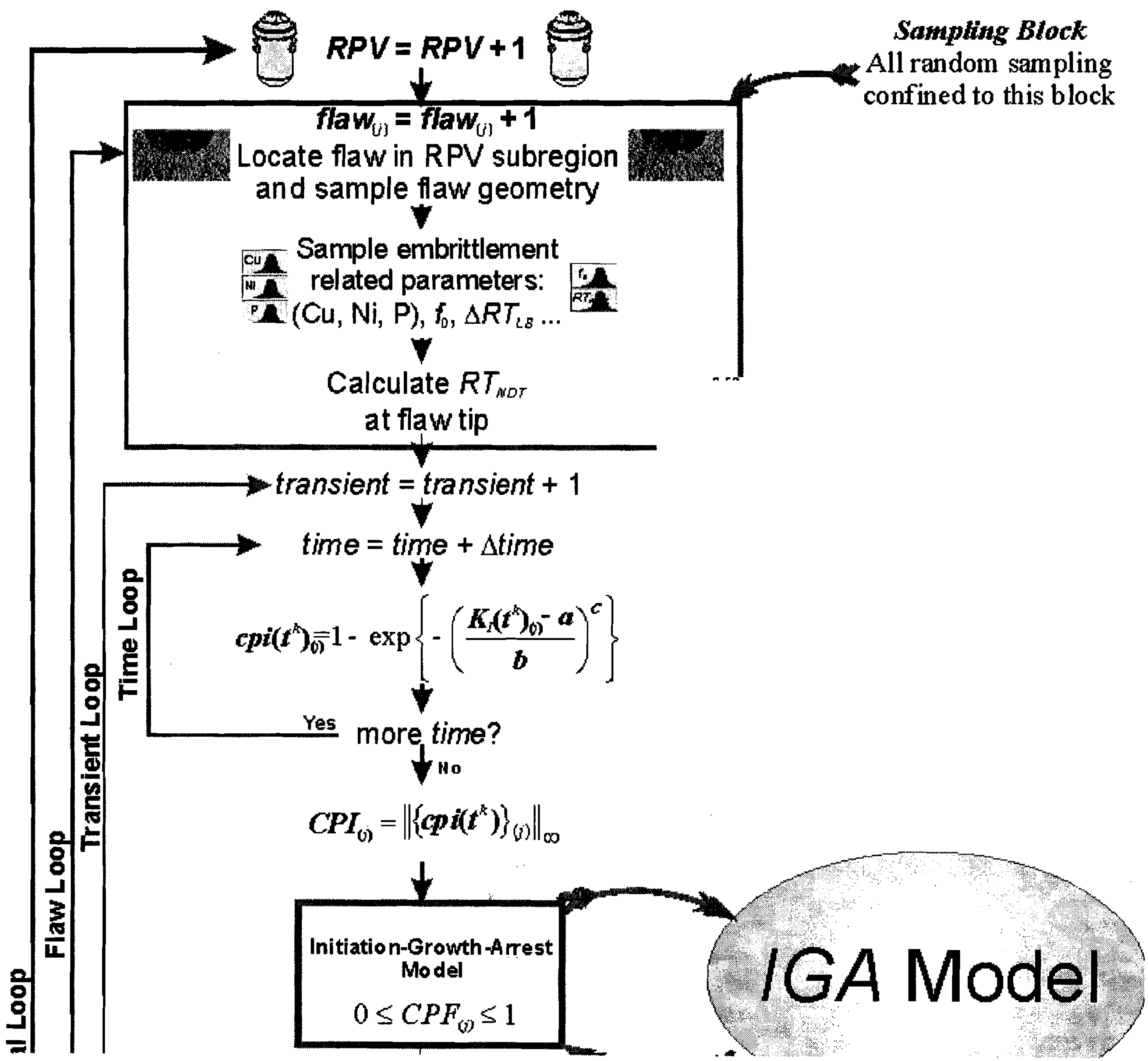


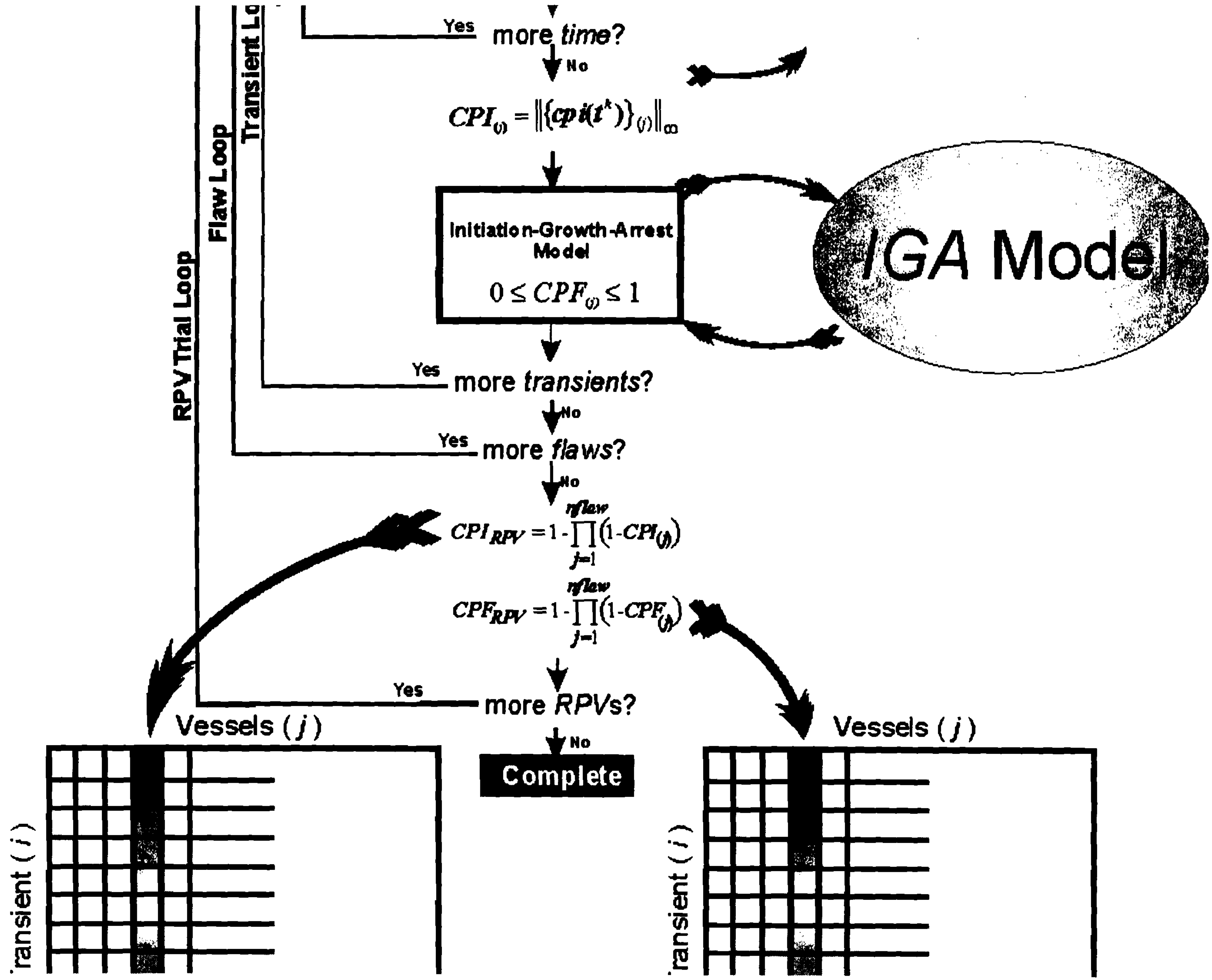
SRV reseated at 50 minutes



SRV reseated at 100 minutes

Example (PFM)

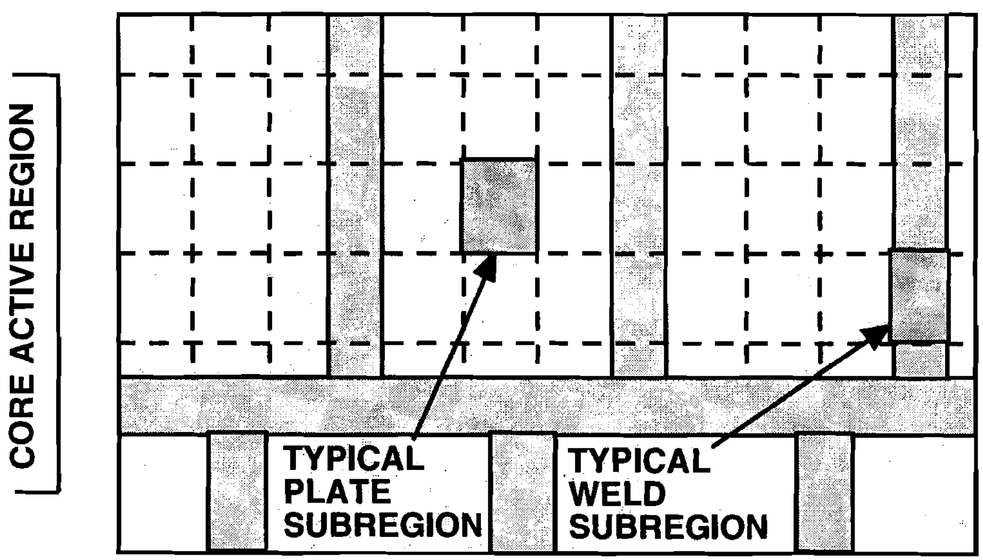

RPV = RPV + 1
Sampling Block
All random sampling confined to this block
flaw(j) = flaw(j) + 1
Locate flaw in RPV subregion and sample flaw geometry
Sample embrittlement related parameters:
(Cu, Ni, P), f0, ΔRTLS ...
Cu
Ni
P
Calculate RTNDT at flaw tip
transient = transient + 1
time = time + Δtime
cpi(t^k)(j) = 1 - exp{-((KI(t^k)(j) - a)/b)^c}
more time?
Yes
No
CPI(j) = ||{cpi(t^k)}(j)||∞
Initiation-Growth-Arrest Model
0 ≤ CPF(j) ≤ 1
IGA Model
Time Loop
Transient Loop
Flaw Loop
l Loop



