



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

April 22, 2003

MEMORANDUM TO: William Shack, Vice Chairman, Materials
and Metallurgy Subcommittee

FROM: Ramin Assa, Senior Staff Engineer
Technical Support Staff

A handwritten signature in black ink, appearing to read "Ramin Assa".

SUBJECT: CERTIFICATION OF THE MINUTES OF THE ACRS SUBCOMMITTEE
MEETING ON MATERIALS AND METALLURGY - FEBRUARY 5, 2003,
ROCKVILLE, MARYLAND

I hereby certify that, to the best of my knowledge and belief, the Minutes of the subject meeting issued April 22, 2003, are an accurate record of the proceedings for that meeting.

A handwritten signature in black ink, appearing to read "William Shack".

W. Shack, Vice Chairman

4/23/03

Date



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

April 23, 2003

MEMORANDUM TO: ACRS Members

FROM: Ramin Assa, Senior Staff Engineer
Technical Support Staff

A handwritten signature in black ink, appearing to read "Ramin Assa".

SUBJECT: CERTIFIED MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON
MATERIALS AND METALLURGY - FEBRUARY 5, 2003, ROCKVILLE,
MARYLAND

The minutes of the subject meeting, issued on April 22, 2003, 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: J. Larkins
S. Bahadur
R. Savio
H. Larson
S. Duraiswamy
S. Banerjee
F. Moody
V. Schrock
ACRS Staff Engineers

CERTIFIED BY: W. Shack

Certified on: April 23, 2003

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MINUTES OF ACRS SUBCOMMITTEE MEETING ON
MATERIALS AND METALLURGY
FEBRUARY 5, 2003
ROCKVILLE, MD

INTRODUCTION

The ACRS Subcommittee on Materials and Metallurgy held a meeting on February 5, 2003, at 11545 Rockville Pike, Rockville, Maryland, in Room T-2B3. The purpose of the meeting was to hold discussions with representatives of the Office of Nuclear Regulatory Research (RES), relating to technical basis for revisions of the pressurized thermal shock (PTS) screening criteria in the PTS rule. Mr. Ramin Assa was the cognizant ACRS staff engineer for this meeting. The meeting was convened at 8:30 a.m. and adjourned at 4:50 p.m. on the same day.

PARTICIPANTS:

ACRS

W. Shack, Vice Chairman
M. Bonaca
P. Ford
T. Kress
G. Leitch

V. Ransom
S. Rosen
G. Wallis
S. Banerjee, Consultant

NRC Staff

D. Bessette
E. Hackett
M. Kirk
M. Mayfield

J. Rosenthal
N. Siu

NRC Contractor

A. Kolaczowski, SAIC

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

INTRODUCTION

Dr. William Shack, Vice Chairman of the ACRS Materials and Metallurgy Subcommittee, presiding, convened the meeting and stated that the purpose of the meeting was to review staff's draft NUREG report on the technical basis for revising the PTS rule (10 CFR 50.61.) Dr. Shack then called upon NRC staff to begin.

NRC STAFF PRESENTATION

Introduction: Mr. Michael Mayfield, RES

Mr. Mayfield started his opening remarks by saying that the PTS Project has been a major undertaking for RES. He then introduced Mr. Siu and asked him to begin with the presentation.

OVERVIEW OF PTS RE-EVALUATION PROJECT - Messrs. Nathan Siu, Edward Hackett, and Mark Kirk, RES

Mr. Siu stated that this project has been supported by industry, specifically the Materials Reliability Program (MRP) of the Electric Power Research Institute (EPRI.)

Dr. Wallis noted that the draft NUREG report appeared to have been written by different people and was not well integrated. Mr. Siu acknowledged this point.

Dr. Ford questioned whether a thorough peer review was conducted as stated in the report's cover letter. Mr. Hackett stated that this activity was in progress and expected to be completed in 2003.

Drs. Ransom and Wallis pointed out that the NUREG does not provide a clear relationship between itself and referenced reports by University of Maryland and Oregon State University (OSU.) Mr. Bessette stated that the results of the OSU report were implicit in the NUREG.

Mr. Kirk noted that the objectives of the meeting were to review the draft NUREG and show a strong case to support rulemaking. Results of the plant-specific evaluation of two of the most embrittled plants in the fleet had shown that these plants had more margin against failure by PTS than previously believed.

Dr. Wallis stated that figure 1.1 in the report was very confusing. Mr. Kirk acknowledged and stated that the two sigma margins were misrepresented. Mr. Hackett added that there has been a fair amount of confusion over this issue over the years and RES' goal was to clarify this issue during the meeting. Mr. Wallis reiterated the need for peer review to identify and correct errors before issuance of the final report. Mr. Kirk acknowledged that the project was not over yet and needed additional reviews, including a response from Office of Nuclear Reactor Regulation (NRR).

Dr. Ford raised the issue of plants that were approaching RT_{PTS} screening criteria and were interested in applying for license renewal. If the current 10 CFR 50.61 rule is not changed, these plants could not easily request a license extension. According to Mr. Hackett, Palisades Plant is the closest to and is projected to reach the screening criteria around 2011.

Mr. Rosen noted that the report only provides the technical basis for a change to the current PTS rule and asked about the criteria used for deciding whether to proceed with a rule change. Mr. Hackett responded that a petition for rulemaking from the industry could initiate this activity but the allocation of resources would be the determining factor.

ANALYSIS APPROACH - Mr. Kirk, RES

Mr. Kirk presented a brief background of the PTS project. The licensee for Yankee Rowe power plant had predicted that the vessel embrittlement would reach the current PTS screening limit before the end of the plant's licensing life (EOL) and had attempted to follow the provisions of Regulatory Guide (RG) 1.154 to support operations at embrittlement levels greater than those implied by the screening criteria in 10 CFR 50.61. However, their efforts were not successful and the plant was permanently shut down in 1991. Following the difficulties with implementing RG 1.154, the Commission directed the staff to revise the RG and associated rule.

Since the original PTS rule was issued, improvements in Probabilistic Risk Assessment (PRA) analysis, thermal-hydraulics studies, and probabilistic fracture mechanics calculations, suggest that the current rule may be overly conservative. In the analysis supporting the development of the original rule, it was shown that a shift in mean value of the fracture toughness transition temperature to 210°F corresponds to yearly through-wall cracking frequency of 5×10^{-6} . A mean transition temperature of 201°F corresponds to a transition temperature of 270°F computed following RG 1.99, Rev. 2 because the RG 1.99, Rev. 2 temperature includes a margin term. The figure on page six of the handouts represented the distribution of plants that were close to the current screening criteria. Mr. Kirk stated that plants get closer to the RT_{PTS} limit by about one degree Fahrenheit per year of operation.

The staff selected Calvert Cliff, Oconee, Beaver Valley, and Palisades for plant-specific studies. These plants represent each of the major pressurized water reactor (PWR) manufacturers. Two of the plants were projected to be the closest to the current PTS screening criteria limit at EOL.

The staff's estimate of the through-wall cracking frequency starts with an events sequence analysis. This analysis defines both the combination of events (scenario) that can lead to a PTS challenge to the vessel and the frequency of such events. The thermal-hydraulic conditions associated with each scenario are determined using the RELAP Code. These analyses give the temporal variations of pressure, temperature, and heat transfer coefficient acting on the embrittled vessel. Probabilistic fracture mechanics analyses, based on linear elastic fracture mechanics techniques, were performed using the FAVOR Code. These analyses calculate the conditional probabilities with which through-wall cracks will occur. These conditional probabilities are multiplied by the sequence frequencies to obtain an estimate of the yearly through-wall cracking frequency.

The probabilistic fracture mechanics analysis treats the pressure, temperature, and heat transfer coefficient variation with time for each scenario deterministically. FAVOR takes as input the pressure, temperature, and heat transfer coefficient values versus time at the vessel surface, calculates the heat conduction in the vessel, and computes the resulting thermal stresses. The stresses are then used to compute the driving force for fracture. At the same time, FAVOR calculates a distribution of fracture toughness of material, which is dependent upon the temperature, the fluence, and embrittlement characteristics. Comparison of the applied driving force with the toughness distribution gives probability of fracture.

PRA ANALYSIS - Mr. Kolaczowski, SAIC

Mr. Kolaczowski provided an overview of the PRA modeling approach and the plant specific PRA models. He stated that the Oconee PRA model is the most complete one, relative to over cooling scenarios. The model identified one hundred eighty-one thousand two hundred fifty eight over-cooling sequences. The initiating event frequencies and equipment failure data in the model were based on industry generic data. The human reliability analysis (HRA) was initially performed by NRC contractors. The Beaver Valley model was the second one prepared by the staff and was simplified based on results from the Oconee analysis which showed that some scenarios were relatively unimportant from a through-wall crack frequency perspective. Palisades was the last model prepared by the staff. Because the Palisades IPE included PTS scenarios, the staff started with a pre-existing model and modified it. Unlike the other two cases, the licensee was the keeper of the model.

The results of the PRA showed that medium and large LOCAs are bigger contributors to PTS than previously taken into account in the 1980s when developing the original PTS rule. Recent analysis also showed that the thermal stress (or temperature) is more dominant than pressure.

During the meeting, there were considerable discussions between the Subcommittee members and the staff regarding operator action and assigning probability values to them.

Mr. Kolaczowski stated that for some over-cooling scenarios operator actions play a key role, either by mitigating or exacerbating the event. However, during a LOCA, which is the dominant event, operator actions have little impact. Thus, in PTS the uncertainties associated with operator actions have relatively little impact on the overall uncertainty in the vessel failure frequency.

THERMAL HYDRAULIC ANALYSIS - Mr. Bessettee, RES

Mr. Bessettee stated that the staff used RELAP 5/MOD 3.2.2 gamma Code to generate downcomer temperature, system pressure, and heat transfer coefficient at the inside of the vessel wall. These results were then used as input to FAVOR Code. Mr. Bessettee presented a comparison between RELAP predicted temperatures and results of ROSA (Westinghouse) and MIST (Babcock & Wilcox) experiments. Members of the Subcommittee questioned the assessment of thermal hydraulic uncertainties, and their impact on the rates of change in the temperatures feeding into the FAVOR Code and asked the staff to present these results clearly and in more detail in the future.

PROBABILISTIC FRACTURE MECHANICS - Mr. Kirk, RES

The pressure, temperature, and heat transfer coefficient are input to an embrittlement and crack initiation model. Other inputs to the model include flaw distribution and their locations, orientation, material properties, composition, and fluence variations around the vessel. The model then calculates a yearly frequency of through-wall cracking. The flaw distribution data came from a variety of sources. According to Mr. Kirk, most of the big flaws (95 to 98 percent) are in the welds. Inspections have revealed that most of these flaws are fusion line flaws. This observation helps in the determination of the flaw orientation.

Generic Letter 92-01 required all licensees to report fluence level and identify limiting materials in terms of RT_{NDT} , and characterize the embrittlement in terms of RT_{NDT} . In addition, confirmatory experimental data were derived from tests at the Oak Ridge National Laboratory and other locations. The staff has recognized that RT_{NDT} is not a precise representation of toughness changes under irradiation. However, even if better characterization of embrittlement were available for all materials of interest, there would still be aleatory uncertainty in the toughness. They developed a model describing both the epistemic and aleatory uncertainties in RT_{NDT} and the aleatory nature of toughness, for both crack arrest and crack initiation. Dr. Wallis noted that the discussion of these uncertainties needed better clarification in the report.

Dr. Wallis asked about the effects of transients and flaw distributions in the stainless steel liner. Mr. Kirk responded that residual stress distribution due to the weld overlay and stresses caused by the differential thermal expansion of the stainless steel relative to the ferritic steel were incorporated in the analysis.

PLANT SPECIFIC STUDIES - Mr. Kirk, RES

Mr. Kirk stated that overall LOCAs are the dominant contributors to PTS failures in PWRs. There is at least three orders of magnitude uncertainty in through-wall cracking frequencies. Two thirds of the contribution come from the uncertainty in the LOCA frequencies and the remaining from uncertainties in the flaw distributions. The distributions are highly skewed and the mean and 95th percentiles are almost equal. Operator action does not play a significant role during most LOCAs because there is little an operator can do in response to it. However, for B&W plants operator action plays a more critical role in response to stuck open primary side valve scenarios. From a materials perspective, the axial weld cracks and weld toughness or the plate properties dominate the RT_{NDT} .

Dr. Ford questioned how could the results of this draft NUREG be applicable to all PWRs based on analysis of only three plants. Mr. Kirk responded that these plants were selected and ranked in terms of irradiation susceptibility and that because the challenge were dominated by LOCA events there is a high degree of consistency in operational challenge among plants.

ACCEPTANCE CRITERIA - Mr. Siu, RES

Mr. Siu described the reactor vessel failure frequency acceptance criteria development process. The strategy for developing the criteria was to be consistent with the original intent of the PTS rule by keeping the risk level low and keeping the relative contribution of PTS risk small compared to the risks associated with other sources. The staff believes that the reactor vessel failure frequency (RVFF) should be defined in terms of through-wall crack frequency rather than the frequency of crack initiation.

The key question was whether there is a margin between the occurrence of a through-wall crack and core damage and large early release associated with a PTS scenario. The staff urged that the challenge to the containment of PTS events is not exceptionally severe as compared to other accident scenarios. The important factor is the relatively low coolant temperature during a PTS events.

Mr. Bessette described several PTS transient scenarios. One scenario starts with a medium size LOCA, followed by a vessel failure in 1000 seconds. The FAVOR calculations and results of the pilot studies showed that the containment failure is unlikely and independent of a PTS event. Other scenarios also show that, overall, there is adequate margin between the occurrence of a PTS induced reactor vessel failure and large early release. For example, the reaction forces resulting from a vessel break are not worse than those analyzed for a cold leg break.

Mr. Siu concluded that the containment pressurization is likely to be less than a design basis LOCA and that choosing reactor vessel failure frequency criterion to be 10^{-6} would be consistent with the intent of the original PTS rule.

PTS SCREENING LIMIT - Mr. Kirk, RES

Mr. Kirk stated that the severity of PTS challenges is remarkably similar among the plants studied, and, the frequency of challenge is also fairly similar but with some greater plant dependencies. From a materials viewpoint, axial weld material and flaws dominate the through-wall cracking frequency and establish the relationship between the embrittlement metric and through-wall cracking frequency. Mr. Kirk described the RT_{NDT} screening criteria graph which gives the relationship between the RT_{NDT} and mean through-wall cracking frequency. The horizontal axis is the ASME RT_{NDT} plus a shift due to irradiation calculated from the Eason formula. The vertical axis is derived from FAVOR calculations and incorporates all the complexities of uncertainties in material properties, thermal-hydraulics, and event frequency.

Using the graph, and taking the reactor vessel failure frequency criterion of 10^{-6} , the resulting screening limit RT_{NDT} comes out to be 290°F. However, RT_{NDT} is not the same as RT_{PTS} . Calculated by Regulatory Guide 1.99, Rev. 2, RT_{NDT} is about 90°F less than RT_{PTS} . This suggests that a 80°F to 110°F increase of the current screening limit is possible.

CONCLUSIONS AND RECOMMENDATIONS

The staff stated that the purpose of this analysis was to show that a PTS event was unlikely and therefore the NRC could raise the criteria to allow the plants to run for a longer time. RES has forwarded the draft NUREG to the Office of Nuclear Reactor Regulation and believes that it can support revising the PTS rule. The staff highlighted that work on this analysis was still ongoing. The Subcommittee noted that the analysis and its conclusions apply to all PWRs. They also commented that the analysis and the draft report needed additional work and strongly recommended a peer review. The staff agreed that the draft report was not final and additional work was necessary. They also committed to present the information in plain language and clearer. The Subcommittee encouraged the staff to proceed with the rulemaking.

STAFF COMMITMENTS

1. The staff committed to perform a thermal hydraulic uncertainty analysis and evaluate the temperature distribution in the downcomer region.
2. The staff committed to perform additional FAVOR runs in terms of sensitivity studies.

4. The staff committed to revise the draft NUREG and perform a comprehensive peer review.

SUBCOMMITTEE DECISION

The Subcommittee decided prepare a letter regarding this matter and submit to the full Committee for consideration. The staff will brief the full Committee at the February 2003 ACRS meeting.

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 and as attachments to the transcript which will be made available in ADAMS.

BACKGROUND MATERIAL PROVIDED TO THE SUBCOMMITTEE

1. "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Criteria in the PTS Rule" (10 CFR 50.61), December 31, 2002
2. Advisory Committee on Reactor Safeguards (ACRS) Letter to William Travers, "Reevaluation of the Technical Basis for the Pressurized Thermal Shock Rule", February 14, 2002.
3. ACRS Letter to William Travers, "Risk Metrics and Criteria for Reevaluation the Technical Basis Of the Pressurized Shock Rule", July 18, 2002.
4. William Travers letter to ACRS, "Risk Metrics and Criteria for Reevaluation the Technical Basis Of the Pressurized Shock Rule", September 3, 2002.

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 view on the Internet at <http://www.nrc.gov/reading-rm/doc-collections/acrs/> can be purchased from Neal R. Gross and Co., 1323 Rhode Island Avenue, NW, Washington, D.C. 20005, (202) 234-4433 (voice), (202) 387-7330 (fax), nrgross@nealgross.com (e-mail).

**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
MATERIALS AND METALLURGY SUBCOMMITTEE MEETING
REEVALUATING THE TECHNICAL BASIS OF THE
PRESSURIZED THERMAL SHOCK (PTS) RULE
FEBRUARY 5, 2003, ROCKVILLE, MARYLAND**

Contact: Richard Savio (301-415-7363, rps1@nrc.gov)
Ramin Assa (301-415-6885, rra@nrc.gov)

-PROPOSED SCHEDULE-

Topics	Presenters	Time
I. Opening Remarks	W. Shack, ACRS	8:30-8:35 a.m.
II. PTS Re-evaluation Project Introduction	M. Mayfield, RES	8:35-8:50 a.m.
III. PTS Project Overview, Background	→ Nathan Siu M. Kirk	8:50-9:35 a.m.
Significance of RELAP differences w/ experiments (assessment results)	D. Bessette	9:35-9:55 a.m.
BREAK		10:05 9:55-10:10 ²⁰ a.m.
IV. Plant Specific Results (Oconee-1, Beaver Valley-1, Palisades) Thermal-Hydraulic characteristics of dominant transients. Uncertainty results.	M. Kirk D. Bessette R. Woods	10:10-12:00 a.m. 12:25 pm
LUNCH		12:00-1:00 p.m. ²⁵
Plant Specific Results (Continued), Applicability Beyond the study plants Generalization and external events	A. Kolaczowski, SAIC D. Whitehead, SNL M. Kirk R. Woods	1:00-2:10 ¹³⁰
V. Risk-Informed Reactor Vessel Failure Frequency Acceptance Criteria Post PTS Vessel Failure considerations (including addressing comments of ACRS) Results of T-H analyses	N. Siu D. Bessette	2:10-3:10 ²⁰
BREAK		3:10-3:25 ²⁰ p.m.
VI. PTS RT _{NDT} based screening limit	M. Kirk	3:25-3:55
VII. Overall summary and conclusions	M. Kirk, E. Hackett	3:55-4:55 ³⁰
VIII. Subcommittee discussion		4:55- 5:10 ^{30 45} p.m.
IX. Adjourn		5:10 ⁴⁵ p.m.

Combined. ↘

NOTE:

- Presentation time should not exceed 50 percent of the total time allocated for specific item. The remaining 50 percent of the time is reserved for discussion.
- 25 copies of the presentation materials to be provided to the Subcommittee

**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON MATERIALS AND METALLURGY**

FEBRUARY 5, 2003

Today's Date

NRC STAFF PLEASE SIGN IN BELOW

PLEASE PRINT

NAME

NRC ORGANIZATION

NATHAN SIM

RES / DRAA / PRAB

SHAH MALIK

RES / DET / MEB

Roy Woods

RES / DRAA / PRAB

Jack Rosenthal

RES / DSARE / SMSAB

Matthew A. Mitchell

NRR / DE / EMCB

Sud Basu

RES / DSARE / SMSAB

Lance Kim

RES / DRAA / PRAB

Kent Welter

RES / DSARE / SMSAB

ED HACKETT

RES / DET / MEB

MARIC KIRK

RES / DET / MEB

"Auxiliary Feedwater System," to better reflect the four train auxiliary feedwater (AFW) system design at STP.

Specifically, the changes specify the same allowed outage time (AOT) for any one inoperable motor-driven pump, regardless of train. The amendments also extend the AOT for one inoperable motor-driven pump from 72 hours to 28 days. A sentence has also been added to Action d. stating that Limiting Condition for Operation (LCO) 3.0.3 and all other LCO actions requiring Mode changes are suspended until one of the four inoperable AFW pumps is restored to operable status. There is also an administrative change in the wording of the LCO to clarify that there are only four AFW pumps in each STP unit.

Date of issuance: December 31, 2002.

Effective date: December 31, 2002.

Amendment Nos.: Unit 1—146; Unit 2—134.

Facility Operating License Nos. NPF-76 and NPF-80: The amendments revised the Technical Specifications.

Date of initial notice in Federal

Register: January 22, 2002 (67 FR 2930). The supplement provided additional information that clarified the application, did not expand the scope as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 31, 2002.

No significant hazards consideration comments received: No.

TXU Generation Company LP, Docket Nos. 50-445 and 50-446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: April 8, 2002.

Brief description of amendments: The amendments revise Technical Specification (TS) 3.4.16, "RCS [Reactor Coolant System] Specific Activity," to lower the Limiting Condition For Operation and associated Surveillance Requirements for Dose Equivalent Iodine-131 in the RCS from a specific activity of 1.0 $\mu\text{Ci/gm}$ to 0.45 $\mu\text{Ci/gm}$.

Date of issuance: January 6, 2003.

Effective date: As of the date of issuance and shall be implemented within 60 days from the date of issuance.

Amendment Nos.: 102 and 102.

Facility Operating License Nos. NPF-87 and NPF-89: The amendments revised the Technical Specifications.

Date of initial notice in Federal

Register: June 11, 2002 (67 FR 40026). The Commission's related evaluation of

the amendments is contained in a Safety Evaluation dated January 6, 2003.

No significant hazards consideration comments received: No.

Virginia Electric and Power Company, Docket No. 50-338, North Anna Power Station, Unit 1, Louisa County, Virginia

Date of application for amendment: December 7, 2001, as supplemented by letters dated June 28 and July 25, 2002.

Brief description of amendment: This amendment permits a one-time extension of the current 10-year Title 10 of the Code of Federal Regulations Part 50, Appendix J, Option B, Type A test interval from April 3, 2003, to April 2, 2008.

Date of issuance: December 31, 2002.

Effective date: As of the date of issuance and shall be implemented within 30 days from the date of issuance.

Amendment No.: 234.

Facility Operating License No. NPF-4: Amendment changes the Technical Specifications.

Date of initial notice in Federal

Register: April 30, 2002 (67 FR 21295). The supplemental letters dated June 28 and July 25, 2002, contained clarifying information only and did not change the proposed no significant hazards consideration determination or expand the scope of the initial application.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated December 31, 2002.

No significant hazards consideration comments received: No.

For the Nuclear Regulatory Commission.

Dated at Rockville, Maryland, this 13th day of January 2003.

John A. Zwolinski,

Director, Division of Licensing Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 03-1161 Filed 1-17-03; 8:45 am]

BILLING CODE 7590-01-P

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards Subcommittee Meeting on Planning and Procedures; Notice of Meeting

The ACRS Subcommittee on Planning and Procedures will hold a meeting on February 5, 2003, Room T-2B1, 11545 Rockville Pike, Rockville, Maryland.

The entire meeting will be open to public attendance, with the exception of a portion that may be closed pursuant to 5 U.S.C. 552b(c) (2) and (6) to discuss organizational and personnel matters

that relate solely to internal personnel rules and practices of ACRS, and information the release of which would constitute a clearly unwarranted invasion of personal privacy.

The agenda for the subject meeting shall be as follows:

Wednesday, February 5, 2003—1 p.m. until the conclusion of business

The Subcommittee will discuss proposed ACRS activities and related matters. 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. Persons desiring to make oral statements should notify the Designated Federal Official named below five days prior to the meeting, if possible, so that appropriate arrangements can be made. Electronic recordings will be permitted only during those portions of the meeting that are open to the public.

Further information regarding topics to be discussed, the scheduling of sessions open to the public, 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 Designated Federal Official, Mr. Sam Duraiswamy (telephone: 301/415-7364) between 7:30 a.m. and 4:15 p.m. (EST). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the proposed agenda.

Dated: January 13, 2003.

Sher Bahadur,

Associate Director for Technical Support, ACRS/ACNW.

[FR Doc. 03-1221 Filed 1-17-03; 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 Subcommittees on Materials and Metallurgy will hold a meeting on February 5, 2003, 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:

Wednesday, February 5, 2003—8:30 a.m. until the conclusion of business

The Subcommittee will meet with representatives of the NRC staff and discuss the risk metric and criteria that can be used for reevaluating the technical basis of the pressurized thermal shock (PTS) rule and the NRC staff's pilot plant studies. 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. Persons desiring to make oral statements should notify the Designated Federal Official 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 Designated Federal Official, Dr. Richard P. Savio (telephone 301-415-7363) between 7:30 a.m. and 5 p.m. (EST). Persons planning to attend this meeting are urged to contact the above named individual at least two working days prior to the meeting to be advised of any potential changes in the proposed agenda.

Dated: January 14, 2003.

Sher Bahadur,

*Associate Director for Technical Support,
ACRS/ACNW.*

[FR Doc. 03-1222 Filed 1-17-03; 8:45 am]

BILLING CODE 7590-01-P

NUCLEAR REGULATORY COMMISSION

Peer Review Committee for Source Term Modeling; Notice of Meeting

The Peer Review Committee For Source Term Modeling will hold a closed meeting on January 28-29, 2003 at Sandia National Laboratories (SNL), Albuquerque, NM.

The entire meeting will be closed to public attendance to protect information classified as national security information pursuant to 5 U.S.C. 552b(c)(1) and as proprietary pursuant to 5 U.S.C. 552b(c)(4).

