ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
479TH MEETING
PTS RULE SCREENING CRITERION REEVALUATION: DETERMINATION OF UNCERTAINTIES
FEBRUARY 1, 2001
ROCKVILLE, MARYLAND

PRESENTATION SCHEDULE

TOPIC	SPEAKER	TIME
I. Introduction	W. Shack, Chairman	8:35 - 8:45 a.m.
II. NRC-RES Presentation: Status of PTS Rule Screening Criterion Re-evaluation • Determination of Uncertainties	M. Mayfield, RES N. Siu, RES	8:45 - 10:05 a.m.
III. Committee Discussion		10:05 - 10:15 a.m.
VI. Recess		10:15 a.m.

CLOSEOUT OF GENERIC SAFETY ISSUE 152
DESIGN BASIS FOR VALVES THAT MIGHT BE SUBJECTED TO SIGNIFICANT BLOWDOWN LOADS

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

KENNETH KARWOSKI
Division of Engineering Technology
Office of Nuclear Regulatory Research
301-415-6933

February 1, 2001

SEEK ACRS ENDORSEMENT OF THE STAFF'S PROPOSAL TO CLOSE THE ISSUE

ORIGIN OF GENERIC SAFETY ISSUE (GSI) 152

GSI-152 was raised by the ACRS in 1989 during the review of activities related to GSI-87, "Failure of HPCI Steam Line Without Isolation"

GSI-87 dealt with the ability of the HPCI steam line isolation valves to isolate a postulated pipe break

GSI-87 was subsequently closed with Generic Letter (GL) 89-10, and its supplements

GL 89-10 focused on the ability of motor operated valves (MOVs) to operate consistent with their approved design basis

The focus of GSI-152 was the adequacy of the design basis for valves that might be subjected to significant blowdown loads (i.e., pipe breaks)

"the requirement for safety-related valves to move against high differential pressures and/or high flows experienced during a large downstream pipe break may not have been specified in the design bases"

GL 89-10

Testing done by the Office of Nuclear Regulatory Research for the closure of GSI-87 showed weaknesses in valve performance attributable both to motors and to valve mechanisms

GL 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance”, did not address the adequacy of the design bases, only adequacy of valve performance in meeting the approved design bases

An examination of the design bases was included as part of GL 89-10 but the evaluation of the adequacy of the design bases was a separate issue, (i.e. GSI-152)

Industry sponsored research which confirmed the weaknesses identified in NRC testing

Licensees, reactor manufacturers, and industry groups developed “working groups” to address these weaknesses

GL 89-10 had 7 supplements spanning from 1989 to 1996

ADEQUACY OF DESIGN BASIS

Although GL 89-10 was focused on the ability of MOVs to operate as designed, the issue of adequacy of the design bases contained in GSI-152 was captured by industry initiatives, and confirmed during NRC inspections which included the reasonableness of the design bases

NRC inspections conducted to examine GL 89-10 programs also addressed the adequacy of design bases and whether they included the differential pressure associated with a large downstream break–the subject of GSI-152

> A priority focus was on the high risk significant valves of HPCI, RCIC, and RWCU MOVs (GL 89-10 Supplement 3)

Lessons learned regarding valves other than MOVs were reported to the licensees by NRC and were voluntarily incorporated by the industry groups into their programs addressing all valve types

CLOSURE OF GSI-152

The ACRS Subcommittee on Mechanical Components was briefed on industry MOV activities and other MOV related items on October 5, 1993

The Subcommittee Chairman who had originally raised the concern stated he was satisfied the issue was adequately addressed and could be closed

RES confirmed the actions taken by licensees by reviewing historical documents including licensee documents submitted in response to GL 89-10, GL 89-10 inspection guidance, and GL 89-10 inspection reports

Based on the actions taken by industry in response to GL 89-10 and confirmed in NRC inspections, there is sufficient evidence to close GSI-152, and no further action is necessary

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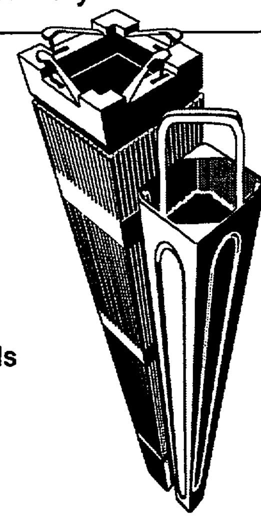
Siemens PWR Appendix K SBLOCA Analysis



Presented by: Jerry Holm
Joe Kelly

Advisory Committee on Reactor Safeguards
February 1, 2001

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Agenda

- Introduction — Jerry Holm
- S-RELAP5 Code — Joe Kelly
  - Relationship to RELAP5
  - Summary of Siemens Enhancements
- Appendix K Small Break LOCA Methodology — Joe Kelly
  - Summary of Methodology
  - Summary of Validation
- Conclusion — Jerry Holm

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Methodology Vision

- PWR Methodology Vision
  - SBLOCA - Appendix K SBLOCA Using S-RELAP5 (under review)
  - LBLOCA - Realistic LBLOCA Methodology Using S-RELAP5 (submittal in 2001)
  - Non-LOCA - Non-LOCA Methodology Using S-RELAP5 (under review)
- BWR Methodology Vision
  - LBLOCA and SBLOCA - Appendix K Using S-RELAP5 (future development)
  - Non-LOCA - Non-LOCA Methodology Using S-RELAP5 (future development)

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Information Provided to Support Review Process

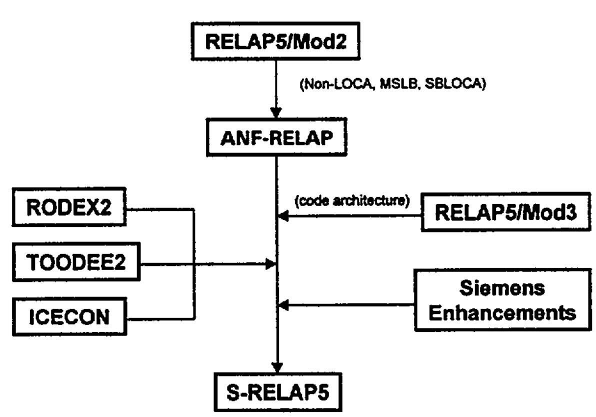
- Topical Report - PWR Appendix K SBLOCA Methodology
- Supporting Documentation
  - Models & Correlations Manual
  - Programmer's Guide
  - Input Requirements Manual
  - Code Source and Executable
  - Sample Problem Input
- Presentation to NRC
- Presentations to ACRS Thermal-Hydraulic Subcommittee
- Response to RAIs

S-RELAP5 Code for Appendix K SBLOCA Analysis

- Outline
  - S-RELAP5 T/H Code
    - Relationship to RELAP5
    - Summary of Siemens Enhancements
  - SBLOCA Appendix K Methodology
    - Methodology Overview
    - Validation Matrix
      - SBLOCA Example: BETHSY 9.1b (ISP-27)

S-RELAP5 Code for Appendix K SBLOCA Analysis

- Relationship to RELAP5 Codes:



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S-RELAP5 Code for Appendix K SBLOCA Analysis

- Summary of Siemens Enhancements:
  - Mass Conservation
    - Improved numerics to minimize mass error during long-term transients
  - Energy Conservation
    - Reformulated energy equation to avoid error associated with flow across large pressure drop
  - Momentum Conservation
    - Implemented 2-D component to avoid flow anomalies associated with cross-flow junctions (e.g., reactor core)
  - Constitutive Models
    - Numerous upgrades (primarily for LBLOCA) and modifications to vertical stratification model that improve loop seal clearing behavior