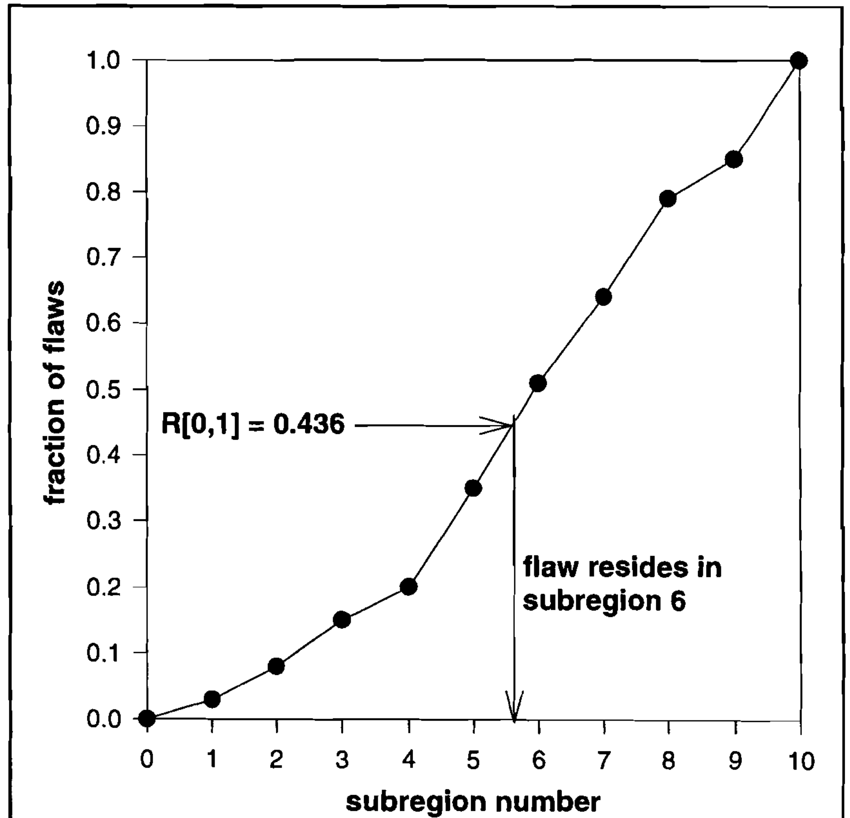
Yes
more time?
No
$CPI_{(j)} = \|\{cpi(t^k)\}_{(j)}\|_\infty$
Initiation-Growth-Arrest Model
$0 \le CPF_{(j)} \le 1$
IGA Model
Yes
more transients?
No
Yes
more flaws?
No
$CPI_{RPV} = 1 - \prod_{j=1}^{nflaw}\left(1 - CPI_{(j)}\right)$
$CPF_{RPV} = 1 - \prod_{j=1}^{nflaw}\left(1 - CPF_{(j)}\right)$
Yes
more RPVs?
No
Complete
Transient Loop
Flaw Loop
RPV Trial Loop
Vessels (j)
Transient (i)
Vessels (j)
Transient (i)

Simulation of Flaw Location

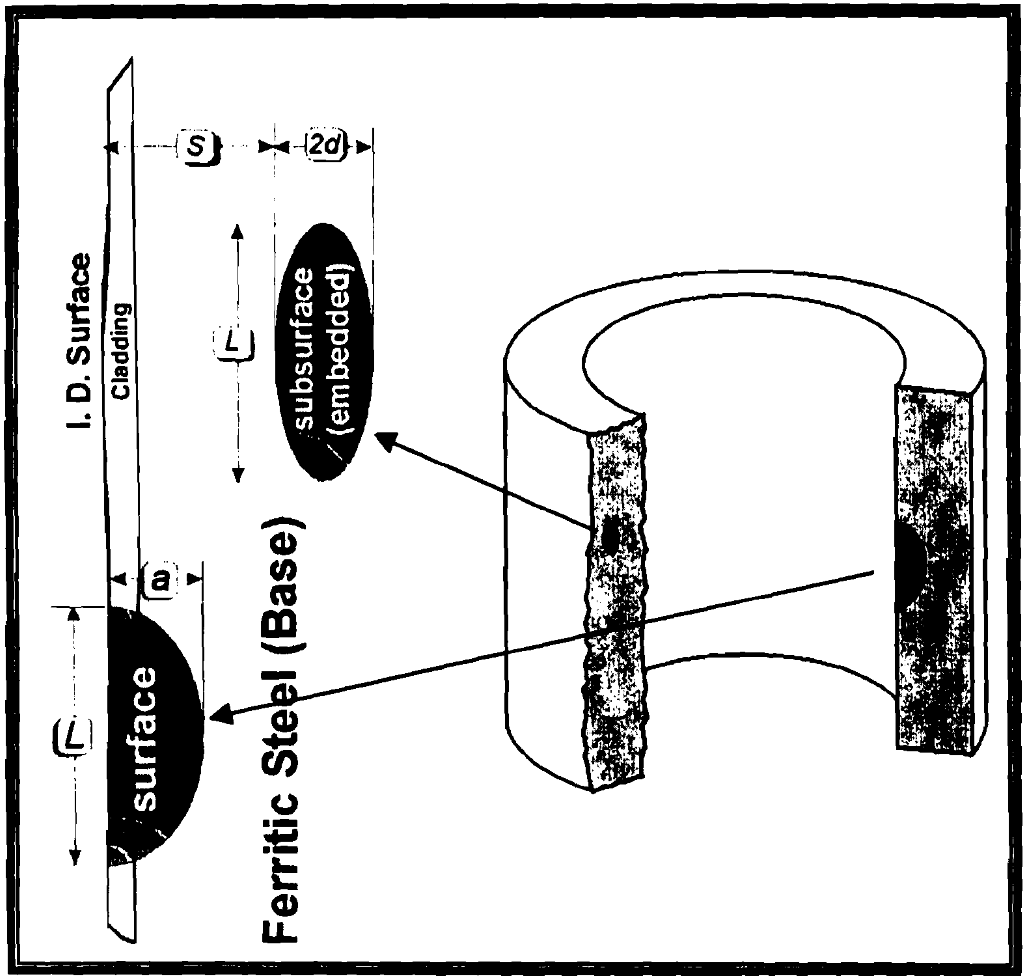
FAVOR locates each flaw in a particular RPV sub-region by sampling from a cumulative distribution function (CDF) that expresses the fraction of total flaws as a function of subregion number



TYPICAL VESSEL BELTLINE LAYOUT



Simulation of Flaw Characteristics



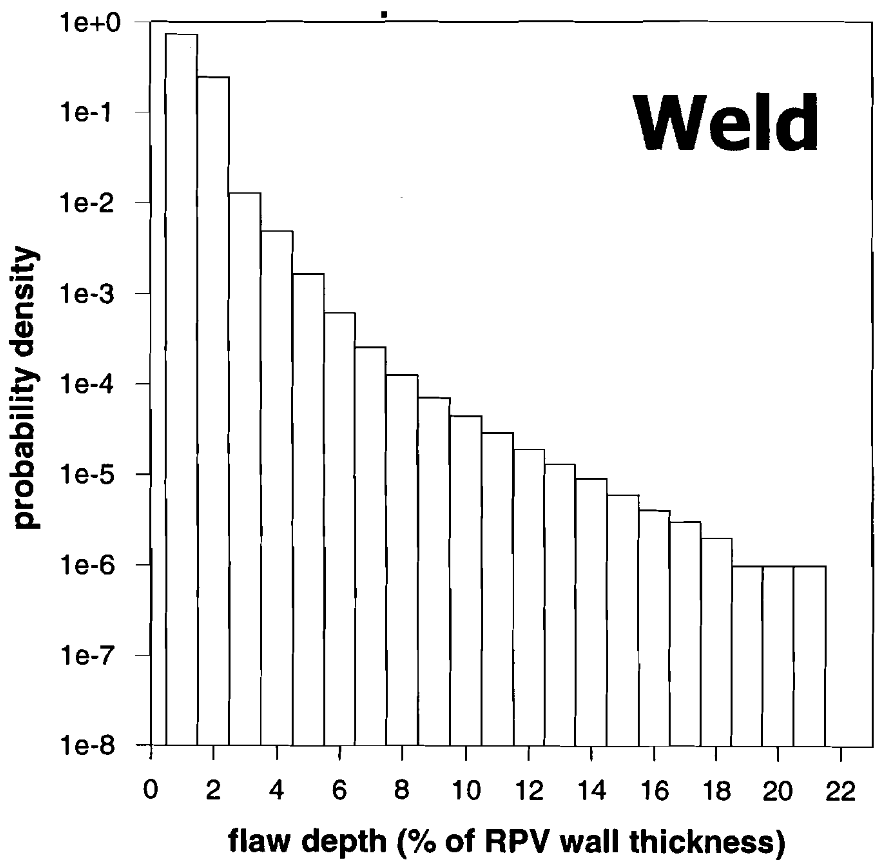
FAVOR determines the characteristics of each flaw by sampling CDFs generated from flaw characterization data to determine:

