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

October 11, 2000

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 TECHNICAL BASIS REEVALUATION PROJECT, SEPTEMBER 21, 2000 - ROCKVILLE, MARYLAND

The minutes of the subject meeting, issued on September 26, 2000, 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
J. Lyons
ACRS Fellows and Technical Staff
E. Barnard

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. William J. Shack, Chairman
Materials and Metallurgy Subcommittee

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS MATERIALS AND METALLURGY SUBCOMMITTEE MEETING CONCERNING THE PRESSURIZED THERMAL SHOCK TECHNICAL BASIS REEVALUATION PROJECT, SEPTEMBER 21, 2000 - ROCKVILLE, MARYLAND

I hereby certify that, to the best of my knowledge and belief, the minutes of the subject meeting issued on September 26, 2000, are an accurate record of the proceedings for the meeting.

Dr. William J. Shack, Chairman
Materials and Metallurgy Subcommittee

10/6/2000
Date

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

September 26, 2000

MEMORANDUM TO: Dr. William J. Shack, Chairman
Materials and Metallurgy Subcommittee

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 TECHNICAL BASIS REEVALUATION PROJECT, SEPTEMBER 21, 2000 - 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: G. Apostolakis
T. Kress
R. Seale

cc via E-Mail:
J. Larkins
J. Lyons

Issued: September 26, 2000
CERTIFIED: October 6, 2000

CERTIFIED

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF SUBCOMMITTEE MEETING ON
MATERIALS AND METALLURGY
PTS TECHNICAL BASIS REEVALUATION PROJECT
SEPTEMBER 21, 2000
ROCKVILLE, MARYLAND

The ACRS Subcommittee on Materials and Metallurgy met on September 21, 2000, to hold discussions with the NRC staff and its consultants concerning the Pressurized Thermal Shock (PTS) Technical Basis Reevaluation Project. The meeting included presentations concerning the activities associated with the initial results of thermal-hydraulic experiments and the development of a flaw distribution, fracture toughness distributions and model uncertainties, embrittlement correlations, and the FAVOR probabilistic fracture mechanics 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 8:35 a.m. and adjourned at 4:00 p.m.

ATTENDEES

ACRS

W. Shack, Chairman
G. Apostolakis, Member
T. Kress, Member
R. Seale, Member
N. Dudley, ACRS Staff

NRC REPRESENTATIVES

E. Hackett, RES
S. Malik, RES
D. Jackson, RES
L. Abramson, RES
M. Kirk, RES
N. Siu, RES
H. Woods, RES
D. Bessette, RES
D. Kalinousky, RES
T. Dickson, ORNL
M. Modarres, University of Maryland

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

INTRODUCTION

Dr. William Shack, Chairman of the Materials and Metallurgy Subcommittee, explained that the purpose of the meeting was to review activities related to the PTS Technical Basis Reevaluation Project. He noted that the staff had briefed the Subcommittee on the PTS Reevaluation Project activities on March 16 and April 27, 2000. He called upon Mr. Edwin Hackett, Assistant Chief of the Materials Engineering Branch, Office of Nuclear Regulatory Research (RES), to begin.

PTS TECHNICAL BASIS REEVALUATION PROJECT - Mr. Edwin Hackett, RES

Mr. Edwin Hackett, RES, provided background information concerning the PTS Reevaluation Project. He described the current status of the activities being performed by the three groups of staff and industry experts that are working in the areas of thermal-hydraulics, probabilistic risk assessment (PRA), and probabilistic fracture mechanics (PFM). Mr. Hackett noted that consideration of large, early release frequency (LERF) and containment integrity is a major departure from the current PTS framework. He stated that although initial indications pointed toward relaxation of current criterion, the final outcome is not yet clear.

The Subcommittee members and the staff discussed the evolution of probabilistic fracture mechanics codes, how uncertainties are characterized and propagated thorough the Fracture Analysis of Vessels: Oak Ridge (FAVOR) code, and the use of insights from Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the License Basis." They also discussed whether the integrity of the containment would be maintained if a PTS event resulted in a reactor vessel failure and core damage.

Dr. Kress noted that reactor vessel failures caused by PTS events result in a source term driven by air oxidation of zircaloy clad instead of steam oxidation. He questioned whether a PTS screening criterion based on 5×10^{-6} events per reactor year would adequately account for the source term resulting from air oxidation. The staff agreed to consider the effect of the air oxidation source term during its derivation of a revised PTS screening criterion.

PROBABILISTIC RISK ASSESSMENT GROUP - Mr. Hugh Woods, RES

Mr. Hugh Woods, RES, explained that the objectives of the PRA working group are to ensure that the overall process is coherent and risk-informed, to develop a screening criterion, and to update the Integrated Pressurized Thermal Shock (IPTS) PTS/PRA studies. He presented the overall framework and status of the PRA event sequence, thermal-hydraulic, and PFM analyses. Mr. Woods explained the event tree and a representative scenario for a PTS accident that would result in reactor vessel failure. He described the information that would be used in a plant-specific analysis. Mr. Woods concluded that the interactions between the three working groups are leading to improvements in the analyses.

The Subcommittee members and the staff discussed the need to develop PTS event trees since most PRAs have screened out these events and the use of A Technique for Human Event Analysis (ATHEANA) in developing the event trees. They also discussed the treatment of aleatory and epistemic uncertainties presented in the paper by Mr. F. Li, University of Maryland, titled "K_{Ic} / K_{Ia} Uncertainty Characterization." The staff agreed to brief the Subcommittee at a future meeting on the treatment of uncertainties in the overall PTS Reevaluation Project as described in the white paper by Mr. Nathan Siu.

THERMAL-HYDRAULIC GROUP - Mr. David Bessette, RES

Mr. David Bessette, RES, explained that the objective of the thermal-hydraulic working group is to update the thermal-hydraulic inputs developed at the time of the IPTS studies if these inputs

are no longer operative. He identified the four participating plants and the thermal-hydraulic issues being evaluated. He described the results of the initial RELAP Code analysis for Oconee and presented initial conclusions [see attached slide #1]. Mr. Bessette described how the results of the RELAP code will be validated and how the thermal-hydraulic uncertainties will be evaluated. He presented the status of experiments being conducted at the APEX facility and the schedule for completing similar thermal-hydraulic calculations for the Beaver Valley, Calvert Cliffs, and Palisades reactors.

The Subcommittee members and the staff discussed the treatment of uncertainties, the stability of the RELAP code during the two hours of transient time that was evaluated, use of the results of the REMAX code, and the differences between the boundary conditions used in the IPTS studies and the more recent RELAP code results.

GENERALIZED FLAW DISTRIBUTION REPORT - Ms. Deborah Jackson and Mr. Lee Abramson, RES

Ms. Deborah Jackson, RES, presented the process used to determine the flaw distribution in reactor pressure vessels. She explained how an expert panel was used to resolve specific technical issues for which there is significant scientific uncertainty. Ms. Jackson described the flaw distribution data obtained from detailed examination of the Pressure Vessel Research User Facility (PVRUF) vessel and the Shoreham vessel, and the comparison of this data to the Marshall flaw data distribution used for previous PTS studies. Ms. Jackson provided details of the expert judgement process and presented the panel's conclusions [see attached slides #2 and #3). She concluded that the staff identified fabrication process factors that are important in considering the introduction of flaws into the reactor pressure vessel.

Mr. Lee Abramson, RES, presented the generalized flaw distribution methodology, the PVRUF flaw distribution, and the Complementary Cumulative Distribution Function for flaws. He described how the flaw distribution would be used as an input to the FAVOR code. Mr. Abramson explained that the generalized flaw distribution combines the densities of flaws, crack depth distributions, and plant specific volumes and areas. He reiterated that the generalized flaw distributions being developed would be based on all available relevant data and that expert judgement would be used where necessary to account for factors not addressed in the data.

The Subcommittee and the staff discussed the following:

- weld processes used to construct boiling water reactor pressure vessels,
- the objective of the expert judgement process,
- clarification of which portions of the flaw distribution are derived from data and which are derived from expert judgement,
- validation of the flaw distribution,
- treatment of overlapping issues,
- how the opinion of each expect is weighted, and
- agreement of flaw distribution with results of the PRODIGAL model.

PROBABILISTIC FRACTURE MECHANICS GROUP - Mr. Shah Malik, RES

Mr. Shah Malik, RES, presented the major activities of the PFM working group. He described the objective, responsible staff, NRC contractors, and status of the following activities for developing:

- a fabrication flaw distribution,
- fracture toughness curves,
- embrittlement correlations,
- material chemistry distributions,
- beltline neutron fluence maps, and
- a revised FAVOR computer code.

Mr. Malik explained that several analysis models are being finalized, the PRA and thermal-hydraulic analyses of Oconee have started, and rigorous uncertainty models are being developed for key variables.

The Subcommittee members and the staff discussed uncertainty models, revising the margin term in Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," changes to parameters as a result of power uprates, and how preliminary results match the results from the IPTS studies.

EMBRITTLEMENT CORRELATIONS - Mr. Mark Kirk, RES

Mr. Mark Kirk, RES, explained how the revised embrittlement curves have application to the PTS rule and Regulatory Guide 1.99. He presented modeling considerations and provided examples of how specific variables were modeled. He presented the gating criteria used to determine which variables would be modeled in the FAVOR Code. Mr. Kirk showed the relative changes between the present Regulatory Guide 1.99 curves and the new trend curves. He outlined a framework for treating uncertainties and suggested potential applications of the revised embrittlement curves.

The Subcommittee members and the staff discussed the relationship between the embrittlement curves and the J_R curve correlation proposed by Prof. Eason, the effect of the revised embrittlement curves on reactor pressure vessel heatup and cooldown curves, and aging effects that are independent of neutron embrittlement.

FRACTURE TOUGHNESS DISTRIBUTIONS AND UNCERTAINTIES - Mr. Mark Kirk, RES

Mr. Mark Kirk, RES, explained that the goal of this activity is to characterize toughness using all available data in a way that is consistent with PRA insights. He presented how NRC contractors collected and assembled data for evaluating fracture toughness and how they developed interim fracture toughness curves. He described how uncertainties associated with the fracture toughness curves were established. Both the interim curves and the associated uncertainties will be inputs to the FAVOR code.

Mr. Kirk introduced root cause diagrams and explained how they could be used to display a complex process in a logical format and help provide a more rigorous treatment of uncertainties. He demonstrated how the root cause diagrams were used to develop more robust procedures for selecting fracture toughness and uncertainty values for a specific reactor pressure vessel. He described a model that predicts the shift in reactor pressure vessel reference temperatures (RT_{NDT}) due to irradiation embrittlement. Mr. Kirk stated that on-going activities included full implementation of uncertainty models in the FAVOR code, determining RT_{NDT} bias correction functions and modeling procedures, and assembling input data to run the models.

The Subcommittee members and the staff discussed how the uncertainties associated with RT_{NDT} should be accounted for in the characterization of the fracture toughness distribution.

STATUS OF FAVOR CODE DEVELOPMENT - Mr. Terry Dickson, Oak Ridge National Laboratory

Mr. Terry Dickson, ORNL, presented how the FAVOR code is applied in the PTS Technical Basis Reevaluation Project. He explained that the FAVOR code was intended to combine the best features of earlier PFM codes that had been used in the development of the original PTS rule. He described how elements of updated technology, such as enhanced neutron flux maps and new statistical models for static initiation and arrest fracture toughness, are being integrated into the FAVOR Code.

Mr. Dickson presented the FAVOR code structure and described the overall PRA methodology. He described the load generator, PFM, and post processor modules of the FAVOR code. He explained the analyses performed by each module and how the final probability of reactor pressure vessel failure is calculated. He concluded the presentation by providing the near term schedule for continued development of the FAVOR code.

The Subcommittee members and the staff discussed what elements of the updated technology results would provide the greatest relaxation to the PTS screening criterion and what the form would be of the thermal-hydraulic inputs to the FAVOR Code. They held an extended discussion on how fracture toughness curves were used to arrive at a probability of reactor vessel failure and the statistical treatment of the data associated with the curves.

SUBCOMMITTEE COMMENTS, CONCERNS, AND RECOMMENDATIONS

Dr. George Apostolakis provided the following recommendations concerning the draft report on the results of the expert judgement process:

- add sections to the report that clearly state the objectives and describes the methodology of the expert judgement process;
- show the calculations that were used to derive the flaw distribution from the experts' inputs;

- explain how the experts' inputs are equally weighted: for example plot the flaw distribution for each expert and take the mean value at different points; and

- provide more context for the increased flaw density.

Dr. William Shack provided the following recommendations concerning the draft report on the results of the expert judgement process:

- explicitly separate which portions of the flaw distribution were derived from data and which were derived from expert judgement,

- add the Pacific Northwest National Laboratory table of flaw data, and

- add the bead size for welds associated with the flaw distribution.

Dr. Thomas Kress suggested that the staff needed to consider an acceptance criterion applicable to PTS events before the PTS Reevaluation Project could be completed. Dr. Kress explained that PTS events result in air-oxidation of the fuel instead of steam-oxidation. As a consequence, the source term may be higher and hence the vessel failure frequency may need to be lowered to ensure the same level of risk for PTS events as the current LERF assessment guideline provides for events where a source term for steam-oxidation is appropriate. Dr. Kress noted that such considerations may markedly lower the acceptance criterion for RPV failure frequency below 1×10^{-6} per year.

STAFF AND INDUSTRY COMMITMENTS

The staff agreed to provide Dr. Apostolakis with a copy of NUREG/CR-5505, "RR-Rodical - A Model for Estimating the Probabilities of Defects in Reactor Pressure Vessel Welds." [received 9/22/2000]

The staff agreed to provide Dr. Apostolakis a copy of NUREG/CR-6471, "Characterization of Flaws in U.S. Reactor Pressure Vessels." [Volumes 1 and 3 received 9/22/2000] [Volume 2 received 9/26/2000]

The staff agreed to consider the effect of the air-oxidation source term during its derivation of the PTS screening criterion.

The staff agreed to brief the Subcommittee on the treatment of uncertainties in the FAVOR Code at a future meeting, including the information presented in the white paper by Mr. Siu and in the paper by Mr. F. Li, University of Maryland, titled "K_{Ic} / K_{Ia} Uncertainty Characterization."

SUBCOMMITTEE DECISIONS

The Subcommittee requested that the staff brief the Committee on the status of the development of the FAVOR Code and on the fracture toughness distributions and uncertainties at the October 5-7, 2000 ACRS meeting.

The Subcommittee requested that the staff brief it on the development of a criterion for deriving the PTS screening criterion.

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:

1. Dickson, T., Oak Ridge National Laboratory, and Malik, S., U.S. Nuclear Regulatory Commission, "An Updated Probabilistic Fracture Mechanics Methodology for Application to Pressurized Thermal Shock," issued in the Proceedings from the IAEA Specialists' Meeting, "Methodology and Supporting Research for the Pressurized Thermal Shock Evaluation," Rockville, MD, July 2000.
2. Jackson, D., and Abramson, L., U.S. Nuclear Regulatory Commission, "Draft Report on the Results of the Expert Judgement Process for the Generalized Flaw Size and Density Distribution for Domestic Reactor Pressure Vessels," received September 7, 2000.
3. Eason, D., and Wright, J., Modeling & Computing Services, "Draft Report on Updated Transition Temperature Shift Model," dated July 28, 2000.
4. Bowman, K., and Williams, P., Oak Ridge National Laboratory, "Technical Basis for Statistical Models of Extended K_{Ic} and K_{Ia} Fracture Toughness Databases for RPV Steels," dated February 2000.
5. Li, F., et. al., University of Maryland, " K_{Ic} / K_{Ia} Uncertainty Characterization," dated June 23, 2000.

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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, 2120 L Street, N.W., Washington, D.C. 20006, (202) 634-3274, or can be purchased from Ann Riley & Associates, LTD., 1025 Connecticut Ave., NW, Suite 1041, Washington, D.C. 20036, (202) 842-0034.

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MEETING OF THE MATERIALS AND METALLURGY SUBCOMMITTEE
PTS SCREENING CRITERION REEVALUATION PROJECT
SEPTEMBER 21, 2000
ROCKVILLE, MARYLAND

- AGENDA -

	TOPIC	PRESENTER	TIME
VI.	Opening Remarks	W. Shack, ACRS	8:30-8:35 a.m.
VII.	Status of Pressurized Thermal Shock (PTS) Technical Basis Reevaluation Project	E. Hackett, RES	8:35-10:00 a.m.
	A. Probabilistic Risk Assessment Group	R. Woods, RES	9:00 – 9:40
	B. Thermal Hydraulics Group	D. Bessette, RES	9:40 – 10:00
	C. Probabilistic Fracture Mechanics Group	S. Malik, RES	
	- BREAK -		10:00-10:15 a.m.
VIII.	Generalized Flaw Distributions Report	D. Jackson, RES L. Abramson, RES	10:15-~~12:00~~ 11:35 noon
	A. Expert Judgement Process		
	B. Results		
	- LUNCH -		12:00-1:00 p.m. 11:35 – 12:00 noon
IX.	Embrittlement Correlations	M. Kirk, RES	1:00-~~2:00~~ 1:45 p.m.
	- BREAK -		~~2:00-2:15~~ 2:10 – 2:25 p.m.
X.	Fracture Toughness Distributions and Uncertainties	M. Kirk, RES	~~2:15-3:00~~ 1:45 – 2:10 p.m.
XI.	Status of FAVOR PFM Code Development	T. Dickson, ORNL	~~3:00~~-4:00 2:25 - 3:45 p.m.
XII.	Discussion	W. Shack, ACRS	~~4:00-4:30~~ 3:45 – 4:00 p.m.
XIII.	Adjournment	W. Shack, ACRS	~~4:30~~ 4:00 p.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
SEPTEMBER 21, 2000
ROCKVILLE, MARYLAND

- AGENDA -

	TOPIC	PRESENTER	TIME
VI.	Opening Remarks	W. Shack, ACRS	8:30-8:35 a.m.
VII.	Status of Pressurized Thermal [illegible] Technical Basis Reevalua[illegible]	E. Hackett, RES	8:35-10:00 a.m.
	A. Probabilistic Risk A[illegible]	[illegible]oods, RES	
	B. Thermal Hydraulics [illegible]	[illegible] Bessette, RES	
	C. Probabilistic Fracture [illegible]	S. Malik, RES	
	- BREAK -		10:00-10:15 a.m.
VIII.	Generalized Flaw Distributions Report	D. Jackson, RES L. Abramson, RES	10:15-12:00 noon
	A. Expert Judgement Process		
	B. Results		
	- LUNCH -		12:00-1:00 p.m.
IX.	Embrittlement Correlations	M. Kirk, RES	1:00-2:00 p.m.
	- BREAK -		2:00-2:15 p.m.
X.	Fracture Toughness Distributions and Uncertainties	M. Kirk, RES	2:15-3:00 p.m.
XI.	Status of FAVOR PFM Code Development	T. Dickson, ORNL	3:00-4:00 p.m.
XII.	Discussion	W. Shack, ACRS	4:00-4:30 p.m.
XIII.	Adjournment	W. Shack, ACRS	4:30 p.m.

Duplicate

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 SUBCOMMITTEE ON MATERIALS AND METALLURGY PTS TECHNICAL BASIS REEVALUATION PROJECT

SEPTEMBER 21, 2000

NRC STAFF SIGN IN FOR ACRS MEETING PLEASE PRINT

NAME	BADGE #	AFFILIATION
Eric Thornsbury	B-8048	RES/DRAA/PRAB
SHAH MALIK	B-7396	RES/DET/MEB
NATHAN SIU	B-8130	RES/PRAA/PRAB
Doug Kalinowsky	B-7796	RES/DET/MEB
Debbie Jackson	B-6296	RES/DET/MEB
J PERSENSKY	B-8413	RES/DSARE/REHFB
Lee Abramson	A-6003	RES/DRAA/PRAB
Mark Kirk	B-8678	RES/DET/MEB.
Roy Woods	B-7313	RES/DRAA/PRAB
Ed Hackett	B-7487	RES/DET/MEB

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

MEETING OF THE SUBCOMMITTEE ON
MATERIALS AND METALLURGY
PTS TECHNICAL BASIS REEVALUATION PROJECT

SEPTEMBER 21, 2000

ATTENDEES - PLEASE SIGN BELOW

PLEASE PRINT

NAME	AFFILIATION
TERRY DICKSON	OAK RIDGE NAT'L LABORATORY
Ron Gamble	Sartrex Corp.
PAUL GUNTER	NIRS
Bob Hardies	CCNPR;

NFD

Commission (Open)—The ACNW will finalize preparations for the next public meeting with the Commission. The meeting is tentatively scheduled for October 17, 2000. Potential topics for discussion include: the development of a Yucca Mountain Review Plan and 10 CFR Part 63 (Disposal of High-Level Radioactive Waste in a proposed geologic repository at Yucca Mountain, Nevada); highlights of the Committee's recent European trip, Risk Informed Regulation in the Office of Nuclear Material Safety and Safeguards; and comments on the staff's Yucca Mountain Site Sufficiency Strategy.

K. *6:00 p.m.–7:00 p.m.: Continue Preparation of ACNW Reports* (Open)—The Committee will discuss the planned ACNW report on the YMRP as well as potential future reports.

L. *7:00 p.m.–7:30 p.m.: Miscellaneous* (Open)—The Committee will discuss miscellaneous matters related to the conduct of the Committee and organizational activities and complete discussion of matters and specific issues that were not completed during previous meetings, as time and availability of information permit.