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S-RELAP5 Code for Appendix K SBLOCA Analysis

- Long-Term Mass Conservation
  - Results from integral assessments and SBLOCA sample problem:

	Transient Time (sec.)	No. of Time Steps	Mass Error (%)
Semiscale	300	$3.0x10^4$	$-1.76x10^{-4}$
LOFT	1500	$1.5x10^5$	$-8.22x10^{-6}$
BETHSY	7690	$1.3x10^6$	$1.66x10^{-3}$
PWR Sample Problem	3500	$3.5x10^5$	$-1.31x10^{-5}$

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S-RELAP5 Code for Appendix K SBLOCA Analysis

- Methodology Overview
  - Siemens defines methodology as the combination of codes used and the application of those codes in the performance of the analysis
  - Methodology is encapsulated in an analysis guideline and quality assurance procedure that:
    - Specifies the plant model nodalization
    - Ensures Appendix K conservatisms are applied
    - Prescribes additional Siemens conservatisms to be applied
      - (e.g., loop seal modeling & diesel start time)
    - Constrains adherence to guidelines by analysts

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S-RELAP5 Code for Appendix K SBLOCA Analysis

- SBLOCA Analysis
  - Four major factors affecting calculated PCT:
    - Determination of limiting single failure
      - Usually loss of diesel generator => only one HHSI available
    - Fuel Cycle
      - Limiting condition is normally EOC with top-skew power profile
    - Break Size
      - Spectrum performed to find limiting case where mass loss is greater than SI make-up and depressurization rate low enough to prolong transient => significant core uncovery
    - Loop Seal Clearing
      - PCT affected by which loop and by how many loops clear

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S-RELAP5 Code for Appendix K SBLOCA Analysis

- S-RELAP5 Validation Matrix
  - General Matrix
    - Selection of separate effects and integral effects tests performed and documented for every code version
  - SBLOCA
    - Integral and separate effects tests that is part of SBLOCA submittal
  - Non-LOCA Transients
    - Integral effects tests that are part of non-LOCA submittal
  - Realistic LBLOCA
    - Extensive PIRT based assessment matrix currently being performed

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S-RELAP5 Code for Appendix K SBLOCA Analysis

- SBLOCA Validation Matrix
  - BETHSY Test 9.1b
  - Semiscale Test S-UT-8
  - LOFT LP-SB-03
  - UPTF Loop Seal Clearing
  - 2-D Flow Tests

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S-RELAP5 Code for Appendix K SBLOCA Analysis

- SBLOCA Validation Example:
  - BETHSY 9.1b (ISP-27)
    - BETHSY is a full-height, 1/100 scale model of a 3-loop PWR
    - Test 9.1b models a 2" break with no HHSI:
      - Deep core uncovery and rod heat-up
  - S-RELAP5 Assessment
    - Input model follows proposed SBLOCA modeling guidelines,
      - Loop seals (broken & 1 intact) modeled so that #2 clears
      - Critical flow model => realistic estimate of break flow
      - 1-D core model

⟶ Excellent comparison with data for core level and PCT

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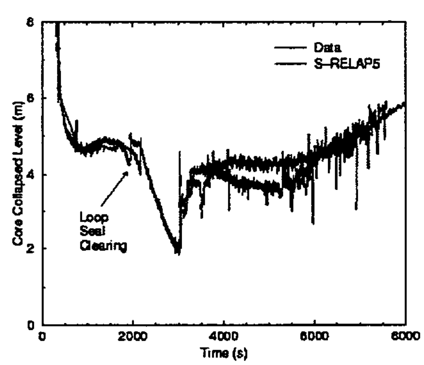
S-RELAP5 Code for Appendix K SBLOCA Analysis

- BETHSY 9.1b
  - Core Collapsed Liquid Level Comparison



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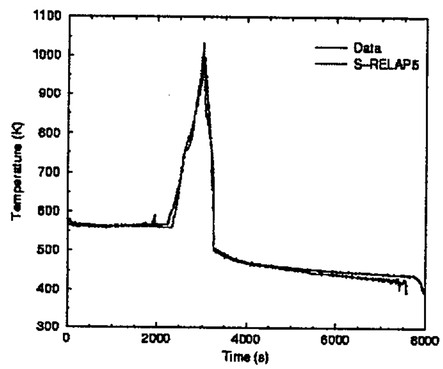
S-RELAP5 Code for Appendix K SBLOCA Analysis

- BETHSY 9.1b
  - Maximum Clad Temperature Comparison



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S-RELAP5 Code for Appendix K SBLOCA Analysis

- Summary
  - Proposed Siemens SBLOCA methodology:
    - Replaces ANF-RELAP and TOODEE2 with S-RELAP5 thereby streamlining the analysis process, and
    - Improves loop seal clearing behavior
  - Results from PWR sample problem and sensitivity study show:
    - Proposed SBLOCA methodology is convergent and robust
  - SBLOCA assessments show S-RELAP5 captures the phenomena important to SBLOCA (loop seal clearing, core boil-off and recovery) with an acceptable level of accuracy
  - Therefore, the proposed S-RELAP5 based SBLOCA methodology is suitable for licensing analysis

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Conclusion

- SER provides Siemens ability to reference topical report in future licensing submittals without further NRC review

- SER has no additional conditions

S-RELAP5
STAFF REVIEW

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
FEBRUARY 1, 2001

RALPH R. LANDRY
REACTOR SYSTEMS BRANCH
DIVISION OF SYSTEMS SAFETY AND ANALYSIS

S-RELAP5 REVIEW
STAFF EVALUATION

TOPICS COVERED

- Milestones
- Code modifications
- Assessment
- Regulatory requirements
- Conclusions

S-RELAP5 REVIEW
MILESTONES

- SIEMENS' REQUEST FOR S-RELAP5 SBLOCA REVIEW: JANUARY 10, 2000
- STAFF REQUESTS FOR ADDITIONAL INFORMATION: DECEMBER 10, 2000
- SAFETY EVALUATION REPORT (DRAFT): JANUARY 2001
- ACRS T/H SUBCOMMITTEE MEETINGS:
  - MARCH 2000 - ACCEPTANCE FOR REVIEW
  - AUGUST 2000 - REVIEW PLANS
  - JANUARY 2001 - DRAFT SER
- ACRS FULL COMMITTEE: FEBRUARY 2001
- FINAL SER: FEBRUARY 2001

S-RELAP5 REVIEW
RELAP5/MOD2 MODIFICATIONS

- Multi-Dimensional Capability - 2-D hydrodynamics
- Energy Equations - Conservation of energy
- Numerical Solution
- State of Steam-Noncondensable Mixture - Ideal gas at low steam quality
- Hydrodynamic Constitutive Models - Interphase friction and mass transfer
- Heat Transfer Model - Consistent use of correlations, replaced Dittus-Boelter
- Choked Flow - Moody critical flow model
- Counter-Current Flow Limit - Conforms with RELAP5/MOD3
- Component Models - EPRI pump performance, ICECON incorporated
- Fuel Model - RODEX2 and TOODEE2 incorporated, Baker-Just metal-water
- Code Architecture - Conforms with RELAP5/MOD3, FORTRAN 77

S-RELAP5 REVIEW
STAFF EVALUATION

CODE ASSESSMENT

- SBLOCA assessment cases defined by NUREG-0737, Section II.K.3.30: Should use data from LOFT and Semiscale facilities
- NUREG-0737 suggests Semiscale Test S-07-10B and LOFT Test L3-1.
- S-RELAP5 assessment done using Semiscale Test S-UT-8, LOFT Test LP-SB-3 instead, plus 2-D Flow Tests, UPTF Tests, and BETHSY Test 9.1b
- Significant parameters for respective tests well predicted
- Substitution of newer tests for those suggested in the NUREG are acceptable
- Caveat: Applications will be reviewed on a case-by-case basis to determine if licensees have in their licensing conditions assessment versus specific tests given in NUREG-0737