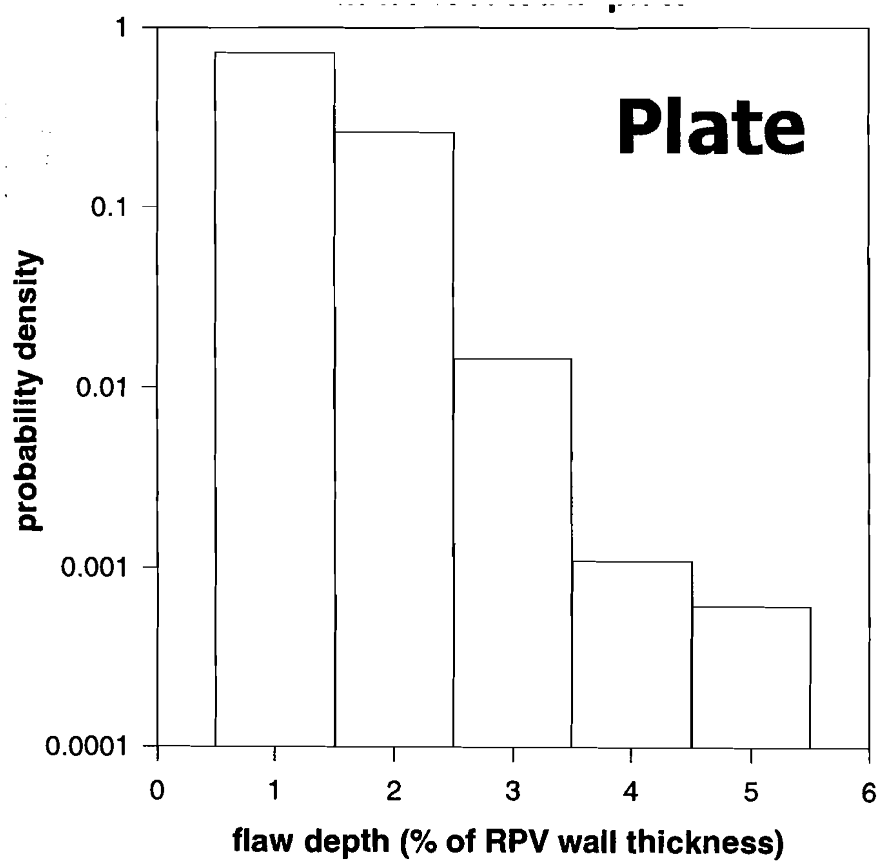
- → **flaw depth**
- → **flaw length**
- → **location of inner crack tip**

Flaw Size & Density Uncertainty

Sample from 1000 different flaw distributions, each containing different flaw densities and flaw size distributions.

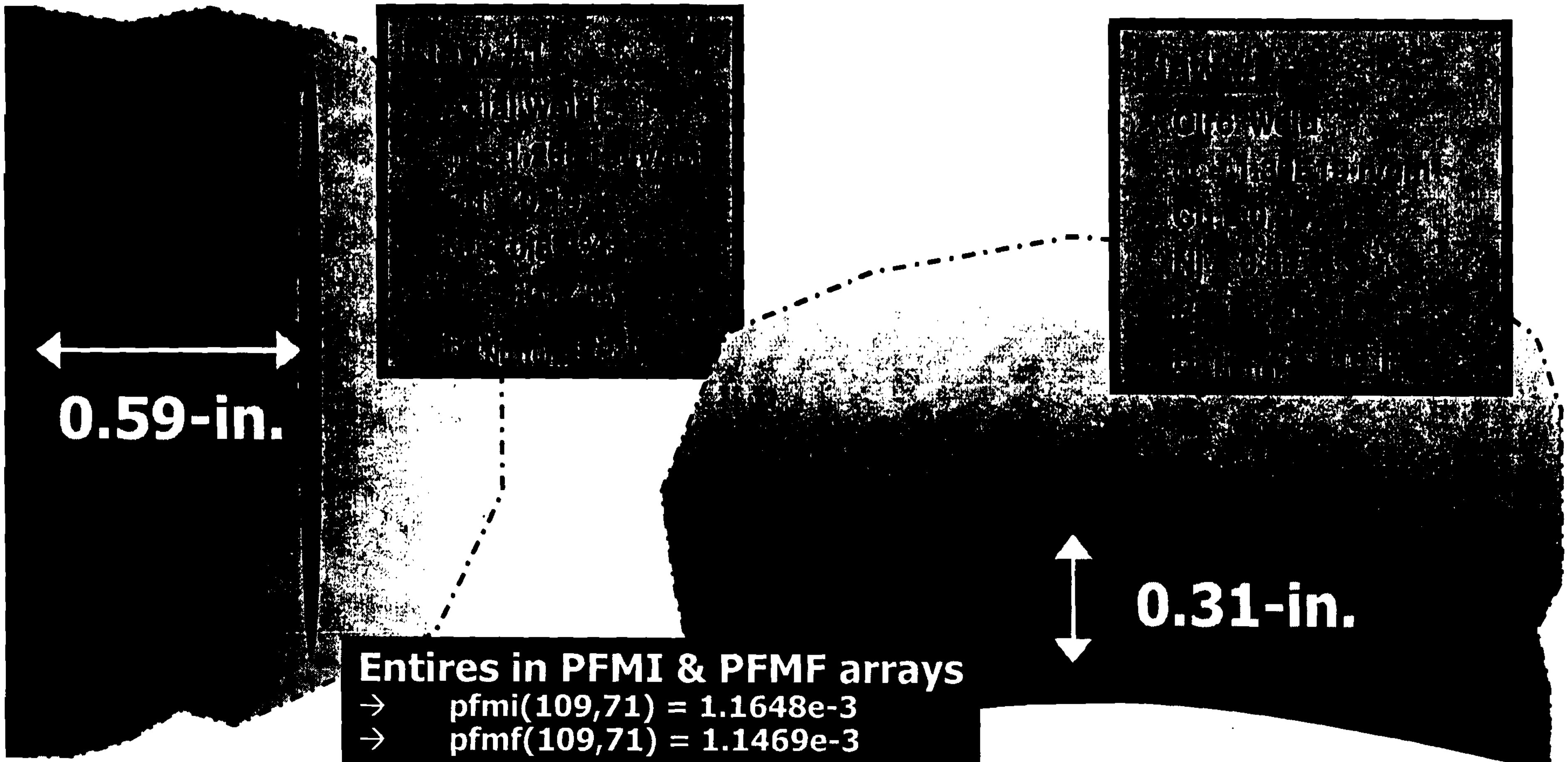
Plate and weld are treated separately.





Demonstration of PFM Methodology

Track the first two flaws subjected to transient #109 that have a non-zero conditional probability of crack initiation (i.e. make some contribution to the estimated vessel failure frequency)



Treatment of Multiple Flaws

- **For 1 flaw in a RPV**
  - **Probability of crack initiation =** **CPI(1)**
  - **Probability of non-initiation =** **(1 - CPI(1))**
- **For 2 flaws in a RPV**
  - **Probability of crack initiation =** **CPI(1) & CPI(2)**
  - **Probability of non-initiation =** **(1 - CPI(1)) * (1 – CPI(2))**
- **For n flaws in a RPV**

$$CPI_{RPV} = 1 - \prod (1-CPI_1)(1-CPI_2)\ldots(1-CPI_n)$$

- **CPI (109,71) = (1-1.144E-3) (1-2.06E-5) = 1.165E-3**
- **CPF(109,71) = (1-1.144E-3) (1-2.7E-6) = 1.147E-3**

Estimation of RT_{NDT}

RT_{NDT} (at the crack tip) is the sum of the initial (unirradiated) value $RT_{NDT(u)}$ and the radiation-induced shift, ΔT_{30}.

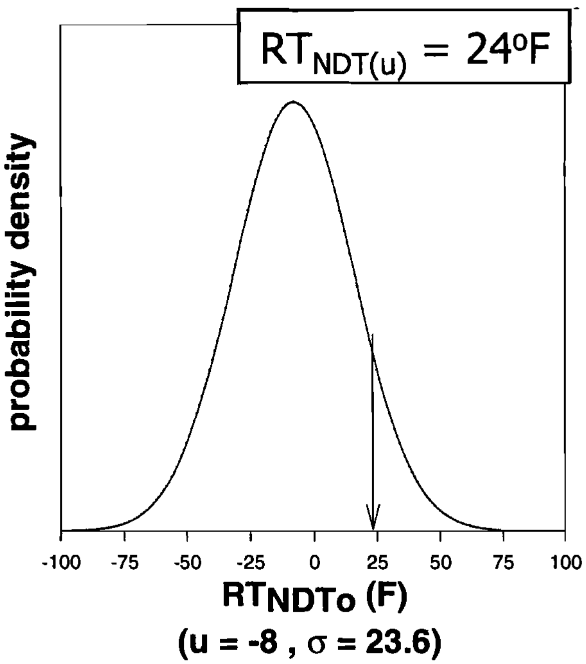
$$RT_{NDT} = RT_{NDT(u)} + \Delta T_{30}$$

for example: flaw 1

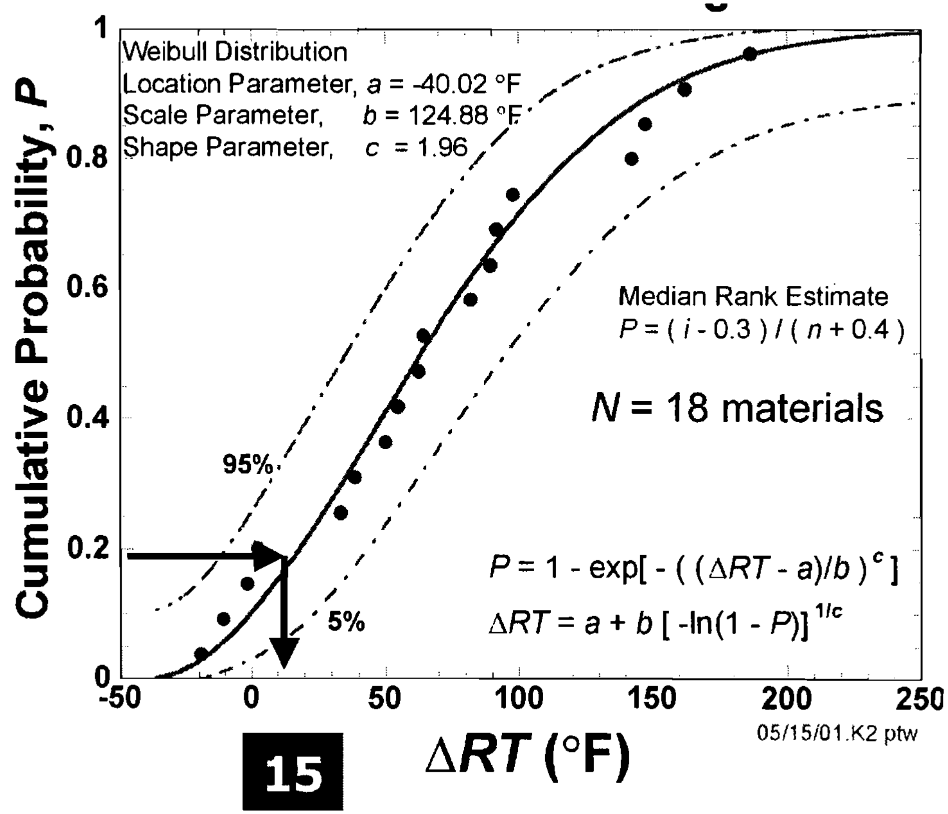
[redacted]

