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The PP&L Approach To Risk Management And Risk Assessment (Second Revision)

January 1989

PENNSYLVANIA POWER & LIGHT COMPANY PP&L

THE PP&L APPROACH

TO

RISK MANAGEMENT

AND

RISK ASSESSMENT

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DISCLAIMER

The material presented here represents the current status of work in progress at PP&L as of August 1988 to address resolution of the severe accident issue. Not all of this material has been formally adopted by PP&L management at this time for application to management of risk at Susquehanna. The material should be considered only to represent current technical efforts at PP&L to resolve the issue of adequacy against the occurrence of severe accidents at Susquehanna. The material is only intended to represent an approach being considered for applicability to Susquehanna.

The material contained in this document is intended for information purposes only. PP&L will not be responsible for use of this information for application or reference by any other organization.

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Acknowledgements

The material presented in this document represents the work of several individuals other than the authors. In particular Mike Carr, Chris Boschetti, Jim Harven, Jack Refling, and Shahin Seyedhosseini are responsible for many of the calculations and studies which are the basis for the material presented. The efforts of Paul Wasson, Tim Hill, and Dennis Raudenbusch of the Engineering and Scientific Systems Programming Section have been invaluable in bringing BWRSAR to operational status and in developing the Risk Analysis Program described in this document. It is equally important to note that PP&L management has provided an environment which encourages and expects performance of the work presented here. In the chain of command involved are Mike Detamore, Jerry Stefanko, Chuck Myers, and Harry Keiser.

We have also benefitted greatly from the assistance from the NRC, Tom Walker, and ORNL, Steve Hodge, and their associates in helping us bring the BWRSAR program to full operational status at PP&L and in assisting us in mastering and interpreting its many intricacies. This program has had a major influence on our risk assessment work.

Finally, the credit for preparation of this document must go to our clerical staff, Evelyn Lugo and Lisa Jacoby.

Preface

In early 1988 PP&L volunteered to present a seminar on the PP&L approach to performance of the Susquehanna Individual Plant Evaluation to the Boiling Water Reactor Owners Group. This offer was accepted by A. R. Diederick, Chairman of the Risk Assessment Issues Committee, and through the efforts of Rick Hill of the General Electric Company arrangements were made for an Individual Plant Evaluation Seminar in August 1988.

The intent of this document is to provide a written version of the presentations which have been prepared. Since the topics to be covered are rather complex, we anticipate that retention of the presentation material will be incomplete and temporary at best. The document is intended to preserve the presentations made for future reference.

The document has been prepared during a period of intense activity in the risk analysis and risk management areas at PP&L. Major activities which are in various stages of planning or completion are:

1. completion of a defense-in-depth evaluation based on the initial Susquehanna IPE,

2. incorporation of Revision 4 of the BWROG Emergency Procedure Guidelines into the Susquehanna EOPs,

3. revision 1 of the Susquehanna IPE,

4. a formalization of the PP&L approach to IPE execution and creation of a computer program to perform the analysis,

5. adoption of the BWRSAR program for accident progression analysis,

6. procurement and adoption of the CONTAIN program for analysis of containment performance in severe accidents.

7. recommendations for formalization of the risk management process at PP&L.

8. recommendations for resolution for outstanding severe accident issues for Susquehanna.

This situation has presented us with several problems in the preparation of the presentations. The fundamental issue here has been:

> "Shall we present only information based on what we have actually completed, or
>
> shall we present our latest thinking and methods which have not yet seen full application?"

We have opted for the latter course in preparing this presentation. We have done so because we believe that otherwise much of the broader view of risk assessment and its value which have been developed at PP&L over the past 3 years would be lost.

The implication of this approach is that much of what will be presented is not actually solidly in place at PP&L at this time, and the actual results presented are based on our original IPE work. This will cause some confusion because the original IPE work, for example, was based on MAAP for accident progression analysis and success criteria obtained from the IDCOR contractors. Presently, we are using the BWRSAR program for this purpose and our new success criteria lead to significantly different results. We have not yet developed in-house experience at the use of CONTAIN, but we anticipate significant changes in the success criteria relating to containment performance as well.

We, nevertheless, believe that the risk of confusion must be accepted in return for presenting what we believe is a much more comprehensive view of the risk analysis process. At this time we are reasonably confident that the above activities which are not yet complete will be successful and that our current view of the expected results are fairly accurate.

The following cautions are nevertheless offered to those who choose to study this material.

1. Our current view of the success rate at saving the reactor vessel is much more favorable than presented in the Susquehanna IPE (∿99% vs. 70%).

2. The dispositioning of the final containment status is more complex due to a greater generation of noncondensible gases by accident processes.

3. The PP&L position on wetwell venting has not yet been resolved.

4. We face a potential for taking additional exceptions to the BWROG EPGs, particularly with regard to suppression pool mass addition.

5. We have not yet developed a comprehensive set of success criteria for the containment. This must await our installation of the CONTAIN program at PP&L.

6. In our IPE revision there will be a great proliferation of support states. While we do not expect any radical change in our results because of this, we cannot be certain of this.

In spite of these rather major cautions we believe the material in this presentation is worthwhile. The material which we feel has enduring value regardless of the cautions above are:

1. the concept of and approach to risk management,

2. the definition of the role played by Emergency Operating Procedures in the risk management process,

3. the treatment of human response in the control room,

4. the conventional PRA conservatisms and their influence, and

5. the PP&L approach to application of the support state method of risk analysis.

We realize that the opinions and value judgements presented here are uniquely applicable only to PP&L and that indiscriminant adoption of these views by another utility may be inappropriate. Nevertheless, we believe there is much here that could be profitably exploited by others should they choose to do so.

Preface to the Revision

Preparation of the original presentation material used for the IPE Workshop in August 1988 suffered from lack of time for proper editing and incorporation of related useful material. Upon completion of the workshop we have had the time to make the necessary repairs to bring the document to a more acceptable level and quality.

In this process, we have not knowingly changed the meaning or intent of any of the original material. We have attempted to remove most of the objectionable grammatic and spelling errors and to improve the clarity of some (but probably not all) of the discussion of the viewgraph material. We have added eight Appendices in order to remove some material from the main body of the document and also to provide some additional material to demonstrate the potential sources of much of the necessary data for analysis of transients.

Preface to the Second Revision

The second revision of this document incorporates new information as follows:

1. modifications to the equations for calculating support systems failure combinations to account for the influence of demand failures,

2. a new Appendix 9 to discuss the assumption that i outages of length T_i can be treated as a single outage of length equal to the total of the T_i, and

3. a new Appendix 10 which outlines the approach to be taken to describe external events such as flooding, fires, and seismic events.

As in the first revision, numerous corrections in spelling and grammar have also been made.

At this time we have made considerable progress on the computer program for calculation of accident sequence frequencies discussed in this document. The calculation part of the program is now operational as a FORTRAN program on an IBM PS/2 machine. Our current efforts are now directed toward the creation of input and output formats and options for the calculation which will give the user some reasonable level of control over the titles and labels for the input and output data. Upon completion of the input/output routines, the program will be ready for trial runs and serious use. The next objective beyond this capability will be the creation of a Monte Carlo driver routine for the calculation which will permit direct propagation of a wide variety of uncertainties through the calculation.

INTRODUCTION

INTRODUCTION(1)

- o **PURPOSE**
  - ACQUAINT OTHER BWR UTILITIES WITH OUR IPE FINDINGS
  - DESCRIBE THE IPE PROCESS
  - DESCRIBE HOW THE IPE METHOD CAN BE EXTENDED TO INCLUDE CONTAINMENT
  - DESCRIBE CRITICAL MODEL ASSUMPTIONS AND THEIR IMPACT
- o **APPROACH**
  - SUSQUEHANNA IS A TWO UNIT BWR4 USING THE MARK II CONTAINMENT
  - OUR APPROACH TO RESOLUTION OF SEVERE ACCIDENT ISSUES HAS SHAPED OUR APPROACH TO RISK ASSESSMENT
  - OUR PRESENTATION WILL COVER OUR APPROACH TO RISK MANAGEMENT
- o **RESERVATIONS**
  - WE DO NOT EXPECT OTHER UTILITIES TO ACCEPT OUR APPROACH TO RISK MANAGEMENT
  - WHILE OUR APPROACH HAS GENERAL APPLICABILITY FOR THE BWR, THE MATERIAL PRESENTED WILL NOT COVER ALL ASPECTS OF BWR PLANTS (BWR2, MARK III)
  - THE PRESENTATION WILL DESCRIBE THE METHODS BEING DEVELOPED FOR THE REVISION TO THE SUSQUEHANNA IPE
  - PP&L HAS USED ACCIDENT ANALYSIS CODES NOT CURRENTLY USED BY OTHER UTILITIES

Introduction(1)

The fundamental purpose of the PP&L presentations on the IPE is to familiarize other BWR utilities with the important elements of the IPE and to demonstrate the nature of the information needed to perform the IPE. In addition, PP&L has discovered that calculations of BWR plant performance in response to a severe accident is critically dependent on a moderate number of modeling assumptions in the analysis. We wish to advise others of these sensitivities.

The NRC has clearly indicated that the IDCOR IPE does not go far enough into the analysis of risk and that they expect the equivalent of a level 2 PRA treatment. PP&L believes that extension of the IPE to disposition all accident sequences to the final spectrum of plant damage states is a relatively simple extension of the methodology. We also believe that it is important to perform this extended analysis with care that conservative simplifying assumptions do not result in an improperly negative representation of containment performance.

The presentations will necessarily be made from the perspective of a BWR4, Mark II plant. To date, all PP&L analysis has been for a single unit of Susquehanna. While we believe our analysis approach is applicable to the BWR plant in general, plants having such features as no jet pumps, isolation condensers, other ECCS networks, motor driven feed pumps, or Mark III containments, may find that the Susquehanna analysis omits features of importance in a specific plant or least requires some modeling changes for a proper representation. We cannot address these issues in our presentations.

An important consideration in the PP&L approach to risk analysis and performance of the IPE, is the PP&L approach to risk management and severe accident management. Many of the features of the analysis may seem to be unnecessary or extreme unless the PP&L approach to severe accident issues is understood. In order to avoid logical problems we will present our approach to severe accident issues in order to provide the rationale for our approach to risk analysis.

While we clearly believe that our approach is the correct approach for PP&L, we are not so unrealistic as to expect any general level of agreement with our approach by other utilities. Nevertheless, there are some real and practical advantages to the PP&L approach to IPE implementation which we believe will benefit any BWR utility which chooses to adopt it.

The very lowest level of benefit that we would expect would be that other utilities will be able to avoid the very negative assessments of BWR containment performance that conventional PRA methods yield. These negative assessments are not a consequence of the approach to risk analysis, but are a consequence of simplifying assumptions, or sometimes deficiencies in existing Emergency Operating Procedures.

INTRODUCTION(2)

MOTIVATION

- o TO DEMONSTRATE THAT THE CONVENTIONAL VIEW OF BWR RISK IS IMPROPERLY NEGATIVE.
- o TO DEMONSTRATE THAT THE ROLE OF THE OPERATOR IS NOT PROPERLY UNDERSTOOD.
- o TO DEMONSTRATE THAT THE CAPABILITIES OF THE BWR PLANT ARE MISREPRESENTED.
- o TO PERSUADE BWR UTILITIES TO BETTER UNDERSTAND THEIR VULNERABILITIES AND TO EXPLOIT THEIR OPPORTUNITIES FOR RISK REDUCTION.

EXPECTATIONS

- o TO ENCOURAGE A CLOSER AND MORE DEFINITIVE LOOK AT EOPs.
- o TO ENCOURAGE AVOIDANCE OF CONVENTIONAL PRA CONSERVATISMS WHICH MISREPRESENT THE BWR.
- o TO ENCOURAGE INCREASED OPERATOR TRAINING ON EOPs.

HOPES

- o TO ENCOURAGE UTILITY PROGRAMS TO MONITOR PERFORMANCE.
- o TO ENCOURAGE OTHER UTILITIES TO ADOPT SOME DEGREE OF RISK MANAGEMENT.

Introduction(2)

The PP&L motivation, expectations, and hopes in preparing the material to be presented here is outlined in the viewgraph. We are aware that our approach to risk analysis differs in many important aspects from previous PRA work for the BWR plant. Further, we are aware that we have used many new analysis and computer models which differ greatly from those in prior use.

We expect these differences to make adoption of much of what we have done very difficult for many BWR utilities. Nevertheless, we believe there are many valuable lessons to be learned from what we have done which can not only improve the calculated performance of other BWR plants in response to severe accident situations, but can also result in a real reduction in the risk of severe plant damage and severe off-site consequences for those plants which choose to heed these lessons and respond. These improvements need not involve total acceptance of the PP&L approach.

PRESENTATION OUTLINE

- THE PP&L APPROACH TO RISK ASSESSMENT
- AN OVERVIEW OF THE PP&L RISK ANALYSIS PROCESS
- INITIATING EVENTS
- EVENT TREE CONSTRUCTION AND SUCCESS CRITERIA
- ACCIDENT SEQUENCE QUANTIFICATION
- ORGANIZATION OF RESULTS
- AN IPE APPLICATION
- HUMAN ERROR IN RISK ASSESSMENT
- CONVENTIONAL PRA CONSERVATISMS
- CATEGORIES OF UNCERTAINTY
- SUMMARY

Presentation Outline

The view graph presents the major topics which will be presented.

THE PP&L APPROACH
TO
RISK ASSESSMENT

FRONT-END AND BACK-END ANALYSIS

FRONT END ANALYSIS.

THE ANALYSIS USED TO DETERMINE CORE DAMAGE FREQUENCY. AS APPLIED BY PP&L THIS ANALYSIS IS EXTENDED TO DEFINE ALL FORMS OF PLANT DAMAGE, SLIGHT TO CATASTROPHIC, OUT TO STABLE OR UNCONTROLLED PLANT CONDITIONS.

BACK END ANALYSIS.

THE ANALYSIS OF THE SPECIFIC NATURE OF LOSS OF CONTAINMENT INTEGRITY (VENTING AND TRIVIAL THROUGH CATASTROPHIC), THE DETAILS OF FISSION PRODUCT TRANSPORT PROCESSES, AND THE NATURE OF THE OFF-SITE CONSEQUENCES OF THE FISSION PRODUCT RELEASE.

Front-End and Back-End Analysis

In probabilistic risk assessment the front-end analysis is generally considered to be only that part of the analysis which determines that plant damage, usually core damage, will occur. This portion of the analysis generally involved consideration of equipment performance and operator performance that has a relatively low level of uncertainty associated with it. Equipment unavailability is reasonably well known, transient codes have been well benchmarked (except for ATWS response), and human performance is imbeeded in the data for initiating events and equipment unavailability. PP&L has extended the front-end analysis to determine the extent of core damage, the status of the reactor vessel, the release of core debris to the drywell floor and the containment status.

This extension introduces new contributions to uncertainty which are considerably greater. These involve:

1. the core damage progression phenomena,

2. the reactor vessel failure phenomena,

3. the core debris interaction with the containment components, and

4. the containment failure mode and magnitude.

While these uncertainties are much greater, the magnitude of their contribution to overall plant damage frequency should be greatly reduced as the extent of the damage increases. This characteristic permits one to control and limit the impact of large uncertainties on the nature of the profile of plant damage derived by a probabilistic risk assessment. This extension, which avoids the even more uncertain back-end analysis phenomena, can be readily accomplished with the IPE methodology and can provide fine resolution of the potential range of plant damage states while minimizing the obscuring influence of the large uncertainties always associated with the back-end analysis.

This permits attention to be focussed on equipment and operator performance in order to optimize Emergency Operating Procedures. This process also permits evaluation of the entire range of plant conditions to which EOPs apply without an artificial interface in the analysis process.

DEFENSE-IN-DEPTH CRITERIA
(FREQUENCY AND EQUIPMENT)

- o ACCIDENT SEQUENCES HAVING HIGH CALCULATED FREQUENCIES ARE NOT ACCEPTABLE.

- o ACCIDENT SEQUENCES HAVING THE LOW CALCULATED FREQUENCIES MUST ALSO HAVE DEFENSE-IN-DEPTH BOTH IN THE FORM OF EQUIPMENT AND PROCEDURES. DEFENSE-IN-DEPTH IS DEFINED FOR EQUIPMENT AND PROCEDURES AS FOLLOWS:

- o EQUIPMENT

  - CORE OR CONTAINMENT DAMAGE SHALL NOT OCCUR WITHOUT MULTIPLE FAILURES OF REDUNDANT OR DIVERSE EQUIPMENT.

  - VESSEL FAILURE SHALL NOT OCCUR FOLLOWING CORE DAMAGE UNLESS ADDITIONAL INDEPENDENT EQUIPMENT FAILURES OCCUR.

  - CONTAINMENT FAILURE SHALL NOT OCCUR FOLLOWING CORE DAMAGE UNLESS ADDITIONAL INDEPENDENT EQUIPMENT FAILURES OCCUR.

  - CONTAINMENT FAILURE SHALL NOT OCCUR FOLLOWING VESSEL FAILURE UNLESS ADDITIONAL INDEPENDENT EQUIPMENT FAILURES OCCUR.

Defense-in-Depth Criteria

(Frequency and Equipment)

PP&L does not consider a low calculated frequency to be a sufficient indication of adequacy of our defenses against a severe accident. We attempt to insure that we have defense in depth against the occurrence of plant damage. If we can demonstrate defense in depth against all Susquehanna accident sequences, we expect a very low calculated frequency of damage from that sequence.

We impose the requirement for defense in depth on the basis that our estimates of initiator frequency, equipment failure rates, and, most important, our understanding of commonalities and interactions may be deficient. Use of the defense in depth concept in combination with symptom based procedures and high quality operator training in their use gives us assurance that the full facilities of the plant will be utilized to avoid or minimize damage regardless of the initiator and the coincident combination of equipment failures.

For equipment we impose a requirement for three stages of equipment protection. These three stages of protection are intended to make the likelihood of a given level of damage less as the severity of the damage level increases. The third and fourth criteria are both directed at assuring containment integrity in the event of core damage. At this time the ability to meet the fourth criterion is in dispute. IDCOR models for reactor vessel failure and core debris behavior indicate a high degree of containment protection is possible while the models of some NRC contractors, indicate a very low degree of success in containment protection. PP&L is actively pursuing this issue at present.

DEFENSE-IN-DEPTH CRITERIA

(PROCEDURES AND INSTRUMENTATION)

- PROCEDURAL

  - NO PROCEDURE SHALL HAVE ADVERSE CONSEQUENCES IN THE CASE OF ADDITIONAL EQUIPMENT FAILURES BEYOND THOSE OCCURRING INITIALLY.
  - THE NECESSARY ANTICIPATORY ACTIONS SHALL BE PERFORMED TO AVOID LOSS OF ADDITIONAL EQUIPMENT BUT SHALL NOT DEGRADE THE EXISTING SITUATION.
  - THE NECESSARY ANTICIPATORY ACTIONS SHALL BE PERFORMED TO PERMIT SUCCESSFUL RESPONSE TO POTENTIAL ADDITIONAL FAILURES, BUT, SHALL NOT DEGRADE THE EXISTING SITUATION.

- INTERFACE

  THE NATURE AND TIMING OF INFORMATION TO THE OPERATOR SHALL BE SUFFICIENT TO ASSURE TIMELY EXECUTION OF ALL APPROPRIATE PROCEDURAL STEPS.

Defense-in-Depth Criteria

(Procedures and Instrumentation)

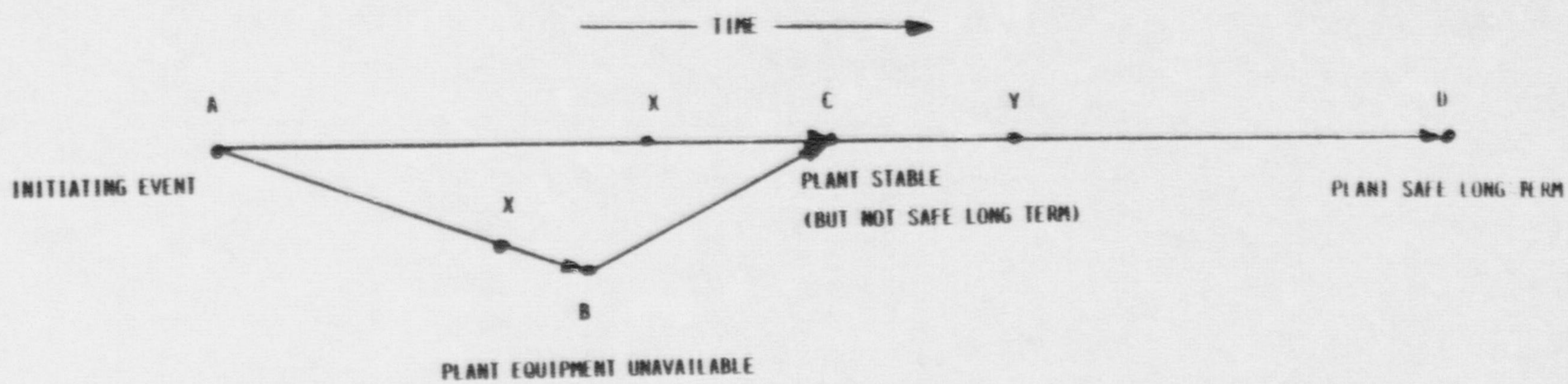
In the BWR plant the close coupling of systems which provide adequate core cooling and those which provide containment heat removal requires direct consideration in the preparation of Emergency Operating Procedures (EOPs). In addition the great redundancy in the means for providing cooling water to the core also require explicit consideration if plant capability is to be fully exploited in reducing the likelihood of core damage.

Three procedural criteria are stated which minimize the likelihood of damage caused by actions taken or additional loss of equipment due to actions taken or not taken. These criteria were developed in late 1981 in the process of maximizing the coping time for Susquehanna in the event of Station Blackout. We have since found these criteria to have a dramatic impact on reducing calculated frequency of plant damage for other initiators as well.

The EOPs are tested against these criteria by examining every accident sequence for Susquehanna against them. While this is a tedious process, it does provide a very thorough check of the adequacy of the EOPs.

The use of symptom based procedures imposed a heavy burden on the plant instrumentation. For this reason it is important to also investigate the actual instrumentation which will be available in the course of an accident sequence to assure that the operators will be prompted to take the necessary actions to protect the plant. Station Blackout has been found to be the most severe event in this regard as a result of the loss of AC power. The relatively slow nature of the Station Blackout transient makes this deficiency less severe by allowing time for appropriate compensating actions.

POTENTIAL ACCIDENT TRAJECTORIES



- AB AND AC ARE GOVERNED BY EOPs.
- EMERGENCY PLAN ACTIONS MAY BE INITIATED AT X.
- BC REPRESENTS COPING ACTIONS INITIATED FROM OUTSIDE THE PLANT. EOPs DO NOT APPLY.
- CD REPRESENTS RECOVERY ACTIONS. EOPs DO NOT APPLY.
- CY REPRESENTS ACHIEVEMENT OF COLD SHUTDOWN. THIS IS NOT POSSIBLE FOR ALL ACCIDENT SEQUENCES.
- XB AND XC REPRESENT THE PERIOD BEFORE PLANT STABILIZATION DURING WHICH EMERGENCY PROCEDURES ARE EXECUTED. THE SHIFT SUPERVISOR IS RESPONSIBLE FOR THESE UNTIL ACTIVATION OF THE TSC.

Potential Accident Trajectories

There appears to be much confusion over the issue of severe accident management and what objectives should be set for it. The diagram on the view graph is intended to define the various phases which can occur in the accident sequences for a plant.

The limits on the definition for Severe Accident Management and EOP applicability are presented in the form of a time line for potential accident sequences in the view graph. The EOPs govern actions taken during those portions of potential accident sequences labeled AB and AC. In the case of the segment AC stabilization and control of the plant are eventually re-established by control room actions, although in some cases with varying degrees of plant damage up to, but not including, the point of containment failure. Venting could take place during the segment AC in order to avoid uncontrolled failure of the containment. Upon reaching point C, a period begins during which recovery actions are initiated. These are actions which would re-establish long term safe conditions in the plant. In a sequence where little or no plant damage is sustained recovery could involve refurbishing as needed for continued operation of the plant. In cases, such as for TMI-2, it could involve clean up of fission products and removal of fuel in preparation for decommissioning of the plant. In all cases, the EOPs completely govern the time segment AC, but do not apply over the time period, CD. There are currently no formal programs or procedures defined for the segment CD other than the Emergency Plan. That plan, however, primarily addresses communications, evacuation, and monitoring of fission product releases.

The alternative time line, AB, represents an accident sequence where control of the plant cannot be regained by application of the EOPs, usually as a consequence of massive levels of equipment failure. In those cases containment would fail, either with or without prior core damage, and control room habitability would be lost. In cases where no core damage had occurred, the possibility of core damage resulting from consequential loss of equipment would require consideration. In such cases, it is likely that control room occupancy and reactor building access would be lost so that further actions would be initiated from outside the control room. At the point where control room actions are no longer effective or cannot be taken, the EOPs are no longer applicable to defining the actions to be taken. Currently, we have defined the actions to be taken during the time segment BC to be coping actions. As in the case of recovery actions, no formal programs or procedures have been defined other than the general provisions for activation of the EOF and General Office support. By the nature of the situation in such cases, a formal definition of actions to be taken to bring about stabilization and control will be difficult to develop.

The view graph then defines the time period for Severe Accident Management (segments AC and AB only), and the EOPs fully define the actions to be taken during those time periods. The EOPs, therefore, are by definition the Severe Accident Management program.

SEVERE ACCIDENT MANAGEMENT

- DEFINITIONS:

  ACTIONS TAKEN TO AVOID OR MINIMIZE PLANT DAMAGE RESULTING FROM AN INITIATING EVENT AND ANY COMBINATION OF ADDITIONAL EQUIPMENT FAILURES.

- APPLICABLE TIME PERIOD:

  FROM THE TIME OF THE INITIATING EVENT UNTIL THE OPERATORS ACHIEVE A LONG TERM STABLE PLANT CONDITION OR UNTIL CONTROL ROOM ACTIONS CAN NO LONGER INFLUENCE THE PLANT CONDITION.

- SOURCE OF GUIDANCE:

  THE EMERGENCY OPERATING PROCEDURES AND SUPPLEMENTAL PROCEDURES REFERENCED BY THE EOPs.

- ROLE OF RISK ASSESSMENT:

  DETERMINE OPTIMAL RESPONSE STRATEGIES FOR STRUCTURING EOPs AND THE DEGREE AND NATURE OF PLANT DAMAGE TO BE EXPECTED IN EACH ACCIDENT SEQUENCE.

Severe Accident Management

The view graph on Severe Accident Management (SAM) represents the PP&L view of SAM. The important feature of this viewgraph is the statement that the EOPs are the source of guidance for SAM.

In the PP&L view, however, this is not a passive relationship. The EOPs are deliberately structured to assure optimum use of all plant capability to avoid or limit the consequences of an accident regardless of the initiating event and the number of equipment failures. The means for accomplishing this is performance of a risk assessment which considers the actual procedures for the plant and the actual equipment capability. The results are examined to assure that all possible plant capabilities have been utilized for all accident sequences.

Thus, the EOPs are the heart of PP&L's Severe Accident Management process.

EMERGENCY OPERATING PROCEDURES

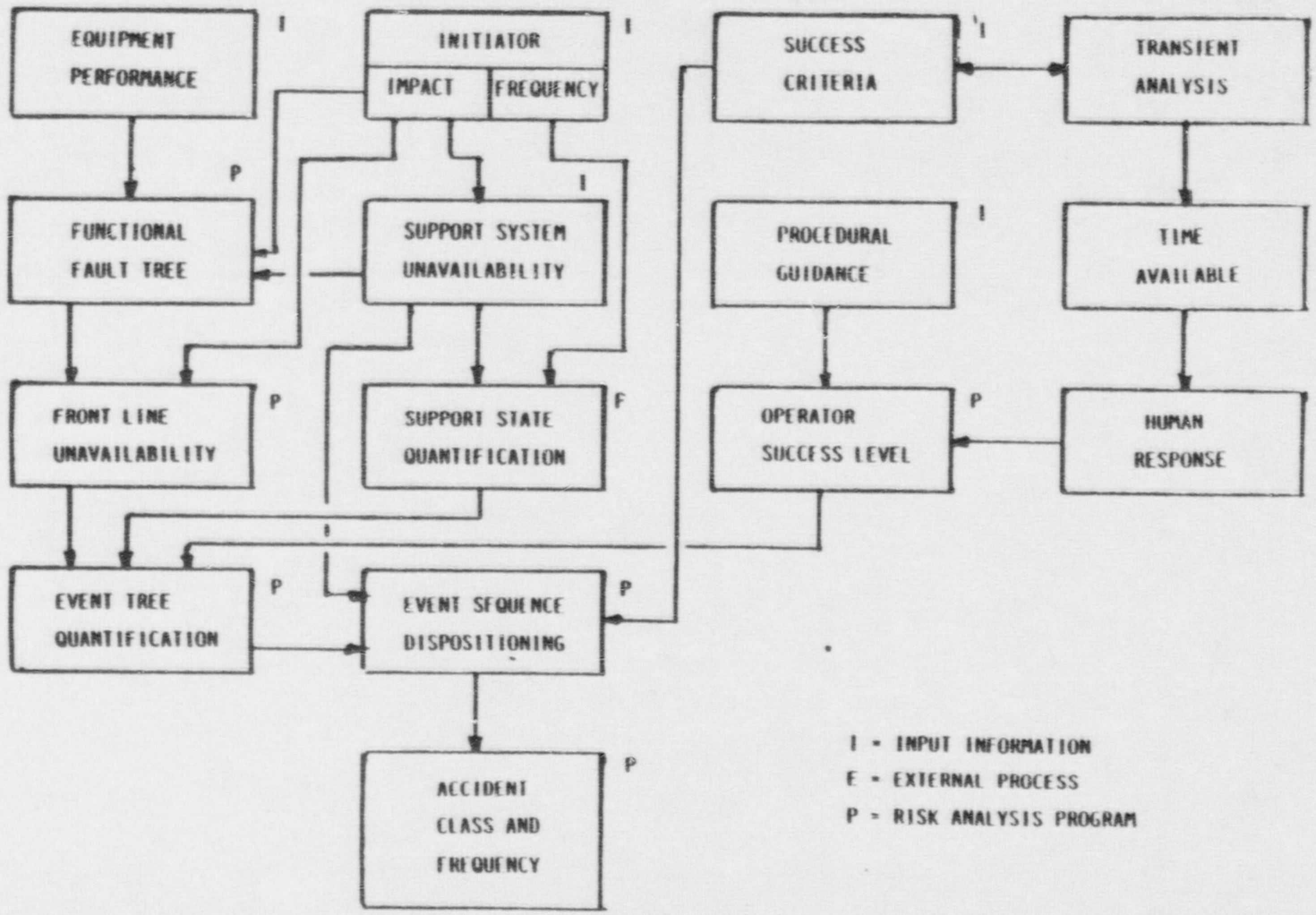
- THE EOPs MUST COVER ALL ACTIONS REQUIRED TO REGAIN CONTROL OF THE PLANT AND TO STABILIZE IT.
- THE EOPs MUST BE FUNDAMENTALLY BASED ON PLANT SYMPTOMS WHICH ARE RELIABLY INDICATED BY PLANT INSTRUMENTATION.
- THE OPTIMAL FORM FOR THE EOPs IS A FLOW CHART FORMAT WHICH PERMITS TIMELY INDICATION OF ALL ACTIONS FOR WHICH TIME IS CRITICAL.
- ACTIONS WHICH ARE NOT TIME CRITICAL MAY BE REFERENCED TO EXTERNAL SUPPLEMENTAL PROCEDURES.
- THE PROCEDURES MUST AVOID COMPLEXITIES AND AMBIGUITIES.
- THE PROCEDURES OR TRAINING MUST ADVISE THE OPERATOR OF TIME CONSTRAINTS AND PRIORITIES.
- THE ACTIONS SPECIFIED MUST BE FEASIBLE.
- THE PROCEDURES SHOULD REINFORCE BASIC OPERATOR TRAINING.
- PROCEDURES AND TRAINING FOR THEM MUST EXIST IF OPERATOR ACTION IS TO BE CONSIDERED SUCCESSFUL AND EFFECTIVE.

Emergency Operating Procedures

This viewgraph is self explanatory and represents PP&L criteria for effective EOPs. The process of assuring effectiveness is not yet complete at PP&L.

At the present time the concepts of Severe Accident Management and Risk Management and their relationship to and demands on EOPs and EOP training are being actively addressed.

The definitions presented here represent the Engineering perspective on this subject.



THE RISK ANALYSIS PROCESS

The Risk Analysis Process

This diagram represents the process which PP&L intends to use to maintain a valid Risk Assessment for Susquehanna.

We have structured our approach after the IDCOR BWR IPE methodology with enhancements which PP&L considers to be important. The most important of these enhancements are:

1. Extension of the support states to represent divisionality and channelization of systems.

2. The dispositioning of each Level 1 event tree end point to the full spectrum of final plant damage states using support state and event tree information.

3. The introduction of accident class definitions which permit a fine resolution of plant damage states and timing.

At the present time PP&L is preparing this analysis method for programming. The portions of the process which will be executed by the Risk Analysis program are indicated by P. The purpose of the diagram is to illustrate the important role that simulator experiments will play in the evaluation of risk at Susquehanna. The Human Response model which will be used is the HCR method with parameters derived from the program of Susquehanna simulator measurements.