The agenda for the subject meeting shall be as follows:

Wednesday, January 28 and Thursday, January 29, 2003—8:30 a.m. until the conclusion of business

The Committee will review SNL activities and aid SNL in development of guidance documents on source terms that will assist the NRC in evaluations of the impact of specific terrorist activities targeted at a range of spent fuel storage casks and radioactive material transport packages including those for spent fuel.

Further information contact: Dr. Andrew L. Bates (telephone 301-415-1963) or Dr. Charles G. Interrante (telephone 301-415-3967) between 7:30 a.m. and 4:15 p.m. (EDT).

Dated: January 14, 2003.

Andrew L. Bates,

Advisory Committee Management Officer.

[FR Doc. 03-1220 Filed 1-17-03; 8:45 am]

BILLING CODE 7590-01-P

SECURITIES AND EXCHANGE COMMISSION

Sunshine Act Meeting

Notice is hereby given, pursuant to the provisions of the Government in the Sunshine Act, Pub. L. 94-409, that the Securities and Exchange Commission will hold the following meetings during the week of January 20, 2003. An Open Meeting will be held on Wednesday, January 22, 2003, at 10 a.m., in Room 1C30, the William O. Douglas Room, and a Closed Meeting will be held on Thursday, January 23, 2003, at 10 a.m.

Commissioners, Counsel to the Commissioners, the Secretary to the Commission, and recording secretaries will attend the Closed Meeting. Certain staff members who have an interest in the matters may also be present.

The General Counsel of the Commission, or his designee, has certified that, in his opinion, one or more of the exemptions set forth in 5

U.S.C. 552b(c)(3), (5), (7), (9)(B) and (10) and 17 CFR 200.402(a)(3), (5), (7), (9)(ii) and (10), permit consideration of the scheduled matters at the Closed Meeting.

The subject matter of the Open Meeting scheduled for Wednesday, January 22, 2003 will be:

1. The Commission will consider whether to adopt new rules 30a-3 and 30d-1 and amendments to rules 8b-15, 30a-1, 30a-2, 30b1-1, 30b1-3, and 30b2-1 under the Investment Company Act of 1940, amendments to rules 12b-25, 13a-15, and 15d-15 and Form 12b-25 under the Securities Exchange Act of 1934, amendments to Form N-SAR under the Exchange Act and the Investment Company Act and the new Form N-CSR under the Exchange Act and Investment Company Act. These new rules and form, and rule and form amendments, would require registered management investment companies to file certified shareholder reports on new Form N-CSR with the Commission, and would designate these certified shareholder reports as reports that are required under sections 13(a) and 15(d) of the Exchange Act and Section 30 of the Investment Company Act. A registered management investment company's principal executive and financial officers would be required to certify the information contained in its reports on Form N-CSR in the manner specified by Section 302 of the Sarbanes-Oxley Act of 2002. The amendments would also remove the requirement that Form N-SAR be certified by a registered investment company's principal executive and financial officers, and would provide that, for registered management investment companies, Form N-SAR would be filed under the Investment Company Act only. In addition, the amendments would implement Sections 406 and 407 of the Sarbanes-Oxley Act by requiring a registered management investment company to provide disclosure on Form N-CSR or Form N-SAR, as applicable, regarding whether the investment company has adopted a code of ethics for the company's principal executive officer and senior financial officers, and whether the investment company has at least one "audit committee expert" serving on its audit committee, and if so, the name of the expert and whether the expert is independent of management.

2. The Commission will consider adopting rules to establish standards of professional conduct for attorneys who appear and practice before the Commission in any way in the representation of issuers. As proposed, the rules would require an attorney to report evidence of a material violation of securities laws, a material breach of fiduciary duty, or similar material violation by the issuer or by any officer, director, employee, or agent of the issuer to the issuer's chief legal officer or the chief executive officer of the company (or the equivalents); if they do not respond appropriately to the evidence, the rule would require the attorney to report the evidence to the issuer's audit committee, another committee of independent directors, or the full board of directors; if the directors do not respond appropriately, the rule would

**ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
SUBCOMMITTEE MEETING ON MATERIALS AND METALLURGY**

FEBRUARY 5, 2003

Today's Date

ATTENDEES PLEASE SIGN IN BELOW

PLEASE PRINT

NAME

AFFILIATION

STAN ROSINSKI

EPRI

Steve Byrne

Westinghouse

Bill Server

ATI CONSULTING

ALAN KOLACZKOWSKI

SATC

Donnie Whitehead

Sandia National Labs.

Yung-Hsien Chang

UMD

W Arcene

ISL

Robert Beaton

ISL

DON FLETCHER

ISL

Terry Dickson

ORNL

Bob Hardies

Constellation Energy

Ken Yoon

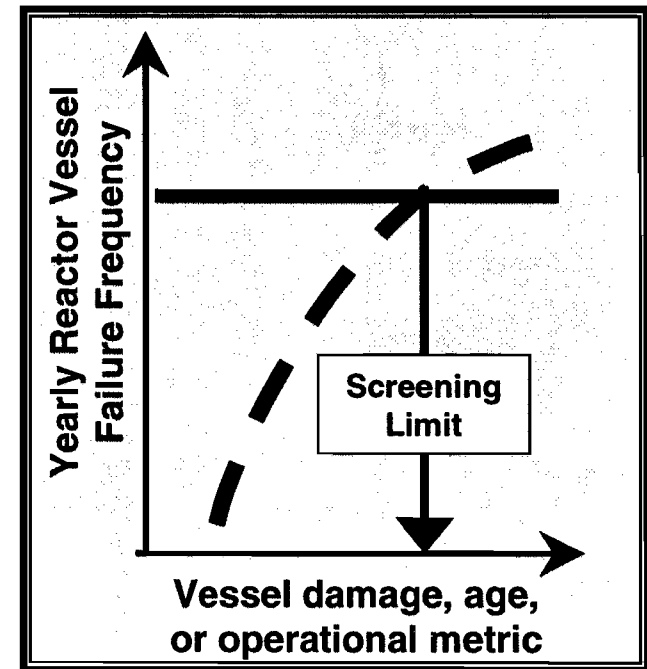
Framatome ANP

Ron Gamble

Sartrex Corp.

Reactor Vessel Failure Frequency (RVFF)

- **RVFF criterion needed for two purposes:**
 - **Support definition of RPV embrittlement criterion**
 - **Provide acceptance criterion for safety analysis**
- **Current metric and criterion established in RG 1.154:**
$$\text{RVFF} \equiv \text{TWCF}$$
$$\text{RVFF}^* = 5 \times 10^{-6}/\text{ry}$$
- **Limited scope activity to revisit metric/criterion in light of recent risk-informed regulation initiatives**



Task Activities

- **Identification of options**
- **Scoping study of post-vessel failure accident progression**
 - Qualitative evaluation of technical issues
 - Review of pilot plant calculations for T/H conditions
 - Limited calculations
- **Status reports and meetings**
 - SECY-02-0092 (5/10/02)
 - ACRS (7/10/02), public meetings (10/17/02; 1/31/03)
 - Chapter 5, draft NUREG (12/31/02)
- **Focus on acceptability => activities are largely independent of plant-specific studies**

RVFF Acceptance Criteria

Principles in Developing and Evaluating Options

- **Consistency with intent of original rule**
 - **Low risk level**
 - **Low relative contribution**

- **Consistency with recent risk-informed initiatives**
 - **Risk metrics**
 - **Risk criteria**
 - **Consideration of defense-in-depth**

RVFF Acceptance Criteria

Options (SECY-02-0092)

■ **Definition of RVFF**

- **RVFF = f(PTS-induced RPV through-wall crack)**
- **RVFF = f(PTS-induced crack initiation)**

■ **RVFF acceptance limits**

- **RVFF* = $5 \times 10^{-6}/ry$**
- **RVFF* = $1 \times 10^{-5}/ry$**
- **RVFF* = $1 \times 10^{-6}/ry$**

Post-SECY Discussions

- **Budgeting process: focus effort on assessing RVFF for pilot plants**
 - **ACRS Letter (7/18/02; ML0220406120)**
 - **RVFF should be based on considerations of LERF (and not CDF)**
 - **Current LERF surrogate goal is not proper starting point**
- "...source terms used to develop the current goal do not reflect the air-oxidation phenomena that would be a likely outcome of a PTS event."**
- **Options:**
 - ✓ **Develop acceptance criterion from prompt fatality safety goal**
 - ✓ **Use a frequency-based approach to develop RVFF* to provide assurance that PTS-induced RPV failures are very unlikely**
 - **ACRS' expectation: RVFF* will be substantially smaller than options proposed in SECY-02-0092**

Definition of RVFF

It is appropriate to define RVFF as the frequency of through-wall cracks (TWCF)

- **TWCF is a more direct indicator of risk than is the vessel cracking initiation frequency**
- **The current technology for predicting crack arrest is reasonably robust**
 - **Laboratory-scale experiments**
 - **Scaled-vessel experiments**

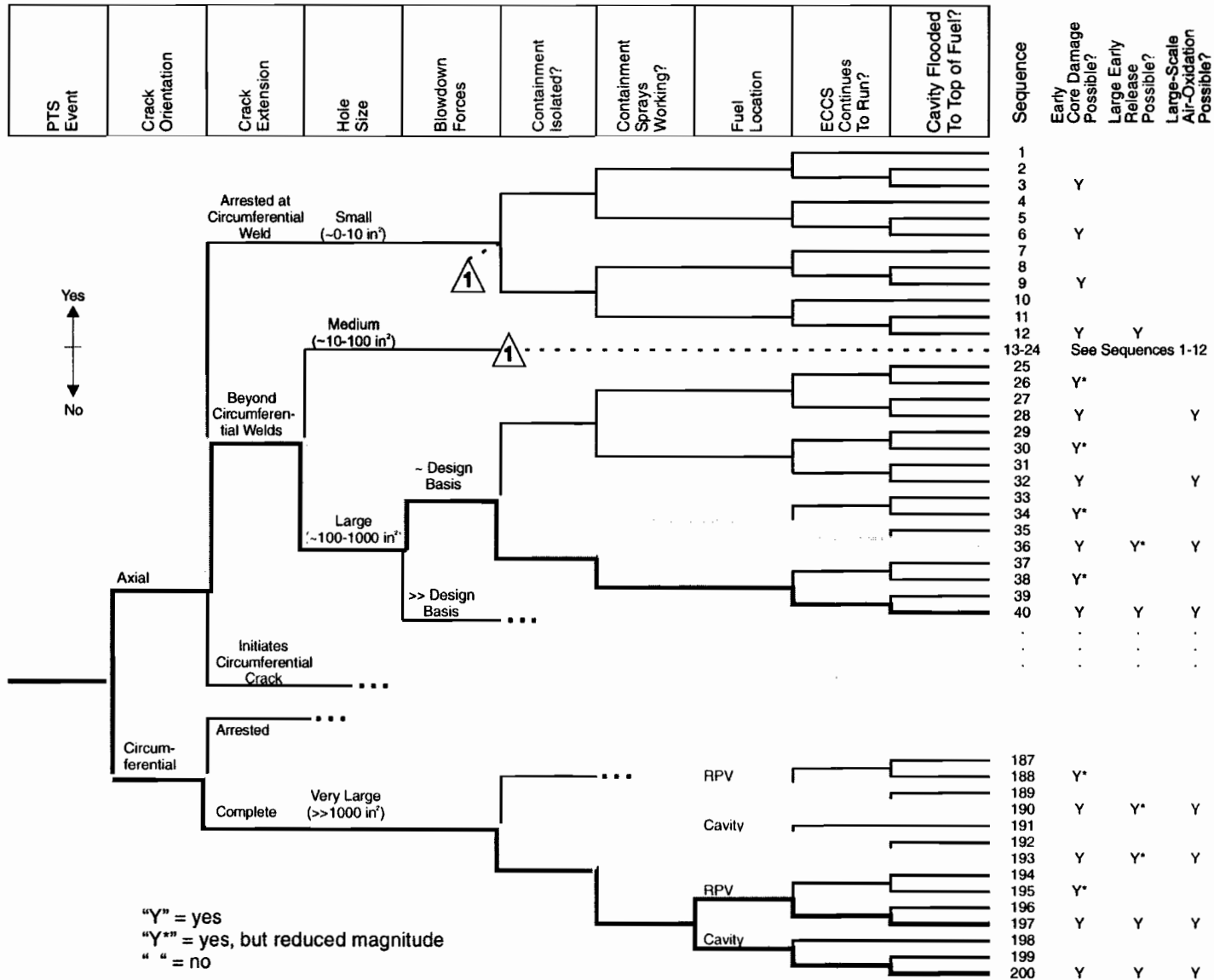
Scoping Study - Key Questions

- **Is a PTS-induced RPV failure likely to lead to melted fuel?**
- **Is a PTS-induced RPV failure likely to lead to a large, early release?**
- **Is the release spectrum (frequency-consequence) for PTS-induced large, early releases significantly worse than that associated with risk-significant, non-PTS-induced scenarios?**

Scoping Study - Approach

- **Refine SECY-02-0092 list of technical issues**
- **Develop accident progression event tree (APET) to support identification, representation and discussion of technical issues**
- **Evaluate current state of knowledge regarding technical issues**
- **Context for evaluations:**
 - **Focus on pilot plants; some consideration of plants addressed in generalization task**
 - **Whether/how PTS changes accident progression**

Accident Progression Event Tree (APET)



PTS Event	Crack Orientation	Crack Extension	Hole Blow Size	Blow Force	Containment Isolated	Sprays Working	Fuel Location	ECCS Runs	Cavity Flooded?							
Yes ↑ No ↓	Axial	Arrested at Circumferential Weld	Small (~0-10 in ²)	1	1											
		Beyond Circumferential Welds	Medium (~10-100 in ²)													
		Circumferential	Initiates Circumferential Crack	Arrested	Large (~100-1000 in ²)					~ Design Basis					
												Complete	Very Large (>>1000 in ²)			
														Cavity	RPV	
																Cavity

"Y" = yes
 "Y*" = yes, but reduced magnitude
 " " = no

Sequence	Early Core Damage Possible?	Large Early Release Possible?	Large-Scale Air-Oxidation Possible?
1			
2			
3	Y		
4			
5			
6	Y		
7			
8			
9	Y		
10			
11			
12	Y	Y	
13-24	See Sequences 1-12		
25			
26	Y*		
27			
28	Y		Y
29			
30	Y*		
31			
32	Y		Y
33			
34	Y*		
35			
36	Y	Y*	Y
37			
38	Y*		
39			
40	Y	Y	Y
...
...
187			
188	Y*		
189			
190	Y	Y*	Y
191			
192			
193	Y	Y*	Y
194			
195	Y*		
196			
197	Y	Y	Y
198			
199			
200	Y	Y	Y

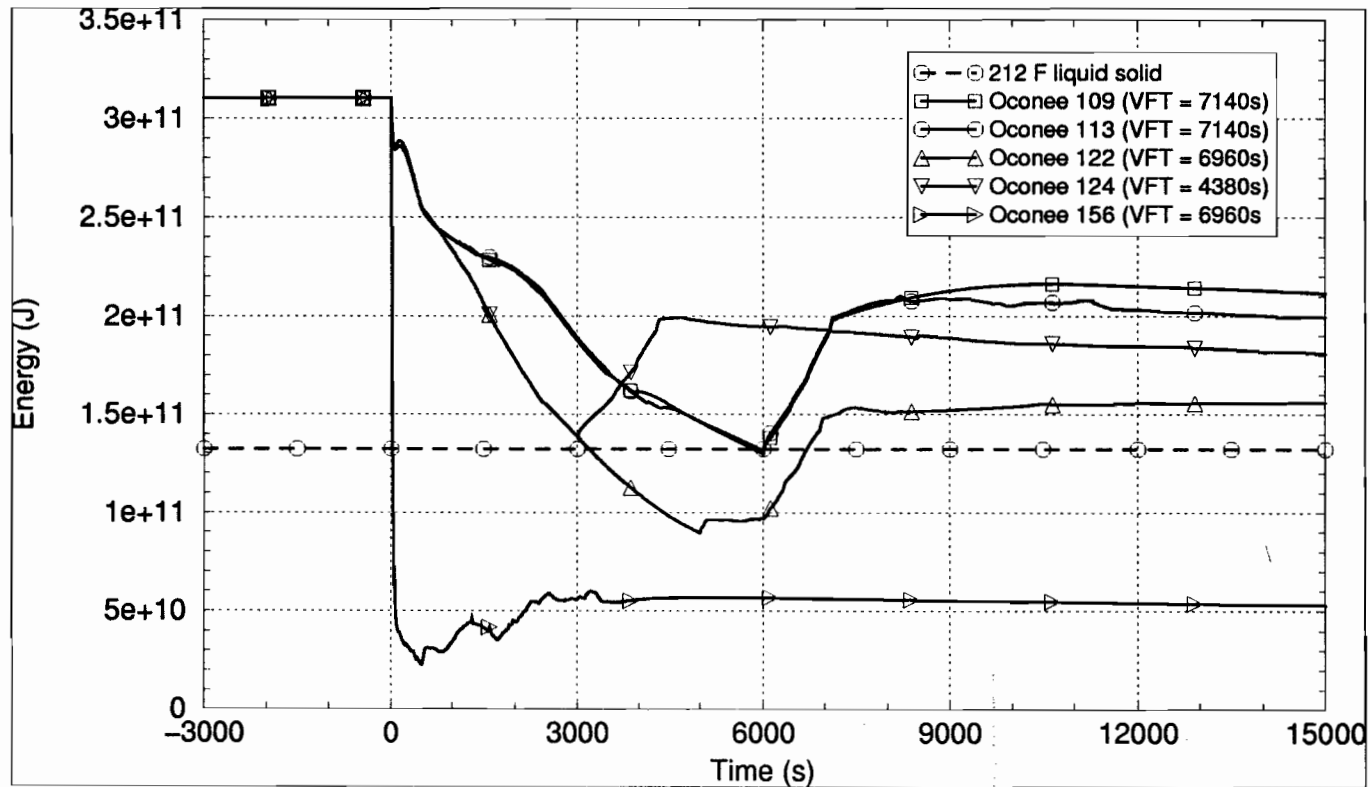
Potential Sources of Dependence Between Top Events

- **Plant systems**
- **RPV movement**
- **Fragments**
- **Fuel movement**

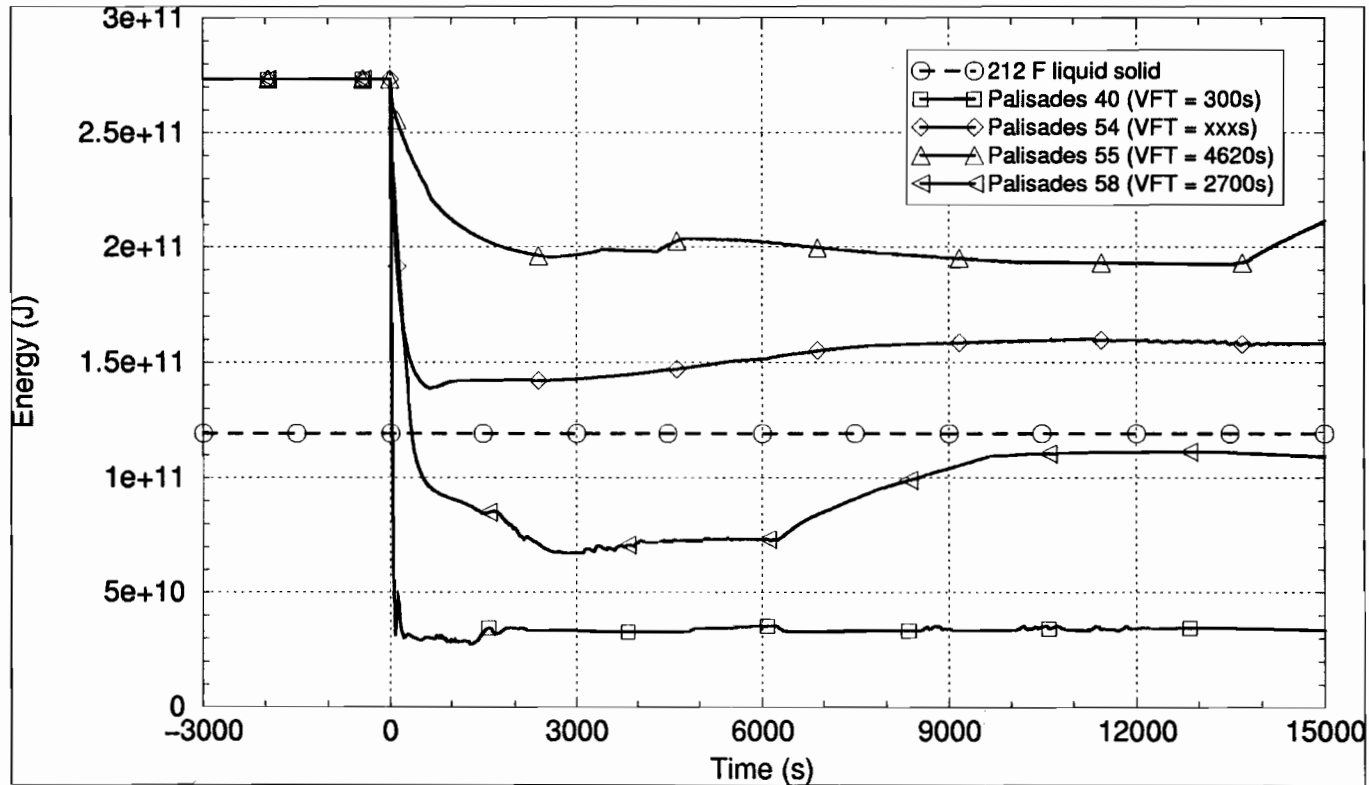
Plant Conditions at RPV Failure

- **Power available, cooling systems running (injection mode)**
- **LOCA events: RCS cooling, depressurizing**
 - **MLOCA - RPV failure at ~15-30 min (40 EFPY)**
 - **LLOCA - RPV failure at ~5-10 min (40 EFPY)**
- **Stuck-open SRV events: RCS at SRV setpoint
RPV failure at ~60-120 min (40 EFPY)**