Estimation of $RT_{NDT(u)}$

Epistemic *(lack of data)*
(for Generic $RT_{NDT(u)}$ only)



Epistemic *(model)*



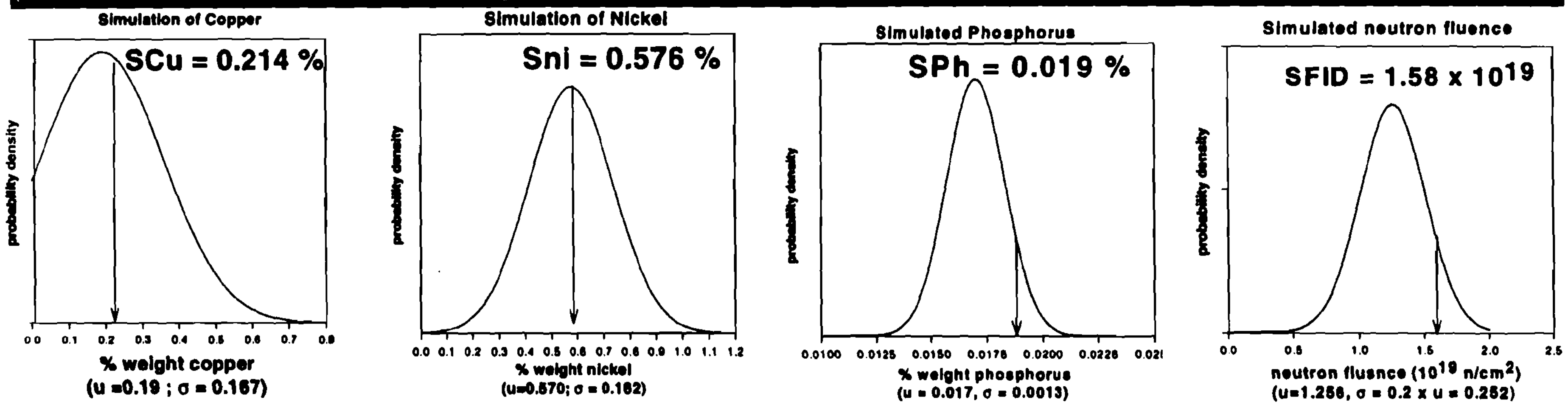
$RT_{NDT(u)} = 24 - 15 = 9°F$

VG 158

Estimation of ΔT_{30}

The mean value of the radiation-induced shift in RT_{NDT} is a function of sampled values of chemistry and neutron fluence

$$(\Delta T_{30})_{mean} = f(S_{Cu}, S_{Ni}, S_{P}, S_{\phi t})$$

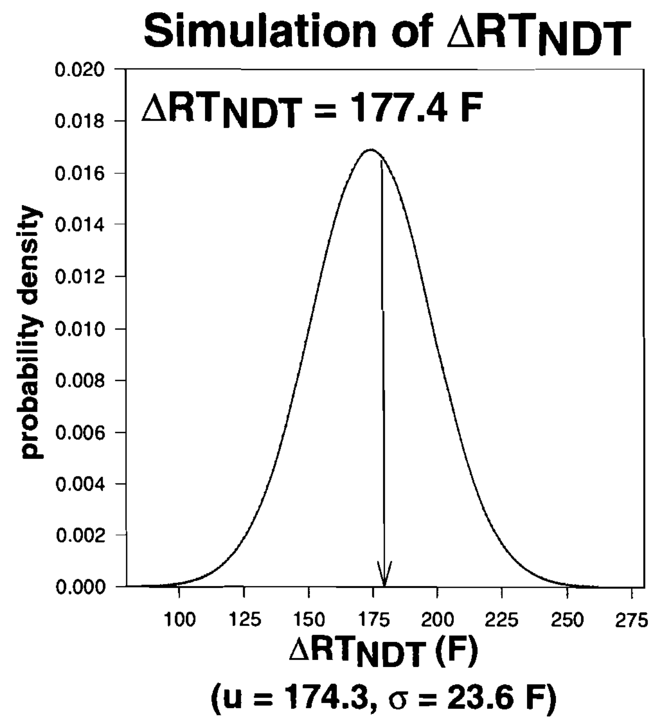


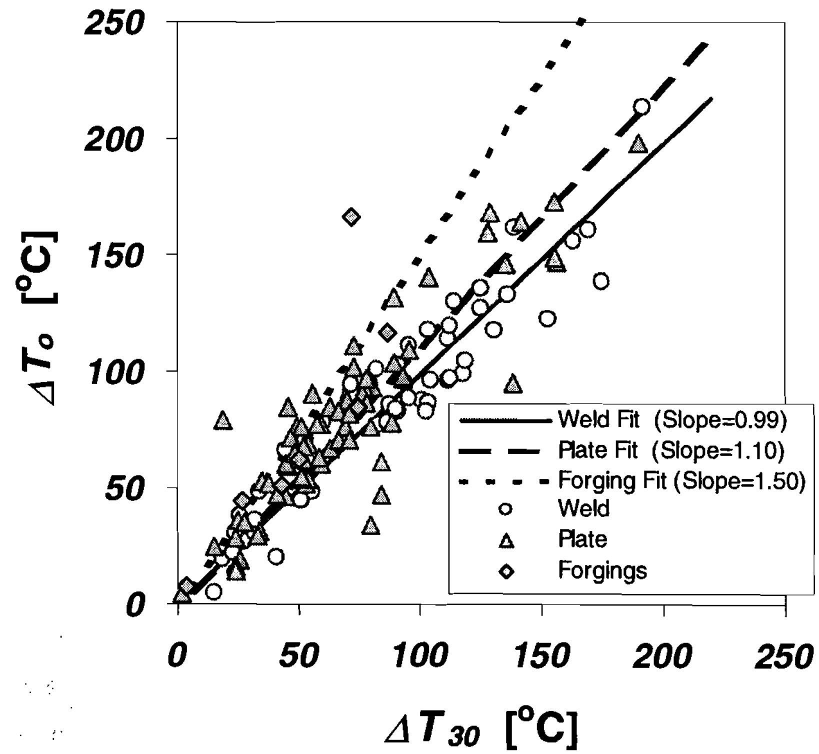
[redacted]

Uncertainty in the embrittlement shift is accounted for by sampling from these distributions of the parameters Cu, Ni, P, and ϕt.

Uncertainty in ΔT_{30}

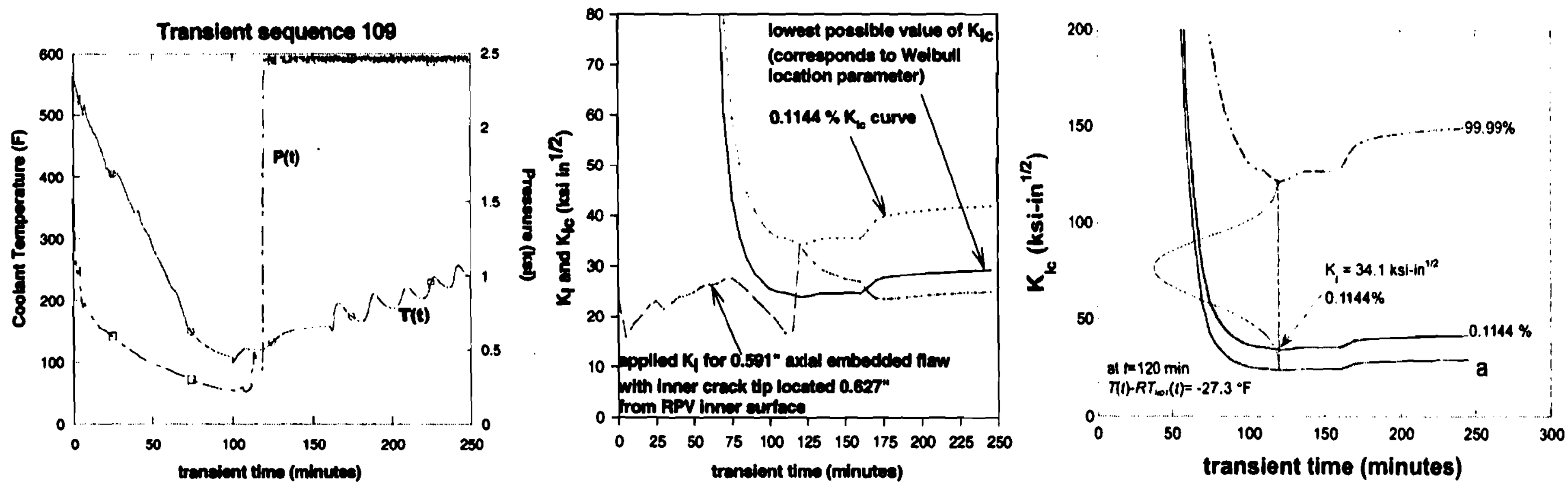
The uncertainty in the radiation-induced shift in RT_{NDT} is determined by first adjusting ΔT_{30} to account for differences between CVN and fracture toughness transition, and then sampling about the mean of this adjusted value to account for the uncertainty in the adjustment.





