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by W. Kerr on 10/10/91

ACRS-2766

ACRS SUBCOMMITTEE MEETING
SUMMARY/MINUTES OF THE INSTRUMENTATION
AND CONTROL SYSTEMS
AUGUST 29, 1991
BETHESDA, MARYLAND

INTRODUCTION

The ACRS Subcommittee on Instrumentation and Control Systems held a meeting on August 29, 1991 in Bethesda, Maryland.

The entire subcommittee meeting was open to public attendance. The documents submitted to the subcommittee Members for this meeting are attached.

ATTENDEES

ACRS

W. Kerr, Chairman
H. Lewis, Member
C. Wylie, Member
M. El-Zeftawy, Staff

OTHERS

K. Barrett, Bechtel
C. Tuley, W
J. Trotter, EPRI
K. Klaube, NUS
P. Ludgate, NUS
C. Swanson, B&W
G. Srikantiah, EPRI
R. Krich, PE Co.
G. Salamon, FP&L
S. Palena, PE Co.
J. Armstrong, PE Co.
L. Hopkins, PE Co.
G. Stewart, PE Co.

NRC

W. Butler, NRR
R. Clark, NRR
T. Kenney, SRI
R. Van Houton, SECY
W. Swenson, NRR
G. Galletti, NRR
E. Imbro, NRR
J. Jacobson, NRR
D. Norkin, NRR
A. Gautam, NRR
J. Gallagher, NRR
C. Doutt, NRR

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Meeting Highlights, Actions, Agreements

1. Dr. Kerr, Subcommittee Chairman, convened the meeting and stated that the purpose of the meeting is to discuss with the EPRI representatives, representatives of Philadelphia Electric (PE) Company, and the NRC staff the following items, respectively.
   - A set-point methodology developed by EPRI
   - Transient Response Implementing Plan (TRIP) procedures developed and implemented by PE Co.
   - Electrical Distribution System Functional Inspections (EDSFIs) currently being performed by the staff.

2. Mr. Srikantiah, EPRI, briefed the subcommittee Members regarding an EPRI procedure for determining the set-points for safety systems. He described a generic process for establishing appropriate and justifiable limits for the reactor monitoring and protection systems set-points. This process was designed to be applicable to utilities with either PWRs or BWRs who plan to assume primary responsibility for performing safety analysis and set-points determination.

   Mr. Srikantiah stated that utilities are familiar with the general aspects of set-point methodology, as well as the results of its use in reactor operation. However, the methods and assumptions used by vendors to determine set points are not readily available for utility use. A plant's set points depend on the operational goals of the licensee and are derived from a consideration of safety analyses and performance considerations. The performance evaluation takes account of the constraints on plant operation, given the limitations imposed by the safety analysis. In turn, the safety analysis verifies the acceptability of the assumed set points. The safety analysis and the determination of set points are interrelated processes.

An important step in the determination of acceptable set-points is the treatment of the uncertainties. These include uncertainties inherent in the actual equipment settings and those in the inputs to the license basis analysis and plant performance evaluations. The estimated values have an associated uncertainty distribution.

EPRI's set-point process is related to the process used in the final safety analysis report (FSAR) design analyses. The FSAR focuses on plant response to design basis events (DBEs) while EPRI's set-point process focuses on monitoring and protection system set-points used in the analyses of the limiting cases in order to assure that acceptable consequences result.

Verification of the monitoring and protection system limits is accomplished through an analytical demonstration of acceptable consequences of DBEs. The analysis plays a central role in the determination of set-points. The results of the event analyses are compared to the event acceptance limits. If the results are acceptable then the set-points are considered acceptable. If the results are not acceptable, then a margin recovery process is undertaken which may include changing set-points, changing the operating envelope, changing model inputs, changing analysis assumptions and/or methods, or changing the plant design. An alternate approach developed by EPRI is based on what is described as a more realistic treatment of uncertainties. The approach involves a statistical combination of uncertainties (SCU). EPRI has developed guidance for determining the appropriate event analysis model inputs from the plant design and determining the operating envelope sources.

The following are EPRI's guidelines for LWR set-point determination;

A. Complete the plant performance evaluation.

B. Identify the plant specific DBEs which must be analyzed and the associated acceptance limits.

C. Identify the plant design model inputs and operating envelope limits to be used in the event analyses.

a) Plant design model inputs

- From the plant design determine the numerical ranges for the model inputs.
- Include in the ranges the appropriate uncertainties.

b) Operating envelope limits

- From the monitoring system determine the numerical ranges for the operating envelope limits.
- Include in the ranges the appropriate uncertainties.
- To these ranges apply any required maneuvering to determine the analysis input ranges over which the event analyses must be valid.
- Select axial power shapes.

D. Model the protection system in the event analyses.

- Identify trips to be used.
- Determine the protection system instrument set-points.
- Apply appropriate uncertainty allowances to the instrument set-points.
- Account for event effects through appropriate allowances or explicit modeling to determine the analytical set-point for the event analyses.

E. Perform the event analyses insuring that limiting cases have been analyzed for each event and all acceptance

limits have been satisfied.

F. Implement margin recovery process if required.

- Change the protection system instrument set-points.
- Change the operating envelope.
- Modify the event analyses models.
- Change the plant design.
- Apply statistical treatment of uncertainties.

Mr. Srikantiah stated that EPRI's statistical combination of uncertainties methodology can be used to reduce excessive conservatism inherent in the existing deterministic analysis process. Use of EPRI's method is expected to decrease the number of scrams and to eliminate some actuations of safety and relief valves, thus increasing operating flexibility. Mr. Srikantiah cited some applications of the EPRI's set-point methodology, such as Oyster Creek high-pressure set-points, Vermont Yankee reload analysis, Diablo Canyon safety valve tolerance, TMI-1 flux/flow set-points, and Palo Verde safety valve tolerance.

Dr. Kerr commented that EPRI's approach appears to be plausible but complicated. Further several steps employ "engineering judgement" making the results unpredictable. EPRI representatives agreed that given the same data, two independent reviewers could arrive at two different conclusions.

Dr. Kerr stated that there is no evidence of any systematic consideration of whether use of EPRI method would have any effect on risk. He commented that it is possible if the method decreases the number of unneeded scrams, and at the same time does not increase the likelihood of exceeding a safety limit, risk may be decreased by its use. He recommended that prior to the NRC approval of this method for

widespread application, a systematic investigation be made of its effect on risk. Dr. Lewis and Mr. Wylie agreed.

3. Mr. L. Hopkins, Limerick Units 1 & 2, presented an overview of the TRIP program. He stated that following Three Mile Island (TMI) accident, the NRC required licensees of operating reactors to reanalyze transients and accidents and to upgrade emergency operating procedures (EOPs). The NRC staff presented guidelines to the licensees to upgrade EOPs and described the use of a "Procedure Generation Package" (PGP) to prepare EOPs. The PGP would include plant-specific technical guidelines, a writers guide, a description of the program to be used for the validation of EOPs, and a description of the training program for the upgraded EOPs.

From this PGP, plant specific EOPs were to have been developed that would provide the operator with the directions to mitigate the consequences of a broad range of accidents and multiple equipment failures. Some licensees such as Limerick Generating Station (LGS) Units 1 and 2 and Peach Bottom have developed and implemented their own EOPs, using guidance formulated by the BWR Owners Group (BWROG), such as the Transient Response Implementing Plan (TRIP). TRIP equals EOPs. LGS is a BWR with a Mark III containment structure.

Mr. J. Armstrong, Philadelphia Electric Company, indicated that the NRC is currently performing inspections on different plants to evaluate the EOPs at the licensee facilities. The NRC staff considers the objectives to be met if review of the following areas were found to be adequate: comparison of the TRIP procedures with the LGS plant specific technical guidelines (PSTG) and the BWR owners group emergency procedure guidelines (EPG); review of the technical adequacy of the deviations from the EPG; control room and plant walkdowns of

the TRIP procedures; real time evaluation of the TRIP procedures on the plant simulator; evaluation of the licensee program on continuing improvements of the TRIP procedures; and performance of human factors analysis of the TRIP procedures.

Dr. Kerr discussed with PE representatives some of the comments made in the staff's report transmitted by a letter of July 3, 1990 to Mr. D. M. Smith (PE Co.). PE representatives consider the comments made by the NRC staff during the inspection as constructive and helpful.

Dr. Kerr commented that examination of some of the comments and PE's responses might lead to the conclusions that PE personnel consider staff suggestions tantamount to an order. Dr. Kerr cited an example in which the staff commented that, "the glare from the Plexiglass covering the flowcharts in the control room is excessive." The PE personnel indicated that they would have not been aware of this problem until the staff called it to their attention and they corrected it immediately.

4. Mr. E. Imbro, NRR, briefed the subcommittee Members regarding the Electrical Distribution System Functional Inspections (EDSFIs). He stated that during multidisciplinary inspections, the NRC has identified many deficiencies related to the electrical distribution system. To address these deficiencies, the NRC has developed an inspection to specifically evaluate the electrical distribution system. The NRC completed 24 EDSFIs at the five NRC geographical regions. During these inspections, the staff found several common deficiencies in the licensees' programs and in the electrical distribution systems as designed and configured at each plant. These deficiencies included inadequate ac voltages at the 480 Vac and 120 Vac distribution levels, inadequate procedures to

test circuit breakers, and inadequate determinations and evaluations of setpoints.

Mr. J. Jacobson, Mr. A. Gautam, and Mr. D. Norkin, NRR, stated that for several nuclear power plants, the staff found that, under certain conditions, the voltage available at the safety buses would be inadequate to operate safety-related loads and associated equipment. These conditions could occur when the plant's electrical distribution systems were being supplied from an offsite grid that had become degraded but that continued to supply voltages that remained above the setpoints at which the degraded grid relays would be activated.

The function of the degraded grid relays is to ensure that adequate voltage is available to operate all Class 1E loads at all distribution levels. In order to ensure that all required Class 1E loads will remain operable during degraded offsite voltage conditions, some licensees are currently reanalyzing the basis for the degraded grid relay setpoints.

The staff identified repetitive deficiencies in licensees' programs to test circuit breakers. These deficiencies included inadequate procedures, inadequate test acceptance criteria, inadequate test equipment, and inadequate control of testing. At the Susquehanna plant, the staff found that the licensee was testing dc molded case circuit breakers with a procedure written for testing ac breakers. The licensee had not established specific acceptance criteria for the dc breakers.

Other findings identified during recent EDSFIs were related to inadequate setpoint determinations. Some licensees have operated equipment outside of acceptable limits because they did not determine proper setpoints and did not evaluate and

account for instrument drift. The staff has identified these circumstances primarily for instruments for which the setpoints were determined by the architect/engineer or the nuclear safety system supplier. Those setpoints not contained in the plant technical specifications were also more frequently found to be deficient. During recent EDSFIs, the staff identified deficiencies in setpoints for diesel day tank level indicators, diesel air start compressor controllers and alarms, invertor low voltage shutdown circuitry, degraded grid relays, and diesel overcurrent relays.

The staff has published Information Notice No. 91-29 on April 15, 1991. The staff believes that its inspections have resulted in an increased sensitivity to the issue throughout the industry. It is planned to continue performing EDSFIs until all plants have been inspected by January 1993.

In response to Dr. Kerr's question regarding the risk reduction that is likely to be achieved by this program, the NRC staff had no answers and indicated that no effort has been made to calculate or estimate the possible risk reduction.

Dr. Kerr stated that he would have liked to see evidence from some source (may be NUREG-1150), that electrical systems in operating plants represent a significant risk contributor, and the result of the EDSFI program will produce some risk reduction. The subcommittee Members commented that it should be a Commission Policy to make decisions on starting new programs based (at least partially) on the need and expectation of risk reduction.

Dr. Kerr asked whether, now that a significant number of plants have been examined and the results of those examinations are available, it would be more effective to make these results available and to ask the remaining plants to conduct their own examinations as part of their IPEs. The

staff appeared to believe that NRC management would not approve of this approach, and the resulting examinations would not be as thorough nor as effective as they would be if performed by the staff.

In conclusion, the subcommittee Members agreed that the plant electrical system is important, and that its reliability and performance are important to both the prevention and the mitigation of accidents.

FUTURE ACTION

The Subcommittee Chairman is planning to brief the full Committee at the September 5-7, 1991 ACRS meeting regarding the above three topics. Pending the outcome of such briefing, the Committee may decide on a course of action.

Attachments:

1. EPRI Setpoints Projects/Methodology and Applications - by G. Srikantiah
2. Philadelphia Electric Co. (slides)
3. EDSFI briefing to ACRS

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NOTE: Additional meeting details can be obtained from a transcript of this meeting available in the NRC Public Document Room, 2120 L Street, NW, Washington, DC 20006, (202) 634-3273, or can be purchased from Ann Riley and Associates, Ltd., 1612 K Street, NW, Suite 300, Washington, DC 20006, (202) 293-3950.

EPRI Setpoints Projects

Methodology and Applications

Presented by

G. Srikantiah

August 1991

EPRI Setpoint Projects Methodology and Applications

OVERVIEW

- *Background*
- *Setpoint Optimization Objectives*
- *Existing Methodology*
- *Safety Analysis Process*
- *Current Treatment of Measurement Uncertainties*
- *Development of Safety Analysis Inputs*
- *Potential Problems*

EPRI SETPOINT METHODOLOGY

EPRI SETPOINT PROJECTS

EPRI Setpoint Projects

Background

- Setpoints are Instrument Settings at which System Automatic or Operator Action Must Occur to Preserve the Assumption in the Plant Safety Analysis
- Setpoints Must Satisfy Plant Safety and Licensing Requirements
- Plant Capacity and Operating Flexibility Should be Optimized

EPRI Setpoint Projects

Setpoint Optimization Objectives

- To Improve Plant Performance through Optimized Instrument Settings while Satisfying Safety and Licensing Requirements
- To Gain Operational Margin
- To Improve Operational Flexibility

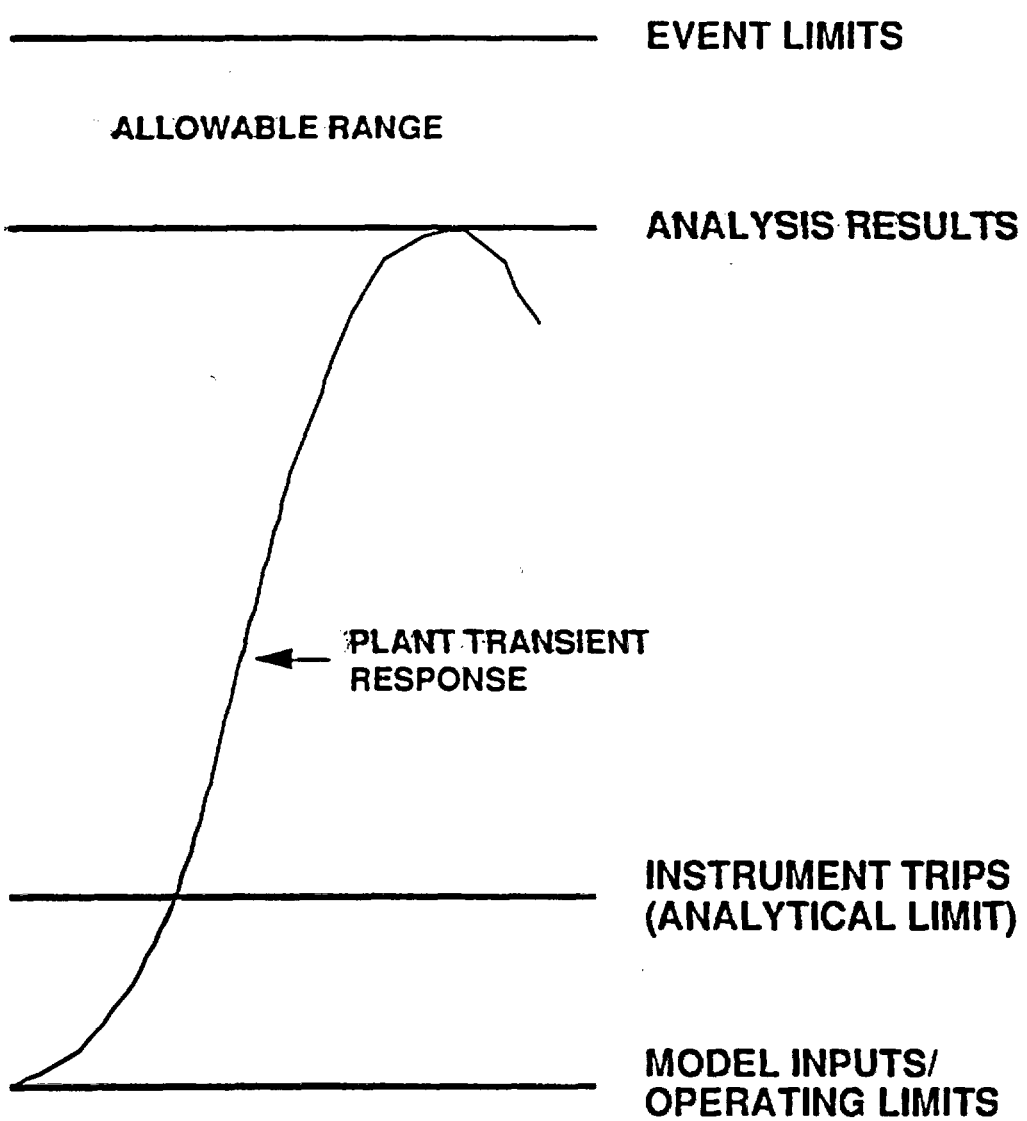
EPRI Setpoint Projects

Existing Methodology

- Existing Methods are Based on Deterministic or License Basis Analysis Assumptions, Including the Most Adverse Plant Conditions.
- Highly Conservative Methodology Restricts Operational Flexibility and May Result in Derates
- Major Improvements Achievable by Quantifying and Combining Uncertainties in Key Variables

CURRENT SAFETY ANALYSIS PROCESS

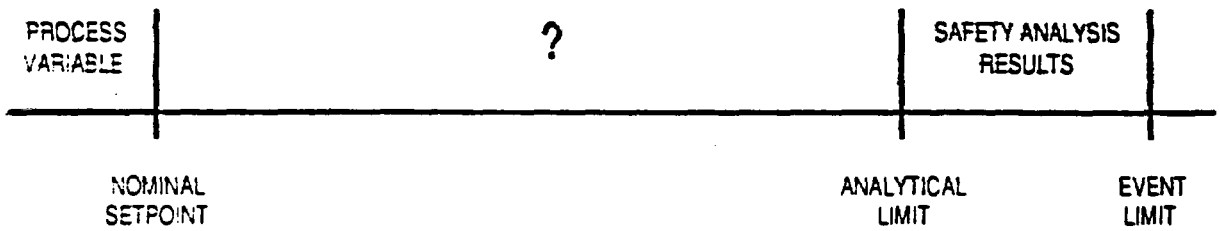
(Deterministic Methodology)



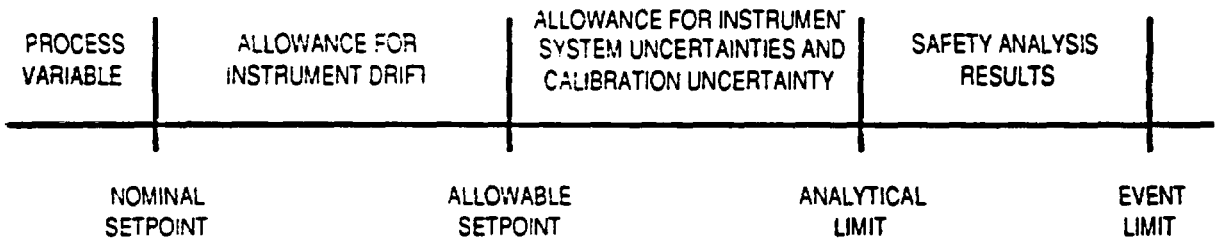
CURRENT TREATMENT OF MEASUREMENT UNCERTAINTIES