Procedures for the conduct of and participation in ACNW meetings were published in the **Federal Register** on September 28, 1999 (64 FR 52352). In accordance with these procedures, oral or written statements may be presented by members of the public, 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 Committee, its consultants, and staff. Persons desiring to make oral statements should notify Howard J. Larson, ACNW, as far in advance as practicable so that appropriate arrangements can be made to schedule the necessary time during the meeting for such statements. Use of still, motion picture, and television cameras during this meeting will be limited to selected portions of the meeting as determined by the ACNW Chairman. Information regarding the time to be set aside for taking pictures may be obtained by contacting the ACNW office, prior to the meeting. In view of the possibility that the schedule for ACNW meetings may be adjusted by the Chairman as necessary to facilitate the conduct of the meeting, persons planning to attend should notify Mr. Larson as to their particular needs.

Further information regarding topics to be discussed, whether the meeting has been canceled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefore can be obtained by contacting Mr. Howard J. Larson, ACNW (Telephone 301/415–6805), between 8:00 A.M. and 5:00 P.M. EDT.

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

Dated: August 29, 2000.

Annette Vietti-Cook,

Acting Advisory Committee Management Officer.

[FR Doc. 00–22645 Filed 9–1–00; 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 September 21, 2000, 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:

Thursday, September 21, 2000—8:30 a.m. Until the Conclusion of Business

The Subcommittee will discuss the status of activities associated with the Pressurized Thermal Shock (PTS) Technical Basis Reevaluation Project. These activities include determining a flaw distribution, embrittlement correlation, and fracture toughness. The purpose of this meeting is to gather information, analyze relevant issues and facts, and to 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 7:30 a.m. and 4:15 p.m. (EDT). 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: August 28, 2000.

Howard J. Larson,

Acting Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 00–22646 Filed 9–1–00; 8:45 am]

BILLING CODE 7590–01–P

PEACE CORPS

Privacy Act of 1974; Systems of Records

AGENCY: Peace Corps.

ACTION: Notice of Modifications to Existing Systems of Records and the Establishment of New Systems of Records.

SUMMARY: Pursuant to the Privacy Act of 1974, the Peace Corps is issuing public notice of its proposal to modify nineteen systems of records and add six new systems of records. This notice provides information required under the Privacy Act on the revised and new systems of records.

DATES: Comments must be received by October 20, 2000. The proposed modified and new systems of records will be effective October 23, 2000 unless the Peace Corps receives comments that require a different determination.

ADDRESSES: Written comments should be addressed to Maggie Thielen, Office of the Chief Information Officer, Peace Corps, 1111 20th Street, NW., Washington, DC 20526. Comments may also be submitted electronically to the following electronic mail address: *mthielen@peacecorps.gov*. Written comments should refer to Privacy Act Systems of Records Notices and, if sent

United States
Nuclear Regulatory Commission

PRA for PTS Rule Revision

H. Woods, N. Siu, E. Thornsbury

Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Presentation to the ACRS Subcommittee on Reactor Safeguards, Materials and Metallurgy
September 21, 2000

PRA Objective

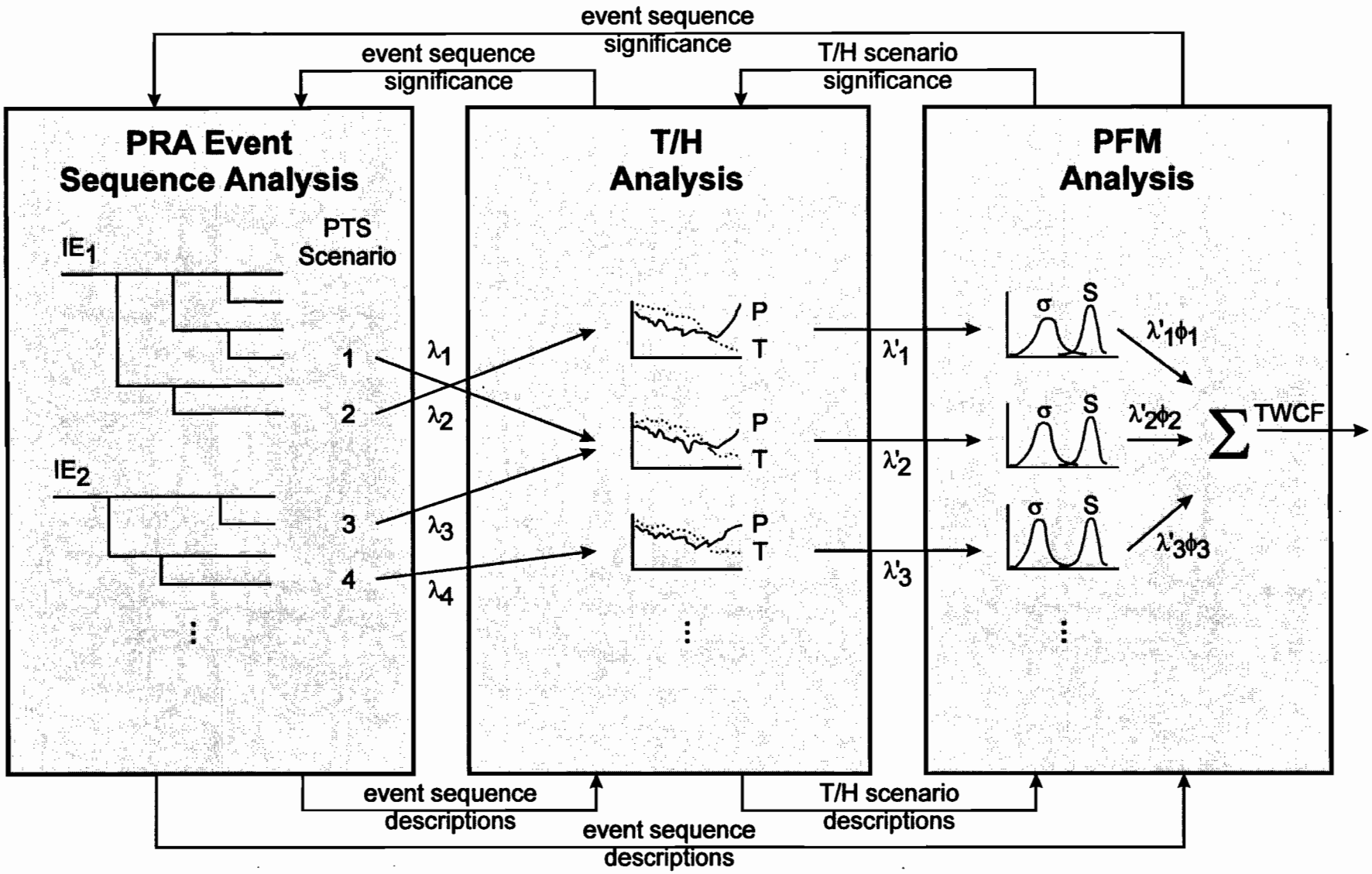
Support development of technical basis for revised rule

- **Ensure overall process is coherent, risk-informed**
  - **Appropriate integration of T/H, PFM, and PRA**
  - **Consistent treatment of uncertainties**

- **Support development of screening criteria**
  - **Derivation of embrittlement criteria from risk figures of merit**
  - **Criteria for risk figures of merit**

- **Update IPTS PTS/PRA studies**
  - **Reflect changes to IPTS plants**
  - **Reflect changes to PRA state of the art, knowledge base**
  - **Address other plants**

Overall Analysis Framework

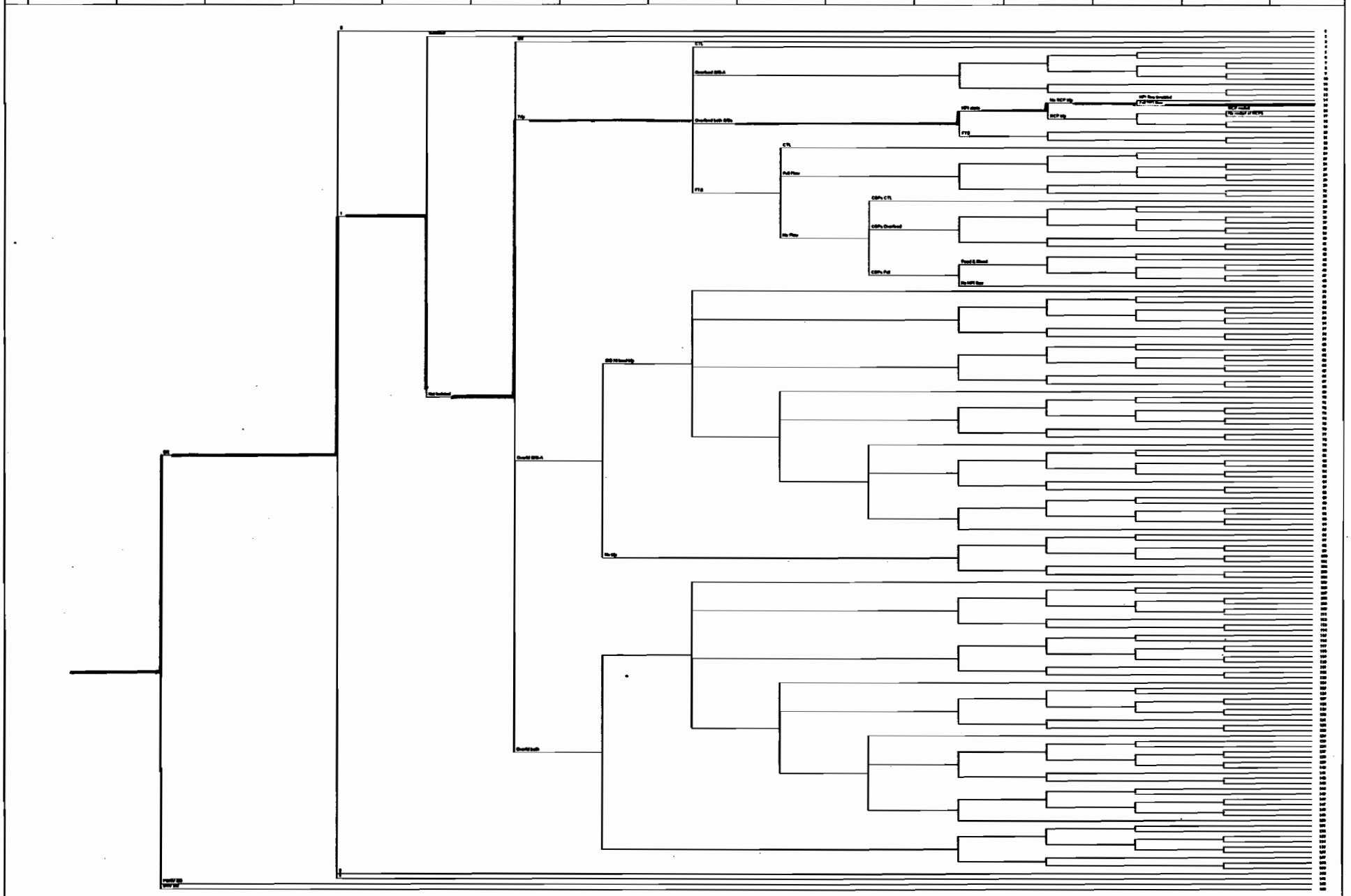


PTS/PRA Analysis - Status

- Oconee and Beaver Valley PTS Scenario Models
  - Event trees developed starting from IPTS studies
  - Generic initiating event frequencies and top event "split fractions" (developed from industry data) used to focus model development, support interactions with T/H and PFM
  - Some fault trees developed from Oconee PRA
  - Potential human failure events developed using ATHEANA
  - Quantification ongoing

- Palisades and Calvert Cliffs - Information Collection & Assessment Underway, No Substantial Review to Date

Rx Trip with TurbineTrip	PORV or SRV Stuck Open	Stuck Open PORV Isolated	TBV or MS-SRV Stuck Open	Stuck Open TBV Isolated	MFW Response to IE	MFW Fails to trip on S/G Hi-L	EFW Response to MFW trip	Fail to Recover from EFW-FTS	Condensate Booster Pumps Fail	HPWF&B Response	RCP Trip (i.e., loss of RCS subcooling)	Fail to Throttle HPI Flow	Fail to Restart RCPs	
RTTT	PORV_SRV_SO	PORV_ISO_F	TBV_SO	TBV_ISO_F	MFW_F	MFW_TRIP_F	EFW_F	EFW_REC_F	CBP_F	HPI_STARTS_F	RCP_TRIP	HPI_THROTTLED_F	RCP_RESTART_F	0



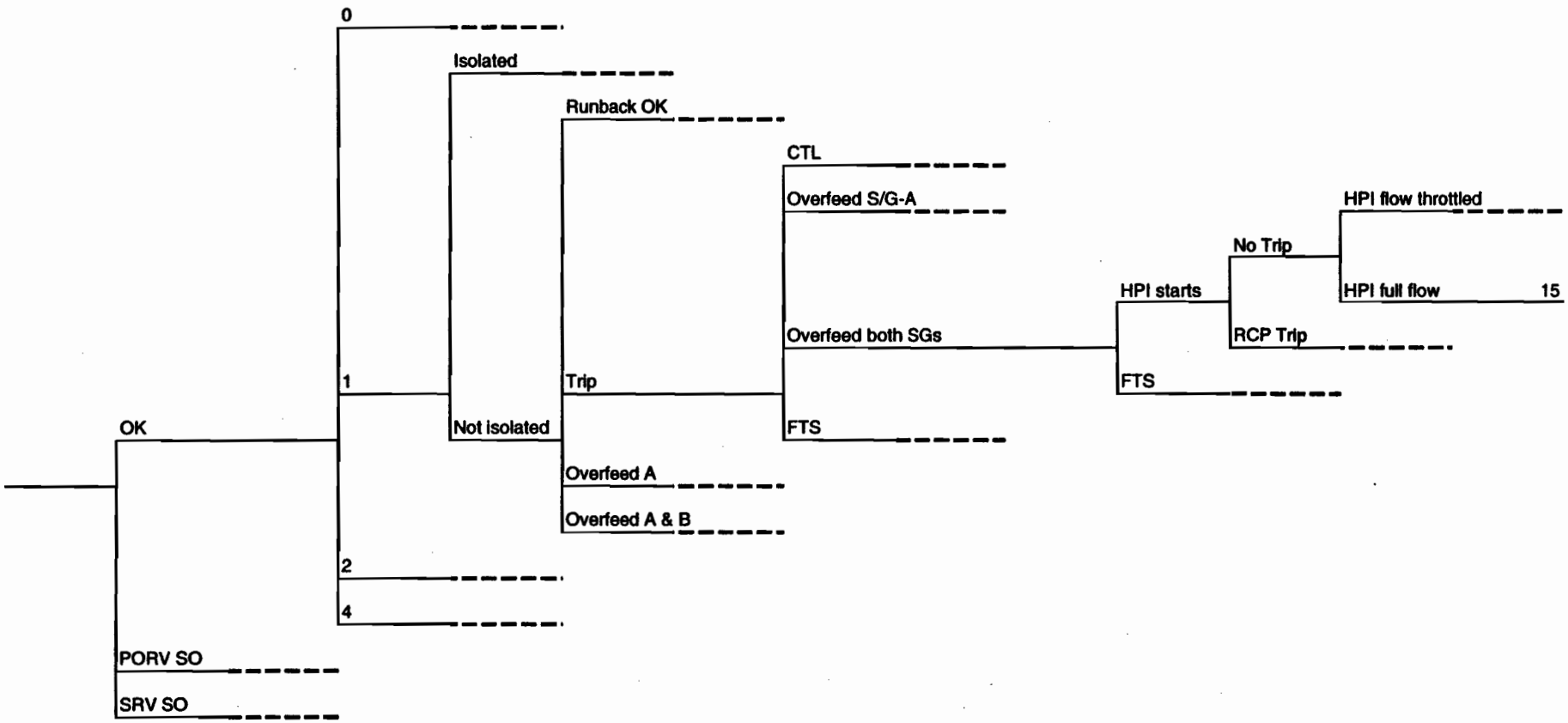
ONS-PTS-RT2 - Oconee PTS Rx-Trip with 1 TBV or MS-SRV Stuck Open

2000/09/12

A Representative Scenario

- **PTS Rx-Trip with 1 TBV or MS-SRV Stuck Open: Sequence #15**

Rx Trip with Turbine Trip	PORV or SRV Stuck Open	Stuck Open PORV Isolated	TBV or MS-SRV Stuck Open	Stuck Open TBV Isolated	MFW Response to IE (Runback)	MFW Fails to trip on S/G Hi-L	EFW Response to MFW trip	Fail to Recover from EFW-FTS	Condensate Booster Pumps Fail	HPI/F&B Response	RCP Trip (i.e., loss of RCS subcooling)	Fail to Throttle HPI Flow	Fail to Restart RCPs
RTTT	PORV_SRV_SO	PORV_ISO_F	TBV_SO	TBV_ISO_F	MFW_F	MFW_TRIP_F	EFW_F	EFW_REC_F	CBP_F	HPI_STARTS_F	RCP_TRIP	HPI_THROTTLED_F	RCP_RESTART_F



A Representative Scenario

- Initiating Event: Reactor Trip [Initiating Event Study]
- PORV or SRV Stuck Open: OK [Industry + Oconee experience]
  - Either not demanded or demanded and reclosed properly
- Stuck Open PORV Isolated: Pass
- TBV or MS-SRV Stuck Open: 1 [Oconee data]
  - TBV normally opens on Rx trip - fails to reclose
- Stuck Open TBV Isolated: Not Isolated [ATHEANA]
  - Identification and diagnosis of stuck open TBV
- MFW Response to IE (Runback): Tripped [Industry data]
  - Either as part of IE or other failure
- MFW Fails to trip on S/G Hi-L: Pass
- EFW Response to MFW trip: Overfeed both SGs
  - I&C Failure [Screening value]
  - Failure of operator control [ATHEANA]
- Fail to Recover from EFW-PTS: Pass

- **Condensate Booster Pumps Fail: Pass**
- **HPI/F&B Response: HPI Starts [HPI Reliability Study]**
  - **Cooldown causes primary pressure drop below 1600 psi**
- **RCP Trip: No Trip [Screening value]**
  - **Subcooling is not lost - RCPs do not need to be tripped**
- **Fail to Throttle HPI Flow: HPI full flow [ATHEANA]**
  - **Concurrent secondary side anomaly**

- **NOTES**
  - **Dependencies between HFEs still under investigation**
  - **Timing of HFEs is important**

Information Used in Analysis (BV Example)

- The BV IPE
- BV P&IDs, FSAR Sections, and Plant Operations Manual Sections for Equipment Related to PTS Events (e.g., CVCS, SI, MStm, FW)
- EOPs (and Their "Bases" Documentation) Related to PTS Events
- Selected AOPs (human actions for some PTS initiators)
- BV TH and PFM Calculations/Summaries Relevant to PTS
- Training Provided to the Operators Related to PTS Events
- BV's PRA Model and Supporting Documentation
- Specific Severe Accident Management Guides (SAMGs)
- Specific Technical Specifications (e.g., MSSVs, MSIVs)
- Specific WCAP Reports (e.g., HBR PTS, Generic PTS, BV PTS)
- BV Operating Experience (e.g., Prim. PORVs and SRVs, Sec. SRVs and SDVs, MSIVs)

What's New

- Use of Increased Operating Experience ⇒ More Realistic Event Frequencies and Failure Probabilities

- Use of Current Plant Design and Operating Procedures (some specifically to avoid overcooling events)

- Provision of Better Coupling Between Event Sequences and TH Analyses

- Use of Better PRA and HRA tools and methods allows:
  - more explicit modeling of sequences (less collapsing)
  - more integrated modeling of the sequence of events (facilitates performing sensitivity calculations)

What's New (cont'd)

- **Identification of Contextual Factors Affecting Operator Performance of Specific Actions**

- **Consideration of New Errors of Commission (e.g., "operator trips RCPs when not required to do so," and "operator isolates wrong SG")**

- **Allowance of More Credit for Recovery Actions (e.g., "operator throttles or terminates SI when SI operation is not desirable")**

Concluding Remarks

- PRA/TH/PFM interactions are leading to analysis improvements
  - Screening of event sequences
  - Better characterization of event sequences [especially important with Oconee - are reducing size of "Other" bin]
  - PRA/TH link [specification of TH runs]
  - Improved treatment of uncertainties

- Issues
  - External events
  - Inconsistencies and generalizations
  - Acceptance criteria

PRESSURIZED THERMAL SHOCK RE-EVALUATION PROJECT

Ed Hackett, Assistant Chief,
Materials Engineering Branch
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Presentation to the
Advisory Committee on Reactor Safeguards
Materials and Metallurgy SubCommittee

Rockville, MD
September 21, 2000

E-Mail: EMH1@NRC.GOV

USNRC/INDUSTRY PTS RE-EVALUATION PROJECT

Background/Impetus

- "Encouraging" materials integrity developments drive need for re-evaluation of PTS rule
  - Improved flaw distributions
  - Revised embrittlement correlations
  - Improved fracture mechanics methods
  - Dickson/Malik ASME PVP Paper (1999) shows potential for significant burden reduction

- Additional developments in thermal hydraulics and PRA
  - Improvements in T-H codes
  - Testing at APEX facility (flow stagnation)
  - Context for PRA (RG 1.174)
  - Explicit consideration of uncertainties

USNRC/INDUSTRY PTS RE-EVALUATION PROJECT

- Initiated in April, 1999 with original planned completion in December, 2001

- Fully participatory with input from stakeholders:
  - NRC (RES, NRR, Contractors)
  - Industry (MRP, EPRI, Vendors)
  - Public
  - ACRS reviews (2/99, 7/99, 3/00, 5/00 ...)