Estimating the Conditional Probability of Crack Initiation

The CPI for each flaw is calculated by solving the Weibull CDF for K_{Ic} for the fractional part (fractile) of the distribution that corresponds to the applied K_I



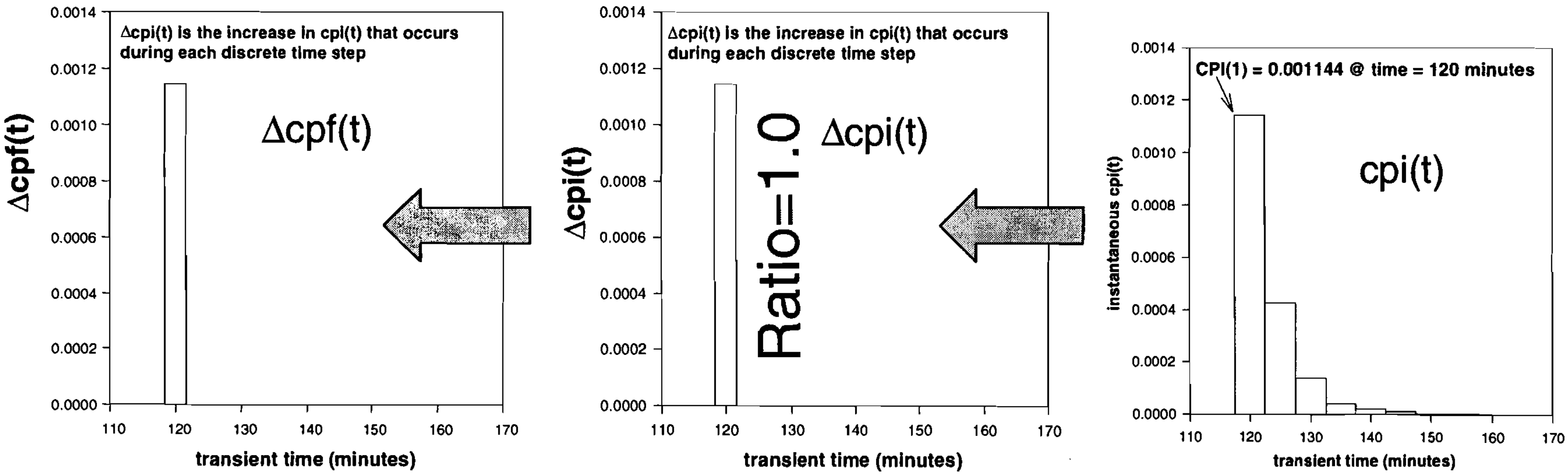
Estimating the Conditional Probability of Thru-Wall Cracking (≡Failure)

$$\Delta cpf_1(t) = \Delta cpi_1(t)\left[Ratio\right]$$

$$\text{where } Ratio = \frac{failures}{initiated\ flaw}$$

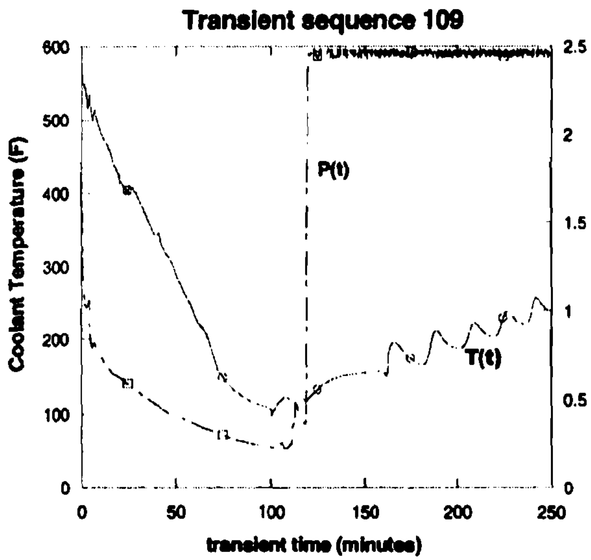
$$cpf_1 = \sum_{t=120\,\text{min}}^{t=120\,\text{min}} \Delta cpf_1(t)$$

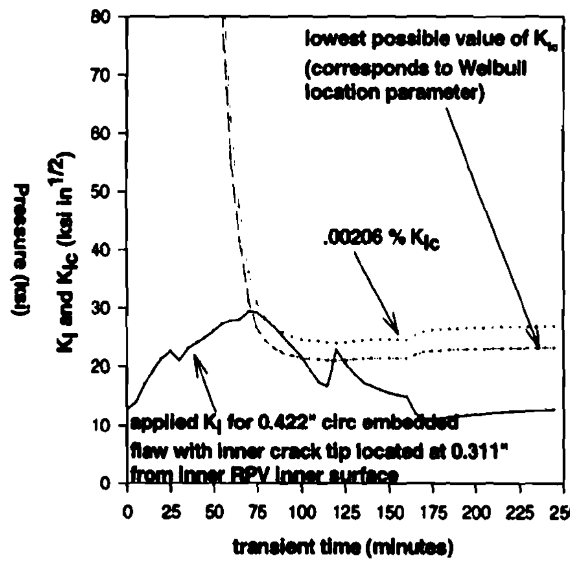
Ratio =1 ➔ All initiated flaws failed at t=120 min. (repressurization)

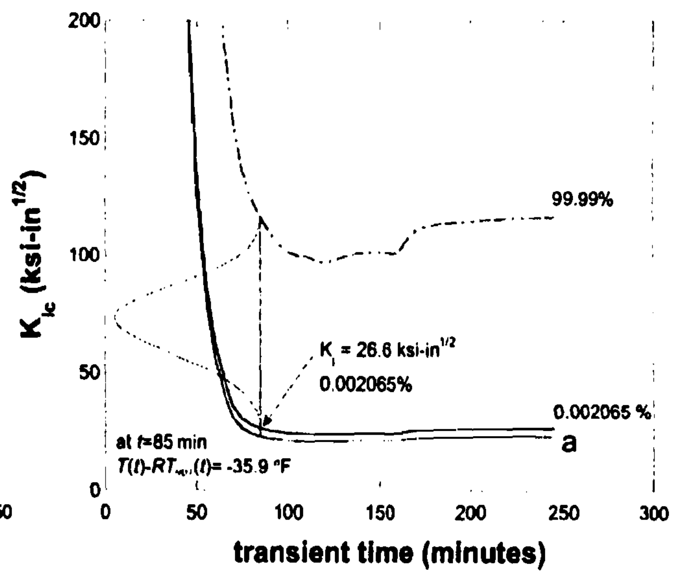


$$cpf_1 = \sum_{t=120\,\text{min}}^{t=120\,\text{min}} \Delta cpf_1(t) = 0.001144$$

CPI Calculation for Flaw #2

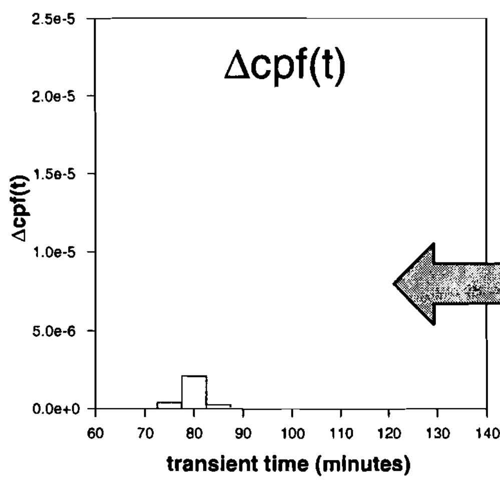


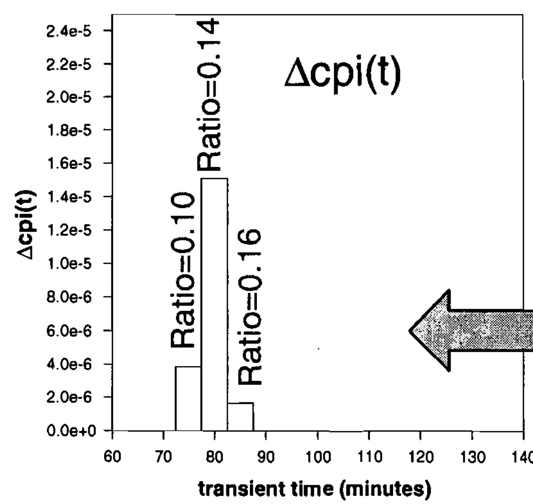


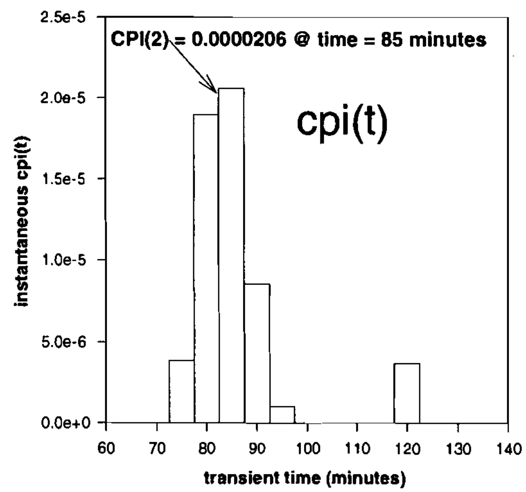


[illegible]