This is particularly important because PP&L does not accept a low calculated plant damage frequency as proof of acceptability. PP&L examines each individual plant damage sequence to assure that the PP&L defense in depth criteria have been met. This process focuses direct attention to EOPs and expected operator performance in executing them. This process results in an intensive review of all EOPs, operator training and operator performance for those accident sequences identified for Susquehanna.

RISK MANAGEMENT

- o DEFINITION:

  THE PROCESS OF MONITORING PERFORMANCE, RISK ASSESSMENT, FEED BACK AND MODIFICATIONS INTENDED TO ASSURE VERY LOW PLANT DAMAGE FREQUENCY AND OPTIMAL USE OF PLANT FACILITIES TO PREVENT OR MITIGATE DAMAGE FROM AN INITIATING EVENT AND ANY COMBINATION OF EQUIPMENT FAILURES.

- o APPLICATION TIME PERIOD:

  FROM START UP THROUGH DECOMMISSIONING.

- o SOURCE OF GUIDANCE

  THE STUDY WHICH DEMONSTRATES THE DEFENSE IN DEPTH PERFORMANCE OF THE PLANT, THE PERFORMANCE MONITORING PROGRAMS, AND THE PLANT RISK ASSESSMENT. PERIODIC UPDATING TO REFLECT CHANGES IN PERFORMANCE, PROCEDURES OR PLANT EQUIPMENT MUST BE DONE FOR THE RISK ASSESSMENT AND THE DEFENSE IN DEPTH STUDY.

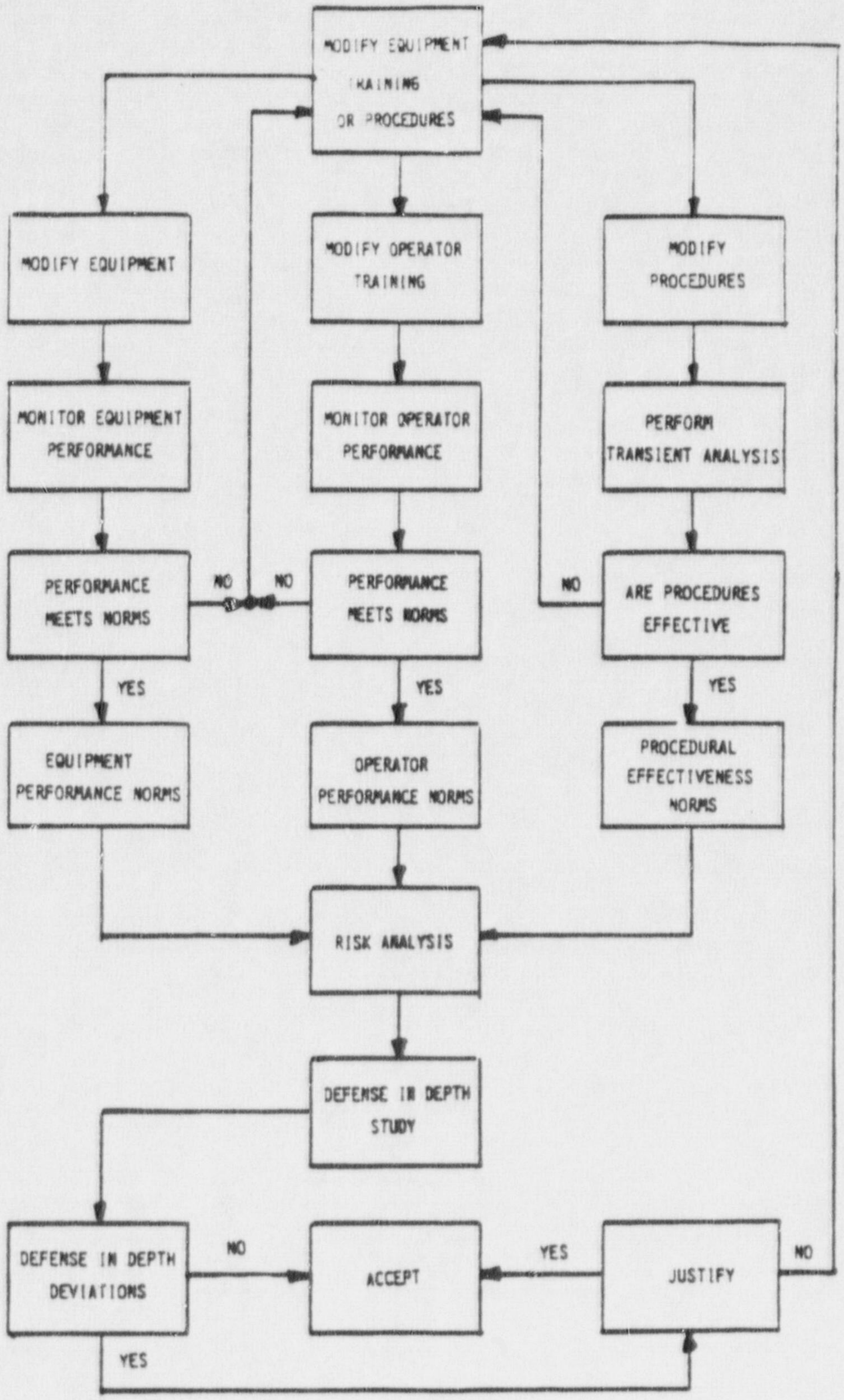
Risk Management

This viewgraph identifies the Risk Management process as defined by PP&L. At the present time we are in the process of performing the first cycle of this continuing process. We are performing an Integrated Risk Reduction Study which will identify all instances in which the PP&L defense in depth criteria are violated. This study is based on the current (and committed) state of the plant and the results of the first Susquehanna IPE performed for IDCOR.

When all deviations from the defense in depth criteria are identified a recommendation for correction (or justification) of all deficiencies will be made for management consideration. This recommendation will include both plant modifications and procedural modifications. These modifications will be selected to eliminate all defense in depth deviations at minimum cost and with minimum impact on plant operations. Several alternative combinations will be identified.

This process will permit us to select modifications which have the maximum impact for reducing the severe accident risk for Susquehanna and will provide the basis for justifying the adequacy of the actions taken.

THE RISK MANAGEMENT PROCESS



The Risk Management Process

The viewgraph is intended to represent a continuing process of risk management. There are five major activities in the overall process. These are:

1. Monitor equipment performance (unavailabilities of systems).

2. Monitor operator performance (All deviations, errors, and execution time in performance of EOPs).

3. Transient analysis (determination of flow and timing requirements against various criteria).

4. Risk analysis (the IPE process to determine the impact of changes).

5. Demonstration of defense in depth (evaluation of all damage sequences from the risk analysis).

This cycle of calculations must be performed on a periodic basis to determine the effects of changes in:

1. Equipment
2. Procedures
3. Operator training
4. Improved analysis methods
5. Improved plant models

This process is intended not only to detect the influence of changes, but also to incorporate the influence of accumulative measurements of plant performance in reducing uncertainties in the input data to the risk analysis process.

PP&L will use this approach to obtaining closure on severe accident issues. We shall consider absence of high frequency damage events and demonstration of defense in depth as demonstration of an adequate level of public safety. We do not intend to perform off-site consequence calculations. We believe that when we have demonstrated defense in depth, any reasonable calculation of off-site consequence will show negligible public risk when calculated event sequence frequencies are considered in combination with the consequence.

Our approach allows us to focus on factors which we can monitor and control - equipment and operator performance.

APPLICATION OF RISK ASSESSMENT(1)

- o BOTTOM LINE APPLICATIONS
  - OFF-SITE CONSEQUENCES
  - CORE DAMAGE FREQUENCY
- o FOR THESE APPLICATIONS THERE IS A TENDENCY TO A CONSERVATIVE BIAS
  - AVOIDANCE OF COMPLEX TRANSIENT ANALYSIS
  - SIMPLIFICATION OF EQUIPMENT AND OPERATIONAL DEPENDENCIES
  - IGNORANCE OF ACTUAL CAPABILITIES OF THE PLANT
- o THE TENDENCY TO CONSERVATIVE REPRESENTATION OF THE BWR IS PARTICULARLY STRONG BECAUSE OF:
  - THE COMPLEX INTERRELATIONSHIPS BETWEEN PLANT SYSTEMS
  - THE MULTIPLE FUNCTIONS OF SOME PLANT SYSTEMS
  - THE COMPLEXITIES OF THE PRESSURE SUPPRESSION SYSTEM

Application of Risk Assessment(1)

There is often a tendency in risk assessment to focus on the "bottom line", the core damage frequency or the extent of off-site consequences. When the "bottom-line" is the dominant objective for performing a risk assessment, there is generally a powerful motivation to use conservative assumptions for a number of reasons. The motivations which are most clearly seen are:

1. Simplification of the analysis, particularly the analysis of the accident transient.

2. Derivation of credible results which cannot be challenged on the basis of optimism in the analysis.

Generally these two motivations go hand in hand, and, while they may be laudable objectives, the conservatisms do not produce results which can withstand close scrutiny when subjected to critical examination in the light of our experience.

The key weakness in any risk analysis is the treatment of:

1. commonalities,

2. interactions, and

3. the effectiveness of procedures and training.

Doubt can be cast on virtually any prior analysis based upon close examination of these factors. Further, our experience tells us that prior risk assessments have had a very poor record in predicting the actual events which have occurred.

While PP&L believes that calculation of a low frequency of plant damage for Susquehanna is mandatory, we do not believe that such a calculation is sufficient. We are certain that our analysis is incomplete in terms of the total array of event combinations which can be expected to threaten a nuclear power station. While we are striving to broaden our evaluations to cover all credible event combinations, we do not expect ever to achieve this goal.

APPLICATION OF RISK ASSESSMENT(2)

- DEMONSTRATION OF ADEQUACY
  - EVALUATE ADEQUACY OF PLANT DESIGN
  - EVALUATE ADEQUACY OF EOPs AND OPERATOR TRAINING
  - EVALUATE IMPACT OF MODIFICATIONS

- EXCESSIVE CONSERVATISM IN THE RISK ASSESSMENT PROCESS IS FULLY AS OBJECTIONABLE AS EXCESSIVE OPTIMISM.

- THE EVALUATION MUST BE AS REALISTIC AS POSSIBLE AND CONTAIN AS FEW ASSUMPTIONS AS POSSIBLE.

- OTHERWISE, ATTENTION WILL BE FOCUSSED ON EVENT SEQUENCES WHICH ARE AN ARTIFICE OF THE ASSUMPTIONS, AND THE MOST PROBABLE EVENT SEQUENCES WHICH MAY REQUIRE EXPLICIT OPERATOR TRAINING MAY BE MISSED.

Application of Risk Assessment (2)

The PP&L objective in performing risk assessment is to demonstrate adequacy of Susquehanna against deficiencies in:

1. plant design,

2. procedures, and

3. training

In addition, we wish to assure that any changes in these items will not have an adverse impact on safe operation of the plant.

In pursuing this objective we have found that excessive conservatism in the risk assessment is fully as objectionable as excessive optimism. The reason for this is that a conservatism will often obscure some lower probability combinations of circumstances which may in fact be the most likely circumstances to be encountered given the occurrence of the accident scenario. This will then prevent such occurrences from being covered either in procedures or in operator training. This then causes a vulnerability to such an occurrence.

For this reason, we must attempt to perform the analysis on the most realistic basis possible so that the most probable event circumstances in accident sequences are not obscured, and we can then be certain to accommodate them in our procedures and training.

THE PP&L APPROACH

TO

RISK ASSESSMENT

INTENT

WE WISH TO ASSURE THAT ALL POSSIBLE CAPABILITY OF THE PLANT TO PREVENT OR MITIGATE THE CONSEQUENCES OF PLANT DAMAGE WILL BE EFFICIENTLY AND EFFECTIVELY APPLIED FOR ANY INITIATING EVENT AND FOR ANY COMBINATION OF EQUIPMENT FAILURES.

The PP&L Approach to Risk Assessment

Our goal is to assure that all possible capability of the plant to prevent or mitigate the consequences of plant damage will be efficiently and effectively applied for any initiating event and any combination of equipment failures. Our realistic approach to risk analysis with incorporation of the results into our procedures and training when combined with symptom based procedures give us the most credible defense against accidents having severe consequences.

This approach also results in calculation of greatly reduced plant damage frequencies. While we do not necessarily believe that our calculated values are a proper representation of reality due to lack of completeness, we do believe that much of the calculated reduction in risk in comparison with the "bottom line" approach to risk analysis is real.

USE OF RISK ASSESSMENT AT PP&L

- WE USE RISK ANALYSIS METHODS TO:

  - DEVELOP OPTIMAL EMERGENCY OPERATING PROCEDURES

  - DEVELOP OPERATOR TRAINING PROGRAMS TO ASSURE OPERATOR KNOWLEDGE OF CRITICAL RESPONSE ACTIONS.

  - IDENTIFY POTENTIAL MINOR MODIFICATIONS WHICH CAN REDUCE THE FREQUENCY OF OR MITIGATE THE CONSEQUENCES OF SIGNIFICANT SEVERE ACCIDENT SEQUENCES.

  - ASSURE "DEFENSE IN DEPTH" IN OUR EQUIPMENT AND OUR PROCEDURES.

  - EVALUATE ALTERNATIVES FOR PLANT MODIFICATIONS.

  - TRAIN AND INFORM MANAGEMENT TO ASSURE PROPER COORDINATION OF RISK IN DECISION MAKING.

  - DEMONSTRATE A HIGH LEVEL OF SAFETY IN SUSQUEHANNA OPERATIONS.

Use of Risk Assessment at PP&L

The Susquehanna IPE, in spite of its known deficiencies, has proven to be a very useful document for PP&L and has had a very significant impact on many of our most important activities. This view graph lists the types of applications which have already been made of this work. In time it is intended that our risk assessment activity will be directly integrated into the process of plant and procedural modifications and operator training.

THE PP&L APPLICATION OF THE IPE METHODOLOGY

- WE USE AN EXPANDED VERSION OF THE SUPPORT STATE METHOD INTRODUCED FOR THE BWR-IPEM.

- WE USE THE FUNCTIONAL FAULT TREE CONCEPT TO DEVELOP SYSTEM UNAVAILABILITIES.

- WE HAVE DEVELOPED SUSQUEHANNA SUCCESS CRITERIA USING.

    - BWRSAR/CONTAIN-ORNL
    - IN-HOUSE CODES FOR ATWS AND SBO
    - SIMPLE END-POINT THERMODYNAMICS

- WE USE THE FRONT-LINE AND SUPPORT SYSTEM STATUS INFORMATION FOR EACH INDIVIDUAL EVENT TREE END POINT TO DISPOSITION IT TO A SET OF FINAL PLANT DAMAGE STATES.

- WE USE AN ACCIDENT CLASSIFICATION SCHEME WHICH GIVES THE TIME, NATURE AND DEGREE OF CORE DAMAGE, REACTOR VESSEL STATUS, THE QUANTITY OF CORE DEBRIS ON THE DRYWELL FLOOR, AND THE TIME AND NATURE OF CONTAINMENT LOSS OF INTEGRITY.

- WE DO NOT BIN SEQUENCES AND WE DO NOT USE A FORMAL CONTAINMENT EVENT TREE.

- WE DO NOT DERIVE OR CONSIDER FISSION PRODUCT SOURCE TERMS.

The PP&L Application of the IPE Methodology

PP&L has incorporated a number of enhancements to the IDCOR IPE methodology which are intended to both enhance the credibility of the results and enhance the ability to utilize its results in practical applications. The most important enhancements are:

1. extension of the support state representations to provide greater resolution of support system failure combinations,

2. development of an accident classification scheme which permits a user defined degree of resolution of plant damage states and timing,

3. a method of dispositioning each Front-line Function Event Tree end point to a full spectrum of final plant damage states individually,

4. new transient analysis methods which we consider to provide more credible results than predecessor methods,

5. methods for directly incorporating response time distribution into the results of the analysis, and

6. methods for incorporating the influence of Technical Specifications into the results of the analysis.

While we have not yet accomplished our goal of creating a computer program to perform these calculations, the mathematical models and computational logic have all been developed.

We admit that what we have done carries the IDCOR method far beyond what was originally intended. Nevertheless, we believe these enhancements have a profound influence on the utility of the final results, and we believe that, if the approach can be computer based, that its application is no more labor intensive than the unenhanced approach.

SUSQUEHANNA IPE
ACCIDENT CLASSES

TIME OF CORE DAMAGE

A. 0 - 2 HRS
B. 2 - 4 HRS
C. 4 - 10 HRS
D. 10 - 24 HRS
E. CONSEQUENTIAL
Z. NO FAILURE

VESSEL STATUS

I - INTACT
F - FAILED
X - NOT DETERMINED

TIME OF CONTAINMENT FAILURE

A. 0 - 4 HRS
B. 4 - 10 HRS
C. 10 - 24 HRS
D. > 24 HRS
V. VENTED
Z. NO FAILURE
X. NOT DETERMINED

TYPICAL CLASSES

AIZ - CORE DAMAGE BEFORE 2 HRS, BUT VESSEL AND CONTAINMENT INTACT

AXX - CORE DAMAGE BEFORE 2 HRS, VESSEL AND CONTAINMENT STATUS NOT DETERMINED

EFD - CORE MELT AND VESSEL FAILURE AS A CONSEQUENCE OF CONTAINMENT FAILURE AFTER 24 HOURS

Susquehanna IPE
Accident Classes

When the IPE was performed for Susquehanna in the latter half of 1985, PP&L believed that an accident classification scheme was needed which would identify the time of core damage and containment failure. In addition we wished to identify the status of the reactor vessel because we believed that it was important to attempt to arrest the core damage progression before reactor vessel failure if possible. By so doing we believed we would achieve a great reduction in the relative frequency of events in which the core on the floor phenomena greatly increase the challenge to containment integrity while adding a high level of uncertainty to these relatively severe sequences. We also wished to segregate the instances where containment integrity was lost through deliberate venting as opposed to overpressure failure. We believe that venting can greatly reduce the likelihood of truly severe accident sequences in which containment failure can release fission products to the environment without scrubbing through the suppression pool. We devised a very simple three letter scheme for classifying all accident sequences. The first letter specifies the time of core damage (but not the degree) including damage that occurs as a consequence of containment failure prior to core damage. The second specifies the status of the reactor vessel and the third specifies the time of containment damage or the occurrence of venting.

Since the IPE event trees are only level 1 event trees many sequences end with only core damage assured, but with the eventual status of the reactor vessel or the containment not determined. The classification scheme also identifies these occurrences.

The use of the support state method in combination with the event tree successes and failures, however, gives essentially complete knowledge of the plant status at the time of core damage and so with this knowledge and knowledge of the initiating event it is a relatively simple matter to calculate the likelihood of success at:

1. Regaining vessel injection capability,

2. Venting the containment, and

3. Regaining or losing containment heat removal capability.

This information was then used to disposition the full spectrum of final plant damage states for each individual event tree end point.

This information is of vital importance in demonstrating defense in depth for all accident sequences of the plant. In addition this approach allows us to isolate the high uncertainty sequences (both those having high potential consequence and those having low potential consequence) so that high uncertainty sequences are not allowed to distort the overall assessment of plant performance in severe accidents.

CAPABILITY FOR SUSQUEHANNA IPE

ACCIDENT CLASS DEFINITION

PLANT DAMAGE STATE [1]

1. NO DAMAGE
2. REACTIVITY TRANSIENTS [2]
3. METAL WATER REACTION
4. CORE MELT AND REACTOR VESSEL FAILURE
5. VENT +1 [3]
6. VENT +2 [3]
7. VENT +3 [3]
8. VENT +4 [3]
9. 1. + COPF [4]
10. 2. + COPF [4]
11. 3. + COPF [4]
12. 4. + COPF [4]
13. COPF + 4. [4]

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(1) TIME DEFINITION IS THE SAME AS FOR THE SSES IPE

(2) MECHANICAL CLAD FAILURE

(3) SEQUENCE NOT IMPLIED BY ORDER

(4) SEQUENCE IMPLIED BY ORDER, COPF = CONTAINMENT OVER PRESSURE FAILURE

Capability for Susquehanna IPE
Accident Class Definition

The actual capability of the simple accident classification scheme used for the Susquehanna IPE is considerably greater than shown in the preceding viewgraph. In this viewgraph it is shown that this simple scheme actually gives the capability of defining twelve damage classes. These twelve classes have been used in sensitivity studies to demonstrate the selective influence of various risk analysis assumptions and models. The IPE methodology as executed by PP&L allows execution of such sensitivity studies with minimal effort.

Based on the investigations we have carried out, we are inclined to believe that the current accident classification scheme does not yet have sufficient resolution. For example, we believe it is worth while to distinguish:

1. the degree and nature of core damage,

2. the quantity of core debris material falling to the drywell floor, and

3. the nature of containment failure (over pressure or over temperature).

This capability will require a more elaborate accident classification scheme and a more elaborate specification of success criteria. It is our intent to provide this additional resolution of plant damage states in our first IPE revision. It is our intent to create a program which will automate the calculations required.

OBJECTIVES FOR PROVIDING DEFINITION OF PLANT DAMAGE STATES(1)

- o UNCERTAINTY ANALYSIS
  - THE FURTHER DAMAGE PROGRESSES, THE HIGHER THE UNCERTAINTY.
  - THEREFORE WE WISH TO SEGREGATE THE EARLY STAGES WITH LOW UNCERTAINTY FROM THE LATER STAGES HAVING HIGHER UNCERTAINTY.

- o UNCERTAINTY ISSUES
  - MECHANICAL CLAD DAMAGE FROM POWER TRANSIENTS. (BELIEVED TO BE TREATED VERY CONSERVATIVELY)
  - CORE MATERIAL RELOCATION. (CONTROL BLADES RELOCATE FIRST)
  - CORE MATERIAL HOLDUP ON THE CORE SUPPORT PLATE. (CORE SUPPORT PLATE FAILS BEFORE UO_2 MELT)
  - REACTOR LOWER HEAD FAILURE. (FAILS LOCALLY AFTER LOWER PLENUM DRY OUT)

Objective for Providing Definition of Plant Damage States(1)

The plant processes which are important prior to plant damage are well understood and have relatively little uncertainty associated with them. As the accident progresses and core damage is initiated, the degree of uncertainty associated with the physical processes involved begins to increase. When large scale relocation of core material begins, the uncertainties become very great indeed. These uncertainties involve not only the rate of the progress of the event, but also the phenomena which are the dominant and controlling influences. These uncertainties become even greater when reactor vessel failure occurs and core debris falls or it ejected to the drywell floor.

Current risk evaluations have made extensive, if not exclusive, use of expert judgement to quantify the effects of such events in lieu of definitive calculations using credible computer models. For the BWR plant, these phenomena which dominate the late phases in an accident sequence have an overwhelming impact on the assessment of the degree of plant damage and on the severity of off-site consequences.

For this reason the PP&L approach involves use of the best available computer programs to calculate such accident transients in detail. While this process removes much of the subjectiveness of expert judgement, it nevertheless conceals a considerable degree of subjectiveness which is exercised when the specific mathematical models and physical data are chosen to perform such calculations. While we believe this process is far superior to expert judgement because it focuses and confines the factors which involve subjectivity, we do not believe that is removes entirely the great uncertainty over the results of such analysis.

Therefore, we wish to develop a sufficient degree of resolution in definition of plant damage categories such that sequences of high uncertainty can be maintained separated from sequences of lower uncertainty. Further, we wish to segregate sequences having high uncertainty resulting from different physical phenomena. While it is not possible to do this with complete success, it is possible to accomplish this to a degree which prevents the influence of the large uncertainties from fully obscuring the overall results of the risk analysis. The definition of the criteria for core stabilization is a major contributor to this type of compartmentalization of uncertainty.

OBJECTIVES FOR PROVIDING DEFINITION
OF
PLANT DAMAGE STATES(2)

- CORE DEBRIS SPREADING ON THE DRYWELL FLOOR
  - ATTACK ON FLOOR
  - RATE OF NON-CONDENSIBLE PRODUCTION
  - ATTACK ON PEDESTAL
  - ATTACK ON FLOOR PENETRATIONS
  - ATTACK ON DRYWELL LINER
- CONTAINMENT FAILURE MODE
  - WETWELL OR DRYWELL
  - OVER PRESSURE OR OVER TEMPERATURE
  - PRESSURE AT TIME OF FAILURE
  - SIZE OF LEAKAGE PATH WAY

Objectives for Providing Definition of Plant Damage States (2)

This is so because we expect a very high degree of success at saving the reactor vessel with relatively low uncertainty so that the high uncertainty phenomena which can lead to high consequence events are sharply limited. The additional resolution provided in terms of the extent of core damages and the quantity of core debris reaching the drywell floor are additional means for compartmentalizing the nature and impact of uncertainties.

IMPORTANT ISSUES OF THE CONTAINMENT FAILURE MECHANISM

- SEVERE CONTAINMENT FAILURE COULD CAUSE LOSS OF REACTOR INJECTION CAPABILITY WITH SUBSEQUENT CORE MELT AND REACTOR VESSEL FAILURE.
- CONTAINMENT FAILURE IN THE DRYWELL IMPLIES RELEASE WITHOUT POOL SCRUBBING.
- A DISTINCTION BETWEEN DRYWELL OVER TEMPERATURE AND CONTAINMENT OVER PRESSURE FAILURE COULD BE MADE ON THE BASIS OF THE ACCIDENT SEQUENCE.
- THE DISTINCTION BETWEEN WETWELL FAILURE AND DRYWELL FAILURE FROM OVER PRESSURE CANNOT BE MADE ON THE BASIS OF THE ACCIDENT SEQUENCE.
- THE MAGNITUDE OF THE FAILURE PROBABLY CANNOT BE MADE ON THE BASIS OF THE ACCIDENT SEQUENCE.
- IF THE CONDITIONAL CONTAINMENT FAILURE PROBABILITY CAN BE SUFFICIENTLY REDUCED, THE CONTAINMENT FAILURE MODE IS REALLY NOT IMPORTANT.
- THE PP&L GOAL FOR CONDITIONAL CONTAINMENT FAILURE PROBABILITY IS AROUND 1% OF ALL CORE DAMAGE EVENTS. WE DO NOT YET KNOW IF THIS GOAL IS PRACTICABLE.
- WHEN THE UNCERTAINTIES IN THE ANALYSIS ARE CONSIDERED, IT IS PROBABLY BETTER TO CONSIDER CONDITIONAL CONTAINMENT FAILURE PROBABILITY INITIATOR BY INITIATOR.

Important Issues
of the
Containment Failure Mechanism

This view graph lists some of the important considerations made when selecting the degree of resolution to be used in plant damage states and success criteria in order to compartmentalize uncertainty. The view graph is intended to indicate what kinds of uncertainty can be compartmentalized and what kinds cannot. These are the types of considerations required for an effective choice of plant damage states.

UNCERTAINTY

- o UNCERTAINTY IN EQUIPMENT PERFORMANCE AND OPERATOR PERFORMANCE CAN BE REDUCED BY PROGRAMS TO MONITOR PERFORMANCE.

- o UNCERTAINTY IN CORE DAMAGE PROGRESSION, REACTOR VESSEL FAILURE, CORE DEBRIS BEHAVIOR, CONTAINMENT FAILURE CANNOT BE MONITORED OR DIRECTLY MEASURED.

- o A STRATEGY WHICH SEGREGATES HIGH UNCERTAINTY SEQUENCES FROM LOW UNCERTAINTY SEQUENCES CAN PROVIDE A BETTER UNDERSTANDING OF DOMINANT RISK CONTRIBUTORS.

- o HIGH UNCERTAINTY, LOW FREQUENCY SEQUENCES SHOULD BE SEGREGATED.

- o THIS SEGREGATION SHOULD KEEP SEPARATE LOW CONSEQUENCE SEQUENCES AND HIGH CONSEQUENCE SEQUENCES.

- o THIS APPROACH AVOIDS APPLICATION OF LARGE UNCERTAINTY BOUNDS TO THE LARGEST FRACTION OF ACCIDENT SEQUENCES AND LIMITS THE EXAGGERATION OF SEVERE CONSEQUENCE EVENTS.

Uncertainty

For the front-end analysis of risk up to the point of severe core damage, equipment and operator performance can be monitored so that the uncertainty in the analysis associated with these inputs can be continually reduced by the availability of a growing database.

Beyond the initiation of core material relocation in the reactor vessel nothing can be done in the plant to further reduce the associated uncertainties. Unless further experimental work is done by industry or the government, improvement in the state of our knowledge of these phenomena cannot be expected. For this reason accident sequences should be segregated on the basis of the extent of damage in risk analysis. This process tends to segregate high consequence sequences from low consequence sequences as well as segregating high uncertainty from low uncertainty sequences.

The purpose, again, is to avoid allowing the uncertainties to obscure the overall nature of the risk associated with plant operation.

OBJECTIVES
OF THE
SUSQUEHANNA INDIVIDUAL PLANT EVALUATION

- WE WISH TO DISTINGUISH THE TIME AT WHICH CORE DAMAGE BEGINS AND TO CHARACTERIZE THE EXTENT OF CORE DAMAGE.
- WE WISH TO DISTINGUISH THE TIME AT WHICH VENTING BECOMES NECESSARY TO SAVE THE CONTAINMENT PRESSURE BOUNDARY INTEGRITY.
- WE WISH TO DISTINGUISH THE TIME OF CONTAINMENT FAILURE.
- WE WISH TO DISTINGUISH THE NATURE OF CONTAINMENT FAILURE (OVER TEMPERATURE OR OVER PRESSURE).
- WE DO NOT INTEND TO DEFINE THE MAGNITUDE OR LOCATION OF CONTAINMENT FAILURE.
- WE DO NOT INTEND TO CALCULATE FISSION PRODUCT RELEASE OR OFF-SITE CONSEQUENCE.
- WE PROVIDE SUFFICIENT DEFINITION TO ALLOW THE HIGH UNCERTAINTY, HIGH CONSEQUENCE SEQUENCES TO BE SEGREGATED.
- SIMILARLY WE WISH TO SEGREGATE THE HIGH UNCERTAINTY, MINOR CONSEQUENCE SEQUENCES.

Objectives for the Revision of the Susquehanna Individual Plant Evaluation

This view graph presents our primary objectives, relative to resolution of plant damage states, which we wish to accomplish in our revision to the Susquehanna IPE. While the resolution provided by the original Susquehanna IPE exceeded that provided by any prior study, we have found that newly developed concerns over a variety of accident phenomena require greater resolution detail yet.

APPLICATION OF THE RISK ASSESSMENT TO DEMONSTRATE DEFENSE IN DEPTH

- o PP&L IS CONDUCTING AN "INTEGRATED RISK REDUCTION STUDY".

- o THE GOAL OF THIS STUDY IS TO DEMONSTRATE "DEFENSE IN DEPTH" FOR EVERY PLANT DAMAGE SEQUENCE DERIVED FOR SUSQUEHANNA (~120 SEQUENCES)

- o BOTH EQUIPMENT AND PROCEDURAL CRITERIA ARE EVALUATED FOR EACH SEQUENCE.

- o EVALUATION OF THE PROCEDURAL CRITERIA INVOLVE AN EXHAUSTIVE REVIEW OF THE SUSQUEHANNA EOPs AGAINST EACH PLANT DAMAGE SEQUENCE.

- o COMPLETION OF THE STUDY WILL RESULT IN IDENTIFICATION OF ALL VIOLATIONS OF "DEFENSE IN DEPTH" FOR THE PRESENT CONDITION OF OUR PLANT AND PROCEDURES EVALUATED AGAINST THE SUSQUEHANNA IPE.

- o MODIFICATIONS TO EQUIPMENT AND PROCEDURES SHALL BE DEVELOPED WHICH MOST EFFICIENTLY AND EFFECTIVELY RESOLVE THE DERIVED "DEFENSE IN DEPTH" VIOLATIONS.

- o RECOMMENDATIONS FOR A SET OF ALTERNATIVE MODIFICATION PACKAGES SHALL BE DEVELOPED FOR MANAGEMENT CONSIDERATION.

Application of the Risk Assessment to Demonstrate Defense-in-Depth

The "Integrated Risk Reduction Study" represents the first systematic and integrated feedback from the Susquehanna IPE to PP&L management to identify potential vulnerabilities in Susquehanna equipment and procedures. This study is the "bottom line" of our risk management process which was described in a previous view graph. The first round of this process, while tedious, has been a reasonable undertaking since the simple support state structure of the IDCOR IPE resulted in only about 120 separate core damage sequences for Susquehanna. We have currently completed cur evaluation of each sequence against our defense-in-depth criteria and have identified a number of deviations.

On the equipment side this evaluation has shown a number of problem areas such as wetwell venting, backup DC power, and others which we have already recognized from our dominant accident sequences. We have also identified a number of procedural problem areas related to suppression pool mass addition, reactor reflood, and blowdown cooling which we are currently evaluating. The report on this work will include recommendations to management for actions to resolve the deviations we have found in an integrated manner. Various alternatives will also be provided with the rationale for each recommendation.

This information will then permit PP&L to demonstrate low core damage frequency, and defense-in-depth for all analyzed accident sequences to respond to the NRC's IPE generic letter and severe accident policy.

This process for our revised IPE will be more complex because with the finer resolution of support states and plant damage states we will have a much larger number of accident sequences. Nevertheless, we expect repetition of the process of demonstrating defense-in-depth to be practicable since any accident sequence more severe that one which just meets defense-in-depth need not be analyzed. This is expected to greatly reduce the number of sequences requiring detailed examination.

In addition we expect relatively few deviations when the initial series of recommendations are implemented.