THREE TYPICAL APPROACHES

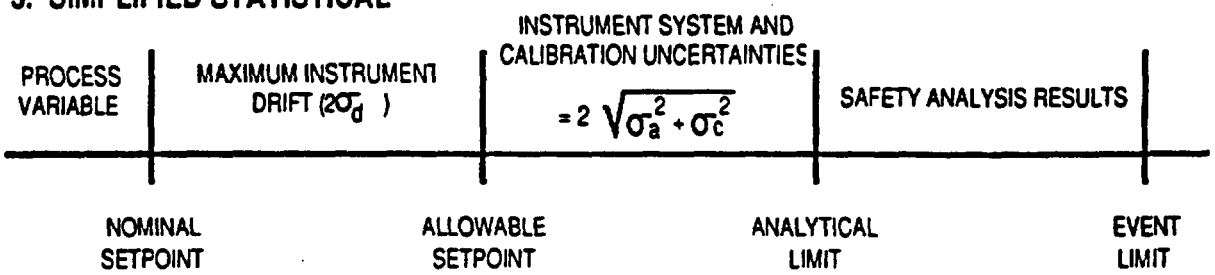
1. HISTORICAL



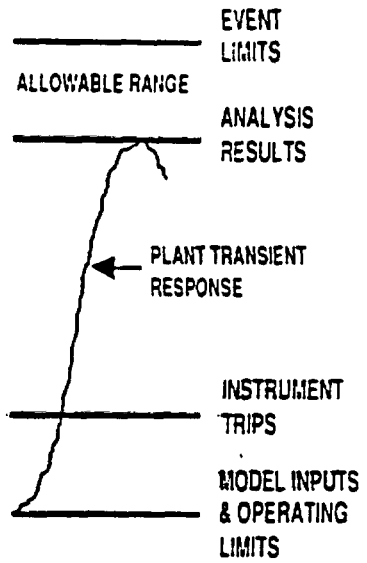
2. ENGINEERING JUDGMENT



3. SIMPLIFIED STATISTICAL



POTENTIAL PROBLEMS

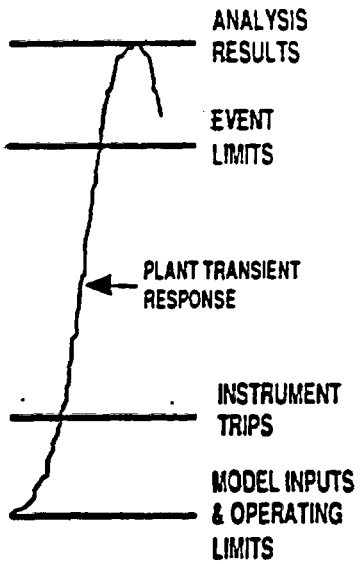


Deterministic Acceptable

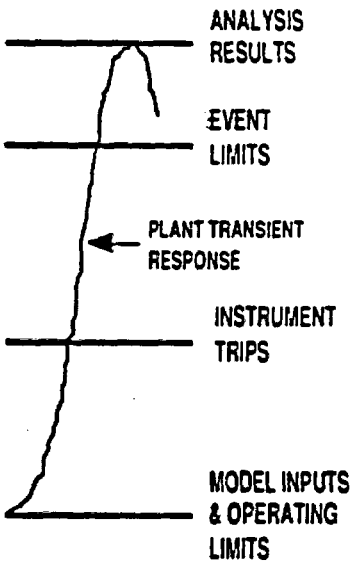
RESULTS ACCEPTABLE

Statistical Analysis May Be Solution

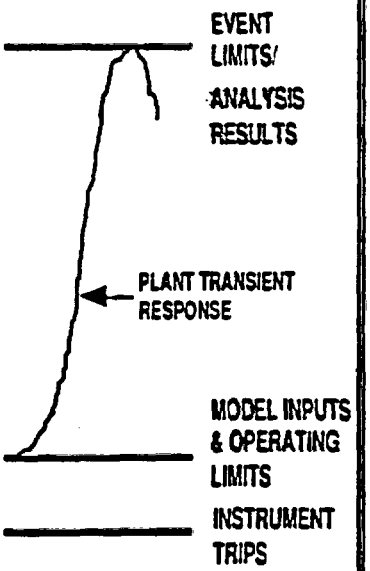
RESULTS UNACCEPTABLE



ANALYSIS RESULTS UNACCEPTABLE



INCREASED OPERATING MARGIN



SETPOINT OVERLAP

EPRI Setpoint Projects Methodology and Applications

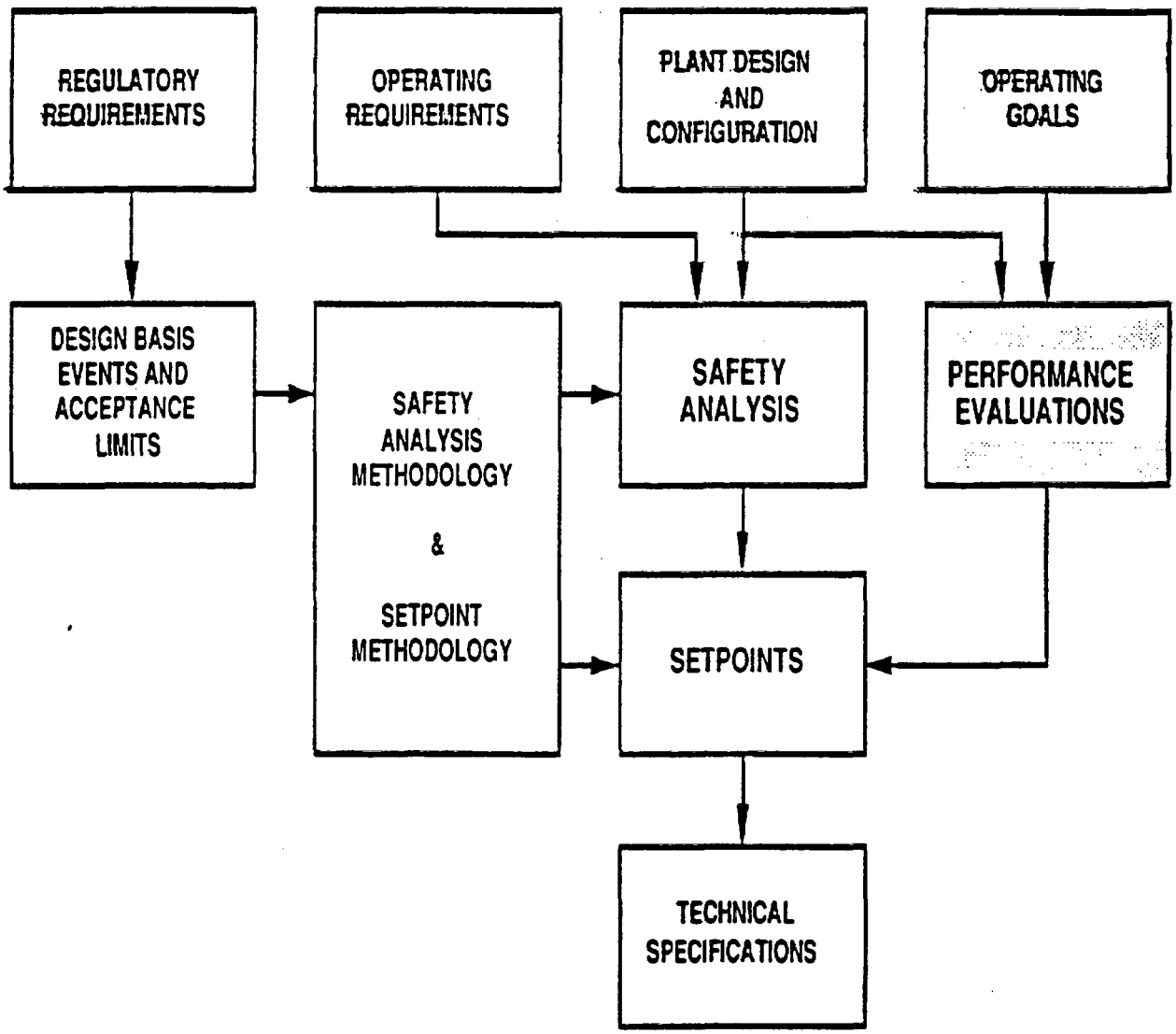
OVERVIEW

EPRI SETPOINT METHODOLOGY

- *Safety Analysis and Setpoint Process*
- *Safety Analysis and Measurement Uncertainties*
- *Setpoint Modification Process*
- *Methodology Options*
- *Project Plan Development*
- *Statistical Analysis Methodology*
- *Statistical Combination of Uncertainties*
- *Network Considerations*
- *Combined Methodologies*

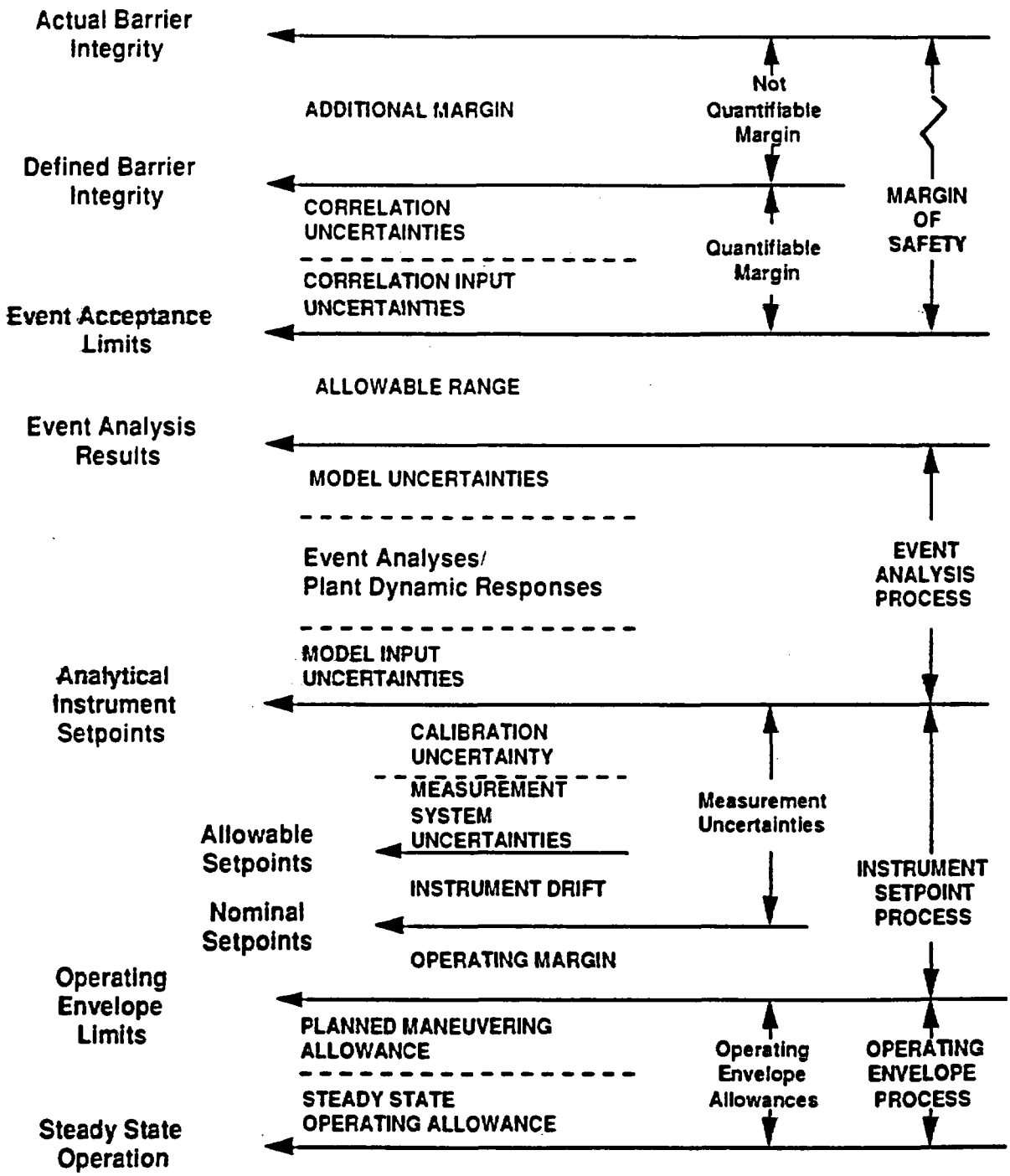
EPRI SETPOINT PROJECTS

GENERALIZED SAFETY ANALYSIS AND SETPOINT PROCESS



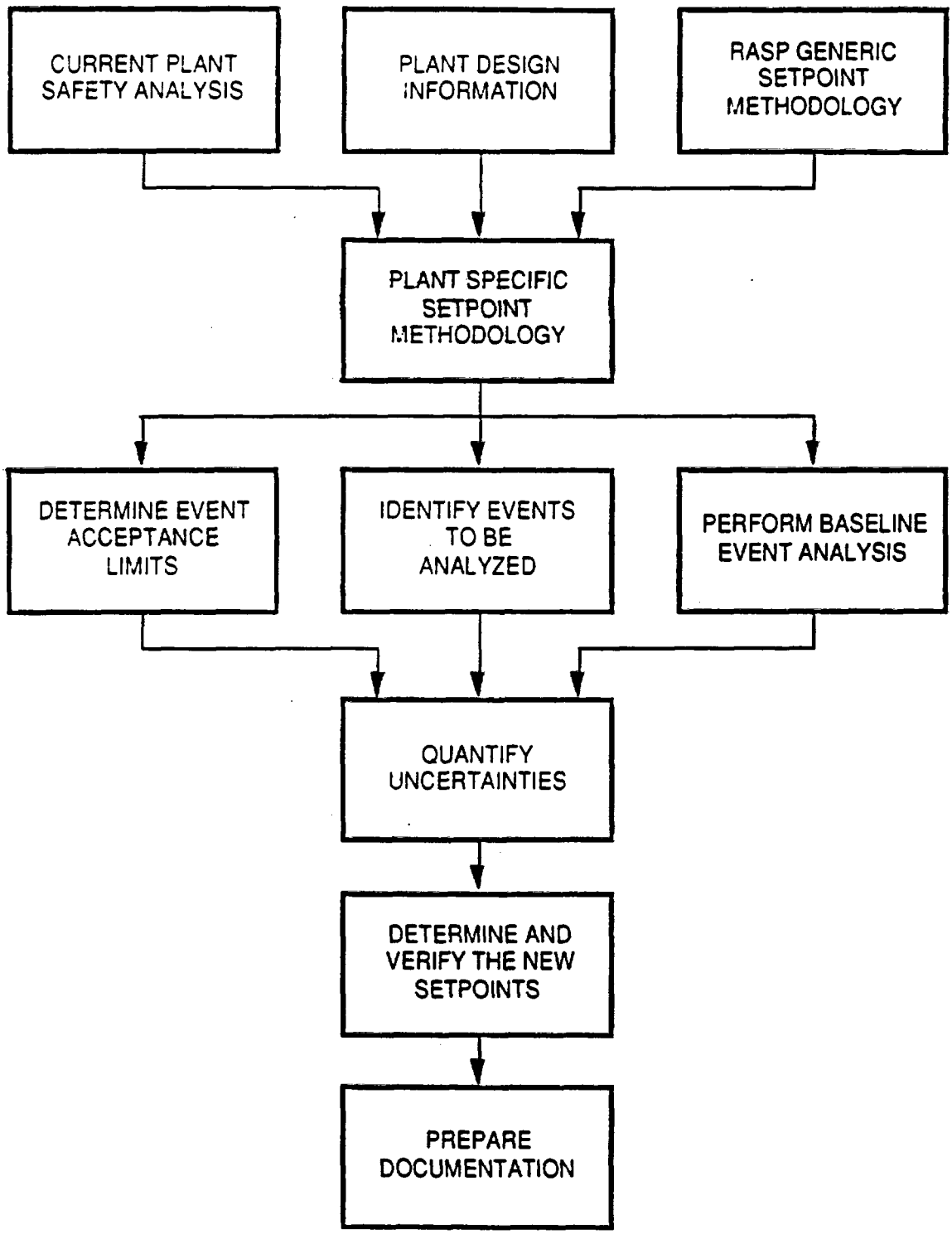
EPRI/NPD

SAFETY ANALYSIS RELATED SETPOINTS AND UNCERTAINTIES



S & R/ Reactor Performance Program

SETPOINT MODIFICATION PROCESS



Description of Methodology Options

Deterministic

- Conservative selection of analysis inputs
- Analytical instrument setpoints at 95% probability
- No additional model uncertainty required
- Results very conservative
- Used in current safety analyses

Complete SCU

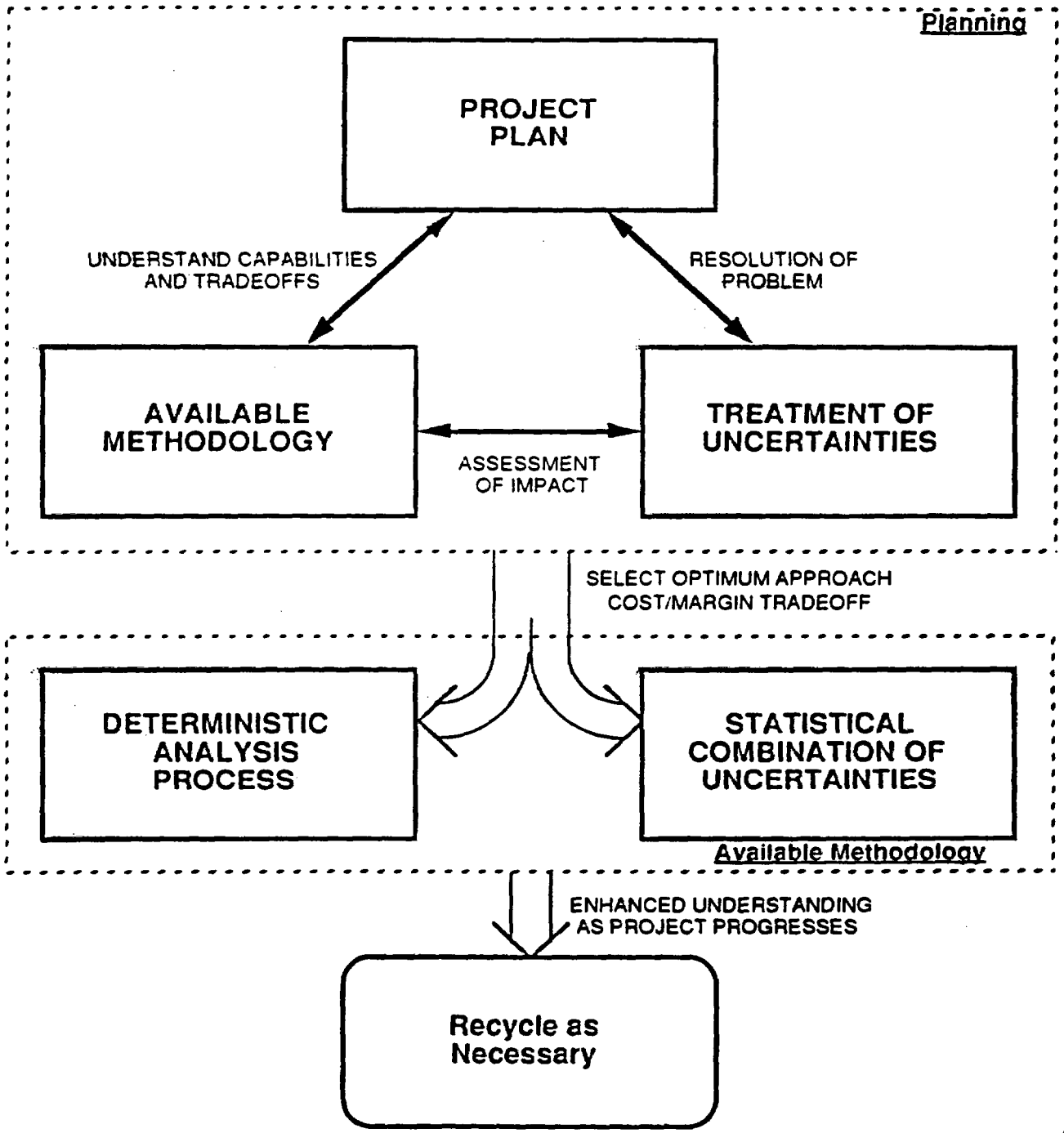
- All analysis inputs statistically treated
- Results determined at the 95% probability/95% confidence level
- Model uncertainty considered in the analysis
- Minimum amount of conservatism in results
- Only used to determine DNBR/MCPR safety limits