S-RELAP5 REVIEW
STAFF EVALUATION

REGULATORY REQUIREMENTS

- Modeling requirements of 10 CFR Part 50, Appendix K, such as Moody critical flow, have been incorporated
- Assessment conforming to the intent of NUREG-0737, Section II.K.3.30 has been performed
- Further assessment beyond that required, including an informal Phenomena Identification and Ranking Table, has been performed
- Sensitivity studies investigating the break spectrum, effect of time step size, loop seal model, pump model, radial flow form loss coefficients, and nodalization have shown the impact of each on the PCT to be less than 5°F
- The solution has been shown to be converged

S-RELAP5 REVIEW
STAFF EVALUATION

CONCLUSIONS

- ANF-RELAP code (approved by the staff) has been modified to incorporate RODEX2, TOODEE2, and ICECON in a single, integrated code
- The code documentation supports the modifications made to the ANF-RELAP code, and the staff accepts those modifications, such as the numeric solution method, heat transfer correlations, assessment cases, and so on
- The staff has noted that errors occur in the documentation and that they will be corrected in the publication of the approved version of the code manuals
- The Siemens Power Corporation has been very responsive to the concerns expressed by the staff and has been very cooperative in the conduct of the S-RELAP5 review
- The staff finds the S-RELAP5 code acceptable for use in satisfying the requirements for analysis of the Small-Break LOCA event under the requirements of 10 CFR Part 50, Appendix K

United States
Nuclear Regulatory Commission

Treatment of Uncertainties in Pressurized Thermal Shock

M. Mayfield, N. Siu

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

Presentation to
Advisory Committee on Reactor Safeguards
February 1, 2001

Introduction

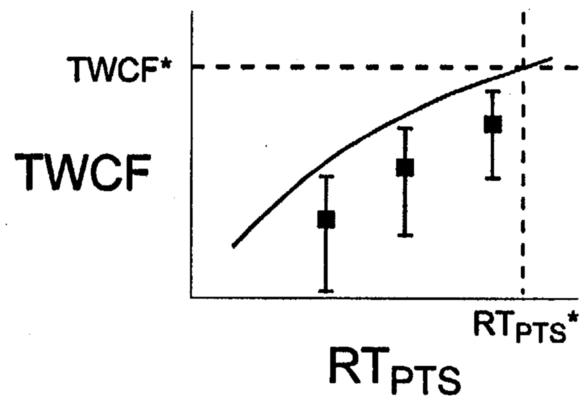
- **Project objective to develop technical basis for potential revision to PTS Rule 10 CFR 50.61 for PWR plants**
- **First major application of risk informed methodology to an adequate protection rule**
- **Evaluating 4 plants in an effort to develop generic approach**
  - **4 Plants: Oconee-1, Calvert Cliffs-1, Palisades, and Beaver Valley-1**
  - **No intention to do plant specific evaluation for PWR fleet**
  - **Use best available tools for analysis**
- **This is one of a continuing series of briefings to**
  - **Provide in-progress summaries in major areas**
  - **Solicit committee feedback**
  - **No letter is requested at this time.**
- **Key issue to be discussed today -- *treatment of uncertainties* in major areas**
- **Some of the comments from the January 18, 2001 Joint Subcommittee briefing are addressed today**

Outline

- Objectives and conceptual approach
- Analysis overview
- Status: PRA, T/H, and PFM
- Draft PRA results
- T/H approach
- PFM approach
- Key issues and summary

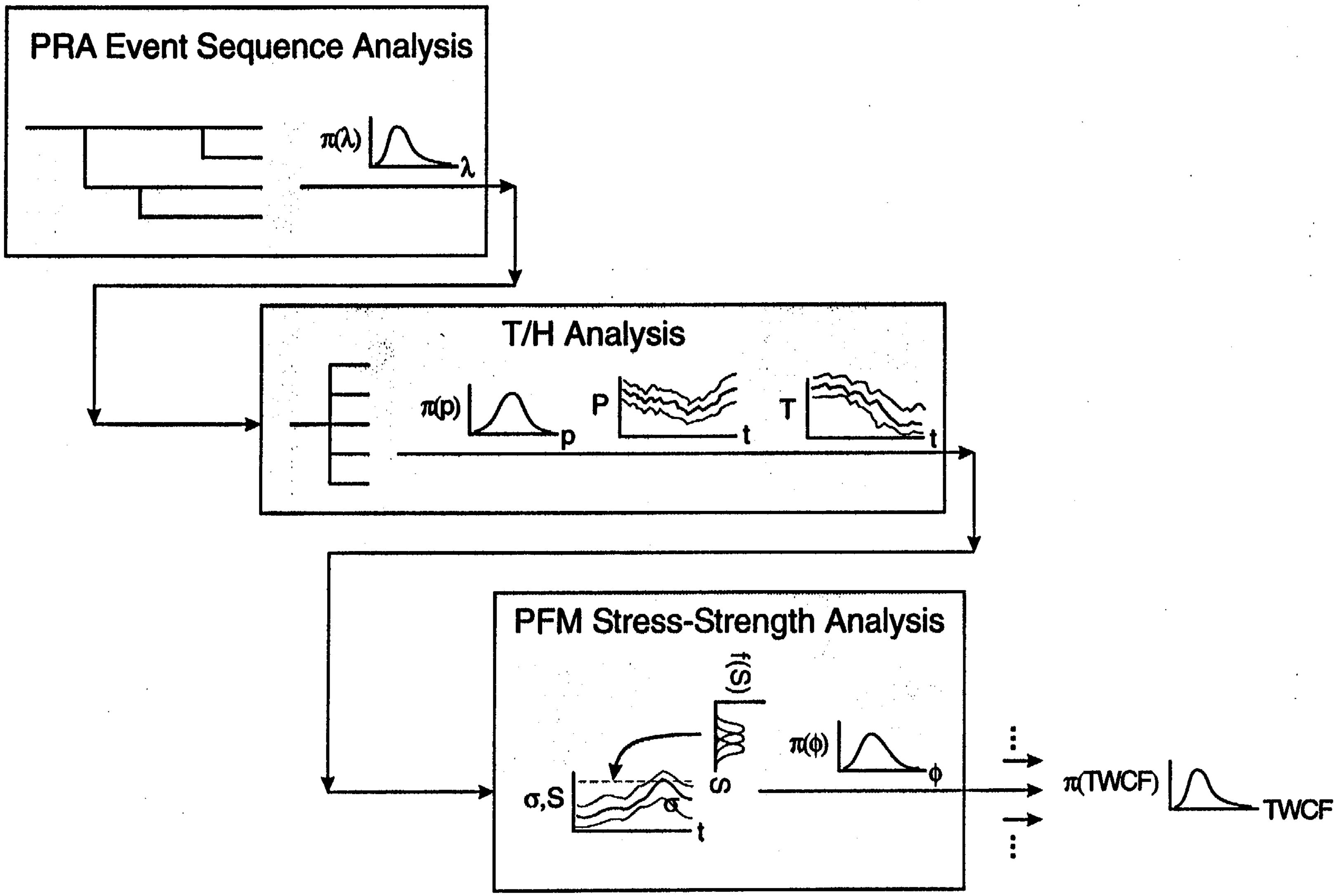
Objective and General Approach

- Assess uncertainties in estimates of PTS risk
- Process supports
  - development of screening criterion, e.g.,