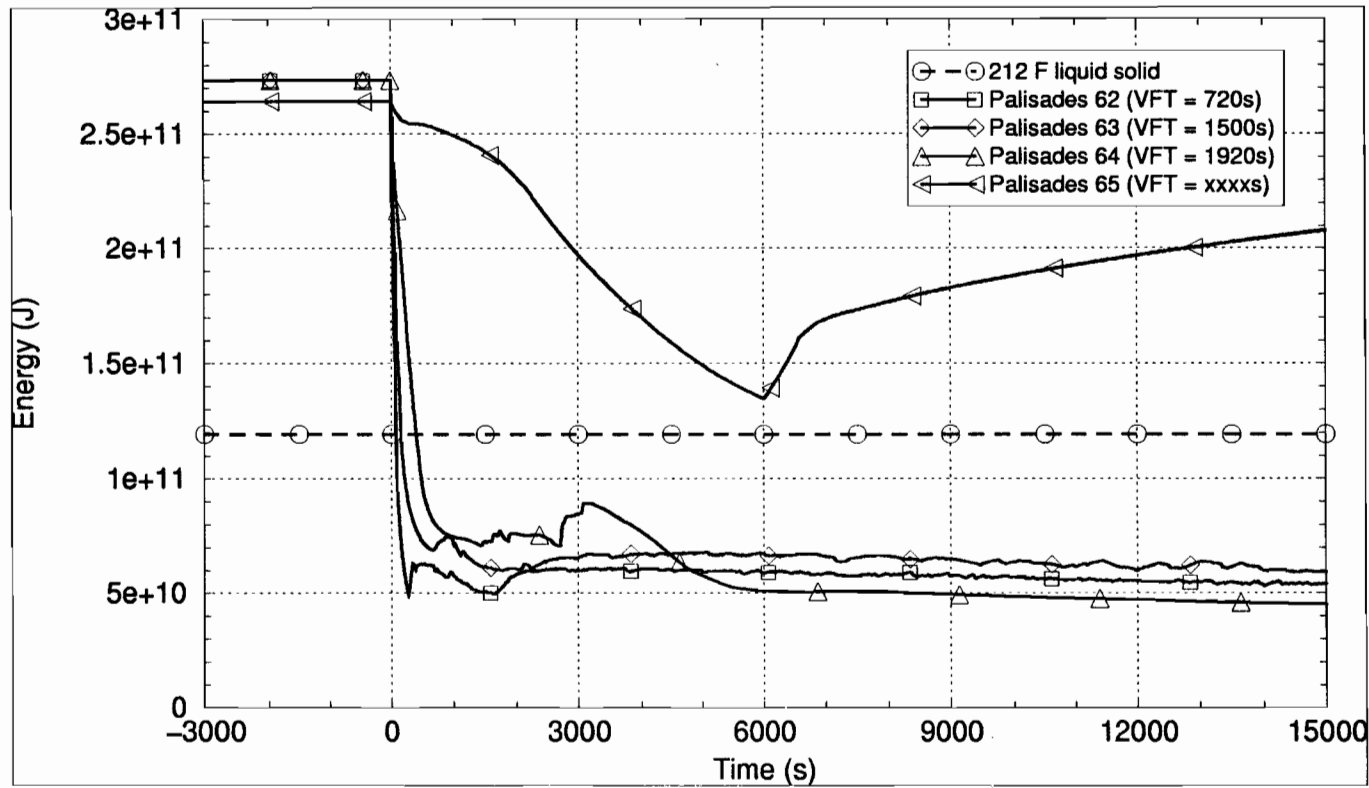
Blowdown Potential After RPV Failure



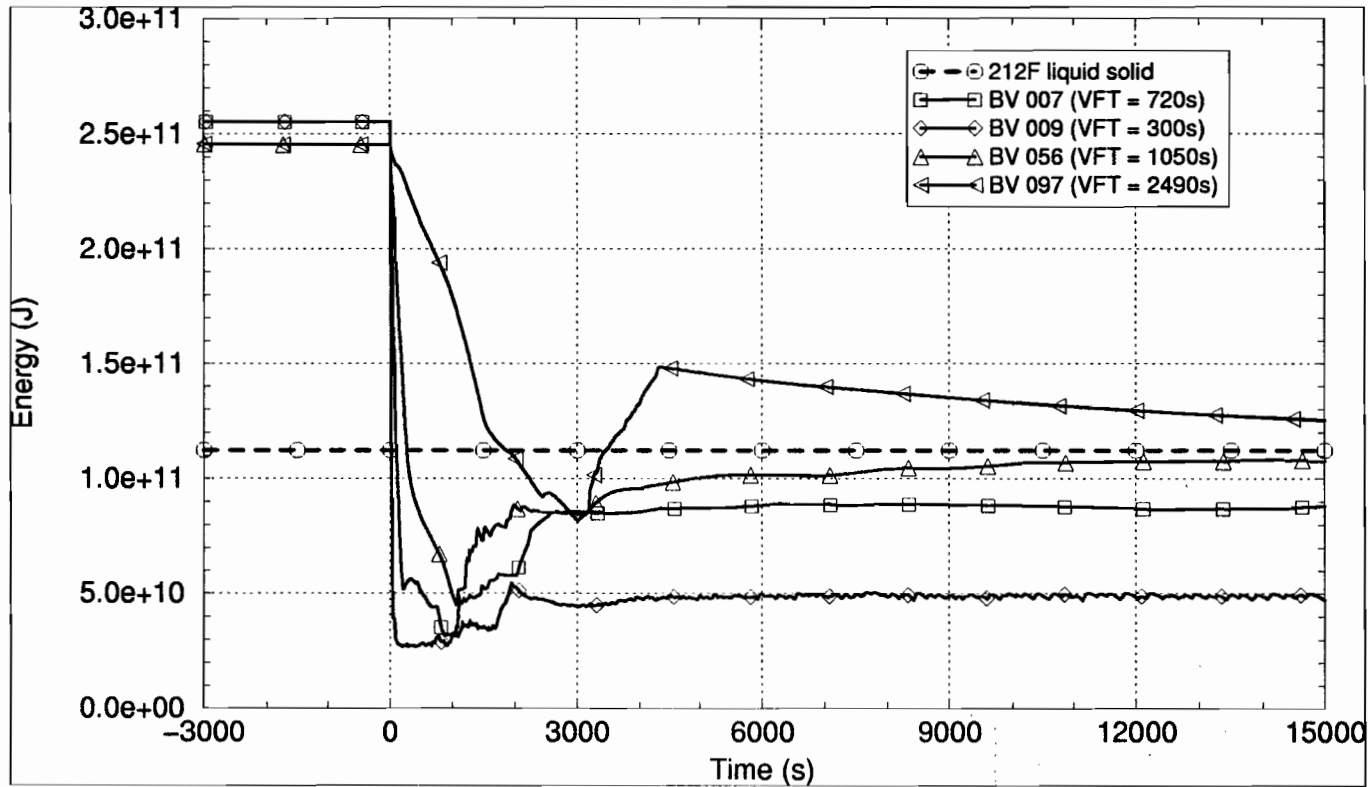
Blowdown Potential After RPV Failure



Blowdown Potential After RPV Failure



Blowdown Potential After RPV Failure



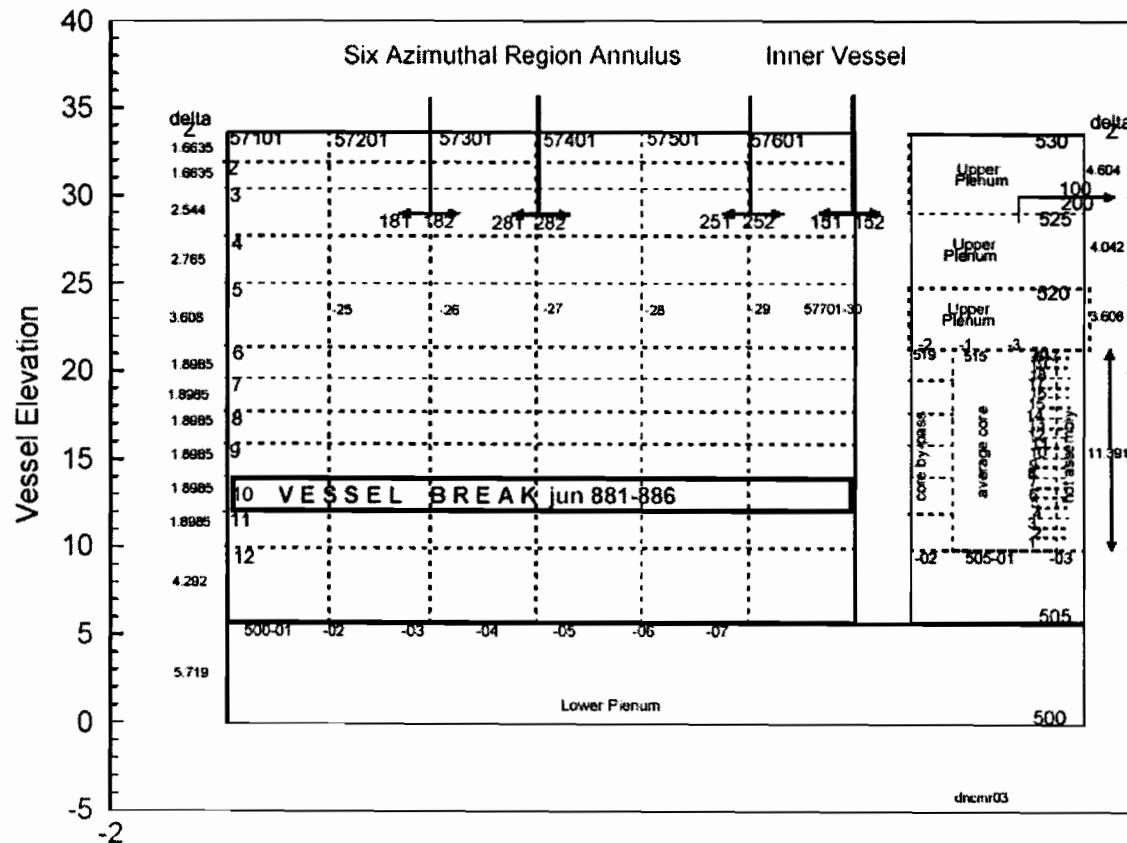
RPV TH Failure Analysis

- **Scoping calculations performed using RELAP5/MOD3.3 of RPV failure for Calvert Cliffs**
- **Two transients analyzed**
 - **4-inch surge line break**
 - **Stuck open pressurizer safety valves (2) that reclose at 6000s**
- **For each transient, two RPV failure modes analyzed**
 - **12 ft² axial break (1 ft x 12 ft)**
 - **360° circumferential break**
- **For each break, three break opening times analyzed**
 - **0.01 s**
 - **0.1 s**
 - **1 s**
- **Results compared to Design Basis LBLOCA**

RPV TH Failure Analysis

Circumferential Break Nodalization

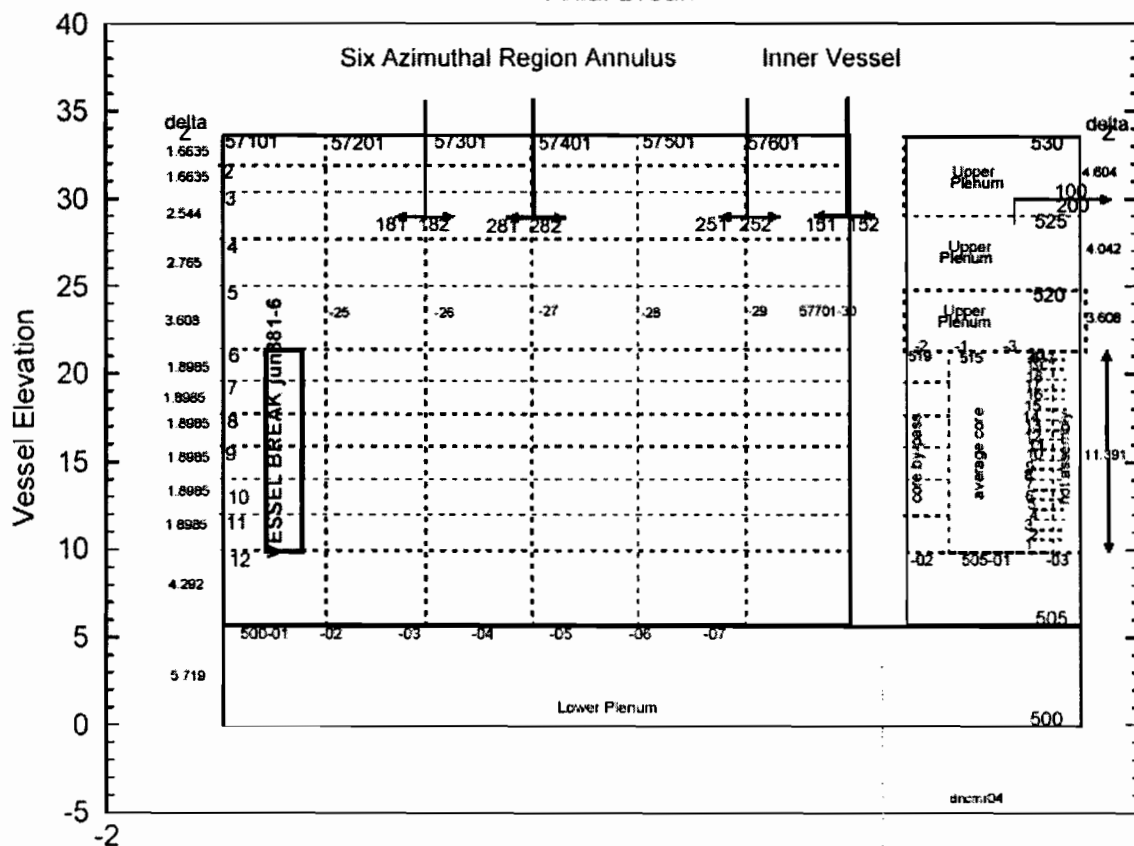
Figure 1. Calvert Cliffs PTS Vessel Noding Diagram
Circumferential Break



RPV TH Failure Analysis

Axial Break Nodalization

Figure 2. Calvert Cliffs PTS Vessel Noding Diagram
Axial Break



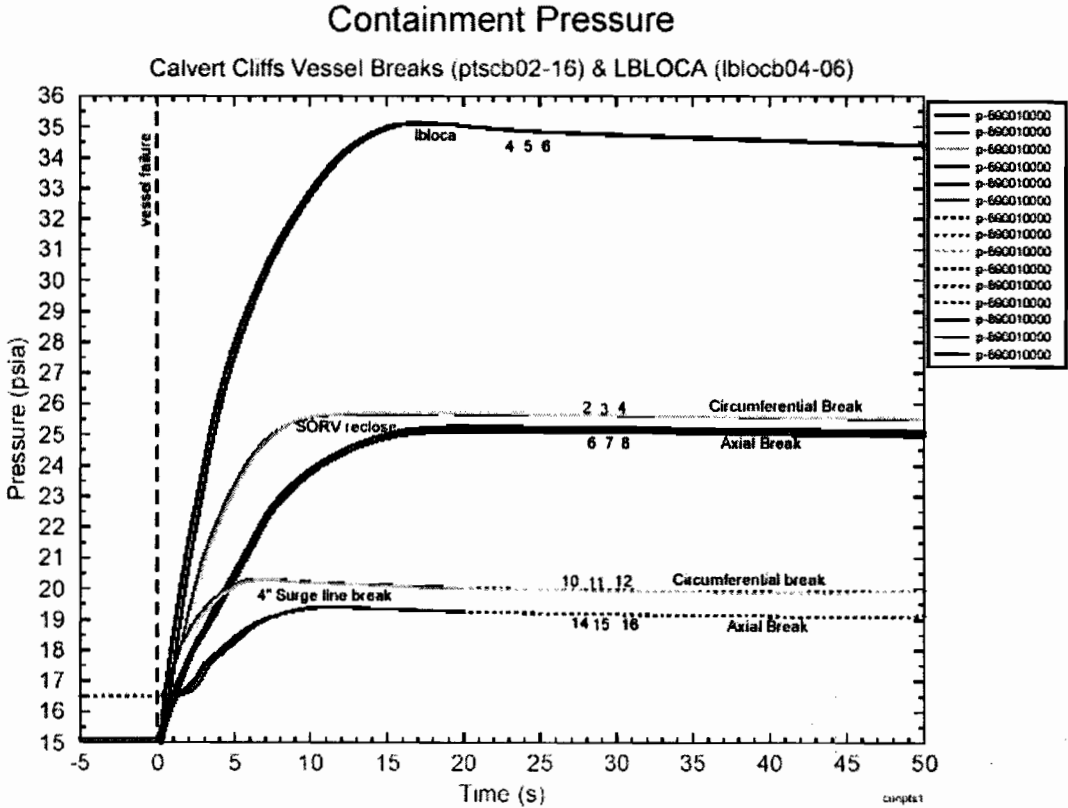
Conditions at RPV Failure

Transient	Break Time (s)	Pressure (psi)	Downcomer Temperature (F)	Specific Enthalpy (Btu/lbm)
4-inch surge line break	2400	200	215 (saturated)	183
Stuck open SRV	8230	2400	355	327
LB LOCA	0	2250	545	543

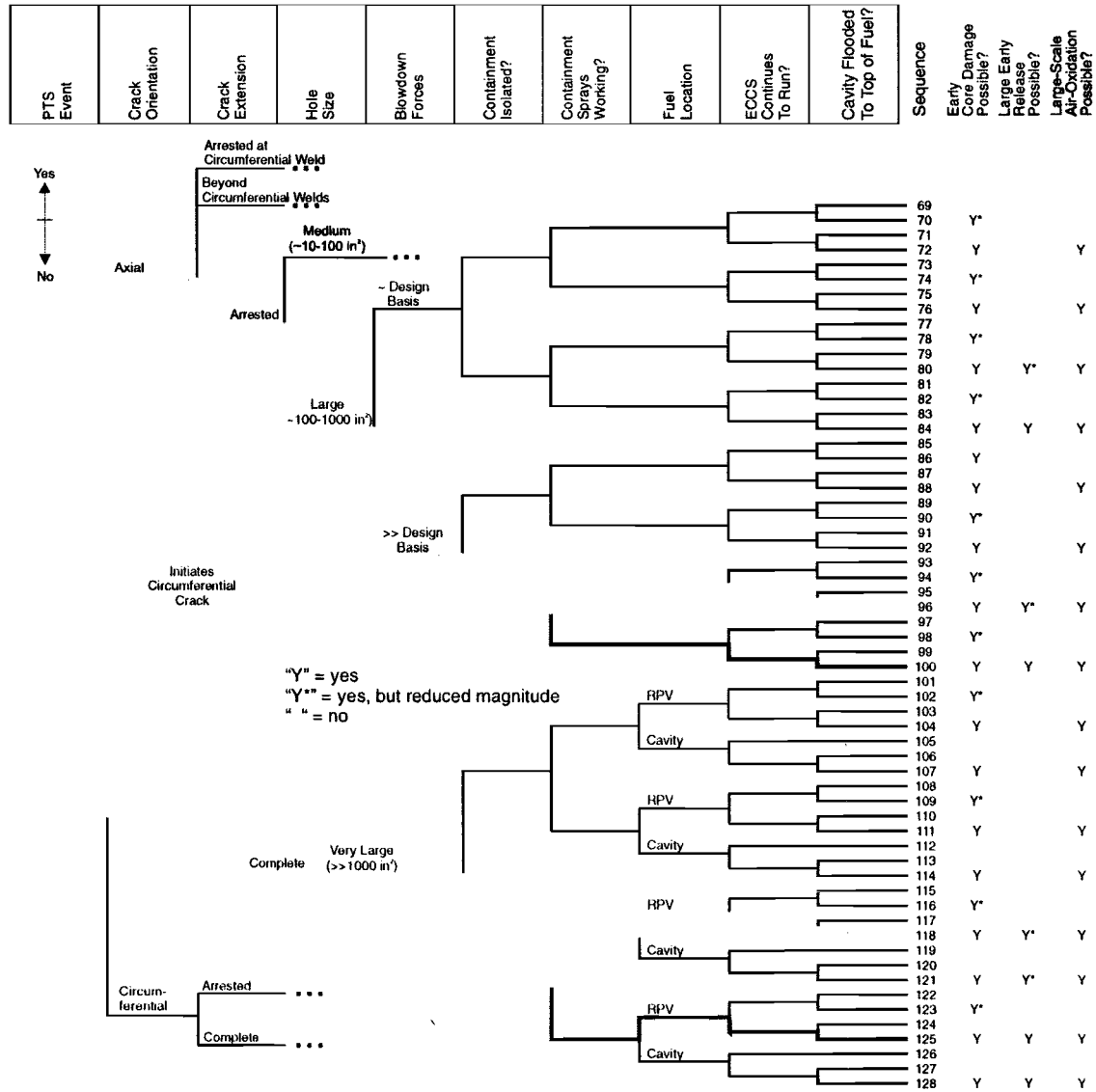
Internal Pressure Differentials

Transient	Vessel Break	Core Barrel ΔP (psi)	Core ΔP (psi)	Downcomer ΔP (psi)	Duration
4-inch surge line break	Axial	10ms	150	60	150
	1s	15	-10	25	12-30 ms 1 s
	Circ	10ms	165	110	35
	1s	45	30	15	20-70 ms 1 s
Stuck open SRV	Axial	10ms	1800	600	1680
	1s	50	-10	40	10-20 ms 130 ms
	Circ	10ms	2140	1460	50
	1s	240	100	-15	10-20 ms 60 ms
LB LOCA	N/A	10ms	1010	240	1110
		1s	-170	-70	-500

Containment Pressure



Key Sequences



PTS Event	Crack Orientation	Crack Extension	Flow Hole Size	Containment Isolation Working?	Sprays	Fuel Cavity	ECCS Runs Flooded	Sequence	Early Core Damage Possible?	Large Early Release Possible?	Large Scale Air-Oxidation Possible?
		Arrested at Circumferential Weld	Medium (~10-100 in ²)	Design Basis				69			
		Beyond Circumferential Welds	Medium (~10-100 in ²)	Design Basis				70	Y*		
			Medium (~10-100 in ²)	Design Basis				71	Y		
			Medium (~10-100 in ²)	Design Basis				72	Y		Y
			Medium (~10-100 in ²)	Design Basis				73	Y		
			Medium (~10-100 in ²)	Design Basis				74	Y*		
			Medium (~10-100 in ²)	Design Basis				75	Y		
		Arrested	Medium (~10-100 in ²)	Design Basis				76	Y		Y
			Medium (~10-100 in ²)	Design Basis				77	Y		
			Medium (~10-100 in ²)	Design Basis				78	Y	Y*	Y
			Medium (~10-100 in ²)	Design Basis				80	Y	Y*	Y
			Medium (~10-100 in ²)	Design Basis				81	Y		
			Medium (~10-100 in ²)	Design Basis				82	Y*		
			Medium (~10-100 in ²)	Design Basis				83	Y	Y	Y
			Medium (~10-100 in ²)	Design Basis				84	Y	Y	Y
			Medium (~10-100 in ²)	Design Basis				85	Y		
			Medium (~10-100 in ²)	Design Basis				86	Y		
			Medium (~10-100 in ²)	Design Basis				87	Y		
			Medium (~10-100 in ²)	Design Basis				88	Y		Y
			Medium (~10-100 in ²)	Design Basis				89	Y		
			Medium (~10-100 in ²)	Design Basis				90	Y*		
			Medium (~10-100 in ²)	Design Basis				91	Y		
			Medium (~10-100 in ²)	Design Basis				92	Y		Y
			Medium (~10-100 in ²)	Design Basis				94	Y*		
			Medium (~10-100 in ²)	Design Basis				95	Y	Y*	Y
			Medium (~10-100 in ²)	Design Basis				96	Y	Y*	Y
			Medium (~10-100 in ²)	Design Basis				98	Y	Y*	Y
			Medium (~10-100 in ²)	Design Basis				99	Y	Y*	Y
			Medium (~10-100 in ²)	Design Basis				100	Y	Y*	Y
			Medium (~10-100 in ²)	Design Basis				115	Y	Y	Y
			Medium (~10-100 in ²)	Design Basis		RPV		116	Y*		
			Medium (~10-100 in ²)	Design Basis		Cavity		117	Y	Y*	Y
			Medium (~10-100 in ²)	Design Basis				118	Y	Y*	Y
			Medium (~10-100 in ²)	Design Basis				119	Y	Y*	Y
			Medium (~10-100 in ²)	Design Basis				120	Y	Y*	Y
			Medium (~10-100 in ²)	Design Basis				121	Y	Y*	Y
			Medium (~10-100 in ²)	Design Basis		RPV		122	Y*		
			Medium (~10-100 in ²)	Design Basis				123	Y		
			Medium (~10-100 in ²)	Design Basis				124	Y	Y	Y
			Medium (~10-100 in ²)	Design Basis		Cavity		125	Y	Y	Y
			Medium (~10-100 in ²)	Design Basis				126	Y		
			Medium (~10-100 in ²)	Design Basis				127	Y	Y	Y
			Medium (~10-100 in ²)	Design Basis				128	Y	Y	Y

Yes
↑
↓
No

Axial

Arrested

Large
100-1000 in²

Initiates
Circumferential
Crack

Circumferential

Arrested

Complete

Very Large
>>1000 in²

Complete

~40%/15%/45%

~90%/10%

Independent

Independent

T/H Calculations: ↑

Unreliability < 10⁻²

Design-Dependent

Potentially Dependent

Observations

- Accident energetics are more benign than those of some other scenarios previously studied (e.g., HPME)
- Containment pressurization likely to be less than design basis LOCA
- Blowdown forces on RPV and internals likely to be the same order of magnitude or bounded by DB LOCA
- Containment spray failure probability may decrease for PTS events (as compared with non-PTS risk-significant accidents)*
- Likelihood of fuel cooling dependent on reactor cavity design
 - Cavity flooding above top of fuel expected for some plants
 - For other plants, ECCS may not be sufficient to cool fuel

*For some plants, this may be dependent on plant changes in response to GSI-191.

Scoping Study Conclusions

- The conditional probability of early fuel damage (given a PTS-induced RPV failure) appears to be
 - Extremely small for plants with cavities likely to be flooded
 - Non-negligible for other plants
- The conditional probability of early containment failure and a large, early release (given a PTS-induced RPV failure) appears to be very small for all plants
- Should a PTS-induced large, early release occur, such a release may involve a large-scale air-oxidation source term

Implications for RVFF*

- **RVFF* = 1×10^{-6} /ry is consistent with philosophy of original PTS rule, with ACRS guidance, and with Safety Goal Policy Statement**
 - **Assures a low level of risk associated with PTS events**
 - **Assures small relative contribution to acceptable risk**
 - **More limiting with respect to core damage than RG 1.174/Option 3 criterion for CDF**
 - **Consistent or conservative with respect to QHOs**
- **Expectation: RPV embrittlement limits will be established in a risk-informed manner**

Summary Conclusions

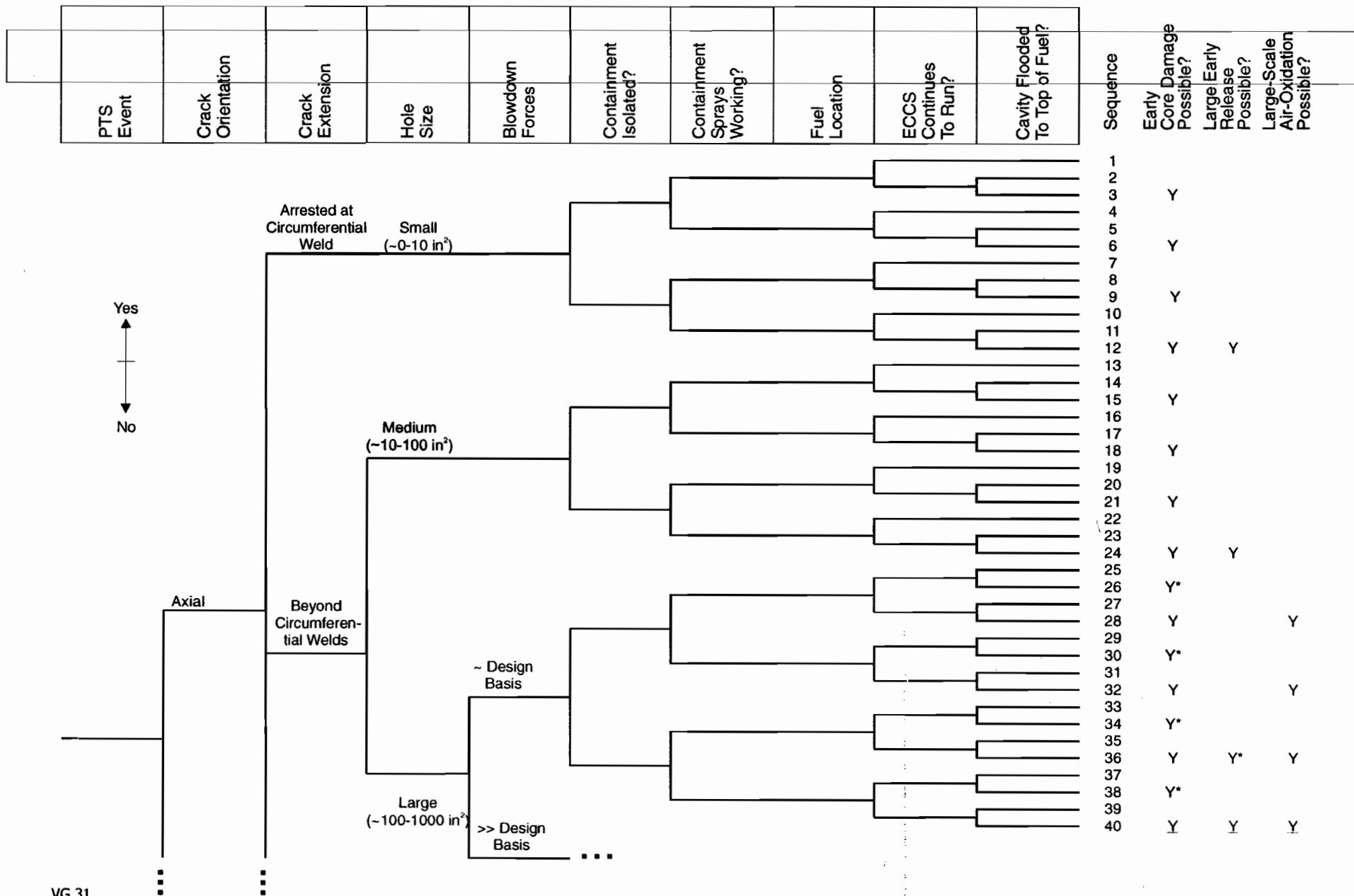
- **RVFF \equiv TWCF**
- **RVFF* = 1×10^{-6} /ry**
- **RVFF* should be compared against mean of plant-specific RVFF distribution**