Mixed SCU

- Limited number of analysis inputs statistically treated
- Remainder of inputs same as deterministic process
- Model uncertainty considered in the analysis
- Results determined at the 95% probability level for the statistically treated parameters
- Intermediate amount of conservatism results
- Increasing usage in the safety analysis process

EPRI/NPD

SETPOINT MODIFICATION PROJECT PLAN DEVELOPMENT



S & R/ Reactor Performance Program

Statistical Combination of Uncertainties

- Methodology
- Data Manipulation
- Relationship to Screening Process
- Analysis Input Screening
- Response Surface Development
- STARS Methodology
- Benefits

STATISTICAL COMBINATION OF UNCERTAINTY METHODOLOGY

IDENTIFY EVENT LIMITS

↓

DETERMINE TREATMENT OF INPUT PARAMETERS - SCREENING PROCESS

1. Identify Analysis Input Parameters to Be Treated Deterministically
2. Identify Analysis Input Parameters to Be Treated Statistically
3. Characterize Statistical Inputs

↓

DEVELOP UNCERTAINTY DISTRIBUTION FOR THE EVENT BEING ANALYZED - SCU PROCESS

1. Develop Response Surface
2. Simulate Event to Obtain Combined Effect of Uncertainties Including the Uncertainties in the Response Surface Fit

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DETERMINE EVENT PROBABILITY AS A FUNCTION OF THE EVENT LIMITS

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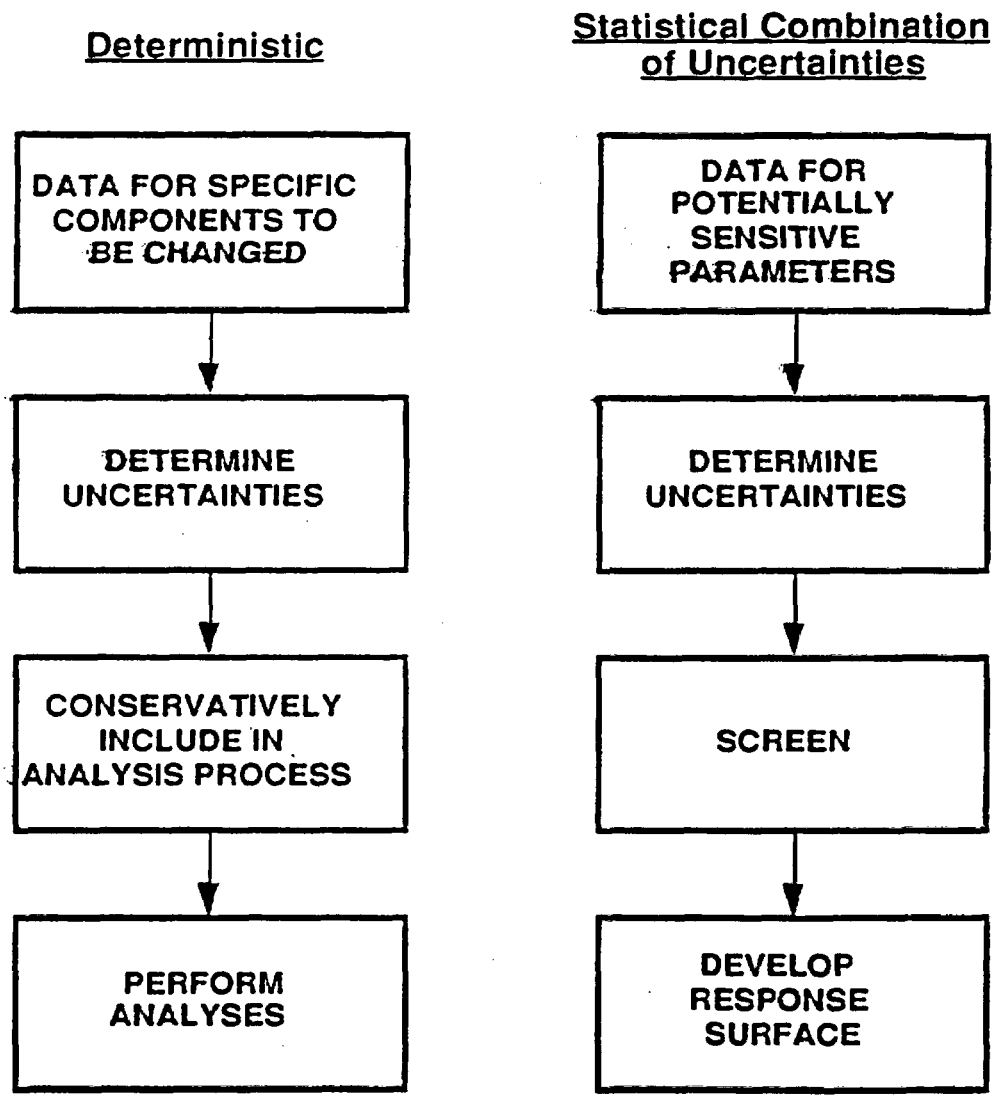
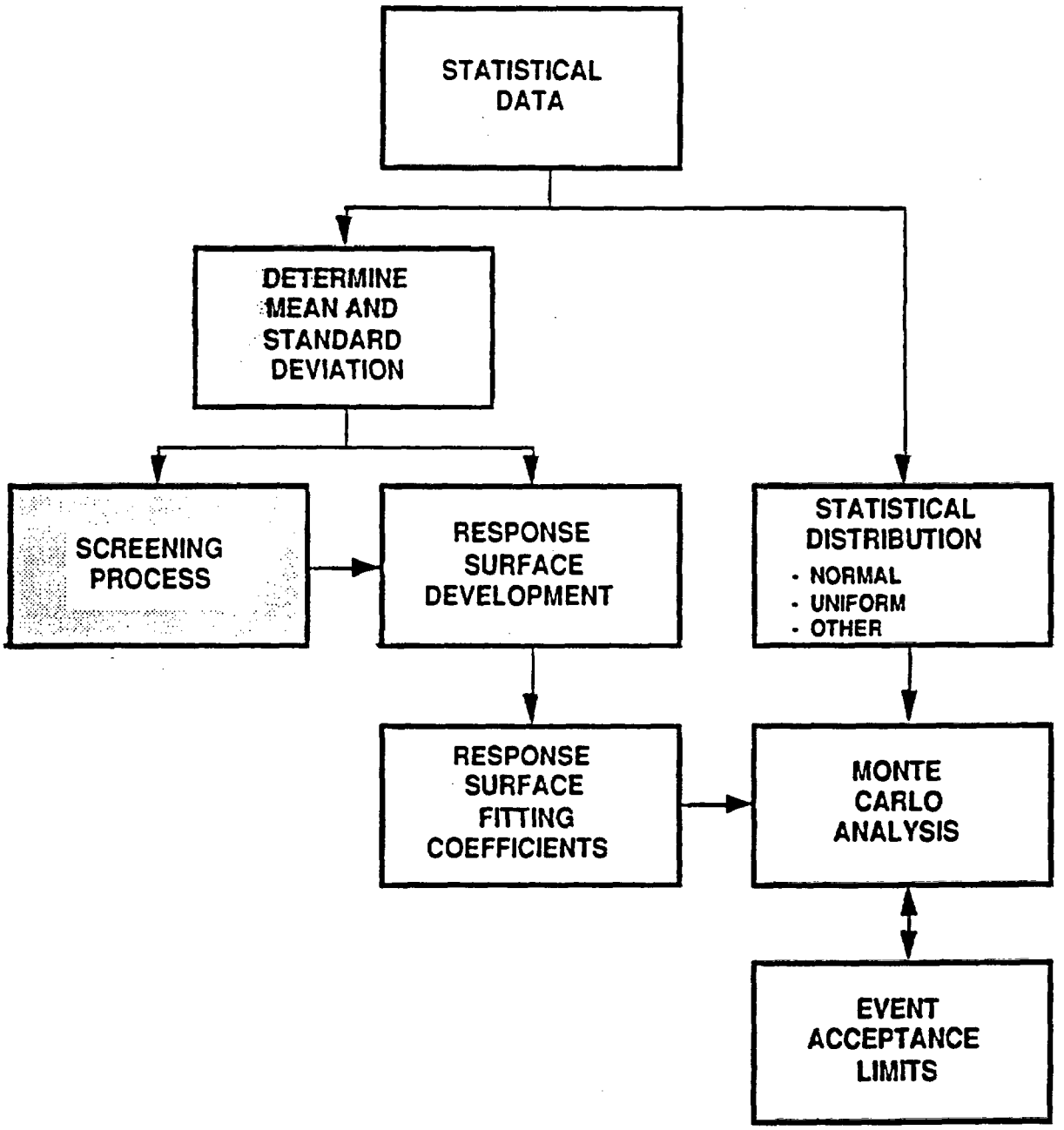
COMPARISON TO THE EVENT ACCEPTANCE LIMITS

DATA MANIPULATION FOR SETPOINT MODIFICATION ANALYSES



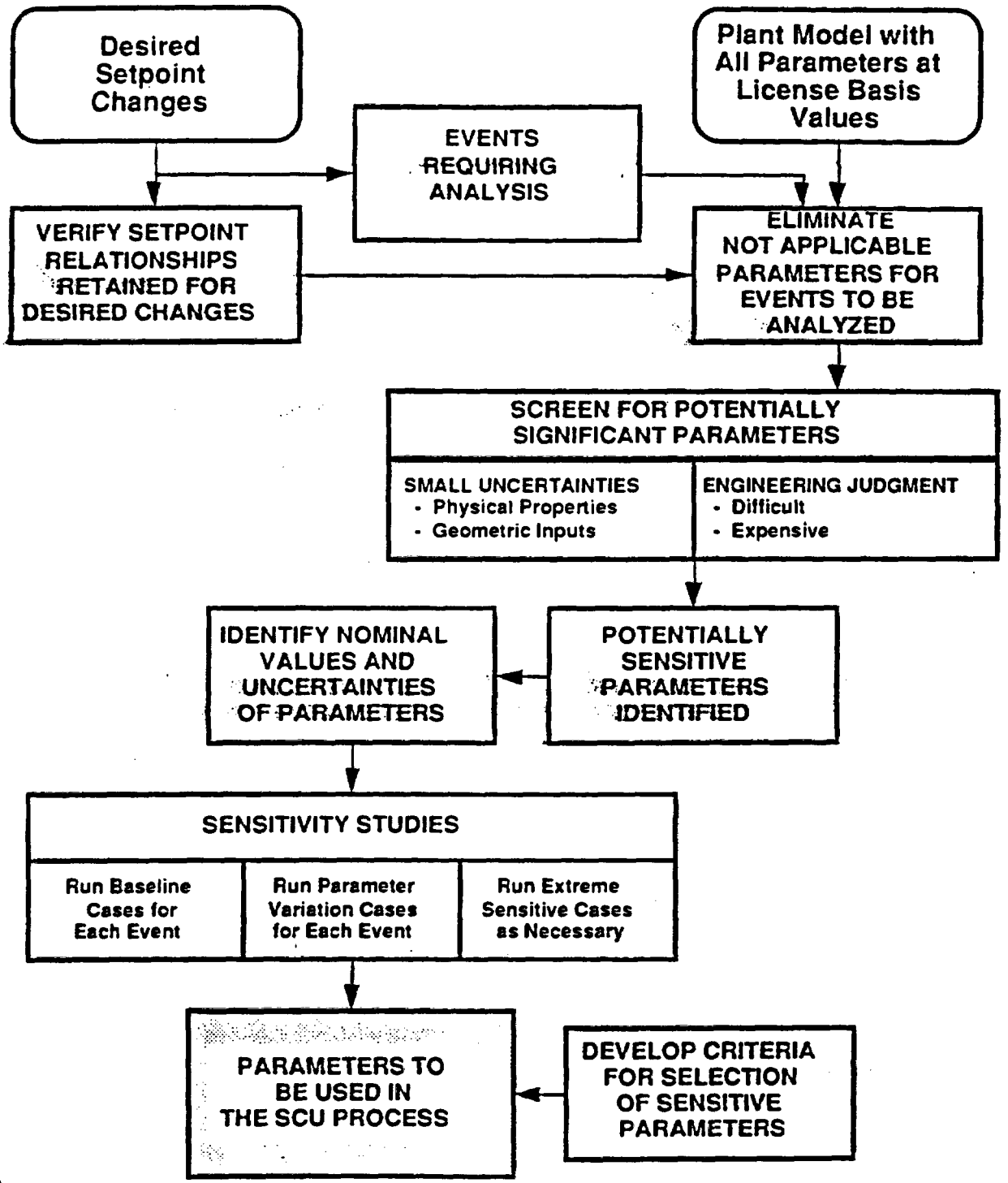
Process is Similar, but Number of Parameters and Their Treatment Is More Extensive for SCU Process