CPF Calculation for Flaw #2



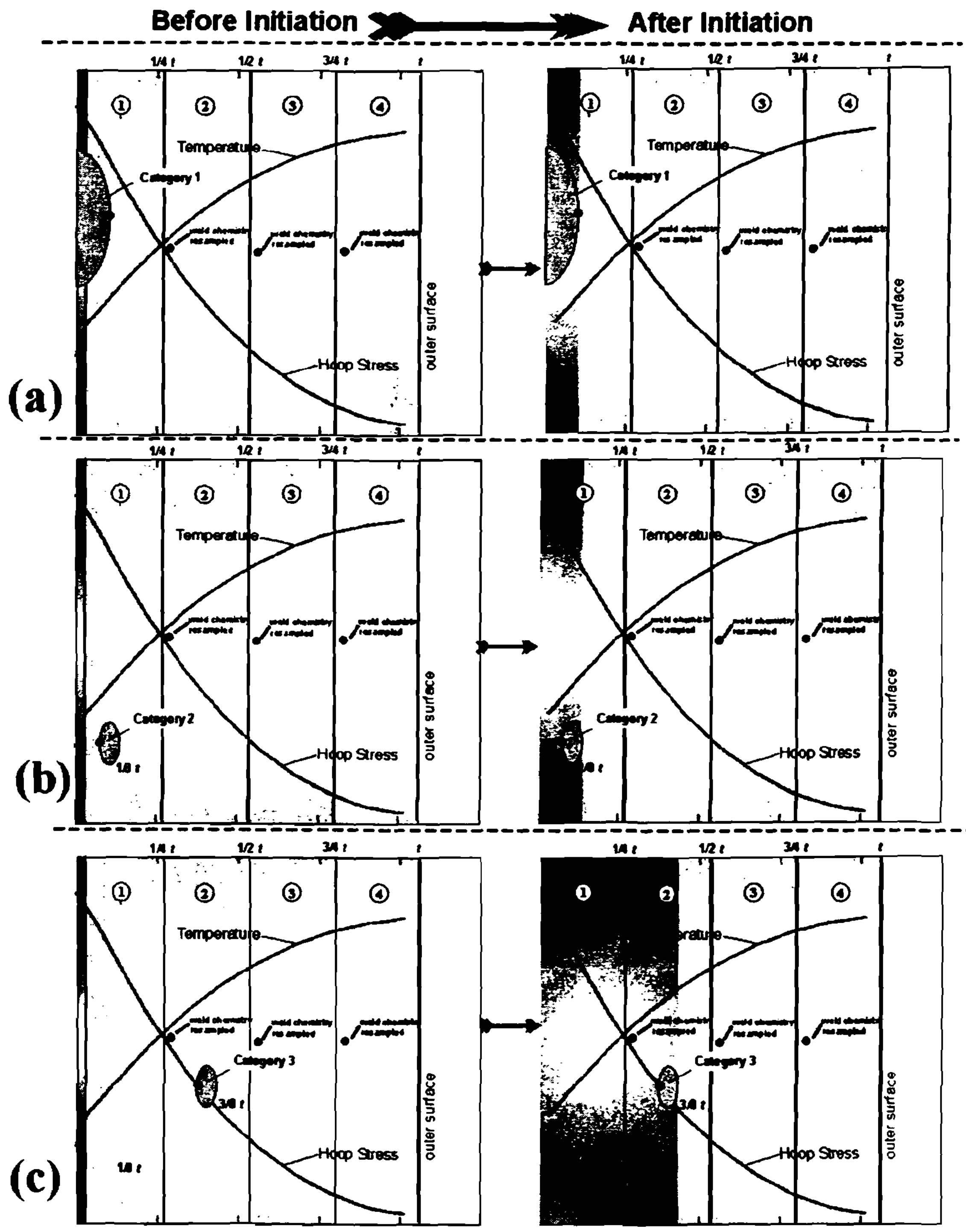




$$cpf_2 = \sum_{t=75\,\mathrm{min}}^{t=85\,\mathrm{min}} \Delta cpf_2(t) = 2.7\times10^{-6}$$

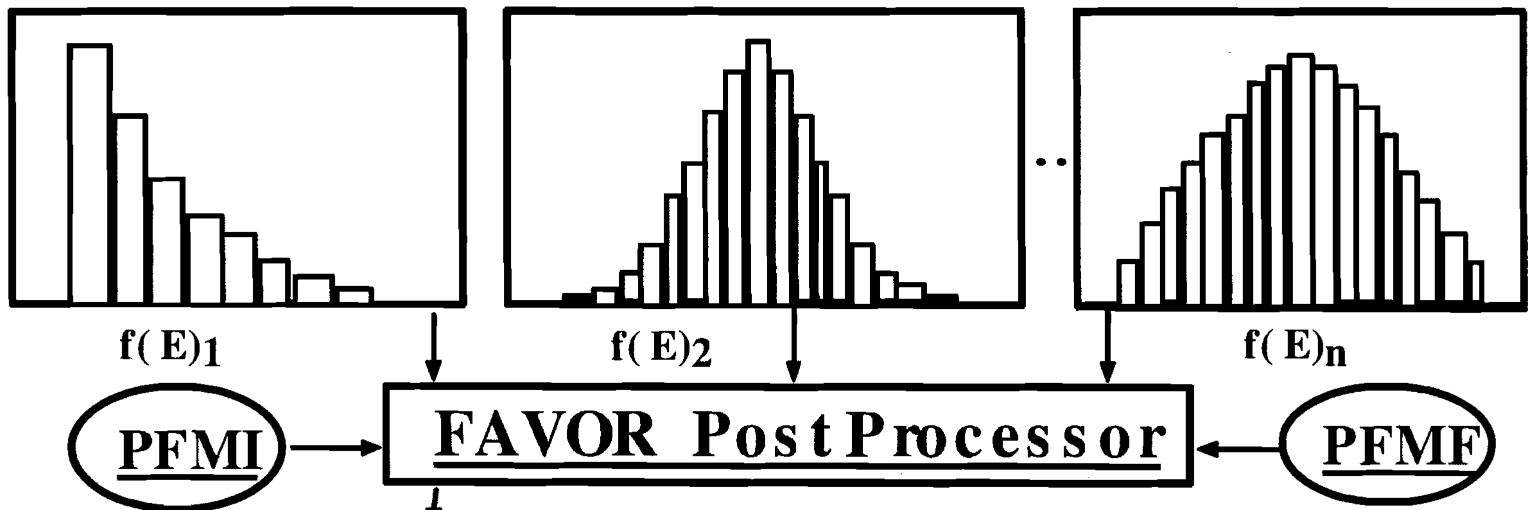
After Crack Initiation

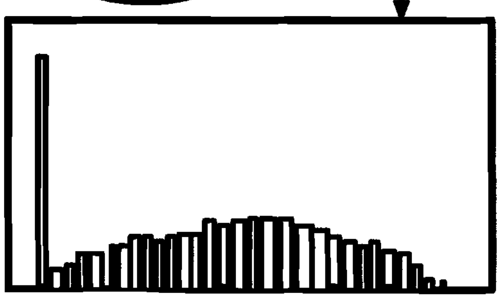
A flaw that initiates in cleavage fracture is assumed to become an infinite-length inner surface breaking flaw



VG 1

The FAVOR Postprocessor Module Integrates the Uncertainties of the Transient Initiating Frequencies with the PFMI and PFMF Arrays to Generate Distributions for the Frequencies of RPV Fracture and RPV Failure





f(F) (Frequencies of RPV fracture/failure)

For each vessel:

1. Sample initiating frequencies, $f(E)_1, \ldots f(E)_n$
2. Combine f(E) with PFM results, $f(F)_i = \Sigma f(E)_T \cdot P_{FM}(T,V)$
3. Generate histogram for f(F) from resulting array of $f(F)_i$.

Summary

ESS

(Emergency Summary Slide)

Transients that generate PTS risk

- Happen less frequently than we thought they would
- Around operators that perform better than we gave them credit for
- To a vessel that is tougher than we thought it was
- That contains smaller cracks than we thought it did

RELAP5 Assessment

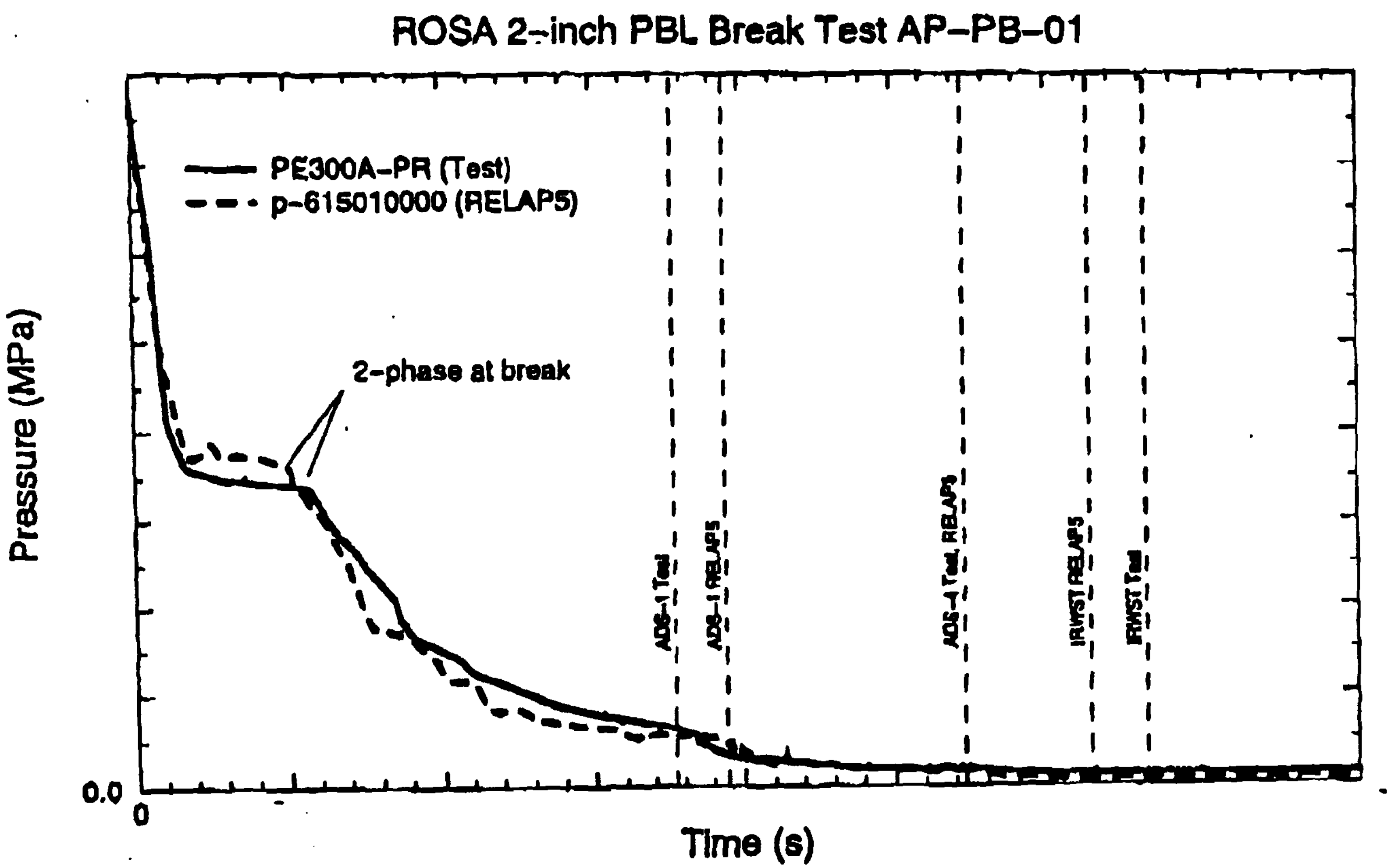
- **Uncertainties in RELAP5 PTS results relating to simulation of reactor vessel internal circulation**
  - **Uncertainties apply for transients in which reactor coolant pumps have been tripped and coolant loop natural circulation has been lost.**
  - **The RELAP5 PTS models employ a multi-dimensional downcomer noding scheme similar to that successfully employed in modeling the AP600 downcomer region.**
  - **RELAP5 is generally capable of simulating circulation upward through the core, through the vent valves and downward through the downcomer. The uncertainty in the vessel circulation rate is addressed via sensitivity studies evaluating effects of bypass size and flow loss coefficient.**