RATIONALE FOR THE PP&L APPROACH

- o WE WISH TO ASSURE OURSELVES THAT:
  - THERE ARE NO INHERENT DESIGN FLAWS IN OUR PLANT.
  - OUR PROCEDURES HAVE ELIMINATED OR GREATLY DIMINISHED THE LIKELIHOOD OF DAMAGE FROM THE MOST PROBABLE COMBINATIONS OF EVENTS AND EQUIPMENT FAILURES.
  - OUR OPERATORS WILL MAKE OPTIMUM USE OF PLANT EQUIPMENT TO AVOID OR MINIMIZE SEVERE DAMAGE TO THE PLANT.
  - PLANT OPERATION POSES A MINIMAL RISK TO THE PUBLIC.
- o WE ALSO WISH TO ACHIEVE CLOSURE ON: "HOW SAFE IS SAFE ENOUGH?"
  - WE DO NOT CONSIDER A LOW BOTTOM LINE DAMAGE FREQUENCY AS PROOF OF ADEQUACY.
  - WE WISH TO DEMONSTRATE "DEFENSE IN DEPTH" AS THE MEASURE OF ADEQUACY OF SUSQUEHANNA OPERATIONAL RISK.
  - IF WE DEMONSTRATE "DEFENSE IN DEPTH" WE BELIEVE:
    - THE FREQUENCY OF PLANT DAMAGE FROM INTERNAL EVENTS WILL BE FOUND TO BE EXTREMELY LOW.
    - OFF-SITE CONSEQUENCES WILL BE FOUND TO BE EXTREMELY LOW BY ANY REASONABLE ANALYSIS.
    - THE REDUCTION IN CALCULATED FREQUENCY OF PLANT DAMAGE IS REAL.

Rationale for the PP&L Approach

PP&L does not wish to demonstrate adequacy of the safety of Susquehanna operations on the basis of a low frequency of severe fission product release and correspondingly low expected consequences from plant operation. We believe this strategy is seriously flawed in several respects.

1. The physical processes and phenomena involved in developing this information have an extremely high degree of uncertainties.

2. The basis for determination of long term health effects cannot be considered to be much better than conjecture by experts without universal acceptance.

3. The process does not focus on what is really important in operation of the plant, but rather on consequences of failure.

4. The result, regardless of its nature, assures a negative public and political reaction.

Since many believe that demonstration of low consequence even with poor containment performance is acceptable, they pay little attention to assumptions and actions that can have a dramatic influence on calculated or actual containment performance.

PP&L believes that the importance of containment integrity is greatest when fission product releases have occurred or can occur, and, therefore, we believe that is essential to demonstrate a low conditional probability of containment failure when fission product release is involved in the event.

The defense-in-depth approach is very much structured to accomplish exactly this goal. Meeting the defense-in-depth criteria assures that three levels of independent equipment failure must occur before severe fission product release to the environment can occur, and it also assures that procedural actions taken or not taken will not cause the sequence to deteriorate. Thus, satisfying the criteria, we believe, assures a very low probability of severe consequence (containment failure). Further, the fact that a wide variety of initiating events in combination with all possible combinations of equipment failure, up to a level where severe consequences will occur, in combination with symptom based procedures, which in general are independent of the nature of the event and equipment failures, gives us the best capability we know how to achieve to accommodate accidents which are unforeseen in the risk analysis. That is, they are unforeseen in the calculation of frequency, but they, or their equivalent, are considered in the development of procedures and the testing of their adequacy.

On this basis we believe that our greatest vulnerability is that our calculated plant damage frequencies may be in error as a result of event sequences we have not considered, but we believe our procedures and training are structured to assure full utilization of the plant's capabilities in response to any event.

RATIONALE FOR THE PP&L APPROACH

- DEMONSTRATION OF ACCEPTABLE OFF-SITE CONSEQUENCES PUTS THE PRIMARY ATTENTION ON THE BACK END ANALYSIS:
  - CORE MELT PROGRESSION
  - VESSEL FAILURE MECHANISMS
  - CONTAINMENT FAILURE MECHANISMS
  - FISSION PRODUCT TRANSPORT

- THIS FOCUS ON THE BACK END ANALYSIS:
  - HAS HIGH UNCERTAINTY
  - ASSURES A NEGATIVE PUBLIC AND POLITICAL REACTION
  - DOES NOT FOCUS ON WHAT IS IMPORTANT IN OPERATION OF THE PLANT

- DEMONSTRATION OF "DEFENSE IN DEPTH" PUTS THE PRIMARY ATTENTION ON THE FRONT END ANALYSIS:
  - EQUIPMENT PERFORMANCE
  - PLANT RESPONSE CHARACTERISTICS
  - EFFECTIVENESS OF PROCEDURES
  - EFFECTIVENESS OF OPERATOR TRAINING

- IF WE FOCUS ON THE FRONT END ANALYSIS:
  - WE CAN MONITOR AND CONTROL THE ELEMENTS INVOLVED IN DETERMINATION OF RISK
  - WE PRESENT A POSITIVE IMAGE OF OUR PERFORMANCE

Our risk management process, in which we

1. monitor equipment performance,

2. monitor operator EOP performance, and

3. test the effectiveness of our procedures against the various accident sequences which have been identified for the plant using the latest analysis methods and programs

gives us the basis for a convincing demonstration that we are properly managing plant operations. This approach has the advantage that we are able to focus on monitoring and controlling equipment and operator performance and that it presents a positive rather than a negative image of our performance.

AN OVERVIEW OF
THE PP&L RISK ANALYSIS PROCESS

AN OVERVIEW OF
THE PP&L RISK ANALYSIS PROCESS

- o DESCRIPTION CATEGORIES
  - INPUT DATA
  - THE PROCESS ELEMENTS
  - EXTERNAL PROCESSES
  - OUTPUT

An Overview of the PP&L Risk Analysis Process

In the following description the PP&L risk analysis process shall be described as a computer program. At this time no computer program to perform this analysis exists, but computational algorithms and program logic have been created which describe the mathematical features of such a program. This approach is taken because it is believed that the descriptions will be easier to understand, particularly in terms of the logical structure and interrelationships of the various parts of the process.

In the material which follows, the descriptions are organized into the following categories.

1. Input data

2. The process elements (subroutines)

3. External processes

4. Output

This order is chosen for the presentation because it is necessary to understand the nature of the various blocks of input data before the computational processes can be properly understood. This does introduce the question, during the input description process, as to why a particular item is needed. The need will become apparent when the computational processes are described.

The descriptions offered here will not be detailed mathematical descriptions, but will be functional descriptions of what the calculations do and why they are done in a particular fashion. More detailed mathematical descriptions will be presented in a later part of the presentation agenda. This descriptive material is primarily intended to present an overview of the process, separate from the more detailed descriptive material, in the hope of minimizing the confusion which would otherwise result.

INPUT DATA(1)

- INITIATING EVENTS
  - EACH TYPE HAS INPUT
  - ANNUAL FREQUENCY
  - IMPACT ON SUPPORT SYSTEMS AND FRONT-LINE SYSTEMS

- SUPPORT SYSTEM UNAVAILABILITIES
  - OPERATING SYSTEMS
    - FAILURE CAN BE AN INITIATING EVENT
    - OTHERWISE FAILURE FREQUENCY AND ALLOWABLE OUTAGE TIME MUST BE SPECIFIED
  - STANDBY SYSTEMS
    - FAILURE ON DEMAND REQUIRED

Input Data(1)

This description will deal only with major groups of input data. It will not deal with input which controls program logic, output formats, or similar input required for a viable program.

Initiating Events

All initiating events to be considered in the analysis must be input. This input must include the frequency of the event and the impact of the event on plant equipment and systems. For example, a Station Blackout will disable all AC power systems in the plant, except uninterruptable systems, and this impact must be input to the calculation. This is accomplished by means of a vector array for the initiator, which indicates which systems have been disabled and which are not, in the form of ones and zeros corresponding to the disabled and functioning systems.

It is also necessary to segregate events which are similar in nature but different in impact. As an example, LOCA may occur as a result of a recirculation line break which eliminates the injection capability from one division of RHR. This type of event must be kept separate from a LOCA which does not have this impact.

In a similar manner, initiating events of a similar nature may involve failures or processes which require different specification of success criteria. A typical example would be the difference in requirements for responding to a large liquid break compared to a small steam break.

The specification of initiating events and their associated input data must be carefully chosen to reflect these considerations.

Support System Unavailabilities

Support systems are those systems which are required to allow operation of the front-line systems. This includes such items as AC power, DC power, compressed gas, cooling water, etc. Typically each of these must be broken up into subsystems in order to adequately describe the possible failure combinations which can result in differing consequences for the plant.

It is also necessary to distinguish between standby systems and operating systems. In the case of operating systems, failure could result in a plant trip in which case it would be classed as an initiating event. If, on the other hand, the plant could continue operating, it is necessary to consider the Allowable Outage Time (AOT) permitted by Technical Specifications. The reason for this is that an initiating event could occur during the AOT and the response of the plant could be quite sensitive to the specific sequence of events. In the case of the operating system the frequency of failure and AOT both must be input.

In the case of standby systems it is necessary to provide the failure on demand probability as input data.

INPUT DATA(2)

- o **FRONT LINE SYSTEM UNAVAILABILITIES**

  - FRONT-LINE SYSTEMS ARE THOSE WHICH CONTRIBUTE TO A FRONT-LINE FUNCTION.
  - SYSTEM UNAVAILABILITY MUST BE PROVIDED
  - THIS UNAVAILABILITY MUST NOT INCLUDE UNAVAILABILITY CAUSED BY SUPPORT SYSTEMS

- o **SUPPORT SYSTEM INTERDEPENDENCIES**

  - IN GENERAL, SUPPORT SYSTEMS DEPEND ON OTHER SUPPORT SYSTEMS FOR OPERATION
  - FIRST ORDER DEPENDENCIES MUST BE INPUT
  - HIGHER ORDER DEPENDENCIES ARE AUTOMATICALLY DERIVED

- o **FRONT LINE - SUPPORT SYSTEM DEPENDENCIES**

  - FRONT-LINE SYSTEMS REQUIRE PROPER OPERATION OF CERTAIN SUPPORT SYSTEMS
  - THESE DEPENDENCIES MUST BE INPUT

Front-line System Unavailabilities

Front-line systems are those systems which contribute to a front-line function. The front-line functions are those functions which are required to restore the plant to a safe and controllable condition after the occurrence of an initiating event. The front-line functions are the top level events in the front-line event trees. As an example, high pressure injection may be specified as a front-line function for which there may be success or failure. Contributing to success could be RCIC, HPCI, Feedwater pumps, or CRD pumps. The particular combination needed for success is dependent upon the initiating event in general, and, further, the availability of a system may depend on the initiating event. The input data required for the system is its inherent unavailability.

In the case of operating systems, availability is generally assumed unless disabled by the initiating event since such systems are needed for normal plant operation. As an example we would assume the feedwater system to be available in non-isolation transients which do not over burden the control system. Standby systems may be partially or completely out of service, however. The frequency of such conditions and the allowable outage time must be input data for the calculation.

The impact of the initiator is input separately as initiating event input as described in previously.

Support System Interdependencies

In general, each support system will depend on some other support system. For example, cooling water systems will need AC power from various systems to provide motive power and power for logic circuits. The system, in general, will also require DC power for its logic circuits as well. Further, the AC systems involved will require DC power for breaker functioning.

It is only necessary to provide as input the direct dependencies, however. The indirect dependencies are automatically derived by the program. The input is made in the form of a square matrix having a dimension equal to the number of support systems. The first order dependencies are simply put in as zeros or ones to indicate no dependence or dependence respectively.

Front-line - Support System Dependencies

The front-line systems similarly require the support systems to assure proper functioning. This data is input by means of a rectangular matrix having dimensions given by the number of front-line systems and the number of support systems. Dependency is similarly indicated by zeros or ones.

INPUT DATA(3)

- FUNCTIONAL FAULT TREE (FFT) LOGIC

  - FFTs ARE USED TO DETERMINE THE FAILURE PROBABILITY FOR FRONT-LINE FUNCTIONS

  - THE FFT USES INPUT FROM THE INITIATING EVENT, FRONT-LINE SYSTEM UNAVAILABILITIES, AND THE SUPPORT STATE

- SUPPORT STATE EVENT TREE (SSET) LOGIC

  - THE PURPOSE IS TO DEVELOP THE FREQUENCY OF ALL POSSIBLE COMBINATIONS OF INDEPENDENT SUPPORT SYSTEM FAILURES

  - A SEPARATE SSET IS SPECIFIED SEPARATELY FOR EACH INITIATING EVENT AND INCLUDES THE IMPACT OF THE INITIATOR ON SUPPORT SYSTEMS

  - THE SSET END POINTS BECOME INPUT TO THE FRONT-LINE FUNCTION EVENT TREES FOR THE SUPPORT STATE REPRESENTED BY THE ENDPOINT

- FRONT LINE FUNCTION EVENT TREE (FFET) LOGIC

  - THE FFET REPRESENTS THE SERIES OF PLANT ACTUATIONS OR OPERATOR ACTIONS REQUIRED TO AVOID SOME LEVEL OF DAMAGE TO THE PLANT

  - THE SEQUENCE OF TOP LEVEL EVENTS MUST BE DEFINED TO REPRESENT THE VARIOUS POSSIBLE SEQUENCES OF SUCCESSES OR FAILURES FOR THE INITIATOR AND SUPPORT STATE

Functional Fault Tree Logic

The front-line event tree top event functions are typically provided by combinations of various front line systems. These combinations may also include support systems as well. The logic which determines the availability of sufficient system capacity to provide success in the top event function must be input to the program. This input actually determines the structure and content of the functional fault tree (FFT) from which the failure probability of the top event function is calculated.

The dependence of front-line systems on the various support systems must be considered in determining the frequency of the failure condition for the top function in the functional fault tree. This is accomplished by means of input data to describe the dependencies of each front line system on the various support systems.

With this information, the actual failure probability for the top front-line function can be calculated for each support state corresponding to a given set of support system failures.

Support State Event Tree Logic

The support state event tree (SSET) has the purpose of developing all combinations of support system failures which occur as a consequence of random failures in a support system. Each initiating event has a separate support state event tree, and this tree is structured so that the impact of the initiating event on the support systems is accounted for.

In the calculation of support state frequencies, the fact that the various support systems may depend one another must be considered. This is accomplished by input of a support system dependency matrix which includes all of the first order (direct) dependencies. Second order and higher level dependencies are calculated by the program.

The end points of the support state event tree, in combination with its initiator frequency, are used as entry events for the front-line function event trees for the same initiator. An end point frequency of the SSET is simply the entry frequency for the front line event tree for that support state. End points which have an identical or equivalent set of failed support systems may be accumulated into a single support state in order to reduce the total number of support states and their corresponding front line function event trees.

The input data for support system unavailabilities are used to determine the end point frequencies of the Support State Event Tree.

Front-line Function Event Tree Logic

The successful response to an initiating event for a given support state will require success of a number of front-line functions to avoid plant damage. The sequencing of these top functions for the event tree should be done so that no top event depends on any subsequent event. This structure is necessary in order to avoid cut set algebra in the quantification of the event tree.

INPUT DATA(4)

o **SUCCESS CRITERIA**

- SUCCESS CRITERIA ARE USED TO DEFINE THE VARIOUS LEVELS OF PLANT DAMAGE WHICH CAN OCCUR

- THE DEGREE OF RESOLUTION IN PLANT DAMAGE STATES DETERMINE THE COMPLEXITY OF THE SUCCESS CRITERIA

- THE SPECIFICATION OF SUCCESS CRITERIA IS DERIVED FROM ANALYSIS OF THE ACCIDENT TRANSIENT

Input Data(4)

Success Criteria

The success criteria for the tree must be structured to reflect the top event functions for the tree so that varying degrees of plant damage can be determined.

The structure for the front-line function event tree is created by input logic to specify the top event functions and their sequence.

Success criteria are derived by a process external to the risk analysis process, namely transient analysis of the accident sequences. Success criteria, in general, consist of two parts. The first is a flow rate, for example, required to re-cover the core within a fixed period of time. The second part of the input is the time by which injection must be restored. A separate criterion may be specified for each increasing level of damage, for example as a function of the time at which injection is restored. As the complexity of the plant damage state definition increases, the complexity of the input must also increase.

In selecting the degree of resolution desired in plant damage states, the strategy given priority should be an attempt to allow segregation of those plant damage states for sequences having high uncertainty.

Increasing complexity of the plant damage resolution does not always imply a corresponding increase in the amount of transient analysis required since a single transient run can provide the latest time for operator action, which would be the flow rate and the time available to provide it. The information would come from an analysis of the accident transient in which it is determined how late the core can be reflooded before a certain level of damage is reached. In such a calculation several levels of damage may be defined including core melt and reactor vessel failure. If the program used also has containment analysis capability various success levels can be defined for the containment as well as for the fuel and reactor vessel. In more simple cases a success criterion may simply be a minimum flow to the reactor to assure adequate core cooling. In general success criteria answer one of two questions pertaining to top level functions in the front-line event trees. These are:

1. what is the minimum level of performance needed for success, or

2. what is the latest time for action of a certain magnitude?

The structure of the success criteria must be chosen to reflect the degree of resolution in plant damage states. Success criteria input will be in the form of logic statements which specify operator actions required and the corresponding time constraints.

INPUT DATA(5)

- o **PROCEDURAL GUIDANCE**

  - THE EOPs ARE THE BASIS FOR PROCEDURAL GUIDANCE
  - AN ACTION MAY ONLY BE CONSIDERED IF THE EOPs SPECIFY IT FOR THE SITUATION CONSIDERED
  - THE LEVEL OF SUCCESS OF THE ACTION IS DETERMINED BY THE TIME AVAILABLE FOR IT

- o **TIME REQUIRED FOR ACTION**

  - THIS INPUT SPECIFIES THE TIME AVAILABLE FOR AN ACTION
  - THE INFORMATION IS DERIVED FROM ANALYSIS OF THE ACCIDENT TRANSIENT

Procedural Guidance

Procedural guidance is based on the plant Emergency Operating Procedures (EOPs). This information is not used directly as input to the risk analysis but is used to constrain the specification of operator actions. The rules used by PP&L are:

1. If the EOPs specify an action, it should be included.

2. If the EOPs do not specify an action, it should not be included.

3. If the EOP specifies an action and the symptom which initiates it will be available, it should be assumed that the operator will execute it successfully if time is available.

4. The only limitation on operator success is time available, and the level of success is calculated by the HCR method.

This approach is essential if a useful evaluation of plant capability against severe accidents is to be derived. If one assumes some level of operator violation of procedural instructions, it then becomes necessary to consider why the violation occurred. There is no credible guidance for such a consideration, and, more important, the implication for the effectiveness of plant operators for response to severe accidents is extremely negative. Development of EOPs which have been examined for the severe accident sequences which characterize a plant and shown to be effective in combination with demonstrated high quality operator response in following EOPs is essential to demonstrating the ability to effectively exploit the plant's capabilities in response to a severe accident.

Time Required for Action

As an input to the HCR model it is necessary to specify the median or characteristic time required for the operator to take an action. In addition it is also required to specify the nature of the action; skill, rule or knowledge based in the HCR methodology. This information must be specified for each action where time is considered to be limiting to operator success.

THE PROCESS ELEMENTS

- o INITIATING EVENTS
- o SUPPORT STATE EVENT TREES
- o FUNCTIONAL FAULT TREES
- o FRONT-LINE FUNCTION EVENT TREES
- o SUCCESS CRITERIA
- o PLANT DAMAGE STATES

The Process Elements

This section will describe the six basic components which make up the overall risk analysis program. These components are:

1. Initiating Events
2. Support State Event Trees
3. Functional Fault Trees
4. Front-line Function Event Trees
5. Success Criteria
6. Plant Damage States

The purpose of these descriptions is to provide information on the function of these components of the calculation and to point out the important features of each.

INITIATING EVENTS

- o REQUIRE OPERATION OF FRONT-LINE SYSTEMS TO RESTORE PLANT STABILITY AND CONTROL

- o MUST BE DIFFERENTIATED ON THE BASIS OF

  - IMPACT ON PLANT SYSTEMS

  - FRONT-LINE SYSTEM OPERATION REQUIRED TO AVOID PLANT DAMAGE

- o NUMBER SHOULD BE MINIMIZED TO AVOID EXCESSIVE NUMBER OF ACCIDENT SEQUENCES

Initiating Events

The initiating event portion of the calculation is actually only input data necessary to quantify the initiator frequency for a given support state event tree. The initiating event frequency is used as the initiating frequency of the support state event tree for that initiator.

The initiator may have an impact on support system and front-line system availability. This impact is input as a part of the initiating event definition. Since each initiating event will have a support state event tree of its own which will define several support states, each of which has its own front-line event tree, it is important to keep the number of initiating events at a minimum. The primary incentive for this is to limit the final number of front-line event tree end points. The number can become extremely large. This tendency also is severely aggravated by increasing the resolution detail in the plant damage spectrum.

In selecting the set of initiating events it is important to separate initiators which have:

1. a different impact on support or front-line systems or

2. require different combinations of equipment functioning or operator actions to avoid or minimize plant damage.

Typically the basic initiating event types identified are:

1. Transients
2. LOCAs
3. Station Blackout
4. ATWS
5. Combinations of these

Actually, Station Blackout and ATWS are special cases of the transient and LOCA classes and the initiating event input need not identify them separately. They may be derived by accumulation of appropriate end points of the transient initiators and LOCA initiators. They are kept segregated because these events are generally perceived to represent the most serious threat of severe plant damage.

There are, in general, several transient initiators and LOCA types. In addition, there are special initiators which can be caused by operating or standby system failures which can either directly or indirectly lead to a requirement for shutdown. In many cases such events can be shown to be equivalent to a specific support state event tree end point. In such cases it is highly advantageous to combine such an initiator with the appropriate support state end point.

SUPPORT STATE EVENT TREES

- o THE SUPPORT STATE EVENT TREE DETERMINES THE FREQUENCY OF VARIOUS COMBINATIONS OF SUPPORT SYSTEM FAILURES, INDEPENDENT AND AS A CONSEQUENCE OF THE INITIATOR

- o THE SSET END POINTS ARE USED AS INITIATORS FOR THE FRONT-LINE FUNCTION EVENT TREES

- o THE IDCOR REPRESENTATION OF SUPPORT SYSTEMS LACKS SUFFICIENT RESOLUTION

- o SUPPORT SYSTEMS MUST BE RESOLVED AT LEAST TO THE DIVISION AND CHANNEL LEVEL

- o SUFFICIENT RESOLUTION WILL RESULT IN AN EXTREMELY LARGE NUMBER OF SUPPORT STATES

- o THE NUMBER CAN BE REDUCED BY:

  - COLLAPSING SIMILAR SUPPORT STATES

  - USING A LOW FREQUENCY CUT-OFF

Support State Event Trees

The support state event trees defined by the IDCOR BWR Individual Plant Evaluation Methodology are extremely coarse representations of the BWR plant support systems. As a result, it is difficult to represent the two division, four channel nature of BWR4 plant systems. For this reason PP&L has expanded the definition of the support state structure to permit this level of detail in the support states for the plant. As a result of this expansion, the complexity of the support state event tree is enormously increased.

Because of this increase in complexity, we have developed a matrix algebra method for derivation and definition of the support states. Event trees are not actually used although it is convenient for conceptual purposes to think in terms of event trees.

The use of the support state concept has an enormous advantage over the more commonly used event sequence fault tree method. This advantageous characteristic is the property of segregating the support system failures caused by the initiating event and random failures into groups having identical characteristics for propagation through the front-line function event trees. This property permits numerical quantification of the front-line function event tree without any requirement for cut set manipulations. For this reason the end points of the front-line function event tree represent fully defined states of the plant rather than combinations of differing equipment failure combinations. This definitive knowledge of the plant state then permits propagation of each front-line function event tree end point into a full spectrum of the potential final plant damage states.

The input to the support state event tree is the frequency of a given initiating event. The support state event tree partitions this frequency into the various support states for that initiator. The end points of the support state event tree are then frequencies for that support state and initiating event which acts as the input to the corresponding front-line function event tree.

It is seen that each initiator therefore has many front-line function event trees, one for each support state defined. Since the number of support systems which must be considered is large, and a support state is simply a combination of support system failures, a very large number of support states is possible. In order to control the magnitude of the computational burden and the magnitude of the output information, it is important to collapse support states together which are functionally equivalent. This process is not expected to result in a sufficient reduction, however, and a cutoff frequency for support states to be considered may be specified.

FUNCTIONAL FAULT TREES

- THESE ARE GENERALLY SYSTEM BASED FAULT TREES WHICH DETERMINE THE UNAVAILABILITY OF FRONT-LINE FUNCTIONS. THEY QUANTIFY THE TOP EVENTS IN THE FFETs

- THE UNAVAILABILITY INCLUDES THE IMPACT OF SUPPORT SYSTEMS AND INITIATORS. THERE IS A SET OF FFTs FOR EVERY INITIATOR-SUPPORT STATE COMBINATION

- THE EQUIPMENT REQUIRED FOR SUCCESS IN THE TOP EVENT OF THE FFET IS DEPENDENT ON THE INITIATING EVENT, IN GENERAL

Functional Fault Trees

The purpose of the functional fault trees (FFTs) is to develop the unavailability for the front-line functions. In general this unavailability is different for every initiator and every support state. The front-line functions considered in the front-line function event trees are:

1. Reactivity control
2. Reactor pressure control
3. High pressure injection
4. Reactor depressurization
5. Low pressure injection
6. Containment heat removal
7. Core cooling given containment failure

In the case of loss of AC power, further consideration may be given to loss of diesel generators or recovery of off-site or on-site AC power sources. In the case of LOCA it is necessary to consider the functioning of the pressure suppression system. Finally, in the case of ATWS it is necessary to consider the condensate pumps, HPCI and RCIC/CRD separately, SLCS, and manual rod insertion. In general, the event trees must be structured to consider the functioning of equipment needed and the time available to respond to the initiating event.

The purpose of the FFT is to determine the availability of equipment to perform each of these top level functions. The calculation must consider:

1. The inherent unavailability of the equipment,

2. the impact of the initiating event on equipment availability

3. the support state being considered.

This determination is generally quite straight forward. The fault tree elements are complete systems or divisions or channels of systems.

FRONT-LINE FUNCTION EVENT TREES

- FOR A GIVEN INITIATOR AND SUPPORT STATE THE FFET DETERMINES ALL POSSIBLE EVENT SEQUENCES

- SUPPLEMENTAL LOGIC IS USED TO DISPOSITION EACH EVENT SEQUENCE TO THE COMPLETE SPECTRUM OF FINAL PLANT DAMAGE STATES

- THIS IS FEASIBLE SINCE THE INITIATOR, THE SUPPORT STATE, AND THE EVENT SEQUENCE FULLY DETERMINE THE PLANT STATUS SO THAT THE FINAL OUTCOME IS EASILY CALCULATED

- FFET END POINTS ARE NOT BINNED, BUT ARE INDIVIDUALLY DISPOSITIONED

Front-line Function Event Tree

The front-line function event tree is most effective when it is structured to reflect the effect of the initiator on support and front-line systems and the support state to be considered. The structuring of a tree is done by considering the nature of the initiating event and support system failures and the resulting guidance to the operator offered by the EOPs. The tree becomes primarily a reflection of the actions he is instructed to take and considers his success or failure at executing these steps. If the problem is failure of the equipment called for, consideration is given to the possibilities for correction of the failure. If time is available, the corrective action is considered. In all cases where time for an action is limited, the distribution of execution times, determined by the HCR method, is used to calculate the level of success or failure in accordance with the success criteria structure which has been devised. This process then allows determination of the actual spectrum of success (or failure) to be expected.

We believe that this approach is very important because very often the probability of failure falls off very rapidly in time, and, even though there may be a fairly high probability of sustaining some degree of damage, the probability of severe damage is low. This consideration is the primary motivation that caused PP&L to originally consider the possibility of arresting core damage before reactor vessel failure.

Using the typical level 1 approach to the structuring of front-line function event trees, the tree ends when some form of damage to the plant has occurred. In general, these trees do not consider containment status unless there is no threat to the core before containment failure. This approach does not provide a complete picture of plant damage, and for this reason we have extended the typical level 1 methodology. This is accomplished by use of the knowledge of event sequence timing which is inherent in the end point determination through the success criteria and the knowledge of the complete plant status which comes from the support state and the front-line failures which have occurred. With this information, the time available for further operator corrective or repair actions may be determined, and the level of success in executing them determined.

In the formulation of this process for a computer program, the event tree concept is not actually used, but logic equations which are their equivalent are developed and used for the quantification. It is useful nevertheless to think of the process in terms of event trees.

SUCCESS CRITERIA

- SUCCESS CRITERIA ARE DERIVED FROM TRANSIENT ANALYSIS OF THE ACCIDENT SEQUENCE

- IF TIME FOR AN ACTION IS CRITICAL, THE TIME DETERMINES THE DEGREE OF PLANT DAMAGE

- IN MANY CASES A SINGLE CALCULATION MAY PROVIDE A SERIES OF TIME CONSTRAINTS FOR ESCALATING PLANT DAMAGE STATES

- IN SOME CASES WHERE MULTIPLE ACTIONS ARE INVOLVED SIMPLIFICATIONS MAY BE REQUIRED

Success Criteria

Success criteria are primarily an expression of the results of performing an analysis of accident transients. The nature and extent of transient analysis required depends, to some extent, on the complexity of the success criteria devised. In the case of a simple criterion for avoidance of reactor vessel failure, it would only be necessary to perform an analysis of the event to determine the latest time at which make up flow to the reactor could be restored in order to maintain a coolable geometry for the core and core debris in the reactor. If it is desired to consider the challenge to the containment should makeup not be restored in time, it is necessary to extend the calculation through reactor vessel failure and to determine the influence of core debris challenges to containment integrity in terms of non-condensible gas generation or attack on critical containment components. For each of these, criteria must be devised to determine the degree of gas generation or critical component degradation which can be tolerated. In combination with these criteria, actions must be defined which will terminate the progress of the accident transient.

In general a single transient analysis can provide an entire sequence of success criteria because a single operator action of the proper type will terminate the accident progression and avoid further damage to the plant. In these cases, it is simply a matter of performing an analysis of the entire accident sequence and determining the various times at which such actions would terminate the sequence and therefore determine the level of plant damage.

In some cases, however, it may be found that a single action does not suffice to terminate the accident progression. In such cases a multiplicity of accident trajectories may be possible which reflect the influence of the timing and sequence of the actions. In other cases, the timing of an accident sequence may be influenced by the time of occurrence of a failure in that the decay heat level is monotonically decreasing with time. In such cases it may be necessary to select somewhat conservative criteria to reduce the analysis burden to an acceptable level. Careful consideration must be given to the consequence of such simplifications, both in terms of characterizing the dominant accident sequences for the plant and the process of developing optimized EOPs.

A very important consideration in the PP&L approach to developing success criteria structures is the desire to isolate phenomena or calculations which have a relatively high degree of uncertainty associated with them. The purpose is to permit such events to be segregated from the general population of accident sequences so that the relatively high degree of uncertainty is not improperly attributed to the entire population of accident sequences.

PLANT DAMAGE STATES

- o IN DEFINING PLANT DAMAGE STATES THE OBJECTIVES ARE:
  - TO IDENTIFY THE TIME OF CORE DAMAGE AND CONTAINMENT FAILURE
  - TO DISTINGUISH BETWEEN DIFFERENT LEVELS OF SEVERITY AND DAMAGE TYPES
  - TO SEGREGATE HIGH UNCERTAINTY SEQUENCES FROM LOW UNCERTAINTY SEQUENCES

- o THIS APPROACH PERMITS THE INFLUENCE OF HIGH UNCERTAINTY EVENTS OR PHENOMENA TO BE BOUNDED

- o THIS APPROACH ALSO PERMITS THE INFLUENCE OF VARIATIONS IN RESPONSE TIMES TO BE PROPERLY REPRESENTED

Plant Damage States

The definition of a Plant Damage State (PDS) is achieved by analysis of the accident transient. The results are translated into a requirement for a certain action of a specific magnitude prior to a specific time in order to avoid exceeding a specific level and type of plant damage. The purpose of defining a high resolution system for accident classification is to:

1. identify the time of core damage and containment failure,

2. distinguish between differing levels of severity of the sequence and between different types of damage, and

3. permit segregation of sequences having high uncertainty, mild consequence, or severe consequence.

We believe that it is important to do this in order to establish proper and credible bounds on the importance of such phenomena as core damage progression, reactor vessel failure, core concrete interactions, and core debris attack on critical containment components.

The Susquehanna IPE accident class definitions established this capability in part, but failed to provide adequate resolution on the extent of core damage and the severity of core debris challenge to containment integrity. We believe that it is necessary to correct this deficiency in future risk assessment work.

In the case of core damage progression, we wish to identify not only the time at which the damage initiates but also to identify the nature of the damage and the extent of it.

There are two basic types of core damage to be considered:

1. mechanical clad damage and
2. metal-water reaction damage.

The first of these results from severe or repeated power transients or unstable limit cycle operation such as might be encountered in ATWS events, particularly those involving inadequate crew response or equipment failure. This form of damage involves mechanical interactions between the fuel and the clad with consequential high stress and loss of clad integrity. These events are believed to be capable of releasing very large fractions of the noble gas inventory and considerable fractions of the volatile fission product inventories, particularly in the case of severe power cycling continuing for an extended time period. The uncertainties associated with this type of fuel damage are very large, but the potentially large release of fission products has a significant impact on EOP structure to minimize the potential consequences of such releases. For this reason we wish to segregate and, if possible, quantify the severity of this type of damage in any analysis of the impact of uncertainties on optimal EOP development, crew training, and mitigating modifications to plant equipment.