- Four full scale plants being analyzed:
  - Oconee-1
  - Calvert Cliffs-1
  - Palisades
  - Beaver Valley-1 (replacing H.B. Robinson)

USNRC/INDUSTRY PTS RE-EVALUATION PROJECT

Current Status

- Work progressing in major technical areas, schedule issues being addressed

- Finalization of materials inputs for revised PFM code (FAVOR) and initiation of FAVOR runs for four plants - Oct/Nov, 2000

- T-H Code validation through testing in APEX facility (4th Q, 2000)
  - Conditions for flow stagnation/mixing

- Progress in PRA aspects includes:
  - Process for explicit consideration of uncertainty in key inputs
  - Completion of Commission Paper on acceptance criteria (July, 2000)

- ACRS Meetings in March/May, 2000
  - Risk approach similar to RG 1.174
  - Likely re-set of risk criteria to 1E-6
  - Containment integrity could be considered to set criteria lower

USNRC/INDUSTRY PTS RE-EVALUATION PROJECT

Summary/Conclusions

- First major application of new risk-informed methodology to revise an adequate protection rule
- Good progress thus far. Schedule impacts currently being assessed
- Consideration of LERF and containment integrity is a major departure from current PTS framework
- Although initial indications pointed toward relaxation of current criteria, the final outcome is not yet clear
- Near-term scoping study for Oconee should complete in December 2000 - Initial indication of probable outcome.

David Bessette
Safety Margins and Systems Analysis Branch
Office Of Nuclear Regulatory Research

Status of Thermal Hydraulic Input to PTS

ACRS
Subcommittee on Materials and Metallurgy

Rockville, MD
September 21, 2000

Objective

- To ensure that, for the risk-significant classes of events, the thermal hydraulic inputs developed at the time of the Integrated Pressurized Thermal Shock Study (IPTS) conducted during 1982-1985 are still operative, or, are otherwise updated as needed

- Additionally, to provide an estimate of the uncertainty of these values

- In the IPTS study, three PWRs were selected for analysis, one from each vendor:

  1. Oconee Unit 1 (Babcock and Wilcox)

  2. Calvert Cliffs Unit 1 (Combustion Engineering); and

  3. H.B. Robinson (Westinghouse). In the current study we have switched to a similar 3-loop plant, Beaver Valley Unit 1

Thermal Hydraulic Issues

- Single and two phase loop natural circulation
- Criteria for interruption of loop flow, which causes flow stagnation
- Number of cold legs which must be flowing to assure mixing in the downcomer
- Local fluid mixing and onset of thermal stratification in the cold leg
- Plume mixing in the downcomer
- These are being studied in a experimental program in the APEX facility

RELAP Analyses of Oconee Transients

- Total of 25 transients were calculated for Oconee-1 thus far with RELAP5/MOD3 (version 322γ)
- Transients were run to 10,000s. Significant improvement from IPTS study
- Downcomer modeled with 6 channels instead of single channel

RELAP Analysis of Oconee Conclusions

- Useful interchange between PRA and thermal hydraulics. Good progress made on Oconee-1
- The break spectrum that was calculated for Oconee-1 covers the range of interest from where the primary system pressure stays near the steam generator secondary pressure to where the primary system pressure drops below accumulator pressure and further to low pressure injection
- The results are sensitive to the trip criteria for the reactor coolant pumps on loss of subcooling. Once subcooling is lost in a small break LOCA it will generally not be reestablished unless the break can be isolated.
- Stuck open valves on the secondary combined with a small break on the primary help to maintain subcooling and, therefore, the reactor coolant pumps are not tripped
- When the reactor coolant pumps are tripped, stagnation beginsand the downcomer cools in response to high pressure injection
- Cold leg breaks slightly less severe than hot leg breaks (similar to IPTS results)

TRAC Analysis of Oconee

- Coupling REMIX with TRAC
- 2 inch (to be run also with REMIX-RELAP)
- Main steam line break, Normal trip of main feedwater and turbine driven emergency feedwater pump upon low steam generator pressure, full uninterrupted flow of emergency feedwater to the broken generator, normal level control of emergency feedwater to the intact generator normal actuation of HPI, failure to trip HPI, normal trip of RCPs if subcooling is lost
- 1.414 inch with two stuck open safety valves
- Stuck Open PORV, Turbine trip with PORV fails to close, normal response of main and emergency feedwater to steam generator level control, normal actuation of HPI, normal trip of RCPs if subcooling is lost
- Results will be obtained by 10/00

Thermal Hydraulic Uncertainty Evaluation

- Performed by University of Maryland for Oconee-1
- Based on simplified treatment based on energy and mass conservation supplemented by RELAP analyses

APEX PTS Testing

- Objective is to provide experimental data on thermal hydraulic PTS issues and for code assessment
- Scaling evaluation performed to compare APEX to Palisades
- Facility was modified to add loop seals, HPI connections to cold legs, additional thermocouples in cold legs and downcomer
- Pre-test calculations performed using RELAP, REMIX
- First test conducted in August, 2000
- Remaining test to follow during the Fall, 2000

APEX PTS Test Matrix

1. Inject HPI into cold legs and measure plume effects in the vessel downcomer. The loop flow conditions will be stagnant

2. Depressurization/repressurization test simulating one of the most risk dominant sequence from the IPTS studies (stuck open PORV followed by closing of the block valve)

3. Flow interruption in one steam generator prior to the other

4. Small (2 inch) hot leg break where the operator is assumed to follow the emergency operating procedures

5. Small hot leg break similar to preceding test, but simulating Westinghouse HPI, accumulators, and LPI, instead of Combustion Engineering

6. Small cold leg break. Similar to hot leg break, but to assess the difference between the same break in the hot leg versus the cold leg. RELAP analysis of Yankee Rowe (4-loop plant) cold leg break in bottom of loop seal indicated possibility that broken loop could remain liquid solid. This needs to be checked by experiment and further analysis.

7. Main steam line break

Status of Calculations of Beaver Valley, Calvert Cliffs and Palisades

- No calculations performed thus far beyond exercising input models for H.B. Robinson and Calvert Cliffs using TRAC and RELAP
- We have begun conversion of the H.B. Robinson decks to model Beaver Valley
- Calculations with Beaver Valley deck are scheduled by 1/2001
- Calvert Cliffs and Palisades calculations scheduled for 5/2001

Conclusions

- Oconee RELAP results ready for transmittal to ORNL
- Using modified H.B. Robinson models, we expect to be able to provide thermal hydraulics analysis for Westinghouse 3-loop plant in a schedule consistent with requirements of PRA and fracture mechanics (early 2001)
- Calvert Cliffs and Palisades calculations to follow Beaver Valley (mid-2001)

RELAP Analyses of Oconee-1

- **Break Spectrum (hot leg breaks located in the pressurizer surge line)**

  1. 1 inch (diameter)
  2. 1.414 inch
  3. 1.5 inch
  4. 2 inch (to be run also with REMIX using boundary conditions from the RELAP calculation)
  5. 2.828 inch
  6. 4 inch

- **Break Plus Additional Failure**

  7. 1 inch with one stuck open safety valve
  8. 1 inch with two stuck open safety valves
  9. 1.414 inch with two stuck open safety valves

- **Break Location**

  10. 1.414 inch cold leg break (compared to hot leg break)
  11. 2 inch cold leg break (compared to hot leg break

- **Subcooling Criteria for Tripping Reactor Coolant Pumps**

  12. 1.414 inch with tripping the pumps at 5F subcooling instead of 0.5F

- **Subcooling Criteria for Throttling HPI**

  13. Trip HPI when 100F subcooling is reached for 1 inch LOCA with one stuck open safety valve
  14. Trip HPI when 100F subcooling is reached for 1 inch LOCA with two stuck open safety valves

- **Low Decay Heat**

  15. 1-inch LOCA with Low Decay Heat (0.3 x normal) [variation on earlier calculation]
  16. 1-inch LOCA with one stuck open safety valve with low decay heat (0.3 x normal) [variation on earlier calculation]

- **Stuck Open PORV**

  17. Turbine trip with PORV fails to close, normal response of main and emergency feedwater to steam generator level control, normal actuation of HPI, normal trip of RCPs if subcooling is lost

RELAP Analyses of Oconee-1 (cont'd)

- **Secondary Side Breaks**

  18. One stuck open steam dump valve with continued feed to the open steam generator

- **Steam Generator Overfeed**

  19. Emergency feedwater continued to 96% operating range
  20. Emergency feedwater continued until the separator is flooded.
  21. Turbine trip, normal reduction in feedwater, and with failure of level control, uninterrupted flow of feedwater
  22. Turbine trip, no reduction in main feedwater from full power flow rate, and with failure of level control, uninterrupted flow of feedwater HPI starts normally upon demand and is never terminated]

- **Main Steam Line Break**

  23. Normal trip of main feedwater and turbine driven emergency feedwater pump upon low steam generator pressure, full uninterrupted flow of emergency feedwater to the broken generator, normal level control of emergency feedwater to the intact generator normal actuation of HPI, failure to trip HPI, normal trip of RCPs if subcooling is lost

- **Other**

  24. 1-inch LOCA with no HPI, normal feedwater control, after 15 minutes steam dumps to go full open to cool down to LPI pressure

  25. Turbine trip, main feedwater trip, and reactor coolant pump trip. Emergency feedwater on to bring steam generator level to natural circulation setpoint. Terminate emergency feedwater and allow steam generators to dry out. When primary side reaches PORV setpoint and steam generators are dry, open steam dumps to allow secondary side to depressurize to allow condensate pumps to inject. Keep steam dumps open and keep feeding the steam generators. Allow HPI to actuate as normal. Turn off HPI if subcooling becomes > 100F.

Development of a Generalized Flaw Distribution for U.S. Reactor Pressure Vessels

Presented to: ACRS

September 21, 2000

By: Deborah A. Jackson, Lee Abramson

Office of Research

U.S. Nuclear Regulatory Commission

Background

- US NRC is re-evaluating the guidance and criteria in the Code of Federal Regulations as it relates to reactor vessel integrity, specifically PTS
- Fracture mechanics calculations are used to address the consequences of transients in reactor pressure vessels
- Reactor pressure vessel flaw distributions are an important input to fracture mechanics calculations

Background (con't)

- Fabrication processes of reactor pressure vessels is a significant factor for flaw introduction

Approach Used to Develop Generalized Flaw Distribution

- Expert judgment is needed to review, interpret and supplement available information on reactor vessel fabrication processes and reactor vessel flaw distributions
- Expert panel may resolve specific technical issues for which there is significant scientific uncertainty
- Structured expert elicitation process used to obtain responses from experts

Domestic Reactor Pressure Vessel Fabricators

- Combustion Engineering
- Babcock and Wilcox
- Chicago Bridge and Iron
- Rotterdam
- Societe Creusot
- New York Shipbuilding

RPV Material Selected for Generalized Flaw Distribution

	Midland	PVRUF	Shoreham	River Bend Unit 2	Hope Creek Unit 2
Manufacturer	B&W	CE	CE	CB&I	CB&I
LWR Type	PWR	PWR	BWR	BWR	BWR
Weld Metal	4 meters	20 meters	24 meters	15 meters	3 meters
Base Metal	0	0.9 m^3	6.8 m^3	1.0 m^3	0.6 m^3
Years of Construction	1968-1974	1976-1981	1968-1974	1974-1978	1971-1975
Beltline Material	Ring Forgings	Bent Plate	Bent Plates	Bent Plates	Bent Plates
Welds Inspected	Circ	Circ	Circ and Axial	Circ and Axial	Circ and Axial

Reactor Vessel Material

- Categorization of Flaws
  - Region of the vessel
    - Inner region
    - Mid region
    - Outer region
  - Volumetric vs. planar
  - Weld, clad, and base metal
  - Repair weld vs. non repair weld

Expert Judgment Process

- Define specific issues/scope to be addressed
- Determine level of complexity
- Identify an expert panel
- Send strawman of issues to panel
- Panel meets to agree on scope and issues
- Elicitation training

Expert Judgment Process

- Identify elicitation team
- Elicitation of experts
- NRC staff processes results
- Expert panel meets to review responses and rationales
- NRC staff aggregates responses and summarizes rationales

Expert Judgment Process

- Successfully completed 17 individual elicitation sessions
- Evolving process
- Preliminary review of the results revealed the need to re-elicit the experts
- Quantitative and qualitative characteristics

Expert Judgment Process
Areas of Expertise

- ASME Construction Code
- Failure Analysis
- Metallurgy
- NDE
- Reactor Vessel Fabrication
  - Base metal fabricators
  - Weld material suppliers
- Statistics
- Welding

Expert Judgment Process Definitions

- Flaw
  - An unintentional discontinuity that has the potential to compromise the reactor vessel integrity and is in the vessel after preservice inspection

Expert Judgment Process Definitions

- Small Flaw
  - Weld metal and cladding
    - Less than or equal to 1 bead thickness for a stated weld process
  - Base metal
    - Less than or equal to ¼" (6mm)
- Large Flaw
  - Weld metal and cladding
    - Greater than 1 bead thickness for a stated weld process
  - Base metal
    - Greater than ¼" (6mm)

Issues/Characteristics

- Product Form
- Weld Processes
- Flaw Mechanisms
- Field vs. Shop
- Repairs
- Weld Procedure
- Weld Materials
- Welder Skill
- Inspection Procedure
- Inspector Skill
- Base Metal Properties
- Surface Prep and Parameters
- Flaw Location

Characteristics Presented to Expert Panel - Page 1

Characteristic	Rank	LMH Value	Flaw Size/Fabricator
1. Product Form			
Forgings			
Plate			
Cladding			
Weldment			
2. Weld Process			
SMAW			
SAW			
ESW			
SMAW			
Cladding			
3. Flaw Mechanisms			
Base Metal			
Weldmetal			
4. Field vs. Shop Fab			
5. Repairs			
Weld metal			
Base metal			
Cladding			
6. Weld Procedure			
Axial Welds			
Girth Welds			
Repairs			
Cladding			

Characteristics Presented to Expert Panel - Page 2

Characteristic	Rank	LMH Value	Flaw Size/Fabricator
7. Weld Materials			
Axial Welds			
Girth Welds			
Repairs			
Cladding			
8. Welder Skill			
Axial Welds			
Girth Welds			
Repairs			
Cladding			
9. Inspection Procedure			
Forgings			
Plate			
Weld metal			
Repairs			
Cladding			
10. Inspector Skill			
Forgings			
Plate			
Weld metal			
Cladding			
Repairs			
11. Base Metal Properties			
12. Surface Prep and Parameters			
13. Flaw Location			

Relative Assessments

- Rank characteristics in order from highest to lowest in terms of contributing to or having a flaw after preservice inspection
- Comparison with highest ranked for relative likelihood of a flaw
- Quantitative and Qualitative responses

Expert Judgment Process

- Quantitative characteristics
  - The experts were able to provide numerical comparisons and
  - Necessary records for use in generalized flaw distribution are available
- Qualitative characteristics
  - The experts were unable to meaningfully quantify or
  - Necessary records are unavailable

Some Conclusions of the Expert Judgment Process

- A generalized flaw distribution for domestic RPVs can be developed but with a wide range of uncertainty
- The flaw density of base metal is substantially less than that for weld metal
- Discontinuities in the cladding that do not affect the clad base metal interface may not be of concern for RPV integrity

Some Conclusions of the Expert Judgment Process (con't)

- Large flaws are usually caused by loss of control of the welding process and would be detected by NDE
- Variations in weld quality are greatly affected by welder skill and inspector skill
- Weld processes are an important factor in the introduction of flaws

Concluding Remarks

- Expert judgment process is complex
- Significant amount of information was obtained from the experts during the process
- As a result of the process, NRC staff was able to identify important factors of the fabrication process which are important in considering the introduction of flaws into the RPV

Generalized Flaw Distribution Methodology

- Flaw densities
- Volumes or areas
- Distributions of crack depths

Generalized Flaw Distribution Methodology (contd.)

$N_S(x)$ = number of small flaws > x, for $x \leq b$

$N_L(x)$ = number of large flaws > x, for x > b

x = crack depth

b = bead thickness

PF = Product Form (Weld Metal, Cladding, Plate, Ring Forgings)

WP = Weld Process (SMAW, SAW, ESW; Strip, Single & Multi Wire)

R = Repair State (Unrepaired, Repaired)

Generalized Flaw Distribution Methodology (contd.)

ρ_S(PF, WP, R) = density of small flaws per unit volume or area

ρ_L(PF, WP, R) = density of large flaws per unit volume or area

V(PF, WP, R) = volume or area of material.

$$N_s = \Sigma\ \rho_S(PF, WP, R) \bullet V(PF, WP, R)$$

$$N_L = \Sigma\ \rho_L(PF, WP, R) \bullet V(PF, WP, R),$$

Generalized Flaw Distribution Methodology (contd.)

$G_S(x)$ = ccdf for small flaws = Prob {crack depth > x}, where $x \leq b$

$G_L(x)$ = ccdf for large flaws = Prob {crack depth > x}, where $x > b$.

Each GFD is the product of the number of flaws and the corresponding crack depth distribution.

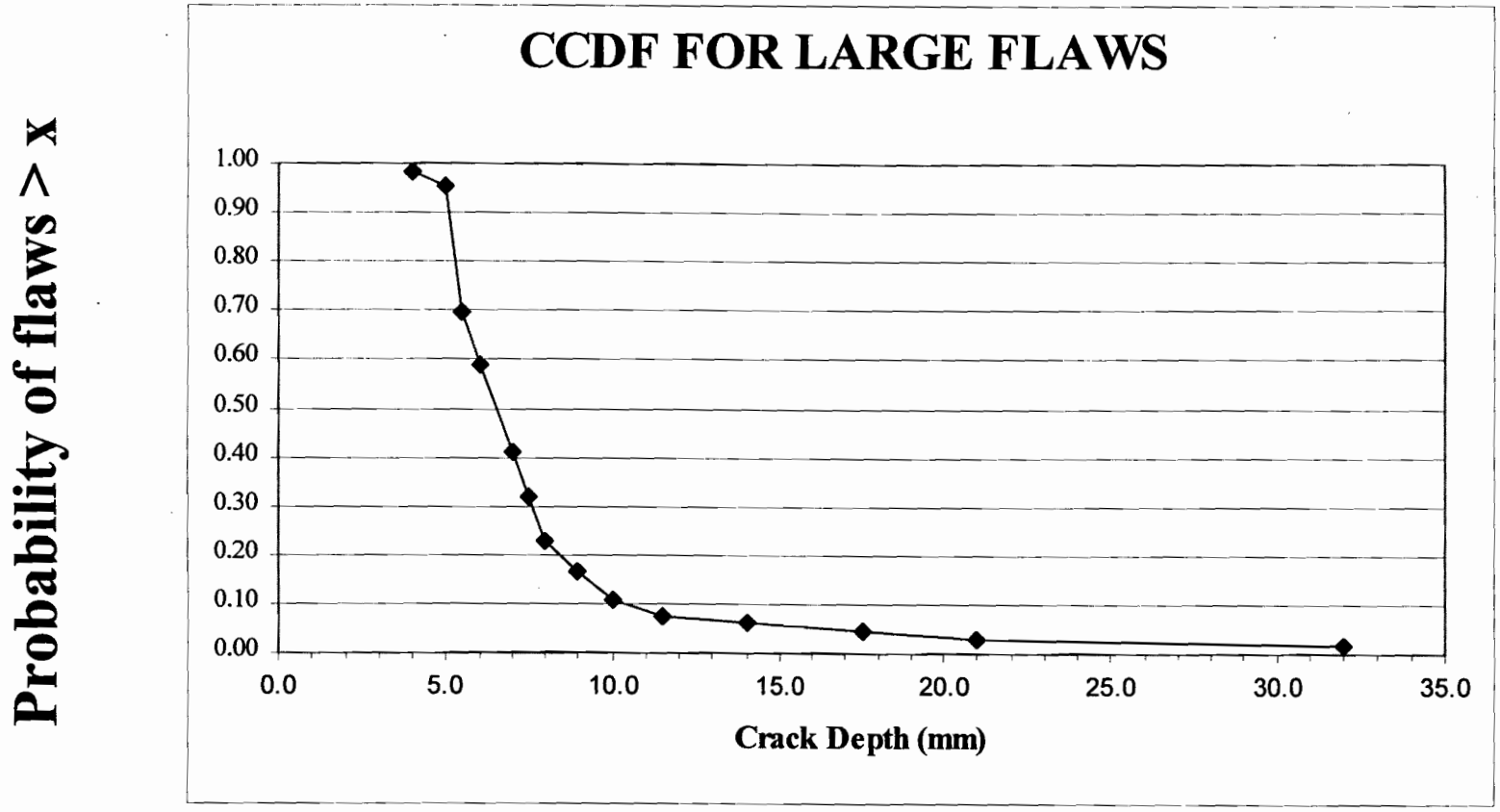
$$N_s(x) = N_S \bullet G_S(x) = [\, \Sigma\, \rho_S(PF, WP, R) \bullet V(PF, WP, R)\,] \bullet G_S(x)$$

$$N_L(x) = N_L \bullet G_L(x) = [\, \Sigma\, \rho_L(PF, WP, R) \bullet V(PF, WP, R)\,] \bullet G_L(x).$$