Backup Slides

Technical Issues

- **Definition of RVFF**
- **Dominant plant damage states**
- **Relative contribution of axial and circ welds**
- **Crack propagation, hole size, hole location**
- **Blowdown forces**
- **Containment isolation**
- **Missiles**
- **ECCS status (injection, recirculation)**
- **Containment spray status**
- **Core status (intact, distorted, disrupted)**
- **Fuel dispersal**
- **Fuel coolability**
- **RPV water level**
- **Fuel environment (steam, air)**
- **Early overpressure**

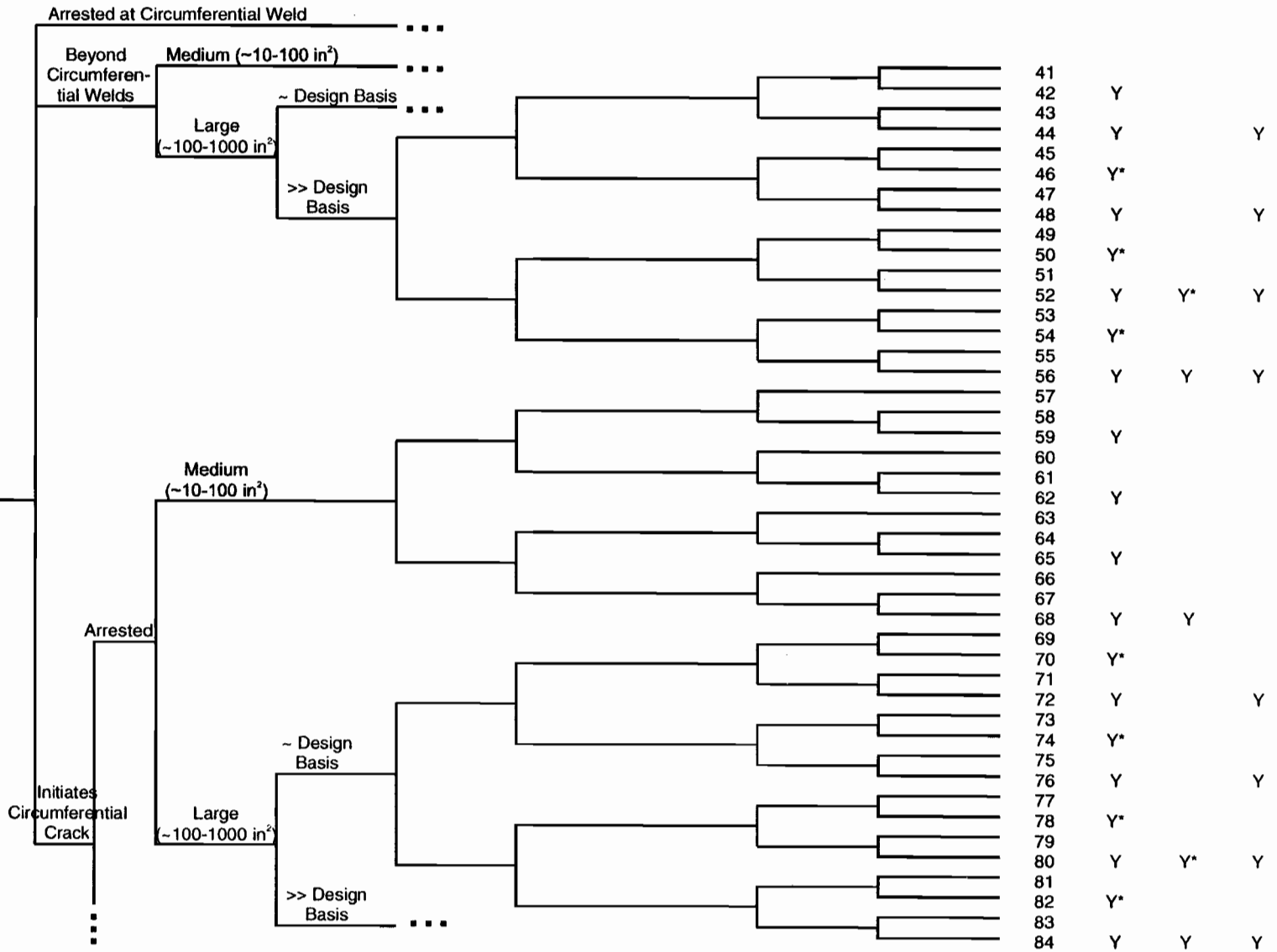
APET (Page 1 of 5)



APET (Page 2 of 5)

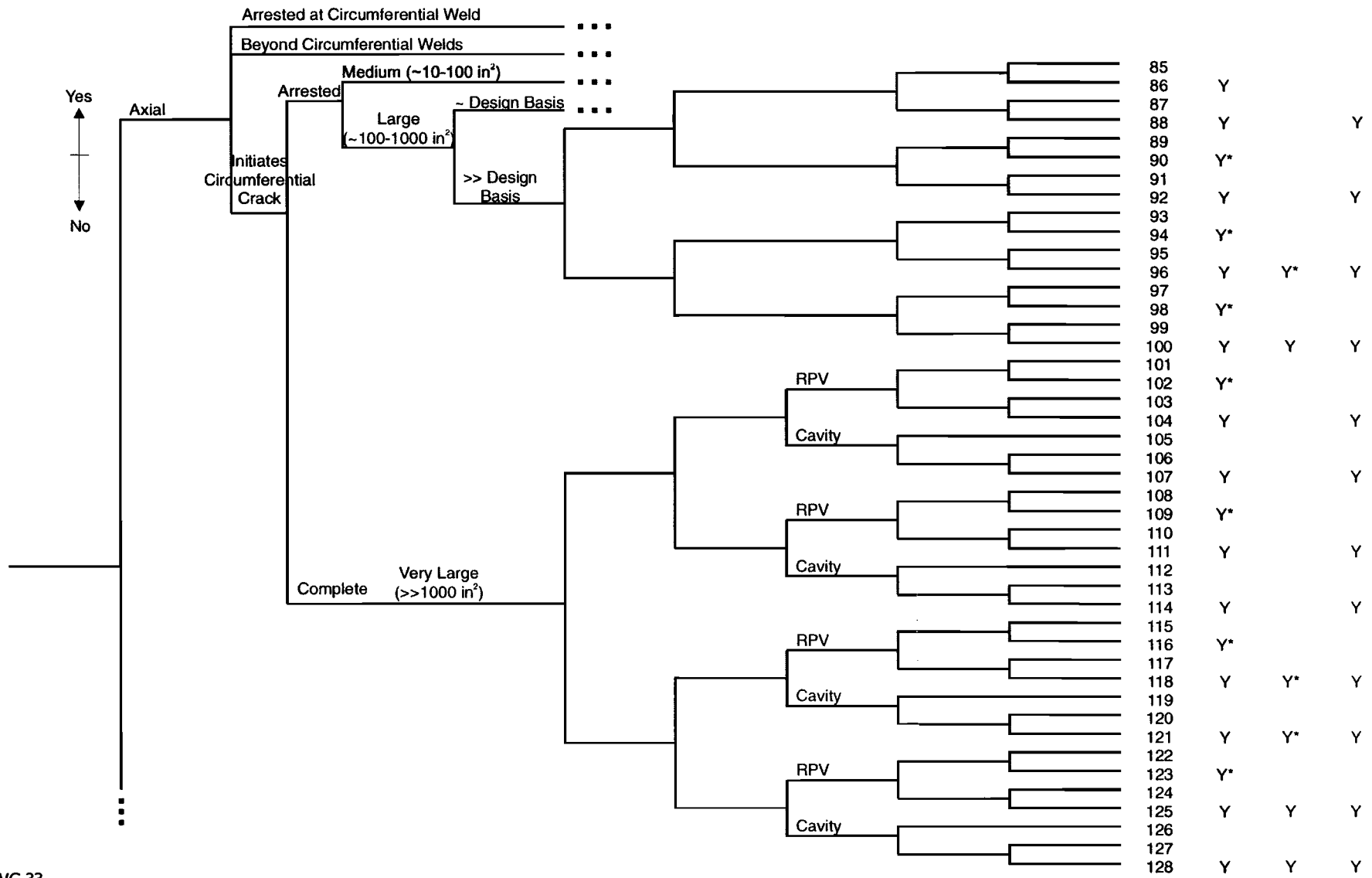
PTS Event	Crack Orientation	Crack Extension	Hole Size	Blowdown Forces	Containment Isolated?	Containment Sprays Working?	Fuel Location	ECCS Continues To Run?	Cavity Flooded To Top of Fuel?	Sequence	Early Core Damage Possible?	Large Early Release Possible?	Large-Scale Air-Oxidation Possible?
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Yes
↑
↓
No



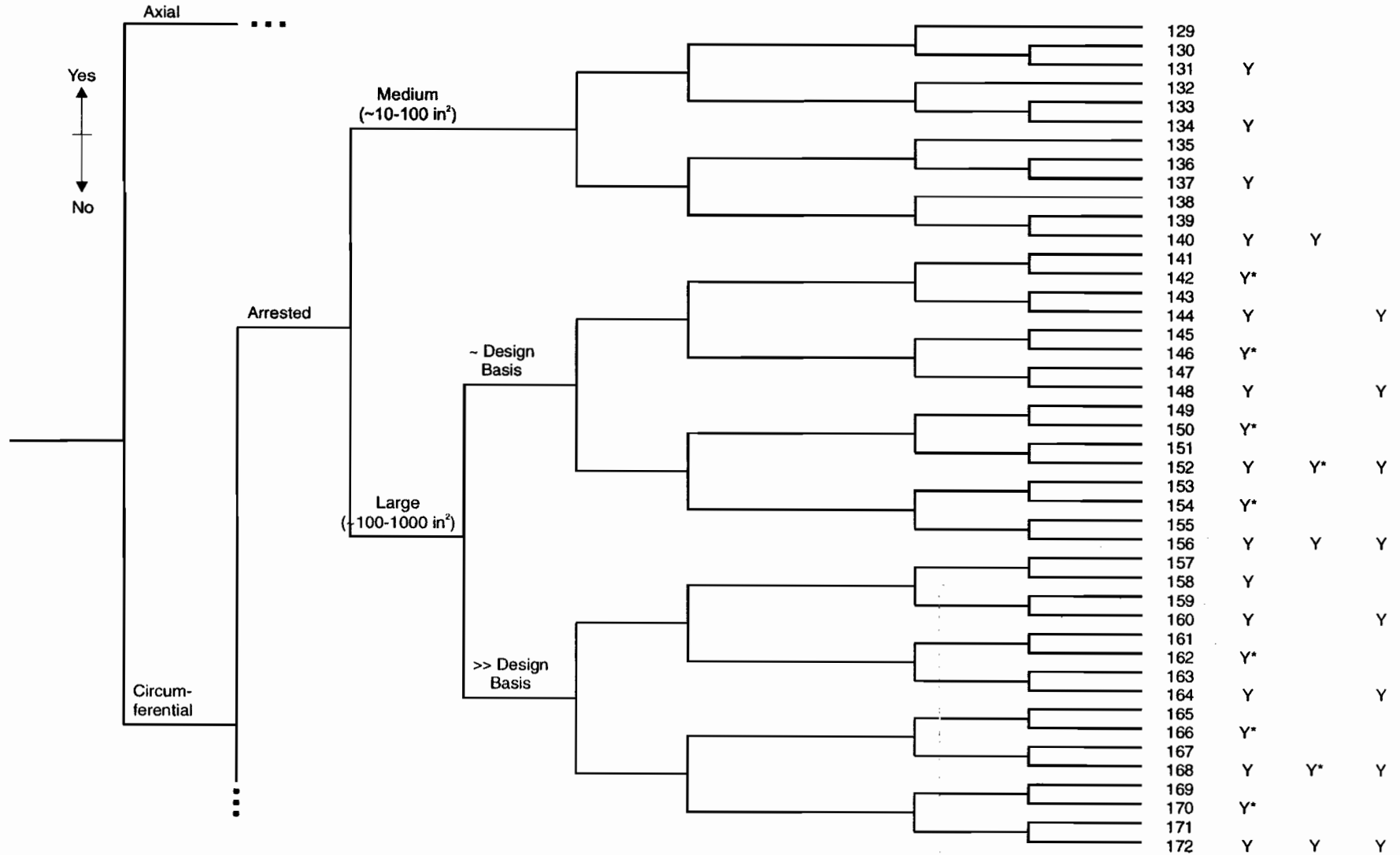
APET (Page 3 of 5)

PTS Event	Crack Orientation	Crack Extension	Hole Size	Blowdown Forces	Containme Isolated?	Containme Sprays Working?	Fuel Location	ECCS Continues To Run?	Cavity Flooded To Top of Fuel?	Sequence	Early Core Damage Possible?	Large Early Release Possible?	Large-Scale Air-Oxidation Possible?
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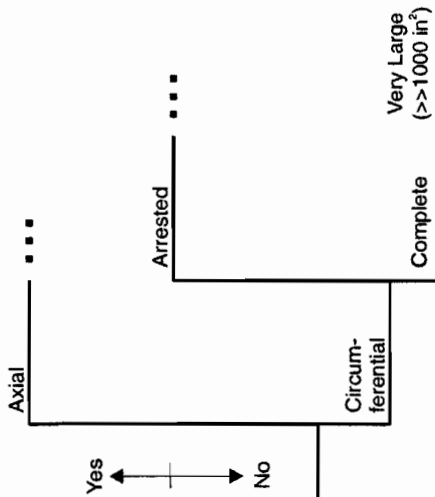
APET (Page 4 of 5)

PTS Event	Crack Orientation	Crack Extension	Hole Size	Blowdown Forces	Containment Isolated?	Containment Sprays Working?	Fuel Location	ECCS Continues To Run?	Cavity Flooded To Top of Fuel?	Sequence	Early Core Damage Possible?	Large Early Release Possible?	Large-Scale Air-Oxidation Possible?
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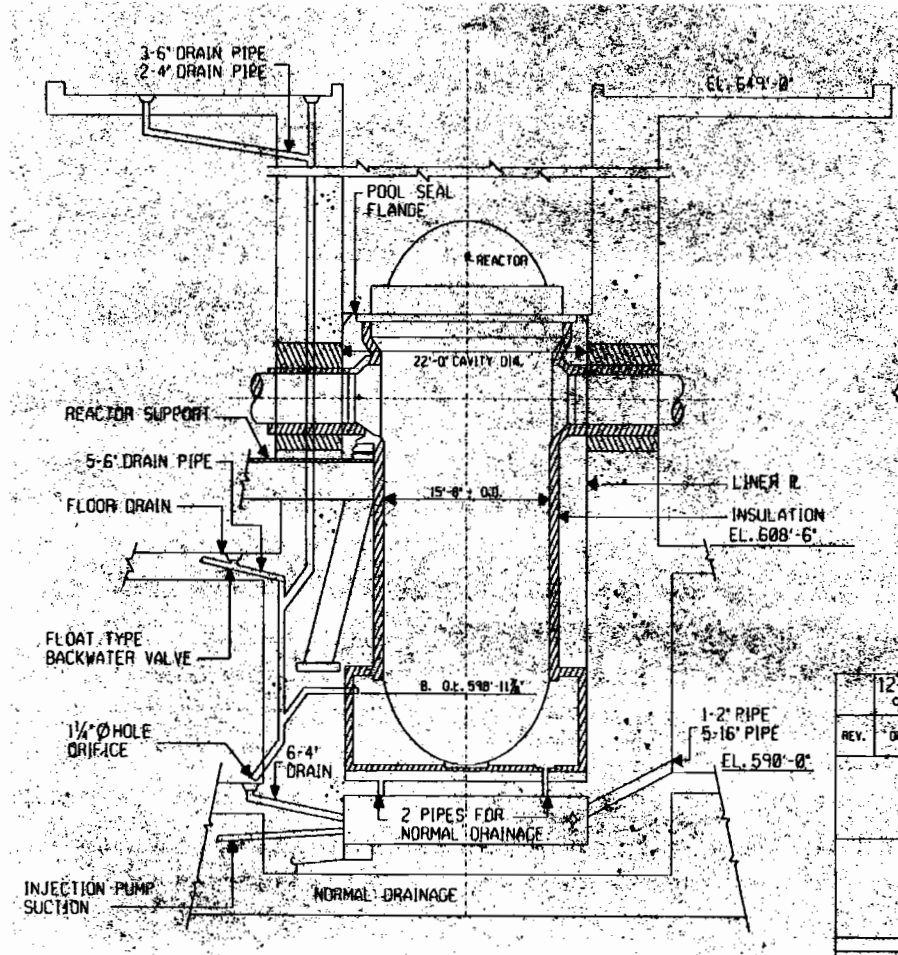


APET (Page 5 of 5)

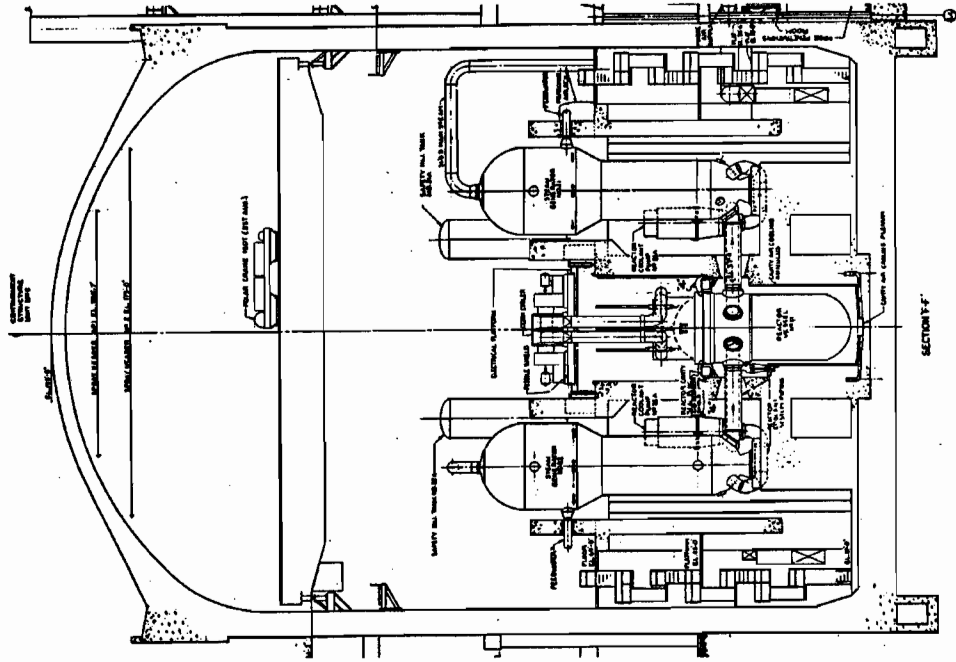
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	Axial	...								173			
		Arrested	...				RPV			174			
							Cavity			175			
										176	Y		Y
										177			
										178	Y		Y
										179			
							RPV			180			
							Cavity			181	Y		
										182			
										183	Y		Y
										184			
										185	Y		Y
										186			
										187	Y		
							RPV			188	Y		
							Cavity			189		Y	
										190	Y		Y
										191			
										192	Y		Y
										193			
							RPV			194	Y		
							Cavity			195		Y	
										196	Y		Y
										197			
										198	Y		Y
										199			
										200	Y		Y



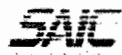
Palisades Reactor Cavity



Calvert Cliffs Containment



PTS Re-Evaluation Project Briefing



Mark Kirk, Ed Hackett

*Probabilistic Fracture Mechanics
(RES/DET/MEB)*

Nathan Siu, Roy Woods, Donnie Whitehead, Alar Kolaczowski

*Probabilistic Risk Assessment
(RES/DRAA/PRAB)*

David Bessette

*Thermal Hydraulics
(RES/DSARE/SMSAB)*

ACRS Materials Subcommittee Meeting on PTS Re-Evaluation
USNRC Headquarters • Rockville, MD • 5th February 2003

VG 1

AGENDA

	Topic	Presenters	Time
I.	Opening Remarks	P. Ford, ACRS	8:30-8:35 a.m.
II.	PTS Re-evaluation Project and Staff Introductions	M. Mayfield, RES	8:35-8:50 a.m.
III.	PTS Project Overview, Background. Significance of RELAP differences w/ experiments	M. Kirk, D. Bessette	8:50-9:55 a.m.
BREAK			9:55-10:10 a.m.
IV.	Plant Specific Results (Oconee-1, Beaver Valley-1, Palsades) Thermal-Hydraulic characteristics of dominant transients. Uncertainty results	M. Kirk, D. Bessette, R. Woods, D. Whitehead, A. Kolaczowski	10:10 a.m - 2:10 pm
LUNCH			12:00-1:00 p.m.
V.	Risk-informed Reactor Vessel Failure Frequency Acceptance Criteria Post PTS Vessel Failure considerations Results of T-H analyses	N. Siu, D. Bessette	2:10-3:10
BREAK			3:10-3:25 p.m.
VI.	PTS RT_{wor} based screening limit	M. Kirk	3:25-3:55
VII.	Overall summary and conclusions	M. Kirk, E. Hackett	3:55-4:55
VIII.	Subcommittee discussion		4:55-5:10p.m.
IX.	Adjourn		5:10 p.m.

VG 2

Broad Government and Industry Participation

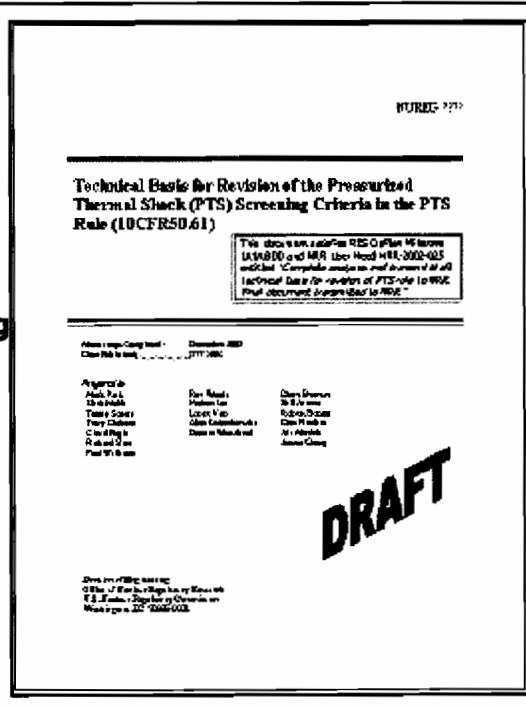


VG 3

Meeting Objectives

- **Review Draft NUREG issued on 1231-02 from RES to NRR**
- **Detail technical basis to support rulemaking**
- **Discuss on-going activities**
- **Address ACRS concerns**
- **Request ACRS letter**

VG 4



Conclusions

- **These analyses provide a technical basis to recommend revision of the PTS rule**
 - **Two of the most embrittled plants in fleet have a TWCF at or below 5×10^{-6} at end of license extension (60 years)**
 - **At the 10CFR50.61 RT_{NDT} screening limits these plants have a TWCF of 1×10^{-6} (vs. RG 1.154 at 5×10^{-6})**
- **Analysis supports a revised screening limit of**
 - **290°F on a weighted RT_{NDT} value**
 - ✓ Axial welds & plates dominate
 - ✓ Circ welds and forgings minor contributors
 - **This limit is 80F to 110F higher than current 10CFR50.61 limits on RT_{PTS}**

VG 5

5

On-Going Activities

- **RES activities**
 - **Calvert cliffs**
 - **Generalization to all plants**
 - **Sensitivity studies & a more detailed examination of current results**
 - **Favor V&V**
 - **External peer review of project**
 - **Implications for operational limits (10CFR Appendix G)**
- **NRR activities**
 - **RES Draft NUREG sent to NRR on 1-21-02**
 - **NRR comments due by 3-31-03**
 - **Decision to proceed with rulemaking?**

VG 6

Briefing Overview

- **10CFR50.61 (the PTS rule)**
 - Background & current implementation
 - Motivations for revision

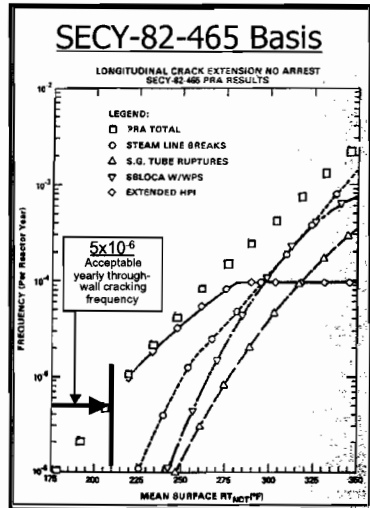
- **PTS re-evaluation project**
 - Scope of analysis
 - Plant specific results
 - ✓ Analysis approach
 - ✓ Results
 - Risk informed reactor vessel failure frequency acceptance criteria
 - Conclusions
 - ✓ Rulemaking
 - ✓ Considerations regarding a new PTS screening limit
 - On-going activities

VG 7

Background

10CFR50.61

(Background & Current Implementation)



VG 9

10CFR§50.61

If beltline materials are projected to exceed the RT_{NDT} screening limit at EOL, the licensee must either implement flux reduction and/or perform vessel specific analysis to justify continued operation.