RELATIONSHIP OF SCREENING PROCESS TO SCU METHODOLOGY



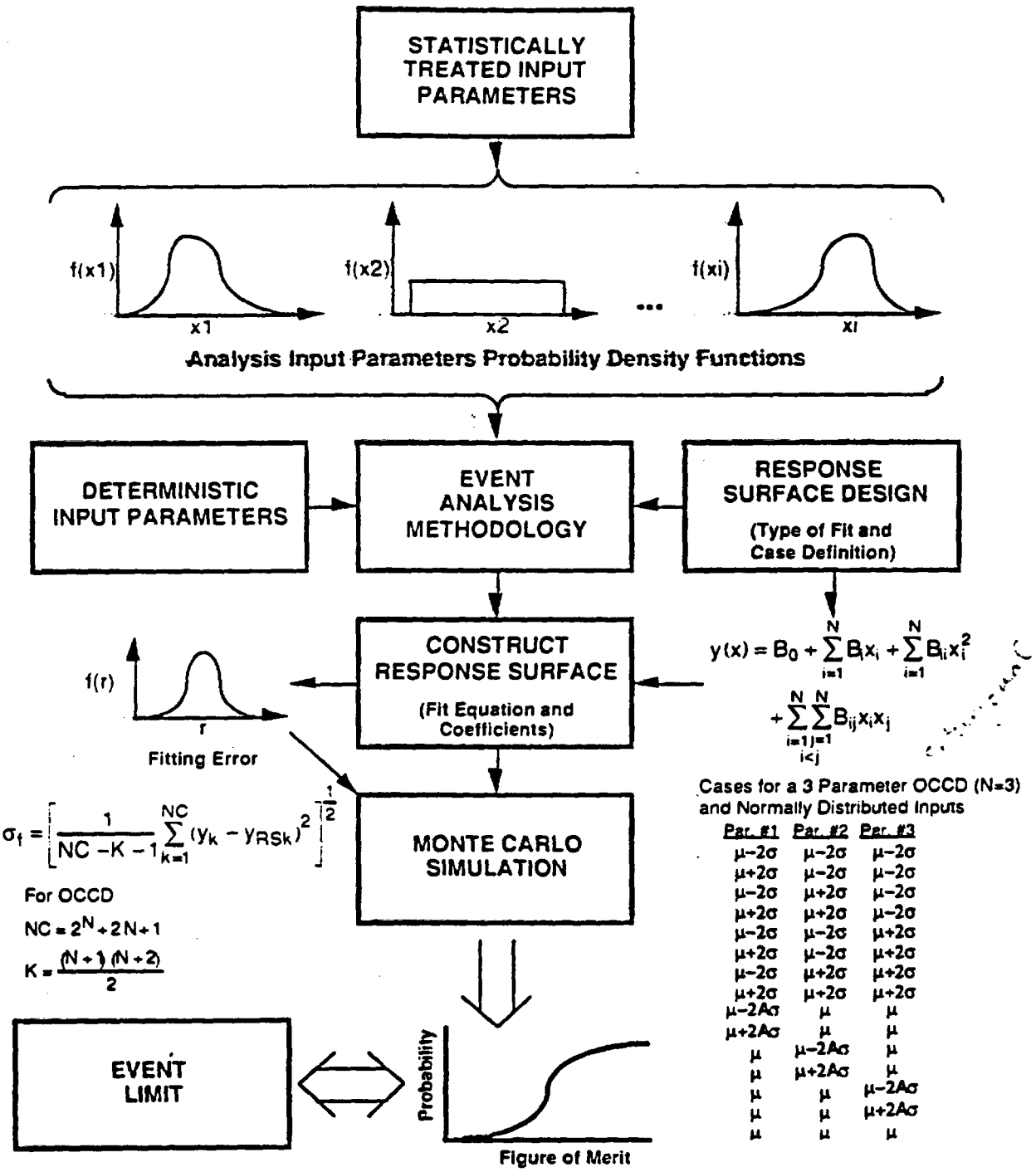
EPRI/NPD

ANALYSIS INPUT SCREENING PROCESS

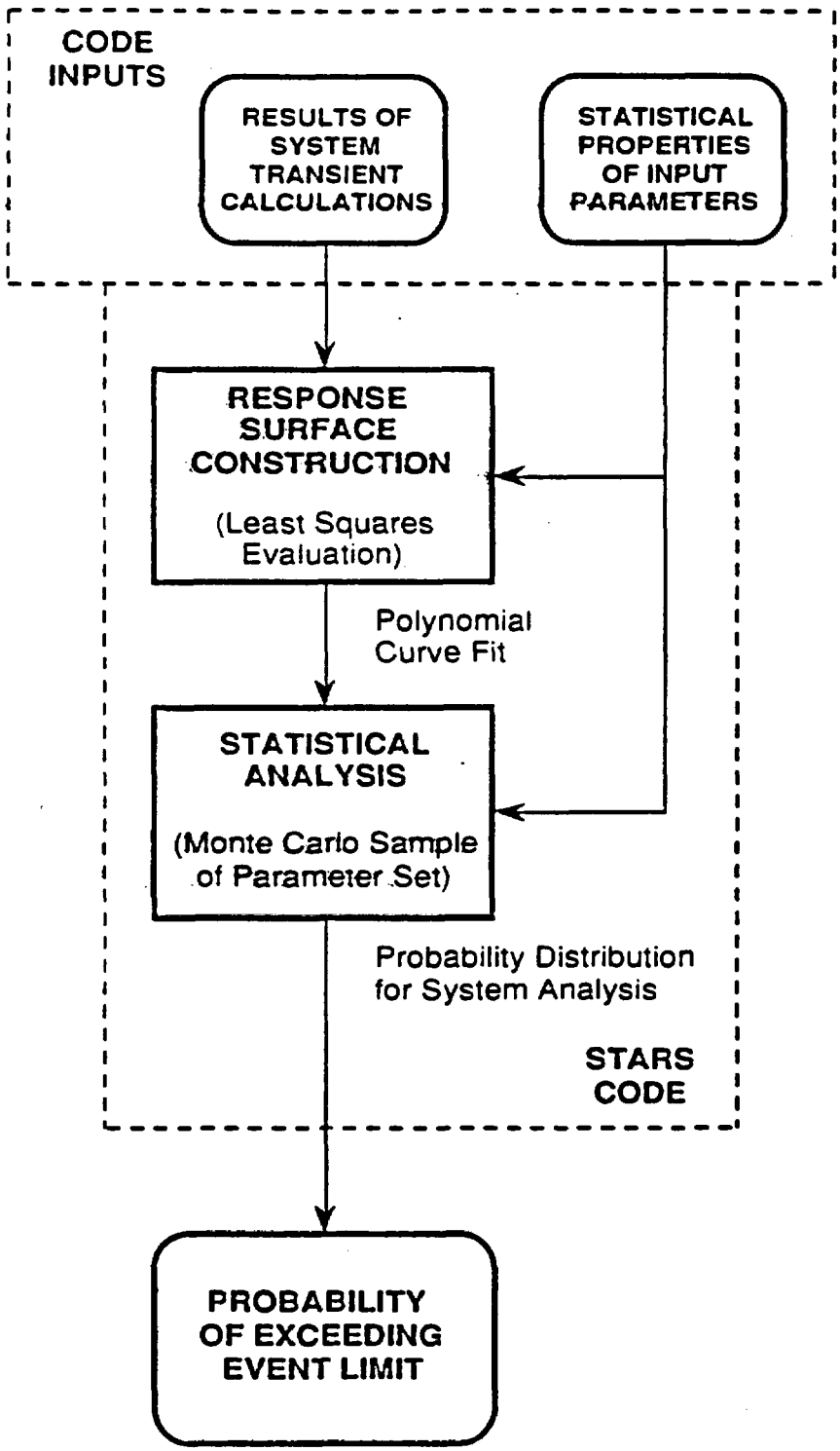


S & R/ Reactor Performance Program

STATISTICAL COMBINATION OF UNCERTAINTIES PROCESS RESPONSE SURFACE DEVELOPMENT



STARS METHODOLOGY



BENEFITS OF STATISTICAL COMBINATION OF UNCERTAINTIES METHODOLOGY

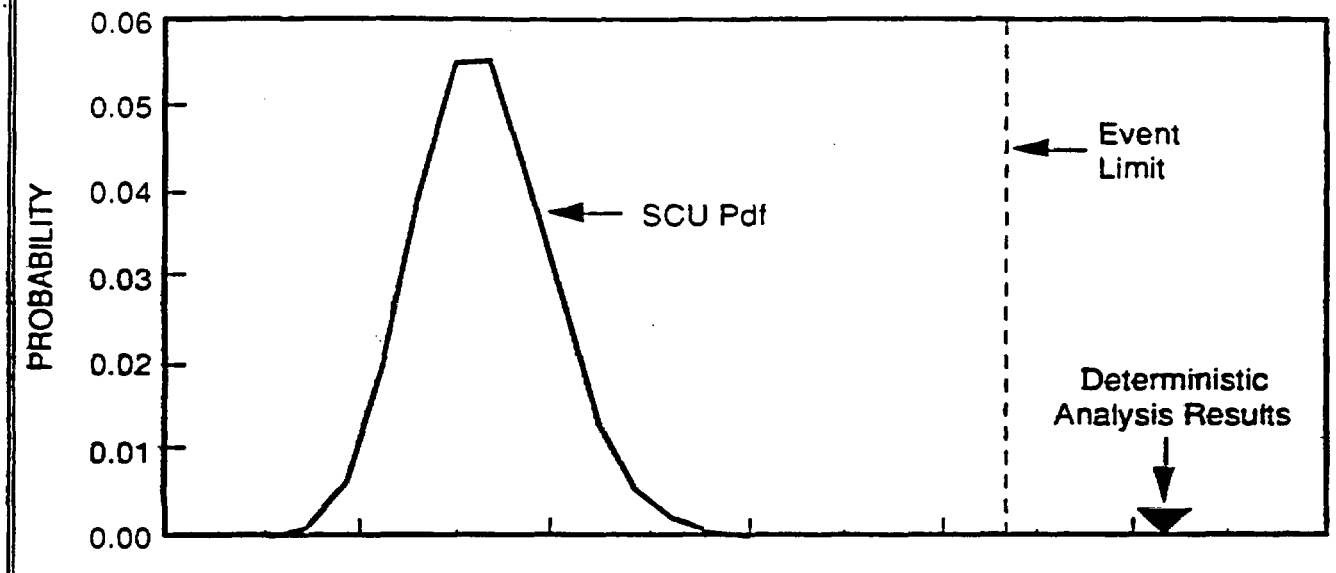


FIGURE OF MERIT
Probability Density Function

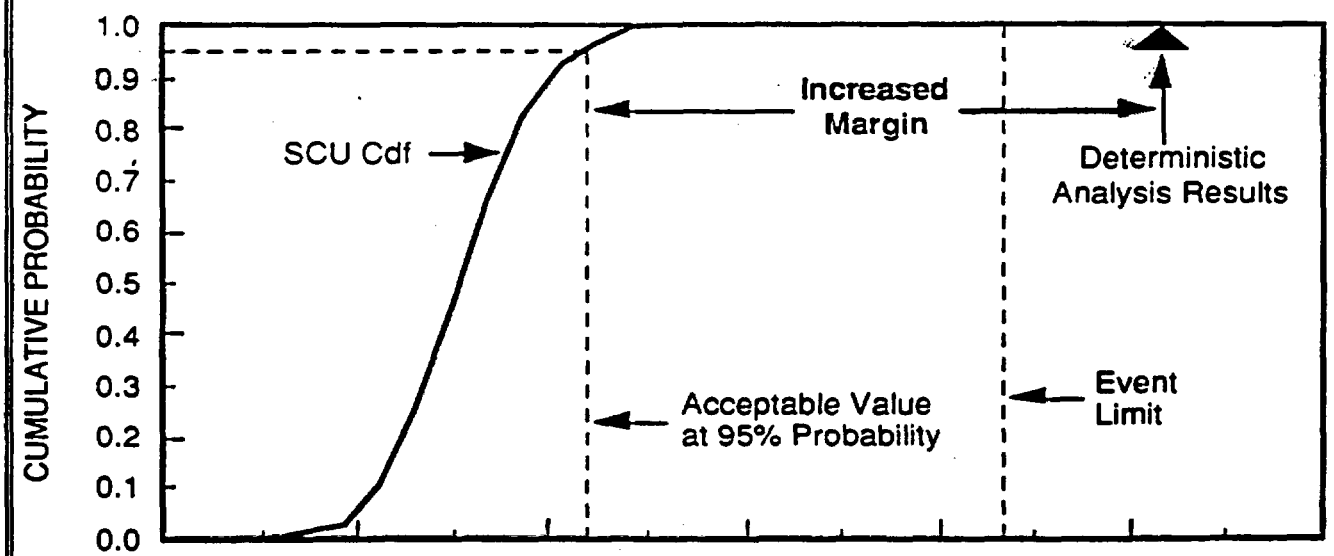


FIGURE OF MERIT
Cumulative Distribution Function

Network Analysis

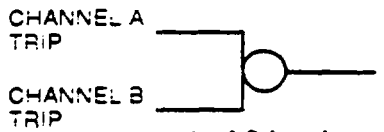
- Network Considerations
- Typical Component Networks
- Development of Analysis Inputs
- PLANETS Methodology
- Typical Individual Component Performance
- Examples of Network Performance
- Event Analysis Inputs

Network Considerations

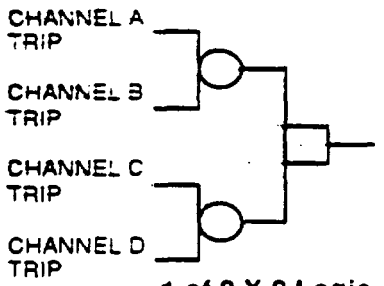
- Component networks are frequently encountered in nuclear power plant design
  - Single failure considerations
  - Avoidance of spurious trips or system startups
  - Improved system reliability
- Safety Analyses Typically Treat Component Networks as Individual Components
  - Deterministic methodology philosophy
  - Computer code model limitations
- Component Networks
  - Superior to individual components
  - Methodology available to quantify benefits

TYPICAL COMPONENT NETWORKS

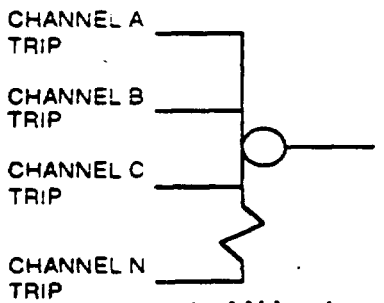
1 of 1 Logic



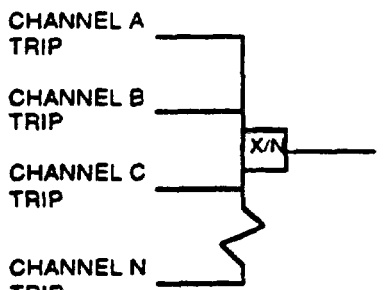
1 of 2 Logic



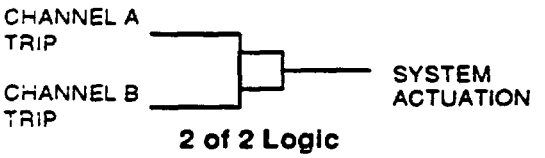
1 of 2 X 2 Logic



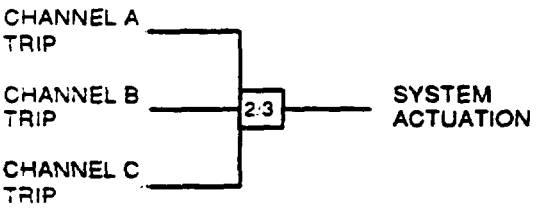
1 of N Logic



X of N Logic



2 of 2 Logic



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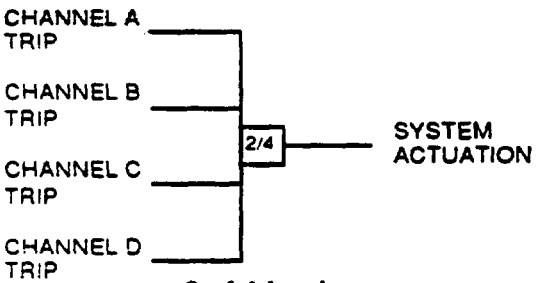
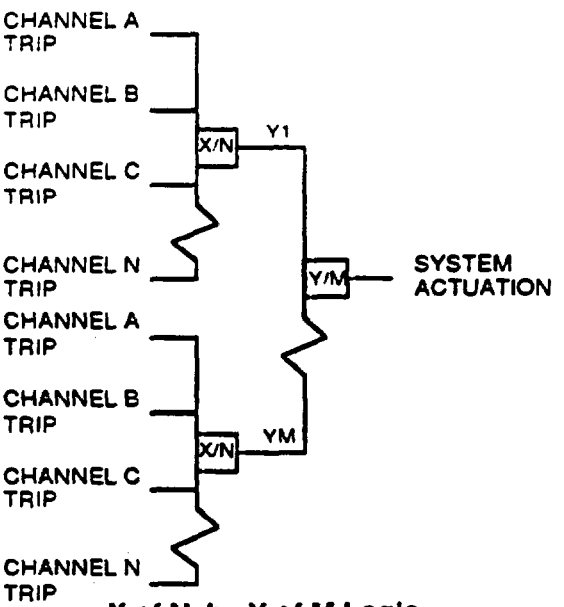


2 of 4 Logic



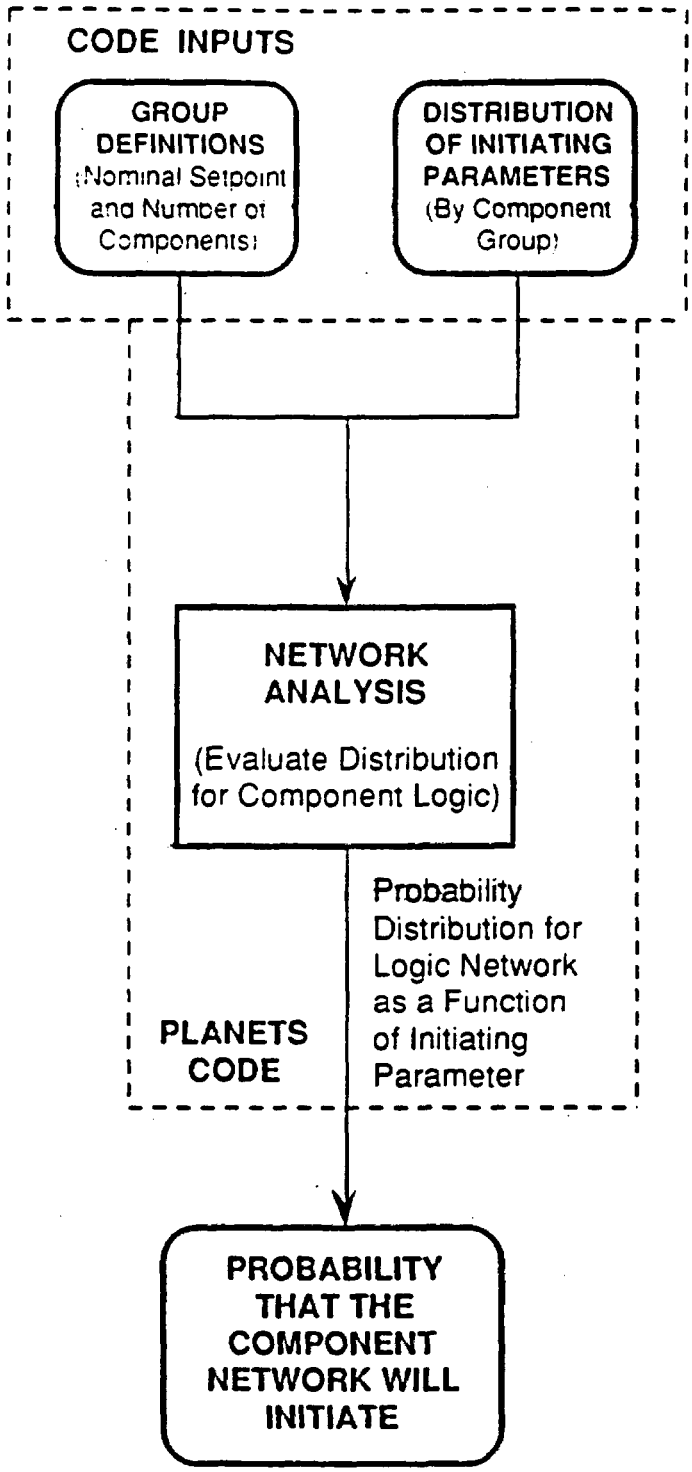
X of N by Y of M Logic

Development Of Analysis Inputs For Component Networks

- Analyses Typically Simulate One Component - Even for Component Networks
- Component Performance can be Statistically Based (95% Probability)
- Network Performance is Superior to Individual Components
- Equivalent Network Performance can be Defined
- Network Inputs can be Developed Consistent with Deterministic or Statistical Combination of Uncertainties Methodology