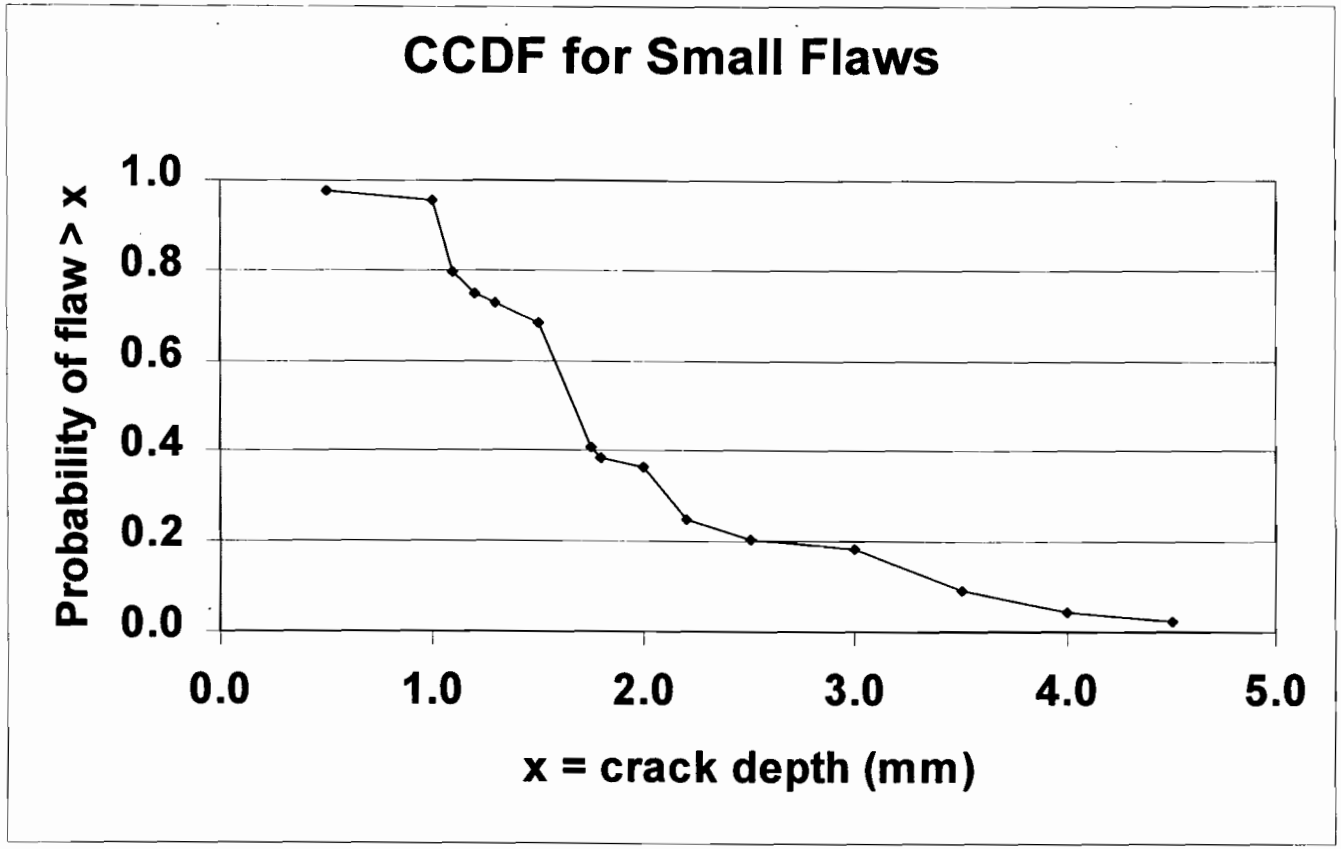
PVRUF Flaw Distribution

Product Form/ Weld Process/ Repair State	PVRUF Volume (m^3)	Density of Flaws		No. of Flaws	
		Small/m^3	Large/m^3	Small	Large
Weld Metal/SMAW/NR	0.025	14,000	280	350	7
Weld Metal/SAW/NR	0.45	7,900	22	3,600	10
Plate/NR	16.7	790	0.55	13,200	9
Weld Metal/SMAW/R	0.007	13,000	1,000	120	70
Plate/SMAW/R	--	7,900	5.5	--	--
Total Weld Metal & Plate				17,270	96
	PVRUF Area(m^2)	Small/m^2	Large/m^2	Small	Large
Cladding/SMAW/NR	2.9	57	1.7	160	5
Cladding/Strip/NR	72	69	2.1	5,000	150
Cladding/MultiWire/NR	0	N/A	N/A	N/A	N/A-
Cladding/SMAW/R	unk	106	5.3	--	--
Total Cladding				5,160	155
TOTAL				22,430	251

CCDF for Large Flaws



CCDF for Small Flaws



Input to FAVOR

	Large Flaws	Small Flaws
Weld Material	$\{x_i\}_{WL}$	$\{x_i\}_{WS}$
Plate Material	$\{x_i\}_{PL}$	$\{x_i\}_{PS}$

x_i = crack depth

Concluding Remarks

- Generalized flaw distribution combines:
  - Densities (generic)
  - Crack depth distributions (generic)
  - Volumes/Areas (plant specific)
- Generic inputs based on all available data and expert judgment

Backup Slides

List of Experts

- Frank Ammirato — EPRI NDE Center
- Francis Berry — F.C. Berry & Associates, Retired Chicago Bridge & Iron
- Spencer Bush — Review & Synthesis Associates, Retired PNNL
- Domenic Canonico — Consultant, President ASME BPV
- Vic Chapman — O.J.V.C. Consultancies, Retired Rolls Royce
- Robert Denale — Naval Surface Warfare Center

List of Experts

• Jack Lareau	Westinghouse (formerly ABB-CE)
• Carl Lundin	Materials Applications Inc., Professor University of Tennessee
• Harry Lunt	Consulting Metallurgist, Retired Burns & Roe
• Edward Nisbett	Consulting Metallurgist, Retired National Forge
• Robert Pond	Consulting Metallurgist, Professor Johns Hopkins University

List of Experts

• Stan Rosinski	EPRI NDE Center
• James Sims	Consulting Welding Engineer, Retired Chicago Bridge & Iron
• Kenneth Stuckey	Framatome Technologies
• Helmut Thielsch	Thielsch Engineering, Inc
• R. David Thomas	R. D. Thomas & Co., Retired Arcos Corporation
• Robert Warke	Southwest Research Institute

Quantitative Characteristics

- Product form
  - base metal
    - ring forgings
    - Plate
  - cladding
  - weldmetal
- Weld processes
  - SAW
  - ESW
  - GMAW
  - SMAW
- Flaw mechanisms
  - base metal
  - weld metal
- Repairs
  - basemetal
  - weld metal
  - cladding
- Flaw location
- Flaw size

Qualitative Characteristics

- Field vs. Shop fabrication
- Weld procedure
  - plate to shell
  - shell to shell
  - repairs
  - cladding
- Weld materials
  - plate to shell
  - shell to shell
  - repairs
  - cladding
- Welder skill
  - plate to shell
  - shell to shell
  - repairs
  - cladding

Qualitative Characteristics (con't)

- Inspection procedure
  - plate to shell
  - shell to shell
  - repairs
  - cladding
- Inspector skill
  - plate to shell
  - shell to shell
  - repairs
  - cladding
- Base metal properties
- Surface preparation and parameters

Re-elicitation of Experts

- Flaw size
- Cladding vs. weldmetal
- Basemetal vs. weldmetal
- Repairs vs. non-repairs (small and large flaws)
- Underclad cracking
- Weld processes (weldmetal and cladding)

Large Flaws in PVRUF and Shoreham

Flaw Size (mm)										BT (mm)	PVRUF
4	5	5-6	6	7	7-8	8	9	10	>10		
2	2									3.5	
					4					6.5	
		2**			2					6	
		5							11.5, 17.5	3.5	
2	2	7			6				11.5, 17.5	#=19	Total PVRUF
Flaw Size (mm)										**BT (mm)**	**Shoreham (NV)**
4	5	5-6	6	7	7-8	8	9	10	>10		
			4							5	
	10*									5	
				1					21, 32	5	
								1		6	
				1					14	5	
	1*		1	1		2	3			5	
	4*		7	3		2	1	1		5	
	15		12	6		4	4	2	14, 21, 32	#=46	Total Shoreham
2	17	7	12	6	6	4	4	2	11.5, 14, 17.5, 21,32	#=65	Total

Quantitative Assessment

X = Quantity to be assessed

X_L = Low value

X_H = High value

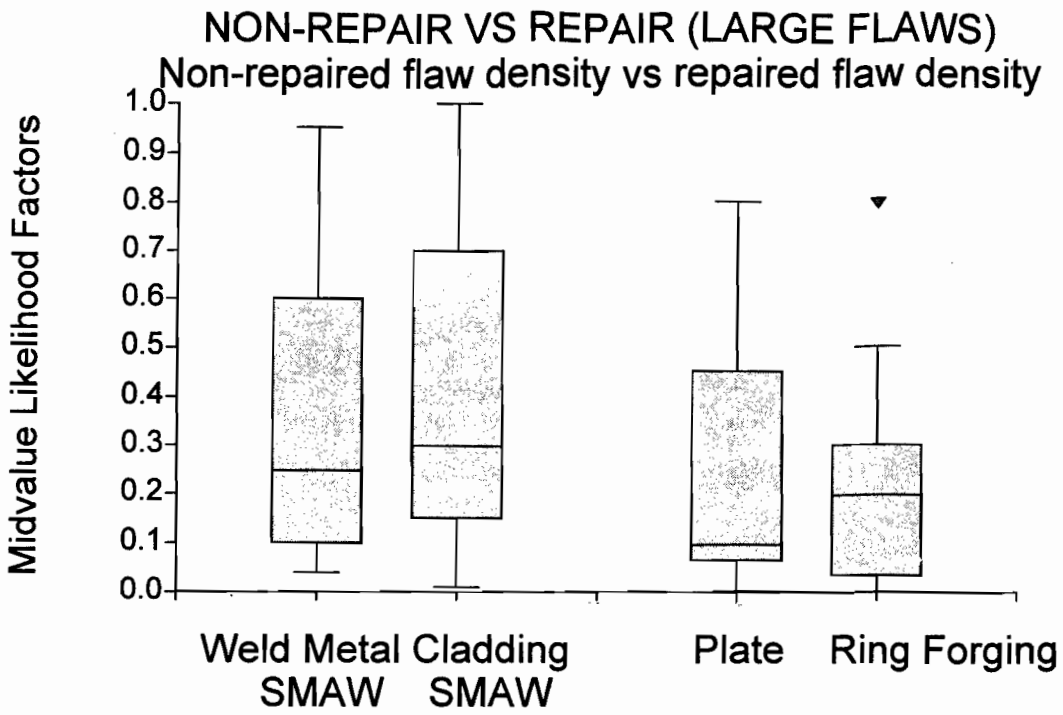
X_M = Midvalue

Chance { $X < X_L$ } ÿ 5%

Chance { $X > X_H$ } ÿ 5%

Chance { $X < X_M$ } ÿ Chance { $X > X_M$ } ÿ 50%

(X_L , X_H) is an approximate 90% coverage interval for X



	Weldmetal SMAW	Cladding SMAW	Base Metal Plate	Base Metal Forging
MIN	.04	.01	.002	.0004
LQ	.1	.15	.06	.03
MED	.25	.3	.1	.2
UQ	.6	.7	.5	.3
MAX	.95	1.0	.8	.8

Status of Pressurized Thermal Shock Screening Criterion Re-evaluation Project

Shah Malik
Materials Engineering Branch
Office of Nuclear Regulatory Research,
U.S. Nuclear Regulatory Commission

Presentation to:
Advisory Committee Reactor Safeguards,
Materials and Metallurgy Subcommittee
September 21, 2000

PTS Re-evaluation Project
Probabilistic Fracture Mechanics
Presentation Outline

- Major PFM technical areas
- Progress in major PFM technical areas
- Concluding remarks

Probabilistic Fracture Mechanics
Major Technical Areas

1. Fabrication Flaw Distributions in RPV beltline
2. Rigorous statistical representation of fracture toughness, K1c and K1a, data as functions of ($T-RT_{NDT}$), and uncertainties
3. Improved irradiation embrittlement correlations to predict shift in RT_{NDT}, and uncertainty
4. Improved statistical distributions for material chemistry (Copper, Nickel, Phosphorus)
5. Beltline neutron fluence maps, and uncertainty
6. PFM computer code, FAVOR, revision

Fabrication Flaw Distributions

- Objective: Determine generalized flaw size, density (# of flaws/unit volume), and location distributions of fabrication flaws in welds, plates and forgings in RPV beltline region, using:
  - NDE/DE techniques and experts' judgement process
- <u>RES Staff</u>: Deborah Jackson, Lee Abramson
- <u>NRC Contractor</u>: PNNL

Fabrication Flaw Distributions (Contd.)

- NDE/DE of welds in several RPVs completed, some still continuing
- NDE/DE of limited plate material continuing
- Experts' elicitation completed in April '00
  - Public workshop held on June 27-28
- Generalized flaw distributions developed during July-Sept. '00
- Distributions for use in FAVOR code being developed during Sep.-Oct. '00

Fracture Toughness (K1c, K1a) Curves

- Objective: Revise fracture initiation and arrest toughness distributions based on expanded LEFM-valid data (ASTM E-399 standard) and rigorous statistical methods
  - Distributions used previously were based on:
    - limited 1970's/80's toughness data
    - Adhoc distributions developed from lower-bound ASME toughness curves
- RES Staff: Mark Kirk, Shah Malik, Nathan Siu
- NRC Contractor: ORNL, Univ. of Maryland
- EPRI-funded work: PEAI -- develop uncertainties

Fracture Toughness (K1c, K1a) --Contd.

- ORNL searched and collected additional test data (database increased considerably)
  - Weibull distributions developed based solely on test data
- Uncertainty in normalizing parameter, RT_{NDT} being investigated
- Univ. of Maryland and EPRI assisting in:
  - separating epistemic and aleatory uncertainty
  - Material variability and model uncertainty
- Completion in Nov. '00

Irradiation Embrittlement Correlations

- Objective: Develop revised model to predict shift in RT_{NDT} (ΔRT_{NDT}) due to irradiation embrittlement using up-to-date data and statistical methods
  - Consistency with RG 1.99 Rev. 3 draft development
- Current correlations in RG 1.99 Rev. 2 are based on early-1980's data
- RES Staff: Mark Kirk, Carolyn Fairbanks
- NRC Contractors: Modeling & Computing Services, Univ. of California-Santa Barbara, ORNL, Univ. of Maryland
- EPRI-funded work: PEAI -- develop uncertainties

Irradiation Embrittlement Correlations (Contd.)

- **Progress:**
  - **Mean correlation developed in July '00**
  - **Uncertainty characterization started**
    - **Completion in Nov. '00**

Material Chemistry Distributions

- <u>Objective</u>: Use NRC and industry data to determine --
  - Heat-specific distributions for Copper, Nickel and Phosphorous
  - Investigate local variability
  - 4 PTS plants have
    - 15 weld heats, 2 with Nickel addition
    - 16 plate heats
- <u>RES Staff</u>: Doug Kalinousky, Tanny Santos, Lee Abramson

Material Chemistry Distributions (Contd.)

- Weld heat-specific distributions
  - **Normal**
- Weld local variability
  - **Logistic for Copper and Nickel**
  - **Normal for Phosphorus**
- Plates
  - **Limited data for the heats in PTS plants**
    - **Chemistry values taken as Heat Estimate**
- Plate local variability – limited data from CE
  - **Normal distributions**

Beltline Neutron Fluence

- <u>Objectives</u>:
  - Determine up-to-date EOL fluence maps for the plants using currently available cycle-by-cycle fuel loading histories and plant data
  - estimate uncertainty in fluence calculations
- Dosimetry Draft Guide-1053 (1999), and draft NUREG/CR-6115 methodology used
- <u>RES Staff</u>: William R. Jones
- <u>NRC Contractor</u>: Brookhaven National Lab.

Beltline Neutron Fluence (Contd.)

- **Plants analyzed:**
  - **Oconee-1, Palisades, Calvert Cliffs-1**
  - **Beaver Valley-1 (plant-data being received)**
- **Very refined axial, circumferential, radial grids used:**
  - **218 axial, 60 x 8 circum. for Oconee-1**
  - **205 axial, 97x8 circum. for Palisades**
  - **142 axial, 79 x 8 circum. for Calvert Cliffs-1**
- **Uncertainty in calculated fluence estimated to be:**
  - **1σ fluence $\cong$ 15% of the mean**
- **Need for evaluating nonlinear interactions among fluence parameters is being considered**
  - Vessel diameter, core inlet temp., nuclear cross-section, core neutron source

FAVOR Computer Code Revision

- FAVOR (Fracture Analysis of Vessels - Oak Ridge)
- Implement refined PFM methodology and up-to-date materials data
- Make it consistent with current PRA and thermal-hydraulics output data and methods
- Participants include:
  - RES Staff: Shah Malik, Nathan Siu, Lee Abramson
  - NRC Contractors: ORNL (Terry Dickson),
    Univ. of Maryland
    (PRA: Modarres, Mosleh)

Concluding Remarks

- Several analysis models being finalized
  - Scoping studies continuing
  - Implementation of finalized models underway
- Application to first plant (Oconee-1) has started in PRA and TH areas
  - PFM analysis expected to start in March '01
- Additional time and resources being used to develop rigorous uncertainty models for key variables.

Revision of ΔT_{30} Embrittlement Trend Curves

→ Status Report to ACRS ←

Mark Kirk
RES/DET/MEB

Rockville, Maryland
September 21, 2000

VG 1

Application of this Information

- PTS rule (10CFR50.61) revision
  - PTS screening criteria
- Reg. Guide 1.99 Revision 3
  - PTS assessment methodology
  - Heat up / cool down calculations

VG 2

Parts of Reg. Guide 1.99 Rev. 3

Part of the Reg. Guide	RG 1.99 R3 Status / Proposed Resolution	PTS Re-Eval. Need?
Transition Shift, Mean Trend Curve	Work Done. Tech Basis Doc Being Written.	Yes
Transition Shift, Uncertainty	Work underway to develop a PRA-consistent uncertainty framework	
Thru Wall Attenuation	Consider new information	
Plant Specific Data: Surveillance Credibility	No technical basis for credit. Use heat average chemistry.	
Plant Specific Data: Ratio Procedure	Not needed if all shifts are based on heat average chemistry.	
Upper Shelf Energy: Mean Trend	Work Done	No
Upper Shelf Energy: Uncertainty	Work Done	

VG 3

Reg. Guide 1.99 Revision 2

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

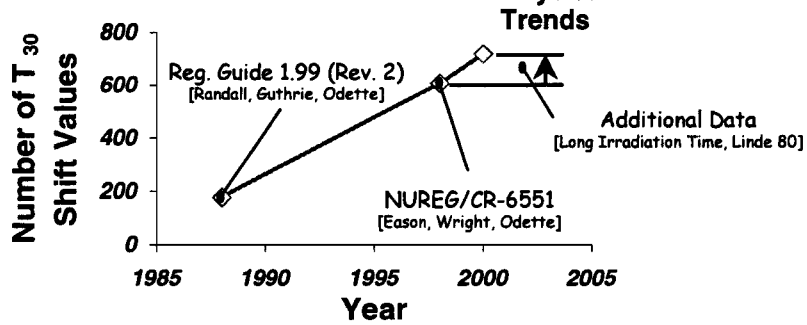
- Chemistry factor depends on
  - Cu
  - Ni
  - Product form

VG 4

Reg. Guide 1.99 Rev. 3 Activity

Objective: Develop a model to predict the shift in T_{30} due to irradiation embrittlement

- More data
  - Better coverage of primary variables
  - Longer time exposures
  - Higher fluences
- Statistical methods
- Physical motivation for Trends