10CFR 50.61: A multi level structure

- Compare deterministically computed RPV embrittlement (RT_{TS}) against screening criteria
- If necessary, employ reasonably practicable flux reduction measures
- If necessary, perform plant specific analysis (RG 1.154) to justify continued operation

10CFR50.61

(Motivations for Revision)

Yankee Rowe

- In late 1980s the Yankee Rowe nuclear power plant was predicted to exceed the 10CFR50.61 PTS screening criteria before EOL
- The Yankee Atomic Energy Company followed the provisions of Regulatory Guide 1.154 in an attempt to build a case supporting operation to embrittlement levels beyond the screening criteria
- Yankee Rowe was permanently shutdown in September of 1991
- The difficulties experienced with evaluation of the Yankee RG1.154 analysis led the Commission to direct the staff to revise the regulatory guide and associated rule

VG 10

10CFR50.61
(Motivations for Revision)

Technical Improvements made in the last 20 years suggest conservatism of the current rule.

- **PRA**
 - Use of latest PRA/HRA data
 - More refined binning
 - Operator action credited
 - Acts of commission considered
 - External events considered
 - Medium and large break LOCAs considered
- **TH**
 - Many more TH sequences modeled
 - TH code improved



- **PFM**
 - Significant conservative bias in toughness model removed
 - Spatial variation in fluence recognized
 - Most flaws now embedded rather than on the surface, also smaller
 - Material region dependent embrittlement props.
 - Non-conservatism removed in arrest and embrittlement models removed

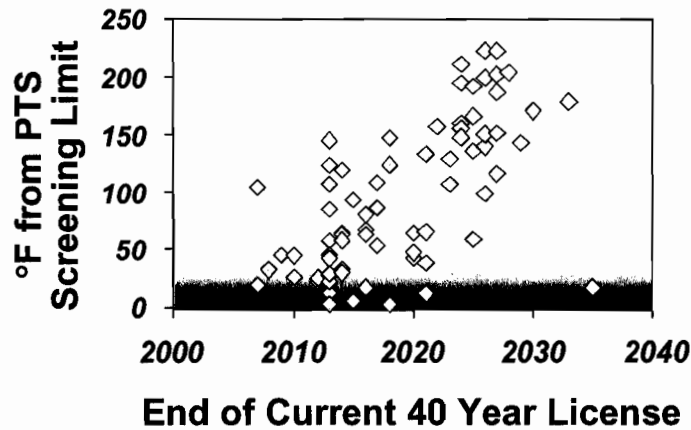


State of art analysis methods adopted throughout

VG 11

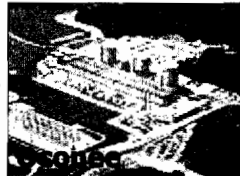
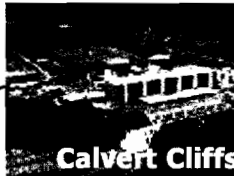
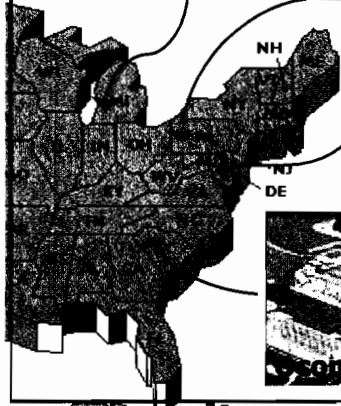
10CFR50.61
(Motivations for Revision)

Some plants "close" to the current screening criteria → licensee exemption requests without a systematic process to address them.



VG 12

Scope of Analysis



- All PWR manufacturers
 - 1 Westinghouse
 - 2 CE
 - 1 B&W
- 2 plants from original (1980s) PTS study
- 2 plants very close to the current PTS screening criteria
- All potential initiating event sequences considered

Analysis Approach

Analysis Approach

2 main components

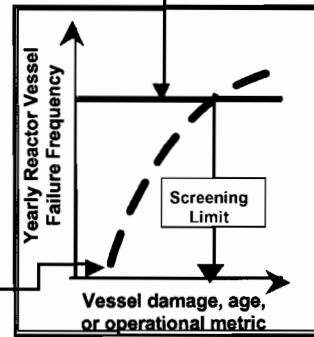
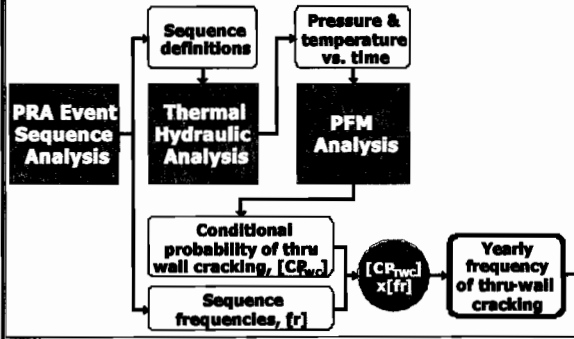
- Plant TWC estimates
- Acceptable TWC frequency

Acceptance Criterion for TWC Frequency

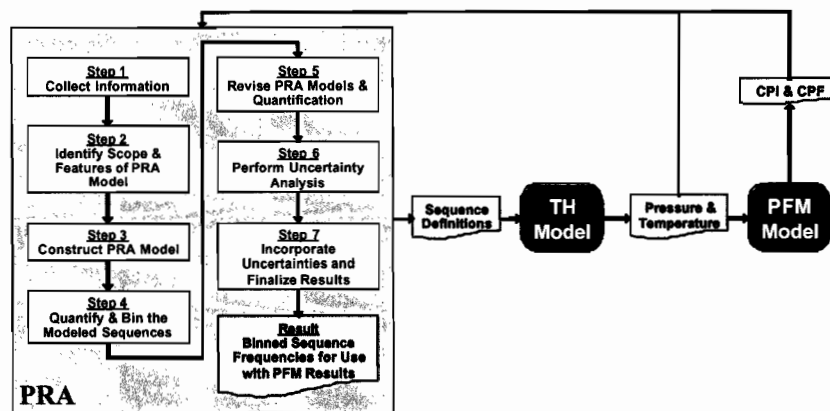
- Consistent with
- 1986 Commission safety goal policy statement
 - June 1990 SRM
 - RG1.174

PLANT TWC ESTIMATES

Uncertainties addressed and quantified as an integral part of the analysis process



Details of PRA Event Sequence Analysis



VG 16

Step 1: Collect Information

- **Started with previous PTS PRA analyses**
 - NUREG/CR-3770 (Oconee)
 - WCAP-15156 ("Beaver Valley")
 - NUREG/CR-4183 (H. B. Robinson)
 - NUREG/CR-4022 (Calvert Cliffs)
- **Collected plant specific information for three plants analyzed (Oconee, Beaver Valley, and Palisades). Examples include:**
 - Emergency and abnormal operating procedures, including PTS relevant training material
 - Plant design information,
 - Existing PRA documentation,
 - Observed simulator exercises
- **Periodic interactions with and feedback from licensees**

VG 17

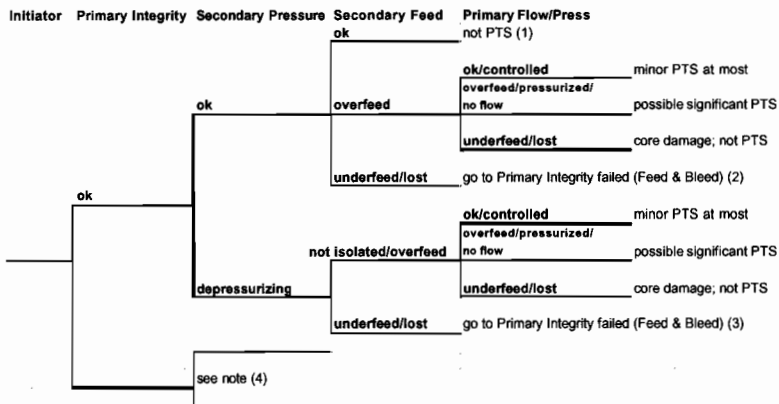
Step 2: Identify Scope & Features of PRA Model

- **Initiators**
 - **LOCAs: small, medium, large**
 - **Transients: all types including support system initiators**
 - **SGTR**
 - **SteamlineBreaks: small, large**
- **Types of accidents**
 - **Overcooling with lowering or otherwise controlled RCS pressure**
 - **Overcooling with high RCS pressure**
 - **Overcooling with repressurization**
 - **RCS faults, secondary faults, and combinations of RCS & secondary faults**
 - **At full power and at hot zero power**

VG 18

Overview of Accident Scenario Modeling

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 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

VG 19

Step 2: Identify Scope & Features of PRA Model (Continued)

- **Operator Actions**
 - **Successes**
 - **Errors of omission**
 - **Acts of commission (procedure driven)**

VG 20

Classes of Human Failures

Primary Integrity Control	Secondary Pressure Control	Secondary Feed Control	Primary Pressure/Flow Control
<ul style="list-style-type: none"> Operator fails to isolate an isolable LOCA in a timely manner (e.g., close a block valve to a stuck-open PORV) Operator induces a LOCA (e.g., opens a PORV) that induces/enhances a cooldown 	<ul style="list-style-type: none"> Operator fails to isolate a depressurization condition in a timely manner Operator isolates when not needed (may create a new depressurization challenge, lose heat sink...) Operator isolates wrong path/SG (depressurization continues) Operator creates an excess steam demand such as opening turbine bypass/atmospheric dump valves 	<ul style="list-style-type: none"> Operator fails to stop/throttle or properly align feed in a timely manner (overcooling enhanced or continues) Operator feeds wrong (affected) SG (overcooling continues) Operator stops/throttles feed when inappropriate (causes underfeed, may have to go to feed and bleed & possible overcooling that way) 	<ul style="list-style-type: none"> Operator does not properly throttle/terminate injection to control RCS pressure Operator trips reactor coolant pumps (RCPs) when not suppose to and/or fails to restore them when desirable Operator does not provide sufficient injection or fails to trip RCPs appropriately (modeled as leading to core damage rather than a PTS concern)

VG 21

Step 3: Construct PRA Model

- Oconee and Beaver Valley**
 - **Event tree— small fault tree models used for both power and hot zero power conditions**
- Palisades**
 - **Event tree— fault tree, where fault trees incorporated more component detail**
 - **Power and hot zero power combined in same model**

VG 22

Oconee PRA Model Development

- **First model to be constructed by NRC contractors**
 - **HRA initially performed by NRC contractors with review by licensee**
 - **Initiating event frequencies and equipment failure data based on industry generic data**
- **No preliminary TH or PFM information available during initial model construction**
- **Hence, modeled "all" over cooling scenarios**

VG 23

Beaver Valley PRA Model Development

- **Model developed by NRC contractors using lessons learned from Oconee analysis**
 - **HRA initially performed by NRC contractors with review by licensee**
 - **Initiating event frequencies and equipment failure data based on industry generic data**
- **Utilized results from preliminary TH and PFM information**
- **Therefore, PRA model could be simplified**

VG 24

Beaver Valley PRA Model Simplifications

■ Sequences involving:

- Certain combinations of stuck open pressurizer PORVs or SRVs were not modeled
- Certain combinations of secondary valve and simultaneous pressurizer PORV/SRV stuck open events were not modeled
- Only secondary valve (single or multiple) stuck open events were not modeled
- Only a single SG overfeed from AFW were not modeled
- Secondary depressurization downstream of the MSIVs were not explicitly modeled
- Steam generator tube ruptures were not modeled including even those involving lack of proper feed control and even with RCPs shutdown (possibly inducing RCS loop stagnation)

VG 25

Beaver Valley PRA Model Simplifications

(Continued)

■ Other sequences were screened from modeling on a case-by-case basis if the sequence frequency could be conservatively estimated at lower than $\sim 1E-8$ /yr

- Justification:
 - When coupled with the highest CPFs being calculated for any type of sequence (in the $1E-7$ range), this would yield a through wall crack frequency of $< 1E-11$ /yr range (thus would clearly not be important to the overall PTS results since some other sequences were known to involve through wall crack frequencies in the $1E-8$ /yr range for reasonable EFPYs).

VG 26

Palisades PRA Model Development

- **Started with licensee's preexisting Palisades PRA model**
- **Modified by licensee to include NRC contractor input**
- **Collaborative HRA effort**
- **Utilized initiating event frequencies and equipment failure data contained in licensee's model**

VG 27

Step 4: Quantify and Bin Modeled Sequences

- **Individual accident sequences quantified**
- **Combined "like" sequences into preliminary TH bins**
- **Developed new TH bins as necessary (an iterative process)**
- **Quantified pointestimate frequencies for all TH bins**

VG 28

Step 5: Revise PRA Models and Quantify

- **Models and preliminary results reviewed by**
 - Licensees
 - Internal project staff
- **Purpose of reviews was to determine:**
 - Whether inaccuracies existed in the models, and whether additional potential PTS sequences needed to be modeled,
 - Whether additional TH bins should be created,
 - Which human actions should be reexamined to produce even more realistic (i.e., less conservative) human error probabilities (HEPs), and
 - What combination of the above that could be accomplished within the constraints of the project.
- **Models were modified and quantified on the basis of these reviews**

VG 29

Step 6: Perform Uncertainty Analysis

- **Each scenario (TH bin) 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**
$$\text{Frequency}_{\text{scenario}} = \text{Frequency}_{\text{init Event}} \times \text{Probability}_{\text{Equip Response}} \times \text{Probability}_{\text{Op Actions}}$$
each with *epistemic* uncertainties described by a distribution
- **The various scenarios & their 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**

VG 30

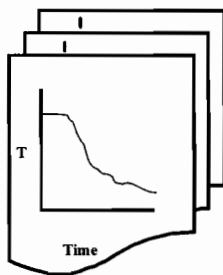
Step 7: Finalize Results

- **Selected aleatory uncertainties were dealt with quantitatively**
 - **Size of the LOCA within a LOCA category plus other factors (e.g., initial injection water temperature),**
 - **Size of the opening associated with a single or multiple stuck open SRV(s),**
 - **Time at which a stuck open SRV closes, and**
 - **Time at which operators take or fail to take action.**

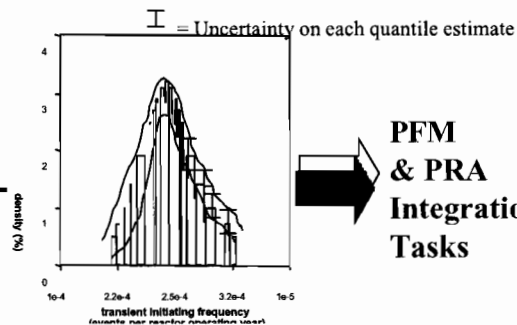
VG 31

General Form of the Results

**Set of T-H Curves
for Bin (Scenario)**



+



**PFM
& PRA
Integration
Tasks**

**Bin (Scenario) Frequency
(per year)**

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

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

VG 32

Thermal Hydraulic Analysis Approach

- **Purpose of thermal hydraulic analysis:**
 - Generated downcomer temperature, system pressure and heat transfer coefficient at the inside of the vessel wall for input FAVOR.
- **Code used for all analysis:**
 - RELAP5/MOD3.2.2 gamma released in June 1999
- **Applied previously developed models as the starting point:**
 - Oconee - model dates from original IPTS study
 - Palisades – developed from model provided by Siemens Power Corporation
 - Beaver Valley – W substantially revised H.B. Robinson IPTS model to reflect Beaver Valley
 - Two-dimensional downcomer model added and models revised to reflect current plant setpoints and operating procedures

VG 33

Assessment of RELAP5 for PTS Applications

- **Assessment presented at the 12/11/02 Thermal Hydraulic Subcommittee meeting based on:**
 - developmental assessment cases: Marviken, MIT Pressurizer, Semiscale Natural Circulation, UPTF
 - integral test data: MIST, LOFT, ROSA, ROSA-AP600
- **Review and update assessment results from Subcommittee Meeting**
 - Focus on Tests MIST-100B2, ROSA-AP600 Test APCL-03, AP-CL-09, and ROSA-IV Test SBCL-18
- **Show that:**
 - RELAP5 provides good agreement for downcomer temperature and system pressure
 - Effect of differences between code and experiment on conditional probability of vessel failure

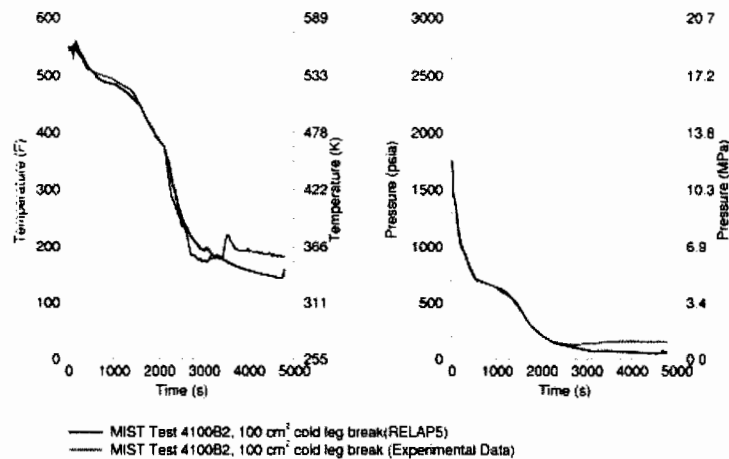
VG 34

MIST Overview

- **MIST (Multiloop Integral System Test)**
 - Full height full pressure, integral system experimental facility (power scaling factor is 817, volume scaling factor is 620)
 - B&W lowered loop design with two hot legs and four cold legs.
 - Major plant components modeled in MIST
 - Boundary systems provided simulation of the HPI, emergency feedwater, vents, controlled leaks, and steam generator tube ruptures.
- Transient assessed is Test 4100B2 which is a 4.4 inch (100 cm²) cold leg break

VG 35

MIST Results (4.4-inch cold leg break)



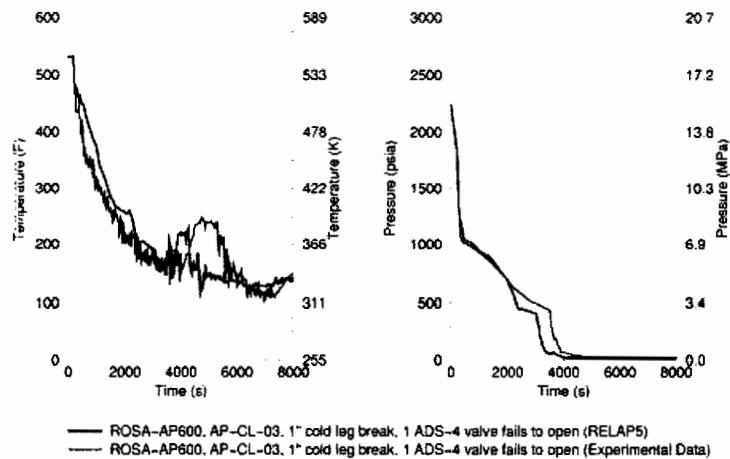
VG 36

ROSA-AP600 Overview

- **ROSA Facility**
 - **1/30 volumescaled, full height, fullpressure representation of a Westinghouse AP600 passive safety PWR**
 - **Major plant components modeled in ROSA**
- **Transients assessed:**
 - **1-inch diameter break on bottom of cold leg (CL3)**
 - **AP-CL-09 - same as APCL-03 except with multiple failures**

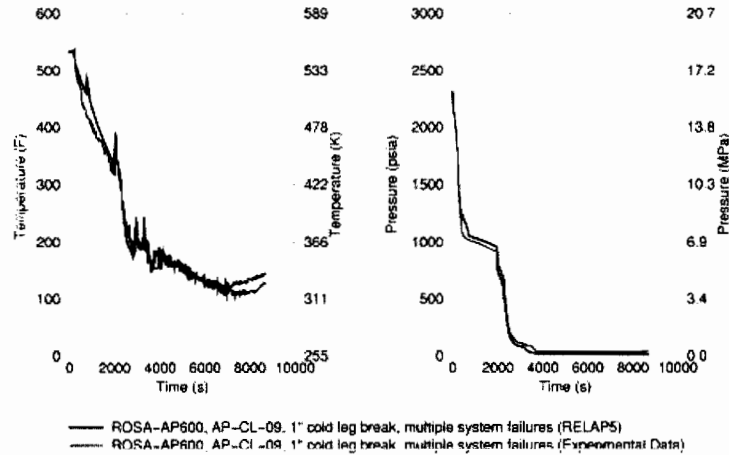
VG 37

ROSA AP-CL-03 Results



VG 38

ROSA AP-CL-09 Results



VG 39

Effect of Differences Between RELAP5 and Experiment

Case	Mean Pressure Error (MPa)	Std Dev Error (MPa)	Mean DC Temp Error (K)	Std Dev DC Temp Error (K)
MIST - 4100B2	-0.2	0.3	-4	11
ROSA - AP-CL-03	-0.4	0.7	-1	16
ROSA AP-CL-09	-0.1	0.2	0	9
ROSA-IV SB-CL-18	0.2	0.2	-1	8

Mean values and standard deviations of the pressure and average downcomer fluid temperature error defined as:

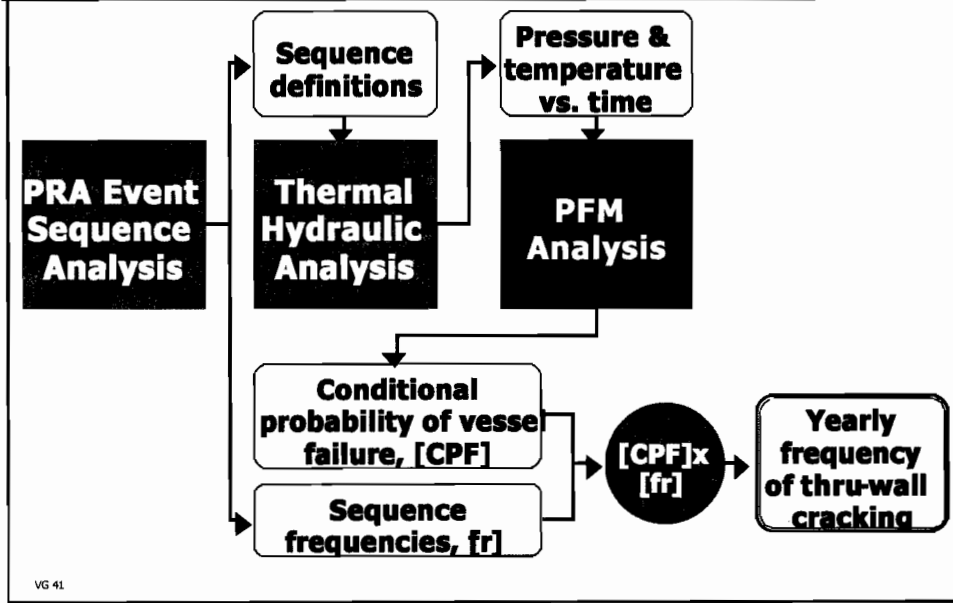
$$\text{Pressure Error} = P_{\text{RELAP}} - P_{\text{DATA}}$$

$$\text{Average Temperature Error} = T_{\text{AVE RELAP}} - T_{\text{AVE DATA}}$$

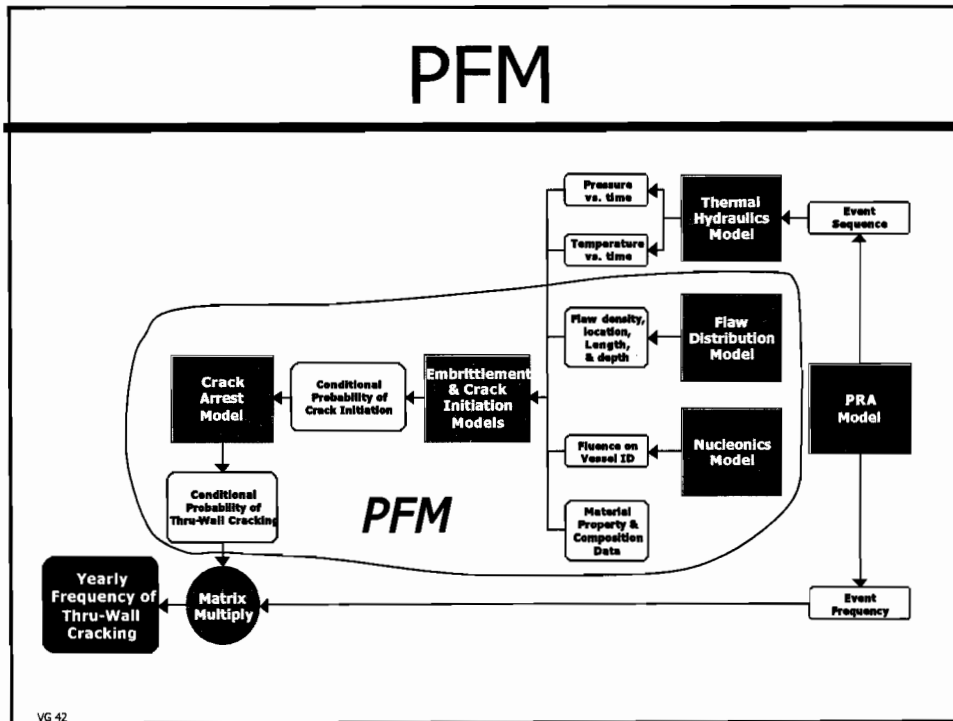
An examination of the effect of these differences on conditional probabilities of vessel failure as calculated by FAVOR is underway

VG 40

PFM in the Overall Process



PFM



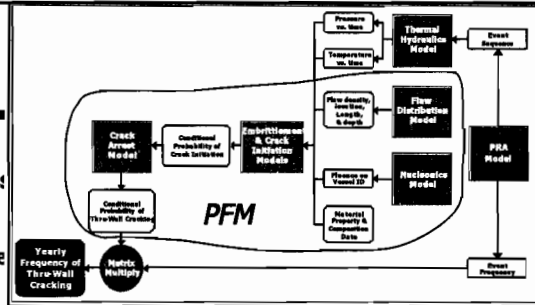
Probabilistic Fracture Mechanics Summary

Toughness

- Referenced to toughness data & physical understanding
 - Significant conservative bias in unirradiated 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