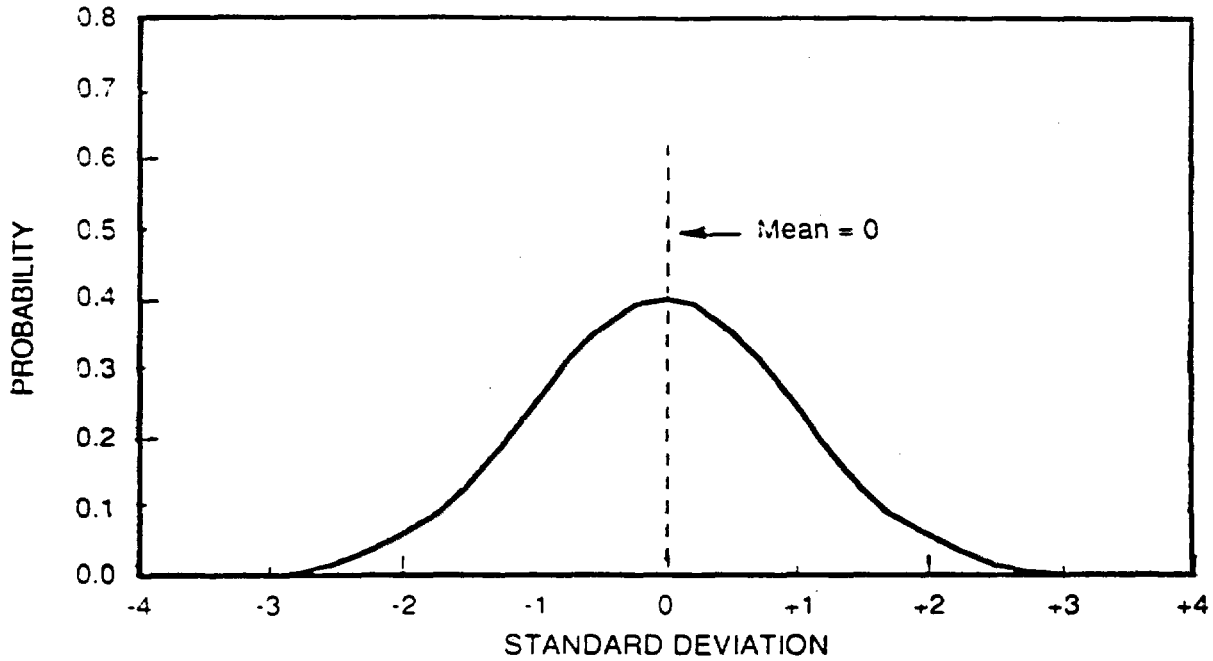
EPRI/NPD

PLANETS METHODOLOGY

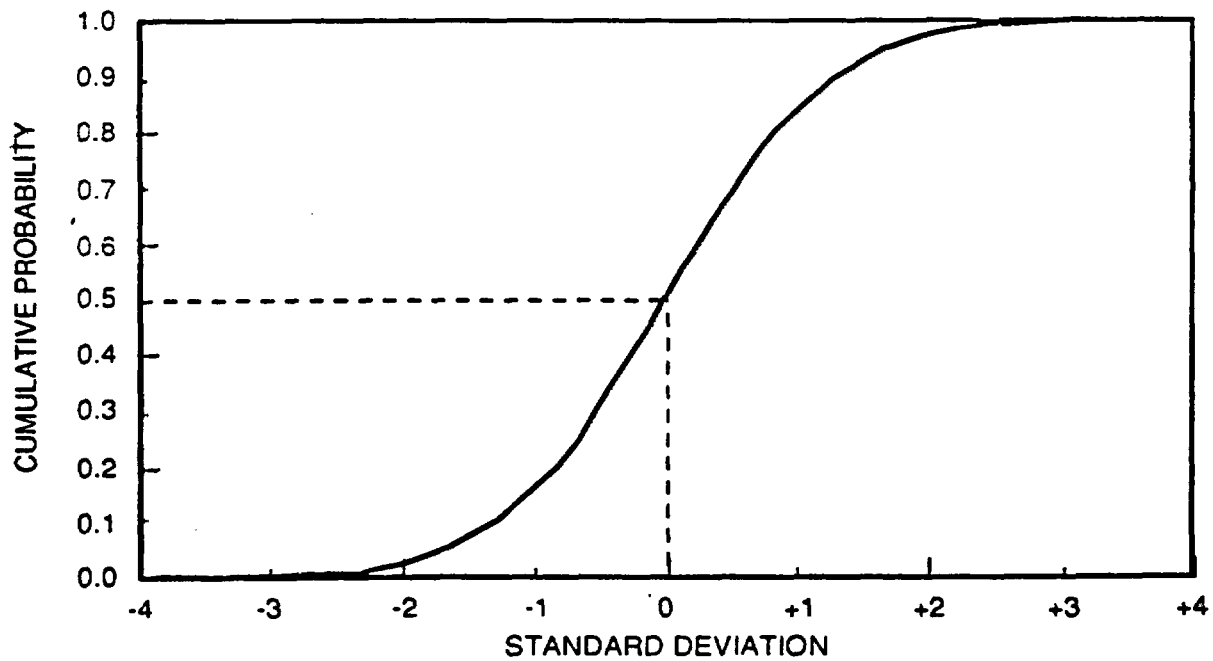


S & R/ Reactor Performance Program

TYPICAL INDIVIDUAL COMPONENT PERFORMANCE

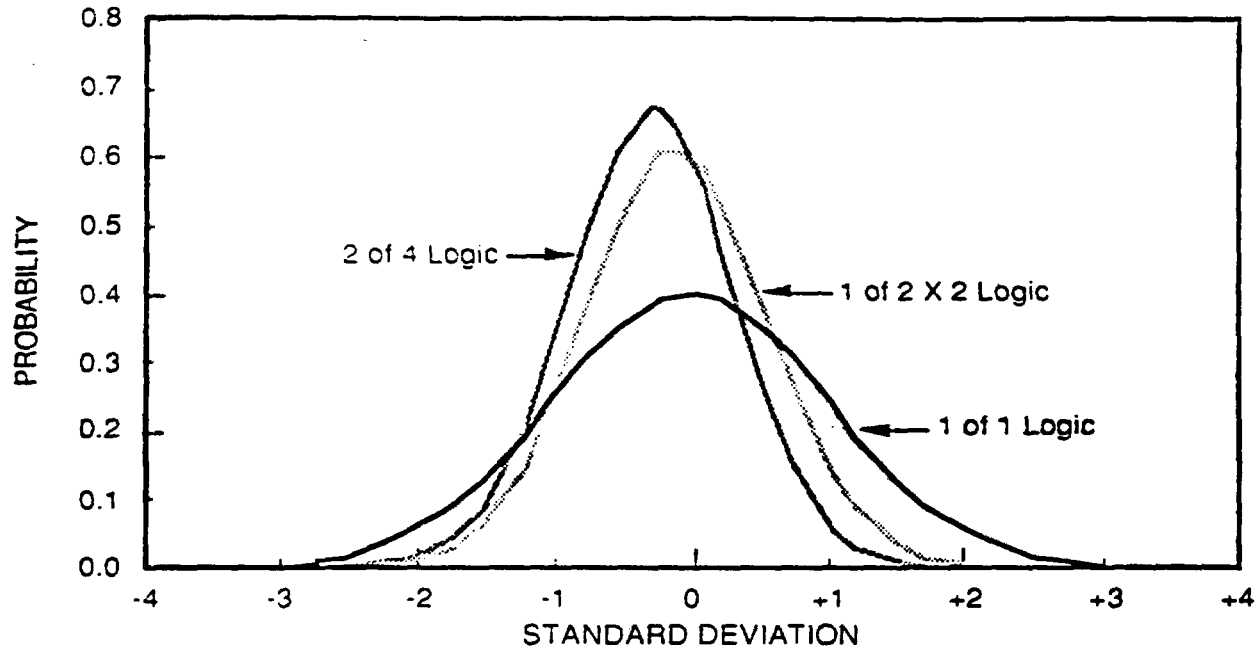


Probability Density Function

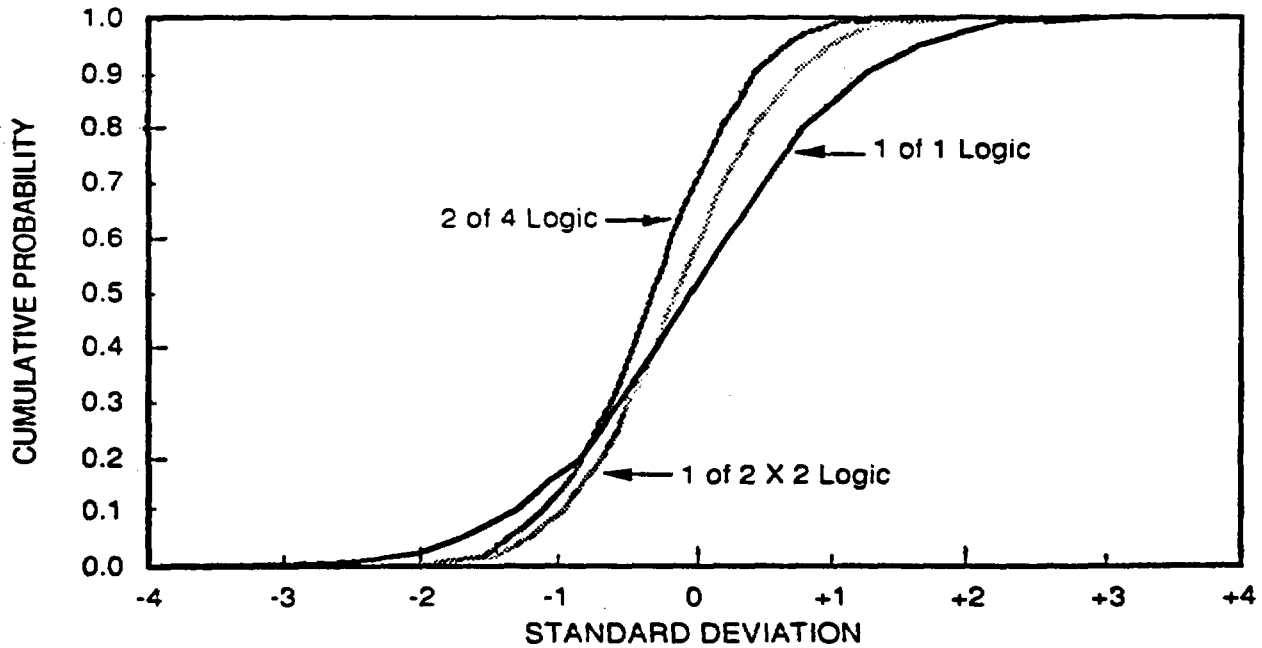


Cumulative Distribution Function

EXAMPLES OF NETWORK PERFORMANCE



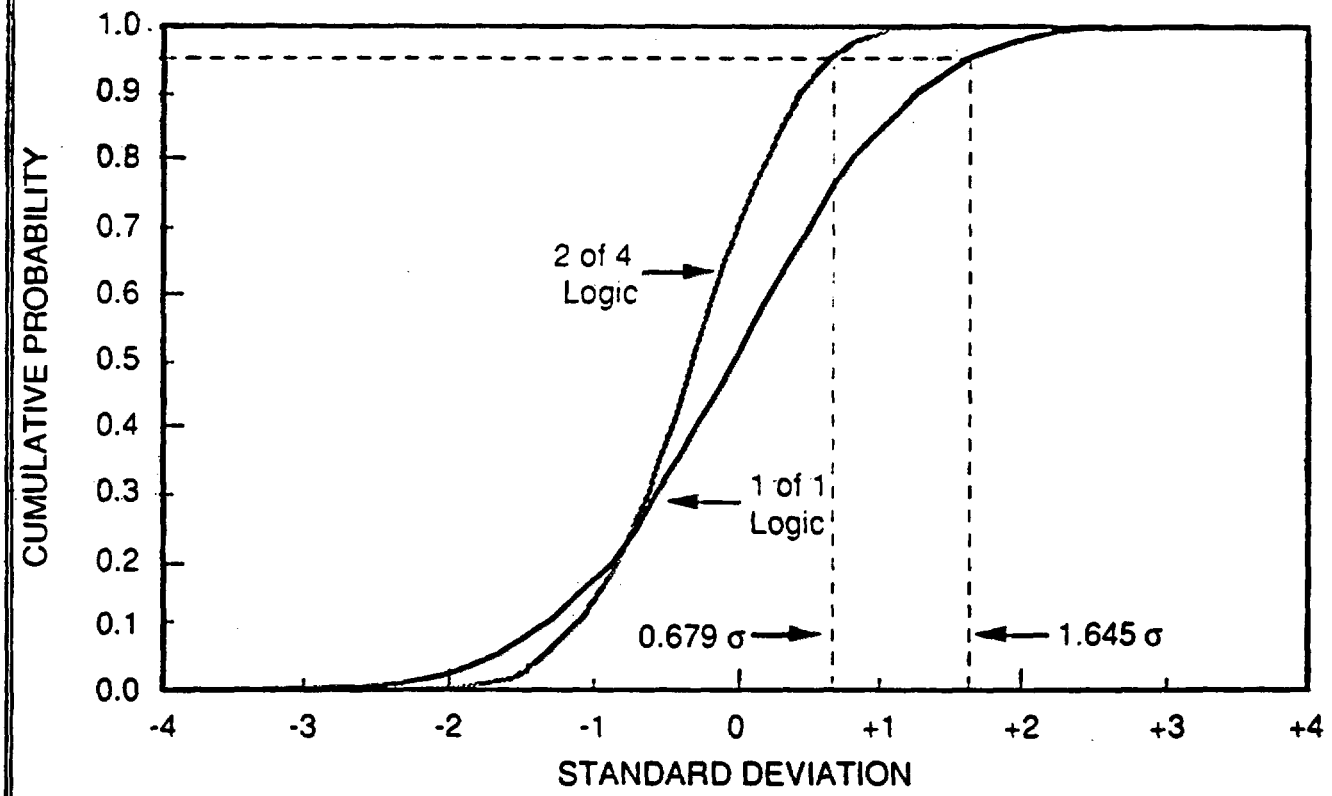
Network Probability Density Function



Network Cumulative Distribution Function

DEVELOPMENT OF EVENT ANALYSIS INPUTS FOR SIMULATING COMPONENT NETWORKS

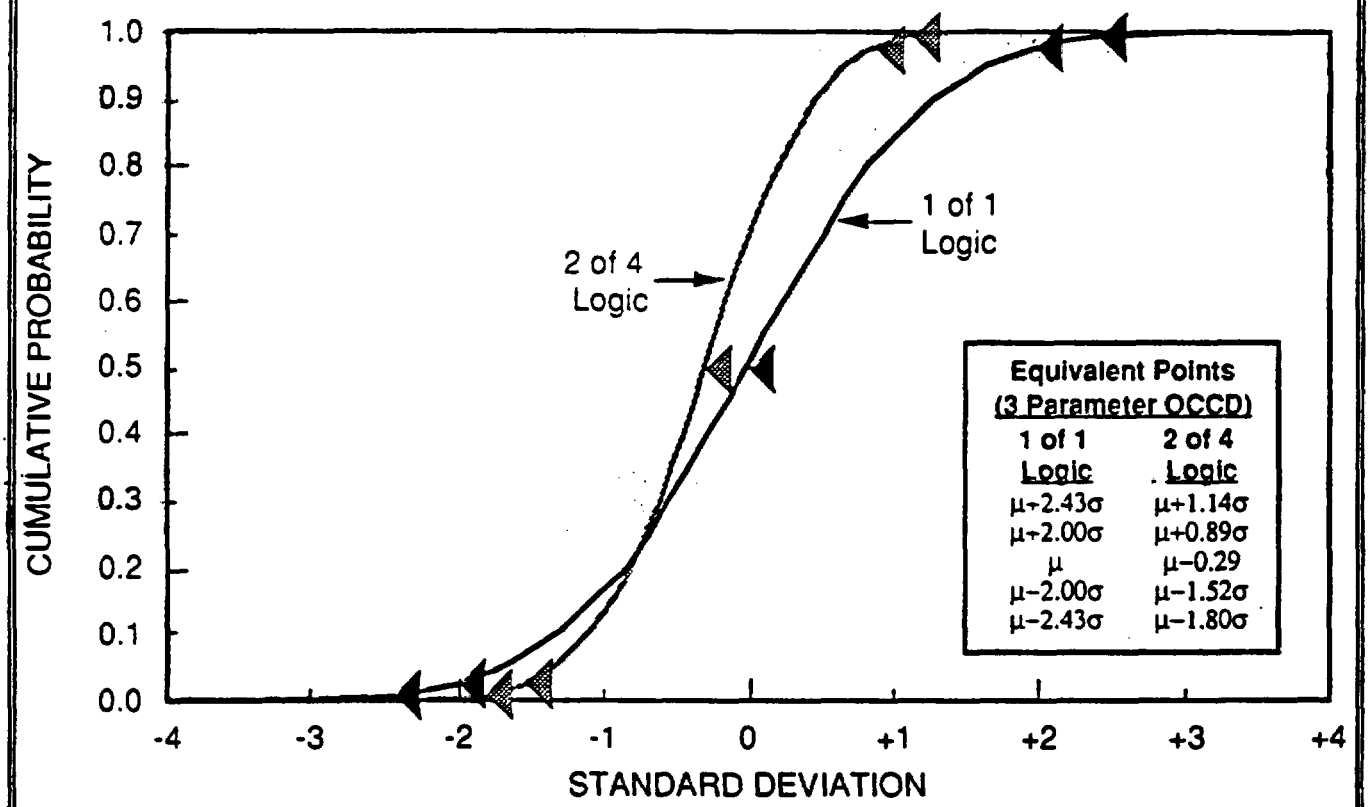
(Deterministic Methodology)



Network Cumulative Distribution Function

DEVELOPMENT OF EVENT ANALYSIS INPUTS FOR SIMULATING COMPONENT NETWORKS

(Statistical Combination of Uncertainties Methodology)

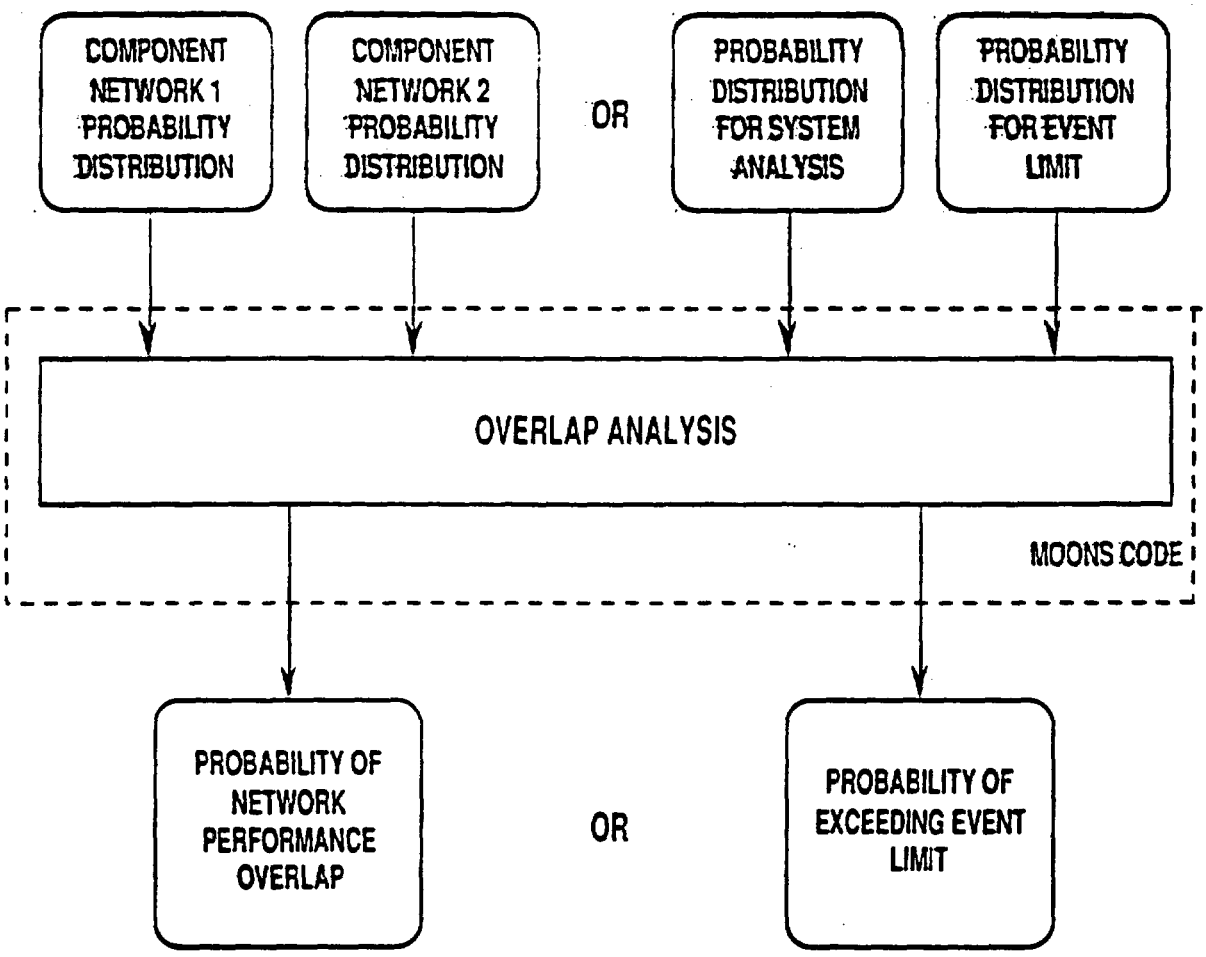


Network Cumulative Distribution Function

Combined Methodologies

- Overview
- Network Performance Overlap
- Overlap Example
- Code Relationships
- Evaluation of Statistical Limits