The second type of damage is a consequence of inadequate core cooling and results from high clad temperatures leading to metal-water reactions with the clad and other components. We suspect that quantification on the basis of the

extent of the reaction is probably not an effective approach. The reason is that in the absence of adequate core cooling, the decay heat and the initial chemical reaction energy results in a gradual increase in clad temperature, but with relatively small reaction fractions. Once this temperature reaches a threshold value ($\sim 2500^{\circ}F$) the rate of reaction and the temperature rise rapidly, and the conditional probability of inhibiting this process by recovered water injection capability is relatively small. Once started, the reaction goes from a small fraction of the metal involved to a plateau value which is achieved when the metallic materials have relocated and been quenched in cooler regions of the reactor vessel.

This process does not occur across the entire core simultaneously, however, but proceeds by regions of similar and increasingly lower power generation history. This process, furthermore is characterized by the following sequence of events:

1. melting of the control rod materials and relocation of the material to the region of the core support plate or lower plenum,

2. auto-catalytic reaction of the channel box zirconium with extensive metal water reaction, melting, and melt relocation to the core support plate or lower plenum,

3. auto-catalytic reaction of the clad zirconium and relocation of molten zirconium to the lower regions of the core,

4. eventual loss of columnar integrity of the UO_2 rods and collapse into a rubble bed supported by the core support plate,

5. failure of the core support plate and associated structure by high temperature caused by decay heat release and stored energy of the unmolten fuel material, and

6. relocation of the fuel material to the lower plenum and quench by the remaining water inventory.

In this progression, the debris bed in the lower plenum retains a coolable geometry and does not threaten the integrity of the lower head until lower head dry out and heat up of the debris to a temperature which no longer permits quench even with water addition.

This process takes a considerable period of time before the lower head integrity is threatened, and an even longer time before complete relocation of all core material to the lower plenum. There is a potential hazard of initiating makeup flow to the reactor vessel during this process. That is the possibility of reflooding fuel bundles after the control rod material has relocated. The fact that this process occurs over an extended time period means that the conditional probability of core reflood of unrodded bundles may not be low.

We believe, based on fuel bundle reflood experiments performed at Argonne, that the reflood process will cause the high temperature fuel to shatter and form a non-critical debris bed. This view, however, is subject to challenge and represents a source of uncertainty in the accident progression and the severity of the energy input into the reactor vessel. Because of this, we do believe that it is important to track the fraction of fuel bundles that have

undergone control rod material relocation. Similarly we must also track what fraction of the unrodded bundles have reached a temperature sufficient to cause collapse of the fuel rods into a non-critical geometry. In this manner, we can characterize the potential magnitude of the threat from unrodded bundles and thereby bound the magnitude of the potential threat. The BWRSAR code makes tracking the status of the core very simple in this regard.

In the case of reactor vessel failure, not all core debris material is believed to relocate to the pedestal room floor immediately on failure. In fact the BWRSAR models indicate that the initial pour of material from the vessel will be the molten metallic components of the debris which have a considerably lower temperature than the UO_2 melting point, but have a higher melt temperature than the reactor vessel material. Initial failure is believed likely to be at instrument tube penetrations, the drain penetration, or possibly CRD penetrations rather than general high temperature failure of the lower head. In this case, the release of material to the pedestal room floor is a relatively long term process involving a considerable change in debris content (and decay heat content) with time. The initial pour does have sufficient temperature and energy content to initiate release of water vapor from the concrete which in turn reacts with the metals of the pour releasing considerable energy. This process can potentially become entirely self-driven and can result in considerable damage to containment structures and release of large quantities of non-condensible gases.

On the positive side, the presence of water on the drywell floor should quench the molten metal and prevent this reaction, or initiation of injection of water into the vessel or the drywell could stop the progression of the reaction early in the process if the metal is released over a reasonably long time period. In all cases, injection of water into the vessel should stabilize a large fraction of the remaining fuel and greatly reduce the challenge from decay heat driven attack of the drywell concrete.

For this reason, we wish to characterize the vessel failure process on the basis of the amount of metal and UO_2 released from the vessel before injection is restored and the process of release from the vessel terminated.

Finally, we wish to discriminate between the nature of the various challenges to containment integrity. The classifications which we believe should be considered are:

1. wetwell venting,

2. attack on the LOCA vent tubes or SRV discharge line penetrations which could cause pool bypass and defeat of pressure suppression,

3. direct attack on the drywell liner which could cause an unscrubbed release from the containment,

4. high drywell temperature combined with pressure causing drywell failure as a consequence of uncooled core debris on the drywell floor, and

5. simple over pressure failure of the containment caused by high suppression pool temperature, large quantities of non-condensible gases generated by the accident, or a combination of the two.

It may be desirable to split item 5 above into cases, with and without non-condensible gas generation. We believe the major challenges to containment will be simple inadequate decay heat removal and that there is relatively small uncertainty in such sequences. Similarly, we believe venting will be a dominating mode of loss of containment integrity if venting capability is provided. Once again, the uncertainty is small and also, the consequences relatively minor.

In the case of the other categories, however, the uncertainties are high primarily due to the phenomena involved which can also strongly influence the accident progression. Nevertheless, we believe the relative frequency of events of this type will be low so that we wish to segregate them, both because of high uncertainties and high potential consequence.

We believe that the original classification scheme devised for the Susquehanna IPE will need to be extended from a three letter code to a four or five letter code to provide the degree of resolution desired. The decision on what specifically shall be provided will be based on examination of a number of BWRSAR and CONTAIN runs. This decision will only lock in the nature of the characterization and the complexity of the class identification scheme. The actual quantification of event severity within the scheme would be user selected.

EXTERNAL PROCESSES

- o THERMODYNAMIC END POINT CALCULATIONS
  - SIMPLE BOIL DOWN TIME
  - SIMPLE DEPRESSURIZATION
  - SIMPLE CONTAINMENT PRESSURE RISE
- o IN-HOUSE CODES
  - STATION BLACKOUT
  - ATWS (CHEXAL-LAYMAN POWER)
- o RETRAN (LIMITED USE)
- o SIMULATE
  - ATWS POWER
  - SINGLE VS. DOUBLE SDV FAILURE
  - ATWS REFLOOD RATE LIMIT (INTENDED)
- o MAAP (NO LONGER USED)
- o BWRSAR (ORNL)
  - EVENTS WITH DECAY HEAT ONLY
  - CORE DAMAGE EVENTS
  - CONTAINMENT TRANSIENT WITHOUT REACTOR VESSEL FAILURE
- o SAMBA (EPRI PRE-RELEASE)
  - ATWS TRANSIENTS
  - STABILITY LIMITS
  - LIMIT CYCLE OPERATION
- o CONTAIN (REQUESTED FROM NRC/ORNL)
  - CONTAINMENT TRANSIENTS WITH REACTOR VESSEL FAILURE

External Processes

The phrase, external processes, is used to refer to the various types of transient analysis calculations. Several types of analysis are included in this category. Th e are:

1. the BWRSAR program,

2. an in-house program for analysis of ATWS transients,

3. simple end point thermodynamics calculations, and

4. the CONTAIN program.

We have in addition used an in-house program for Station Blackout, RETRAN, SIMULATE, SAMBA, and, in earlier work, MAAP.

These programs and analysis methods are used to determine constraints on timing, flow rates, and pressure required to avoid a specific form of damage or to limit the accident transient not to exceed a specific level of plant damage. Typically end point thermodynamics calculations are used to calculate boildown time for a subcritical reactor with makeup inadequate to maintain level. This type of calculation can also be used to calculate the capability for reactor pressure reduction without reaching a condition of inadequate core cooling.

The BWRSAR program is used to track boildown and depressurization transients in more detail than is possible with end point thermodynamics analysis. The more important use of BWRSAR is to track core damage progression sequences and to track containment pressurization sequences up to the point of reactor vessel failure. The program cannot follow a sequence beyond reactor vessel failure since it does not have core concrete interaction models. This program is extremely important in allowing success criteria having considerable resolution to be developed. For example the following characteristics of an accident sequence can be tracked:

1. the extent of metal water reaction,

2. the number of control rods which have melted and relocated,

3. the number of fuel bundles which have collapsed to the core support plate,

4. the number of free standing fuel bundles which are unrodded.

5. the extent of core support plate local failure with fuel debris relocation to the lower plenum,

6. the time of reactor vessel failure,

7. the rate and composition of core debris pours from the reactor vessel, and

8. the rate of containment pressure increase up to the time or reactor vessel failure.

These characteristics of an accident transient which can be quantitatively extracted from a BWRSAR calculation may be used to develop time quantifications for success criteria. It is easily seen that there is wide latitude in the definition possible in the plant damage states. It is also seen that a single transient calculation can define multiple action times for termination of a given type of damage progression, such as core damage or containment pressure buildup. After such an action, the plant may not be stabilized and further actions may be required to establish a stable condition. In such cases, multiple transient cases may be required, but engineering judgement and acceptance of mild conservatisms may greatly reduce the quantity of calculations required.

PP&L has developed an in-house transient calculation based on the Chexal-Layman correlation for power in an ATWS event which allows rapid and efficient evaluation of alternative operator response strategies in response to the event and time limits for operator success. In addition to calculations of this type, very simple calculations can be made assuming linear power reductions from a stable ATWS power level at full makeup flow to decay heat level power in cases where boron is injected or control rods are manually inserted using CRD drive flow. Calculations of these two kinds allow a reasonably reliable assessment of a wide variety of ATWS response strategies and time limits for successful operator action.

Finally, CONTAIN calculations allow determination of time constraints for mitigating actions to arrest damage resulting from core debris falling to the drywell floor. PP&L has not yet performed calculations of this type, but we are certain that development of a range of quantified success criteria to terminate such accident progressions should be possible.

OBJECTIVES OF TRANSIENT ANALYSIS

DETERMINE:

- TIME AVAILABLE FOR OPERATOR ACTION
- MINIMUM ALLOWABLE FLOW RATES
- REQUIRED DEPRESSURIZATION RATES
- INFLUENCE OF BORON MIXING EFFICIENCY
- MASS LOSS FOR RPV BLOWDOWN
- POOL MASS AND TEMPERATURE INCREASE FOR RPV BLOWDOWN
- CONTAINMENT PRESSURE TRANSIENT
- STAGES OF CORE DAMAGE PROGRESSION
- TIME OF REACTOR VESSEL FAILURE
- STAGES OF CORE DEBRIS RELEASE TO THE DRYWELL FLOOR
- TIME OF CONTAINMENT FAILURE

Objectives of Transient Analysis

The basic purpose of transient analysis is to determine the time available for the operator to take an action to avoid or minimize plant damage. The transient analysis must also determine the necessary level of equipment performance to result in a successful operator action.

Much of the analysis of this type is directed to a determination of the minimum level of make up flow to the vessel required to meet one of the various success criteria that may be specified. In some cases, the rate of depressurization and water mass loss from the reactor vessel before low pressure pumps can function is an important issue for resolution. In the case of ATWS sequences, the influence of SLCS initiation time and boron mixing efficiency is important in establishing the probability of success in avoiding plant damage.

Finally, if plant damage cannot be avoided, it is important to establish the timing for occurrence of various levels of plant damage in order to determine the probability of terminating the damage progression at various stages in the transient. The information required from the various analyses performed is strongly dependent on the success criteria structure selected.

TRANSIENT CALCULATIONS(1)

GENERAL FOR ALL INITIATORS

- o BOILDOWN TIME AFTER LOSS OF INJECTION
- o DEPRESSURIZATION WITH LIMITED INJECTION
- o MINIMUM INJECTION RATE AND LATEST TIME FOR INJECTION TO AVOID:
  - FUEL DAMAGE
  - CONTROL ROD RELOCATION
  - FUEL RELOCATION TO CORE SUPPORT PLATE
  - CORE SUPPORT PLATE FAILURE
  - LOWER PLENUM DRY OUT
- o GENERATION RATE OF NON-CONDENSIBLES
  - METAL/WATER REACTION
  - CORE/CONCRETE INTERACTION
- o CONTAINMENT PRESSURE TRANSIENT
  - NON-CONDENSIBLES IN DRYWELL
  - NON-CONDENSIBLES IN WETWELL
  - NON-CONDENSIBLES SPLIT
- o CORE DEBRIS ATTACK ON CRITICAL COMPONENTS
  - LOCA DOWNCOMER VENTS
  - SRV DISCHARGE LINE PENETRATIONS
  - DRYWELL LINER

Transient Calculations(1)

In this viewgraph the type of information that must be generated in general for all initiators is listed. The items listed are not intended to represent the specific information needed, but only the type of information that can be extracted from the analyses performed. The specific information required will be related to the parameters listed, but will depend on the success criteria selected.

A given set of transient runs could in principle allow several differing sets of success criteria to be examined. These different sets, for example, could simply specify differing degrees of resolution in the plant damage states. At this time we have not performed a sufficient range of analyses to select the optimum set of success criteria. It is most effective to select plant damage levels that will be reasonably well separated in time so that the rate of reduction in probability from one level to the next is depicted in reasonably uniform steps.

TRANSIENT CALCULATIONS(2)

ATWS

- o LATEST TIME FOR BORON INJECTION
  - ONE PUMP
  - TWO PUMPS
- o LATEST TIME FOR HPCI BYPASS ACTIONS
- o MINIMUM FLOW TO AVOID FUEL DAMAGE
  - WITH BORON INJECTION
  - WITHOUT BORON INJECTION
- o TIME AND FLOW RATE FOR INCREASING POOL MASS

STATION BLACKOUT

- o DRYWELL TEMPERATURE AND PRESSURE
- o HPCI/RCIC ROOM TEMPERATURE
- o CONTROL ROOM TEMPERATURE
- o TIME AND FLOW RATE FOR INCREASING POOL MASS

Transient Calculations(2)

For ATWS and Station Blackout information unique to the initiator is required. In the case of ATWS, the ability to determine the latest time for boron injection is important. In the case that timely boron injection is not available, we need to know how much time is available to perform bypass operations on HPCI logic to avoid loss of the system due to high suppression pool temperatures or high containment back pressure.

The Chexal-Layman correlation has made possible the use of very simple global transient calculations for the reactor vessel thermodynamic state which permits a great deal of the necessary information to be generated. There are deficiencies in the treatment of boron injection and core inlet subcooling effects in this correlation which must be examined, however.

A potentially important strategy for extending the time before containment failure pressure is reached in ATWS transients is to add water to the suppression pool. This action has the effect of drastically reducing the rate of pressure increase and provides much more time for operator actions.

In the case of Station Blackout several important pieces of equipment are subjected to transient conditions which will ultimately cause their loss. These are drywell, HPCI/RCIC, and control room temperatures. These analyses have been done using very simple single temperature models for the room atmosphere. These crude models indicate that sufficient time to failure is available that such losses have only a small impact.

The addition of water to the suppression pool is also a potentially effective strategy for slowing containment pressure increase for Station Blackout sequences as well as ATWS sequences. These calculations require a reliable containment program.

OUTPUT

- o THE BASIC CALCULATION OUTPUT SHALL BE THE FREQUENCY FOR PLANT DAMAGE STATE FOR EACH:
  - INITIATOR
  - SUPPORT STATE
  - FFET END POINT

- o FOR USEFUL DISPLAY OF THE OUTPUT A VARIETY OF EDIT FORMATS ARE REQUIRED:
  - GLOBAL DAMAGE FREQUENCY
  - TOTAL DAMAGE FREQUENCY BY PLANT DAMAGE STATE
  - TOTAL DAMAGE FREQUENCY FOR EACH PLANT DAMAGE STATE BY INITIATOR
  - ETC.

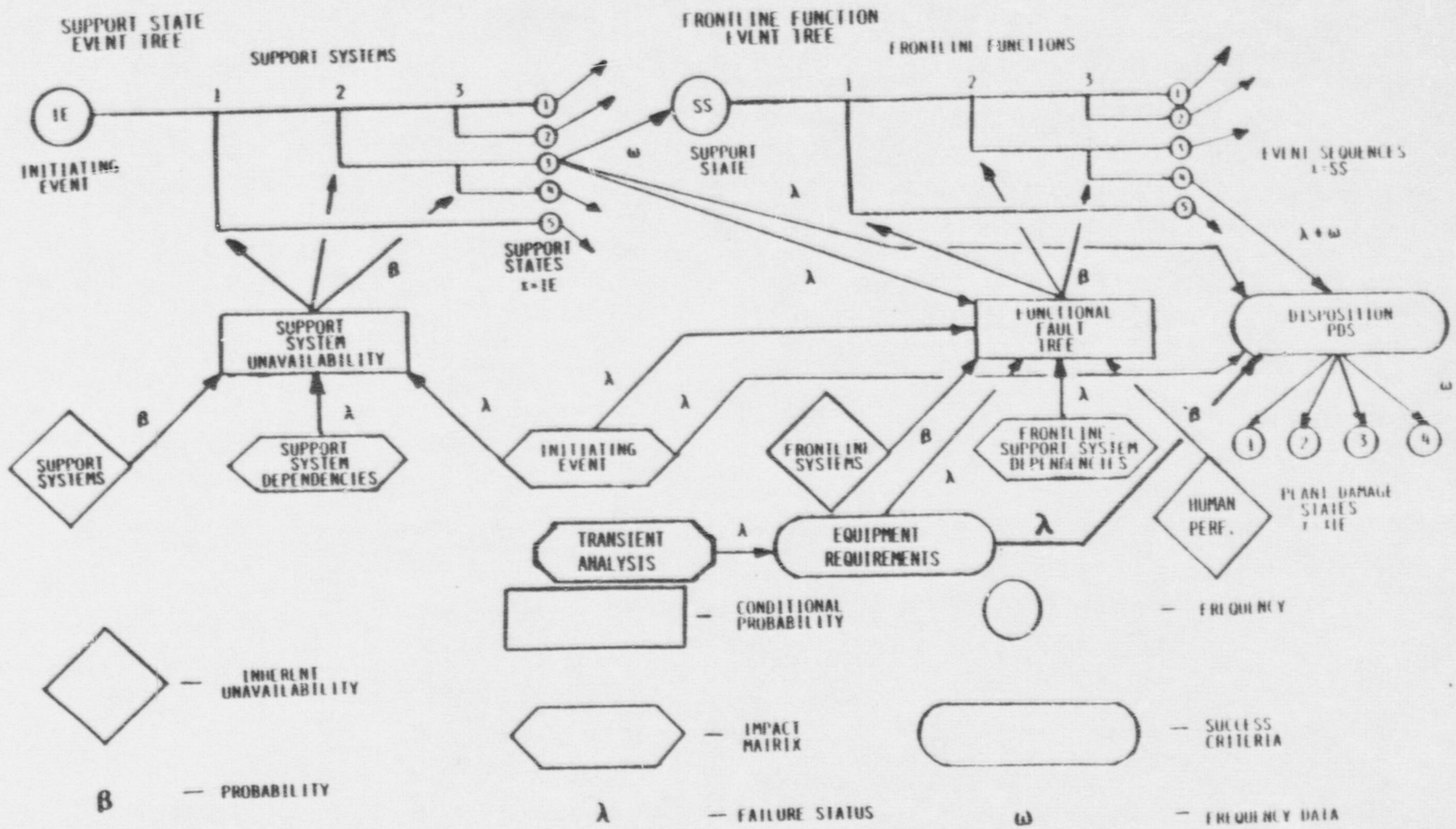
Output

The raw output of the calculation shall be organized by initiator and support state. For each initiator/support state combination the output shall be a set of plant damage states and their corresponding frequency for each end point of the front-line function event tree - that is for each combination of front-line function failures.

This type of output data organization implies an extremely large volume of output data with an extremely wide range of frequencies. Except for possible truncation or simplification of the support state combinations considered, no cut-off frequency will be used to drop sequences or plant damage state contributions to a sequence. All data will be retained. In order to organize the output in a more useful form, a variety of edit routines shall be provided. These routines shall permit grouping of the sequences and plant damage states into a variety of formats. For example, a display of all plant damage states for all support states combined for a given initiator would be a potentially useful form of edit. Similarly an edit of all plant damage states for all initiators and their support states combined would also be of interest. A variety of edits of this type would be provided.

Ultimately, we believe it will be feasible to provide distribution functions for each of the input parameters to the calculation and to create a Monte Carlo sampler and driver for the risk analysis calculation. For this type of calculation special edit programs would be required to collapse the output data into the desired format and to provide statistical distribution parameters for each element of the output format.

SUPPORT STATE METHOD INFORMATION FLOW



Support State Method Information Flow

The view graph diagram is intended to demonstrate the overall flow of information through the calculation process. The input data for the calculation are.

1. initiating events, frequency and impact on equipment,

2. support system unavailabilities and interdependencies,

3. front-line system inherent unavailabilities and dependencies, and

4. success criteria.

It should be noted that the actual success criteria may have to be derived before constructing the FFET. As an example, actions which are not effective should not be incorporated into the FFET. The effectiveness of a potential action can only be determined by an analysis of the transient in question. Thus, to some extent, development of the FFET and development of success criteria must go hand in hand. We have ignored this consideration in this diagram.

With the above input information and the SSETs, FFETs and FFTs, we may start the calculation. First, the initiator is partitioned by the SSET into Support State frequencies. These Support States represent support system failure combinations which have an equivalent effect on the front-line systems. The branching probabilities in the SSET are simply determined on the basis of the inherent unavailabilities of the support systems, the interdependencies of the support system, and the influence of the initiator on the various support systems. These calculations are very straightforward, but the selection of support system failure combinations having an equivalent impact on front-line systems does involve a knowledge of the Front-line System dependencies on the Support Systems. This link is also not shown on this diagram. The fact that a number of SSET endpoints may have an equivalent effect on front-line systems is not shown in the diagram. This process involves binning of such equivalent endpoints and their frequencies. This binning process does not introduce any error or obscure the nature of the sequence.

In the computer program based calculation an SSET is not actually used due to the very large number of support systems. The computation is actually performed on the basis of Boolean logic which is much more efficient than an event tree.

Nevertheless, the SSET is a useful conceptual device which clearly shows the partitioning of the initiating event frequency into an array of support state frequencies. For each support state, then, a Front-line Function event tree is prepared which takes into account equipment which is disabled by the initiating event and the Support System failures. The Support State frequency is then partitioned by the FFET into a series of event sequences representing various success and failure combinations of the Front-line Functions. The determination of the timing and nature of plant damage resulting from one of these sequences must be determined from the success criteria. The event tree, and the success criteria associated with it, however, only will determine the nature of the initial damage to the plant, initial core damage or containment

failure. They will not and, as a practical matter, cannot determine the eventual nature of plant damage as the event sequence progresses beyond the events covered by the FFET.

There are, however, additional success criteria provided which are structured to determine the spectrum of final plant damage states for each endpoint of the FFET. These additional success criteria are based on the initiating event and the combination of support system and front-line function failures which define the sequence. With this information, the various probabilities for equipment recovery to terminate the accident progression can be calculated which then permits allocating the frequency of the endpoint in question among the possible plant damage states. This process cannot readily be accomplished by an extension of the FFET since the logic involved is too complex for the simple event tree structure (see Appendix 6).

The flow of information in the risk analysis process is quite complex, but if taken a step at a time is actually very straightforward. The diagram can be useful, within the limits of the cautions stated above, in developing a comfortable understanding of the workings of the Support State method.

INITIATING EVENTS

DEVELOPMENT OF INITIATING EVENTS

DEFINITION OF INITIATING EVENTS

TYPES OF INITIATING EVENTS CONSIDERED

FREQUENCY OF INITIATING EVENTS

Develop of Initiating Events

Initiating events are discussed in this section. The discussion includes a definition of initiating events, the types of initiating events considered, and the estimation of initiating event frequency.

DEFINITION OF INITIATING EVENT

AN INITIATING EVENT IS THE OCCURRENCE OF SOME EVENT THAT PLACES THE PLANT IN A STATE THAT REQUIRES THE ACTUATION OF STANDBY SYSTEMS TO MITIGATE ITS EFFECT.

Definition of Initiating Events

It has been demonstrated that steady state operation of a nuclear power plant represents an insignificant safety hazard to the general public. Therefore, to create a hazard, the plant must at least be perturbed away from its steady state operating point. The plant is designed, however to adjust process parameters and thus compensate for the majority of the perturbations experienced by the plant. On occasion the deviation from the steady state is so severe that the plant control systems are ineffective at compensation. These situations require prompt intervention by operators or standby systems to bring the plant back to a steady state condition; usually plant shutdown. These severe deviations from the steady state are defined to be initiating events. The initiating events may be the direct result of failure or misoperation of plant equipment which results in a prompt automatic or manual shutdown of the reactor or an indirect result of failure or misoperation of plant equipment that cause an orderly forced plant shutdown.

TYPES OF INITIATING EVENTS

TRANSIENT INITIATORS

- MANUAL SHUTDOWNS
- REACTOR SCRAM WITHOUT CONTAINMENT ISOLATION
- INADVERTENT OPENING OF A SAFETY RELIEF VALVE
- REACTOR SCRAM WITH CONTAINMENT ISOLATION
- LOSS OF OFFSITE POWER

LOSS OF COOLANT ACCIDENTS

- RECIRCULATION SYSTEM LINE BREAKS
- CORE SPRAY LINE BREAKS
- INSTRUMENT LINE BREAKS
- OTHERS TO BE DETERMINED

ANTICIPATED TRANSIENTS WITHOUT SCRAM

- WITHOUT CONTAINMENT ISOLATION
- WITH CONTAINMENT ISOLATION

STATION BLACKOUT

SPECIAL TRANSIENTS

- LOSS OF A VITAL DC BUS
- LOSS OF SWITCHGEAR ROOM COOLING
- LOSS OF CONTROL STRUCTURE HVAC

Types of Initiating Events

The number of events which satisfy this definition is quite large, thus evaluation of each specific initiator type becomes impracticable. Many of these initiating events have an identical or perhaps very similar impact on the plant. Therefore, it is logical to group these similar events into classes. Traditionally these initiators have been classified by the plant's transient response to the event. The transient response is used to define what processes are required to successfully restore the plant to a steady state - that is, the success criteria. Transients with the same or similar success criteria are grouped into the same class.

In the past, evaluations of initiating events have been limited to events that directly result in a reactor scram. In more recent studies the initiating events have been expanded to events which degrade plant systems and result in a plant shutdown. In these situations the plant transient response may be identical, however, the equipment available to respond to the event may be much different. Thus plant equipment availability is also a logical means by which to classify initiators.

Many sources of operational, design, and transient data exist which can be used to identify and categorize initiating events. Plant operational and design data represents the best source of information for defining initiating events and their frequencies. Unfortunately, much of this data is difficult to obtain, insufficient for making inference, or nonexistent. Therefore, other sources of data must be used to determine the type and frequency of initiating events. Compilations of industry wide initiating event data have been prepared which represent an excellent starting point for such evaluations. A limited set of reports contain the bulk of the data. A synopsis of these reports appears in Table 1.

These data sources were used when preparing the Susquehanna IPE. The initiating events were divided into principal categories: transients and loss-of-coolant accidents (LOCAs). Transients are events involving a perturbation to the plant without an uncontrolled loss of reactor coolant into the drywell and include the inadvertent opening of a safety relief valve. LOCAs are events involving an uncontrolled loss of reactor coolant into the drywell. These two principal categories of initiating events are distinguished by a different configuration of the reactor and different demands on mitigating systems. Each of these categories are further divided into finer classes.

Transients are divided into the following classes: manual reactor shutdown, reactor scram without containment isolation, stuck open safety relief valve, reactor scram with containment isolation, and loss of offsite power. The reactor transient response for these initiating events is similar, however the equipment available is different. Each case represents a degraded state of the plant. A manual reactor shutdown is an orderly plant evolution and therefore represents a very benign event. A reactor scram without isolation is more severe than a manual shutdown since it requires prompt action by operators or equipment to reestablish control of the plant. Because the main condenser is available as a heat sink and the feedwater-condensate system is used for inventory control, these events are still benign. The case where an SRV inadvertently opens while at power represents a direct challenge to the primary containment. For this reason a plant scram is required if the SRV

cannot be closed in 2 minutes. The suppression pool is at an elevated temperature and the high pressure turbine injection systems are lost early in the event due to the reactor depressurization. Thus this event represents a more significant threat to the plant. Reactor scrams with containment isolation result in loss of the main condenser and the feedwater pump.[1] Thus a layer of defense in both heat removal and inventory control is lost as a result of this event. A loss of offsite power is even more severe in that only the onsite generators are available to provide power to safety related equipment.

The second general category of initiators considered in the Susquehanna IPE were LOCAs. LOCAs and their characterization were not treated mechanistically in the Susquehanna IPE. This was seen as a deficiency which is being rectified in our revision activity. LOCA's are classified in terms of the time required to lower the reactor vessel pressure to the low pressure ECCS operating range and the equipment that is lost as a result of the pipe break. Each vessel penetration is broken at the vessel and at the first isolation valve. A LOCA calculation is then performed to determine the transient response. Breaks having similar response are then grouped into a break class.

As an example a break of the core spray line would result in a rapid reactor depressurization, the loss of RCIC and HPCI due to the depressurization, and the loss of a core spray loop due to the break.

The anticipated transient without scram (ATWS) and the station blackout (SBO) event are identified as unique initiators since they represent a significantly different plant condition than those identified thus far. The ATWS represents a significant challenge to the containment and requires prompt specific operator response if a favorable outcome is to be realized. ATWS is subdivided into the two categories: non-isolation and isolation. In the non-isolation case the main condenser is still available as a heat sink. If the power is reduced to the turbine bypass capability no immediate threat to the containment exists. The isolation case is much more demanding in that all the heat generated in the reactor must be rejected to the suppression pool.

Occurrence of the SBO on the other hand represents an indirect challenge to the plant in that most of the plant's safety systems are lost. Therefore, these events are treated as unique initiators.

A final class of Initiating Events to consider involves equipment malfunctions which degrades the plant's ability to respond to transients and results in a reactor scram. These events are called special transients. These events include: loss of a vital DC bus, loss of switchgear room cooling, and loss of control structure HVAC. These events will result in a reactor scram and disable safety related equipment. Since they are the cause of reactor scram, they must be considered as an initiator.

[1]This may not be true if motor driven feedwater pumps are utilized.

TABLE 1

Synopsis of Reports Containing Initiating Event Data

Report Identifier	Synopsis
NUREG/CR-2497 June 22 NUREG/CR-3591 NUREG/CR-4674 Vol. I-VI years 82 and 83 are missing	These reports contain an evaluation of LER from 1969 to the present. They identify the types of initiating events that have occurred, a historical estimate of the frequency and a description of the plant response for each event reviewed.
NSAC-80 NSAC-111	This report describes all the loss of offsite power events, and their durations through the present.
EPRI-NP-801 EPRI-NP-2230	This report gives a description of the possible transient initiators and their frequencies.
NUREG-1032	This report offers a methodology for assessing the frequency of loss of offsite power events lasting T hours.
Final Safety Analysis Report Chapters 6 and 15	This report describes the type of accidents considered when designing the plant.

INITIATING EVENT FREQUENCIES

INITIATING EVENTS	GENERIC DATA	SUSQUEHANNA SPECIFIC DATA		
		UNIT 1	UNIT 2	COMPOSITE
MANUAL SHUTDOWNS	4.3	4.22	3.40	3.86
REACTOR SCRAM WITHOUT CONTAINMENT ISOLATION	6.47	2.13	1.73	1.94
INADVERTENT OPENING OF AN SRV	0.07	0.30	0.1	0.18
REACTOR SCRAM WITH CONTAINMENT ISOLATION	0.93	1.82	0.42	1.24
LOSS OF OFFSITE POWER	-	0.059	0.059	0.059
ANTICIPATED TRANSIENT WITHOUT SCRAM WITHOUT CONTAINMENT ISOLATION	1.54×10^{-4}	1.21×10^{-4}	9.3×10^{-5}	1.06×10^{-4}
ANTICIPATED TRANSIENT WITHOUT SCRAM WITH CONTAINMENT ISOLATION	4.5×10^{-5}	9.8×10^{-5}	2.3×10^{-5}	6.7×10^{-5}
STATION BLACKOUT	1.88×10^{-4}	1.88×10^{-4}	1.88×10^{-4}	1.88×10^{-4}

Initiating Event Frequencies

This section covers how the frequency of each initiating event is estimated. Generic data was used in the SSES-IPE to estimate initiator event frequency. Tables 7.9-2 through 7.9-4 of the SSES-IPE identify what events were considered and the rationale for inclusion into a particular class. This information is then used to estimate the frequency of a given transient initiator. NUREG-1032 was used to estimate the loss of offsite power frequency. The LOCA frequencies were based upon WASH-1400. The ATWS probability was derived from the transient initiator frequency and the NUREG-0460 without scram probability. The station blackout frequency is just the LOOP frequency from NUREG-1032 times the failure probability of the onsite AC power system. If sufficient plant specific data sources are not available, then this approach seems reasonable.

Since publishing the Susquehanna IPE we have examined our plant shutdown history. We plan to use this data in our revision where possible. A comparison of generic data to SSES specific data is presented in the accompanying view graph.