  - development of technical basis for revising RG 1.154
- Analysis involves
  - categorization of sources of uncertainty
  - construction of aleatory model
  - propagation of epistemic uncertainties through aleatory model

A Conceptual Model



Simplifications

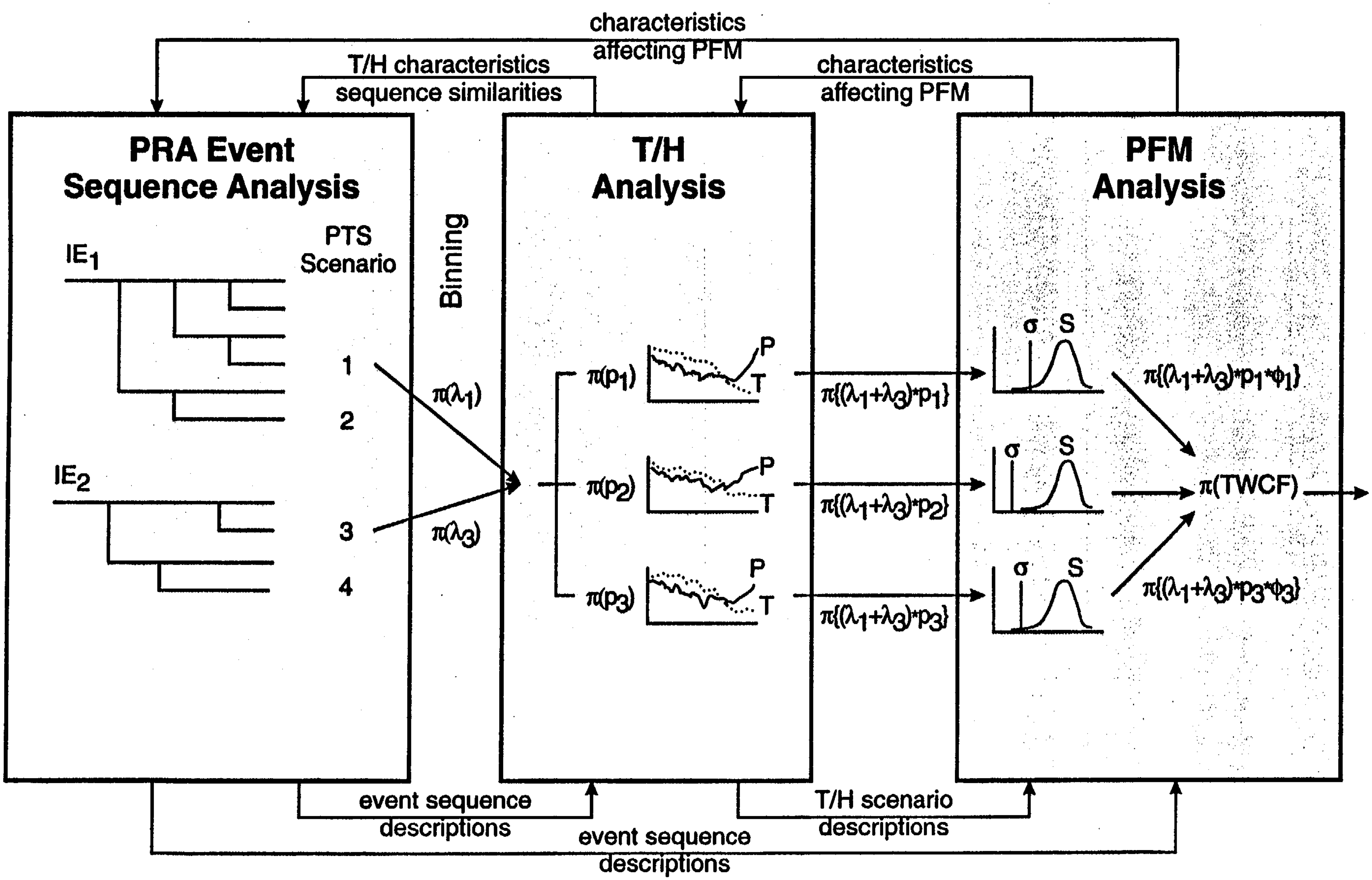
- **Resource constraints:**
  - RELAP5 run times
  - Pre- and postprocessing requirements

=> Bin similar sequences

- **Model uncertainties**
  - Formal methods under development
  - Limited data

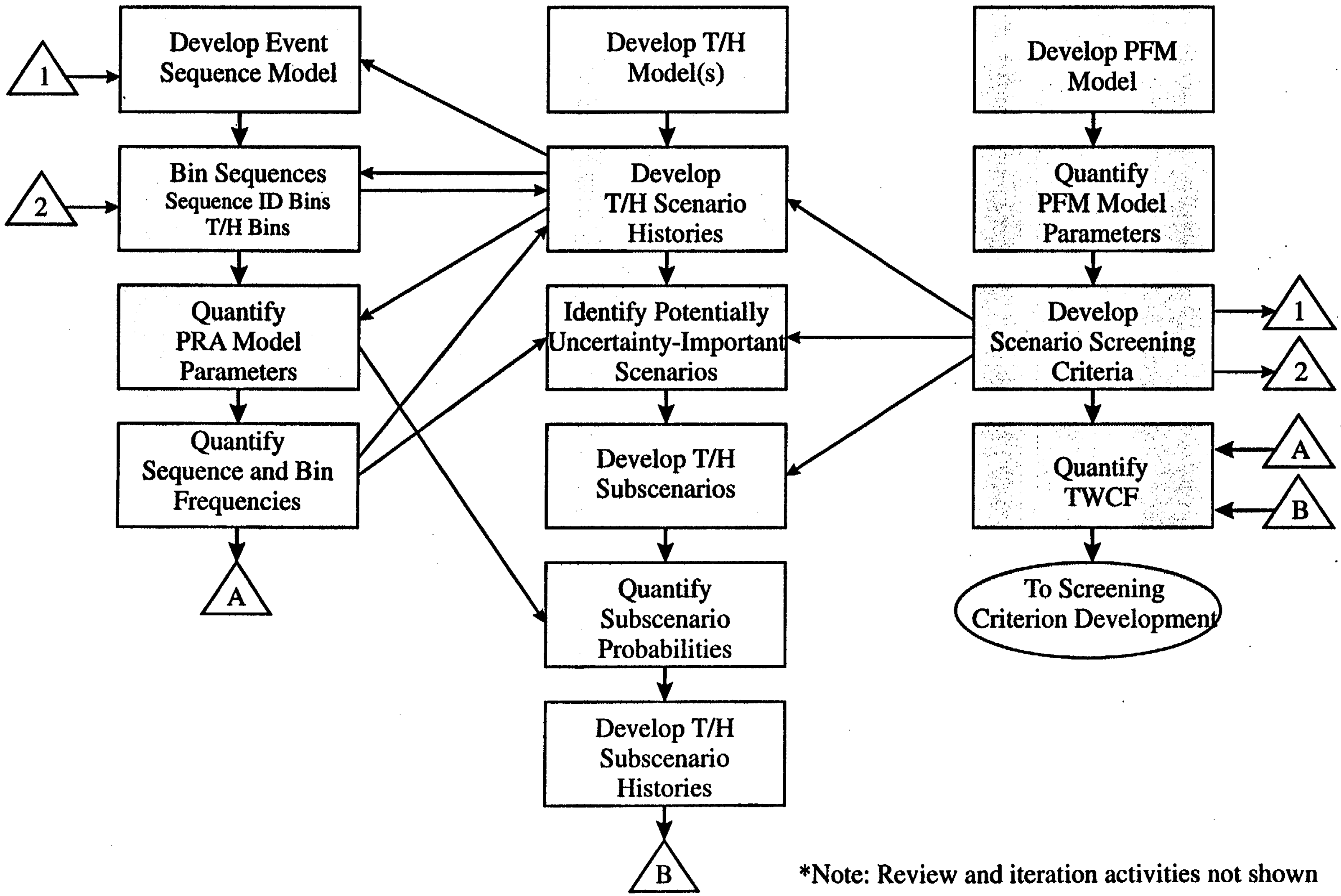
=> Quantify parameter, boundary condition, and submodel uncertainties
=> Supplement with information from experiments and selected sensitivity studies