A VG14

RELAP5 Assessment

- **RELAP5/MOD3 recently underwent extensive assessments demonstrating its adequacy for simulating accidents in AP600**
  - **AP600 applications are generally more challenging to simulate than existing-plant applications because of the passive safety systems**
  - **AP600 accident phenomena and behavior include those found in PTS accident scenarios for existing plants**
  - **RCS pressure and temperature comparisons with AP600-related experimental data indicate a general RELAP5 capability for simulating these key PTS parameters**

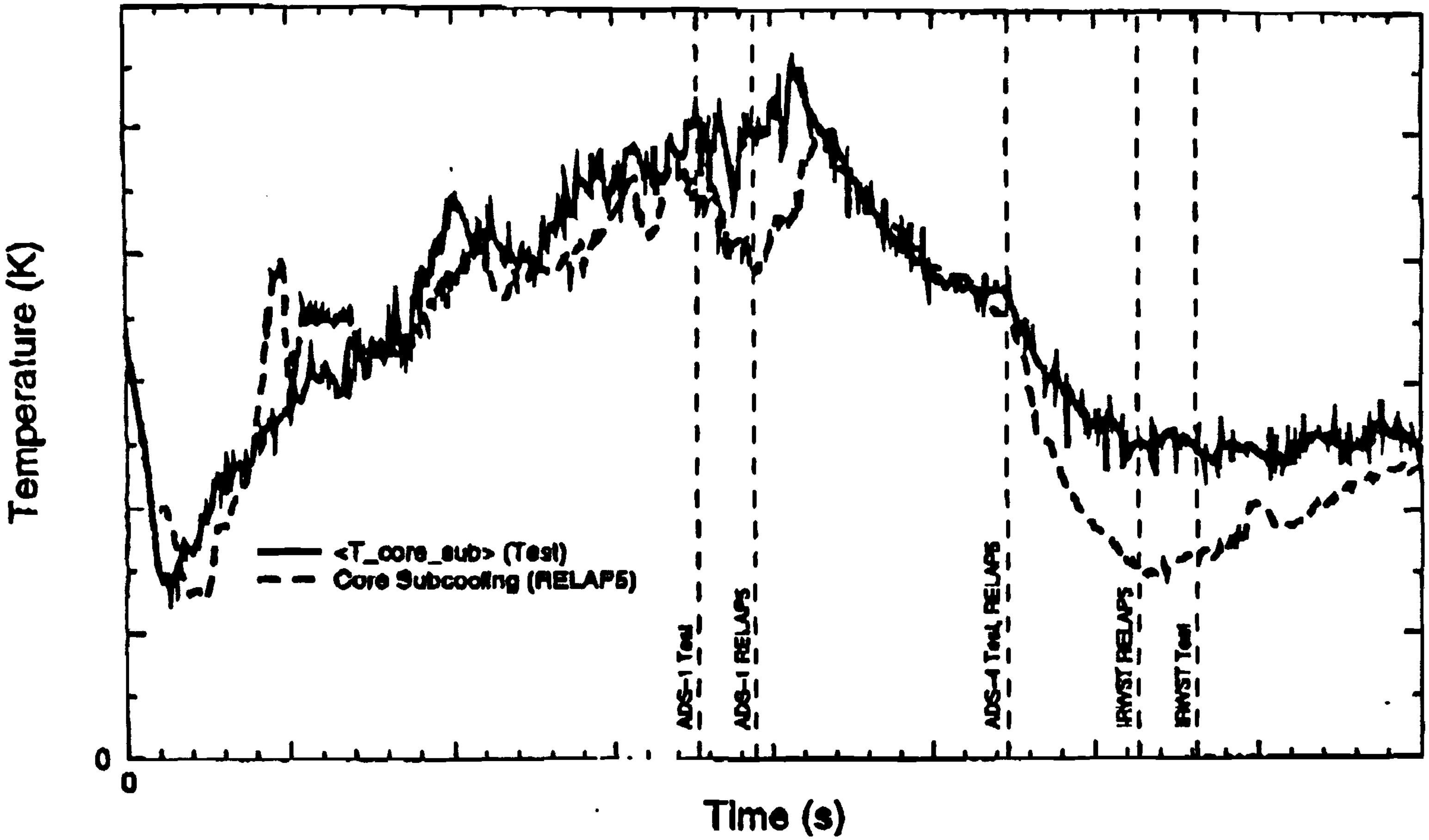
RELAP5 Assessment



A VG 11

RELAP5 Assessment

ROSA 2-inch PBL Break Test AP-PB-01

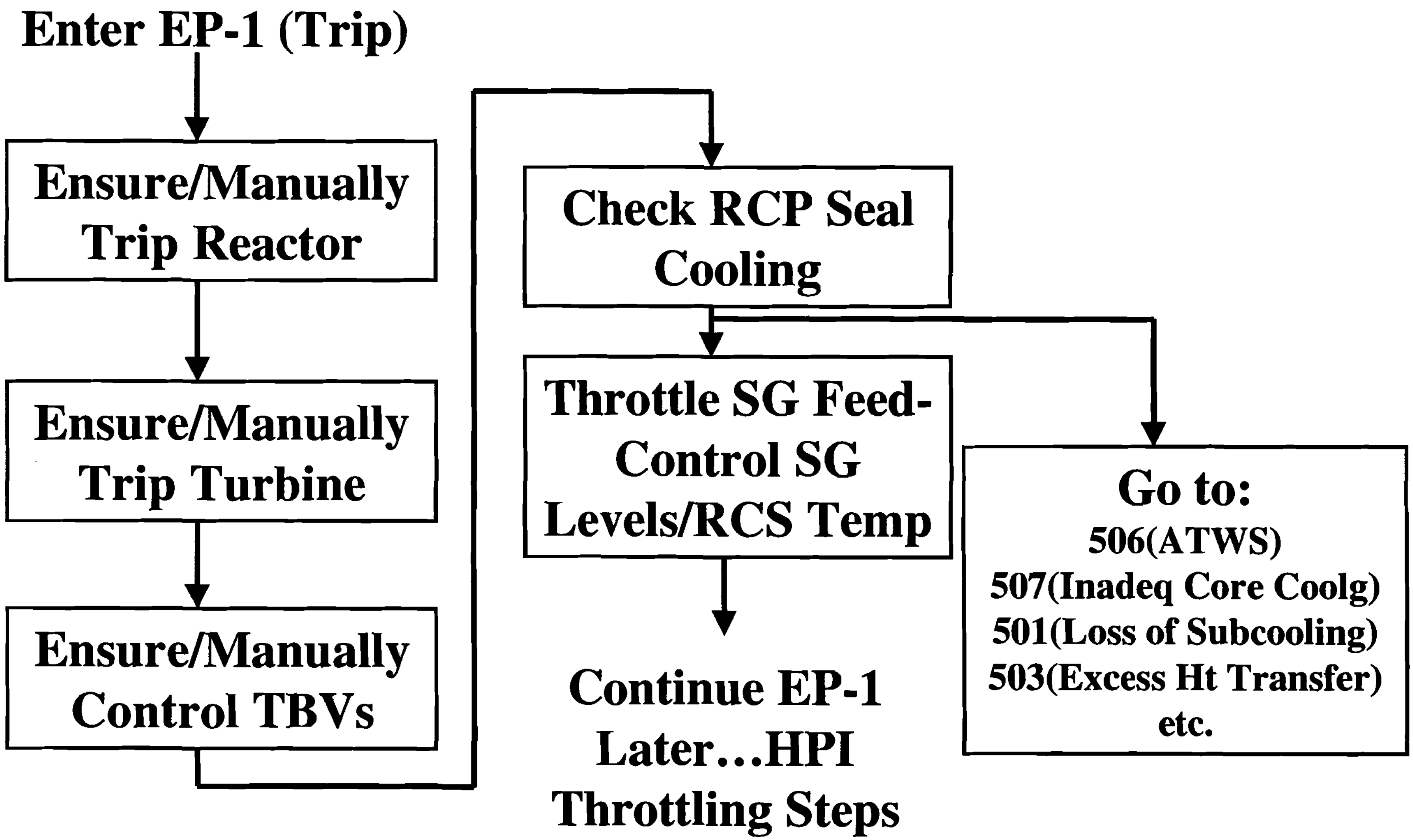


A VG12

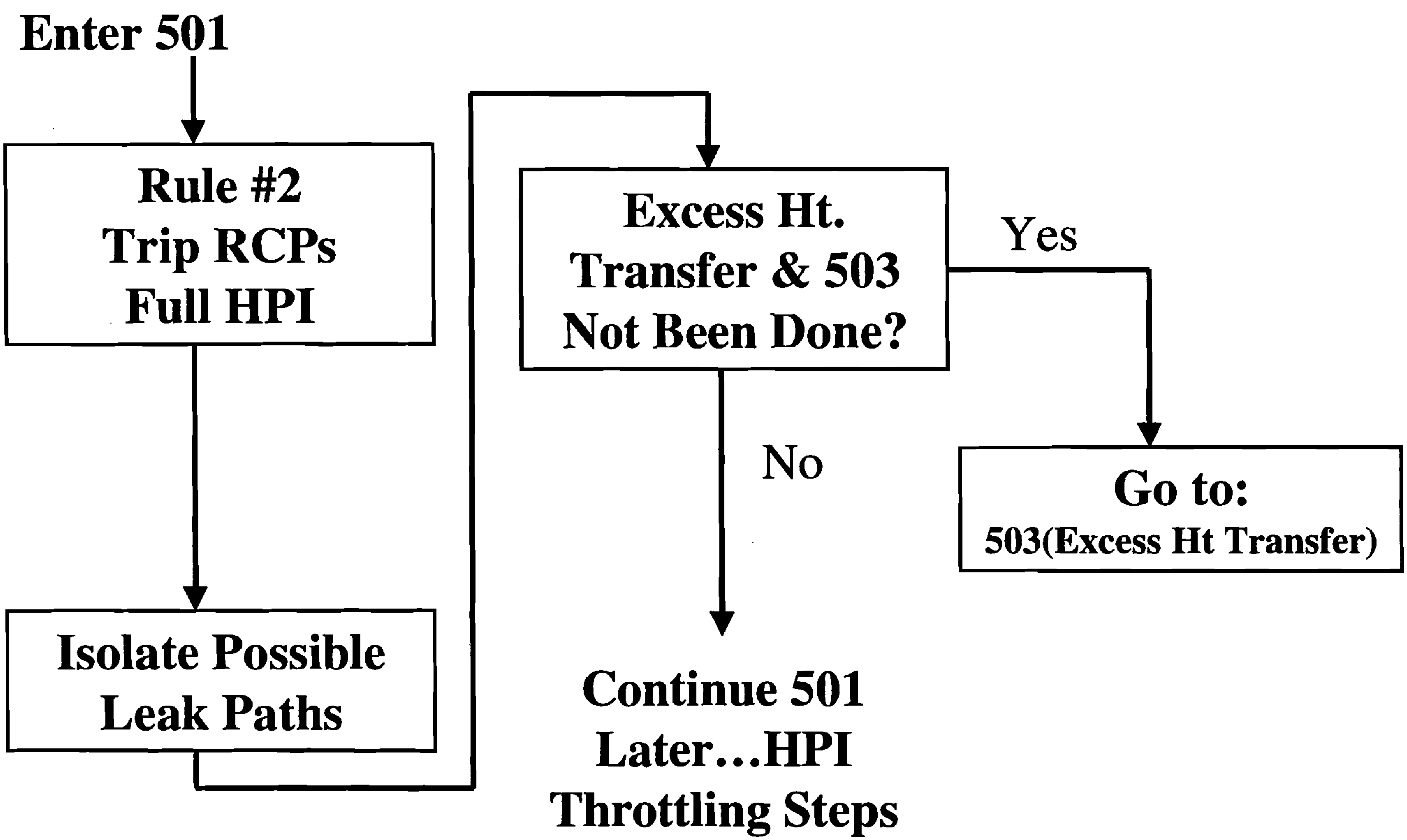
RELAP5 Assessment

- **Uncertainty in RELAP5 PTS results relating to simulation of system outflows**
  - **The outflow rate can be affected by break location (pipe, elevation and circumferential orientation) and can influence both RCS pressures and temperatures (because ECCS injection rates are typically functions of RCS pressure)**
  - **This uncertainty is generally acknowledged and is addressed in the PTS study in the typical manner (break spectrum sensitivity studies covering break locations and sizes)**

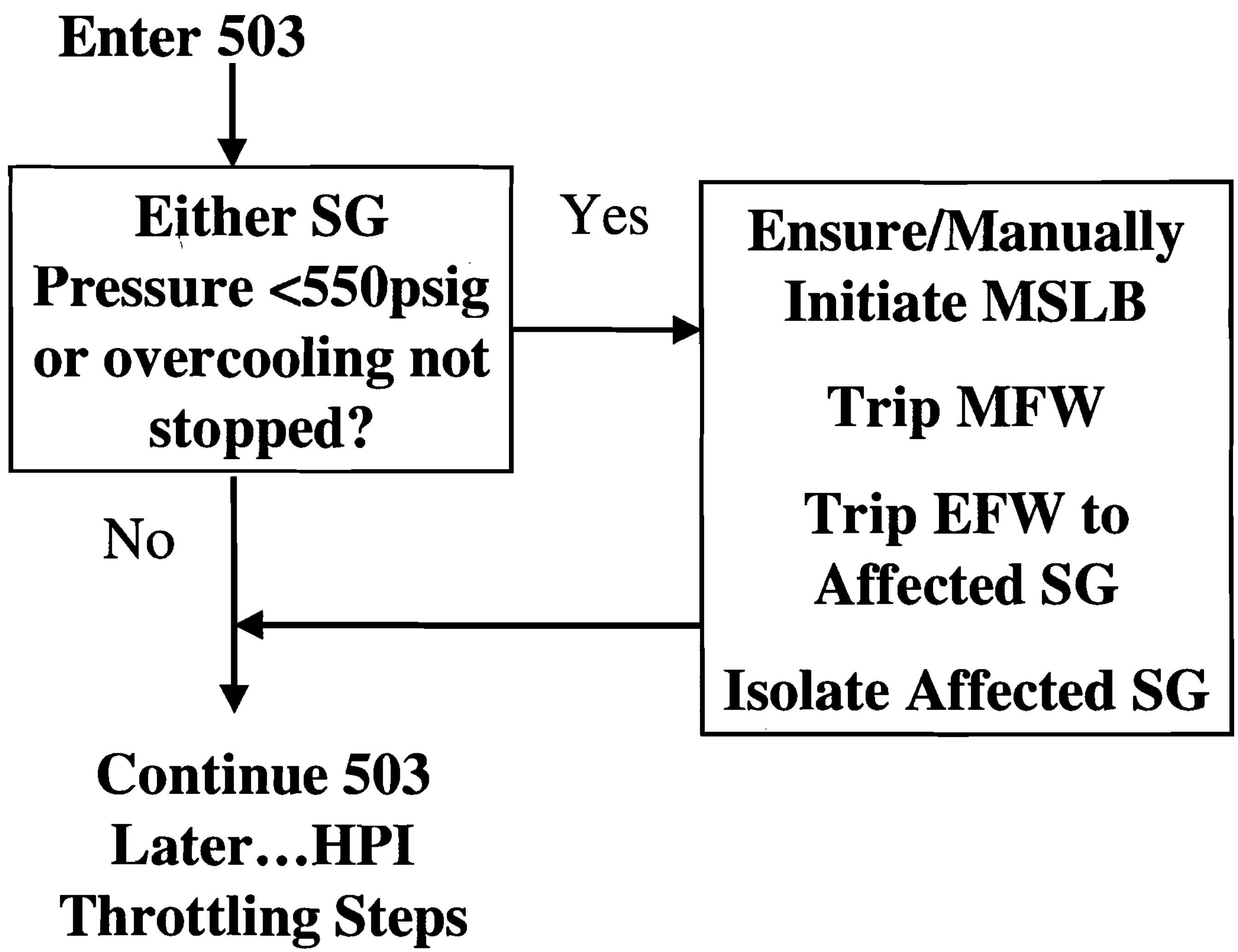
Beginning Steps in EP-1 (Trip)



Beginning Steps in 501 (Loss of Subcooling)



Beginning Steps in 503 (Excess Heat Transfer)



Crew Fails to Isolate Faulted SG