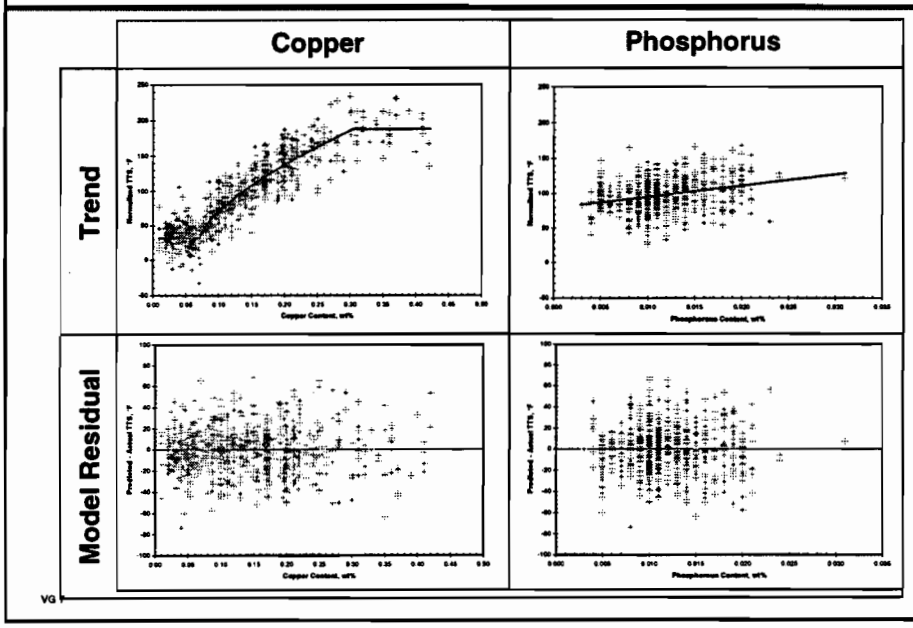
VG 5

Modeling Considerations

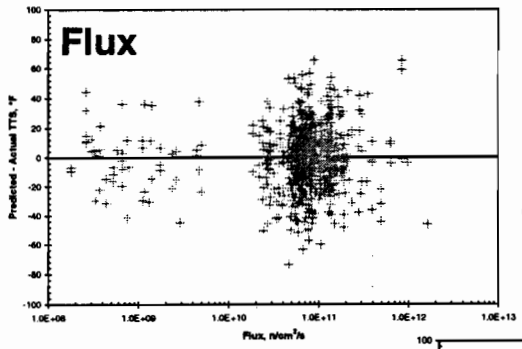
- Nonlinear fitting
  - Some coefficients based on all of the data set, some based on subsets
  - Some functional forms physically motivated
- A "good" fit has
  - Minimum standard error
  - Residuals
    - ✓ Average = 0
    - ✓ Balanced +/-
    - ✓ No trend with
      - Modelled variables
      - Un-modelled variables
- Statistical significance
  - Physical understanding suggests appropriateness of a one or two tailed test
- Model stability checked
  - Relative to initial estimates
  - Relative to data used to calibrate

VG 6

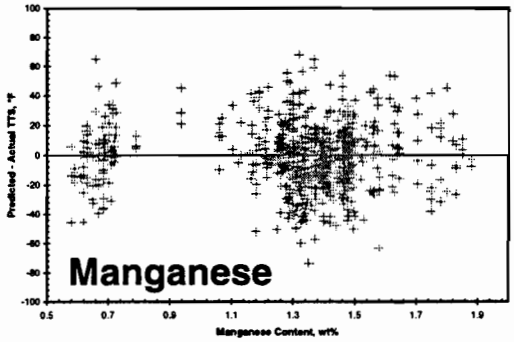
Variables Modeled, Examples



VG



Variables Not Modeled, Examples



VG 8

Gating Criteria for Effect Inclusion

- Effects considered (recently)
  - Copper saturation
  - Phosphorus
  - Flux – time
  - Long time
  - Vessel Fabricator

Gating Criteria

Physical Rationale	Statistical Basis		
	Strong (>95%)	<95% & >90%	Weak (<90%)
Well Accepted	Include	Maybe	Maybe
Plausible	Include	Maybe	Exclude
Not Established	Probably	Exclude	Exclude

VG 9

Gating Criteria for Effect Inclusion

- Effects considered (recently)
  - Copper saturation 99%
  - Phosphorus 99.9%
  - Flux – time 99%
  - Long time 99.8%
  - Vessel Fabricator 99.9%

Gating Criteria

Physical Rationale	Statistical Basis		
	Strong (>95%)	<95% & >90%	Weak (<90%)
Well Accepted	Include	Maybe	Maybe
Plausible	Include	Maybe	Exclude
Not Established	Probably	Exclude	Exclude

Deference to Empirical Evidence

VG 10

Deference to Empirical Evidence, Part 1

1860: Wöhler's fatigue experiments on German railway axles.

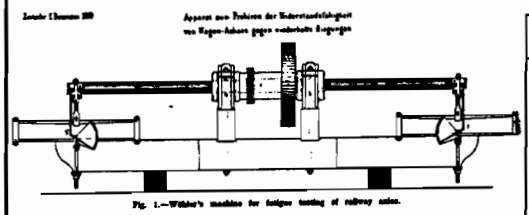


Fig. 1.—Wöhler's machine for fatigue testing of railway axles.

Physical Understanding

The metal crystallized, and so it broke.

Emprirical Evidence

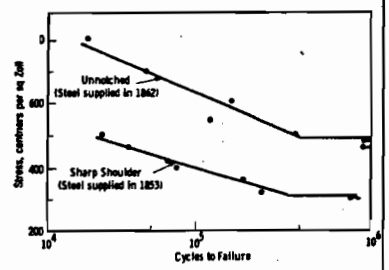
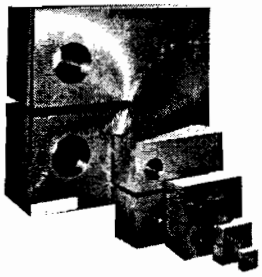


Fig. 2.—Wöhler's S-N curves for Krupp axle steel.

VG 11

Deference to Empirical Evidence, Part 2

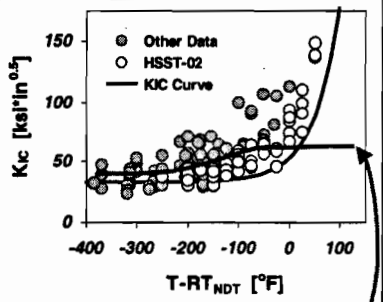
1972: A LEFM K_{IC} curve used then (and now) for analysis of nuclear RPVs.



Physical Understanding

Not *quite* so refined as we enjoy today.

Empirical Evidence



Circa-1972 physically motivated prediction

VG 12

New Embrittlement Correlation

$\Delta T_{30} =$ SMD (Phos., fluence, product form, coolant temp.) +

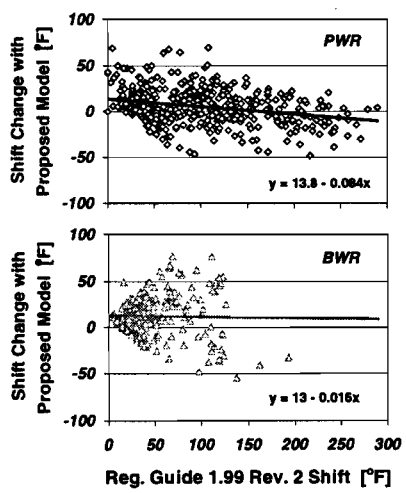
CRP (Cu, Ni, fluence, time, product form, mfgr.) +

BIAS (time)

[Eason & Wright, 2000]

VG 13

Changes Relative to RG 1.99 Rev. 2



- ✓ Mean shift somewhat higher
- ✓ Fit uncertainty reduced for welds

Data subset	Standard Deviation °F	
	New Trend Curve	RG 1.99 R2
Forgings	19.3	17
Plates, CE vessels	21.0	
Plates, non-CE vessels	19.5	
Welds, Linde 80 or Linde 0091 flux	24.0	28
Welds, Linde 1092 or other flux	23.6	
All Materials	21.5	

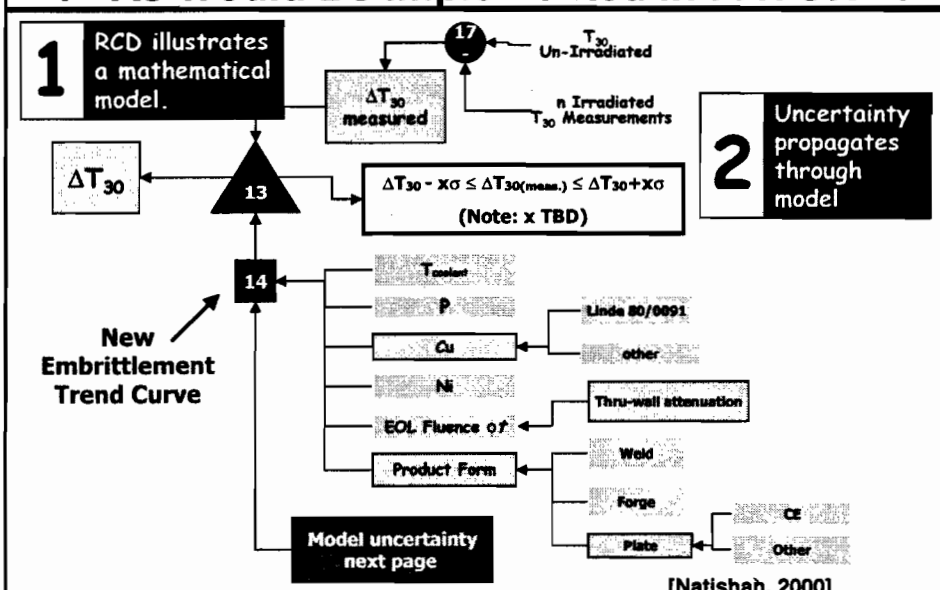
VG 14

Uncertainty Framework

- Developed using same methodology as employed to characterize RT_{NDT} / K_{IC} uncertainty
  - Data assembled and curve fit developed (MCS / UCSB)
  - Nature of uncertainties understood, framework for mathematical model developed (PEAI, EPRI Contractor)
  - Mathematical model developed consistent with PRA (UM)

VG 15

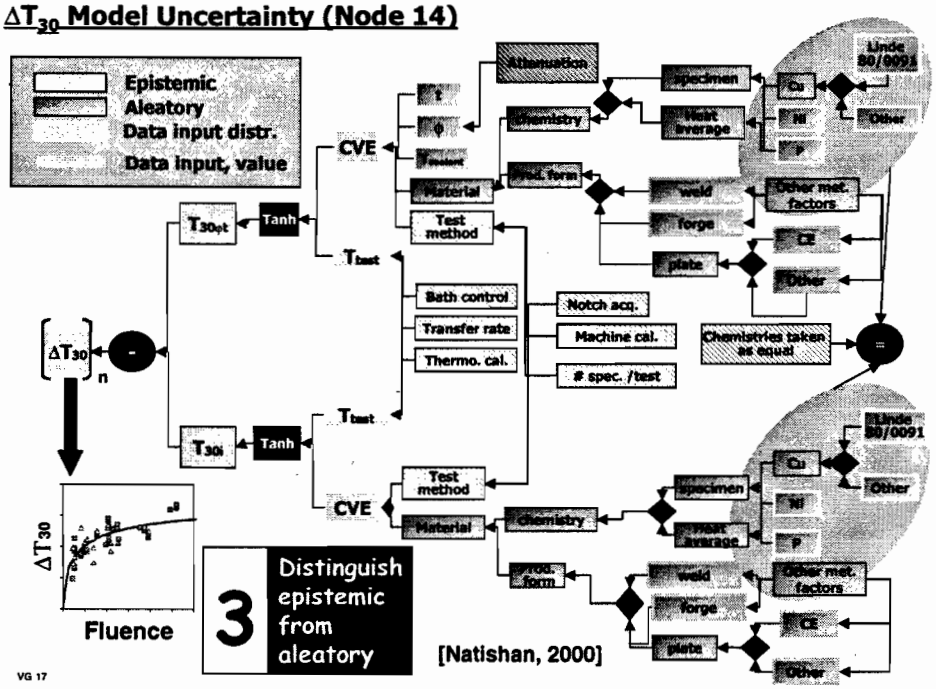
Uncertainty in ΔT_{30} Prediction
→ As Would Be Implemented in FAVOR ←



[Natishah, 2000]

VG 16

ΔT_{30} Model Uncertainty (Node 14)



Treatment of Surveillance Data

- Currently, there is no technical basis relating uncertainty reduction to the availability of surveillance data
- Consequently,
  - Continued "credit" for surveillance data by reducing the σ_Δ margin term (a.k.a RG 1.99 Rev. 2) cannot be justified
  - Proposal: Use ΔT_{30} measured from surveillance merely as a check on the embrittlement trend curve
    - ✓Define value of x in "xσ"
- A basis for a technically defensible uncertainty reduction may / could emerge from the uncertainty analysis

13

$\Delta T_{30} - x\sigma \le \Delta T_{30(meas.)} \le \Delta T_{30\,(pred.)} + x\sigma$

(Note: x TBD)

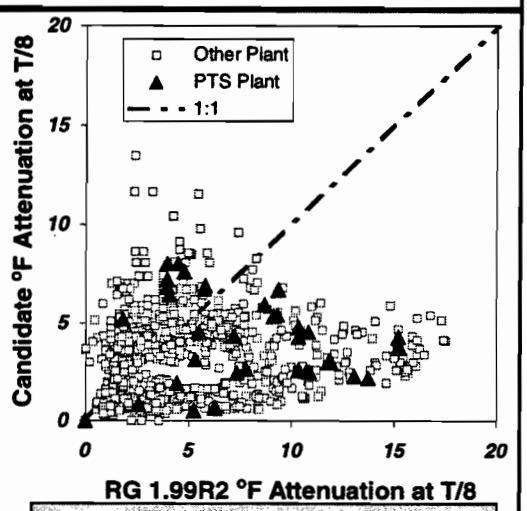
VG 18

Through Wall Attenuation

- For now use RG 1.99 R2 function to attenuate ϕt in embrittlement trend curve
  - Do not attenuate t_i
  - Do not attenuate bias

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

- For the RG consider
  - New data basis for revising this function – or not
  - Appropriate damage measure (dpa vs. ϕt w/ E>1MeV)
  - May have to await Rev. 4



Attenuation a More Significant Consideration for HU/CD & for BWRs

VG 19

Next Steps – On Going Work

- PRA-type uncertainty analysis for PTS project (Natishan and Modarres)
- Regulatory impact analysis
- Staff consideration of
  - Treatment of surveillance
  - Thru-wall attenuation
- Tech. basis document being written

VG 20

Fracture Toughness Distributions and Uncertainty Analysis

→ Status Report to ACRS ←

Mark Kirk
RES/DET/MEB

Rockville, Maryland
September 21, 2000

VG 1

Overview

- Goal and participants in cooperative effort
- Approach
- New data
- Uncertainty framework
- Current results
- Summary and future work

VG 2

Goal and Participants in Cooperative Effort

Goal: Characterize toughness using all available data in a way that is PRA consistent.

	Kirk Malik Siu	***Rosinski***
	Modarres Li Mosleh	***Natishan***
	Williams Dickson Bass; Bowman Merkle Nanstad	

VG 3

Approach

Goal: Characterize toughness using all available data in a way that is PRA consistent.

Assemble Available Valid K_{IC} and K_{IA} Data → Establish Sources of Uncertainty → Description of K_{IC} / K_{IA} for FAVOR

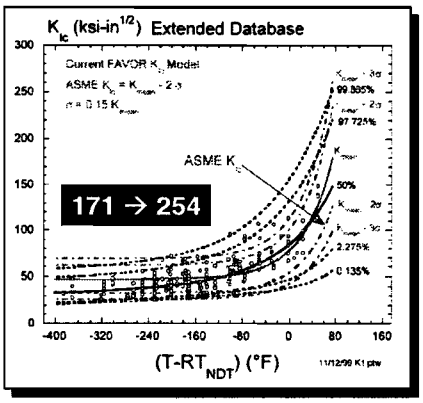
ORNL

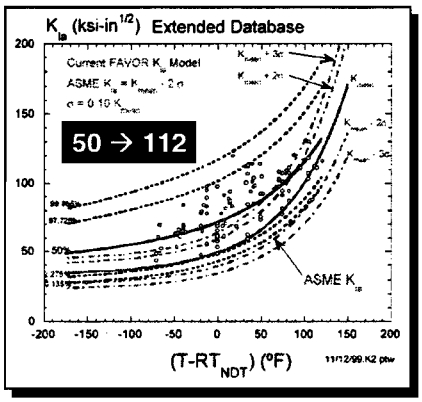
- Purely statistical fit

University of Maryland / PEAI

- Root cause analysis
- Physical basis
- Distinguish
  - Aleatory
  - Epistemic
- Procedure to treat parameter and model uncertainty
  - Consistent with PRA methodologies

Fracture Toughness Characterization
→ *New Data Collected by ORNL* ←





Old *FAVOR* scatter-bands too narrow

Effect on Probability of Vessel Failure Depends on Transient Considered

VG 5 **[Bowman and Williams, 2000]**

Uncertainty Analysis

Root Cause Analysis

- ✓ **Identify sources of uncertainties**
- ✓ **Physically based material model for toughness curves in ductile-to-brittle transition temperature range**
- ✓ **Distinguish epistemic (state of knowledge) and aleatory (randomness) parts of uncertainties**
  - **Natishan (EPRI funding)**

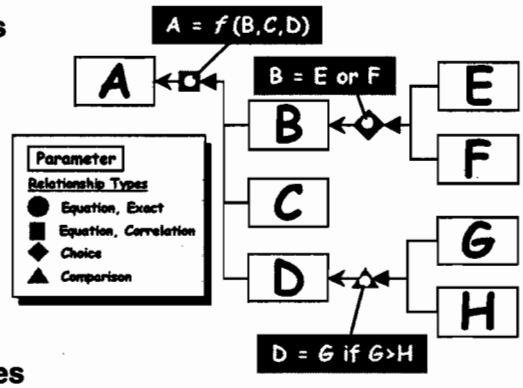
PRA Framework

- ✓ **Distinguish epistemic (state of knowledge) and aleatory (randomness) parts of uncertainties**
- ✓ **Mathematical model to quantify uncertainties**
- **Product:** ✓ **Recommended program structure for FAVOR**
  - **Modarres, Fei, and Mosleh (NRC funding)**

VG 6

Root Cause Analysis

- Describe uncertainties
  - Data
  - Model / equation
  - Expert judgements
- Classify uncertainties
  - Aleatory
    - ✓ Non-reducible
  - Epistemic (state-of-knowledge)
    - ✓ Reducible
- Propagate uncertainties through model



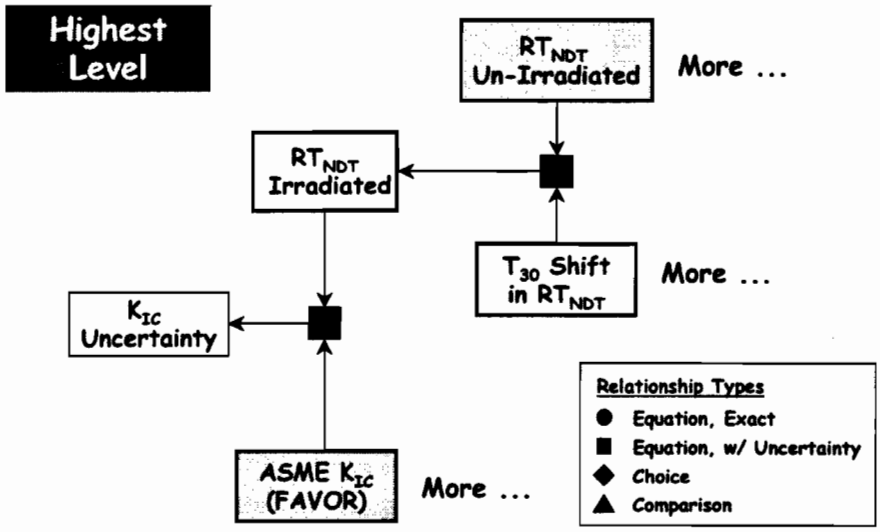
[Natishan & Modarres, 2000]

The BIG Change from the Old Ways

Input uncertainties propagate to output uncertainties via a systematic and critiquable process, rather than margins being prescribed to the analysis *a priori*.

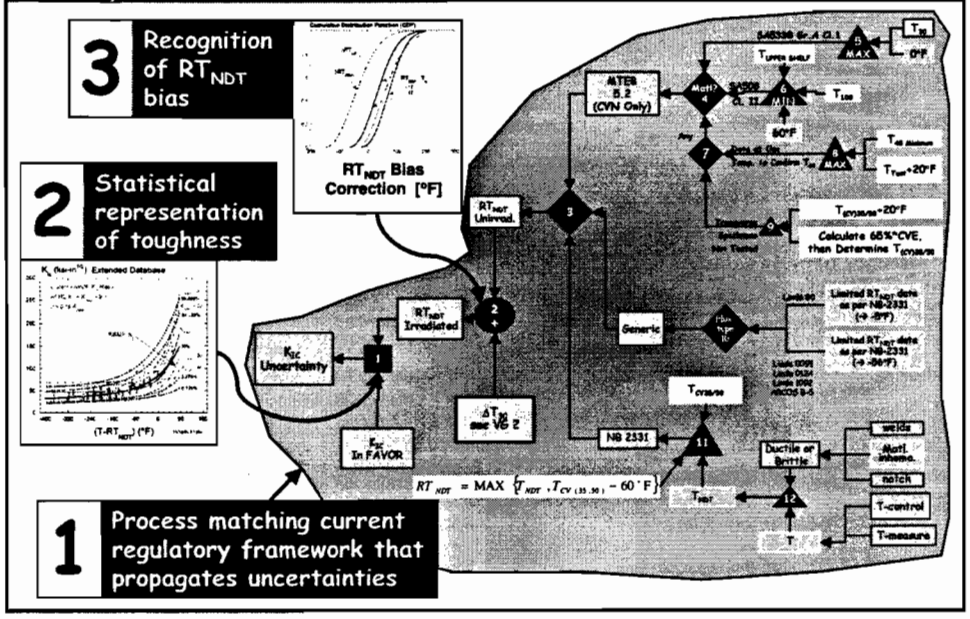
VG 7

Root Cause Analysis of K_{IC} / RT_{NDT} Process

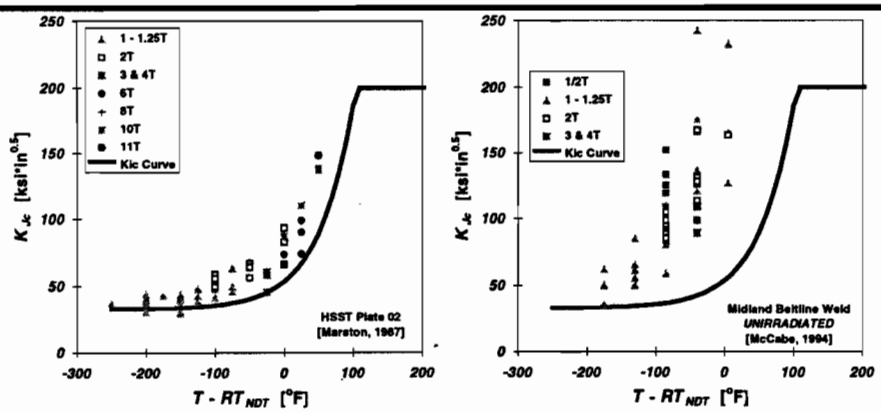


VG 8

Root Cause Analysis of K_{IC} / RT_{NDT} Process
→ New or Significant Features ←



Bias in RT_{NDT}



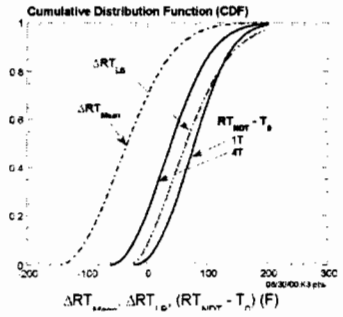
Expected since RT_{NDT} designed to be a bounding estimate of transition temperature, but …

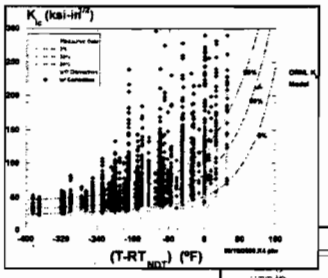
Inconsistent with a PRA approach that relies on "best estimates."