Framework



characteristics
affecting PFM
T/H characteristics
sequence similarities
characteristics
affecting PFM
PRA Event
Sequence Analysis
IE1
PTS
Scenario
1
2
IE2
3
4
Binning
π(λ1)
π(λ3)
T/H
Analysis
π(p1)
π(p2)
π(p3)
P
T
π{(λ1+λ3)*p1}
π{(λ1+λ3)*p2}
π{(λ1+λ3)*p3}
PFM
Analysis
σ
S
π{(λ1+λ3)*p1*φ1}
π(TWCF)
π{(λ1+λ3)*p3*φ3}
event sequence
descriptions
T/H scenario
descriptions
event sequence
descriptions

Overall Analysis Process and Key Interactions*



*Note: Review and iteration activities not shown

Current Status: Treatment of Uncertainties

- Aleatory model developed
- Model parameters categorized
- PRA event sequence analysis
  - distributions for scenario frequencies developed for Oconee scoping study
  - distributions to be revised as part of iteration process
- T/H analysis
  - have identified classes of scenarios where boundary condition uncertainties dominate model structure uncertainties
  - potentially important parameters identified
  - process for quantifying subscenario probabilities (split fractions) proposed
  - process to be demonstrated as part of Oconee analysis

Current Status (cont.)

- **PFM analysis**
  - **distributions for most model parameters (e.g., flaw number and characteristics, fluence, chemistry) quantified**
  - **approach for treating uncertainties in fracture toughness and RT_{NDT} being developed**
  - **FAVOR undergoing modification**

Draft PRA Results Overview - Oconee 1

Description	T/H Run	Preliminary Estimates* Challenge Frequency (/yr) 5th	Mean	95th	P{TWC|S}
SBLOCA (2"), HPI on full	3	1E-5	3E-5	7E-5	TBD
SBLOCA (2.8"), HPI on full	4	4E-8	3E-7	1E-6	TBD
Large MSLB, EFW feeding faulted SG, HPI on full	25	4E-5	9E-5	2E-4	TBD
Large MSLB, EFW feeding faulted SG, HPI throttled	27	9E-5	1E-4	1E-4	TBD
SBLOCA (1"), 2 MS-SRV stuck open, HPI on full, HZP	31	2E-11	9E-6	3E-5	TBD
Reactor trip, SRV stuck open, HPI on	34	2E-5	8E-5	2E-4	TBD

*Estimates will be revised following a review of the initial integrated analysis results

Example - Draft PRA Results Decomposition

- Large main steam line break
  - 9 top events modeling isolation, feedwater response, high pressure injection, reactor coolant pumps
  - multiple potentially challenging end states possible
  - a contributing sequence (frequency ~ 5 x 10^{-6}/yr)

    Large steam line break

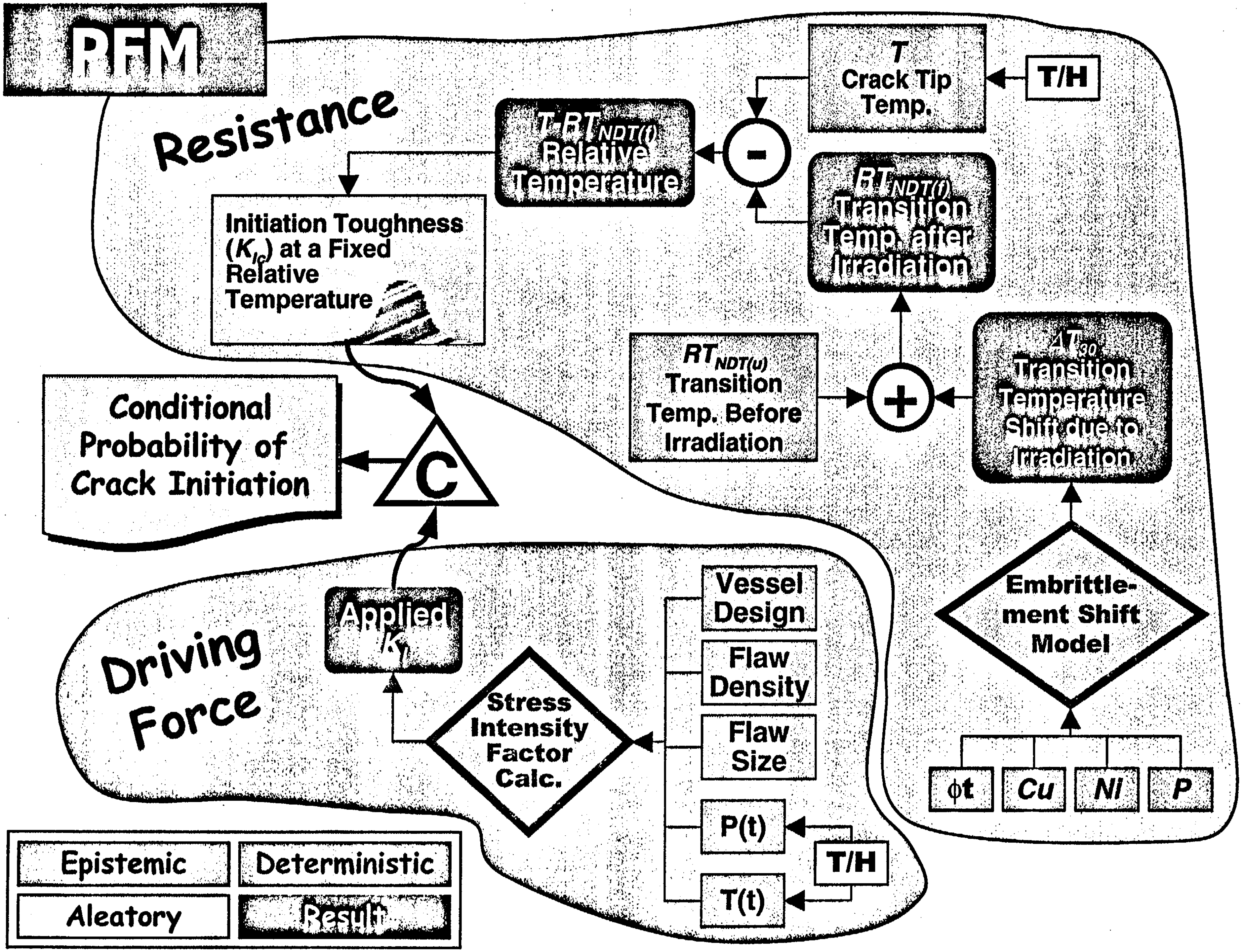
    Operators fail to isolate break

    Operators fail to isolate flow to faulted S/G

    Operators fail to throttle HPI flow
- Binned into T/H Run 25
- Issues to be addressed
  - binning of sequences
  - time frame for operator actions
  - dependencies
  - uncertainties