The LOCA frequency is estimated by counting pipe sections for a particular pipe run and then multiplying it by a generic pipe break frequency per pipe section. This is shown with the following equation.

$$\text{LOCA Frequency} \leq \text{Pipe Section Count in pipe run} \times \frac{\text{frequency of break}}{\text{pipe section}}$$

EVENT TREE CONSTRUCTION

AND

SUCCESS CRITERIA

EVENT TREE CONSTRUCTION AND SUCCESS CRITERIA

- HEALTH AND SAFETY OF GENERAL POPULATION GUARANTEED BY ENSURING BARRIER INTEGRITY
- RULES OF EVENT TREE CONSTRUCTION
- PERFORMANCE REQUIREMENT FOR CORE INTEGRITY
- PERFORMANCE REQUIREMENT FOR CONTAINMENT INTEGRITY WITH THE VESSEL INTACT
- PERFORMANCE REQUIREMENT FOR VESSEL INTEGRITY
- PERFORMANCE REQUIREMENT FOR CONTAINMENT INTEGRITY VESSEL NOT INTACT
- STATION BLACKOUT EXAMPLE

Event Tree Construction and Success Criteria

The requirements for siting a nuclear power station are presented in 10CFR100. These requirements specify a set of reference radiation exposure limits which are used to determine the acceptability of a site. These radiation exposure limits are not given as acceptable levels of exposure for the general public, but are meant to be limiting values of exposure which could occur in extremely low probability accidents in a nuclear plant. The presumption is, in the evaluations made, that severe core damage has occurred with a release of "appreciable quantities of fission products". This release is normally taken to be 100% of noble gases and 25% of halogens released to the containment atmosphere. The containment is nevertheless considered to remain intact and functional. In the calculation of public exposure, it is presumed that the population in a "low population (LPZ)", zone can be evacuated within a time period of not less than two hours after the start of the fission product release such that the reference exposures are not exceeded, and such that those beyond the LPZ will not receive exposures greater than the reference values with no evacuation or any other form of protection from the airborne fission products.

We shall use a rather loose form of the guidance given in 10CFR100 to distinguish between accident sequences which represent a threat to public health and safety and those which do not. In general, we shall assume:

1) if no core damage occurs, there is no consequence,

2) if core damage occurs, but containment integrity is not lost, the consequences are acceptable, but LPZ equivalent evacuation may be required to assure conformance to 10CFR100 limits, and

3) if core damage occurs and containment integrity is lost, off-site consequences must be considered.

The loss of containment integrity can take many forms from wetwell venting to overpressure failure in the drywell corresponding to relatively benign failure type and the most severe failure type respectively. For this reason the severity of off-site consequence depends not only on the time and nature of core damage but also on the form of containment loss of integrity.

It must be recognized that with these definitions, the avoidance of off-site consequence could require some level of evacuation even in the event that containment integrity is not lost because of the design value of the containment leak rate. Even in the realm of probabilistic risk assessment where we attempt to use realistic analysis models, some accident sequences could result in sufficient containment leak rates that 10CFR100 limits could be threatened without some level of evacuation.

In this discussion we have identified only the fuel clad and the containment as fission product barriers, and in the strictest sense this is a proper assessment. Nevertheless, the reactor vessel also represents an important barrier in that if the fuel damage can be confined within an intact reactor vessel, the core debris does not directly threaten containment integrity. If the core damage progresses to a point where high temperature debris causes vessel failure with release of the debris to the drywell floor, the containment is threatened by several new phenomena.

For this reason we explicitly consider the performance of three barriers in our analysis:

1) the fuel clad,
2) the reactor vessel, and
3) the containment.

This presentation will address the various issues which must be considered in the protection of these barriers, and will describe the analysis process used to determine the outcome of the various event sequences.

In the construction of the Front-line Function Event Tree (FFET), the tree will be terminated when adequacy of core cooling or lack of it, has been determined. In the case of ATWS the tree is terminated when successful shutdown, or failure to shutdown, has been determined. If shutdown is not accomplished, containment overpressure failure is the consequence. Overpressure failure in turn can result in reactor injection capability and therefore lead to inadequate core cooling and core damage with the containment failed.

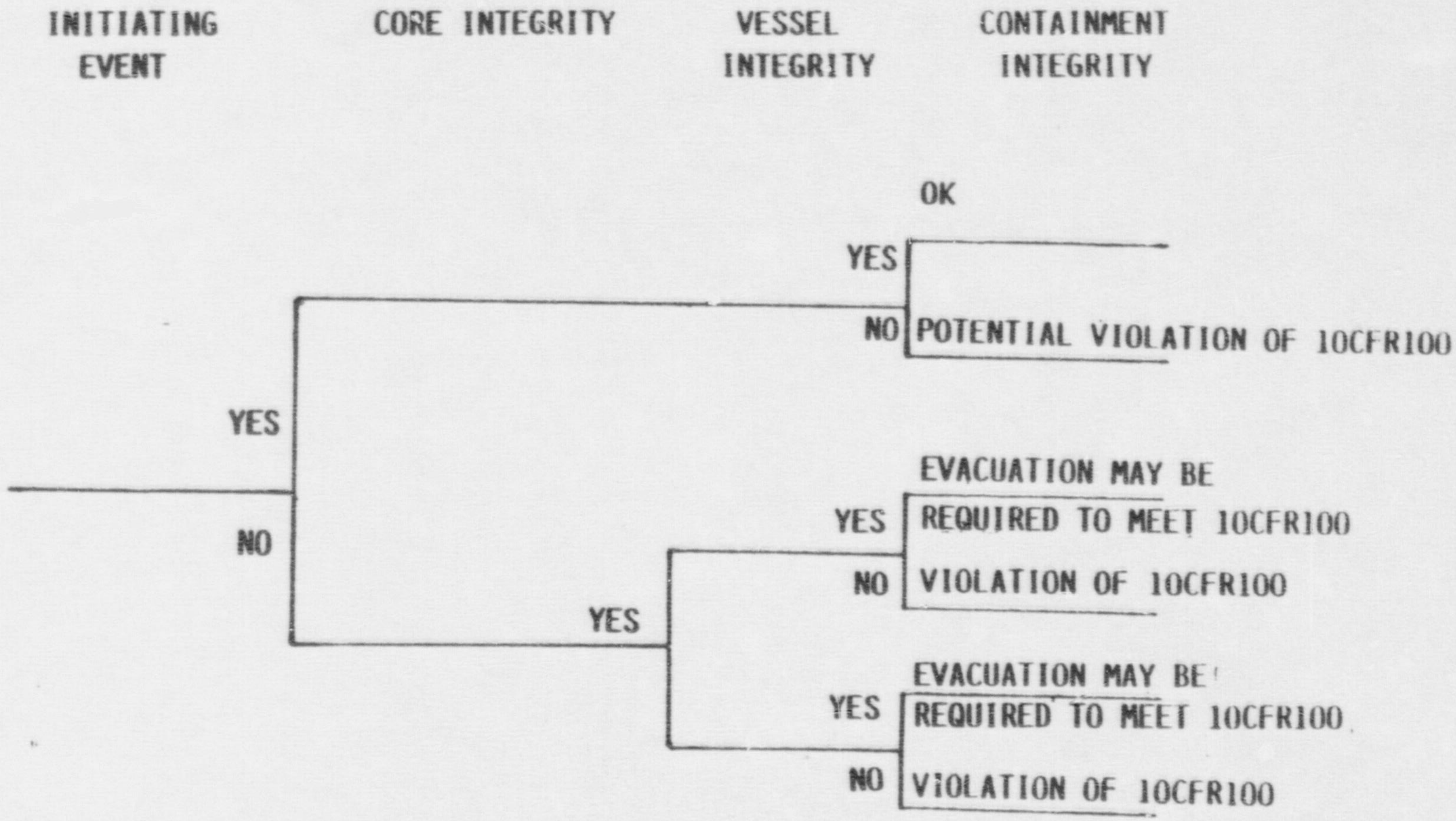
In cases where reactor shutdown has occurred and an extended period of inadequate core cooling with loss of clad integrity has not occurred, an extension of the FFET to consider successful restoration of decay heat removal or venting can be used to determine the eventual status of the containment.

In those cases where inadequate core cooling has occurred and core damage results, the situation is much more complex and cannot readily be represented as an extension of the FFET because of the complex interactions which must be considered. These cases can be fully dispositioned to the full spectrum of final plant damage states however by relatively simple logic equations which account for what equipment is unavailable and the probability of recovery in time to avoid specific levels of plant damage.

For this reason the FFET is always terminated without establishing the final reactor vessel or containment status. In the case where adequate core cooling is assured, however, reactor vessel integrity may be inferred. The subsequent viewgraphs and discussion will attempt to outline the approach taken to represent the various accident sequence types.

This information will then be used to given an example of the process for Station Blackout

GENERAL EVENT TREE



General Event Tree

The event tree depicted in the viewgraph does not represent the actual nature of the event trees used by PP&L in risk analysis. It is, rather, a conceptual device used to present the overall flow of the risk analysis process. The intent here is to show that there are three barriers to be considered in the accident progression (only two of which are properly considered fission product barriers) and to show the necessary sequence for their evaluation. In actual practice an event tree type of logic is used only to the point of determining whether core damage has occurred. For sequences where core damage has occurred an event tree structure is not well suited to describe the logic which determines the various branchings which can occur and the actions which influence them. The basic reason for this is that a unique set of top event questions which will apply to all branches in the sequence of events cannot be developed. The logic of the various event sequences is, nevertheless, straight forward and tractable.

For this reason we do not use an event tree structure beyond the point of core damage, but rather develop a set of logic equations which can properly describe and quantify the various possible outcomes as the sequence progresses. The logic required when no core damage has occurred is much simpler than when the reactor vessel integrity is subjected to a threat by inadequate core cooling. For that reason these two cases will be treated separately in the following discussion.

The conceptual event tree presented here does indicate the crude degree of accident classification which can be made on the basis of success and failure combinations in preserving barrier integrity. The following discussion presents some of the considerations involved in developing a logical structure for determining the various courses an accident may follow.

The event tree is a graphic method of logically linking questions about the status of the fission product barriers to the impact on them from the initiating event. The event tree for a specific initiator is developed by deducing specific questions about the plant response from general questions concerning barrier integrity. The most general event tree is identified in the view graph. The first heading identifies the initiating event being considered. The initial conditions that the plant must respond to are identified at this point.

The questions are ordered to preserve the natural dependence of the damage progression. For example, the evaluation of containment integrity depends upon the status of the reactor vessel. If the vessel is intact then the containment integrity is only challenged by pressurization. If on the other hand the vessel bottom head is breached by the melted core, containment challenges due to core concrete interactions, which may be overpressure or overtemperature or both, must be evaluated. Therefore, the status of the vessel must be established prior to considering containment integrity. Vessel integrity, however, can only be challenged if core cooling is lost. Therefore, the integrity of the core must be considered first. If core cooling is maintained then vessel integrity is implied. The containment integrity, however, is not guaranteed by ensuring clad integrity since it can still be challenged by pressurization, but only by loss of decay heat removal capability. Therefore, the status of containment must be questioned even if

core damage has been avoided. If loss of cooling causes the integrity of the clad to be lost, then the vessel integrity cannot be implied. Therefore, if clad integrity is lost vessel integrity must be evaluated. If the vessel integrity is preserved then the challenge to the containment is limited to pressurization from loss of decay heat removal, hydrogen generation, and the additional chemical energy release. If on the other hand the vessel is lost then the containment will be challenged by core concrete interactions in addition. The core concrete interactions may involve

- additional chemical energy release,
- additional pressurization due to noncondensible gas generation,
- higher drywell temperatures which weaken components and increase gas pressure,
- failure of the drywell-suppression pool pressure barrier and loss of suppression pool quenching, and
- direct core debris attack on the drywell liner causing loss of containment integrity.

We also consider venting to reduce containment pressure as loss of containment integrity.

Specific questions concerning the plant response to the initiating event are deduced from the general questions on barrier integrity. This is performed by determining what performance is required to maintain barrier integrity for the plant state associated with the initiating event. The success criteria are defined by the performance requirements. The plant design data and procedures are used to identify what plant equipment can be used to satisfy the given performance requirement. The performance of this equipment then is specified as the top events identified in the event tree.

The general rules for event tree construction are as follows.

1. Determine the plant initial conditions associated with the initiating event.
2. Identify the performance requirements sufficient to ensure barrier integrity.
3. Use the plant design data and the operating procedures to identify equipment that satisfy the performance requirement. Operation of specified equipment becomes the success criteria.
4. Group equipment by performance requirement.
5. Successively fail equipment identified by the success criteria until barrier integrity is lost.

In the following discussion performance requirements are developed for the clad, the vessel and the primary containment. The station blackout event is used to demonstrate equipment success criteria which satisfy these performance criteria. This information is used to build an event tree for station blackout.

BWR FRONT-LINE FUNCTIONS

- REACTOR SHUTDOWN
  - REDUCES HEAT GENERATION RATE TO SHUTDOWN COOLING SYSTEM LEVELS.
  - AVOIDS DAMAGE FROM DEGRADED ATWS SEQUENCES.
- REACTOR VESSEL MAKE UP
  - ASSURES TWO PHASE CORE COOLING AND AVOIDS HIGH TEMPERATURE FUEL FAILURES.
- CONTAINMENT HEAT REMOVAL
  - ASSURES PROTECTION AGAINST CONTAINMENT OVERPRESSURE FAILURE GIVEN ADEQUATE PERFORMANCE OF REACTOR SHUTDOWN AND REACTOR VESSEL MAKE UP.
- DEBRIS COOLING
  - MINIMIZES THE THREAT TO CONTAINMENT INTEGRITY GIVEN INITIAL FAILURE OF REACTOR VESSEL MAKE UP.

BWR Front-Line Functions

In the preceding view graph the general event tree which poses the questions from which the degree and nature of the threat to the health and safety of the public are derived is presented. In developing the specific questions from which core, vessel, and containment integrity are determined it is necessary to consider the front-line functions which protect the health and safety of the public. These are:

1) Reactor Shutdown,

2) Reactor Vessel Make Up,

3) Containment Heat Removal, and

4) Debris Cooling.

The questions to be posed must address each of these front-line functions. They are briefly discussed individually below.

Reactor Shutdown

Failure to shutdown the reactor when the turbine has tripped exposes the plant to loss of clad integrity and loss of containment integrity and their consequences if the backup systems which must operate on failure to scram also fail. Such failures can cause loss of clad integrity due to potential reactivity transients or unstable behavior when ARI and SLCS have failed in response to ATWS.

Since the shutdown heat removal systems can only accommodate decay heat levels it is essential to reduce the fission power to avoid containment failure when ARI and SLCS have failed. For non-isolation cases this is accomplished by recirculation pump trip and feed water runback to a rate which can be accommodated by the steam by pass system. For isolation cases actions must be taken to extend the time available to allow manual rod insertion. These are:

1) trip of the recirculation pumps

2) reduction of reactor pressure,

3) addition of mass to the suppression pool, and

4) if available and permissible, use of the wetwell vent.

Reactor Vessel Make Up

If sufficient reactor vessel make up is not provided in a timely manner, the core will lose two phase cooling and fuel temperatures will begin to rise due to the reduction in heat transfer and transport rates. If high pressure systems are available, it is necessary to derive the minimum flow necessary to maintain adequate core cooling and the time at which such flow must be provided. Factors which can influence this are:

1) initial vessel inventory,

2) reactor shutdown, and

3) a loss of coolant causing vessel pressure reduction.

These issues are presented in subsequent view graphs and simplified hand calculations for some cases are presented.

Containment Heat Removal

The normal and preferred method for removal of heat from the containment after a turbine trip is directly from the reactor, in the form of steam, to the main condenser via the main steam bypass line. If suppression pool heat up has occurred, the bypass system cannot remove the heat involved, and the suppression pool cooling system must be used. The suppression pool cooling system, however, can only stabilize the pool temperature at decay heat power levels. For this reason it is essential to either shutdown the reactor or maintain availability of the main condenser. Even when the main condenser is available, however, it is essential to achieve reactor shutdown as quickly as possible since it is necessary to operate the reactor with a makeup flow in adequate to maintain the normal reactor water level, a poorly analyzed and untested mode of reactor operations.

Debris Cooling

If adequate core cooling is not provided initially and core damage initiates, the operator response strategy must be:

1) to restore vessel makeup at the earliest possible time to terminate the damage progression before reactor vessel failure, and

2) to prepare for reactor vessel failure by flooding the drywell floor and operating drywell sprays.

If the reactor vessel can be saved, the threat to containment integrity is greatly reduced. This threat involves:

1) generation of large quantities of non-condensible gases,

2) severe heating of the drywell atmosphere and components, and

3) potential direct thermal attack on critical drywell components by core debris

If vessel failure cannot be prevented, preparation for its failure can reduce the magnitude of the threat by cooling the debris as it falls to the pedestal room floor.

CONSIDERATIONS FOR CLAD INTEGRITY

- DEFINITIONS OF REQUIREMENTS FOR CLAD INTEGRITY

- MAKE UP FLOW REQUIREMENTS

- BOILDOWN TRANSIENTS

- BLOWDOWN TRANSIENTS

- MECHANICAL CLAD INTEGRITY

Considerations for Clad Integrity

When we speak of core integrity, we are actually speaking of integrity of the fuel clad. As long as the clad maintains integrity, the fission products cannot escape and the potential for hazardous levels of radiation exposure to the public is quite low.

Clad integrity may be lost by two different mechanisms,

1. severe clad oxidation and

2. mechanical failure of the clad.

The first of these is caused by lack of two phase cooling for some period of time. In this event, cooling is achieved only by heat transfer to the steam environment and as a result, the fuel and clad heat up to temperatures at which chemical reaction rates between the zirconium clad and the steam becomes significant. Since this reaction is exothermic and the rate increases with temperature, a temperature level can be reached at which the heat produced by the chemical reaction exceeds the decay heat level. At that temperature, the rate of temperature increase increases rapidly until the melting point of the clad material is reached and relocation of molten material reduces the reaction rate locally. The molten material flows or falls to lower regions of the core where lower temperatures may cause refreezing may cause refreezing of the melt or water may quench the melt and terminate the chemical reaction.

Mechanical loss of clad integrity is primarily a result of stresses in the clad caused by internal gas pressure or mechanical interference with the UO_2 pellets or a combination of these two sources of stress. The only credible source of this type of clad damage are:

1. a brief period of inadequate core cooling with a rapid quench by refloods or

2. an ATWS event involving unstable operation or a severe reactivity transient.

In the case of the first, we do not make a distinction between this form of clad damage and that caused by severe oxidation. The time period in which clad is vulnerable to this form of loss of integrity as opposed to severe oxidation or melt loss of integrity is very short and the conditional probability of vessel injection recovery in that time period is relatively small.

Mechanical clad damage in ATWS events may occur when the reactor is taken into a range of operating conditions where it becomes unstable or when a rapid insertion of reactivity occurs. In the case of unstable operation the reactor goes into an oscillatory power mode of operation. These oscillations can become quite large and the resulting cycles in thermal expansion and contraction can result in eventual mechanical failure of the clad by crack formation. In the case of reactivity insertions, the cause may be a sudden increase in the rate of injection of water.

The criteria for occurrence of each of these forms of clad damage is discussed below.

Core Damage Due to Oxidation

The reactor core is composed of three major components: zircaloy clad fuel rod, zircaloy channel boxes and stainless steal control blades. These components continuous experience some extremely small amount of oxidation at normal operating temperatures. Thus, some amount of oxidation can occur without considering the core to be damaged. The oxidation reaction rate is strongly temperature dependent. If core cooling is interrupted, the core temperature will increase which in turn increases the clad oxidation rate. This oxidation reaction causes the zircaloy to become brittle and thus more susceptible to thermal shock upon reflood. Experimental data indicates that at temperatures greater than 2000°F and with more than 30% of the clad oxidized the fuel clad shatters upon reflood, (ref. 1) forming a rubble bed. Clearly this represents major core damage. This condition is very unlikely to occur in an actual reactor accident sequence, however, since at 2000°F the fraction of clad oxidized is only a few percent and the temperature is rising rapidly. The maximum clad temperature which would characterize a 30% oxidation level would be far in excess of 2000°F and would probably involve some degree of clad melt.

The data also demonstrates that restoration of core cooling prior to exceeding the 2200°F and 17% equivalent clad reaction limits set by the Code of Federal Regulations (ref. 2) assures integrity of the clad barrier. The integrity of the other core components is inferred. Therefore, if adequate core cooling is lost, core damage will not occur if adequate core cooling is restored prior to exceeding 2200°F and 17% equivalent clad reaction.

Adequate core cooling is assured as long as the fuel is cooled by two phase flow. Two phase cooling of the fuel is assured if the downcomer water level is above the top of the active fuel (TAF). Because of the swell in the core due to the boiling process, two phase cooling can be demonstrated, in the short term, at downcomer water levels of 2/3 core height. This level is also known as the jet pump throat elevation. Therefore adequate core cooling for the short term is defined in terms of a downcomer water level above the jet pump throat elevation. Because of the limited water volume inside the core shroud, core dryout may occur only a short time after water level falls to the jet pump throat elevation.

At later times in the transient, the 2/3 core height criterion becomes invalid as a result of the decreased amount of swell due to boiling as decay heat decreases. In such cases a downcomer level approaching TAF may be required to assure adequate core cooling. In such a case, high temperatures would be experienced in the low powered bundles first as a consequence of reduced swell.

In Susquehanna cores the axial power profile may also influence the definition of adequate core cooling since various bundle types have differing axial profiles. Some bundles may have power peaked toward the top of the bundle while others are strongly bottom peaked. This difference will strongly influence the degree of swell and the vapor superheat developed above the top of the two phase level.

Adequate core cooling can be lost in two ways. First, injection flow to the vessel can be lost, in which case the water inventory will be boiled off and transported as steam through the SRVs to the suppression pool. In this situation adequate core cooling is assured until the downcomer water level

falls down to the jet pump throat elevation. Second, a rapid depressurization or an unisolated break in the primary system piping can occur which will blowdown the reactor pressure vessel. The difference in phenomena between these sequence types imposes different timing and performance requirements on the plant equipment.

Mechanical Failure of the Clad Pressure

Mechanical failure of the clad pressure boundary can result from extensive severe power cycling of the fuel or from a severe power excursion. Rapid gross failure of this nature is not expected to occur except for ATWS events. The fundamental technical issue, relative to BWR ATWS is the lack of any experimental data for reactor operation in the physical regimes appropriate to ATWS combined with a deficiency of adequate analytical tools to describe reactor behavior by theoretical means. Certain limits on reactor response to ATWS can be reliably derived from conservation considerations. As an example, it is obvious that, within certain limits, the power level of a critical reactor in natural circulation flow will be just sufficient to transform the makeup water into steam for release from the reactor at a rate which just equals the makeup rate. What we cannot derive, with fully benchmarked methods, is the downcomer water level and core flow rate that will result. Perhaps even more important, we cannot assure that the reactor will not operate in an oscillatory, limit cycle mode under the conditions selected for evaluation with a "time averaged" mass balance being maintained.

The two technical issues which have been identified as important to success in mitigation of ATWS events are:

1. Limit cycle operation.
2. Reactivity excursions during vessel reflood.

The criterion that we have defined for avoidance of this type of clad damage is that if the reactor is not depressurized while critical, no damage will occur. If depressurization becomes necessary due to exceeding the HCTL or failure of high pressure injection while the reactor is still critical, clad damage is postulated to occur. This damage is not related to inadequate core cooling, however. The need to depressurize exposes the reactor to potential reactivity excursions in the case of loss of high pressure injection due to the fact that a core dryout and reflood sequence is likely to be generated. In the case of depressurization because of exceeding HCTL, the pressure reduction is know to reduce stability when the reactor is already known to be operating in a region of marginal stability. The assumption of oscillations of sufficient severity to result in loss of clad integrity may be excessively conservative, but at this time we are unable to perform quantitative analyses of the consequences.

In subsequent view graphs we will discuss:

1. make up for requirements,
2. boildown transients,
3. blowdown transients, and
4. mechanical clad integrity.

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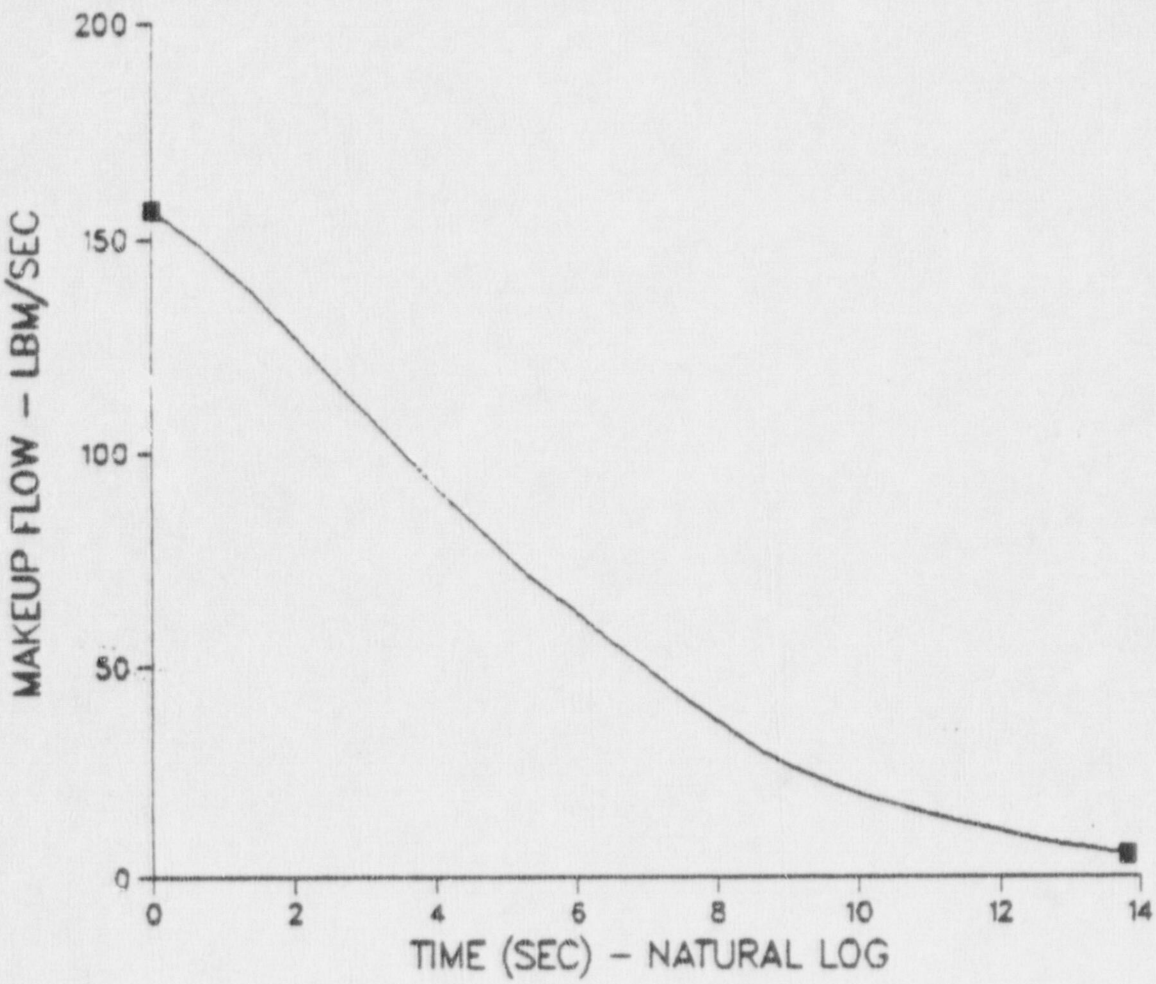
ref. 1 Embrittlement Criteria for Zircaloy Fuel Cladding Applicable to Accident Situations in Light Water Reactors: Summary Report, H. M. Chung and T. F. Kassner, NUREG/CR-1344, January 1980.

ref. 2 10CFR50 Appendix K.

MAKE UP FLOW REQUIREMENTS



Makeup Flow Requirements

Makeup flow can be lost at any time during the accident sequence and at any point in the operating pressure range. The loss of makeup may also be complete or partial. To evaluate the adequacy of core cooling in this situation it is necessary to know the minimum makeup flow required at the time makeup is lost and how much time is available to restore makeup. Both the minimum makeup flow rate and the boildown time can be determined using end point thermodynamics and the decay heat curve. As recommended by the USNRC, this evaluation utilizes the more realistic decay heat curve generated by Eric Haskin of Sandia. This data is presented in Appendix 1.

The minimum makeup flow must be sufficient to replace the water boiled off by decay heat. If it is less than this value boildown will commence. If a constant pressure saturated system is assumed and the reactor water level remains constant, the minimum makeup flow can be computed using the following equation:

$$W_{min}(t) = \frac{q(t)}{(h_g - h_{mu})}, \qquad (1)$$

here,

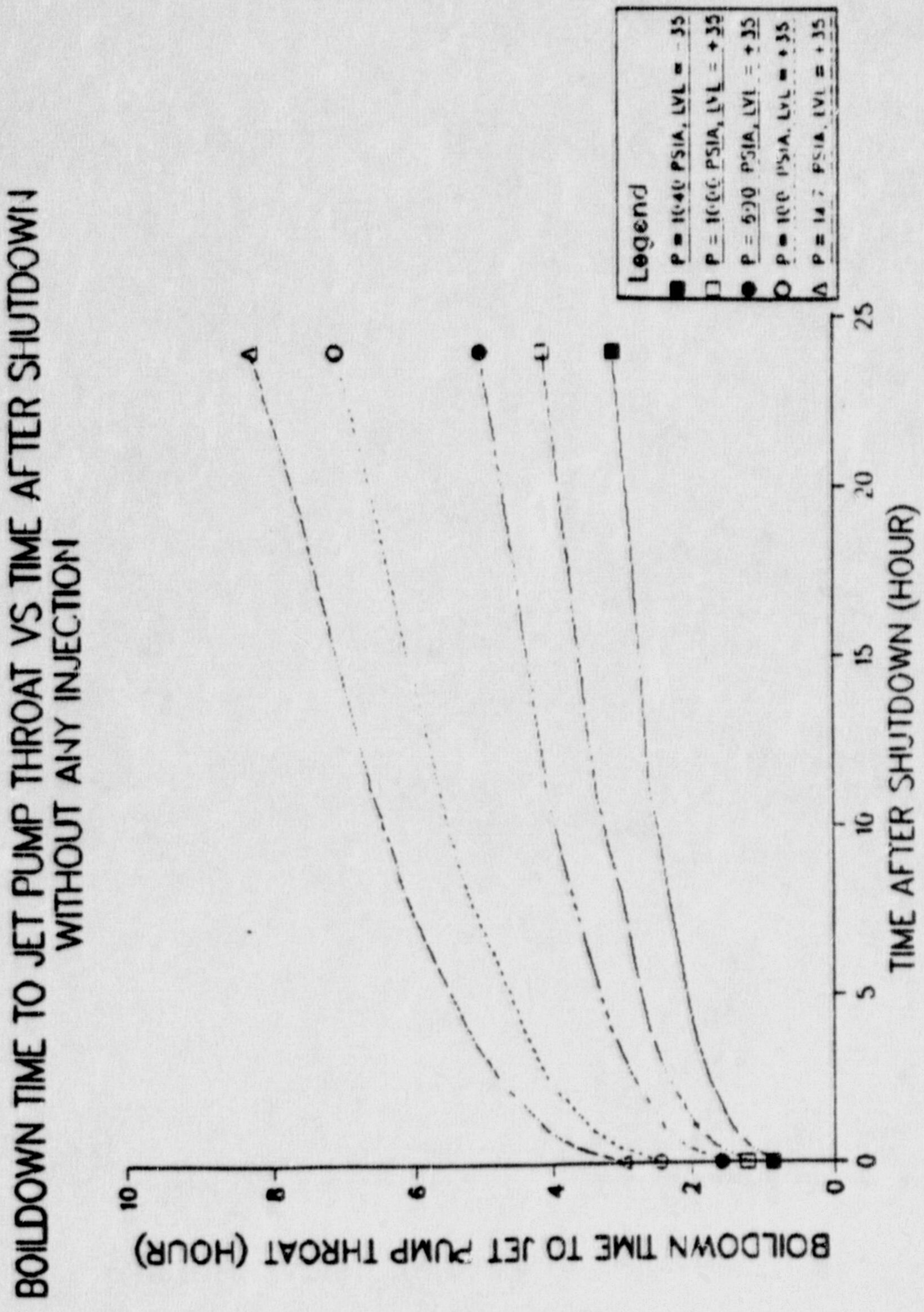
$W_{min}(t)$ is the makeup flow required at time t, #/sec.

$q(t)$ is the decay heat at time t obtained from Table 1, BTU/sec.

h_g is the enthalpy of saturated steam at the reactor pressure, BTU/#, and

h_{mu} is the enthalpy of the makeup flow, 58 BTUs/lbm at 85°F.

The steam enthalpy varies only slightly over a broad pressure range and so a constant value can be used without significant error. Makeup flow from sources external to the primary containment rarely exceed 85°F. If 85°F water is assumed to be used for makeup the minimum flow is only a function of decay heat. The flow rate for these conditions is plotted as a function of time after shutdown in the view graph. This data is also presented numerically in Appendix 2.