VG 10

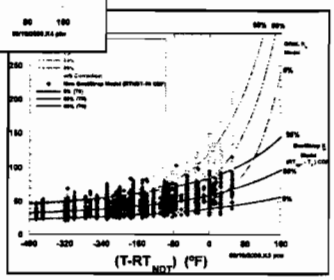
Bias in RT_{NDT}

Candidate Correction Functions





Different Correction Procedures



Appropriate correction and procedure to correct still being developed.

VG 11

Summary and Future Work

- **Completed**
  - **Statistical transition fracture toughness model**
    - ✓ **Data collected**
    - ✓ **Fit completed**
  - **PRA-uncertainty framework**
    - ✓ **Current processes understood using root-cause diagram approach**
    - ✓ **Mathematical models developed**
      - ➢ **Details of FAVOR implementation discussed / clarified**
- **On-going**
  - **Full implementation of uncertainty model into FAVOR**
  - **Resolution of RT_{NDT} bias correction function and modeling procedure**
  - **Assembly of input data to run the models**

VG 12

Status of the FAVOR Code Development

Terry Dickson, Richard Bass and Paul Williams
Heavy-Section Steel Technology Program
Oak Ridge National Laboratory

Advisory Committee on Reactor Safeguards
Meeting of the Materials and Metallurgy Subcommittee
Pressurized Thermal Shock Screening Criterion Re-evaluation

September 21, 2000
Nuclear Regulatory Commission
Rockville, Maryland

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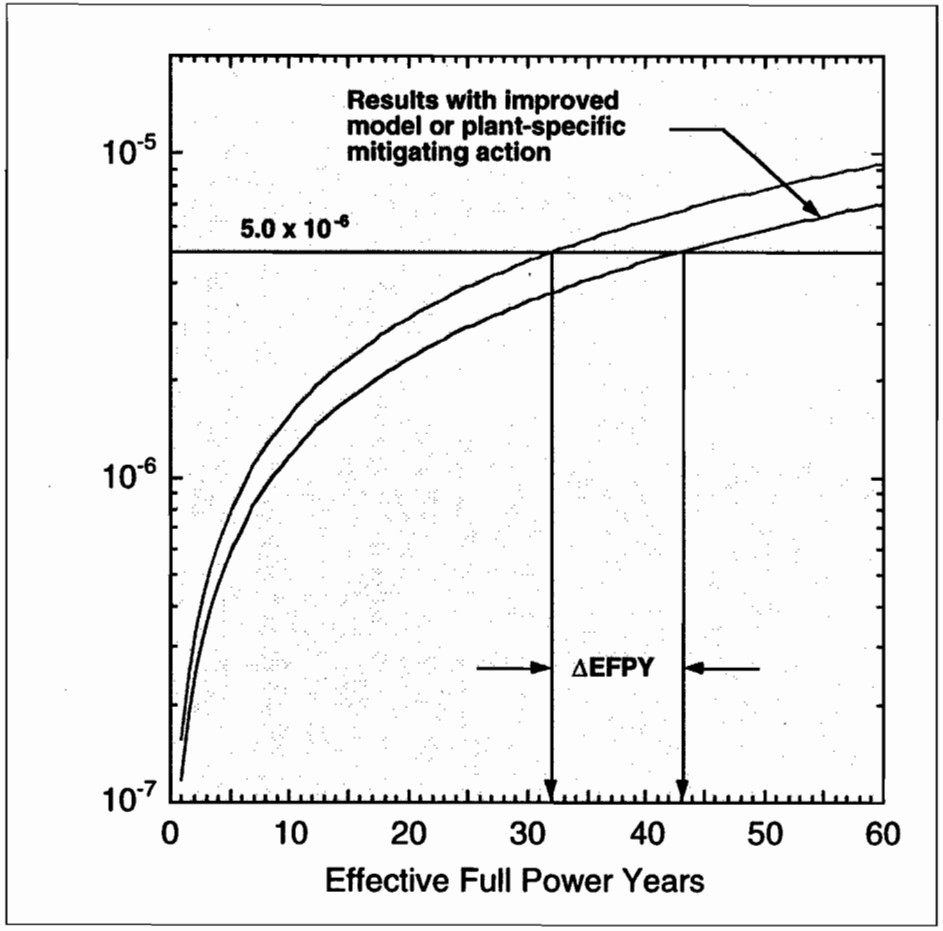
Application of FAVOR to PTS Re-evaluation Addresses the Following Two Questions



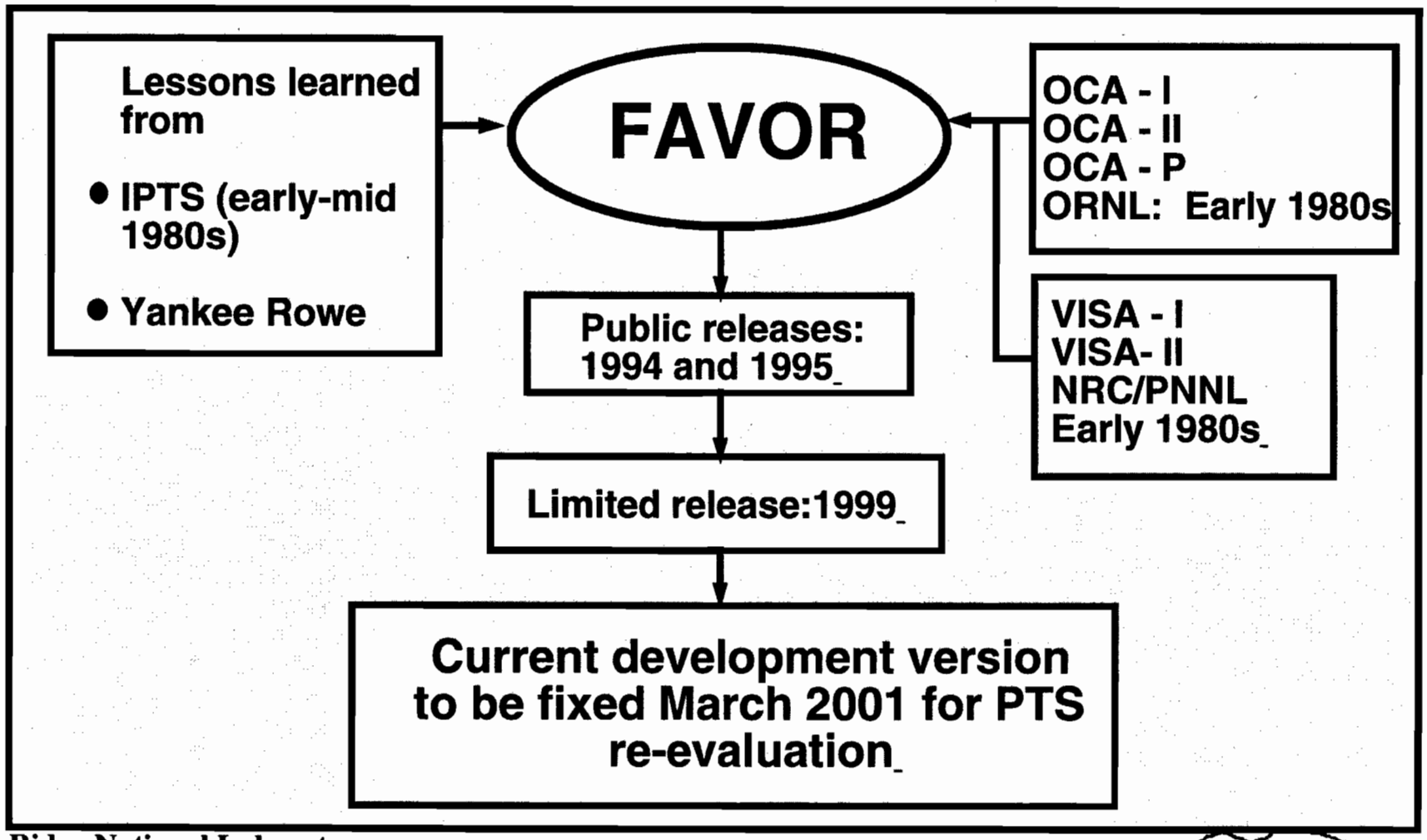
- At what time in operating life does frequency of RPV failure exceed acceptable value (currently 5 x 10e-06)?
- How does integration and application of advanced technology affect the calculated result?

Near-Term Schedule for Development of the FAVOR Code has been Defined

- Current schedule specifies FAVOR to be ready for PTS re-evaluation analyses on March 1, 2001

- In the interim period:
  - models are being finalized
  - finalized models are being implemented
  - scoping studies are being performed

Development of the FAVOR Code was Initiated in Early 1990s by Combining Best Attributes of OCA/VISA with Evolving Technology



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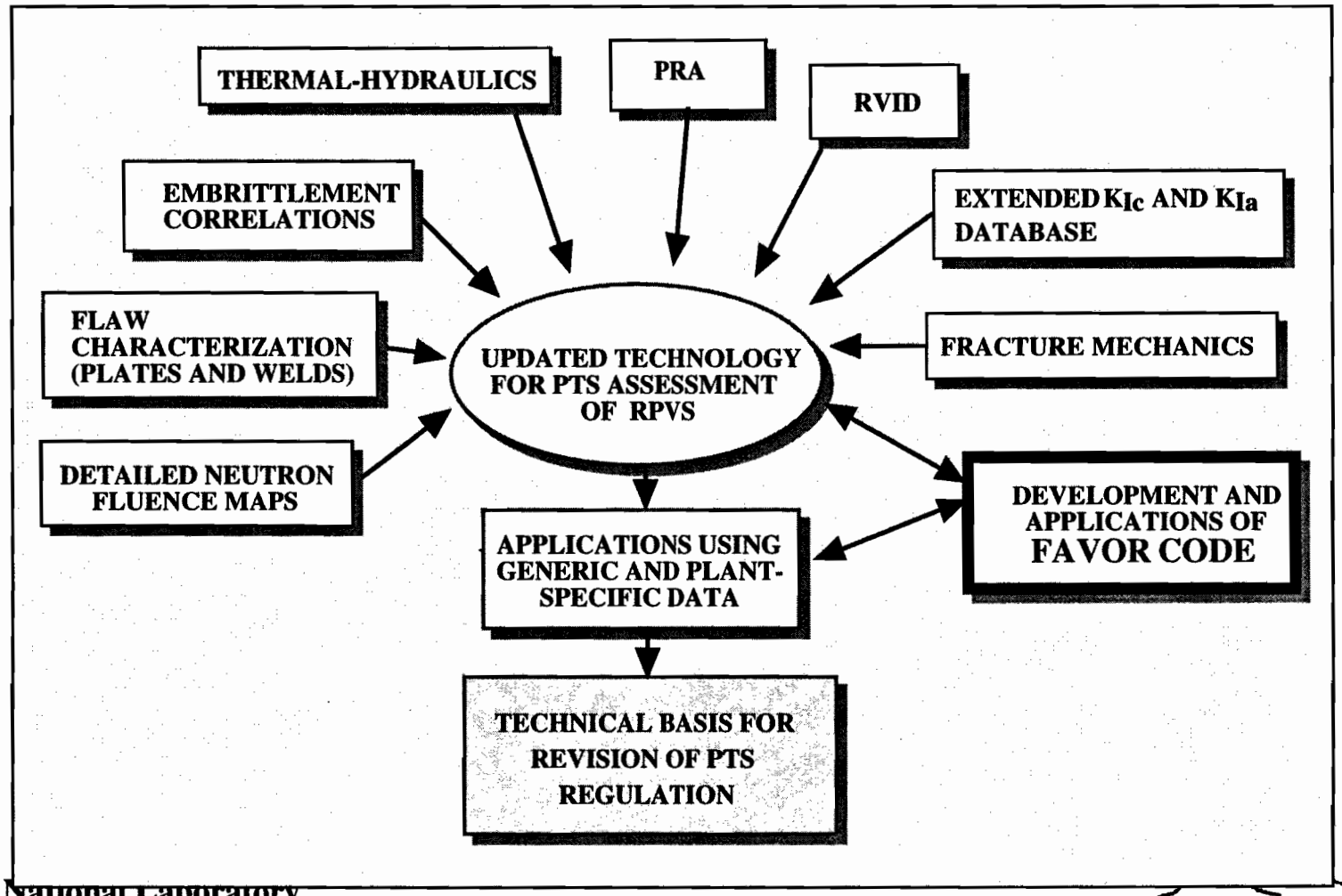
Elements of updated technology are currently being integrated into the FAVOR computer code to re-examine the current PTS regulations*

(Fracture Analysis of Vessels: Oak Ridge)



Advanced Technology is Integrated into FAVOR to Support Possible Revision of PTS Regulation

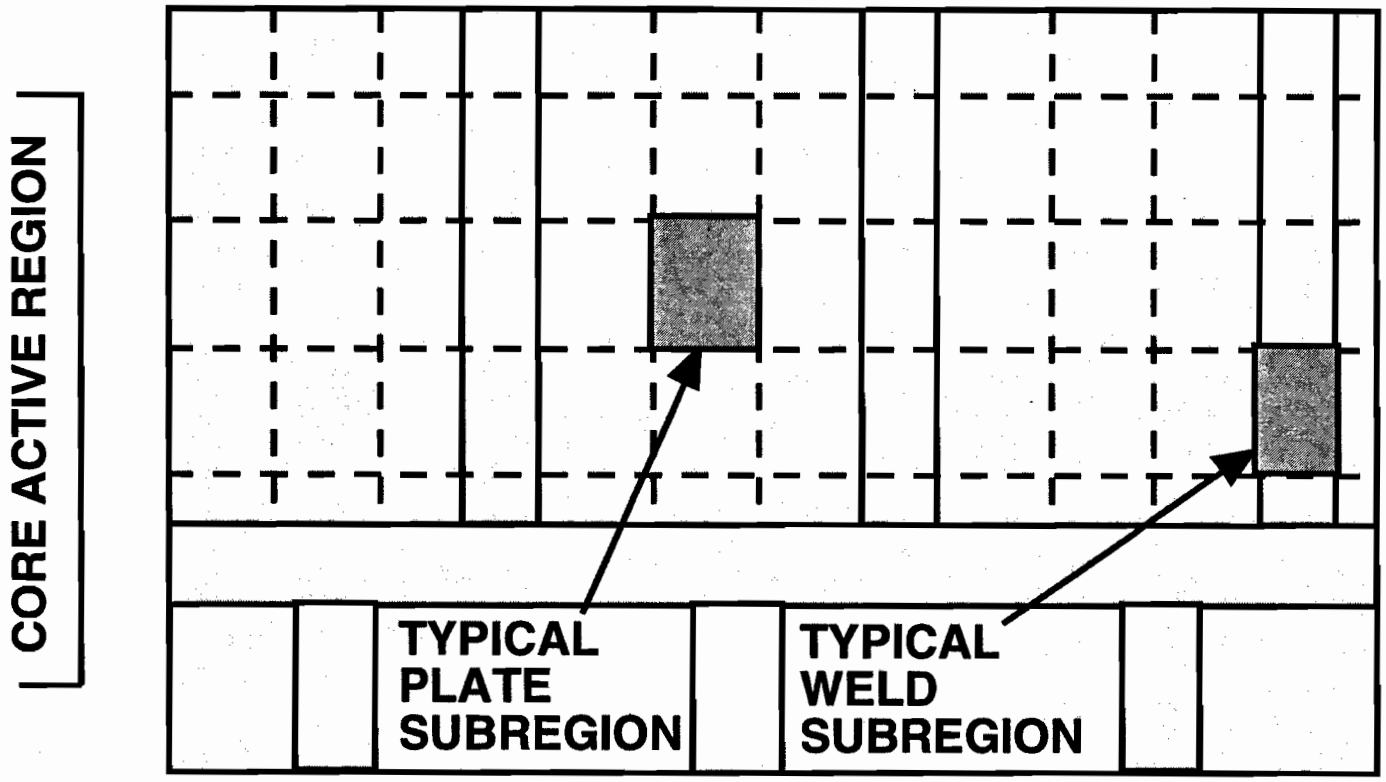
- Flaw characterizations from NRC research (plates and welds)
- Detailed fluence maps
- Embrittlement correlations
- RVID
- Fracture toughness models
- Surface-breaking and <u>embedded</u> flaws
- Inclusion of through-wall weld residual stresses
- New PFM methodology

A Significant Improvement Since the Derivation of Current PTS Regulations is Flaw Characterization*

** analyses assumed all flaws were inner-surface breaking flaws*

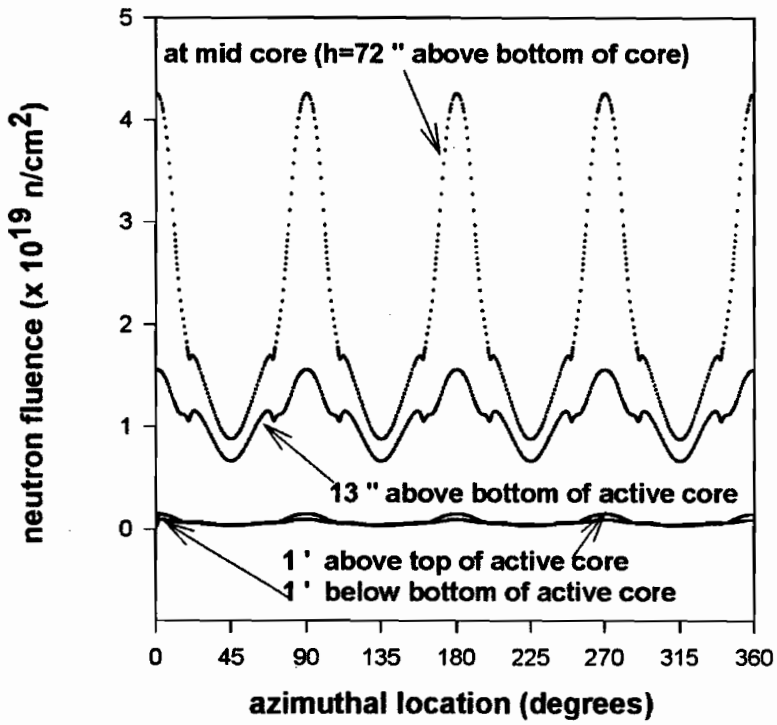
- Recent NDE and DE of RPV material at PNNL has established an improved technical basis for flaw-related data used as input for PFM analyses
- A significantly higher number of flaws were found than was postulated in PFM analyses from which current PTS regulations were derived; however, all flaws detected thus far are embedded
- Application of PVRUF flaw densities to commercial PWR results in over 3500 flaws in 1st 3/8 thickness of RPV wall