- Number, location, and readability of SG pressure RCS temperature indications make depressurization easily discernable
- Isolation is early in procedure guidance
- If pressure drop is slow/partial, operators taught to err on side of isolation
- Training strongly oriented toward following procedures with sensitivity to overcooling
- Use of "BAGS" could catch error in later times, if not done early
- Takes only one action to isolate (close EFW control valve) unless verification checks show that an auto action failure requires multiple actions to completely isolate
- Simulated events - isolation occurred in ~1-2 min
- NRC T-H runs show that shortest time period of interest ~10 min.
- Time of day, day of shift – not a strong influence on operator response
- 1st cut : Mean: 0.5 failure is likely

  0.1 failure is infrequent

  0.01 failure is unlikely

  0.001 failure is extremely unlikely

  Uncertainty: Generally assumed lognormal, with error factor of 5 or 10 using THERP guidance

Crew Fails to Isolate Faulted SG

- Secondary depressurization is only problem
  - Fail to isolate within 10 min; mean=0.001

- Additional anomaly in scenario **(e.g., concurrent LOCA)**
  - Fail to isolate within 10 min; mean=0.01
  - Fail to isolate within 20 min; mean=0.001

For very important operator events…

- A more detailed expert elicitation assessment was performed with experts providing histograms for the failure probability

Quantiles	1%,	10%,	25%,	50%,	75%,	90%,	99%
Failure Probability	0.001	0.003	0.01	0.05	0.2	0.5	0.8