BOILDOWN TIME TO JET PUMP THROAT VS TIME AFTER SHUTDOWN
WITHOUT ANY INJECTION
BOILDOWN TIME TO JET PUMP THROAT (HOUR)
TIME AFTER SHUTDOWN (HOUR)
Legend
0
2
4
6
8
10
0
5
10
15
20
25

Analysis of Boildown Transients with Scram

The boildown time is required to determine how much time is available to restore makeup. If the reactor pressure is assumed to remain constant, the boildown time is computed using a simple energy balance. The integrated decay heat is used to boil off the liquid water mass in the vessel. This energy balance takes the following form:

$$\left[\frac{V_{os}}{\mu_f} + V_{is}(1-\alpha)\frac{1}{\mu_f} \right] h_{fg} + W_{in} t_f (h_g - h_{in}) = \int_o^{t_f} dtq(t)$$

here

V_{os} is the water volume outside the shroud, above jet pump throat, ft.
V_{is} is the water volume inside the shroud, above jet pumpt throat, ft.
α is the average void fraction inside the shroud,
W_{in} is the inlet flow rate (less than W_{min}), #/sec.
t_f is the boildown time, sec.

$\int_o^{t_f} dtq(t)$ is the integrated decay heat at the boildown time given in the decay heat table, BTU

μ_f is the specific volume of the liquid, ft^3/#,
h_{in} is the enthalpy of the inlet flow, BTU/#, and
h_g is the enthalpy of the steam, BTU/#.

The water volumes inside and outside the shroud can be obtained from the weights and volume drawing (Appendix 3). When applying this equation, equilibrium thermodynamics and a constant pressure are assumed. A constant pressure is not realistic. However, the pressure will vary only by about 100 psi. The thermodynamic properties of water do not change radically over this range thus the constant pressure assumption does not introduce much error. Because the boildown time is a limit of integration and is also used to determine the total makeup mass, the solution of this equation is iterative. In the special case where the makeup flow is zero the boildown time can be read directly off the integrated decay heat curve. This information is used to determine what equipment provides adequate core cooling.

ANALYSIS OF BOILDOWN WITHOUT SCRAM

FULL ATWS

60-180 SECONDS TO TAF

PARTIAL ATWS

UP TO 780 SEC FOR TURBINE TRIP

Analysis of Boildown without Scram

As in the scrammed condition adequate core cooling is assured if the core is cooled by two phase flow. Again two phase flow can be expected along the entire length of the core if the water level is kept above the jet pump throat. During an ATWS condition, however, the water level and fission power are dependent upon one another. A transient reactor kinetics code is required to determine this relationship. ORNL has studied the water level as a function of steady state power at various reactor pressures. If the reactor power remained constant at a set water level and pressure, the minimum makeup flow rate could be computed using a formulation similar to the one presented in the previous decay heat discussion. However, the boiling water reactor develops thermal-hydraulic instabilities as the core flow is reduced. Because of these instabilities we cannot be sure all channels are receiving two phase cooling. Because of this, the minimum water level that will avoid severe oxidation is restricted to top of the active fuel.

If makeup flow is lost during the ATWS transient the boildown is very fast due to the high power state of the core. This problem must also be solved using a reactor transient code. Several such calculations for the full ATWS case have been performed by the industry and the NRC. These calculations indicate that the time required for the water level to fall to TAF is between 60 and 180 seconds. At the other extreme, if the boildown starts at the feedwater high level trip setpoint during a partial ATWS, the boildown could take as long as 780 seconds. The evaluation of adequate core cooling for the ATWS transient requires a transient computer code.

ANALYSIS OF BLOWDOWN

BLOWDOWN RESULTS IN RAPID LOSS OF COOLANT INVENTORY

AMOUNT OF COOLANT LOST DEPENDS ON

- PRESSURE REDUCTION
- RATE OF DEPRESSURIZATION
- TIME AFTER SHUTDOWN

MUST REFLOOD PRIOR TO VIOLATING CLAD INTEGRITY

Analysis of Blowdown

If the reactor is rapidly depressurized through either the safety relief valves or a primary system pipe break, liquid phase will be lost as mass flowing out the break depressurizes the vessel. The blowdown may be intended to allow a low pressure pump to be used for vessel makeup when ADS is activated or undesired as in the case of a pipe break. In any event, the important phenomena to consider are the rate of depressurization and the mass lost during the depressurization. The rate of depressurization and the mass inventory lost depends upon the phase of water being discharged and the size of the opening. Steam breaks will depressurize the reactor more rapidly and result in less mass loss than a liquid break. This is because latent cooling is a much more effective than sensible cooling. The reactor vessel will begin to depressurize when the energy flow out the break exceeds the energy generation rate. There are two sources of energy to consider. These are the decay heat and the sensible heat stored in the reactor water, the reactor core, the vessel, and the vessel internals. If cold water is being injected during the depressurization it must also be considered in the energy balance since it will reduce the enthalpy of the water in the RPV and therefore reduce the reactor pressure. This is a very complicated process which is best analyzed by a transient computer code. Several LOCA cases are documented in the FSAR as part of the design basis analysis. GE has also documented detailed LOCA analysis for each of the product lines in NEDO 24708. These analyses are success oriented in that they demonstrate what equipment combinations are sufficient to mitigate a LOCA without exceeding the criteria specified in the Code of Federal Regulations. The necessary requirements, however, must be inferred.

Vessel depressurization calculations are best done using suitable computer codes designed for the purpose. If such codes are not available, the special case for a pure steam blowdown can be evaluated approximately by hand calculations. To perform such a calculation, we must be select the time interval required for the blowdown. The calculation is most valid for depressurization events which are relatively slow (several minutes or greater) in order to assure a high degree of thermal equilibrium at the end of the process. More rapid depressurization events can also be evaluated, but generally some allowance for a non-equilibrium final state should be made.

The calculation is performed by consideration of conservation equations for energy, mass, and reactor volume. These three equations permit derivation of an equation for the blowdown steam mass as a function of the initial state of the system (fully known), the mass and energy of makeup water added during the process, the decay heat released, and the final thermodynamic state of the system (temperature and pressure only). The solution of these equations and the symbol definitions used are presented in Table 1 and Table 2 of Appendix 4 respectively.

It should be noted that an allowance may be made for failure to fully achieve thermodynamic equilibrium in the blowdown process through the provision of the factors f_V, f_F, f_I which account for the fact that these large masses may not fully come into thermal equilibrium with the liquid and vapor phases in the vessel if the depressurization event is too rapid. In reality, the reactor vessel almost certainly will never be in complete equilibrium when the depressurization process is complete. That portion in contact with the liquid

phase would probably be close to the same temperature as the final liquid phase, but the portion of the vessel in contact with steam only would probably be at a considerably higher temperature due to relatively poor heat transfer. The portion of the vessel in contact with steam gives up its energy as superheat to the steam, in any event, which was not considered in the derivation. For this reason the value of f_V chosen should reflect the fraction of vessel mass in contact with steam only using an average water level over the depressurization period in the determination.

A similar situation may also apply to some vessel internals such as the steam dryer which is well above the water level. Some portion of the steam separators may also be involved. Nevertheless, the value of f_I chosen should usually be much nearer unity than f_V.

In the case of f_F it is probably best to assume two phase cooling always exists unless the final water level calculated is extremely low.

The calculation of final water level is quite simple. Using the final value of liquid mass and density, the liquid volume is calculated and this is used to determine the final water level from a table of free volume in the vessel as a function of elevation. This table is fairly simple to develop. An example for Susquehanna is given in Appendix 3. The influence of core voiding on final elevation is neglected since the pressure losses due to natural circulation flow are low. This means that collapsed water level inside the shroud must equal that outside the shroud. This assumption is slightly conservative in terms of estimating final water level. For final water level which is calculated to be below the jet pump throat elevation, some caution must be used since the downcomer/recirc loop volume becomes uncoupled from the lower plenum and the volume inside the shroud.

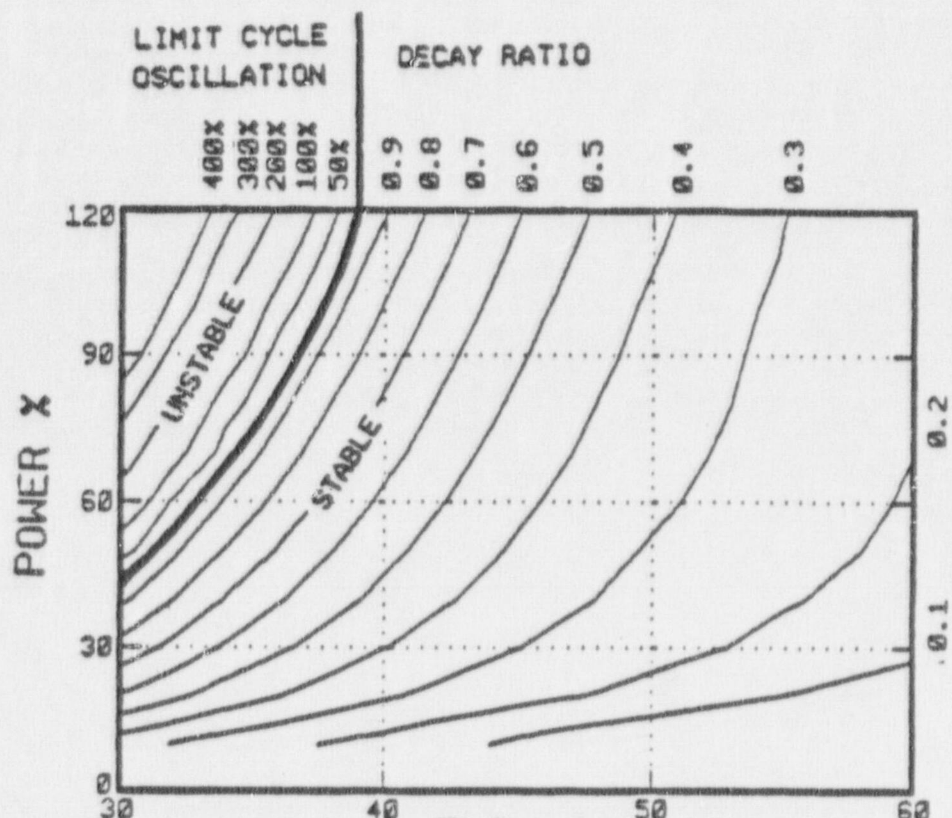
Caution should also be used in the values of M_{Li} and h_{Li} used. The M_{Li} should consider all liquid phase including the recirculation loops. The value of h_{Li} should be an average value which reflects the variations in initial liquid enthalpy throughout the reactor vessel.

LOSS OF CLAD PRESSURE BOUNDARY

THE LOSS OF THE CLAD PRESSURE BOUNDARY IS POSTULATED TO OCCUR FOR EITHER OF TWO REASONS

LIMIT CYCLE OPERATION, OR

REACTIVITY EXCURSIONS



Loss of Clad Pressure Boundary

Limit cycle operation represents an unstable mode of reactor operation in which the system non-linearities limit the magnitude of the unstable excursions and result in a bounded oscillatory mode of operation. The magnitude of the oscillations are dependent upon the operating parameters of the system, and the magnitude may be quite large under some conditions. Limit cycle operation has been observed in BWRs under natural circulation flow conditions with normal downcomer water levels. The concern then is that as downcomer water level falls, as it must in an ATWS event, is there an increasing trend to limit cycle operation, and, if so, what are the magnitude and the frequency of the resulting oscillations in reactor parameters.

We do not expect resolution of this issue within the next several years, and we are forced to rely on our judgement to assess the importance of this issue. Our approach has been to limit the degree to which downcomer level is allowed to drop, except under extreme conditions of equipment failure, in order to minimize the probability and magnitude of limit cycle operation. It is our expectation that oscillatory behavior will be sufficiently mild that there will be no significant risk of core damage or loss of instrument readability. In the event that limit cycle operation does occur, we would not expect damage more severe than mechanical failure of the clad in some bundles.

For our evaluation of ATWS we have assumed that severe extended limit cycle operation does not occur when adequate makeup flow is available and when reactor pressure is maintained at the design value. In this situation adequate makeup flow is defined in terms of sufficient flow to keep the water level above the top of the upper plenum. This is full HPCI flow.

If the reactor pressure vessel is ever depressurized while the reactor is still critical the void formation as a result of the blowdown will cause the core to become subcritical. If the reactor is reflooded using the high volume low pressure pumps, a prompt critical insertion could occur. Such insertions have been calculated using the TRAC-G code. The magnitude of the power excursion depends on the rate and magnitude of reactivity insertion. Sufficiently powerful excursions can cause clad failure from the combined effects of pellet-clad interface forces and increased internal gas pressure.

If such severe reactivity insertions are avoided, the low pressure of the new operating power nevertheless assures LCO. In this case we have postulated some degree of clad damage dependent upon the duration of this condition.

In the view graph diagram we have shown lines of constant decay ratio in the stable operating regime (to the right of the heavy line corresponding to D.R.=1.0), and lines of constant power variation magnitude to the left of the heavy line on a power flow map derived for design pressure. At lower flows than 30%, the reactor was found to go into an aperiodic mode of power variation having widely varying amplitudes. This mode of operation is probably more severe than pure limit cycle operation behavior.

If reactor pressure is reduced, the boundary between stable and unstable operation will move to the right as will the flow at which aperiodic behavior initiates. For this reason we ascribe mechanical clad damage to event sequences in which reactor pressure is reduced while the reactor is still critical.

CONTAINMENT INTEGRITY WITH ADEQUATE CORE COOLING

- REJECT DECAY HEAT PRIOR TO EXCEEDING THE CONTAINMENT ULTIMATE STRENGTH

- TIME TO ESTABLISH DECAY HEAT REMOVAL ESTIMATED BY CONVERTING INTEGRATED DECAY HEAT TO VAPOR PRESSURE

- DECAY HEAT REMOVAL MUST BE ESTABLISHED PRIOR TO VAPOR PRESSURE AND NONCONDENSIBLES IN THE WETWELL AIR SPACE EXCEEDING THE CONTAINMENT ULTIMATE STRENGTH

  WITHOUT MASS ADDITION
  WITH MASS ADDITION

Containment Integrity with Adequate Core Cooling

In accident sequences where hydrogen is not generated by metal water reaction the ultimate status of the containment may be determined by consideration of the rate of suppression pool heatup. In LOCAs and transients this is simply a matter of considering the stored energy released from the vessel and the decay heat. Frequently, this calculation may be done without use of transient computer programs if containment heat sinks are neglected. Consideration of heat sinks can extend the time before containment overpressure failure, or a need to vent to avoid it, by a significant amount, however.

In the case of ATWS events similar calculations may also be done, but the fission power will represent a much greater rate of heat generation. Approximate hand calculations may be used to determine the time available for boron injection of manual rod insertion simply by assuming a linear decrease in power from the stable level associated with initial makeup flow down to the decay heat level when shutdown boron has been injected or sufficient rods have been inserted for hot shutdown. The transient boildown energy from the start of the ATWS transient must be added to this.

If a conservative (and upper bound) calculation is desired, constant power can be assumed out to the time of hot shutdown. In the case of manual rod insertion, this conservatism is not recommended.

In all cases, LOCAs, transients, and ATWS, mass addition to the suppression pool can greatly extend the time to containment overpressure failure or the time to venting to avoid such failure. A simple calculation can be made to estimate the degree of time extension.

Initially in a transient, stored energy in the reactor and decay heat are rejected to the suppression pool. This energy raises the temperature of the suppression pool water, and thus increases the vapor pressure. The suppression pool is a large volume of water and therefore, the pressurization of the containment is slow. If this energy is not eventually rejected from the containment, it will fail on overpressure when the pressure reaches the containment's ultimate strength.

The containment ultimate strength must be evaluated prior to establishing a containment performance requirement. Such an evaluation has been performed for the Mark I by the BWROG. An analysis of the Mark II is presented in Appendix J of the Limerick PRA.

With this information established the time to reach containment overpressure failure can be estimated using the following simple energy balance. Two cases are reviewed: without and with mass addition.

Increasing the suppression pool mass increases the containment heat sink and therefore decrease the containment pressurization rate. Studies performed by PP&L show that the time to containment failure can be increased up to a factor of 3.

The time to containment failure can be estimated using the following equations.

Case 1, no supplemental mass addition to the pool:

$$\text{Specific Enthalpy} = \frac{M_o h_o + M_s h_s}{M_o + M_s}$$

where

M_o = initial pool mass, #

M_s = mass of steam condensed in pool, #

h_o = initial pool specific enthalpy, BTU/#

h_g = specific enthalpy of steam, BTU/#

Case 2, supplemental mass added to pool:

$$\text{Specific Enthalpy} = \frac{M_o h_o + M_a h_a + M_s h_s}{M_o + M_a + M_s}$$

where

M_a = mass of water added to the pool, #

h_a = specific enthalpy of added water, BTU/#

The maximum allowable final pool mass is 15.2×10^6#.

The specific enthalpy is that of the bulk suppression pool. The vapor pressure, however, is determined by the pool surface temperature. If the pool stratifies the pool temperature could be 20 to 30°F higher than the bulk. This is accounted for by adjusting the allowable bulk temperature to account for the stratification and then use the specific enthalpy of the adjusted bulk temperature in the evaluation of the time to containment failure. The time to containment failure is obtained from the integrated decay heat curve.

As stated above, the calculation of time to containment failure can be done with simple hand calculations in this case where no significant hydrogen generation has occurred. The vapor pressure of the suppression pool is easily calculated as a function of time from the equations above and the decay heat curve of Appendix 1. Some care must be used however in the treatment of containment non-condensibles.

In the case of LOCA the non-condensibles are driven from the drywell into the wetwell by the flow of steam through the LOCA vents. If reflood of the vessel and subsequent makeup is controlled so that decay heat is subsequently removed from the core by boiling and release of steam through the break to the drywell, the non-condensibles will remain contained in the wetwell air space and the vapor pressure in the drywell will equal the combined partial pressure of the suppression pool vapor pressure and the non-condensibles in the wetwell.

On the other hand, if heat is removed from the core sensibly by supplying a sufficient quantity of water to flood the core and remove heat without boiling, the vapor pressure in the drywell could exceed that of the

suppression pool only by a small amount, depending on the flow rate used, and some portion of the non-condensibles would then flow back into the drywell via the vacuum breakers. A precise calculation of this situation would require use of a transient containment code, but bounding calculations can be made by arbitrary assignment of the non-condensible partitioning.

In the case where LOCA has not occurred and decay heat is removed from the core by steam which is routed to the suppression pool via SRVs, the entire non-condensible inventory will ultimately be driven into the drywell by water vapor pressure above the pool. If drywell temperatures can be approximated, this case also is amenable to hand calculation estimates for time to failure.

SEQUENCES WITH INADEQUATE CORE COOLING

CONTAINMENT INTEGRITY LINKED TO CORE DAMAGE PROGRESSION.

THIS REQUIRES VESSEL AND CONTAINMENT INTEGRITY TO BE EXAMINED SIMULTANEOUSLY

MUST COMPUTE PROBABILITY OF CORE DAMAGE STATES WITH CONTAINMENT FAILURE MODES

- VESSEL INTACT
- VESSEL FAILED
- CONTAINMENT INTACT
- CONTAINMENT VENTED
- CONTAINMENT OVERTEMPERATURE FAILURE
- CONTAINMENT OVERPRESSURE FAILURE

Sequences With Inadequate Core Cooling

In accident sequences where loss of injection capability leads to inadequate core cooling a number of complexities arise. These include

1. metal-water reactions add non-condensibles to the containment free volume,

2. if injection is not recovered, the core damage will progress to the point of reactor vessel failure,

3. if the reactor vessel fails, additional non-condensibles are produced and various critical containment components may be attacked,

4. there is commonality in pumps which can supply the reactor vessel, drywell sprays, and provide suppression pool cooling.

These factors create a series of inter-relationships which are too complex to adequately represent with an event tree. The process of dispositioning the various outcomes of such a sequence is best demonstrated by listing the allowable outcomes of the sequence. These are:

1. core damage with reactor vessel and containment intact,

2. core damage with reactor vessel intact but with containment vented,

3. core damage with reactor vessel intact but with containment overpressure failure,

4. reactor vessel failure with containment intact,

5. reactor vessel failure with containment vented,

6. reactor vessel failure with overtemperature failure of the drywell, and

7. reactor vessel failure and containment overpressure failure.

Reactor vessel failure here implies that core damage has progressed to the point of lower plenum dryout and high temperature failure in the lower head area. In general, venting implies that overtemperature failure has been avoided or will be avoided by drywell spray operation and that overpressure failure will be avoided. In general venting will not prevent drywell overtemperature failure caused by liner attack or suppression pool bypass, and such cases result in unscrubbed releases. It can avoid failures caused by a combination of temperature and pressure and it can scrub the early releases before the debris causes liner attack or suppression pool by pass. Venting will generally prevent containment overpressure failure, except possibly in the case of ATWS events.

While these seven final plant states are considered to be the only possibilities, there are degrees of damage that may be considered. For example, BWRSAR permits characterization of the extent of core damage before recovery of vessel injection and core stabilization at several stages of the

damage progression before reactor vessel failure must occur. Similarly we believe that an analogous situation exists for the core debris pour from the failed reactor vessel either in terms of the points at which the debris pour can be terminated or the point at which the debris can be stabilized on the drywell floor, or both. We do not currently have experience with the CONTAIN code so that we cannot state any firm conclusions in this regard.

Consideration of the seven final plant states which are allowable and the various degrees of damage progression which may be derived for them clearly indicates that the probability of a given final state depends on the probability of one or more successful operator actions within a given period of time. These actions involve:

1. restoring vessel injection,

2. initiating drywell sprays,

3. initiating suppression pool cooling, and

4. operation of the containment vent.

The times by which these actions must be taken are determined by the calculation of the accident transient and by specification of the actions needed to avoid a given level of plant damage. The process by which the conditional probabilities are derived is presented in Appendix 6.

CORE DAMAGE PROGRESSION

BUNDLE BLOCKAGE MODEL

RESTORE CORE COOLING PRIOR TO ANY CORE NODE EXCEEDING 3000°F

- BOILDOWN

  48 MINUTES AFTER 2/3 CORE HEIGHT
  16 MINUTES AFTER 2200°F

- RAPID BLOWDOWN

  14.4 MINUTES AFTER CORE DRYOUT

COMPONENT RELOCATION MODEL

RESTORE COOLING PRIOR TO BOTTOM HEAD DRYOUT

- BOILDOWN

  158 MINUTES AFTER 2/3 CORE HEIGHT
  112 MINUTES AFTER 2200°F

- RAPID BLOWDOWN

  83.9 MINUTES AFTER CORE DRYOUT

TIMES OBTAINED USING BWRSAR

Core Damage Progression

The determination of the initiation of core damage is a relatively simple matter with very little uncertainty involved in the case of inadequate core cooling. The simple hand calculations described or referenced in previous view graphs are quite simple, and a number of transient programs exist which also provide reliable results. Description of the core damage progression is not nearly so straight forward. There are, in fact, alternative models for this process which produce drastically different results. In this presentation we shall discuss only the two models with which PP&L has had experience, MAAP and BWRSAR.

The MAAP code uses only two temperatures to characterize a core node, one for the fuel, clad, and channel box, and a second for the control rod. The relocation of material of a core node only begins when the node temperature reaches a pre-specified temperature. Relocation of different materials at different temperatures is not described. It is further assumed that for some degree of relocation the channel becomes completely blocked and neither water or steam can any longer cool the bundle. The blockage also causes the metal-water reaction to terminate from lack of a steam supply.

The molten material eventually relocates to the lower plenum where the melt quickly attacks instrument or CRD penetrations and causes failure of the penetration and flow of the melt from the vessel.

The criterion given by IDCOR for stopping the damage progression was that the core should be reflooded before any core node reaches 3000°F. Shortly after some part of the core reaches 3000°F, it is assumed that the core is essentially uncoolable because of the blockage phenomenon. This criterion imposes a very severe limit on the time available to restore vessel injection. Further, if vessel injection is restored after the 3000°F limit, the melt progression and vessel failure cannot be prevented. For Susquehanna accident sequences we determined that we could save the reactor vessel in only 70% of all core damage sequences.

We believe that the MAAP core damage progression models and material relocation models are not realistic, however, and that the criterion imposed for core stabilization (preventing reactor vessel failure) is overly conservative.

In 1987 PP&L began trial calculations using the BWRSAR code developed at ORNL. The models used in BWRSAR seemed to provide a more realistic (although still crude) representation of the process of core heatup and the subsequent relocation of materials. It uses a four temperature model for each core node (UO_2, clad, channel box, and control rod) and provides the capability for describing the different temperatures at which various materials initiate relocation. The model also does not presume channel blockage as a consequence of melt relocation, and this difference does result in much greater degrees of metal-water reaction. Considering the nature of the BWR core, we believe this representation is more credible than a model which presumes a fully coherent blockage which cuts off all flow of steam and water.

The use of multiple relocation temperatures allows a description of heat up and melt relocation of control rod and channel box materials while the fuel rods remain in place (consistent with the results of the DF-4 experiment).

The UO_2 material, clad by a thin ZrO_2 sheath, remains in place after the clad zirconium has all reacted with steam or relocated downward by melting. The UO_2 rods continue to heat up and eventually lose their strength and collapse in a mostly solid state to the core support plate. Depending on the axial power profile and the nature of the accident sequence the core plate may already have failed due to water boil off and heatup to the failure temperature by falling control rod, channel box and clad material. If not, the collapse of the UO_2 structures quickly leads to this result, and in either case, the UO_2, and other debris, fall into the lower plenum in an essentially solid debris form and are quenched by the water in the lower plenum. The resulting debris bed is then considered to remain coolable until dryout.

This criterion, that the debris bed remains coolable until dryout, results in an allowance of much longer time period to regain vessel injection capability. Equally important, it is not necessary to reflood the core region to do this since the debris to be cooled is in the lowest part of the reactor vessel.

This allows the reactor vessel to be judged as saved with a much smaller injection flow capability (a few hundred gpm) much later than for the MAAP criterion. Preliminary evaluations have lead us to expect a 99% rate of saving the reactor vessel using BWRSAR and its criterion as opposed to 70% using MAAP.

This difference has far reaching implications for containment challenges since less than 1% of all core damage sequences will result in core debris threatening containment integrity.

The dramatic differences in these two models is illustrated by the view graph. The results of a BWRSAR run has been used here to derive the timing for MAAP, but this provides a valid comparison since material relocation has not occurred in either model when a core node temperature of 3000°F is reached and the heat transfer models used are essentially equivalent.

In the case of simple boildown calculations the gain is 96 minutes and in the case of rapid blowdown the gain is nearly 70 minutes. The conditional probability of regaining vessel injection in that increment of time approaches unity and the amount of flow required is much less which allows success with low flow systems such as the fire pumps.

CONTAINMENT INTEGRITY WITH VESSEL FAILURE

1) HEAD FLANGE FAILURE

2) DRYWELL LINER MELT THROUGH

3) LOSS OF VAPOR SUPPRESSION FROM DOWNCOMER MELT THROUGH

4) OVERPRESSURE FAILURE DUE TO LOSS OF DECAY HEAT REMOVAL

5) PEDESTAL ABLATION

Containment Integrity with Vessel Failure

If vessel integrity cannot be guaranteed, then challenges to containment integrity from core debris on the drywell floor must be considered. Five failure modes are postulated. These are:

- drywell head flange failure,
- drywell liner melt through,
- loss of vapor suppression from downcomer melt through,
- overpressure failure due to loss of decay heat removal.
- pedestal failure from ablation,

Each of these are discussed below.

As the core debris pours onto the drywell floor from the reactor vessel it spreads until it freezes. Water vapor is driven out of the drywell floor concrete as it heats up. This vapor reacts with zirconium in the core debris generating H_2 gas and heat. The debris also reacts with the concrete to produce CO_2 gas. These hot gases transfer their heat to the drywell liner. In Mark I and II containment the drywell liner and the containment head are flanged together with a seal to form a pressure boundary. Test data in NUREG/CR-4977 indicates that this seal fails catastrophically at 730°F. Thus if the convective heat transfer from the hot gases and radiation heat transfer from debris bed cause the seal to exceed 730°F, it will fail and containment integrity will be lost in the drywell.

If the debris does not freeze or remelts after freezing, it will spread until it intersects the drywell liner. If the heat flux from the debris bed to the liner is sufficient the liner will fail. In steel containments, integrity will again be lost in the drywell. In steel lined concrete containment it represents a loss of the containment pressure integrity and thus some fission products may escape.

If the debris spreads to and fails the LOCA downcomer vents or SRV penetrations in the Mark II containment, the drywell and wetwell air space will communicate, thus vapor suppression will be lost. As the core debris penetrates the downcomer and falls into the suppression pool, large quantities of steam will be generated. In the absence of vapor suppression the pressure will rapidly rise until the containment fails on high pressure. This failure mode represents a containment failure with suppression pool bypass.

If the heat flux into the containment structures is insufficient to cause a loss of containment integrity, then a challenge from overpressure nevertheless still exists. The reaction of the debris with water vapor and concrete produces H_2 and CO_2 gas. Vapor may also be produced if water is used to cool the debris. The presence of additional noncondensible gas reduces the amount of water vapor pressure that can be accommodated prior to overpressure failure. Therefore, decay heat removal must be established earlier if containment overpressure failure is to be avoided.

Finally as the core debris pours from the vessel it intersects the lower pedestal. Pedestal ablation occurs, and could be so significant that the pedestal could no longer support the reactor vessel. If this is the case the collapse of the vessel would have to be investigated.

ISSUES EFFECTING CONTAINMENT INTEGRITY

- o CONCRETE COMPOSITION

- o PRESENCE OF WATER ON THE DRYWELL FLOOR

- o AMOUNT OF UNREACTED Zr

- o RATE OF POUR OUT OF THE BOTTOM HEAD

Issues Effecting Containment Integrity

Four issues influence the behavior of core debris on the drywell floor. These are:

- the concrete composition,
- the amount of unreacted zirconium in the pour,
- the presence of water on the drywell floor, and
- the rate of pour out of the bottom head.

A discussion of each of these issues follows.

Two types of concrete have been evaluated by the USNRC. These are limestone common sand and high limestone concrete. Limestone common sand concrete is typical of most concretes. It consists of 1 part cement, 2 parts quartz sand, and 3 parts limestone aggregate. In high limestone concrete the quartz sand is replaced with a limestone based material. The limestone common sand concrete has a much lower ablation temperature. This results in a greater mass of concrete ablating which dilutes the core debris. If the debris is low in unreacted zirconium, the debris will freeze. If the core debris is rich in unreacted zirconium the water vapor degassing from the concrete will react with the zirconium and generate sufficient heat to prevent the debris from freezing. If the concrete is high limestone, the amount of ablation is much less. In this case dilution does not occur and the debris remains liquid independent of the amount of unreacted zirconium.

The presence of water on the drywell floor has a significant impact on the containment's response to core debris. For the case of limestone common sand concrete, the addition of water to the drywell floor provides sufficient cooling to prevent head flange failure, drywell liner melt through and downcomer melt through. Some form of decay heat removal is still required to prevent overpressure failure.

Finally the rate at which the debris pours out of the vessel has a significant effect on the containment response. The core damage progression model directly influences the core debris pour rate. In the bundle blockage model the core first melts behind the postulated block. It finally melts through the block rapidly pour to lower plenum, melts the bottom head and pours onto the drywell floor.

In this model a major fraction of the entire core is on the floor in a matter of minutes. Freezing cannot be expected prior to intersecting metal components in the drywell. Therefore, containment failure can be expected to occur if the bundle blockage model is used.

In the component relocation model the time frame for damage is much greater. In this model it takes about 40 minutes for the debris metals and several hours for the UO_2 to pour out of the bottom head. Heat transfer from the debris to the containment structure can be expected. This results in a much more favorable containment performance.

STATION BLACKOUT EXAMPLE USING COMPONENT RELOCATION MODEL

INITIAL CONDITIONS

PERFORMANCE REQUIREMENT FOR CORE INTEGRITY

PERFORMANCE REQUIREMENT FOR CONTAINMENT INTEGRITY VESSEL INTACT

PERFORMANCE REQUIREMENT FOR VESSEL AND CONTAINMENT INTEGRITY

THE REACTOR AND PRIMARY CONTAINMENT INITIAL CONDITIONS

REACTOR PROCESS PARAMETERS	VALUE
WATER LEVEL	28 2/5 INCHES ABOVE VESSEL ZERO
PRESSURE	1034.7 PSIA
POWER	DECAY HEAT

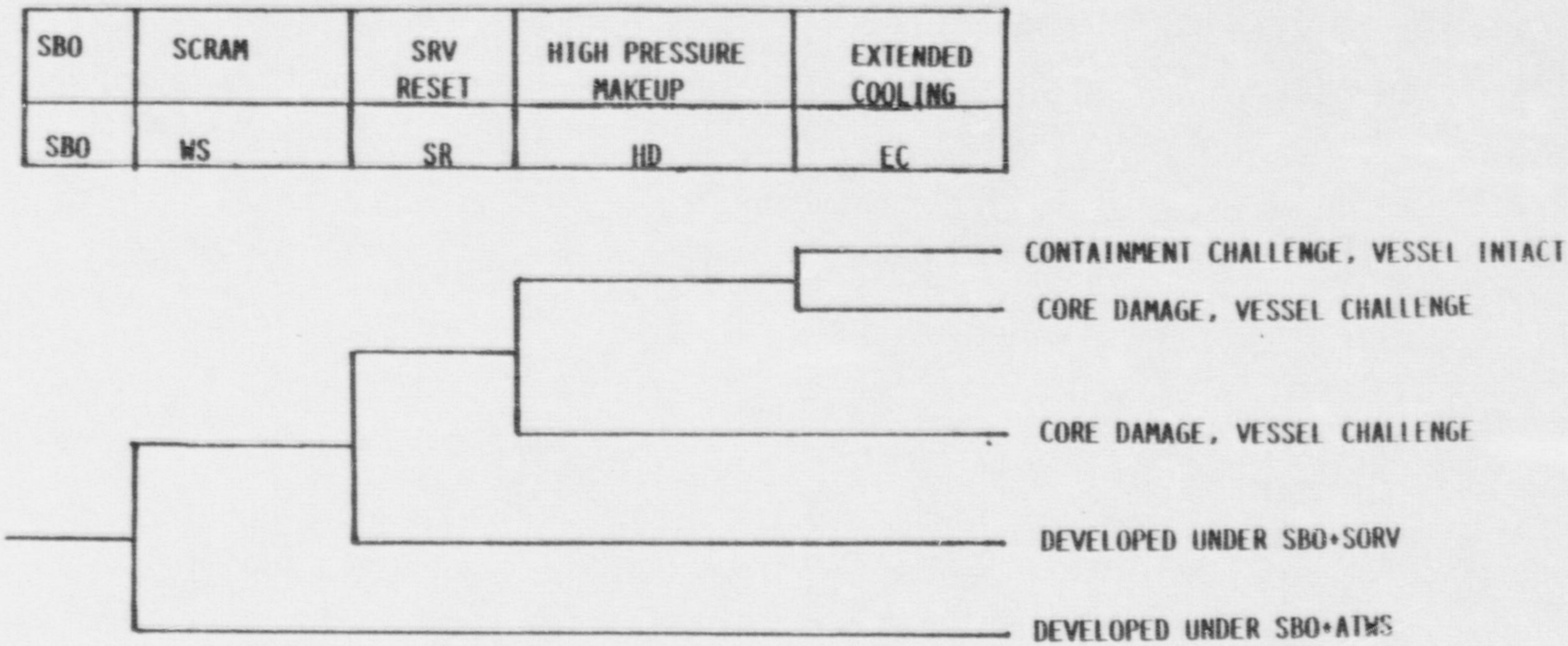
PRIMARY CONTAINMENT PARAMETERS	
DRYWELL PRESSURE	15 PSIA
DRYWELL TEMPERATURE	$130^{o}F$
SUPPRESSION POOL TEMPERATURE	$90^{o}F$
SUPPRESSION POOL LEVEL	22 FEET

The Reactor and Primary Containment Initial Conditions

The initiating event is a station blackout.