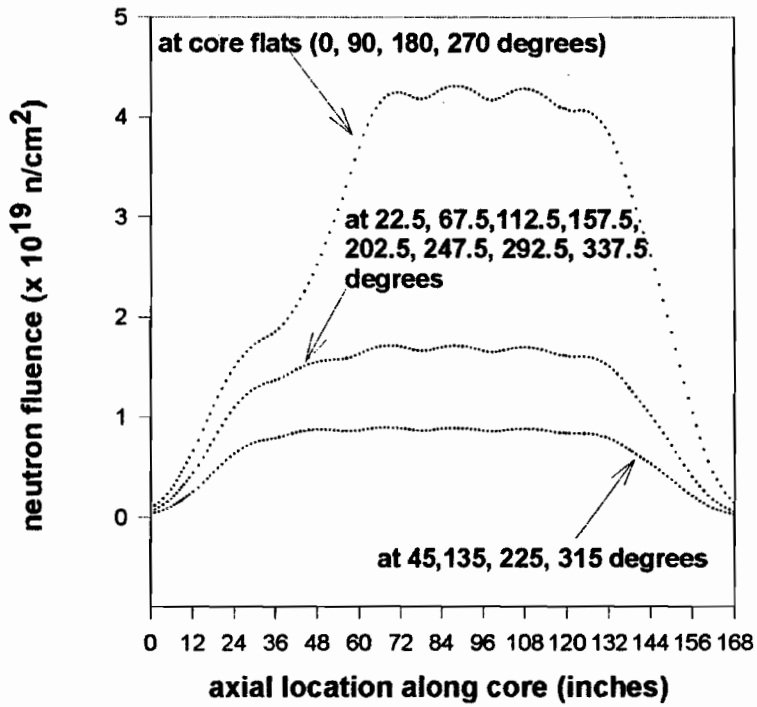
FAVOR utilizes a methodology that allows the RPV beltline to be discretized into sub-regions, each with its own distinguishing embrittlement-related parameters. This accommodates chemistries from the RVID and detailed neutron fluence maps



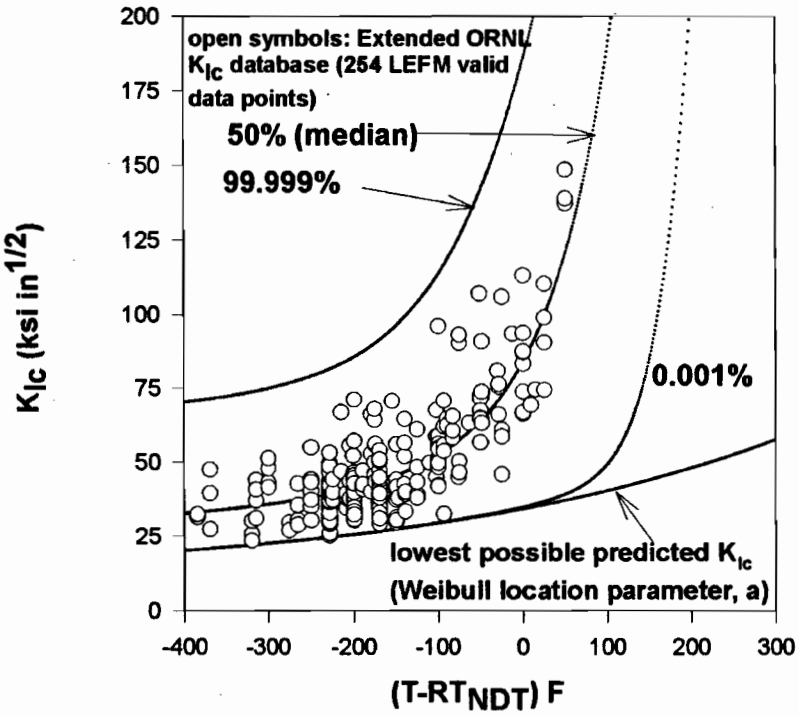
TYPICAL VESSEL BELTLINE LAYOUT

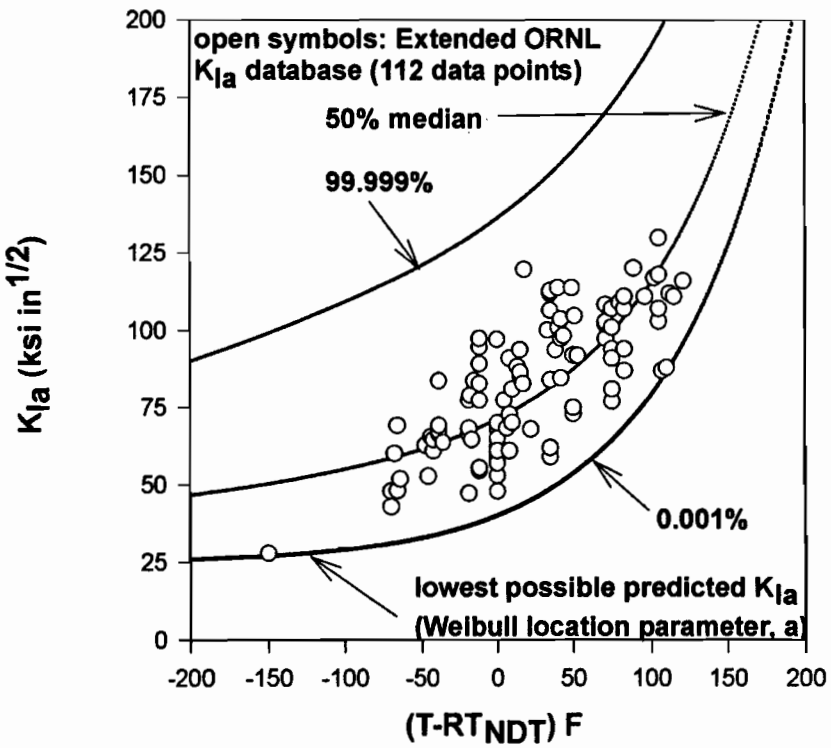
Brookhaven National Laboratory is generating very detailed neutron fluence maps for selected PWRs corresponding to 32 EFPY and 40 EFPY



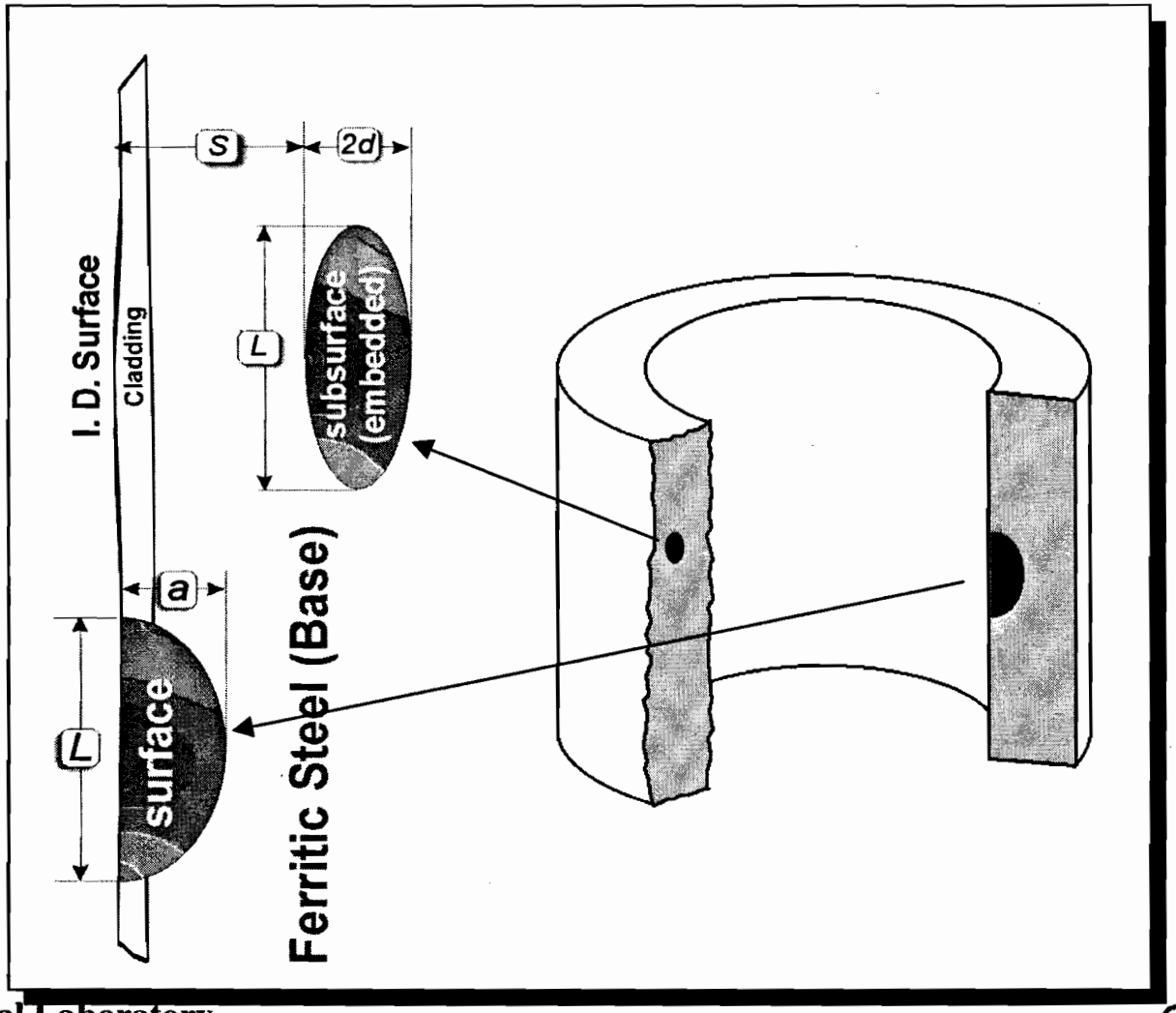


New Statistical Models for Enhanced Plane-Strain Static Initiation (K_{Ic}) and Arrest (K_{Ia}) Fracture Toughness Data Bases were Implemented into FAVOR





The FAVOR PFM Model Now Includes Inner Surface-Breaking and/or Embedded Flaws



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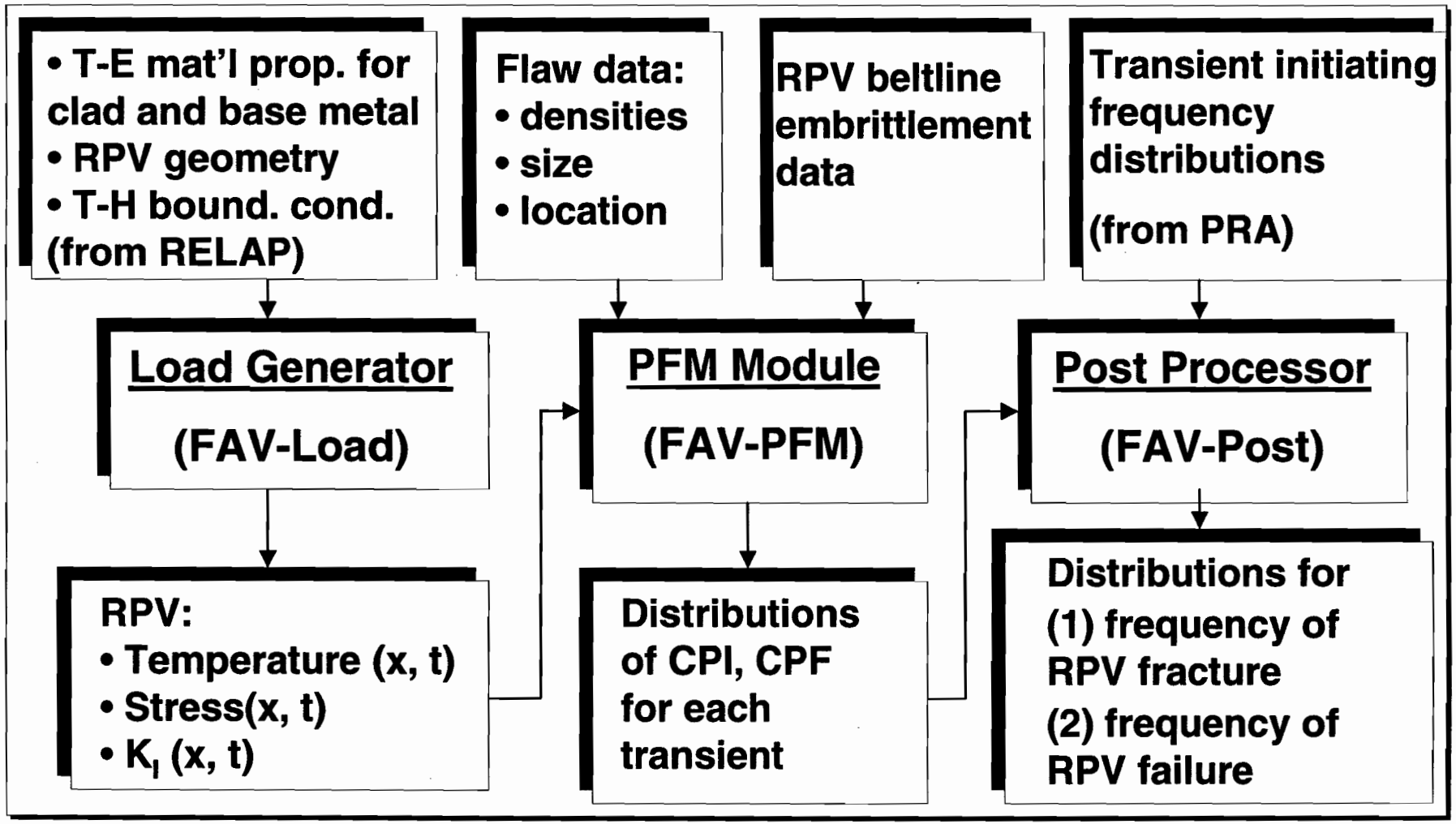
The Current FAVOR Code Consists of Three Separate Modules



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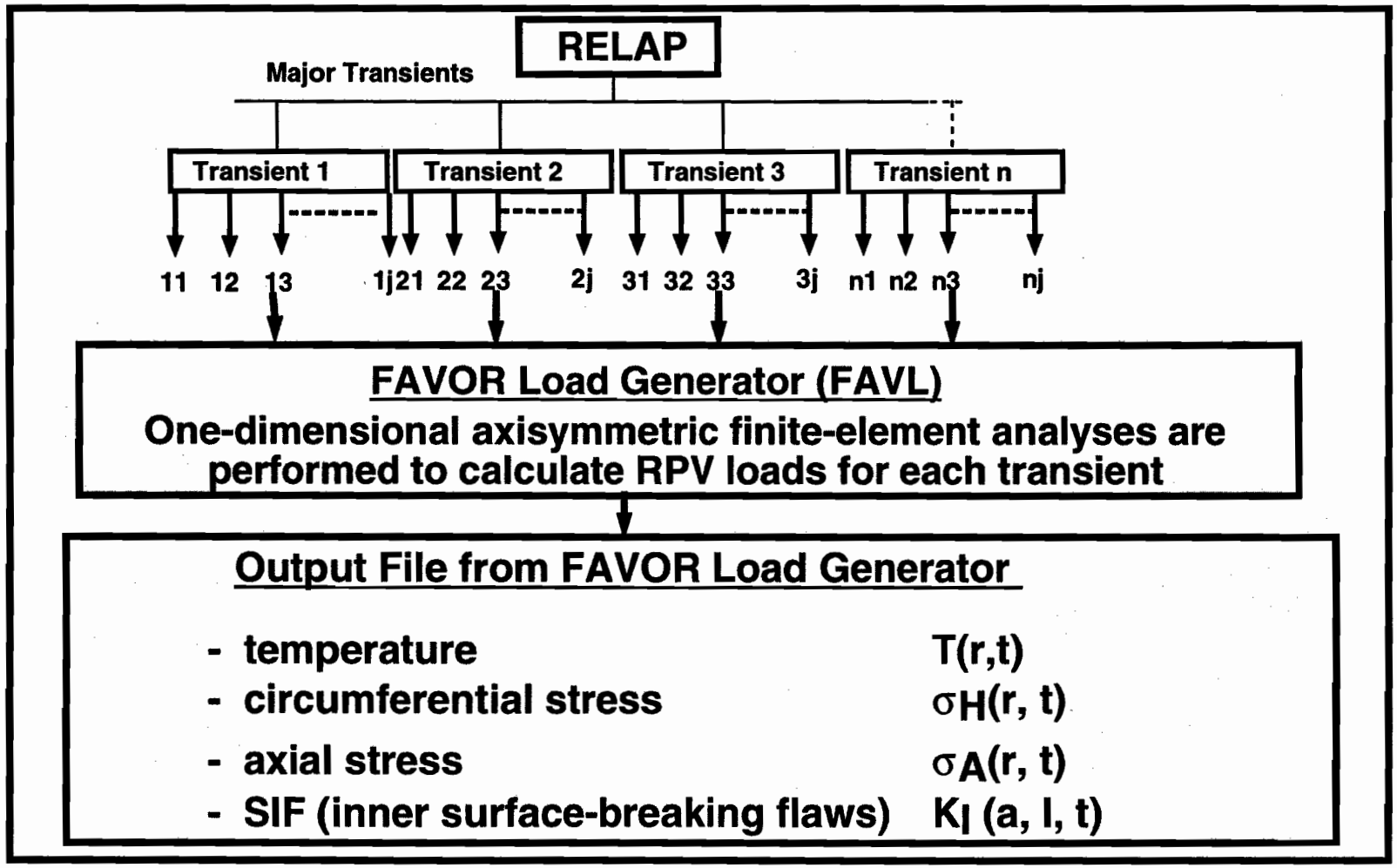
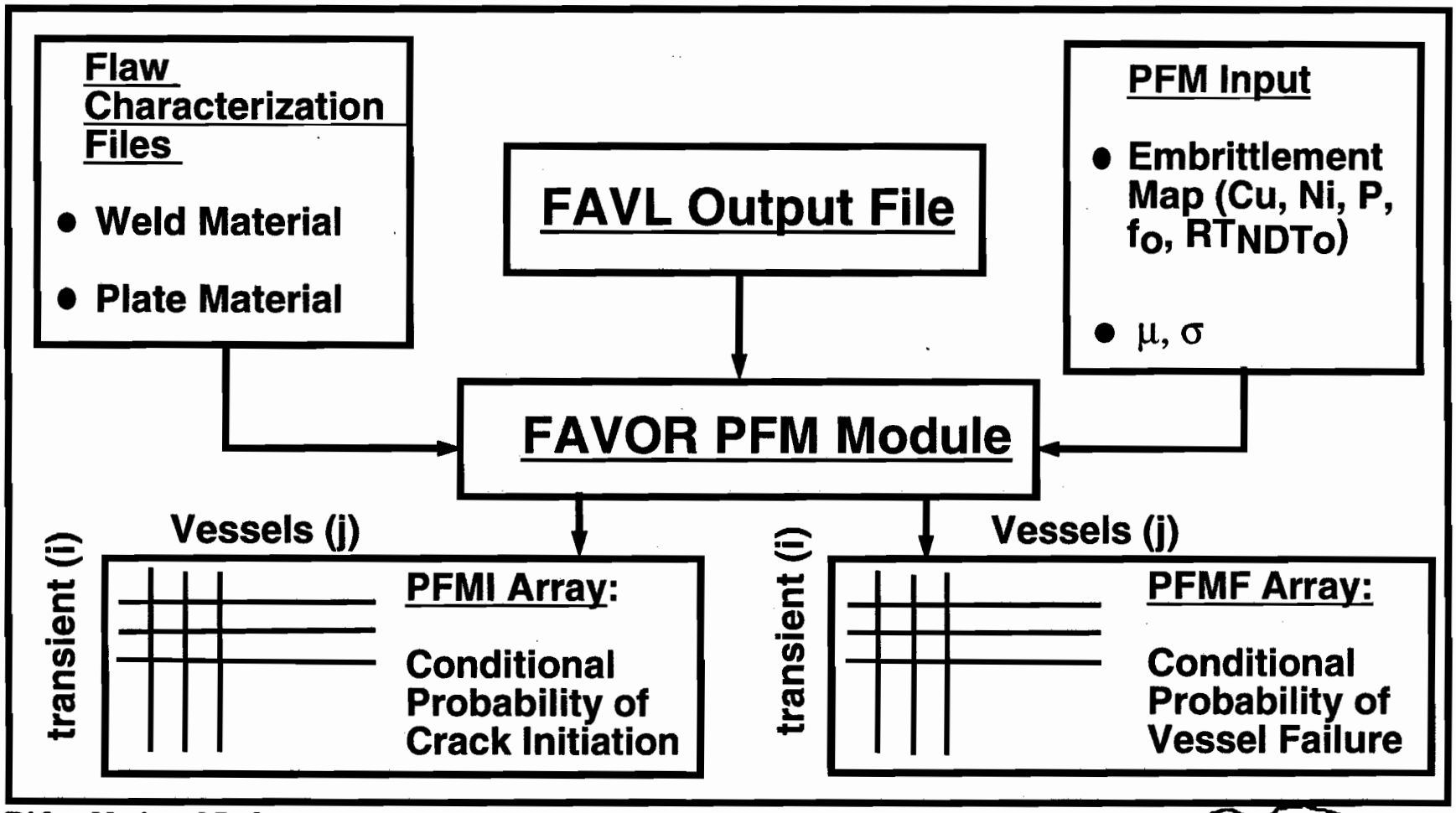
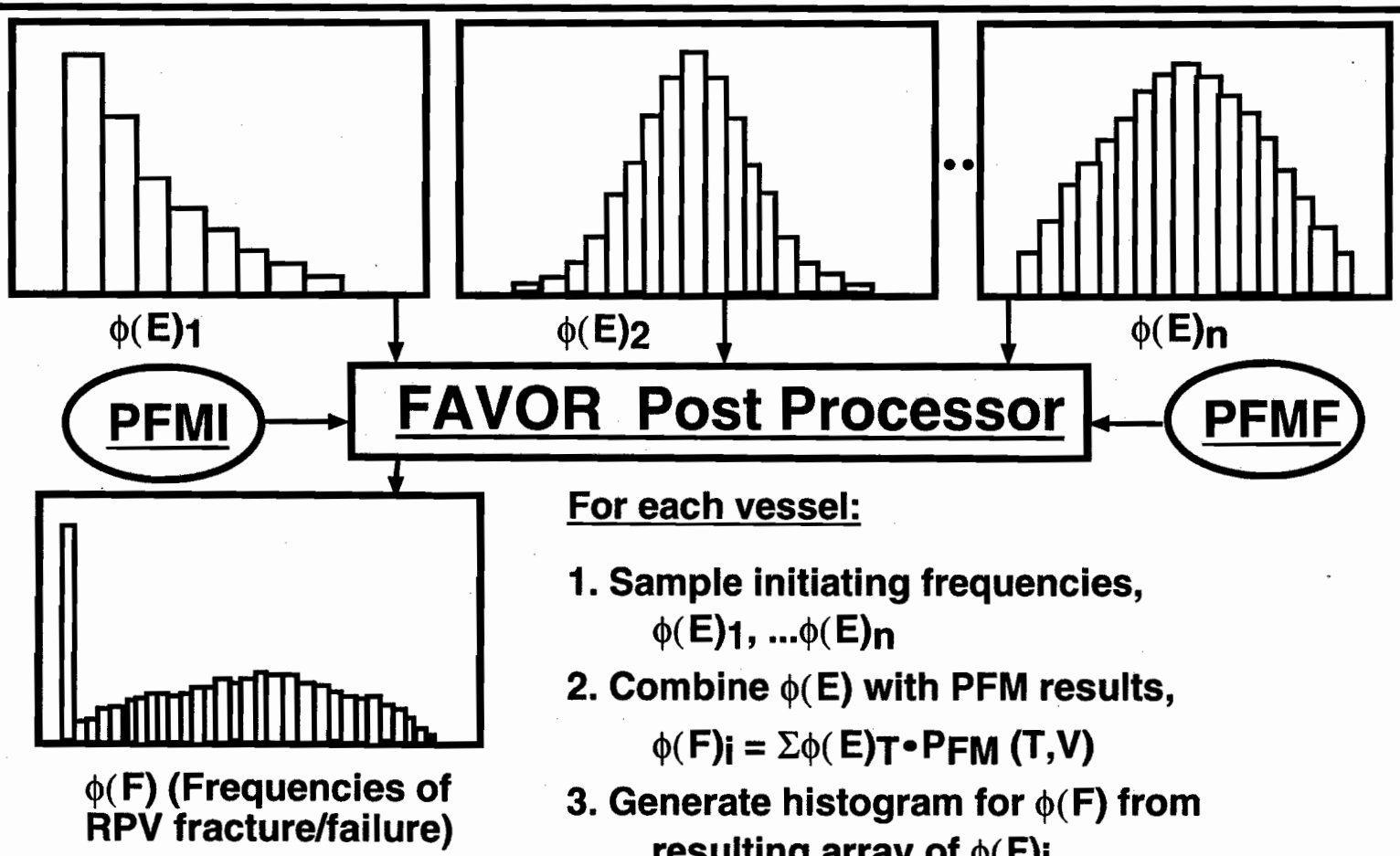
FAVOR Analyses Incorporate Uncertainty Associated with Thermal Hydraulics by Including Variants for Each of the Transients