Uncertainties in T/H Analysis

- Identify key sources of uncertainty
  - boundary conditions (e.g., timing of events, size of breaks)
  - models
- Classify scenarios regarding relative importance of uncertainty sources
  - single-phase
  - two-phase
- For single-phase scenarios which appear to be important contributors to PTS risk
  - use representative boundary condition variations to define subscenarios
  - quantify distributions for subscenario probabilities (split fractions)
  - identify appropriate T/H run or perform additional run
- For two-phase scenarios, follow a similar approach and also investigate potentially dominant sources of model uncertainty


RFM
Resistance
T Crack Tip Temp.
T/H
T-RT_NDT(f) Relative Temperature
RT_NDT(f) Transition Temp. after Irradiation
Initiation Toughness (K_Ic) at a Fixed Relative Temperature
RT_NDT(u) Transition Temp. Before Irradiation
ΔT_30 Transition Temperature Shift due to Irradiation
Conditional Probability of Crack Initiation
C
Embrittlement Shift Model
Driving Force
Applied K_I
Vessel Design
Flaw Density
Flaw Size
Stress Intensity Factor Calc.
P(t)
T(t)
T/H
ϕt
Cu
Ni
P
Epistemic
Deterministic
Aleatory
Result

Key Issues

- PRA event sequence analysis
  - uncertainties in success criteria (especially for human actions)
  - other uncertainties in human failure event probabilities
- T/H analysis
  - model uncertainties (2ϕ scenarios)
  - parameter distributions
- PFM analysis
  - uncertainties in fracture toughness and irradiation shift
  - uncertainties in crack arrest
- Integrated analysis: uncertainties in binning process (especially "sequence identifier" -> T/H runs)

Summary

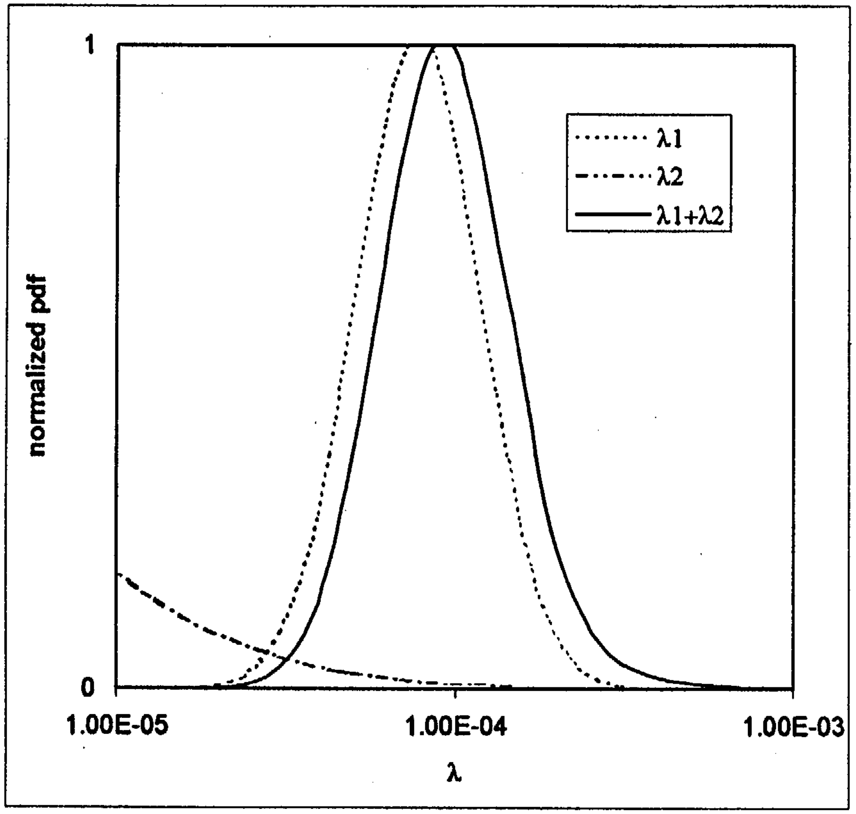
- Analysis consistently treats uncertainties across different disciplines
- Approach quantifies most potentially important sources of uncertainty
  - model parameters, T/H boundary conditions and submodels
  - model structure uncertainties treated qualitatively
  - may need to refine models, depending on results of experiments and sensitivity analyses
- Approach will be documented in white paper update
- Work is in progress
  - currently iterating on initial results
  - scoping results expected later in February; full results in late Spring
- Approach may be useful in other risk-informed applications

Backup Slides

Treatment of Thermal Hydraulic Uncertainties in Pressurized Thermal Shock

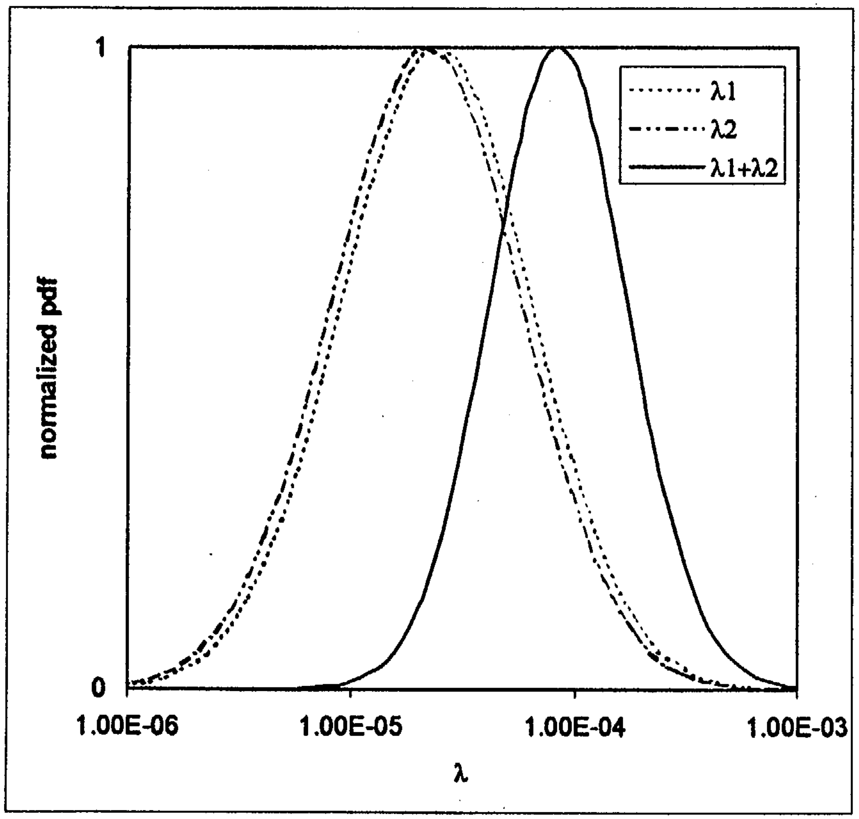
February 1, 2001

Dominant Effect of Dominant Sequences



	Mean	RF	5th	95th
λ1	1.0E-4	2	4.6E-5	1.8E-4
λ2	5.0E-5	20	4.8E-7	1.9E-4
λ1+λ2	1.5E-4	2.3	5.5E-5	3.0E-4