The reactor and primary containment process parameters are listed in view graph. The reactor data represents the state of the reactor at the conclusion of the feedwater coast down 36 seconds after scram. The containment process parameters are obtained from normal operational data. The containment parameters have a fair amount of inertia and do not change much over 36 seconds.

PERFORMANCE REQUIREMENTS SUFFICIENT TO ENSURE BARRIER INTEGRITY



Performance Requirements Sufficient to Ensure Barrier Integrity

The first barrier to consider is the clad. The clad can suffer damage in two ways: loss of mechanical integrity or oxidation. A different performance requirement is applied to avoiding each of these failure modes.

Mechanical Failure

Mechanical clad damage is avoided if the reactor is subcritical or if the water level is maintained above the top of the upper plenum and design pressure is maintained if scram has failed. If either of these requirements are satisfied then mechanical clad damage is not expected. Thus the first question to consider is whether the reactor is scrammed or not. If the answer is yes then clad oxidation must be considered. If a scram has not occurred then the reactor water level and pressure must be explored. In this illustration only the scrammed case is developed. Therefore, oxidation is considered next.

Oxidation

Oxidation of the fuel clad, the channel boxes, and the control blades occurs when cooling is lost. Cooling is ensured as long as adequate makeup flow is supplied to the reactor vessel prior to the water level dropping below the jet pump throat. The requirements for adequate makeup depends upon the time after shutdown. The time required for the water level to fall to the jet pump throat elevation depends upon whether the water is lost through blowdown or boildown. During a station blackout no pipes are assumed to be broken. However, the relief valves cycle and can stick open resulting in a reactor vessel blowdown. If the valves cycle without failure then the accident timing is set by boildown. Thus the second question to consider is the status of the relief valve. Cases where the relief valves stick open are not developed in this illustration. Therefore, the boildown case is considered.

Boildown

The minimum flow required to ensure adequate core cooling depends upon the time after shutdown and the water level when the boildown commences. Examining the boildown plots and the initial conditions, the following performance requirement is developed,

> Provide 250 gpm at 1000 psia to the reactor vessel within 50 minutes of the reactor scram.

The following equipment is identified in the emergency procedures that satisfies this requirement:

Feedwater,
RCIC,
HPCI

All of this equipment operates at high pressure, therefore the success criteria for high pressure injection becomes: provide flow from any of these pumps by 50 minutes. Reactor isolation occurs in a station blackout and therefore the feedpumps are unavailable due to lack of motive steam and condensate pump inoperability.

If this success criterion is satisfied, then core cooling must be satisfied in the long term. Cooling can be performed by either the RCIC or HPCI systems or should they fail the diesel fire pump if the reactor is depressurized. The SSES procedural response to station blackout requires the reactor be depressurized to 150-200 psig. Thus, should the high pressure pumps fail, the fire pump is available for injection. Both the high pressure pumps and the relief valves which maintain the reactor at low pressure require DC control power. DC power is normally provided by station batteries. These batteries however are depleted within about 4 hours. This problem has been circumvented at SSES by providing 480 V AC power from a mobile generator to the battery chargers. The chargers then feed the DC loads. If the generator and at least one injection source are successful then core cooling is assured. With core cooling assured the vessel is not challenged, and the containment must be considered.

If the mobile generator fails, the batteries will deplete, injection will be lost, and boildown will commence. If adequate cooling is not restored prior to the downcomer water level falling below the jet pump throat, core damage is assumed to occur. Adequate cooling is again defined using the boildown plots. If injection is assumed to be lost at 4 hours due to battery depletion, the water level will reach the jet pump throat in 2.5 hours. The makeup flow rate required at 6.5 hours (4 hours + 2.5 hours) is 145 gpm. The performance requirement becomes.

Provide 145 gpm to the reactor by 6.5 hours.

All the equipment identified in the procedures requires either AC motive power or DC control power. Thus with a station blackout and loss of DC power core damage will occur unless AC power is recovered. The success criteria for long term cooling includes recovery of AC power before 6.5 hours. Failure to provide core cooling will result in a vessel challenge.

If the high pressure injection pumps fail, the procedures instruct the operator to depressure the reactor and use low pressure pumps to provide cooling. Review of the blowdown analysis section results in the following performance requirement.

> At 50 minutes rapidly depressurize the reactor pressure vessel and inject 800 gpm into the reactor vessel.

The following equipment is identified in the emergency operating procedures which satisfies this requirement.

<u>Depressurization</u>

Safety Relief Valves
Turbine Bypass Valves

<u>Injection</u>

Condensate pump
Core Spray pumps
Low pressure coolant injection pumps

Success requires both depressurization and injection. The injection sources identified all require AC power. Therefore AC power must be recovered to

satisfy this success criterion, and AC power must be recovered within 30 minutes or the vessel will be challenged.

The information discussed above is depicted in the event tree. This event tree dispositions the status of the reactor core integrity and identifies if additional barrier challenges exist. The first sequence is an extended station blackout with no additional failures. This sequence does not challenge the core integrity. However, unless decay heat is rejected from the containment it will fail on overpressure. This challenge to the containment will be discussed next. The remaining two sequences result in a challenge to the vessel integrity. These are discussed following the discussion on containment.

Containment Integrity Vessel Intact

System Arrangement	Method of Cooling Reactor	Method of Cooling Containment	Power Source	Cooling Water	Ultimate Heat Sink
1. Main Condenser	latent	N/A	13 kv system	SW	cooling tower
2. RHR System					
a. Shutdown cooling	sensible	N/A	13 kv system or 4 kv ESS	ESW	spray pond
b. Suppression pool	latent	sensible	13 kv system or 4 kv ESS	ESW	spray pond
3. RWCU blowdown mode	sensible	N/A	13 kv system	SW or ESW	cooling tower
4. Vent	latent	latent	N/A	N/A	direct to the environment

Containment Integrity Vessel Intact

In cases where the vessel integrity is preserved, containment integrity can still be challenged by pressurization due to storage of decay heat in the suppression pool. Decay heat removal is normally accomplished by rejecting heat through the main condenser. If the main condenser is unavailable, energy is stored in the suppression pool and then transferred to the spray pond by the RHR system. If the RHR system is also unavailable, a blowdown path from the reactor to either the main condenser or liquid radwaste exists through the reactor water cleanup (RWCU) system. Addition of mass to the suppression pool during the blowdown will result in reduced peak containment pressures, but, even without mass addition, the peak pressure is below the estimated containment rupture pressure. Finally procedures instruct operators to vent the containment prior to exceeding its design pressure. Venting the wetwell represents a loss of containment integrity and is used only as a last resort. It is preferable to containment failure in that the suppression pool is used to scrub the release.

These methods of rejecting decay heat are both redundant and diverse. The diversity spans both the type of equipment and the phenomena used to transfer heat, as described in the view graph.

Performance for Wetwell Vent

		Time to Reach 30 psia in the Wetwell (hr)	Drywell Gas Temperature (^oF)	Partial Pressure of non-Condensible Gas in the Drywell (psia)	Wetwell Water Temperature (^oF)	Wetwell Gas Temperature (^oF)	Partial Pressure of non-Condensible Gas in the Wetwell (psia)	Wetwell Water Mass (lbm)
1.	1 SRV Blowdown at 4 hours, no mass addition	10.9	190	20.6	220	220	17.3	9.1×10^6
2.	1 SRV Blowdown at 15 minutes, no mass addition	11.9	187	21.3	227	223	10.5	9.1×10^6
3.	1 SRV Blowdown at 15 minutes, add mass to wetwell	16.4	192	21.4	208	190	20.5	1.3×10^7
4.	1 SRV Blowdown at 15 minutes, add mass through drywell sprays	23.9	210	10.9	220	220	3.6	1.1×10^7

Performance Requirement for Wetwell Vent

Under Station Blackout conditions all decay heat removal (DHR) systems requiring AC power are unavailable. If the wetwell vent can be operated without AC power, it may be considered for the DHR function. At PP&L the vent is considered to be a last ditch option so that vent opening is delayed as long as possible. At the present time the vent path available involves HVAC duct work which can withstand internal pressures of only 10 psid before major failure. This allows a containment pressure of only 30 psia before vent opening is required since we will not consider venting massive quantities of water vapor into the reactor building.

The view graph presents the time available for opening the vent before containment pressure reaches 30 psia.

We would prefer to recover AC power to permit use of the RHR system as opposed to venting. The view graph clearly demonstrates that there should be a very good chance of AC power recovery before venting becomes necessary. This time can be greatly extended if water is deliberately added to the wetwell. Further, if this water is added via drywell sprays, which keep the drywell temperature down, nearly 24 hours are available before venting is required.

The cases which must be considered, then, in dispositioning end point 1 of the SBO event tree are:

1) AC power recovery with no water addition before 11.9 hours,

2) venting at 11.9 hours with no water addition,

3) recovery of AC power before containment failure with no venting and no water addition,

4) failure to recover AC power before containment failure with no venting and no water addition,

5) AC power recovery with water addition before 23.9 hours (assuming use of drywell sprays),

6) venting at 23.9 hours with water addition,

7) recovery of AC power before containment failure with no vent but with water addition, or

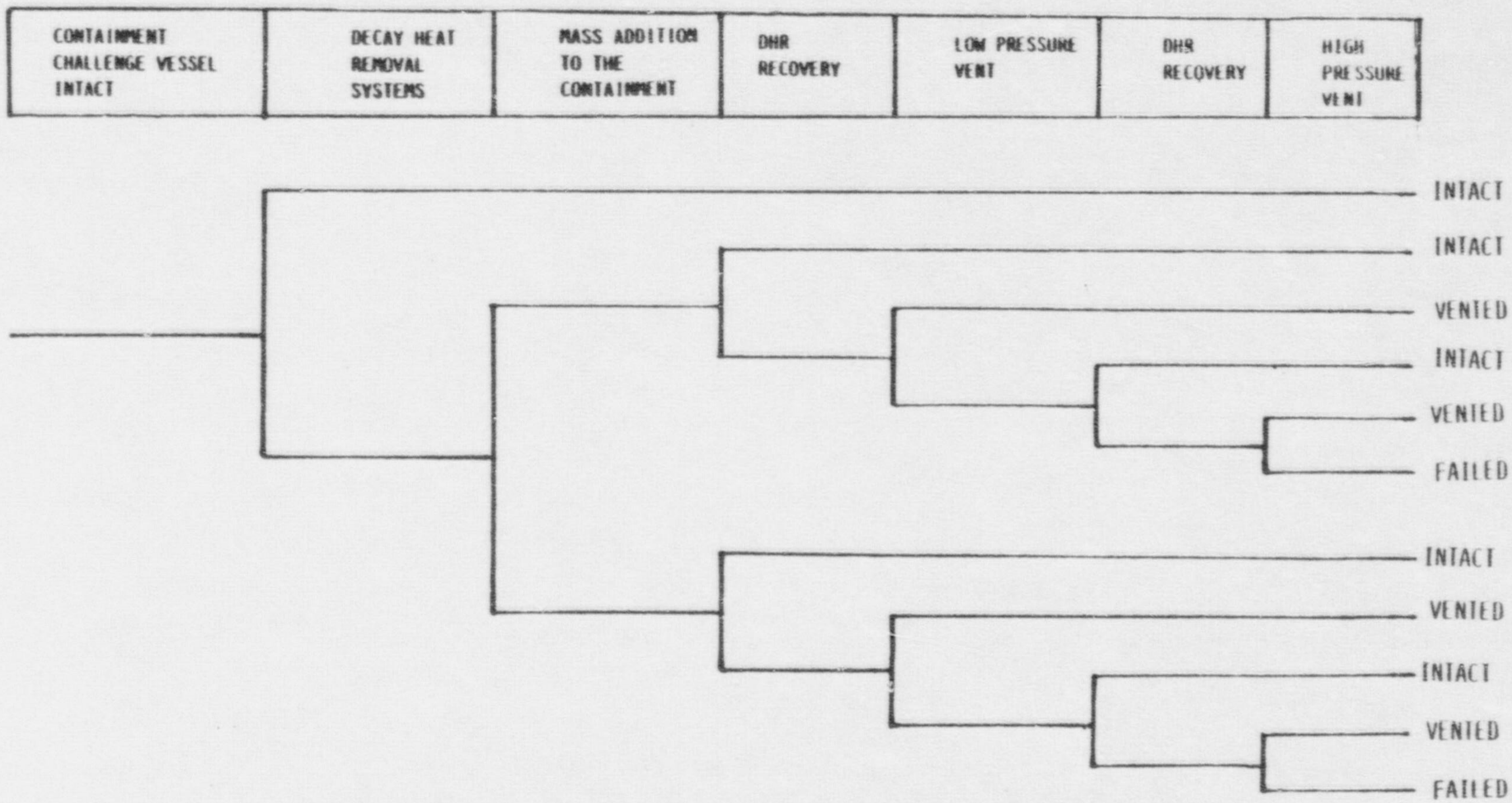
8) Failure to recover AC power before containment failure with water addition but no vent.

If it is necessary to consider the case where water is added directly to the suppression pool additional cases must be considered. The specification of success criteria determine what combinations will be considered.

In order to quantify the equations for conditional probability that are derived it is necessary to have an AC power recovery curve (Appendix 7) and transient calculations to determine the time of containment failure for the three potential water addition cases.

The risk analysis program being developed will only need as input the success criteria for various final plant damage states, the time constraints for the various actions, the AC power recovery curve, and the success rate for venting when necessary.

CONTAINMENT CHALLENGE; VESSEL INTACT



Containment Challenges; Vessel Intact

An event tree organizing these questions is illustrated in the view graph. The containment status is identified as: intact, vented, or failed. The latter two cases represent a loss of containment integrity. In the vented case, however, credit for suppression pool scrubbing can be taken.

At SSES the venting issue is under evaluation by management. No credit is taken for venting for this reason.

In the situation being considered, Station Blackout, no AC power is available. Review of previous view graph indicates that all the decay heat removal systems at SSES require AC power. Therefore, no decay heat removal systems are available. The next question to consider deals with the addition of mass to the suppression pool. To answer this question the emergency operating procedures must be consulted. At SSES the procedure instructs the operator to spray the drywell on high pressure and also to align the fire protection system to the RHRSW system to be used for injection. Therefore, the success criteria for this event becomes:

Add mass to the containment using the fire pump.

Finally, if containment failure is to be avoided, decay heat removal must be established. This requires the recovery of AC power. The amount of time available for AC power recovery depends upon the water mass added to the pool. If no mass is added to the pool beyond that of condensed steam from the SRV's, then the containment failure pressure is reached in about 1 day. If the maximum amount of mass is added, then this time can be extended to 3 days. Therefore, at SSES the success criteria for this event becomes:

> restore AC power within 1 day if no mass is added and 2.5 days if maximum mass is added.

Failure to satisfy this criterion will result in containment failure.

PERFORMANCE REQUIREMENT FOR VESSEL AND CONTAINMENT INTEGRITY

VESSEL INTEGRITY

PROVIDE 120 GPM PRIOR TO BOTTOM HEAD DRYOUT AT 3.5 HOURS

CONTAINMENT INTEGRITY - OVERTEMPERATURE

PROVIDE 500 GPM TO DRYWELL PRIOR TO VESSEL PENETRATION AT 5.5 HOURS

CONTAINMENT INTEGRITY - OVERPRESSURE

PROVIDE DECAY HEAT REMOVAL PRIOR TO EXCEEDING THE CONTAINMENT ULTIMATE STRENGTH

Performance Requirement for Vessel and Containment Integrity

The integrity of the reactor vessel and primary containment are linked through the gross generation of noncondensible gas. Performance requirements, however, can be written which will prevent the phenomena challenging the integrity of each the last two barriers.

In the case of the reactor vessel, the stabilization criterion requires injection prior to bottom head dryout. Prior to bottom head dryout the debris bed is quinched. Therefore, it is only necessary to reject decay heat. Based upon this criterion the performance requirement to ensure vessel integrity becomes:

> Provide 120 gpm to the reactor vessel prior to bottom head dryout at 3.5 hours.

If sufficient injection is not provided to the bottom prior to dryout the reactor vessel is postulated to fail and the core debris will pour onto the drywell floor. Many containment challenges associated with core debris interactions with the containment have been discussed. Two different phenomena challenge the containment integrity once the vessel has been breached. These are overtemperature and overpressure failure. Stabilization criteria for each case are reviewed below.

In the case of overtemperature failure the presence of sufficient water on the drywell floor is a sufficient condition to preclude overtemperature. At this point PP&L is still evaluating the containment response in vessel failure cases. However if water is injected onto the drywell floor prior to vessel penetration failure it is reasonable to assume that the debris will be quinched. Therefore the following criterion is suggested as an interim stabilization criterion for containment overtemperature failure.

> Provide 500 gpm to the drywell prior to vessel penetration of 5.5 hours.

The 500 gpm is the SSES drywell spray flow rate.

Finally, if the containment temperature is stabilized below the failure point, the containment pressure must still be controlled. The criteria for overpressure failure is unaffected by vessel penetration. However, the timing associated with the event is changed. Core concrete interaction produce large volumes of noncondensible gas which reduces the amount of vapor generation allowable prior to overpressure failure. The following criterion is suggested as a stabilization criteria for containment overpressure failure.

> Provide decay heat removal prior to exceeding the containment ultimate strength.

At this point PP&L has not performed sufficient calculation to estimate the partial pressure of the noncondensible gas generated during core concrete interactions. Therefore, timing information is undetermined.

ACCIDENT SEQUENCE QUANTIFICATION

ACCIDENT SEQUENCE QUANTIFICATION

DEFINITION OF BOUNDARIES

SUPPORT STATE METHOD

CONSTRUCTION OF FUNCTIONAL FAULT TREES

DEVELOPMENT OF SYSTEM DEPENDENCY MATRICES

DEVELOPMENT OF SUPPORT STATE

ACCIDENT SEQUENCE QUANTIFICATION

STATION BLACKOUT EXAMPLE

Accident Sequence Quantification

Quantification of the accident sequences defined by the event trees requires that the probability of the event tree top events be computed. These probabilities are then combined with the initiating event frequency to yield the accident sequence frequency. Discussion of the process requires a common set of terms and an outline of the support state method. Therefore, these topics are discussed first in this section. With a common understanding of the terms and the method, details of the calculation can be discussed. This discussion includes;

the construction of functional fault trees,

the development of system dependency matrices,

the development of the support state, and the

the accident sequence quantification,

These topics will follow the discussion of definition and methodology.

DEFINITION OF BOUNDARIES

FRONT-LINE SYSTEM BOUNDARIES

COMPONENTS ARE GROUPED WITH FRONT-LINE SYSTEMS IF THEY ARE DEDICATED TO THE FRONT-LINE SYSTEMS OPERATION

SUPPORT SYSTEM BOUNDARIES

COMPONENTS ARE GROUPED WITH SUPPORT SYSTEMS IF THEY ARE REQUIRED TO SUPPORT MULTIPLE SYSTEMS

DIVISION BOUNDARIES

COMPONENTS ARE GROUPED WITH A DIVISION IF THEY ARE REQUIRED TO SUPPORT AN ENTIRE DIVISION OF A SYSTEM

CHANNEL BOUNDARIES

COMPONENTS ARE GROUPED WITH A CHANNEL IF THEY ARE ONLY REQUIRED TO SUPPORT A CHANNEL IN THE DIVISION OF A SYSTEM

Definition of Boundaries

Front-line and Support System Boundaries

Prior to constructing plant models, boundaries must be defined to assure components are properly treated in the models. When drawing boundaries two classes of systems considered: front-line and support systems. Front-line systems perform a specific safety function such as injecting water into the vessel. Support systems supply necessary facilities to enable a front-line system to perform its safety function, e.g., AC power provides motive force to perform the safety function. A failure is grouped with the front-line system if its failure only causes the front-line system to fail. A failure is grouped with was support system if its failure causes more than one front-line system to fail. As an example, if the circuit breaker for the RHR injection valve failed it would be collected with the RHR system because it only affected the RHR system. If, however, the motor control center failed, it would be collected with the AC power system, because its failure would cause other systems to become unavailable.

Divisions and Channels

Each SSES unit may be considered as having two divisions of safety equipment. One division is sufficient to perform the intended safety function should a failure disable the second division. Some systems, such as RHR, have redundancy within a division. This intra-divisional redundancy is called a channel. Whereas channel usually refers to electrical systems, in this presentation it is also used for mechanical systems. The PRA model is structured in terms of divisions and channels. The components are therefore grouped along these lines. Failures are grouped by channel if they affect only one channel in a division. A failure is grouped by division if the failure affects an entire division. As an example, failure of the breaker to the "A" RHR injection valve is grouped with Division I of RHR. A failure of the "A" RHR pump, however, is grouped with channel "A" of RHR.

SUPPORT STATE METHOD

- o CONSTRUCT FUNCTIONAL FAULT TREES FROM FRONT-LINE FUNCTIONS
- o IDENTIFY SYSTEM COMMON IN FUNCTIONAL FAULT TREES
- o COMPUTE FRONT-LINE SYSTEM FAILURE PROBABILITY FOR EACH SUPPORT SYSTEM FAILURE COMBINATION CONSIDERED.
- o GROUP SUPPORT SYSTEM FAILURE COMBINATIONS BY SIMILAR IMPACT ON THE FRONT-LINE FUNCTIONS
- o PARTITION INITIATING EVENT BY SUPPORT STATE
- o COMPUTE ACCIDENT SEQUENCE FREQUENCY BY ACCIDENT SEQUENCE.

Support State Method

The event tree is used to link functions which fulfill the performance requirement for barrier integrity. Computation of the accident sequence frequency requires that the failure probability of these functions be estimated. These functions are composed of systems that uniquely or in combination satisfy the performance requirement. The functional fault tree identifies the set of system failures which must occur if the function is not satisfied. Estimates of the function failure probability then becomes the probability of the set of system failures occurring. Failures in these systems can result from loss of either system components or loss of support systems. In the support state methodology, support system failures are treated explicitly. This is performed by computing the conditional probability of the top event for a given support system failure. This is executed for the various combination of support systems. Support system combinations which have a similar impact on the front-line functions are grouped into a support state. The initiating events are partitioned by these support states. The accident sequence frequency for a given support state is then computed using the conditional failure probability of the front-line function for the given support state. This event tree quantification process is summarized with the following rules.

1. Construct functional fault trees for the front-line functions identified in the event trees.

2. Identify support system dependencies common to the functional fault trees.

3. Compute the conditional probability of failure for the front-line function for each support system failure combination considered.

4. Group the support system failure combinations that have a similar impact on the front-line function. These groups are called support states.

5. Partition the initiating event frequency by the probability of each support state.

6. Compute the accident sequence frequency for each support state using the front-line system failure probability computed in step 3.

A discussion of each of these steps follows. The station blackout sequence developed earlier is used for illustration.

REFERENCES ON FAULT TREE ANALYSIS

1. FAULT TREE HANDBOOK, USNRC NUREG-0492, W. E. VESELY, ET. AL., JANUARY 1981

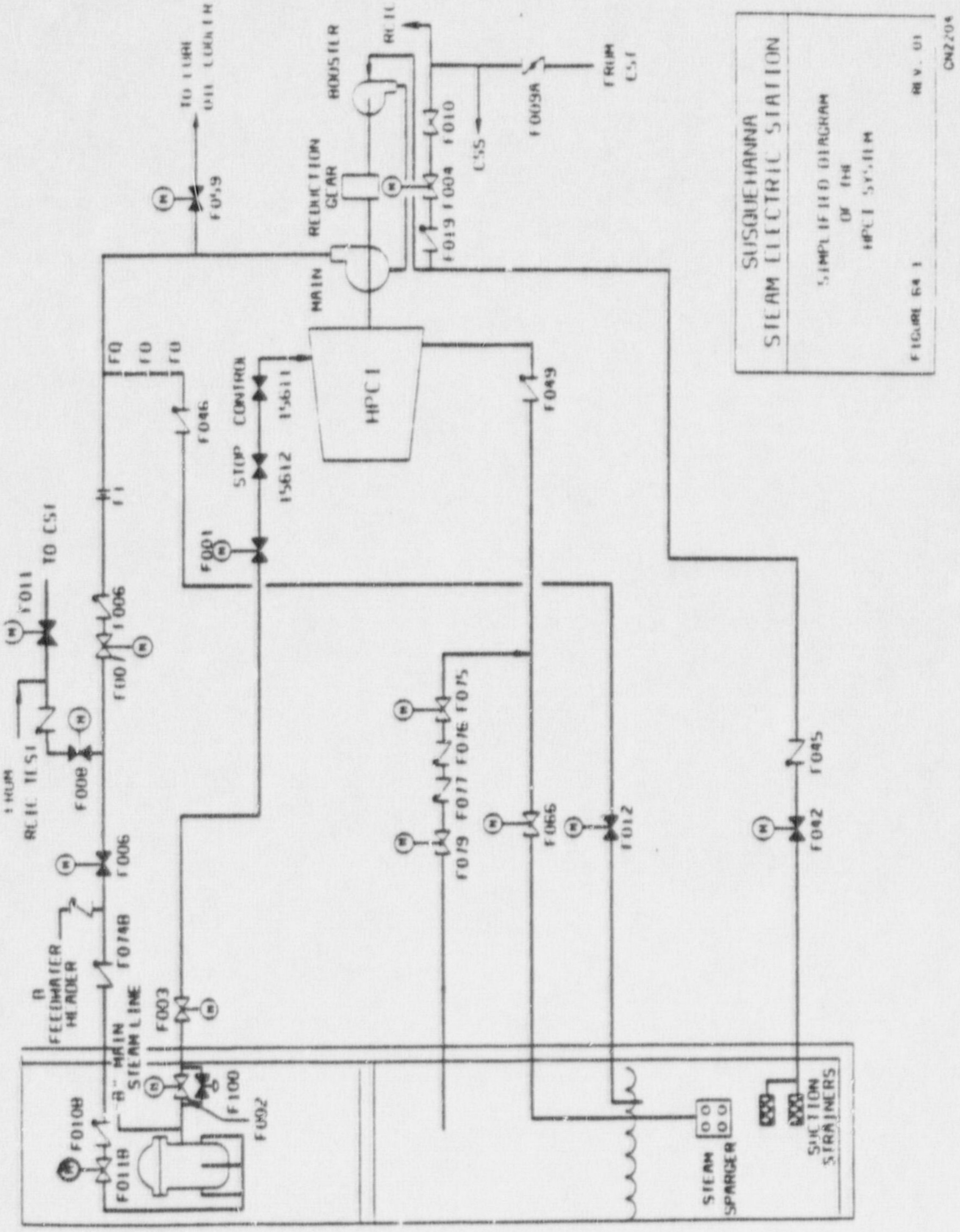
2. STATISTICAL THEORY OF RELIABILITY AND LIFE TESTING R. E. BARLOW AND F. PROSCHAN, TO BEGIN WITH PRESS, 1981

3. RELIABILITY AND RISK ANALYSIS, NORMAN J. McCORMICK ACADEMIC PRESS, 1981

4. DISCRETE MATHEMATICAL STRUCTURES WITH APPLICATIONS TO COMPUTER SCIENCE, J. P. TREMBLAY AND R. MANOLAR, McGRAW-HILL COMPUTER SCIENCE SERIES

References on Fault Tree Analysis

The failure probability of the front-line functions are required when computing the accident sequence frequency. Little failure data exist at the function level, therefore, the function failure probability must be deduced from the function's basic constituents upon which data exist. The fault tree is a deductive logic method traditionally used in PRA. It is a hierarchial structure in which a complex failure is represented in terms of faults in more basic constituents. The statement of faults must be concise and allow for only two outcomes, success or failure. Many references exist on fault tree analysis. The attached Table list a few. Reference 1 of this Table lists concise rules for construction. These references should be consulted if more discussion on fault tree construction is desired. A description of the functional fault tree for high pressure make up, however, is presented to illustrate the process.

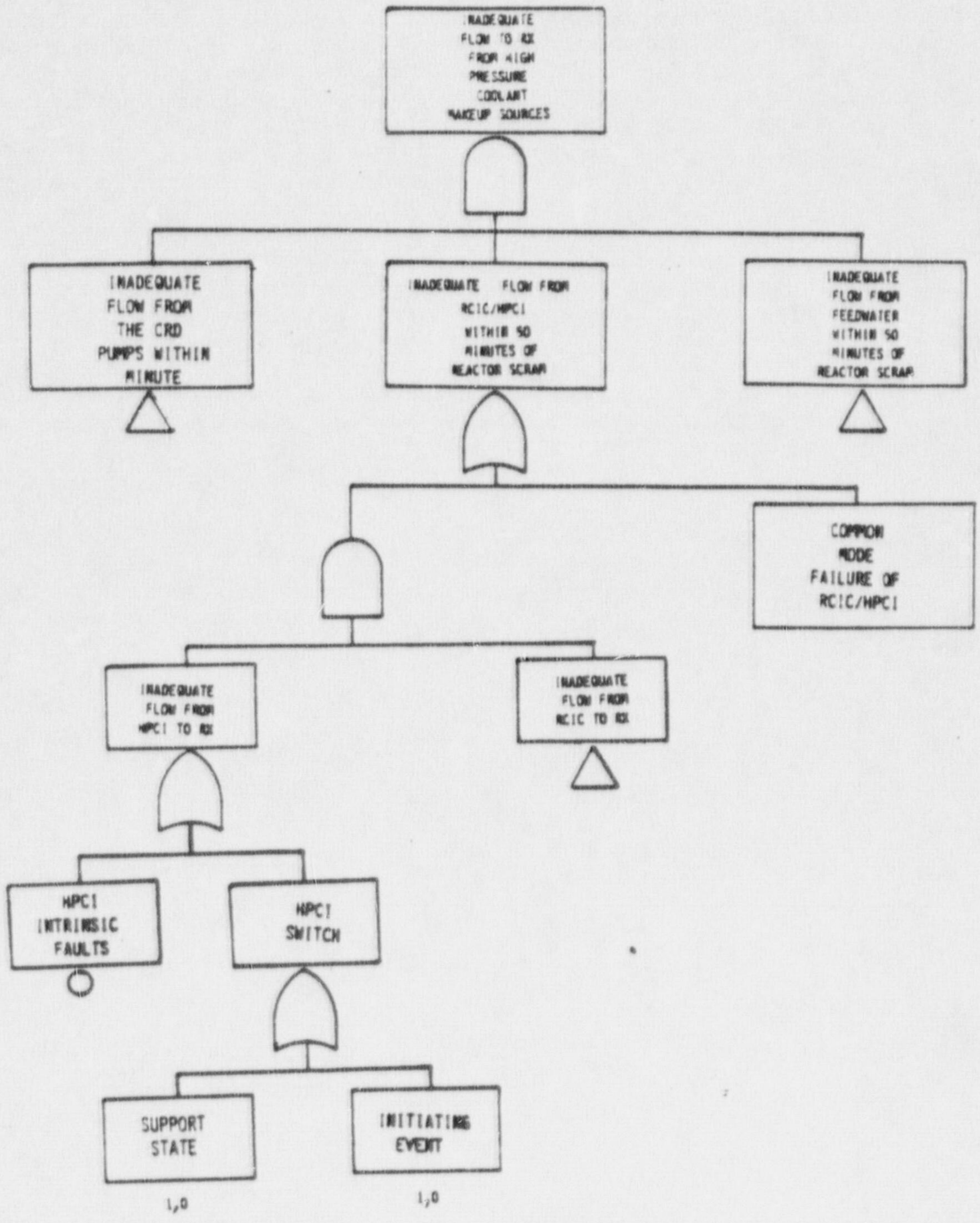

SUSQUEHANNA
STEAM ELECTRIC STATION
SIMPLIFIED DIAGRAM
OF THE
HPCI SYSTEM
FIGURE 64.1
REV. 01
CN2204
A
FEEDWATER
HEADER
"A" MAIN
STEAM LINE
FROM
RCIC TEST
TO CST
TO LUBE
OIL COOLER
REDUCTION
GEAR
BOOSTER
MAIN
HPCI
STOP
CONTROL
RCIC
CSS
FROM
CST
STEAM
SPARGER
SUCTION
STRAINERS
F010B
F011A
F092
F100
F003
F074B
F006
F008
F011
F007
F001
F046
F059
F049
F019
F004
F010
F009A
F079
F077
F076
F075
F066
F012
F042
F045
F002
FE
FO
HV155F012
HV155F011

Simplified Diagram of the HPCI System

The HPCI system is a single loop system consisting of turbine-driven injection and booster pumps, piping, valves, controls, and instrumentation. A simplified flow diagram is shown in Figure 1. The system is designed to pump water into the reactor vessel with a maximum capacity of 5000 gpm for a range of reactor pressures between 150 and 1150 psig. Upon initiation of the HPCI system, a normally closed injection valve, F006, automatically opens, allowing water to be pumped into the reactor vessel through the main feedwater header B. The HPCI turbine is driven by reactor steam. The inboard and outboard HPCI isolation valves, F002 and F003, in the steam line to the HPCI turbine are normally open to keep the piping to the turbine at an elevated temperature, thus permitting rapid startup of the HPCI system. Steam is admitted to the HPCI turbine through supply valve F001, turbine stop valve 15612, and turbine control valve 15611, all of which are normally closed and are opened by a HPCI initiation signal. Exhaust steam from the turbine is discharged to the suppression pool, while condensed steam from the steam lines and leakage from the turbine gland seals are routed to a barometric condenser.