The FAVOR PFM Analysis Module Generates Arrays Containing Conditional Probabilities of Initiation (PFMI) and Failure (PFMF) for Vessel(j) Subjected to Transient(i)



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



For each vessel:

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

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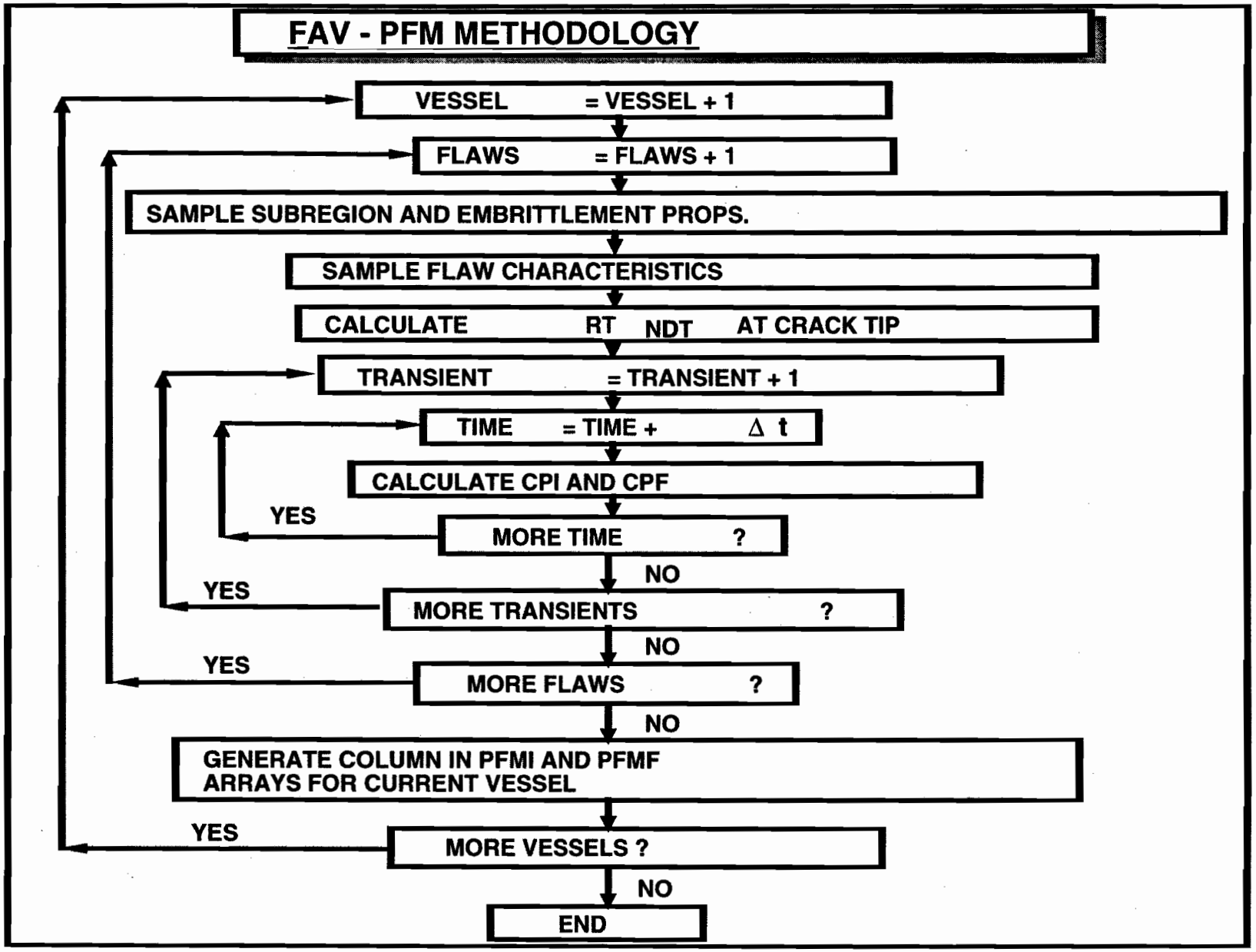
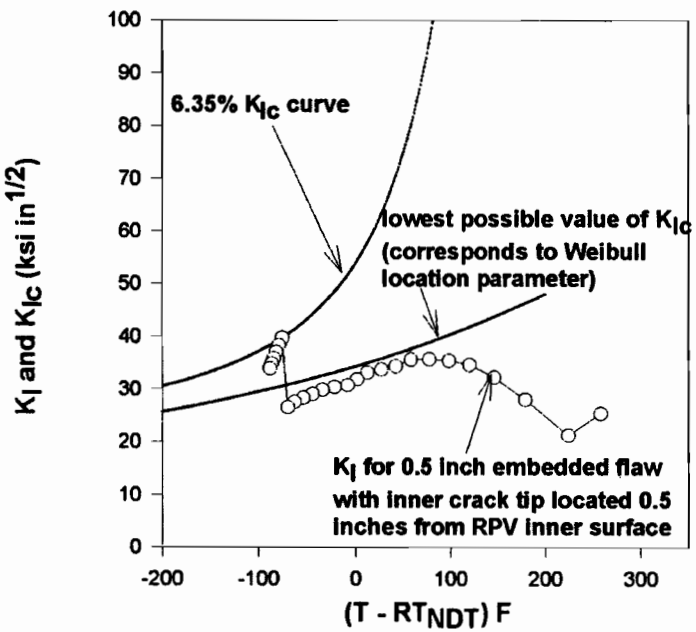
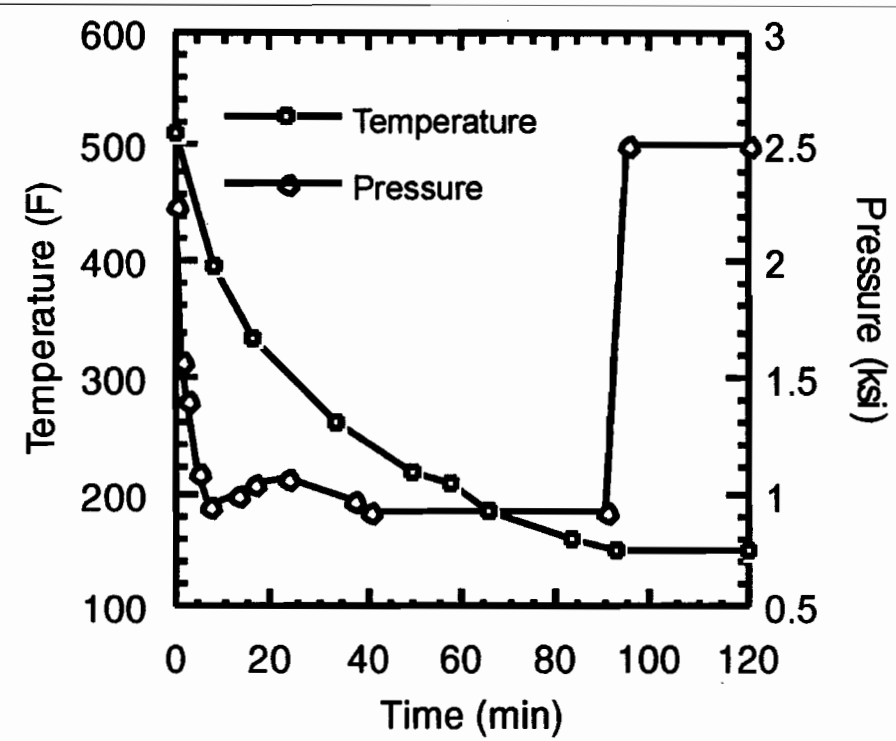

FAV - PFM METHODOLOGY
VESSEL = VESSEL + 1
FLAWS = FLAWS + 1
SAMPLE SUBREGION AND EMBRITTLEMENT PROPS.
SAMPLE FLAW CHARACTERISTICS
CALCULATE RT NDT AT CRACK TIP
TRANSIENT = TRANSIENT + 1
TIME = TIME + Δ t
CALCULATE CPI AND CPF
YES
MORE TIME ?
NO
YES
MORE TRANSIENTS ?
NO
YES
MORE FLAWS ?
NO
GENERATE COLUMN IN PFMI AND PFMF
ARRAYS FOR CURRENT VESSEL
YES
MORE VESSELS ?
NO
END

The Conditional Probability of Initiation is Calculated by Solving the Weibull K_{Ic} CDF for the Fractional Part (Percentile) of the Distribution that Corresponds to the Applied KI(t)

$$\text{CPI (transient, time, flaw)} = P = 1 - \exp\{-[((K_I(t) - a)/b)^c]\}$$

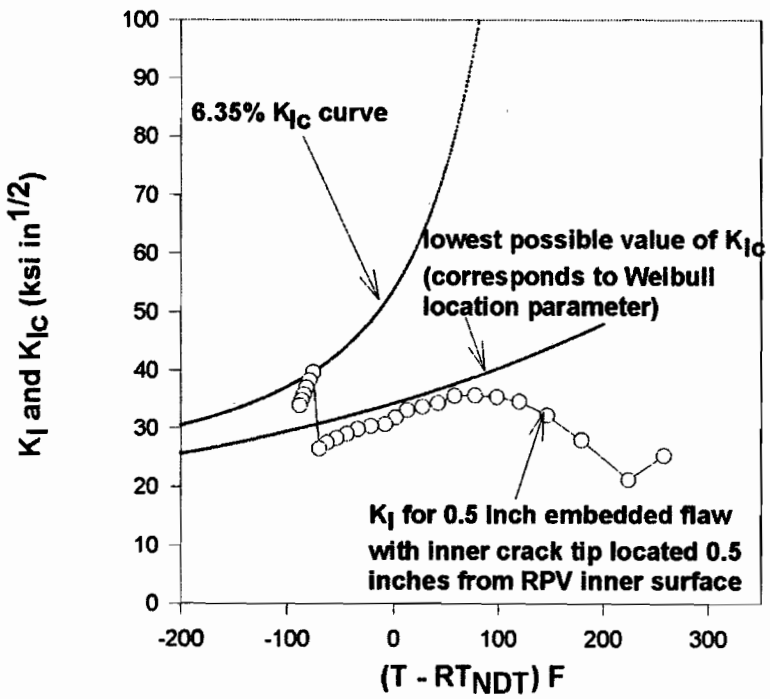


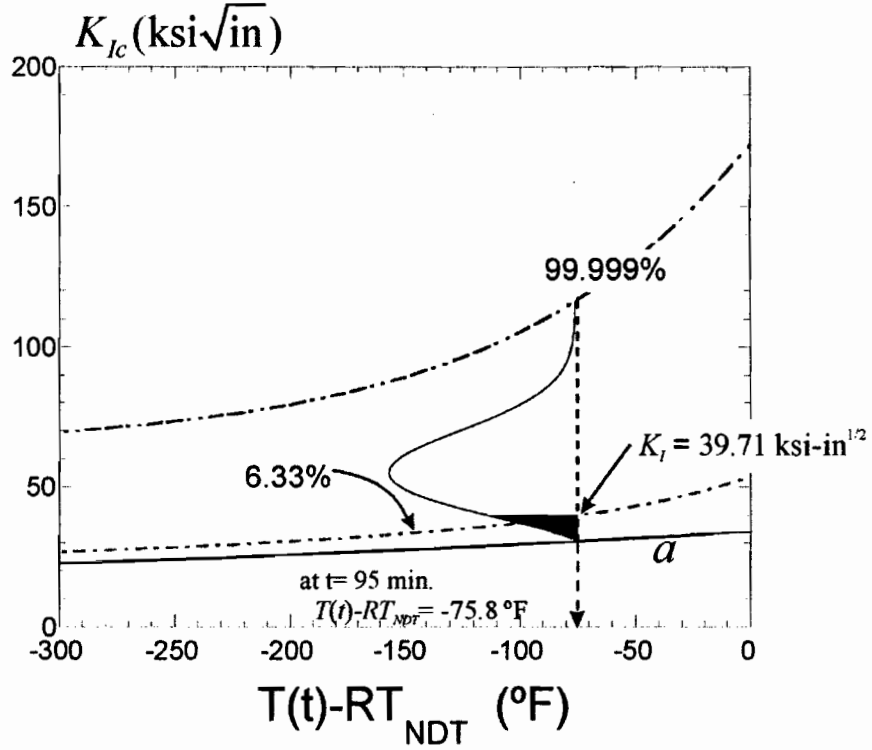


In the Illustrative Example Problem, at the time of Re-pressurization:

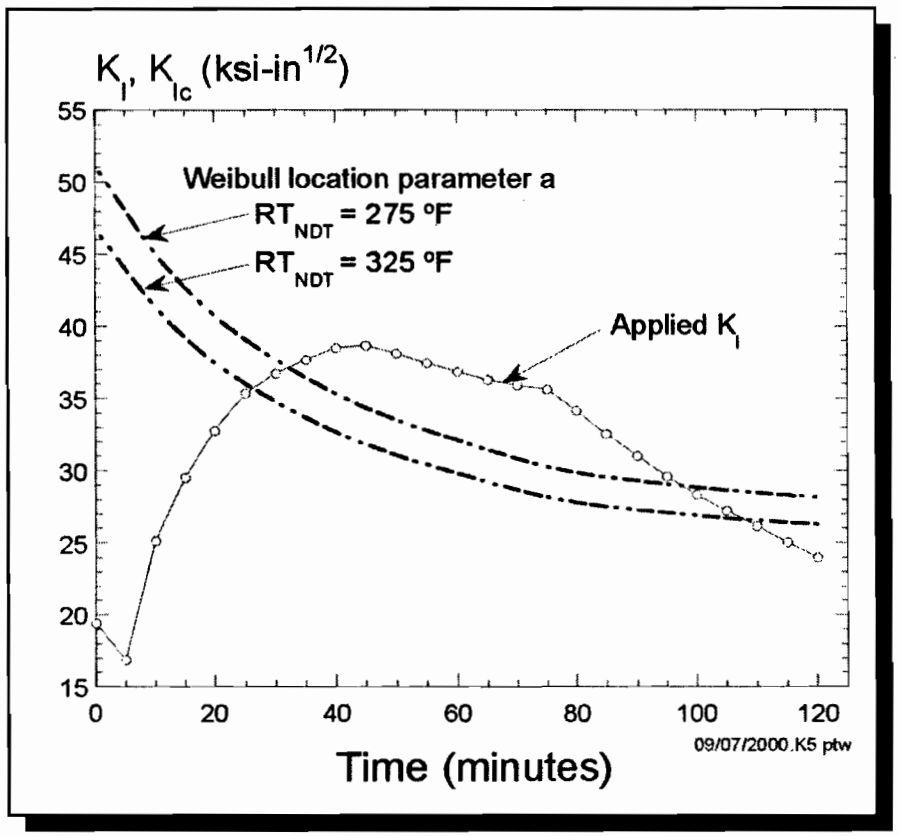
K_I > 0.0635 of the Weibull K_{Ic} distribution at [T(t)-RTNDT] = -75.8;

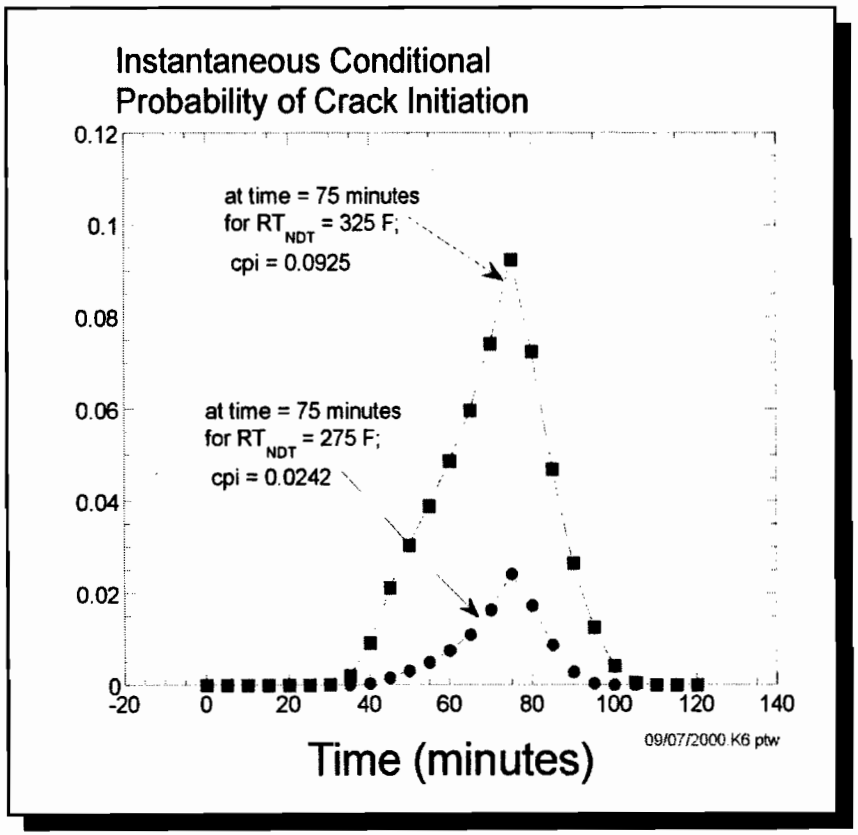
therefore, the CPI = 0.0635 for this flaw at this time





As embrittlement increases over the operating life of an RPV, Conditional Probability of Initiation (CPI) Increases for Given Flaw and Transient





The following computation processes array CPI *for each vessel to account for multiple flaws*

$$PFMI(transient\,i, vessel\,j) = \left(1 - \prod_{flaw=1}^{nflaw} [1 - CPI(transient\,i, \max, flaw))\right)$$

where:

$$\prod_{k=1}^{nflaw} \left[1 - CPI(i, \max, k)\right] = \left[1 - CPI(i, \max, 1)\right]\left[1 - CPI(i, \max, 2)\right] \ldots \left[1 - CPI(i, \max, nflaw)\right]$$

= Probability that none of the *nflaw* flaws initiates in cleavage fracture

$\therefore PFMI(transient\,i, vessel\,j)$ = Probability that at least one of the *nflaw* flaws (in vessel j) initiates in cleavage fracture when subjected to transient i

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