COMBINED METHODOLOGIES OVERVIEW

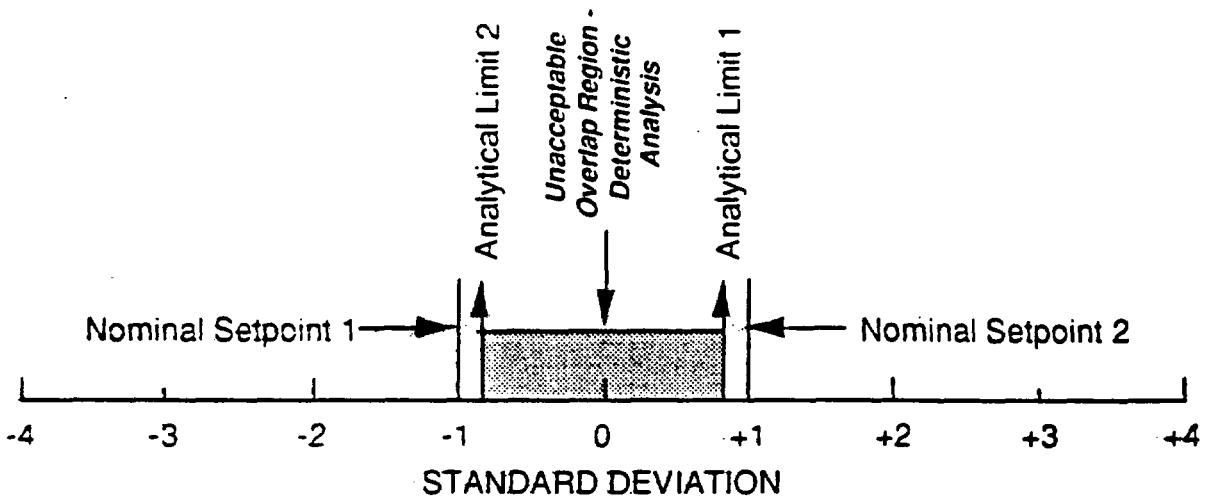


Network Performance Overlap

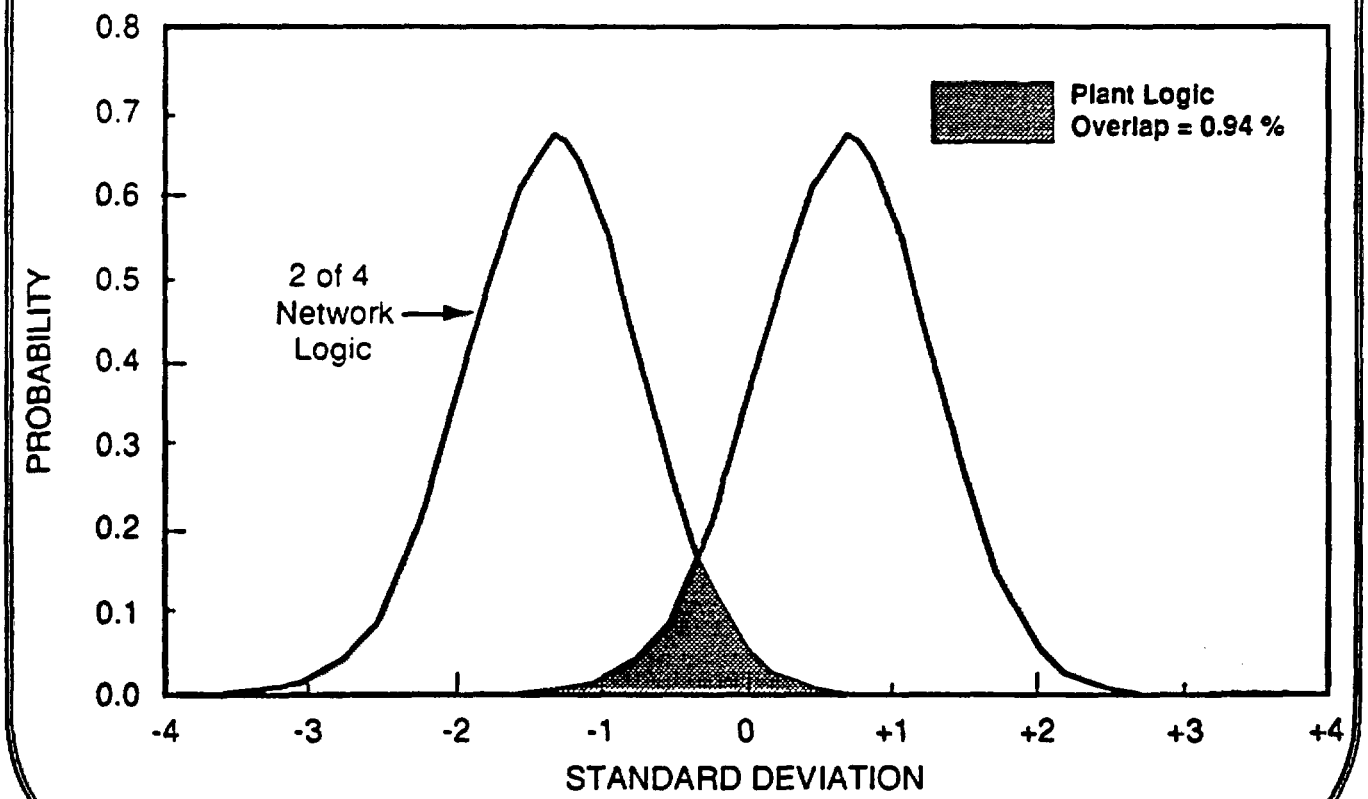
- System actuation overlaps can occur
  - Nominal setpoints in close proximity
  - Large measurement uncertainties
  - Associated with deterministic treatment of analytical limit
- Network performance may be acceptable
  - Low probability of overlap acceptable for many applications
  - Statistical assessment of overlap feasible

EPRI/NPD

NETWORK PERFORMANCE OVERLAP

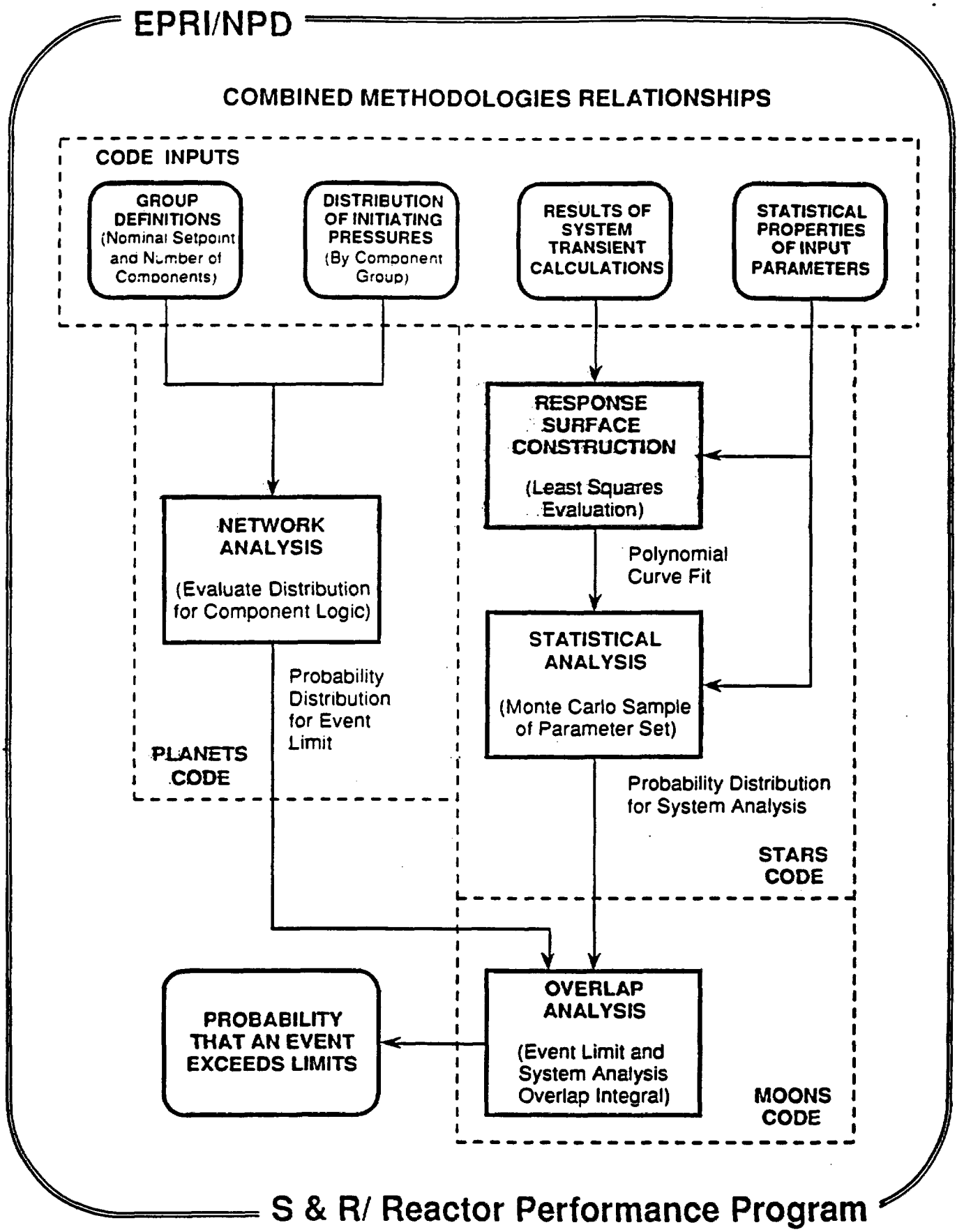


Deterministic Setpoints



Network Probability Density Function

S & R/ Reactor Performance Program


EPRI/NPD
COMBINED METHODOLOGIES RELATIONSHIPS
CODE INPUTS
GROUP DEFINITIONS (Nominal Setpoint and Number of Components)
DISTRIBUTION OF INITIATING PRESSURES (By Component Group)
RESULTS OF SYSTEM TRANSIENT CALCULATIONS
STATISTICAL PROPERTIES OF INPUT PARAMETERS
RESPONSE SURFACE CONSTRUCTION (Least Squares Evaluation)
NETWORK ANALYSIS (Evaluate Distribution for Component Logic)
Polynomial Curve Fit
STATISTICAL ANALYSIS (Monte Carlo Sample of Parameter Set)
Probability Distribution for Event Limit
PLANETS CODE
Probability Distribution for System Analysis
STARS CODE
PROBABILITY THAT AN EVENT EXCEEDS LIMITS
OVERLAP ANALYSIS (Event Limit and System Analysis Overlap Integral)
MOONS CODE
S & R/ Reactor Performance Program

EVALUATION OF STATISTICAL LIMITS

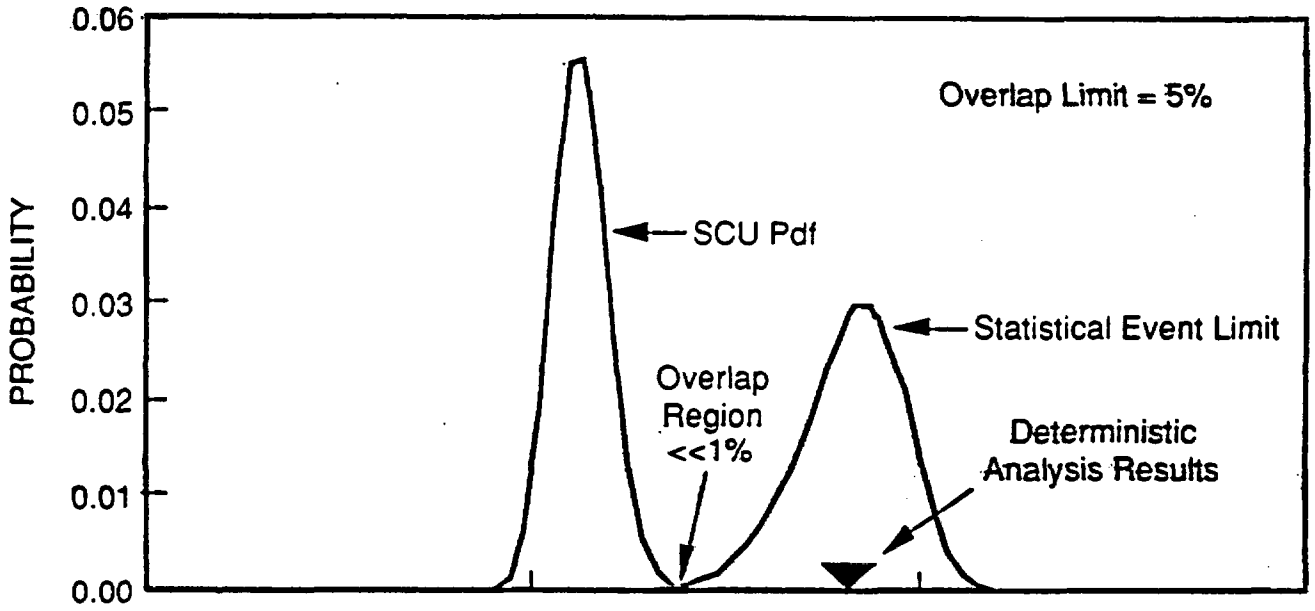


FIGURE OF MERIT
Probability Density Function

EPRI Setpoint Projects Methodology and Applications

OVERVIEW

EPRI Setpoint Methodology

EPRI Setpoint Projects

Oyster Creek High Pressure Setpoints

- Application Overview
- High Pressure Setpoint Relationships
- Current and Proposed Setpoints
- Analysis Considerations
- Licensing Basis Events Selection Process
- Plant Performance Events Selection Process
- Event Assessment
- Analysis Results
- Approach to Safety Valve Opening Probability
- Conclusion

EPRI Setpoint Projects Methodology and Applications

Vermont Yankee Reload Analysis
- Application Overview
- Approach and Results

Diablo Canyon Safety Valve Tolerance
- Application Overview
- Approach and Results

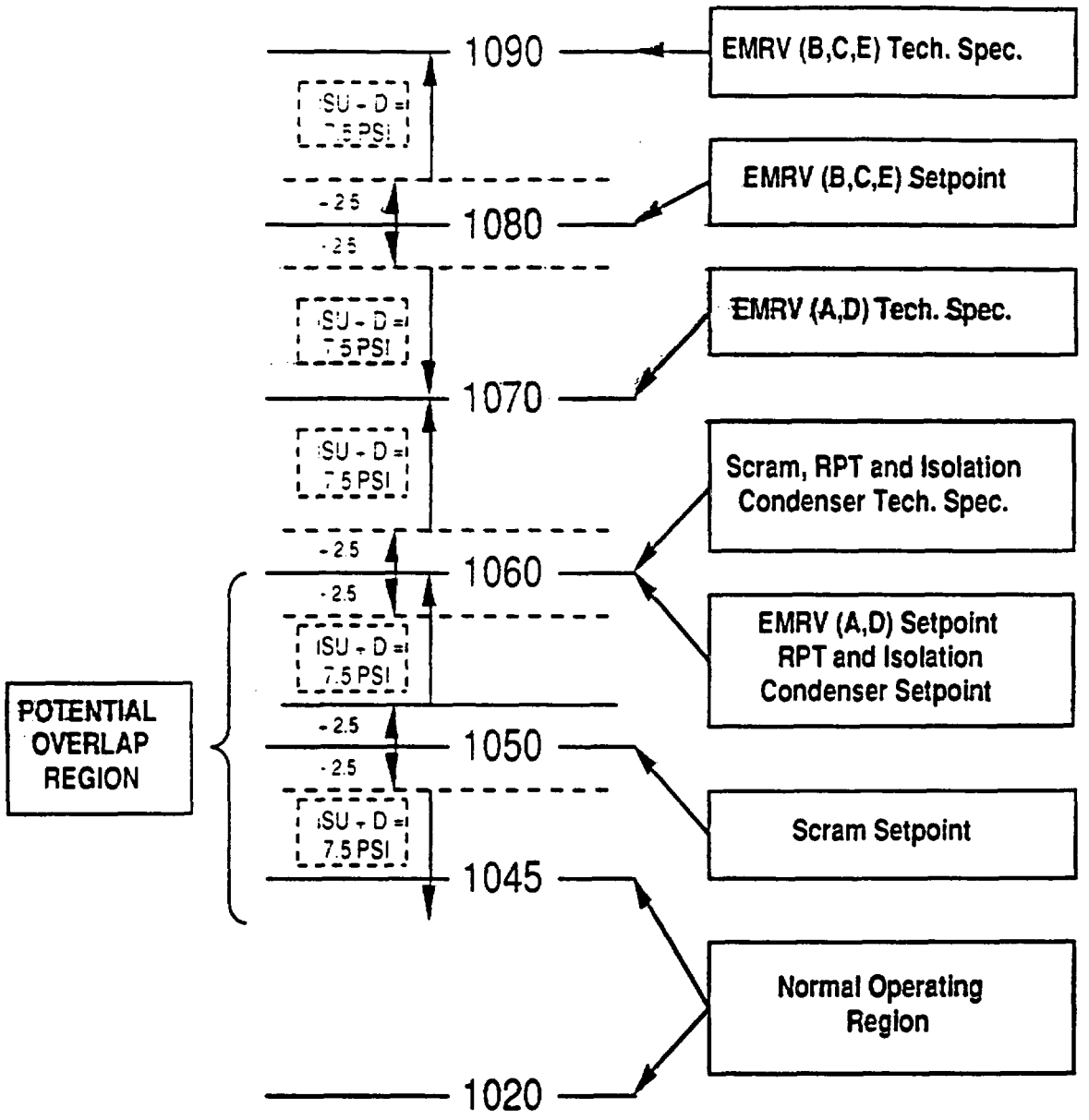
TMI-1 Flux/Flow Setpoints
- Application Overview
- Current Flux/Flow Setpoints
- Proposed Flux/Flow Setpoints
- Code Sequence
- Project Status

Palo Verde Safety Valve Tolerance
- Application Overview
- Project Approach

Oyster Creek High Pressure Setpoints Overview

- High pressure setpoints did not provide sufficient margin for instrument drift and calibration tolerances
- Safety valve setpoint drift resulted in frequent and costly testing of valves