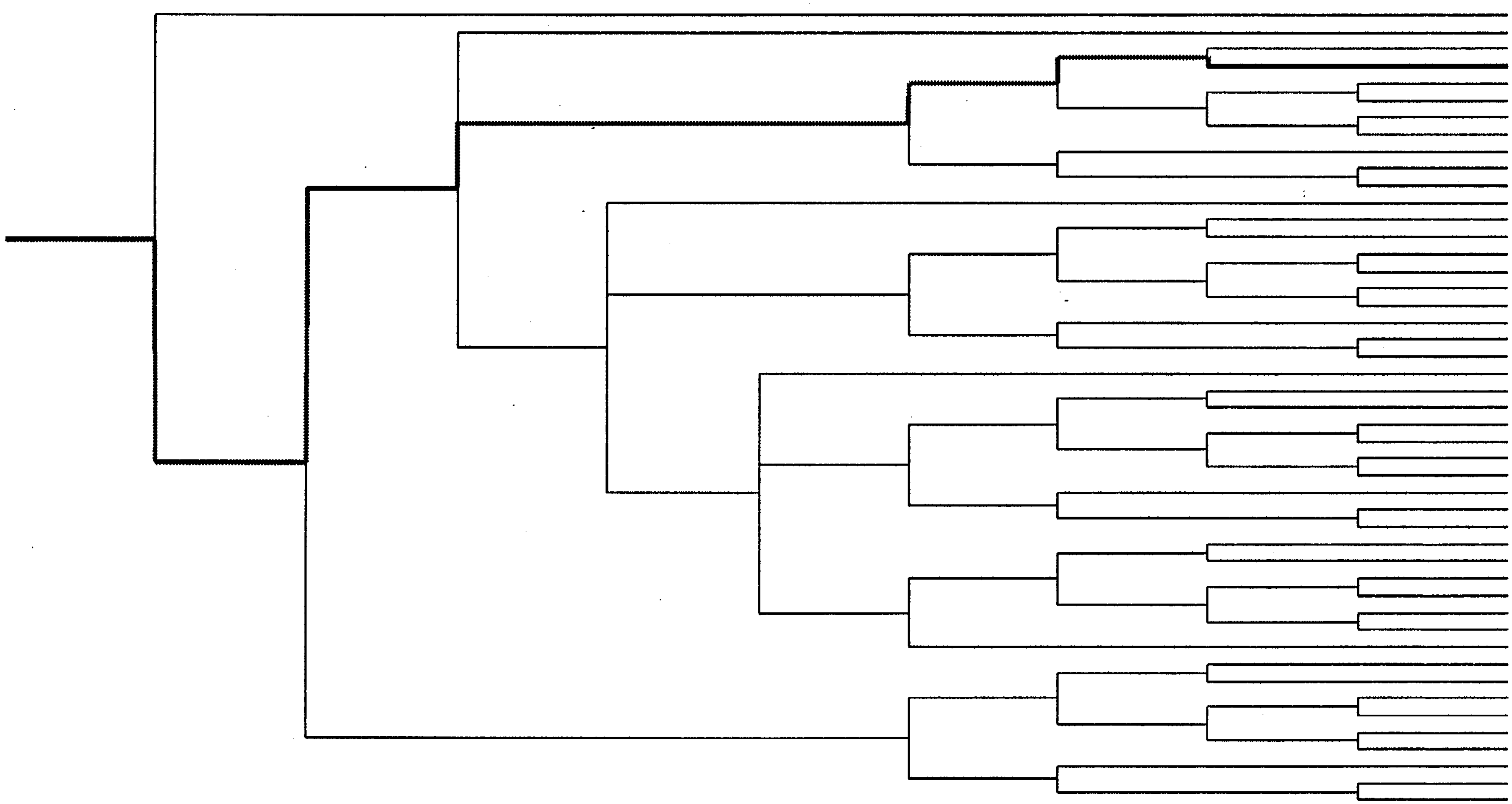
Narrowing Effect of Adding Sequences



	Mean	RF	5^{th}	95^{th}
λ1	1.0E-4	5	1.2E-5	3.1E-4
λ2	9.0E-5	5	1.1E-7	2.8E-4
λ1+λ2	1.9E-4	3	4.4E-5	4.5E-4

MSLB Event Tree

Large Steam Line Break	Fail to Isolate SLB2	MFW Response to IE	EFW Response to MFW Trip	Fail to Recover from EFW-FTS	Condensate Booster Pumps Fail	HPI/F&B Response	RCP Trip (Loss of RCS Subcooling)	Fail to Throttle HPI Flow	Fail to Restart RCPs