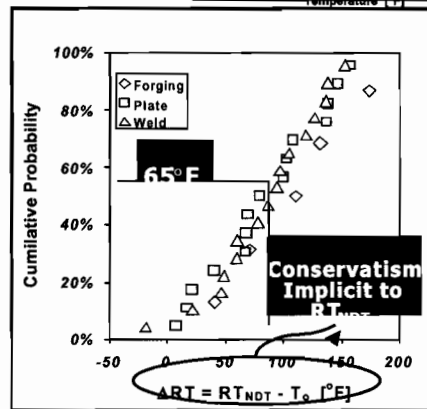
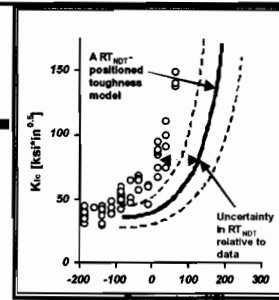
Flaws

- Based on significantly more data than before
- Most flaws now embedded rather than surface flaws
- More flaws than before

VG 43

RT_{NDT} Bias Correction

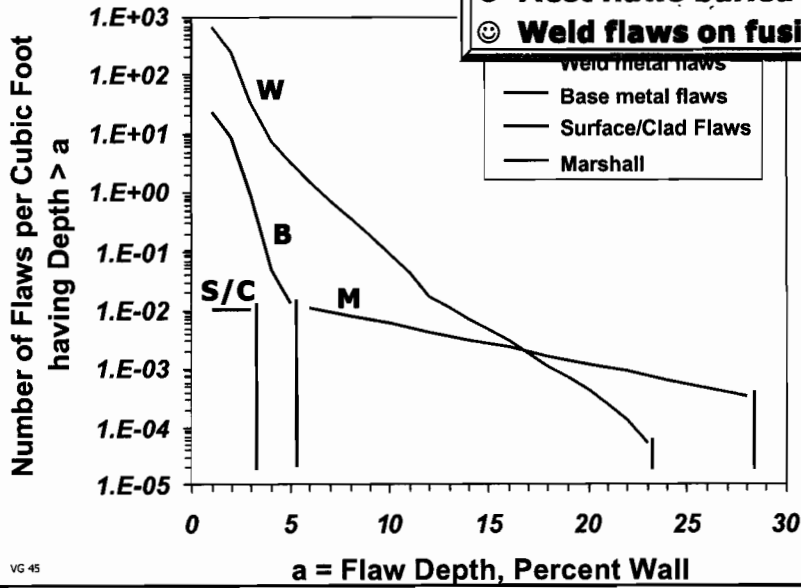
- Quantify how far off RT_{NDT} is from an accurate representation of real toughness data
 - Using a consistent representation of that data
- T₀ best represents "true" fracture toughness transition data
- Adjustment based on CDF of $\Delta RT = RT_{NDT(u)} - T_0$
- ΔRT accounts for all known epistemic uncertainties



VG 44

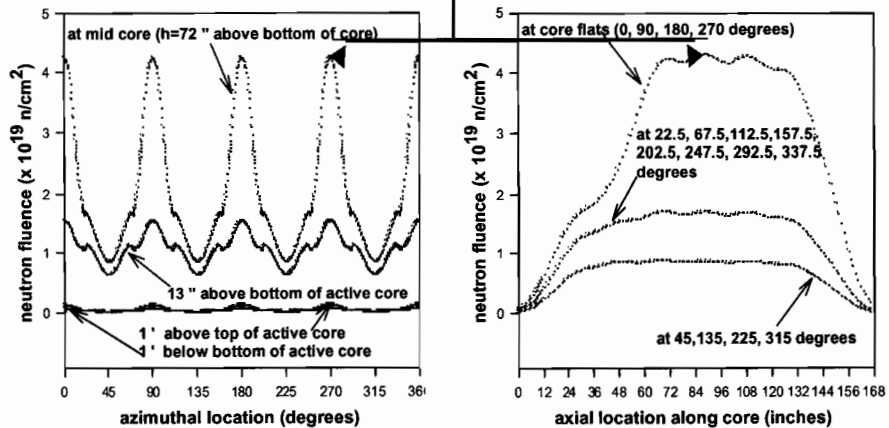
Flaw Distribution

- ☹ **More flaws**
- ☺ **Flaws smaller**
- ☺ **Most flaws buried**
- ☺ **Weld flaws on fusion lines**



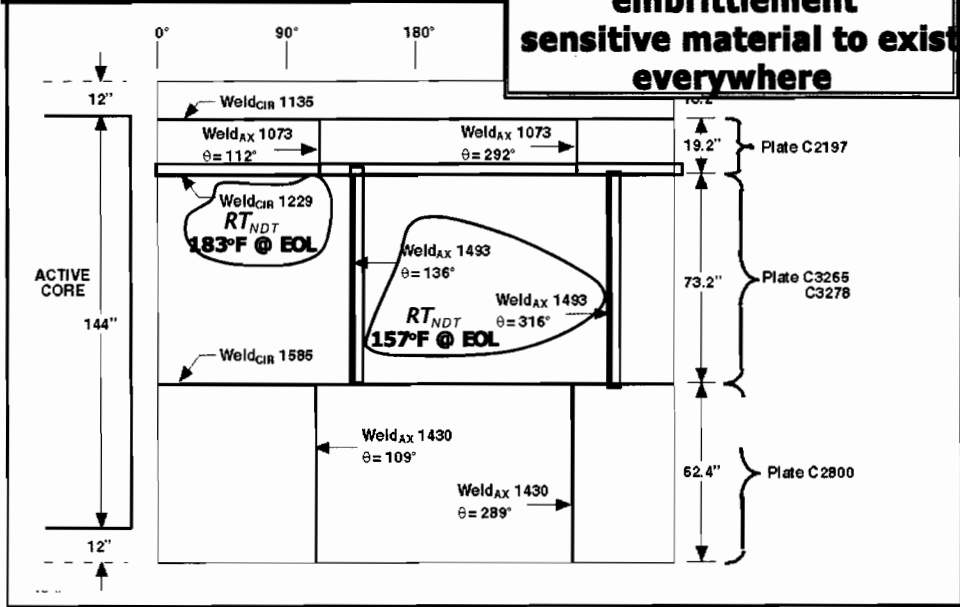
Fluence

Previous (circa 1980s) PTS analyses assumed peak fluence over the entire vessel



Oconee 1 Material Map

**Previous analyses
assumed most
embrittlement
sensitive material to exist
everywhere**



Plant Specific Results

Outline

■ Plant specific analysis features and inputs

- PRA
- TH
- PFM

■ Estimated yearly TWCF

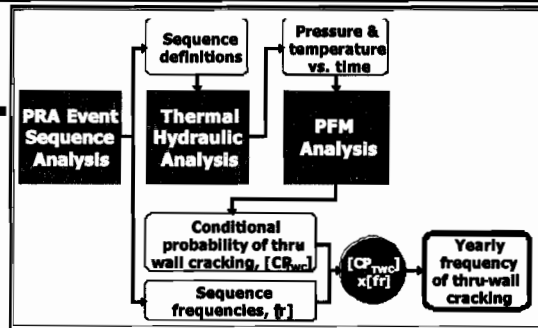
- Values
- Distribution characteristics

■ Dominant contributors to TWCF

- Transients
- Material features

■ Applicability of these results beyond the 3 study plants

- External events
- Generalization to all PWRs



VG 53

Scope Considered PRA

■ Initiators

- LOCAs: small, medium, large
- Transients: all types including support system initiators
- SGTR
- Steamline Breaks: small, large

■ Types of accidents

- Overcooling with lowering or otherwise controlled RCS pressure
- Overcooling with high RCS pressure
- Overcooling with repressurization
- RCS faults, secondary faults, and combinations of RCS & secondary faults
- At full power and at hot zero power

■ Operator Actions

- Successes
- Errors of omission
- Acts of commission (procedure driven)

VG 54

54

Issues Important to Understanding the Results

- **Numerous uncertainties were accounted for**
 - **Break size variation,**
 - **HPI flow and temperature variations,**
 - **Valve size openings, and**
 - **Timing of SRVreclosure**
- **Combinations of these uncertainties yield different TH profiles**
- **Representative cases were selected to depict all these possible TH profiles by the assignment of appropriate split fractions**

VG 55

Issues Important to Understanding the Results (Continued)

For example, the original Palisades medium LOCA bin was subdivided into the following TH bins to represent the possible spectrum of TH profiles using the split fractions provide by UMD.

TH Case No.	TH Case Description	Split Fraction
62	20.32 cm (8 in) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	0.35
63	14.37 cm (5.656 in) cold leg break. Winter conditions assumed (HPI and LPI injection temp = 40 F, Accumulator temp = 60 F)	0.30
64	10.16 cm (4 in) surge line break. Summer conditions assumed (HPI and LPI injection temp = 100 F, Accumulator temp = 90 F)	0.35

VG 55

Plant-Specific TH Features

Characteristic	Oconee	Beaver Valley	Palisades
Plant Type	B&W lowered loop design, 2x4 loops, OTSG	Westinghouse design, 3 loops	CE design, 2x4 loops
Core Power (MWth)	2568	2660	2530
RCP Trip Criteria	all pumps assumed to trip when subcooling < 0.5 °F	$\Delta P < 200$ psid between the RCS and highest SG pressure (normal containment conditions) $\Delta P < 375$ psid (adverse containment conditions)	one pump tripped in each loop if PZR pressure < 1300 psia. All pumps tripped when subcooling < 25 °F
HPI maximum flow	180 lbm/sec	134.8 lbm/sec	184 lbm/sec
HPI shutoff head	> 2600 psia	> 2600 psia	1292 psia
LPI maximum flow	1050 lbm/sec	690.9 lbm/sec	922 lbm/sec
LPI shutoff head	214 psia	215 psia	218 psia
Accum liquid volume	2150 ft ³	3104.1 ft ³	4800 ft ³
Accum disch press	590 psia	648 psia	215 psia
PZR SRV Capacity	489,183 lbm/hr (total for 2 valves)	1,494,618 lbm/hr (total for 3 valves)	690,000 lbm/hr (total for 3 valves)
SG Water Mass	40,000 lbm (HFP) 11,000 lbm (HZP)	118,760 lbm (HFP) 160,470 lbm (HZP)	142,138 lbm (HFP) 210,759 lbm (HZP)
SG SRV Capacity	13.0 Mlb/hr (16 valves)	13.1 Mlb/hr (15 valves)	37.6 Mlb/hr (24 valves)
AFW maximum flow (total)	1390 gpm (motor) 1350 gpm (turbine)	700 gpm (motor) 700 gpm (turbine)	400 gpm (motor) 400 gpm (turbine)

VG 57

Plant-Specific TH Features (Cont.)

- **Reactor vessel vent valves (B&W)**
 - As in Oconee, valves connect upper plenum to downcomer
- **LPI and Accumulator:**
 - Beaver Valley and Palisades LPI and accumulator connections to each cold leg
 - Oconee- low pressure injection and core flood tank (accumulator) discharge connected directly to the downcomer above the cold leg nozzle
 - Palisades- low accumulator initial pressure
- **HPI flow characteristics**
 - Oconee- about 30 percent more flow to the "A" loop compared to the B loop
 - Beaver Valley and Palisades equal flow to all loops

VG 58

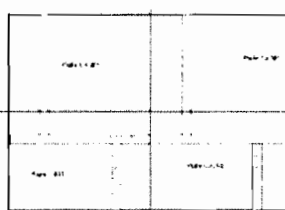
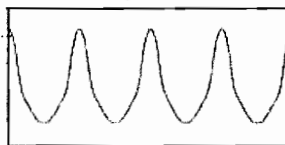
Plant Specific PFM Features

Cladding

- **Oconee** Single layer clad → circ. surface breaking cracks
- **Beaver Valley & Palisades** Multi layer clad → no surface breaking cracks

All plants have plantspecific

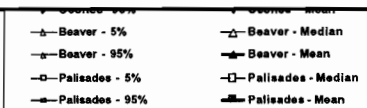
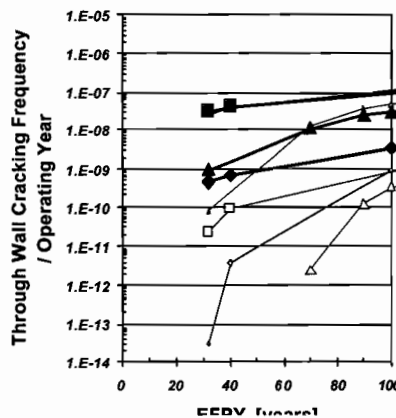
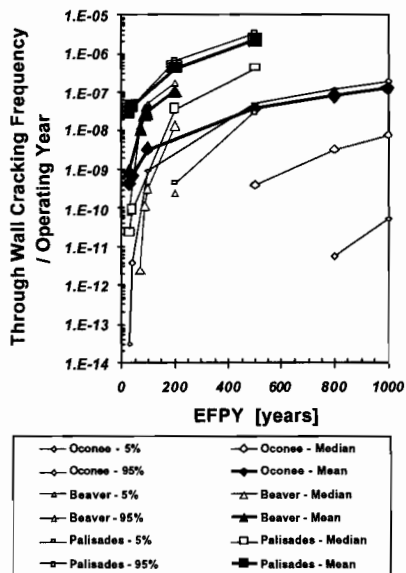
- Dimensions
- Weld / plate placement
- Fluence maps
- Chemistry
- Transition temperature



VG 59

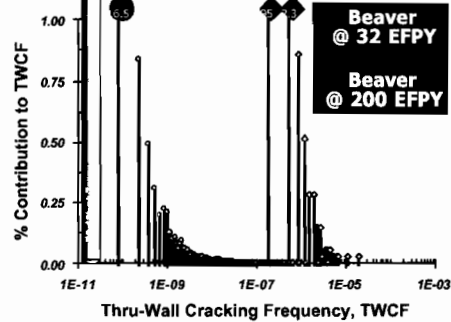
TWCF Estimates

- Over realistic operational lifetimes, the estimated TWCF for these plants is *very small*.
- Values range from 1×10^{-10} to 5×10^{-6}
- Two of these plants are among the most embrittled in operation.



Characteristics of these TWCF Distributions

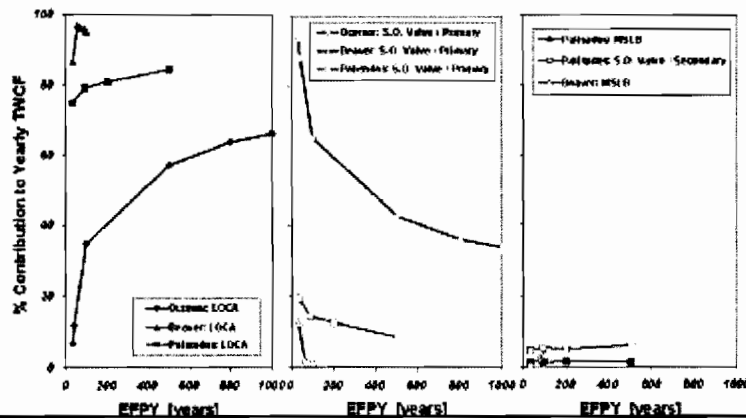
- **Skewed:** the 95th percentile and mean roughly coincide
 - ... because, the physical nature of cleavage fracture produces finite minimum toughness values
 - Therefore, P_r (init or fail) can be, and often is, zero
 - However, sometimes (rarely) P_r (init or fail) is large
 - Severe transients, AND
 - Large flaws, AND
 - High embrittlement, AND
 - These factors produce *skewed* TWCF distributions
- **Broad:** > 3 orders of magnitude separate 5th and 95th percentiles
 - ... for all the same reasons listed under "skewed"
 - Distributions narrow as plant operating time: because material embrittles mitigating (or eliminating) zero contributors to the TWCF



VG 75

Dominant Transients Overview

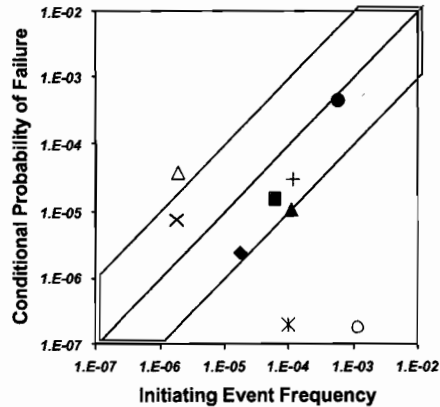
- **LOCAs dominant contributor to risk**
- **Stuck open valves also a contributor in B&W PWRs due to plant design features**
- **Secondary side breaks not important**



VG 76

Contributors to Through-Wall Cracking Frequency

- **TWCF is the product of**
 - The initiating event frequency, IEF, (X-axis), and
 - The conditional probability of failure, CPF, (Y-axis)
- **The contribution of IEF and CPF to the through wall cracking frequency is approximately "balanced"**
 - All but two of the dominant transient categories have IEFs and CPFs that are within about ± 1 order of magnitude



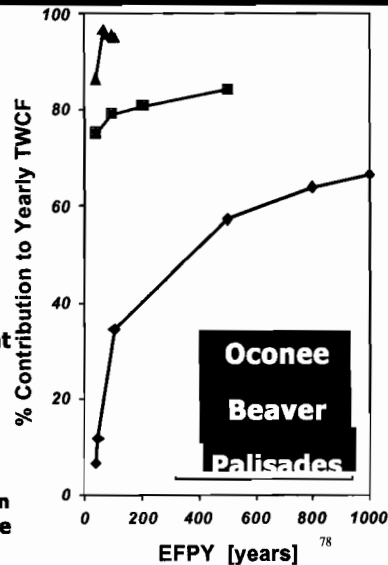
- Oconee - LOCA (Pipe Break)
- Oconee - Stuck Open Valves, Primary Side
- ▲ Beaver - LOCA (Pipe Break)
- × Beaver - Stuck Open Valves, Primary Side
- × Beaver - MSLB
- Palisades - LOCA (Pipe Break)
- + Palisades - Stuck Open Valves, Primary Side
- △ Palisades - MSLB
- Palisades - Stuck Open Valves, Secondary Side

VG 77

Source of Uncertainty in Dominant Transients

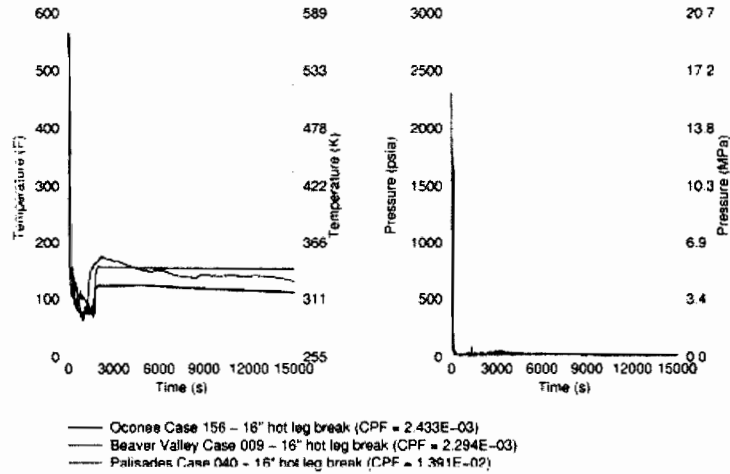
→ LOCAs ←

- LOCAs important in all 3 plants, dominate in Palisades and Beaver
- Relative contribution typically increases (or remains approximately constant at a high value) as EFPY increases
- About 3 orders of magnitude uncertainty in TWCFs driven by:
 - 2 orders of magnitude come from uncertainty in LOCA frequencies (reflect latest NRC expert judgments) propagated thru the analyses
 - T-H uncertainty is handled by different "bins" each representing different LOCA sizes; within any bin- no T-H uncertainty (small)
 - PFM uncertainties account for remainder of uncertainty
 - ✓ 1 order of magnitude: R_{DT} bias adjustment
 - ✓ 1 order of magnitude: flaw distribution
 - Operator actions do not play a key role



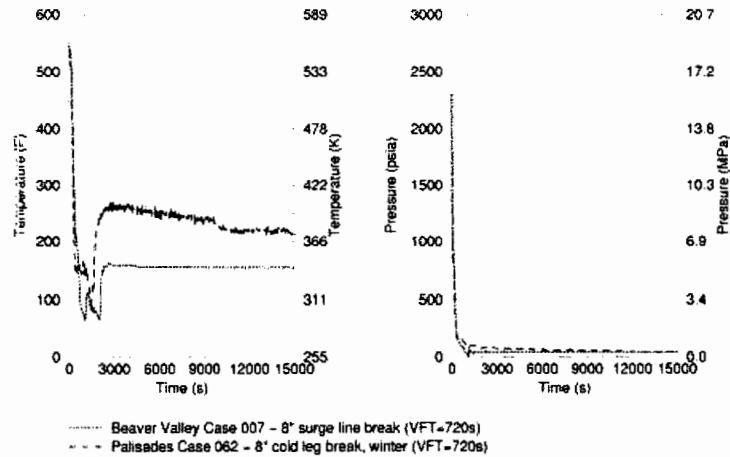
VG 78

Large Break LOCA Results Comparison



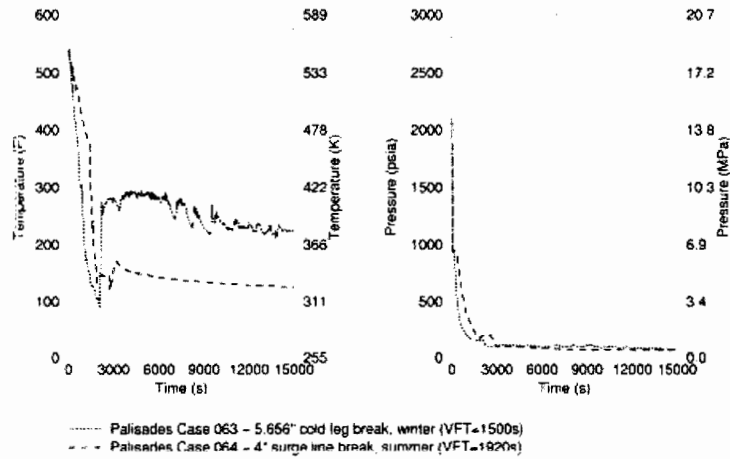
VG 79

Medium-Break LOCA Results Comparison



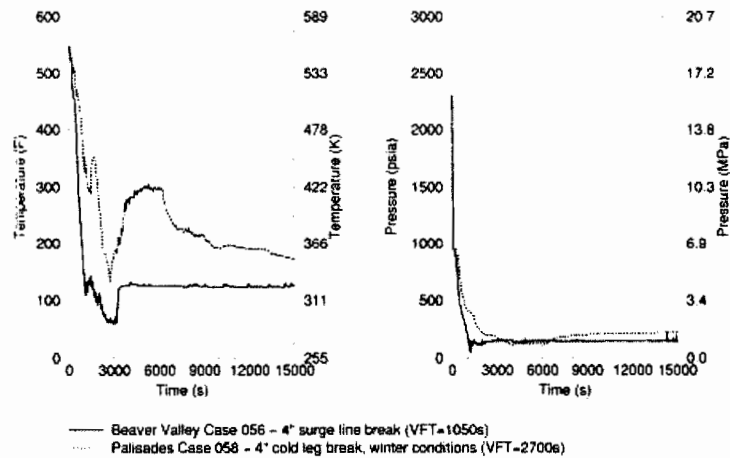
VG 80

Medium-Break LOCA Results Comparison



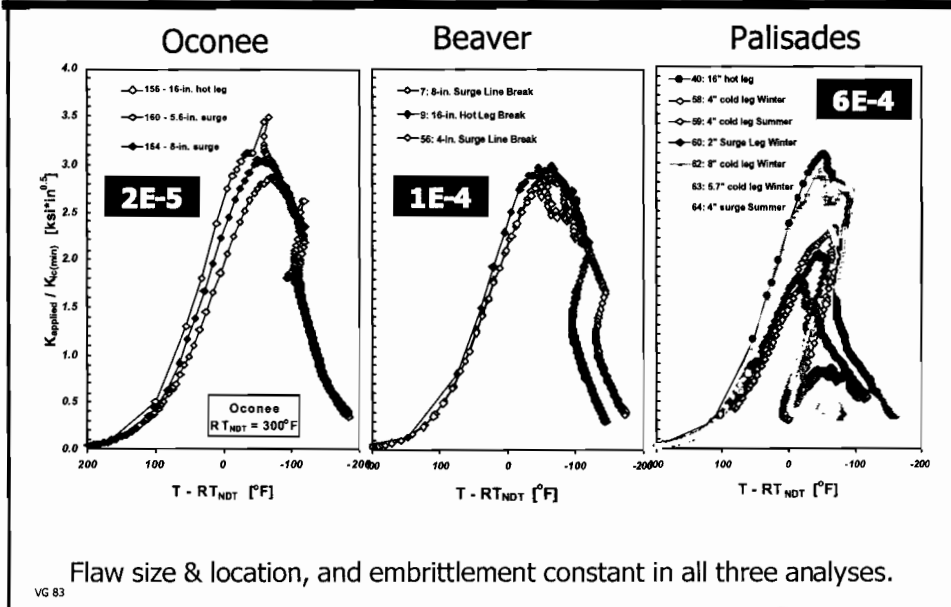
VG 81

Small-Break LOCA Results Comparison



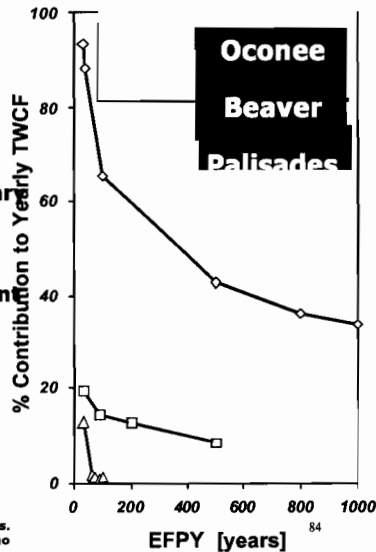
VG 82

Comparison of Dominant LOCA Transients



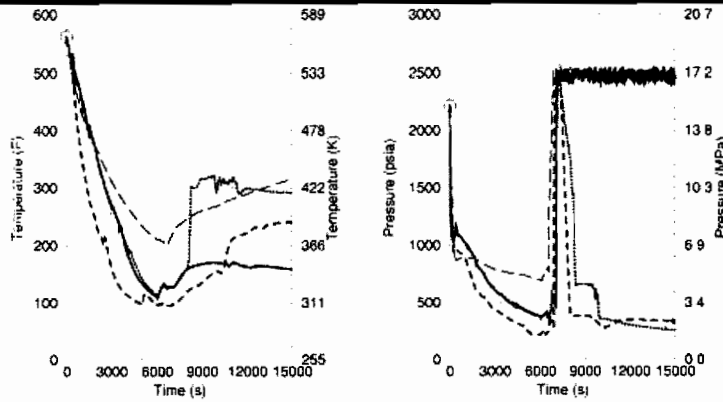
Source of Uncertainty in Dominant Transients → Stuck Open SRVs that Re-Close Later ←

- **Stuck-open SRV / reclosure type scenarios an important class of transients for Oconee only**
 - Relative contribution lowers as EFPY increases
 - Important for Oconee due to greater tendency to decouple RCS from secondary and less heat addition from steam generators into the RCS during event in B&W plants
 - Key uncertainties in this type of transient have been addressed quantitatively
 - ✓ Degree of valve opening
 - Modeled by a split fraction for fraction of valve opening size of interest to PTS assuming any size opening is equally likely
 - ✓ When valve recloses
 - Modeled by two discreet models (bins) reclosure at 3000 sec & reclosure at 6000 sec with 50% probability
 - ✓ How fast operator controls RCS pressurization
 - Modeled by different times and associated probabilities with uncertainties for operator actions. Note: different probabilities used across 3 plants; no considerable credit for success.