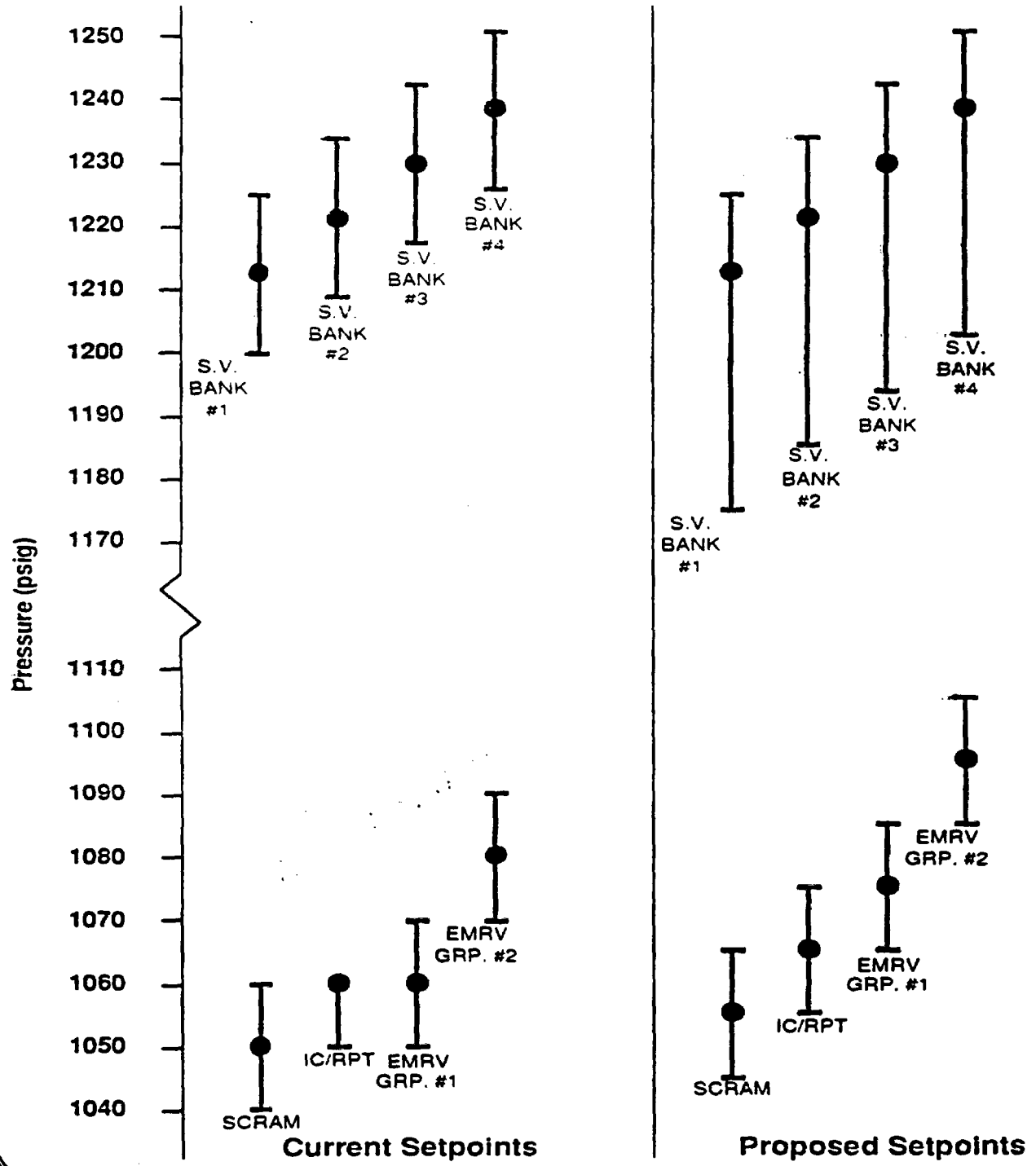
OYSTER CREEK HIGH PRESSURE SETPOINT/ TECHNICAL SPECIFICATION RELATIONSHIPS



ISU = Instrument System Uncertainties
D = Instrument Setpoint Drift

EPRI/NPD

OYSTER CREEK HIGH PRESSURE SETPOINTS



S & R/ Reactor Performance Program

Analysis Considerations

License Basis Requirements

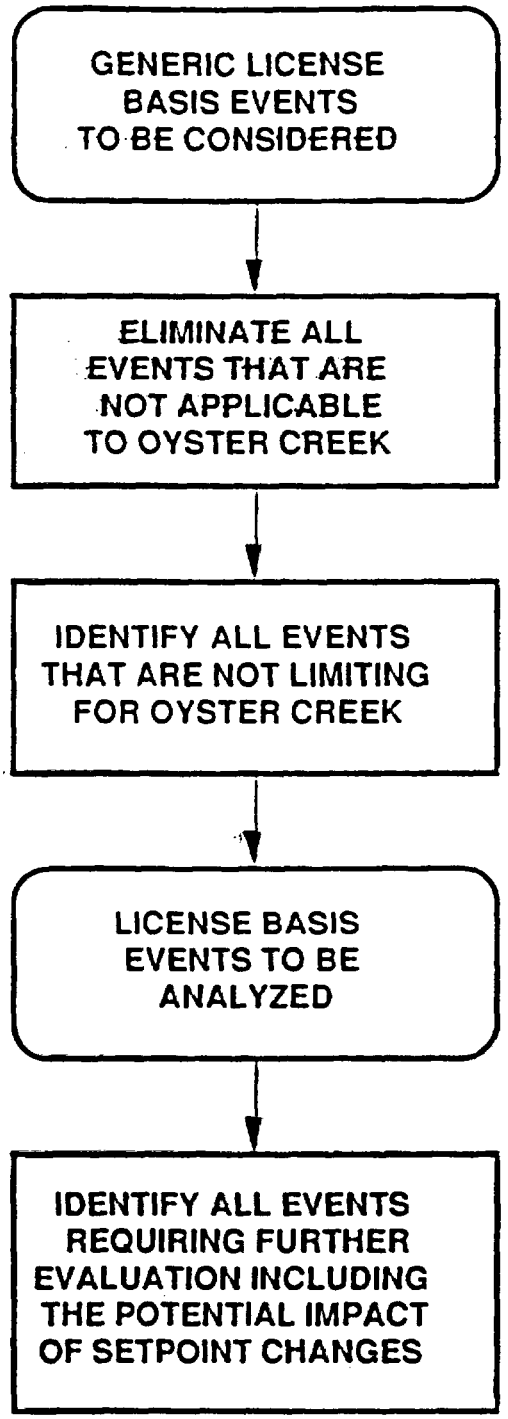
- Evaluation of recirculation pump trip and alternate rod insertion for high pressure setpoints for anticipated transients without scram
- Opening no more than one relief valve without scram
- Analysis to establish minimum critical power ratio operating limit
- Demonstration peak system pressure acceptable for proposed setpoints

Analysis Considerations

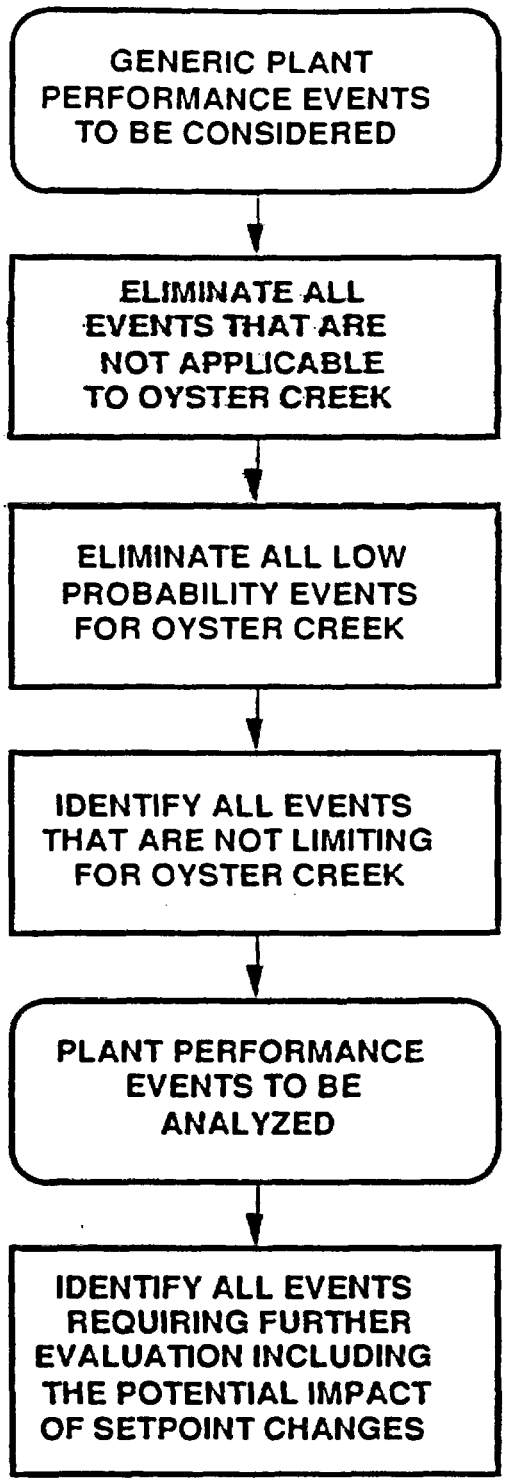
Plant Performance Considerations

- Low probability of relief valve opening prior to isolation condenser initiation
- Low probability of alternate rod insertion prior to isolation condenser initiation
- Low probability or recirculation pump trip prior to scram
- Low probability of relief valve opening prior to scram
- Low probability of a safety valve opening as a result of an anticipated operational occurrence

LICENSE BASIS EVENT SELECTION PROCESS



PLANT PERFORMANCE EVENT SELECTION PROCESS



Assessment of BWR Events that Reach High Pressure Setpoints

Transients	Assessment Plant Performance	Assessment Licse Basis
Inadvertent High Pressure Makeup System Start	Not Applicable	Not Applicable
Pressure Regulator Failure-Closed	Low Probability	Not Limiting
Generator Load Rejection with Bypass	Not Limiting	Not Limiting
Generator Load Rejection without Bypass	Low Probability	Not Limiting
Turbine Trip with Bypass	**Analyze**	Not Limiting
Turbine Trip without Bypass	Not Limiting	**Analyze**
Main Steamline Isolation Valve Closure	**Analyze**	Not Limiting
Loss of Condenser Vacuum	Not Limiting	Not Limiting
Recirculation Pump Trip	Not Limiting	Not Limiting
Recirc. Flow Controller Failure-Decreasing Flow	Not Limiting	Not Limiting
Inadvertent Opening of a Safety or Relief Valve	Not Limiting	Not Limiting
Pressure Regulator Failure-Open	Not Limiting	Not Limiting
Loss of Feedwater Flow	Not Limiting	Not Limiting
Loss of Offsite Power	Not Limiting	Not Limiting
Loss of Shutdown Cooling	Not Limiting	Not Limiting
Feedwater Controller Failure	Low Probability	**Analyze**
Accidents		
Control Rod Drop Accident	Low Probability	Not Limiting
Main Steamline Break	Low Probability	Not Limiting
Small Loss of Coolant Accident	Low Probability	Not Limiting
Recirculation Pump Seizure	Low Probability	Not Limiting
Special Events		
Shutdown from Outside the Control Room	Low Probability	Not Limiting
Overpressure Protection Analysis	Low Probability	**Analyze**
Anticipated Transients without Scram	Low Probability	Not Limiting

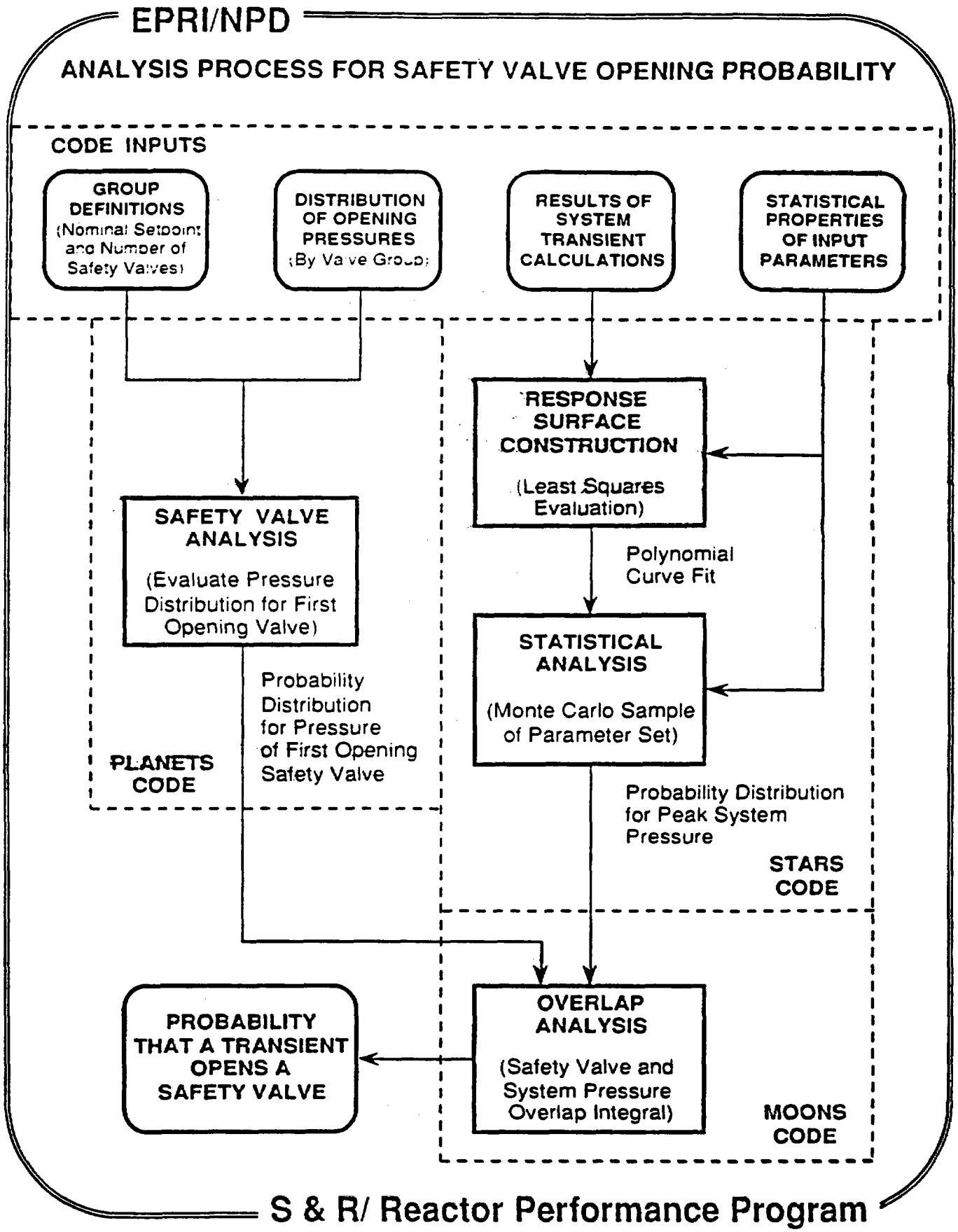
Analysis Results

License Basis (License Basis Methodology)

- Recirculation Pump Trip and Alternate Rod Insertion Performance Acceptable
- No Overlap of Relief Valve and Scram Setpoints
- Current Minimum Critical Power Ratio Operating Limit Acceptable
- Peak System Pressure Below Safety Limit

Plant Performance (PLANETS Analysis)

- Probability of Relief Valve Initiation Prior to Isolation Condenser < 7X10-4 per Event
- Probability of ARI Initiation Prior to Scram < 10-6 per Event
- Probability of Recirculation Pump Trip Prior to Scram < 5X10-2 per Event
- Probability of Isolation Condenser Initiation Prior to Scram < 5X10-2 per Event
- Demonstration of Acceptably Low Probability of Safety Valve Opening for Anticipated Operational Occurrences Most Difficult Task


EPRI/NPD
ANALYSIS PROCESS FOR SAFETY VALVE OPENING PROBABILITY
CODE INPUTS
GROUP DEFINITIONS (Nominal Setpoint and Number of Safety Valves)
DISTRIBUTION OF OPENING PRESSURES (By Valve Group)
RESULTS OF SYSTEM TRANSIENT CALCULATIONS
STATISTICAL PROPERTIES OF INPUT PARAMETERS
RESPONSE SURFACE CONSTRUCTION (Least Squares Evaluation)
Polynomial Curve Fit
SAFETY VALVE ANALYSIS (Evaluate Pressure Distribution for First Opening Valve)
STATISTICAL ANALYSIS (Monte Carlo Sample of Parameter Set)
Probability Distribution for Pressure of First Opening Safety Valve
PLANETS CODE
Probability Distribution for Peak System Pressure
STARS CODE
PROBABILITY THAT A TRANSIENT OPENS A SAFETY VALVE
OVERLAP ANALYSIS (Safety Valve and System Pressure Overlap Integral)
MOONS CODE
S & R/ Reactor Performance Program

Conclusions

- All License Basis Requirements Were Satisfied for the Proposed Changes to the High Pressure Setpoints for the Demonstration.
- No Other Changes to Any Other Current Plant Operating Limits or Technical Specifications Are Required.
- The Plant Performance Requirements Were Satisfied for the Proposed Changes to the High Pressure Setpoints for the Demonstration.
- The EPRI RASP Setpoint Guidelines Can Be Used in Establishing Complex and Interrelated Setpoints.
- Statistical Combination of Uncertainties Methodology Can Be Used to Reduce the Excessive Conservatism Inherent in the Deterministic Analysis Process.
- The Statistical Combination of Uncertainty Methodology is Relatively Simple to Apply.
- Analysis Capability for Complex Valve Networks Is an Important Contribution Made by this Demonstration.
- The Setpoint Demonstration Program Identified a Number of Important Lessons Learned.
- The Experience Gained Though the Performance of this Demonstration Should Be Invaluable to Other Setpoint Applications.

Vermont Yankee Reload Analysis Overview

- Substantial Economic Benefits for High Energy Reload Core Design
- Planned High Energy Reload Constrained by Minimum Critical Power Ration Operating Limit

Vermont Yankee Reload Analysis Approach and Results

- Reduced Critical Power Ratio Margin Requirements by use of 1-D Kinetics Analysis and EPRI Guidelines
- Scram Speed Determined to be Most Sensitive Analysis Parameter
- Applied EPRI's Statistical Combination of Uncertainties Methodology and Combined Scram Speed and Model Uncertainties to Justify Simplified Analysis Approach
- NRC Approval - October 1989

Diablo Canyon Safety Valve Tolerance Overview

Objectives

- Relaxation of Lift Tolerance on Pressurizer and Steam Generator Safety Valves
- Justify Increase of Tolerance to ±3%

Benefits

- Fewer Safety Valve Surveillance Test Failures
- Reduction in Plant Down Time for Valve Testing and Maintenance
- Increase in Operational Flexibility

Diablo Canyon Safety Valve Tolerance Approach and Results

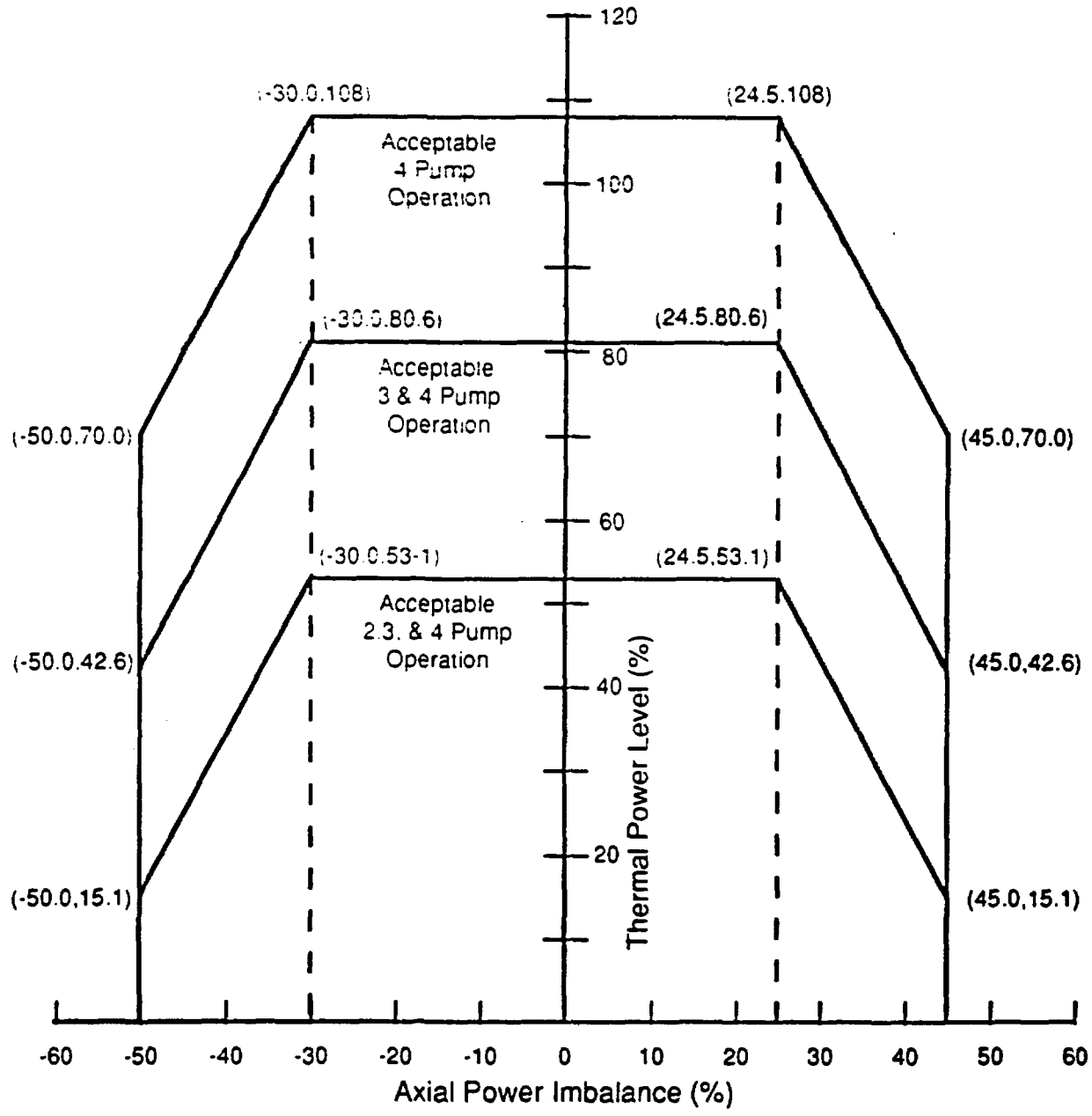
- Applied EPRI Reactor Analysis Support Package Guidelines to Perform Analyses for Setpoint Modification
- Used PLANETS Code to Determine Safety Valve Opening Probability
- Analyses Demonstrated that Vessel Overpressure Protection Requirements Satisfied for Increased Safety Valve Setpoint Tolerance
- Statistical Analysis Demonstrated that Increased Tolerance Covered by Previous Conservative Analyses