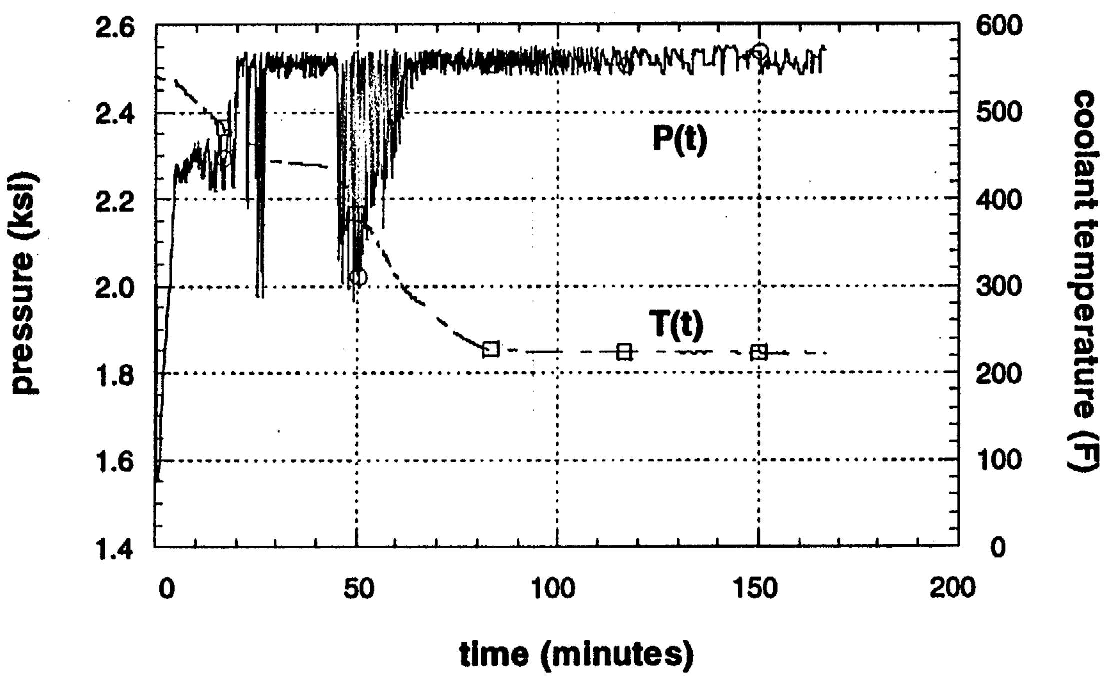


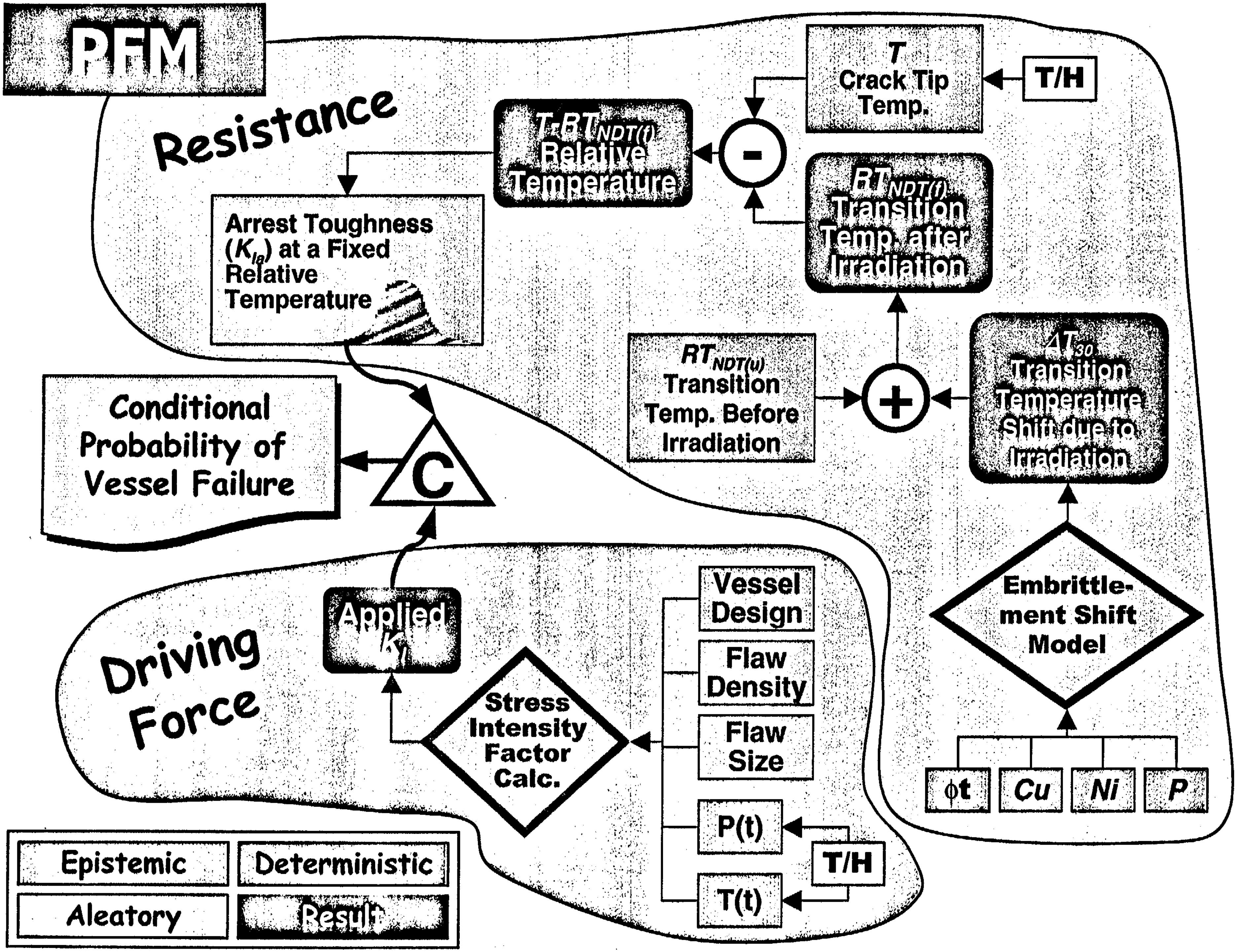
Description: T/H Run 25

- Main Steam Line Break (severed main steam line at S/G-A outlet).
- Steam Line Break (SLB) Isolation circuit trips main feedwater (MFW) pumps, but turbine-driven emergency feedwater (EFW) pump is not tripped by the SLB isolation circuit. (At Oconee, the motor driven EFW pumps are not tripped by the SLB isolation circuit.)
- EFW system provides flow to faulted S/G (i.e., S/G-A) only. Flow controlled to maintain level in S/G. All heat removal is through faulted S/G, therefore no flow to intact S/G.
- High pressure injection (HPI) actuates at 21 seconds into transient.
- Control room operators do not throttle HPI flow.

All other primary and secondary functions are assumed to be normal.

Thermal hydraulics input for Oconee transient case 25




PFM
Resistance
T
Crack Tip
Temp.
T/H
T-RT_NDT(f)
Relative
Temperature
-
RT_NDT(f)
Transition
Temp. after
Irradiation
Arrest Toughness
(K_Ia) at a Fixed
Relative
Temperature
RT_NDT(u)
Transition
Temp. Before
Irradiation
+
ΔT_30
Transition
Temperature
Shift due to
Irradiation
Conditional
Probability of
Vessel Failure
C
Embrittle-
ment Shift
Model
Vessel
Design
Applied
K_I
Flaw
Density
Driving
Force
Stress
Intensity
Factor
Calc.
Flaw
Size
φt
Cu
Ni
P
P(t)
T/H
T(t)
Epistemic
Deterministic
Aleatory
Result

Categorization of Key Parameters

Variable/Parameter	Uncertainty Category
initiating event frequencies	epistemic
component unavailabilities	epistemic
human failure event probabilities	epistemic
event occurrence times	aleatory*
flow areas	aleatory*
flow rates	aleatory*
fluid temperatures (feedwater, HPI)	aleatory*
decay heat	aleatory*
chemistry (Cu, Ni, P)	epistemic
neutron fluence	epistemic
flaw characteristics (density, size, location)	epistemic
K_{Ic} scatter	aleatory*

*all aleatory models may have significant epistemic uncertainties

CLOSEOUT OF GENERIC SAFETY ISSUE 152
DESIGN BASIS FOR VALVES THAT MIGHT BE SUBJECTED TO SIGNIFICANT BLOWDOWN LOADS

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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SEEK ACRS ENDORSEMENT OF THE STAFF'S PROPOSAL TO CLOSE THE ISSUE

ORIGIN OF GENERIC SAFETY ISSUE (GSI) 152

GSI-152 was raised by the ACRS in 1989 during the review of activities related to GSI-87, "Failure of HPCI Steam Line Without Isolation"

GSI-87 dealt with the ability of the HPCI steam line isolation valves to isolate a postulated pipe break

GSI-87 was subsequently closed with Generic Letter (GL) 89-10, and its supplements

GL 89-10 focused on the ability of motor operated valves (MOVs) to operate consistent with their approved design basis

The focus of GSI-152 was the adequacy of the design basis for valves that might be subjected to significant blowdown loads (i.e., pipe breaks)

"the requirement for safety-related valves to move against high differential pressures and/or high flows experienced during a large downstream pipe break may not have been specified in the design bases"

GL 89-10

Testing done by the Office of Nuclear Regulatory Research for the closure of GSI-87 showed weaknesses in valve performance attributable both to motors and to valve mechanisms

GL 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance”, did not address the adequacy of the design bases, only adequacy of valve performance in meeting the approved design bases

> An examination of the design bases was included as part of GL 89-10 but the evaluation of the adequacy of the design bases was a separate issue, (i.e. GSI-152)

Industry sponsored research which confirmed the weaknesses identified in NRC testing

> Licensees, reactor manufacturers, and industry groups developed “working groups” to address these weaknesses

GL 89-10 had 7 supplements spanning from 1989 to 1996

ADEQUACY OF DESIGN BASIS

Although GL 89-10 was focused on the ability of MOVs to operate as designed, the issue of adequacy of the design bases contained in GSI-152 was captured by industry initiatives, and confirmed during NRC inspections which included the reasonableness of the design bases

NRC inspections conducted to examine GL 89-10 programs also addressed the adequacy of design bases and whether they included the differential pressure associated with a large downstream break–the subject of GSI-152

> A priority focus was on the high risk significant valves of HPCI, RCIC, and RWCU MOVs (GL 89-10 Supplement 3)

Lessons learned regarding valves other than MOVs were reported to the licensees by NRC and were voluntarily incorporated by the industry groups into their programs addressing all valve types

CLOSURE OF GSI-152

The ACRS Subcommittee on Mechanical Components was briefed on industry MOV activities and other MOV related items on October 5, 1993

The Subcommittee Chairman who had originally raised the concern stated he was satisfied the issue was adequately addressed and could be closed

RES confirmed the actions taken by licensees by reviewing historical documents including licensee documents submitted in response to GL 89-10, GL 89-10 inspection guidance, and GL 89-10 inspection reports

Based on the actions taken by industry in response to GL 89-10 and confirmed in NRC inspections, there is sufficient evidence to close GSI-152, and no further action is necessary