VG 84

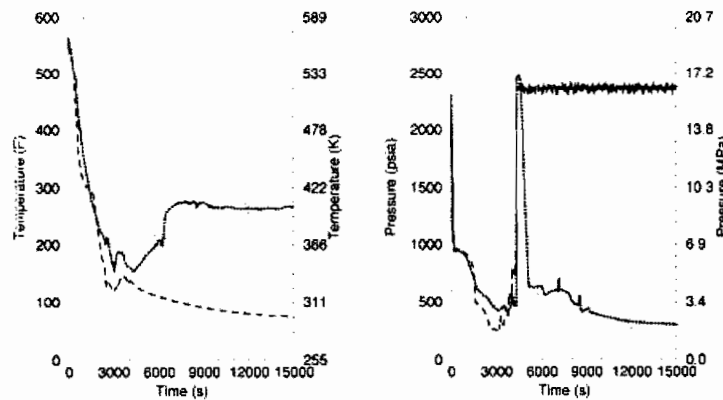
Stuck-Open Primary SRV Results Comparison



- Oconee Case 109 - RTT w/1 SO Pzr SRV (rec1 @ 6000s), no HPI throttling (VFT=7140s)
- Oconee Case 113 - RTT w/1 SO Pzr SRV (rec1 @ 6000s), HPI throttling (10 min delay) (VFT=7140s)
- - - Oconee Case 122 - RTT w/1 SO Pzr SRV (rec1 @ 6000s), HZP, HPI throttling (10 min delay) (VFT=6960s)
- - - Palisades Case 065 - RTT w/1 SO Pzr SRV (rec1 @ 6000s), HZP, no HPI throttling (VFT=xxxx)

VG 85

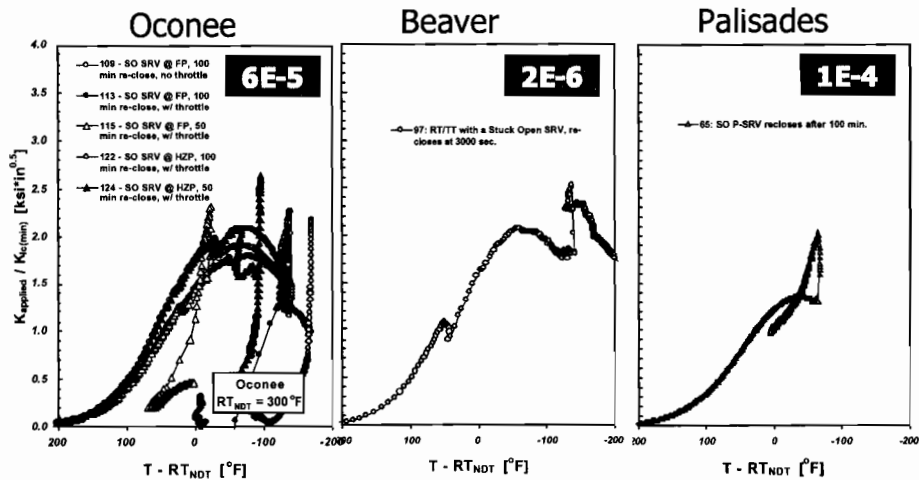
Stuck-Open Primary SRV Results Comparison



- Oconee Case 124 - RTT w/1 SO Pzr SRV (rec1 @ 3000s), HZP, HPI throttling (10 min delay) (VFT=4380s)
- - - Beaver Valley Case 097 - RTT w/1 SO Pzr SRV (rec1 @ 3000s), HZP, no HPI throttling (VFT=2490s)

VG 85

Comparison of Dominant SO Primary



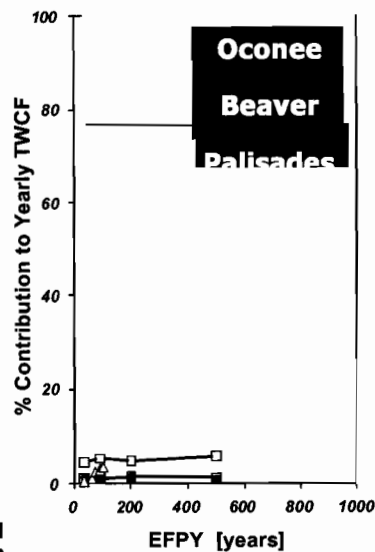
Flaw size & location, and embrittlement constant in all three analyses.

VG 87

Non-Dominant Transients → Main Steam Line Break? ←

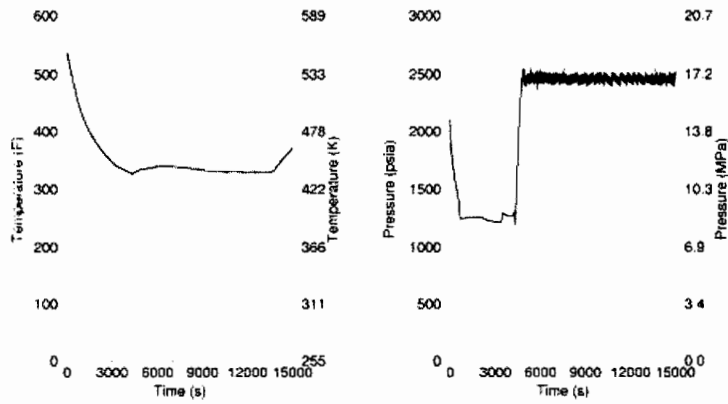
■ Main Steamline Breaks and other secondary faults (stuckopen valves, overfeeds, etc.) are relatively unimportant

- Why? generally:
 - ✓ Binning not as gross as in earlier work (current work separates large breaks from small breaks from valve opening scenarios so there is less conservatism compared with earlier studies)
 - ✓ Not as severe a transient as a LOCA
 - ✓ Realistic credit for operator actions including uncertainties on human action probabilities
- Note: Uncertainties / judgments would have to be significantly different before the small contribution of these



VG 88

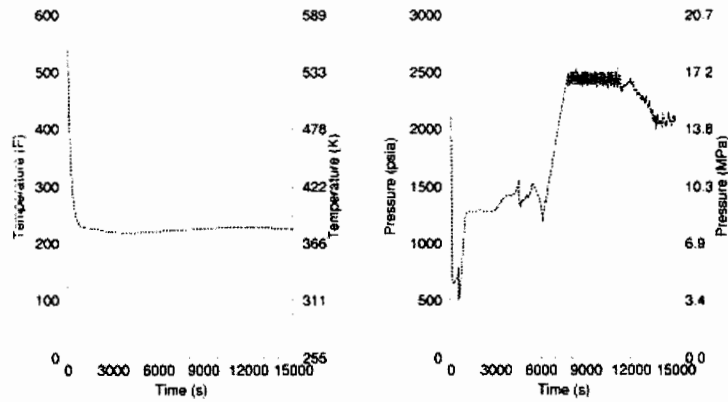
Stuck-Open Secondary SRV Results



----- Palisades Case 055 - RTT w/2 SO ADVs. AFW failure (overfeed) no HPI control (VFT=4620s)

VG 89

MSLB Results



----- Palisades Case 054 - MSLB w/MSIV failure, no HPI throttling, no AFW isolation (VFT=xxxx)

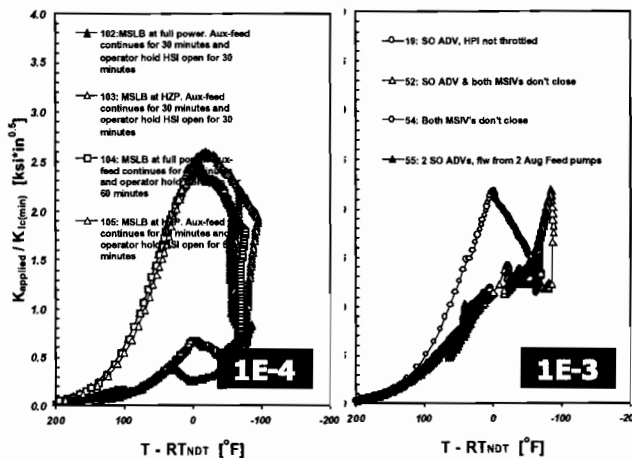
VG 90

Comparison of Dominant Secondary Side

Oconee

Beaver

Palisades

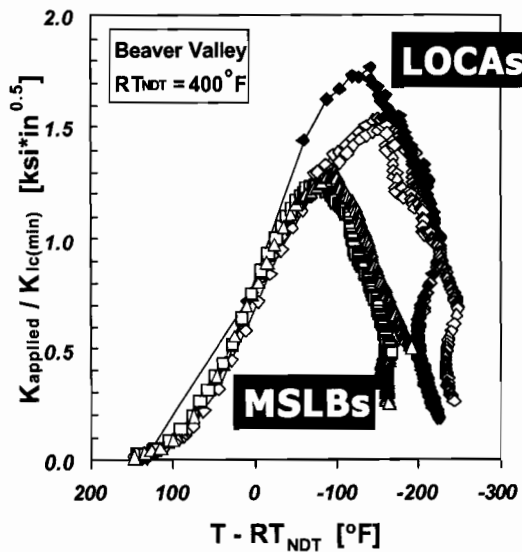


Flaw size & location, and embrittlement constant in all three analyses.

Non-Dominant Transients

→ Main Steam Line Break?

- Not as severe a transient as a LOCA

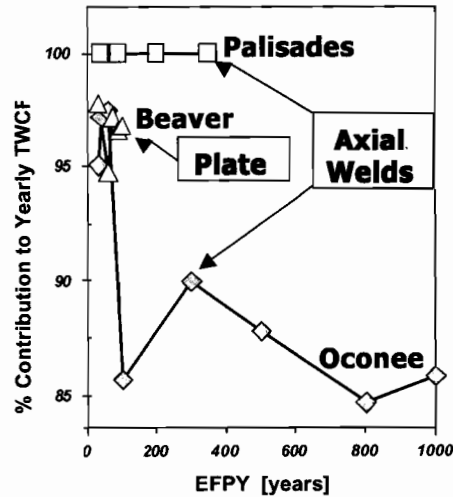


VG 92

- 7: 8-in. Surge Line Break
- 9: 16-in. Hot Leg Break
- 56: 4-in. Surge Line Break
- 102: MSLB at full power. Aux-feed continues for 30 minutes and operator hold HSI open for 30 minutes
- 103: MSLB at HZP. Aux-feed continues for 30 minutes and operator hold HSI open for 30 minutes
- 104: MSLB at full power. Aux-feed continues for 30 minutes and operator hold HSI open for 30 minutes

Dominant Material Contributors

- **Axial weld cracks dominate TWCF (~90%)**
 - Axial weld RT_{NDT} or
 - Plate RT_{NDT}
- **Circumferential weld cracks play a minor role in TWCF (~10%)**
 - Circ. weld RT_{NDT} or
 - Plate RT_{NDT}
 - Forging RT_{NDT}
- **Cracks in plates and forgings too small to play a role**



VG 95

Summary

PTS TWCF very low over any currently anticipated operating lifetime.

- **Operations**
 - LOCAs and stuck/open valves on primary side dominate PTS challenge
 - Secondary side breaks insignificant contributors
 - Holding all material factors constant, operational challenge reasonably consistent between 3 plants analyzed
 - ✓ Probability of challenge occurring
 - ✓ Fracture challenge assuming event occurs
- **Materials**
 - (Nearly) all weld flaws occur on weld fusion line
 - Axial weld flaws dominate through wall cracking frequency
 - ✓ Axial weld toughness
 - ✓ Plate toughness
 - Circ. weld flaws make minor contribution to through wall cracking frequency
 - ✓ Circ. weld toughness
 - ✓ Plate toughness
 - ✓ Forging toughness
 - Flaws in large regions of plate & forging remote from welds too small to matter

VG 96

Generic Applicability of Results

Overview

- **Consideration of external initiating events**
- **Generalization of 3 plantspecific analyses to all PWRs**

Impact of External Events

- **External events (e.g., fires, earthquakes, floods) can also cause overcooling events**
- **Actual experience indicates a small fraction of significant overcooling events involve external events (1 or 2 out of over 100 events)**
- **Performed general bounding analyses to determine worst-case contributions to TWCF (e.g., no operator credit)**
- **Findings:**
 - **Realistically: Not quantified, but judged ~~not~~ to be a significant contributor to TWCFs as compared to internal events**
 - **Bounding results: External events could result in similar TWCFs as for internal events**

99

VG 99

Example of External Event Approach

- **Internal scenario #1 description**
 - **Small LOCA**
- **Corresponding external event scenario**
 - **Seismic-induced pipe break only external event identified**
 - ✓ **0.3 g HCLF assumed (corresponds to 0.5 g median fragility) with uncertainty**
- **Based on above fragility information & seismic hazard inputs, determined seismic-induced small LOCA frequency for two sites**
 - **H. B. Robinson 1.1E-4/yr (mean)**
 - **Diablo Canyon 5.0E-4/yr (mean)**
- **Note: No accounting for seismic effect on HPI (assumed to work) & no credit for operator mitigating actions**

VG 100

Example of External Event Approach (cont'd)

- **Internal scenario #2 description**
 - **Reactor trip with single stuck open PORV**
- **Corresponding external event scenarios**
 - **Seismic-induced PORV opening (e.g., relay chatter)**
 - ✓ **0.3 g HCLF assumed (corresponds to 0.5 g median fragility) with uncertainty**
 - **Fire-induced PORV opening (e.g., hot short)**
- **Based on above fragility information and seismic hazard inputs, determined seismic-induced PORV open scenario frequency for two sites**
 - **H. B. Robinson $1.1E-4$ /yr (mean)**
 - **Diablo Canyon $5.0E-4$ /yr (mean)**
- **Note: No accounting for seismic effect on HPI (assumed to work) & no credit for operator mitigating actions.**

VG 101

Example of External Event Approach (cont'd)

- **Based on $2E2$ /yr fire frequency (Aux Bldg electrical cabinets experience), 0.5 hot short probability, & 0.1 factor to affect specific cabinet/circuit of concern = $1E3$ /yr fire-induced PORV open scenario**
- **Note: no accounting for operator actions such as to close valve/block valve, or other mitigating actions.**
- **Sum of seismic and fire-induced scenario frequencies is $<2E-3$ /yr.**

VG 102

Resulting Comparison of Internal vs. External Event TWCFs

(External Event TWCFs \leq highest Internal Event TWCFs)

Scenario	Internal Event Frequencies	Internal Event CPFs @ ≤ 100 EFPY	Internal Event TWCFs	External Event Frequencies	External Event CPFs @ ≤ 100 EFPY	External Event TWCFs
Small LOCA	< 2E-3/yr	< E-5	< 2E-8/yr	< 5E-4/yr	< E-5	< 5E-9/yr
Stuck-open PORV	< 3E-4/yr	< E-5	< 3E-9/yr	< 2E-3/yr	< E-5	< 2E-8/yr

VG 103

Generalization of Results to Other PWRs

- **Ongoing**
- **Looking at 5 other PWRs that are among the most embrittled plants**
- **Approach:**
 - **Compare plant design and operational features that matter most to the 3 plants analyzed**
 - **Qualitatively judge potential impact on PTS results based on these comparisons**
- **Assuming LOCAs should still dominate, results should be similar since frequencies would not change, T-H responses should be similar to extent plant features are similar, and CPFs should not be drastically different from plants analyzed (Beaver Valley and Palisades are also among the more embrittled plants)**

104

VG 104

Plants Covered in Generalization Step

- Plants ranked in terms of un-irradiated $RT_{NDT} +$ Eason embrittlement shift at 32EFPY.
- Circ. welds NOT considered in ranking

Tolerance to a PTS Challenge	Plant Name	Most Embrittled Material	$RT_{NDT} +$ Irradiation Shift at 40 years [F]
1	SALEM 1	PLATE	204
2	BEAVER VALLEY 1	PLATE	194
3	TMI-1	AXIAL WELD	188
4	PORT CALHOUN	AXIAL WELD	181
5	PALISADES	AXIAL WELD	179
6	DIABLO CANYON 1	AXIAL WELD	171
7	DIABLO CANYON 2	PLATE	170
8	SEQUOYAH 1	FORGING	167
9	WATTS BAR 1	FORGING	164
10	ST. LUCIE 1	AXIAL WELD	164
11	SURRY 1	AXIAL WELD	163
12	INDIAN POINT 2	PLATE	162
13	GINNA	FORGING	161
14	POINT BEACH 1	AXIAL WELD	159
15	FARLEY 2	PLATE	158
16	MCGUIRE 1	AXIAL WELD	158
17	OCONEE 1	AXIAL WELD	157
18	NORTH ANNA 2	FORGING	155
19	SHEARON HARRIS	PLATE	153
20	NORTH ANNA 1	FORGING	153
21	COOK 2	PLATE	152
22	SALEM 2	AXIAL WELD	148
23	CRYSTAL RIVER 3	AXIAL WELD	141
24	CALVERT CLIFFS 2	PLATE	139
25	ROBINSON 2	PLATE	138
26	COOK 1	AXIAL WELD	138
27	FARLEY 2	PLATE	133
28	FARLEY 1	PLATE	133
29	ARKANSAS NUCLEAR 1	AXIAL WELD	129
30			

The estimated tolerance to a PTS challenge increases as the number in the next column increases (i.e., the plants with the lowest ranking have the most embrittled materials).

Notes:
Plants analyzed in the PTS re-evaluation
Plants being considered for re-evaluation

VG 105

General Information Categories Examined in Generalization Step

- Secondary Breaches
- Secondary Overfeed
- LOCA Related
- PORV and SRV Related
- Feed and Bleed Related

VG 106

Secondary Breaches

- **Number of MSIVs**
- **Isolation capability with regard to other paths (e.g., ADVs, SDVs, TBVs)**
- **Identification of procedures, steps, and location of steps within procedures that ensure likelihood of early identification & isolation of faulted steam generators**
- **Operator training or procedural allowances that support early isolation of steam generators**
- **Location/size of steamline flow restrictors**
- **Key assumptions relative to MSLB analysis**
- **AFW/MFW control during steamline break (or similar) [e.g., auto isolate, self-limiting flow, manual only control]**
- **Does turbine driven pump isolate in MSLB?**

VG 107

Secondary Overfeed

- **Information on the feed (MFW and AFW/EFW) capabilities to the steam generators including inventory of water available to continue MFW or AFW/EFW**
- **Information on normal steam generator inventory**
- **Information on possible feed temperatures for all feed sources (especially how cold they could be)**

VG 108

LOCA Related

- Allowable range of safety injection water temperatures
- Information to estimate recirculation water temperature
- Safety injection water source size (i.e., inventory)
- Safety injection flow rate versus LOCA break size
- Charging, HPI, LPI shutoff heads
- Actuation requirements for containment spray and flow rate once running
- Impact on HPI, LPI, charging when sump switchover occurs (which pumps on vs. off)
- Any significant changes in flow rates going from injection to recirculation?
- Accumulator (SIT, CFT) discharge pressure

VG 109

PORV and SRV Related

- Number and sizes of PORVs & SRVs, whether plant operates with PORV block valves normally shut, and if there are any auto operation features of the PORVs
- Instrumentation available (e.g., acoustic monitors, differential pressure, etc.) to identify open PORVs or SRVs and to notice if they have closed
- Procedure for addressing LOCAs resulting from stuck open PORVs or SRVs
- Procedures for addressing the sudden closure of such valves including throttle/terminate SI guidance
- Training material associated with sudden closure events
- Operating characteristics of charging when pressurizer level goes back high (e.g., stop, keep running?)
- How many SRVs must open before likely initiation of containment sprays?

VG 110

Feed and Bleed Related

- **Number of AFW/EFW pumps/flow paths versus minimum success criteria for adequate feed to the steam generators**
- **EOP criteria for initiation of feed and bleed**
- **Number of PORVs opened out of total available (or even SRVs if pumps can open SRVs) when in feed and bleed mode**
- **Number of HPI pumps used in feed and bleed and is actual flow rate equivalent to number of pumps (e.g., at BV, they attempt to use all pumps but design only allows 2 out of 3 pumps to be aligned for injection at any one time)**

VG 111

Summary

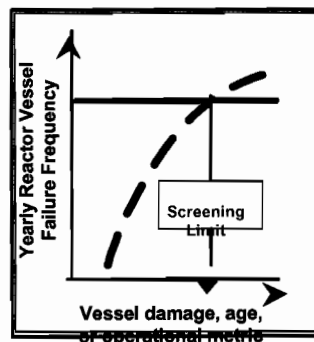
- **External events**
 - **Contribution small relative to internal events**
- **Generalization**
 - **5 plants selected → highest embrittlement**
 - **Question for plants developed based on understanding of important contributors developed so far**

VG 112

Risk-Informed Reactor Vessel Failure Frequency Acceptance Criteria

Reactor Vessel Failure Frequency (RVFF)

- **RVFF criterion needed for two purposes:**
 - Support definition of RPV embrittlement criterion
 - Provide acceptance criterion for safety analysis
- **Current metric and criterion established in RG 1.154:**
$$RVFF \equiv TWCF$$
$$RVFF^* = 5 \times 10^6 / \text{ry}$$
- **Limited scope activity to revisit metric/criterion in light of recent risk-informed regulation initiatives**



Task Activities

- **Identification of options**
- **Scoping study of postvessel failure accident progression**
 - Qualitative evaluation of technical issues
 - Review of pilot plant calculations for T/H conditions
 - Limited calculations
- **Status reports and meetings**
 - SECY-02-0092 (5/10/02)
 - ACRS (7/10/02), public meetings (10/17/02; 1/31/03)
 - Chapter 5, draft NUREG (12/31/02)
- **Focus on acceptability => activities are largely independent of plant-specific studies**

VG 115

RVFF Acceptance Criteria

Principles in Developing and Evaluating Options

- **Consistency with intent of original rule**
 - **Low risk level**
 - **Low relative contribution**
- **Consistency with recent risk-informed initiatives**
 - **Risk metrics**
 - **Risk criteria**
 - **Consideration of defense-in-depth**

VG 116

RVFF Acceptance Criteria Options (SECY-02-0092)

- **Definition of RVFF**
 - RVFF = f(PTS-induced RPV throughwall crack)
 - RVFF = f(PTS-induced crack initiation)
- **RVFF acceptance limits**
 - RVFF* = 5×10^6 /ry
 - RVFF* = 1×10^6 /ry
 - RVFF* = 1×10^5 /ry

VG 117

Post-SECY Discussions

- **Budgeting process: focus effort on assessing RVFF for pilot plants**
- **ACRS Letter (7/18/02; ML0220406120)**
 - RVFF should be based on considerations of LERF (and not CDF)
 - Current LERF surrogate goal is not proper starting point
 - "...source terms used to develop the current goal do not reflect air-oxidation phenomena that would be a likely outcome of a PTS event."
 - **Options:**
 - ✓ Develop acceptance criterion from prompt fatality safety goal
 - ✓ Use a frequency-based approach to develop RVFF* to provide assurance that PTS-induced RPV failures are very unlikely
 - **ACRS' expectation: RVFF* will be substantially smaller than option proposed in SECY02-0092**

VG 118

Definition of RVFF

It is appropriate to define RVFF as the frequency of through-wall cracks (TWCF)

- **TWCF is a more direct indicator of risk than is the vessel cracking initiation frequency**
- **The current technology for predicting crack arrest is reasonably robust**
 - **Laboratory-scale experiments**
 - **Scaled-vessel experiments**

VG 119

Scoping Study - Key Questions

- **Is a PTS-induced RPV failure likely to lead to melted fuel?**
- **Is a PTS-induced RPV failure likely to lead to a large, early release?**
- **Is the release spectrum (frequency/sequence) for PTS-induced large, early releases significantly worse than that associated with risk-significant, non-PTS-induced scenarios?**

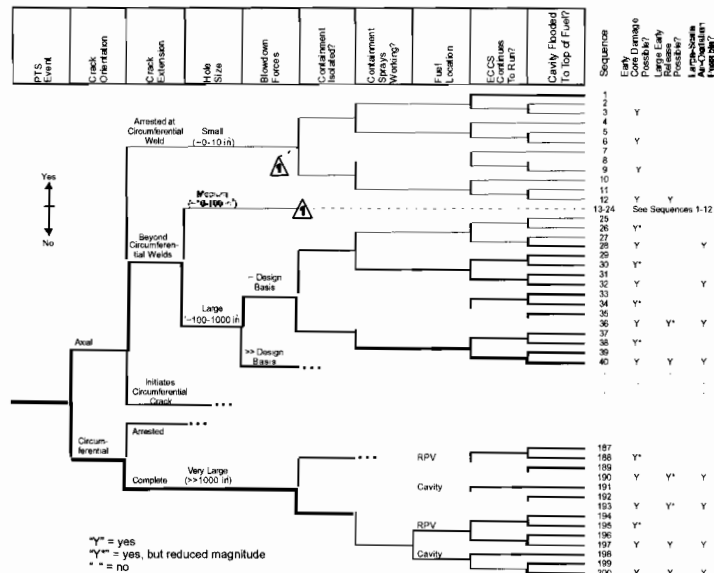
VG 120

Scoping Study - Approach

- Refine SECY-02-0092 list of technical issues
- Develop accident progression event tree (APET) to support identification, representation and discussion of technical issues
- Evaluate current state of knowledge regarding technical issues
- Context for evaluations:
 - Focus on pilot plants; some consideration of plants addressed in generalization task
 - Whether/how PTS changes accident progression