TM-1 Flux/Flow Setpoints

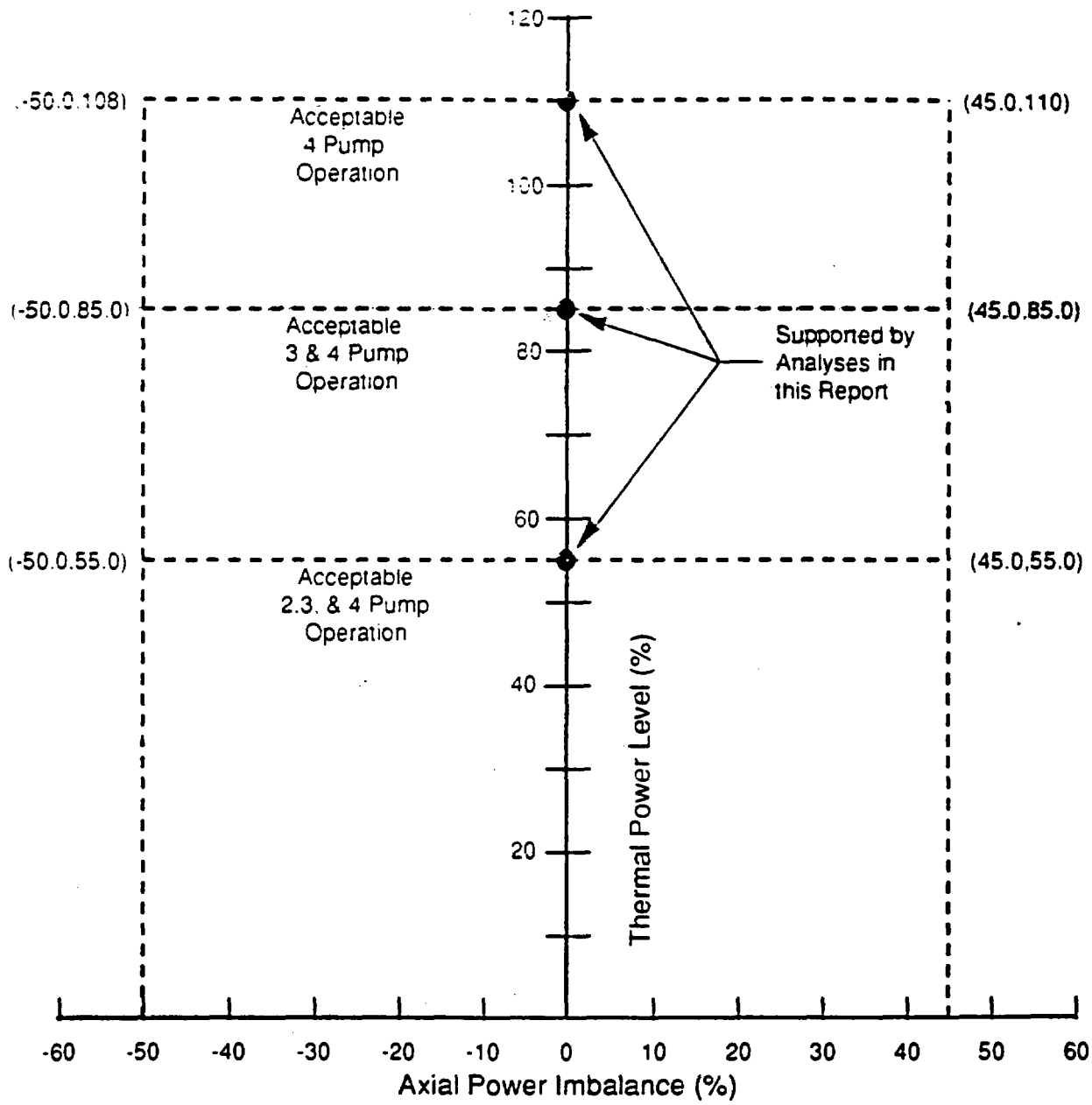
Overview

- Increase in Flux/Flow Operating Margin Desired
- Reduction in Potential for Spurious Scrams

CURRENT FLUX/FLOW SETPOINTS FOR TMI-1

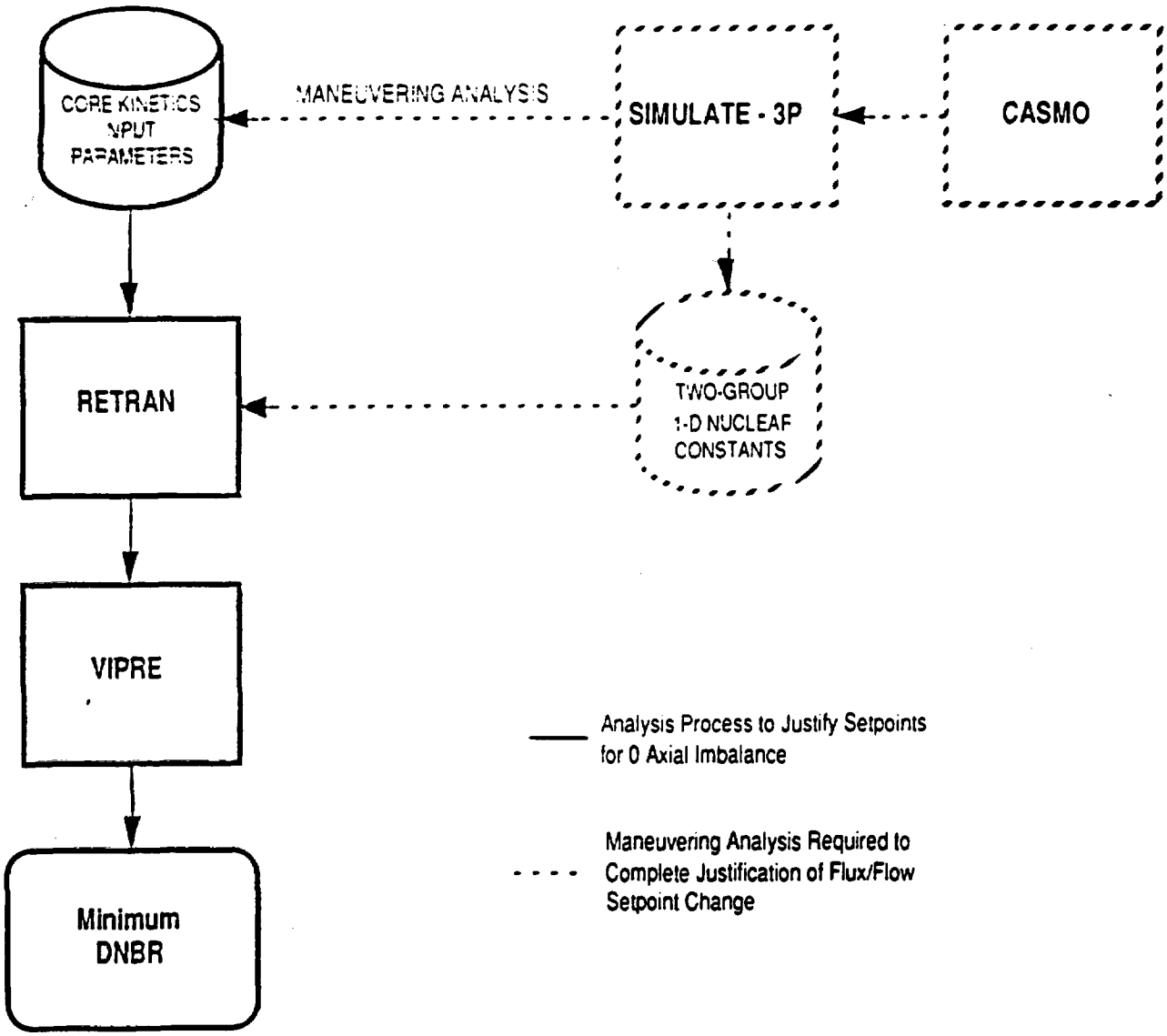


PROPOSED FLUX/FLOW SETPOINTS FOR TMI-1



Requires Maneuvering Analysis for Complete Justification

TMI-1 CODE SEQUENCE FOR FLUX/ FLOW ANALYSES



TMI-1 Flux/Flow Setpoints

Project Status

Phase 1 - Complete

- Safety Analysis for 0 Axial Imbalance Acceptable
- Documentation Being Prepared

Phase 2 - Being Initiated

- Maneuvering Analysis Necessary

Palo Verde Safety Valve Tolerance

Overview

- Test Data on Pressurizer and Main Steam Safety Valves Indicate Drift
- Technical Specification Tolerance of ±1% too Restrictive
- Tolerance Relaxation to ±3% Desired
- Requires Demonstration of Adequacy of Overpressure Protection
- Requires Demonstration of Low Probability of Safety Valve Opening Prior to Scram

Palo Verde Safety Valve Tolerance

Project Approach

- Use EPRI Statistical Methodology to Justify Larger Uncertainty in Valve Opening Pressure
- Use PLANETS Code to Simulate the Behavior of the Valve Network
- Quantify the Overlap in the Probability Distributions of the Safety Valve Opening and Scram Occurrence
- Use License Basis Analysis Methodology to Justify Acceptable Overpressure Protection

ATTACHMENT 2

PHILADELPHIA
SERVICE
ELECTRIC COMPANY

LIMERICK GENERATING STATION

TRIPS

TRIP

Transient Response Implementation Plan

LIMERICK AND PEACH BOTTOM

TRIP

EQUALS

EOP

LGS TRIPS

- T-100 Series Procedures
- T-200 Series Procedures
- TRIP Bases
- PSTG
- PSWG
- V & V

LGS TRIPS
T-100 PROCEDURES

- Flowchart format
- Developed from Revision 4 of NRC approved BWROG EPGS

LGS TRIPS
T-200 PROCEDURES

- Prose format
- Supplement T-100 procedures

LGS TRIPS
TRIP BASES

- Technical justification for T-100 procedural steps
- One Bases per flowchart

LGS TRIPS

PLANT SPECIFIC TECHNICAL GUIDELINE

LGS TRIPS
PSTG

- Used to develop site specific EOPs from generic EPGs

LGS TRIPS
PSTG

- LGS PSTG
- EPG ▷ LGS PSTG
- LGS PSTG ▷ TRIPS
- LGS PSTG Variable List

LIMERICK PLANT SPECIFIC TECHNICAL GUIDELINES

SP/L-1 Maintain suppression pool water level between 24.25 ft. and 22 ft. Refer to SAMPLING PROCEDURE prior to discharging water.

LIMERICK GENERATING STATION
EPG TO PSTG CONVERSION FORM

BWR-OG-EPG STEP:

SP/L-1 Maintain suppression pool water level between {12 ft. 6 in. (maximum suppression pool water level LCO)} and {12 ft 2 in. (minimum suppression pool water level LCO)}; refer to {sampling procedure} prior to discharging water; {suppression pool makeup may be augmented by SPMS}.

LGS-PSTG STEP:

SP/L-1 Maintain suppression pool water level between 24.25 ft. and 22 ft. Refer to SAMPLING PROCEDURE prior to discharging water.

DIFFERENCES:

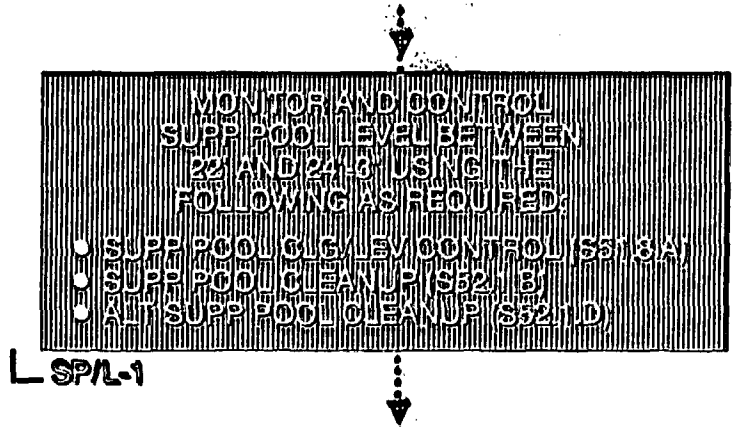
- The maximum suppression pool water LCO at LGS is 24.25 feet.
- The minimum suppression pool water level LCO at LGS is 22.0 feet.
- The reference to SPMS (Suppression Pool Management System) is not included as SPMS is not an installed system at LGS.

LIMERICK GENERATING STATION
PSTG TO TRIP CONVERSION FORM

LGS-PSTG STEP:

SP/L-1 Maintain suppression pool water level between 24.25 ft. and 22 ft. Refer to SAMPLING PROCEDURE prior to discharging water.

LGS-TRIP STEP:



DIFFERENCES:

The letdown procedures referred to in TRIP step SP/L-1 refer to water quality concerns as appropriate.

F998 JIMA11.P 10826 B196-XE

LGS TRIPS

PLANT SPECIFIC WRITER'S GUIDE

LGS TRIPS
PSWG

- Detailed instruction on how to prepare T-100 & T-200 procedures

- Ensures procedures are:
  - Complete
  - Accurate
  - Convenient
  - Readable
  - User friendly
  - Acceptable to personnel
  - Human factored

LGS TRIPS

VERIFICATION

&

VALIDATION

LGS TRIPS
V & V

- Ensures proper procedure preparation
- Ensures usability:
  - Technically
  - By human factoring
- Performed by a multi-disciplined team

LGS TRIPS
TRIP EQUIPMENT AND LABEL PRESTAGING

- Equipment and Tool Cabinets
- Purple Tagging/Labeling
- Surveillance Tests for availability and operability

ATTACHMENT 3

EDSFI BRIEFING TO ACRS

AUGUST 29, 1991

- SCOPE OF EDSFI
  - ADEQUACY OF THE ELECTRICAL DISTRIBUTION SYSTEM
  - EVALUATE ENGINEERING AND TECHNICAL SUPPORT CAPABILITY
- TEMPORARY INSTRUCTION IN USE BY REGIONS
- TO ACHIEVE CONSISTENCY OF TI IMPLEMENTATION
  - FIVE PILOT INSPECTIONS CONDUCTED BY DRIS
  - TRAINING PROVIDED AT TTC
  - DRIS PARTICIPATION IN REGIONAL EDSFIs
- STATUS OF TI IMPLEMENTATION
  - 24 INSPECTIONS COMPLETED AS OF AUGUST 30, 1991
  - TARGET COMPLETION OF TI - JANUARY 1993

EDSFI BRIEFING TO ACRS
AUGUST 29, 1991

- REGIONAL COUNTERPART MEETINGS HELD APRIL 24 AND SEPTEMBER 5, 1991

- REGULATORY INFORMATION CONFERENCE EDSFI SESSION MAY 7

- INFORMATION NOTICE 91-29 ISSUED
  - ADDRESSES INITIAL FINDINGS FROM EDSFIs
    - DEGRADED GRID VOLTAGE RELAY SETPOINTS
    - INADEQUATE SETPOINTS
    - INADEQUATE TESTING OF CIRCUIT BREAKERS

ELECTRICAL DISTRIBUTION FUNCTIONAL INSP

RECURRING PROBLEM AREAS

- DESIGN DEFICIENCIES
- PROCEDURAL WEAKNESSES
- CONFIGURATION MANAGEMENT

ELECTRICAL DISTRIBUTION FUNCTIONAL INSP

RECURRING PROBLEM AREAS

- ELECTRICAL DESIGN DEFICIENCIES
  - FAULT PROTECTION
    - UNDERSIZED AC & DC CIRCUIT BREAKERS
    - LACK OF SHORT CIRCUIT ANALYSES FOR DC CIRCUITS
  - VOLTAGE REGULATION
    - GRID UNDER VOLTAGE CONDITIONS
    - DIESEL GENERATOR TRANSIENT LOADING CONDITIONS
    - LOAD SEQUENCING
  - BUS TRANSFER
  - BREAKER COORDINATION
  - RELAY SETTINGS
  - PHYSICAL SEPARATION
  - RACEWAY FILL

ELECTRICAL DISTRIBUTION FUNCTIONAL INSP

RECURRING PROBLEM AREAS

- MECHANICAL DESIGN DEFICIENCIES
  - TEMPERATURE RELATED PROBLEMS
    - CONTROL OF MINIMUM BATTERY ROOM TEMP
    - EQUIPMENT QUALIFICATION
      - MINIMUM COOLING WATER TEMPERATURE
      - MAXIMUM AMBIENT TEMPERATURE
  - DIESEL GENERATOR FUEL OIL SYSTEM
    - CAPACITY OF FUEL OIL STORAGE TANK
    - FUEL SUPPLY AVAILABLE IN DAY TANK
    - TORNADO PROTECTION
  - SEISMIC QUALIFICATION
    - FUEL OIL TRANSFER SYSTEM
    - DG LUBE OIL AND JACKET WATER HEATERS
    - FIRE PROTECTION SYSTEM IN DG ROOM
    - RELIEF VALVES ON DG AIR START ACCUMULATORS

ELECTRICAL DISTRIBUTION FUNCTIONAL INSP

RECURRING PROBLEM AREAS

- PROCEDURAL INADEQUACIES
  - LACK OF ACCEPTANCE CRITERIA
    - MAINTENANCE PROCEDURES
    - BATTERY TESTING
  - ERRORS OR OMISSIONS
    - INCONSISTENT WITH VENDOR RECOMMENDATIONS
    - EOP ELECTRICAL LOAD TABULATION
  - APPROPRIATE PROCEDURES DO NOT EXIST
    - FUSE CONTROL
    - MAINTENANCE
      RELAY CALIBRATION
  - FAILURE TO IMPLEMENT EXISTING PROCEDURES

ELECTRICAL DISTRIBUTION FUNCTIONAL INSP

RECURRING PROBLEM AREAS

- CONFIGURATION MANAGEMENT

  - LACK OF DESIGN BASES
    - BREAKER/RELAY SETTINGS
    - TRANSIENT ANALYSIS OF VOLTAGE PROFILE DURING LOAD SEQUENCING
  - PLANT DOES NOT CONFORM TO DESIGN DOCUMENTS
    - BREAKER/RELAY SETTINGS

EDSFI BRIEFING TO ACRS
AUGUST 29,1991

- TWO LEVELS OF UNDERVOLTAGE RELAYS
- PSB-1 REQUIREMENTS
  - THE SELECTION OF UNDERVOLTAGE AND TIME DELAY SETPOINTS SHALL BE DETERMINED FROM AN ANALYSIS OF THE VOLTAGE REQUIREMENTS OF THE CLASS 1E LOADS AT ALL ONSITE SYSTEM DISTRIBUTION LEVELS
- GRID STABILITY
- 7 OF 21 PLANTS INSPECTED HAVE HAD INADEQUATE SECOND LEVEL DEGRADED GRID RELAY SETPOINTS

EDSFI BRIEFING TO ACRS
AUGUST 29, 1991

- PROGRAM RESULTS
  - LICENSEE'S CONDUCTING THEIR OWN EDSFIs
  - INCREASED ATTENTION TOWARDS EVALUATING AND IMPROVING THE DESIGN BASIS OF THE EDS
  - IMPROVED THE OVERALL FUNCTIONAL CAPABILITY OF THE EDS

EDSFI BRIEFING TO ACRS
AUGUST 29, 1991

- CONCLUSIONS
  - EDSFI FOUND DEFICIENCIES THAT WERE:
    - INTRODUCED BY PLANT MODIFICATIONS
    - PRESENT IN THE ORIGINAL DESIGN
  - INDICATES NEED FOR BETTER
    - ENGINEERING AND TECHNICAL SUPPORT
    - LICENSEE SELF-ASSESSMENT PROGRAMS
    - UNDERSTANDING OF DESIGN BASES
    - AVAILABILITY OF DESIGN DOCUMENTATION