VG 121

Accident Progression Event Tree (APET)



VG 122

Sequence	Early Core Damage Possible?	Large Early Release Possible?	Large Scale AV-Generator Problem?
1			
2			
3			
4	Y		
5			
6	Y		
7			
8			
9	Y		
10			
11			
12	Y	Y	
13-24	See Sequences 1-12		
25			
26	Y*		
27			
28	Y		Y
29			
30	Y*		
31			
32	Y		Y
33			
34	Y*		
35			
36	Y	Y*	Y
37			
38	Y*		
39			
40	Y	Y	Y
...			
...			
...			
187			
188	Y*		
189			
190	Y	Y*	Y
191			
192	Y		
193	Y	Y*	Y
194			
195	Y*		
196			
197	Y	Y	Y
198			
199			
200	Y	Y	Y

"Y" = yes
 "Y*" = yes, but reduced magnitude
 "*" = no

VG 123

Potential Sources of Dependence Between Top Events

- Plant systems
- RPV movement
- Fragments
- Fuel movement

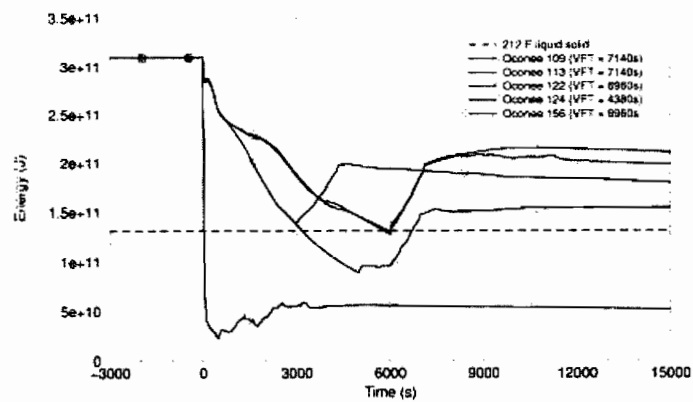
VG 124

Plant Conditions at RPV Failure

- **Power available, cooling systems running (injection mode)**
- **LOCA events: RCS cooling, depressurizing**
 - MLOCA - RPV failure at ~1530 min (40 EFPY)
 - LLOCA - RPV failure at ~510 min (40 EFPY)
- **Stuck-open SRV events: RCS at SRV setpoint**
RPV failure at ~60120 min (40 EFPY)

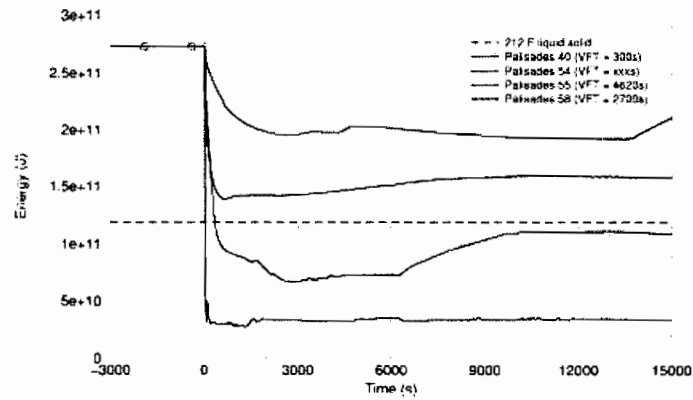
VG 125

Blowdown Potential After RPV Failure



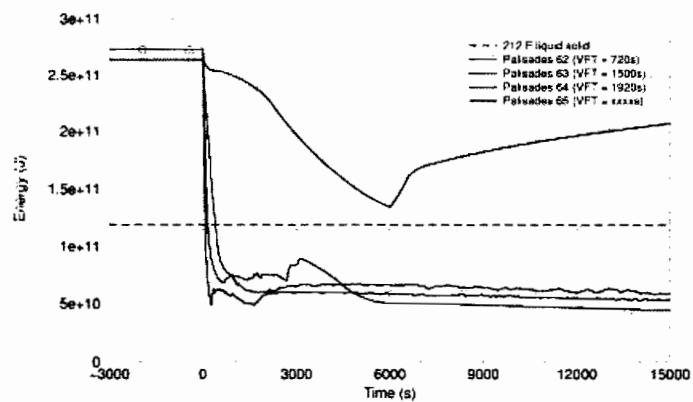
VG 126

Blowdown Potential After RPV Failure



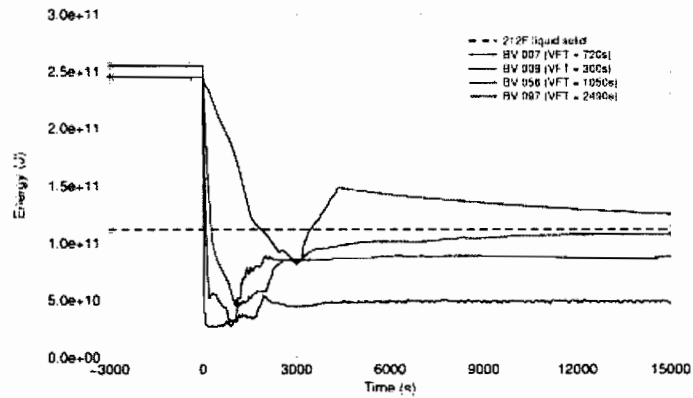
VG 127

Blowdown Potential After RPV Failure



VG 128

Blowdown Potential After RPV Failure



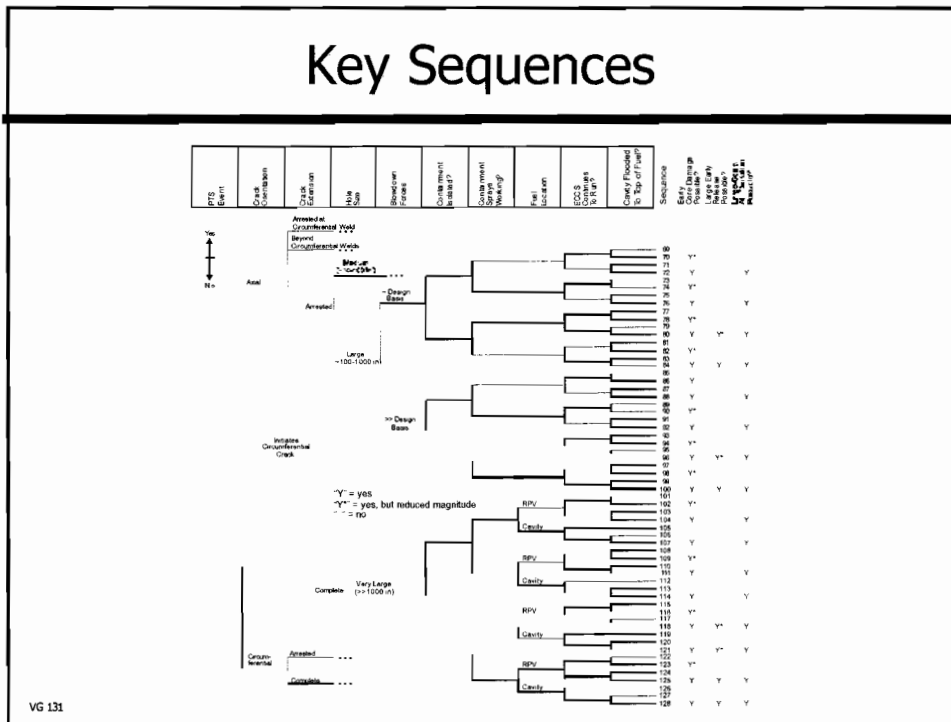
VG 129

Pressure Differentials After RPV Failure

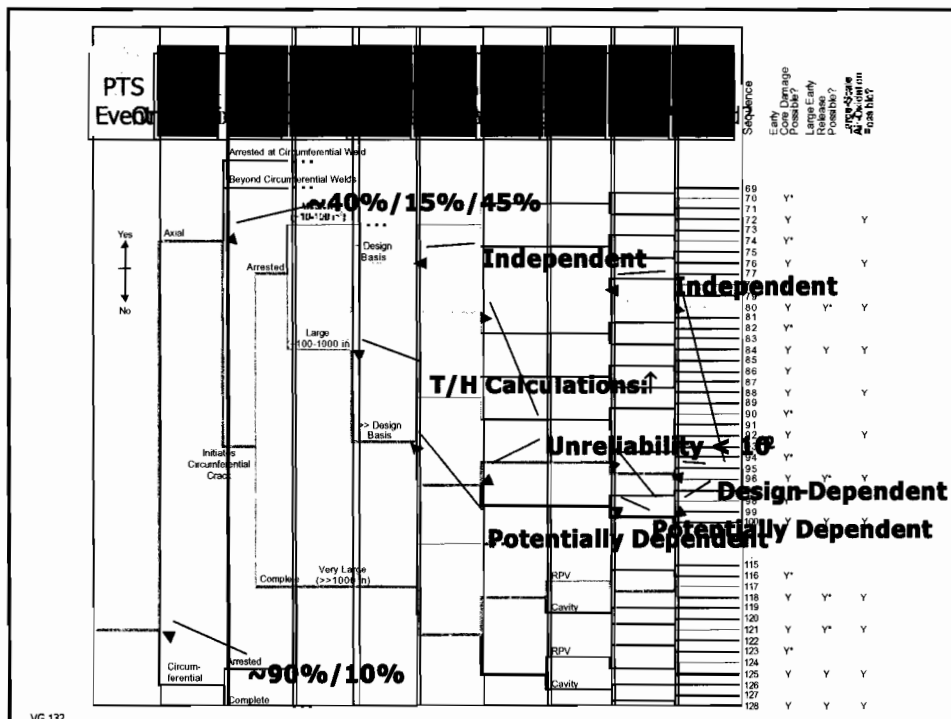
Plug-in from Dave Bessette

VG 130

Key Sequences



VG 131



VG 132

Observations

- Accident energetics are more benign than those of some other scenarios previously studied (e.g., HPME).
- Containment spray failure probability may decrease for PTS events.*
- Likelihood of fuel cooling dependent on reactor cavity design
 - Cavity flooding above top of fuel expected for some plants
 - For other plants, ECCS may not be sufficient to cool fuel
- Blowdown forces on RPV and internals likely to be the same order of magnitude or bounded by design basis LLOCA

*For some plants, this may be dependent on plant changes in response to GSI-191.

VG 133

Scoping Study Conclusions

- The conditional probability of early fuel damage (given a PTS-induced RPV failure) appears to be
 - Extremely small for plants with cavities likely to be flooded
 - Non-negligible for other plants
- The conditional probability of early containment failure and a large, early release (given a PTS-induced RPV failure) appears to be very small for all plants
- Should a PTS-induced large, early release occur, such a release may involve a large-scale air-oxidation source term

VG 134

Implications for RVFF*

- **RVFF* = 1×10^6 /ry is consistent with philosophy of original PTS rule, with ACRS guidance, and with Safety Goal Policy Statement**
 - Assures a low level of risk associated with PTS events
 - Assures small relative contribution to acceptable risk
 - More limiting with respect to core damage than RG 1.174/Option 3 criterion for CDF
 - Consistent or conservative with respect to PHOs
- **Expectation: RPV embrittlement limits will be established in a risk informed manner**

VG 135

Summary Conclusions

- **RVFF \equiv TWCF**
- **RVFF* = 1×10^6 /ry**
- **RVFF* should be compared against mean of plant specific RVFF distribution**

VG 135

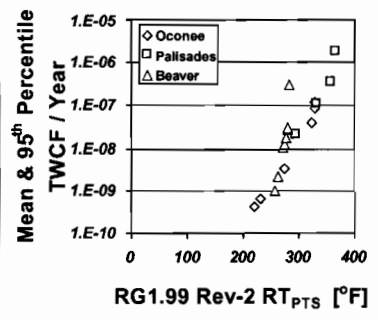
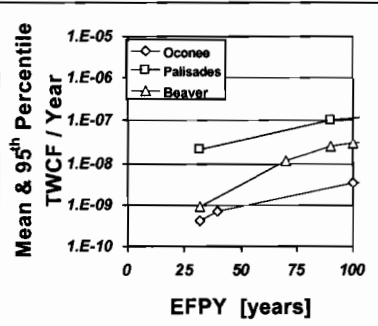
PTS Screening Limit Considerations

Outline

- **PTS risk at likely operational lifetimes**
- **Operating challenge considerations**
- **Materials considerations**
- **A physically motivated embrittlement metric**

PTS Risk is Low

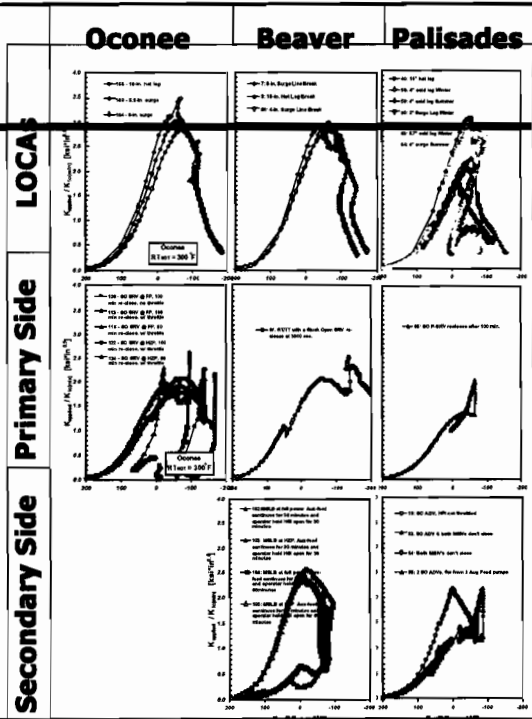
- Over realistic operational lifetimes, the estimated TWCF for these plants is VERY small (from 1×10^9 to 5×10^8)
- At the current screening criteria the yearly TWCF is approximately 1×10^8
- Two of these plants are among the most embrittled in operation



VG 141

Operational Considerations

All material factors held equal, the severity of PTS challenge is remarkably similar between the plants studied



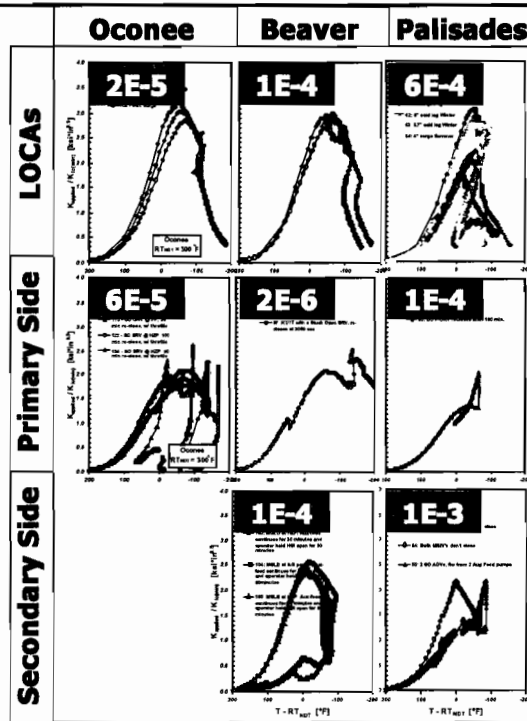
VG 142

Operational Considerations

All material factors held equal, the severity of PTS challenge is remarkably similar between the plants studied

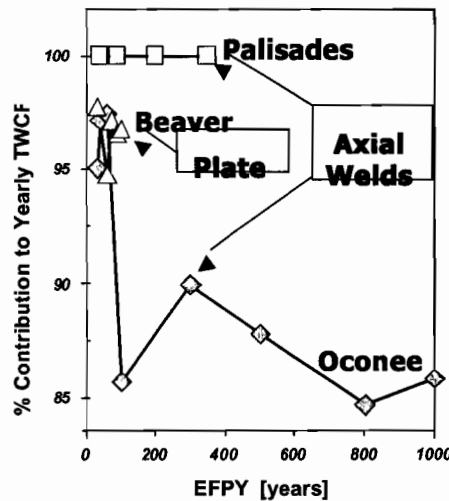
The frequency of challenge is also similar between plants

VG 143



Axial Weld Flaws Dominate TWCF

- **Axial weld cracks dominate TWCF (~90%)**
 - Axial weld RT_{10T} or
 - Plate RT_{10T}
- **Circumferential weld cracks play a minor role in TWCF (<10%)**
 - Circ. weld RT_{10T} or
 - Plate RT_{10T}
 - Forging RT_{10T}
- **Cracks in plates and forgings too small to play a role**



VG 144

Characteristics of a Physically-Motivated Embrittlement Metric

A causal relationship should exist between the embrittlement metric and TWCF ... so ...

- Axial weld / plate properties dominate the metric,
- Circ weld / forging / plate properties play minor role
- Relevant fluence is that along the welds
- Large regions of plate / forging remote from welds don't matter

VG 149

Embrittlement Metric Formula

3-10% Weight on Circ. Weld or Forging Properties

$$W_{P-F-CW} \equiv 0.033 \cdot n_{CIRC}$$

90-97% Weight on Axial Weld or Plate Properties

$$W_{P-AW} \equiv 1 - W_{P-F-CW}$$

$$RT_{NDT}^* \equiv W_{P-F-CW} \cdot RT_{P-F-CW} + W_{P-AW} \cdot RT_{P-AW}$$

Circ. Weld / Forging Reference Temperature Depends on Most embrittled material, number of welds, max fluence along weld

$$RT_{P-F-CW} \equiv \text{MAX} \left\{ (RT_{NDT(u)}^{P-F} + \Delta T_{30}^{P-F}(f_{max}^{CW})), (RT_{NDT(u)}^{CW} + \Delta T_{30}^{CW}(f_{max}^{CW})) \right\}$$

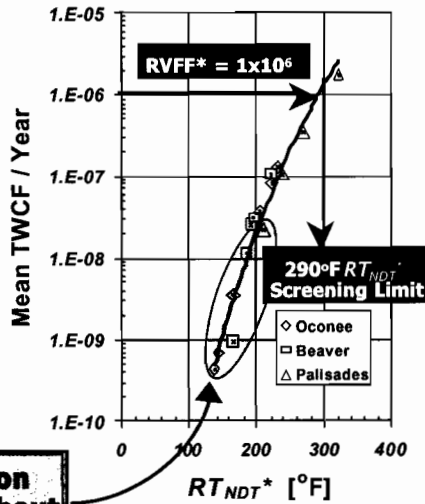
Axial Weld / Plate Reference Temperature Depends on Most embrittled material, length of weld, max fluence along weld

$$RT_{P-AW} \equiv \frac{\sum_{i=1}^{n_{AXIAL}} \{ L_i \cdot \text{MAX} \left[(RT_{NDT(u)}^P + \Delta T_{30}^P(f_{max}^{AW})), (RT_{NDT(u)}^{AW} + \Delta T_{30}^{AW}(f_{max}^{AW})) \right] \}}{\sum_{i=1}^{n_{AXIAL}} L_i}$$

VG 150

Suggested Embrittlement Metric (& Significance)

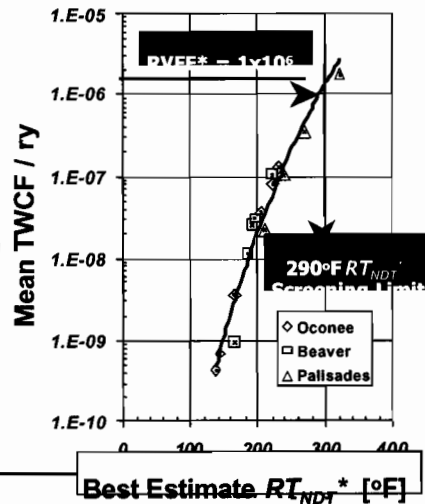
- **VERY LOW** predicted TWCF values suggest that revision of the PTS rule & screening criteria is justified
- A yearly RVFF limit of 1×10^6 events corresponds to a weighted RT_{NDT} value (RT_{NDT}^*) of 290°F
- Since RT_{NDT}^* is about 90°F less than RT_{PTS} , this suggests that a 80°F to 110°F increase of the current 10CFR50.61 screening limit is possible



Results suggest that operation possible for 60 to 80 years without close approach to RVFF* limit.

RT_{NDT}^* Screening Limit for PTS

- **Margin on RT_{NDT}^* neither necessary nor appropriate**
 - Maximum material uncertainties accounted for explicitly in FAVOR calculations— any plant state of knowledge will be better than we simulated
- **290°F RT_{NDT}^* limit pertains only to RT_{NDT}^* estimated from**
 - RVID $RT_{NDT(u)}$ values
 - Cu
 - Ni
 - P



Recommendations & Significance

Conclusions

- **These analyses provide a technical basis to recommend revision of the PTS rule**
 - **Two of the most embrittled plants in fleet have a TWCF at or below 5×10^{-6} at end of license extension (60 years)**
 - **At the 10CFR50.61 RT_{DT} screening limits these plants have a TWCF of 1×10^{-6} (vs. RG 1.154 at 5×10^{-6})**
- **Analysis supports a revised screening limit of**
 - **290°F on a weighted RT_{NDT} value**
 - ✓ Axial welds & plates dominate
 - ✓ Circ welds and forgings minor contributors
 - **This limit is 80F to 110F higher than current 10CFR50.61 limits on RT_{PTS}**

154

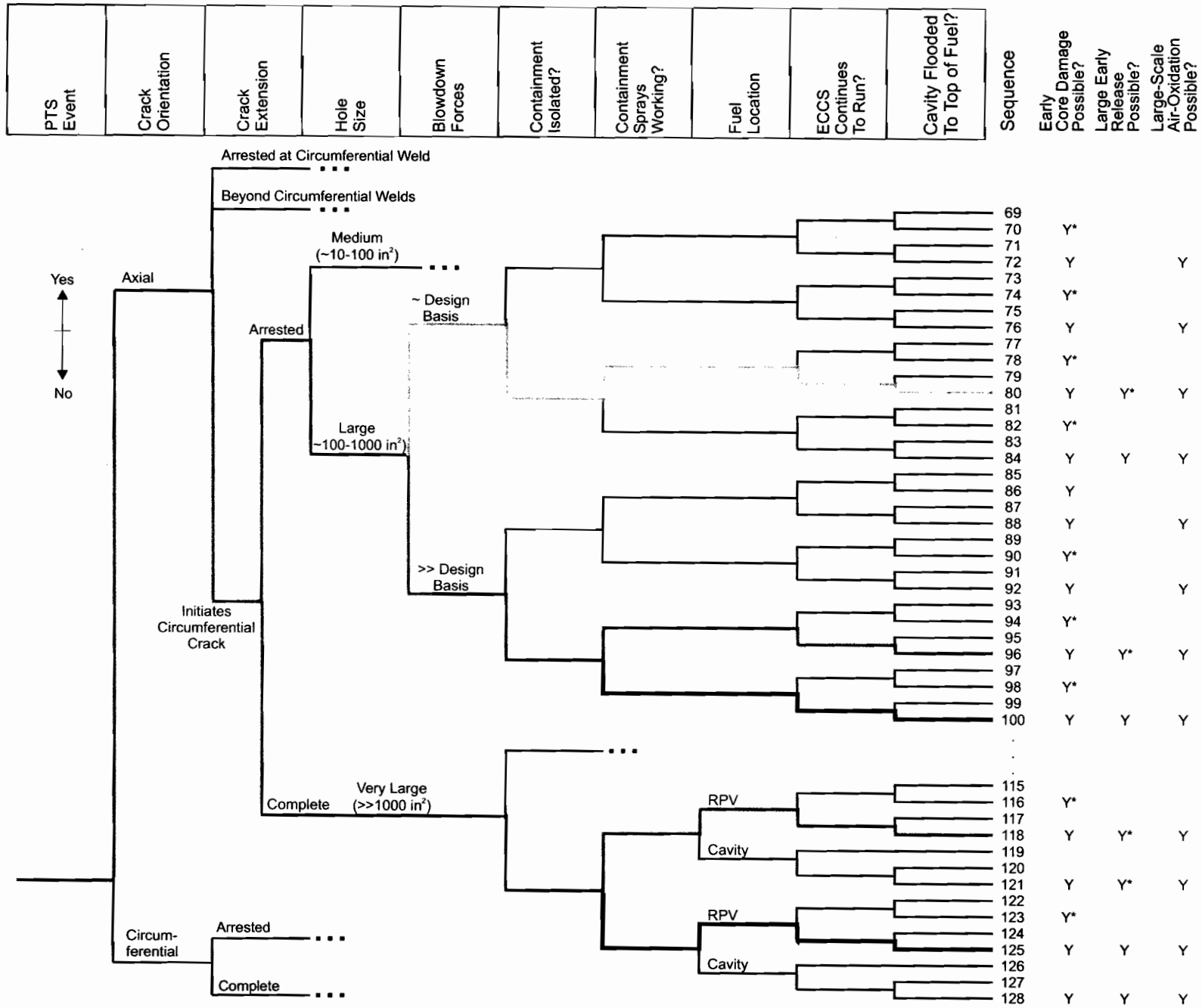
VG-154

On-Going Activities

- **RES activities**
 - **Calvert cliffs**
 - **Generalization to all plants**
 - **Sensitivity studies & a more detailed examination of current results**
 - **Favor V&V**
 - **External peer review of project**
 - **Implications for operational limits (10CFR Appendix G)**
- **NRR activities**
 - **RES Draft NUREG sent to NRR on 1-21-02**
 - **NRR comments due by 3-31-03**
 - **Decision to proceed with rulemaking?**

VG 155

2:00 pm
 Section V.
 Sim.



2nd Ho.
 V 23