The HPCI system is automatically actuated on low reactor water level. If automatic actuation fails, the system can be manually initiated from the control room.

FUNCTIONAL FAULT TREE FOR HIGH PRESSURE MAKEUP



Functional Fault Tree for High Pressure Makeup

This view graph shows the functional fault tree for high pressure make up. The event, inadequate flow from HPCI to the reactor, consists of two inputs: HPCI intrinsic faults and the HPCI switch. An intrinsic fault is one that only prevents the system from performing its function. These faults includes failure of valves, pumps, controls and other components in the HPCI system. The probability of HPCI failure is estimated with a model. The model detail depends upon the amount of system data available. The more data available at the system level the less detail required in the model. If no data exist then all components required for HPCI operation must be included in the fault tree. If a large source of system level data exists, then the model may consist of the single input, HPCI intrinsic faults. The functional fault tree must be detailed enough, however, to preserve the independence of subsystems which are capable of satisfying the performance requirement. The HPCI system is a single train system and therefore meets this definition.

The second input to the event, inadequate flow from HPCI to reactor, is: the HPCI switch. This switch contains faults external to HPCI that influence its operability. These are the influence of the support state and the initiating event. If the particular support state and initiating event have no impact on HPCI operability then the output of the gate is zero and the HPCI failure probability is completely determined by the intrinsic faults. If the support system or the initiating event results in loss of HPCI operability then the output of the gate is one and the HPCI failure probability is 1.0. The influence of the support state on the front-line systems is determined using the dependency matrices. Development of these matrices is the subject of the next discussion.

EQUIPMENT UNAVAILABILITY DATA

TYPE OF DATA

AVAILABILITY OF DATA

Equipment Unavailability Data

Equipment unavailability data is required if the failure probability of the front-line functions are to be estimated. Equipment unavailability data is input into the support state model at the point where the unavailabilities are independent. This usually implies at the division or channel level; however, in some instance it could involve components. Estimates of these unavailabilities can be obtained from plant records or industry data or if necessary synthesized from fault tree models of the channel or division. The type and source of data utilized is reviewed in the next view graphs.

TYPE OF EQUIPMENT UNAVAILABILITY DATA

PREVENTIVE MAINTENANCE

CORRECTIVE MAINTENANCE

TESTING

MODIFICATIONS

FAILURE DURING OPERATION

FAILURE TO START ON DEMAND

Type of Equipment Unavailability Data

There are many causes of equipment unavailability. These include:

- preventive maintenance,
- corrective maintenance,
- testing,
- modifications,
- failure during operation, and
- failure to start on demand.

The first four of these items generally result in unavailability of the equipment involved. Technical Specifications, however, control the degree of degradation of plant capability by imposing constraints on the duration of the outage and requirements for demonstrated operability of redundant systems or equipment. The last two of these contributors to unavailability may occur in response to a surveillance test, in which case the outage is collected as a part of corrective maintenance. Definitions of the above contributors follow.

Preventive Maintenance - Preventive maintenance is performed to ensure that equipment maintains a given level of performance.

Corrective Maintenance - Corrective maintenance is performed when a component's performance is degraded or failed.

Testing - Testing is performed to verify equipment is within specifications. These tests are considered equipment outages if they rendered the equipment functionally inoperable.

Modifications - Modifications are physical changes in the system that render equipment inoperable during the modification process.

Failure during operation - Failure during operation occurs when equipment ceases to perform its function while operating.

Failure on Demand - Failure on demand occurs when standby equipment does not respond when challenged.

AVAILABILITY OF DATA

GENERIC DATA

PLANT SPECIFIC

Availability of Data

There are two sources of data. These are generic data and plant specific data. Generic data is generally derived in two ways. Failure data from many sources is pooled. This pool population is used to make estimates of component or system unavailability. A second method is also employed. This involves soliciting judgement of experts. These judgements are then pooled to make estimates of component or system unavailability.

Another source of data is plant records. While this source is very appealing it is labor intensive and for early life plants may be insufficient for parameter estimation.

GENERIC INDUSTRY UNAVAILABILITY DATA

REACTOR SAFETY STUDY

NUREG LER SUMMARY

IEEE 500-1977

IEEE 500-1984

NUCLAAR

NPRDS

Generic Industry Unavailability Data

The principal sources of generic data used in risk studies are listed in the view graph. The applicable component failure rates published in these references are in the context of the model assumed when generating the estimates. The interpretation of the estimates is not clear in every case, but the following assumptions seem reasonable for the following sources.

1. Reactor Safety Study. Estimates are given as a median value and an error factor on the assumption of a log normal distribution. These distributions were interpreted as representing the generic distribution of component failure rates.

2. NUREG LER Summaries. The estimates are evaluated on the basis of an assumed common value of the failure rate or probability for all components in a particular category. The uncertainties presented in the NUREG reports are statistical uncertainties on the quoted best estimates, obtained by accumulating the data from individual plants.

3. IEEE Std 500-1977. Following the guidelines of paragraph 3.6 of the IEEE document, the quoted maximum failure rate or probability is interpreted as the 95% uncertainty limit on the generic distribution, and the recommended value is interpreted as the mean value of the distribution. The results are given in such a way that it is impossible to be more precise, but this choice does give the broadest distribution.

4. IEEE Std 500-1984. The recommended values are assumed to be mean values. The high and low values are assumed to be the 5% and 95% uncertainty limits on the generic distribution.

5. NUCLAAR. Is an automated data base management system used to process, store and retrieve human and hardware reliability data in a ready-to-use format. A description of this data base is presented in NUREG/CR-4639.

6. NPRDS. A industry data based maintained by INPO. This system contains failure reports on components required to support plant operations. The boundary is drawn at the turbine stop valve. A large volume engineering data on these components is also available.

PLANT SPECIFIC DATA

EQUIPMENT OUTAGE

FAILURE TO RUN

FAILURE ON DEMAND

Plant Specific Data

Plant specific data can be collected and utilized in the risk evaluation. Three types of data can be collected. These are: equipment outage, failure to run, and failure on demand. Each of these are discussed below.

Equipment outage data is the most abundant source of data available. It can be obtained from LCO logs, work authorization and other maintenance records. It is just the sum of the hours the equipment is out of service. This data has been collected at PP&L for the scope of systems in the IPE.

Failure to run data requires two pieces of information: the number of failures and the total run time. If the constant failure rate model is assumed, the maximum likelihood estimate of the failure rate is just the number of failures divided by the total run time. The number of failures is obtained from plant maintenance records. The total number of operation hours may not be available on all components. However, for large electrical equipment it can be estimated from the total hour meter.

Finally the failure on demand probability also requires two pieces of data: the number of failures and the total number of demands. These data are obtained from maintenance records and plant testing schedules. The maximum likelihood estimate of the failure probability on demand is the number of failures divided by the number of demands.

At PP&L we are developing a system that automatically collects all this data for systems that are monitored by the plant history computer. This system should be operational by the end of 1988.

SYSTEM DEPENDENCY MATRICES

FRONT-LINE SYSTEM VS. SUPPORT SYSTEM

IDENTIFIES THE IMPACT OF THE SUPPORT SYSTEM ON THE FRONT-LINE SYSTEMS

SUPPORT SYSTEM

IDENTIFIES THE IMPACT OF SUPPORT SYSTEM ON OTHER SUPPORT SYSTEMS

System Dependency Matrices

The output of the switch gate in the functional fault tree is set by the support state. The support state maps the effect of the support systems onto the front-line system. This mapping allows explicit treatment of the support systems. Construction of the support states require that the dependency between the front-line system and the support systems and the dependencies between the support systems be determined. Only the primary dependencies must be identified. A primary dependency is one where a direct cause and effect relationship exists. As an example a motor operated valve may require AC power from a 480 volt motor control center. This motor control center in turn requires power from a 480 volt load center which is powered by a 4160 volt bus. The bus may receive power from offsite circuits or emergency diesel generators. In this example the motor operated valve has a primary dependency on the 480 volt motor control center. All the others are higher order. These higher dependencies are accounted for by the algorithm which constructs the support state. This example points out that dependencies exist between support systems. The 4160 volt bus supplies power to many 480 volt load centers. The 480V load center in turn provides power to many 480V motor control centers. Finally, these motor control centers may power many front-line functions. These primary dependencies between support systems must also be captured. Therefore, two tables must be constructed. First, the primary dependencies between the front-line systems and support systems must be identified. Then the primary dependencies between the support systems must be identified.

FRONT-LINE FUNCTION - SUPPORT DEPENDENCY MATRIX

	SUPPORT SYSTEMS									
	DC		480V AC				RHR CH			
FRONT-LINE SYSTEMS	125V	250V	1B216	1B226	1B236	1B246	A	B	C	D
HPCI	1	1	0	0	0	0	0	0	0	0
RHR DIV I	0	0	0	1	0	1	1	0	1	0
RHR DIV II	0	0	0	0	1	0	0	1	0	1

Front-line System Support System Dependency Matrix

An example of the front-line system vs. support system dependency table is illustrated in the view graph. In this table the support systems are identified as rows and the front-line systems are identified with the columns. If a primary dependency exist between the front-line system and the support system a 1 is entered in the Table, otherwise a zero is entered. In this example primary dependencies exist between the HPCI system and the 125 and 250 volt buses and the RHR divisions and channels and the 480V motor control centers. Therefore, 1's are entered at the intersections of the front-line systems and support systems in the Table. Notice that the RHR channels are listed with the support systems. At Susquehanna the RHR pumps support many functions, i.e., vessel injection, suppression pool cooling, containment spray etc. This is a sufficient condition for classification as a support system. They, therefore, are classified as such.

The components in the RHR system that perform a single function are classified as front-line systems. The injection valves serve as an example.

SUPPORT SYSTEM DEPENDENCY MATRIX

DEPENDENT SYSTEM

INDEPENDENT SYSTEM		125V				460V				480V LC				RHR				ESW	
		A	B	C	D	A	B	C	D	A	B	C	D	A	B	C	D	1	2
125V DC	A	1				1													
	B		1				1												
	C			1				1											
	D				1				1										
4160V	A					1				1				1				1	
	B						1				1				1				1
	C							1				1				1		1	
	D								1				1				1		1
480V LC	A									1									
	B										1								
	C											1							
	D												1						
RHR	A													1					
	B														1				
	C															1			
	D																1		
ESW	1													1			1	1	
	2														1	1			1

Support System Dependency Matrix

An example of a support system dependency Table is shown in the view graph. This Table is constructed by replacing the front-line systems in the columns by the support systems. If a one to one correspondence exists between the rows and columns a symmetric matrix will result. While not a necessary condition, a symmetric matrix is visually more appealing and simplifies the mathematical manipulation. Again primary dependencies are identified by 1's. If a primary dependency does not exist then a 0 is entered. Notice that additional support system are included in the table beyond what is identified in the front-line system vs. support system dependency Table. This is required because a primary dependency between these systems and those identified in the front-line system dependency table exists. As an example, although the ESW system does not appear in the front-line system vs. support system dependency Table, it provides cooling water to the RHR pump. Without this cooling the pumps are postulated to fail. This primary dependency must be captured. In this table the independent variables are identified by the column and the dependencies are identified by the rows. As an example there is a primary dependency between 4160 volt bus A, and 125 VDC bus A. Note that a primary dependency always exists between a system and itself. In this example 125 VDC depends upon itself.

These tables are the front-line system vs. support system and the support system primary dependency matrices. These matrices form the basis of the support states.

DEVELOPMENT OF SUPPORT STATES

1. DETERMINE ALL SUPPORT SYSTEM INTERDEPENDENCIES

2. DETERMINE IMPACT OF SUPPORT SYSTEMS ON FRONT-LINE FUNCTIONS

3. GROUP SUPPORT SYSTEM COMBINATIONS WHICH HAVE SIMILAR IMPACT ON FRONT-LINE FUNCTIONS.

Development of Support States

The support state maps the effect of support system failures onto the front-line systems. This requires that the interrelationship between the support systems be determined. Once these interrelationships are accounted for, the effect of support system failures on the plant front line systems can be determined. This process utilizes the following algorithm.

1. Identify the dependencies of the support systems upon one another.

2. Evaluate if all front-line systems are unavailable.

3. Pick support system i in the front-line vs. support system dependency matrix.

4. Check to see if support i is dependent on a support system already selected in the string. If not, continue.

5. Check to see if the support system fails additional equipment. If it does, continue.

6. Check frequency of the support system combination.

7. Group support system combinations which have similar impact on the front-line functions.

IDENTIFICATION OF SUPPORT SYSTEM INTERDEPENDENCIES

$$A^N = A^{N-1} \times A^{N-1}$$

IF

$$A^N = A^{N-1}$$

ALL INTERDEPENDENCIES IDENTIFIED

Identification of Support System Interdependencies

As discussed previously many support systems require other support systems to function. As an example AC power provides motive force to many pumps and valves in the plant. The AC power source in many cases requires DC power for control of the AC equipment. Thus a failure of DC power may indirectly fail a front-line system. These interdependencies must be accounted for to properly identify the influences of the support systems upon the front-line systems.

An algorithm has been developed to determine these interdependencies. This algorithm consist of four steps. The first step is input to the process and requires that the primary support system dependency matrix be developed. The second step involves taking higher products of this matrix to determine support system interdependencies. After taking each new product of the matrix it is compared to the previous step to determine if additional dependencies were identified. This process is continued until no additional dependencies are identified.

The interdependencies of support systems must be identified through input. This input is in the form of a square matrix. A one is entered in element i, j if system j depends upon system i. If no dependency exist between system i and j then a zero is entered. Only direct dependencies need be identified. As an example a 480 volt motor control center depends upon a 480 volt load center. This load center in turn depends upon a 4160 volt bus which in turn depends upon DC control power. Only the primary dependency between the motor control center and the load center need to be supplied as input to the support system vs. support system matrix. Higher level dependencies are determined by taking products of the matrix with itself.

It is shown in by induction (see reference 4 on fault tree analysis) that all the support system interdependencies can be determined by taking higher products of the input support system dependency matrix. Since matrix multiplication does not commute the operation must be performed as follows:

$$A^n = A^{n-1} \times A^{n-1}$$

Here

A is the support system vs. support system dependency,

A^n is the product of the support system vs. support system dependency, and,

n is the number of products taken.

These higher level products are taken until the elements in the matrix A^n equals the elements in the matrix A^{n-1}. When this has occurred all intra-support system dependencies are determined. This information is used to map the effects of support system failures into the front-line systems.

SUPPORT STATE DEVELOPMENT

EXAMINE ALL SUPPORT SYSTEM FAILURE COMBINATIONS

DEFINE RULES TO ELIMINATE IMPROPER COMBINATIONS

COMBINE SUPPORT SYSTEM FAILURE COMBINATIONS THAT HAVE SIMILAR IMPACT ON FRONT-LINE SYSTEMS

Support State Development

The identification of front-line system dependencies as a function of the systems requires that the primary dependencies of the front-line support systems on the support systems be input. This is just the front-line system vs. support system dependency matrix discussed earlier. Once this relationship has been established, the support systems are combined to determine their influence on the front-line systems. There is a problem, however, with just examining all possible support system combinations. The support systems depend upon one another and therefore combinations which include dependent support system will result in double counting. In some cases an additional support system failure does not alter the influence the failure combination has on the front-line system. This situation would again result in double counting. Many combinations of support system failures may have an equivalent impact on the plant. This situations lends itself to grouping which substantially reduces the computational burden. The outage of many support systems is controlled by plant technical specifications and their failure will result in a forced shutdown. Thus the order of failure in the combination of support systems becomes important when computing the combination frequency.

An algorithm to generate the support system combinations which accounts for these complications is discussed below. Strings of support system combinations are considered. At each point where an additional support system failure is considered the combination is tested against criteria to properly account for the above problems.

Various forms of support are required for operation of the front-line systems. As an example a pump may require electric power and cooling for operation. These requirements form the basis of the support states. The approach taken is analogous to that used for support systems. The primary dependencies are input in matrix form. This matrix is called the front-line system vs support system dependency table. The columns and rows represent support systems and front-line systems respectively. A 1 in element i,j, identifies that front-line system j requires support system i for its operation. Only the primary support system dependencies need to be input. Higher order dependencies are identified through the support system matrix.

All possible combinations of support system are determined by taking sequential strings of support systems. This method is best demonstrated by example.

Consider the case of four support systems.

The first support system is considered by itself. Then the first and second, then the first, second and third and so forth. Numerically this becomes:

1	1 2 4	2	3
1 2	1 3	2 3	3 4
1 2 3	1 3 4	2 3 4	4
1 2 3 4	1 4	2 4	

RULES FOR SUPPORT SYSTEM COMBINATIONS

SUPPORT SYSTEMS IN STRING ARE DEPENDENT

SUPPORT SYSTEMS HAVE UNIQUE EFFECT ON FRONT-LINE FUNCTIONS

THE SUPPORT SYSTEM STRING HAS A SIMILAR IMPACT ON FRONT-LINE SYSTEMS

ALL COMBINATIONS CONSIDERED

THE SUPPORT SYSTEM FAILURE STRING HAS AN ACCEPTABLE FAILURE PROBABILITY

Rules For Support System Combination

If the support systems are combined as described in the previous view graph the number of combinations would grow to an unmanageable number. Some of the combinations have an equivalent effect on the plant while others do not represent minimal failure combinations. A set of rules are imposed on each new string of support system failures considered.

Each new combination of support systems is tested to ensure the combination is appropriate. Each string is tested to determine:

whether the combination consists of dependent support systems,

the impact of the additional support system failure on the front-line systems,

the equivalence of the present support system combination to a previous combination,

that the consideration of all permutations is considered in the frequency calculation, and

whether the frequency of the combination is below an input cutoff value,

Each of these items is discussed below.

The test for a dependency between support systems in a combination is determined by checking the support system vs. support system dependency matrix for a one in the row that corresponds to the support system under consideration. If support system j is being considered then the previous the elements 1 through (j-1) which appear in the combination are querried in the support system vs. support system matrix. Should a one appear, a dependency exists, and the jth support system will not be included in the string. The checking will be terminated and the calculation will proceed to the j+1 support system. If only zero's exist in these matrix elements the jth support system will be tested for its impact on the front-line systems.

If the jth support system is independent of the j-1 support systems in the combination being considered, it must be tested to see if it results in additional front-line system failures.

This test is performed by complementary addition of the rows in the front-line system support matrix which correspond to the support systems in the j-1 string and the support system being tested. If no additional dependences between the jth support system and the front-line system exist then the result of the complementary additional will remain the same. This implies that the jth support system failure has had no additional effect on the front-line systems and therefore the jth support system will not be included in the string. If additional dependencies exist between the front-line systems and jth support system then the result of the complementing additions will change. This change in status represent additional dependencies.

Once a combination of support systems is identified, the frequency of its occurrence is computed. The following information is required to perform the computation: the support systems in the combination, the failure rate of each of the support systems and the allowed outage time for each support system. The support system combination is provided by the computer program, while the failure rate and the allowed outage time for each of the support system is user input. With this information the frequency of the event can be computed.

We have developed an equation to compute the support state probability. This equation is based upon the following assumptions.

1. All operating system failures follow the constant failure rate model. The failure rate is represented by λ.

2. Each support system i has an allowed outage time AOT which is represented by A_i.

3. Should two or more failures occur the allowed outage time for the combination of support systems is set by a constant time, C. This time is analogous to the 3.0.3 action statement in the plant Technical Specification.

4. After an initiating event I, the equipment must operate for a period sufficient to place the unit in cold shutdown. This time is represented by M.

5. The plant operates for a time T.

6. i outages of length T_i can be treated as one outage who's length is the sum of the T_i's. This assumption is discussed in Appendix 9.

Given a support system failure or an initiating event at time t, the probability of a subsequent event P_s is computed using the constant failure model:

$$P_s = \int_o^T dt_i \, pdf(t).$$

Failures can occur before or after an initiating event. If the failure occurs before the initiating event, the exposure time is set by the Allowed Outage Time of the first failure. If the first failure occurs after the initiating event, then the support systems must function for the mission time. Thus the order of failure must be preserved when performing frequency calculations.

The form of this integral for each failure is the same as the above case with the initiating event I being the point at which the limits of integrate change. Using this fact induction can be used write an expression for n+m failure occurring prior to and after an initiating event as shown below:

The form of the integral becomes:

$$P_s(t) = \int_{o}^{(T-A)} dt_1\lambda_1 \int_{t_1}^{(t_1+A-C)} dt_2\lambda_2 \int_{t_2}^{(t_2+C)} dt_3\lambda_3 \int_{t_3}^{(t_3+C)} dt_4\lambda_4 \ldots + \int_{o}^{(T-A)} dt_1\lambda_1 \int_{t_1+A-C}^{(t_1+A)} dt_2\lambda_2 \int_{t_2}^{(t_1-A)} dt_3\lambda_3 \int_{t_3}^{(t_1+A)} dt_4\lambda_4$$

$$+ \int_{T-A}^{T} dt_1\lambda_1 \int_{t_1}^{T-C} dt_2\lambda_2 \int_{t_2}^{t_2+C} dt_3\lambda_3 \int_{t_3}^{t_3+C} dt_4\lambda_4 \ldots + \int_{T-A}^{T} d_1\lambda_1 \int_{T-C}^{T} dt_2\lambda_2 \int_{t_2}^{T} dt_3\lambda_3 \int_{t_3}^{T} dt_4\lambda_4$$

The pdf for the constant failure rate model is an exponential which can be expanded in an infinite series.

$$pdf(t) = e^{-\lambda t} = (1-\lambda t + (\lambda t)^2/2! \ldots)$$

The AOT is designed to minimize the amount of time the plant is operated with failures, therefore if the λ is small, λt terms are much smaller than 1. This situation allows the pdf to be approximated by just λ. This greatly simplifies the integral. The process can be generalized to deal with demand failures by realizing that operating systems cannot experience demand failure while operating and likewise standby systems cannot experience operating failures while in standby. Both, however, can experience maintenance prior to an initiating event. Using this fact induction can be used to write an expression for n+m failures occurring prior to and after an initiating event as shown below.

No failures prior to an initiating event.

$$P_s = \sum_{j=1}^{r} \left(\frac{\lambda_j}{\lambda_j+P_j}\right) \prod_{i=1}^{r} (\lambda_i+P_1) \quad M^{(r-d)}$$

One failure prior to an initiating event.

$$P_s = \sum_{j=1}^{r} \left(\frac{\mu_j+\lambda_j}{\lambda_j+P_j}\right) \left[\prod_{i=1}^{r} (\lambda_i+P_i)\right] M^{(r-d+d_j-1)} \quad \frac{A[T-(A/2)]}{T}$$

More than one failure prior to an initiating event

$$P_s = \sum_{j=1}^{r} \sum_{k=1}^{r-d+d_j+1} \frac{\mu_j+\lambda_j}{(\lambda_j+P_j)} \left[\prod_{i=1}^{r} (\lambda_i+P_i)\right] \frac{(r-d+d_j+1)!}{(k-1)!(r-k-d+d_j)!} \quad M^k C^{(r-d+d_j-k)}$$

$$\frac{(TA - TC - \frac{A^2}{2} - \frac{1}{K} CT)}{T}$$

Here,

r is the total number of failures in a string

i, j, k are indices,

λ_i is the operating system failure rate of the i^{th} component,

M is the mission time from the start of the initiating event,

C is the allowed outage time associated with two or more failures,

A_j is the allowed outage time for the j failure, and

T is the total time failures are to be considered.

μ = (Total outage time)/AOT/T: and is the maintenance rate.

p is the demand failure probability

$$d = \sum_{i=1}^{r} \frac{P_i}{\lambda_i + P_i}$$ and is equal to the number of demands

$$d_j = \frac{P_j}{P_j + \lambda_j}$$

If a support system failure combination satisfies the above selection rules, and it has a sufficient frequency of occurrence, it is then evaluated to determine if it is represented by an existing support state or if it represents a new support system configuration. This evaluation is performed using the support system failure combination, the front-line system vs. support system dependency table, the functional fault trees and the basic event data. Prior to this evaluation, however, matrices used to evaluate the logical consistency of the support system failure combination are updated.

The functional fault trees are evaluated using the support system failure combinations to determine the influence the support system failure combination has on the event tree top events. The results of this evaluation are compared to the event tree top events for other support states. If the influence of the combination on the event tree top events is the same as another existing support state, the combination being evaluated is included as an element of that support state. If the influence of the combination on the event tree top events is unique, then a new support state is created. In either case the calculation proceeds to the next support system failure combination. When all the support system failure combinations are evaluated the influence of the support systems on the front-line systems is mapped.

CALCULATION OF ACCIDENT SEQUENCE FREQUENCY

$$F_{ACI} = IXP_{SSI} \times P_{TE1,I} \times P_{TE2,I} \cdots P_{TEN,I}$$

Calculation of the Accident Sequence Frequencies

The evaluation of accident sequences requires evaluation of logical relationships of the plant components and estimation of the failure probability. The interdependencies are explicitly treated when developing the event tree top event probabilities in a manner which preserves their independence. Accident sequence frequency evaluation therefore reduces to the product of the initiating event frequency and the event tree top event probabilities in the accident sequence. This process is described by the following equation,

$$F_{ACi} = I \times P_{SSi} \times P_{TE1,i} \times P_{TE2,i} \cdots PH_{TEN,i}$$

Here,

F_{ACi} is the accident sequence frequency for support state i,

I is the initiating event frequency,

P_{SSi} is the probability of support state i, and,

$P_{TE1,i}$ through $P_{TEN,i}$ are the event tree top event probabilities for support state i.

STATION BLACKOUT EXAMPLE

INPUT DATA

DEVELOPMENT OF SUPPORT STATES

ACCIDENT SEQUENCE FREQUENCY CALCULATIONS

Station Blackout Example

This process is illustrated using the station blackout sequence with a successful scram, proper operation of the SRVs, but a failure of HPCI and RCIC. In this evaluation input data is required. This data will be used in developing the support states.

Once the support states are defined the accident sequence frequency for each support state is computed.

Input data for Accident Sequence Calculation

	Item	Value	Reference
1.	SBO frequency	2.2×10^{-4}/yr	NUREG-1032 and SSES-IPE
2.	DC bus failure	3.4×10^{-6}/hr	SSES-IPE
3.	HPCI failure probability	4.2×10^{-2}/dem	SSES-IPE
4.	RCIC failure probability	4.2×10^{-2}/dem	SSES-IPE
5.	Offsite power nonrecovery probability at 50 minutes	0.23	NUREG-1032
6.	Onsite power non-recovery probability at 50 minutes	0.91	SSES-IPE
7.	AC power non-recovery probability at 50 minutes	0.21	Item 6 x Item 7
8.	DC bus AOT	2 hours + 6 hours	SSES Tech Spec
9.	Fire pump failure probability	2.9×10^{-2}/dem	SSES-IPE
10.	Bottom head Dryout	3.5 hours	BWRSAR Calculation
11.	Non-recovery probability at 3.5 hours	0.09	Application of NUREG1032

Input Data for Accident Sequence Calculation

This view graph identifies the initiating event frequency, the system failure data, timing information associated with the event, and the probabilities associated with loss of each barrier. These probabilities are associated with the recovery of AC power in many instances and therefore are called non-recovery probabilities.

Development of Support States

Support State Failure Combination	Frequency (Years^{-1})	Unavailability of High Pressure Injection (demand^{-1})	Extended Cooling	Vessel Non-recovery Probability
1D614	1.1×10^{-4}	4.2×10^{-2}	0.21	0.0027
1D624	1.1×10^{-4}	4.2×10^{-2}	0.21	0.0027
1D614 and 1D624	5.6×10^{-9}	1.0	0.21	0.09
No Failure	0.99978	0.0018	0.21	0.0027

Example Development of Support States

This process is demonstrated using the accident sequence SBO.HP.EC. This accident sequence is defined in the discussion on event tree construction. It represents a station blackout with a failure of HPCI and RCIC and a failure to recover AC Power. Only the DC power support systems will be considered in this example. Additional data required for the analysis is given in the previous view graph. In this example the support system failure combinations to consider are no failures, loss of 1D614, loss of 1D624, and loss of both DC buses. The effect of these support system is shown in the development of the support system view graph. Examining this table reveals that loss of 1D614 and 1D624 have the same effect on the front-line system. Therefore, they are combined. These support states are therefore considered; no failures, loss of 1 DC bus, and loss of 2 DC buses.

Accident Sequence Frequency

Support State	Initiator Year	Support State Year	Core Damage HP	BC	P_{CD}		Vessel VF	Failure P_{vf}
No failure	2.2×10^{-4}	0.9978	0.0018	x0.21	$=8.3\times10^{-8}$	x	0.0027	$= 2.2\times10^{-10}$
1 DC bus	2.2×10^{-4}	2.2×10^{-4}	0.042	x0.21	$=4.3\times10^{-10}$	x	0.0027	$= 1.2\times10^{-12}$
2 DC bus	2.2×10^{-4}	5.6×10^{-9}	1.0	x0.21	$=2.58\times10^{-13}$	x	0.09	$= 2.3\times10^{-14}$
TOTALS					8.3×10^{-8}			2.2×10^{-10}

Accident Sequence Frequency Calculation

This view graph demonstrates how the accident sequence frequencies are computed. The initiating event frequency is partitioned by the support state. In this example the three support states are considered. The frequency of the front-line functions for each support state are used in the calculation. These are obtained from the previous view graph.

Since each probability is independent the accident sequence frequence for a given support state is obtained by multiplication. The plant damage frequency for a given accident sequence is obtained by summing the support states.

ORGANIZATION OF RESULTS

ORGANIZATION OF RESULTS(1)

INPUT DATA

- o INITIATING EVENTS
  - TYPE
  - FREQUENCY
  - IMPACT VECTOR
- o SUPPORT SYSTEMS
  - UNAVAILABILITY
  - DEPENDENCY MATRIX
- o FRONT LINE SYSTEMS
  - UNAVAILABILITY
  - DEPENDENCY MATRIX
- o SUCCESS CRITERIA
  - UNAVAILABILITY OF NEEDED EQUIPMENT
  - TIME AVAILABLE FOR OPERATOR ACTION

Organization of Results(1)

PP&L takes the position that in order for any risk assessment to have credibility, the input data for the calculation must be presented in the final report document. On the basis of practicality we exclude from this presentation the various transient analysis and supporting calculations necessary to the risk analysis. These calculations should be available in the form of auditable records supporting the risk analysis, however. Further, the results of these analyses which are used in the risk analysis should be identified and their origin unambiguously referenced. This requirement for an unambiguous reference applies to all data used in the risk analysis.

The input data for initiating events must include a list of the initiators considered and the frequency of each. In addition the impact vector must be provided. This data specifies the equipment made unavailable by the initiator. The equipment list must include all items from the list of Support Systems and the list of Front-line Systems.

In the case of Support Systems the inherent unavailability and the primary dependencies must be provided. It is probably desirable to present the final dependency matrices as calculated with all dependencies indicated as well as the primary dependencies. This type of information could be useful in evaluating the results of the risk analysis. If the system is a standby system, the unavailability will be on a demand basis. If it is an operating, system, however, the Allowable Outage Time (AOT) must also be provided in order to determine the expected unavailability.

The information required for Front-line Systems is similar in nature.

The Success Criteria, in general, have the form of a set of equipment required and a time constraint for its successful operation. This data is generally derived from the various transient analyses and should be carefully referenced to them.

ORGANIZATION OF RESULTS(2)

OUTPUT DATA

- UNAVAILABILITY MATRICES
  - INITIATOR-SUPPORT STATE X FRONT-LINE FUNCTION

- SUPPORT STATE FREQUENCY FOR EACH INITIATOR

- ACCIDENT SEQUENCE FREQUENCY MATRIX
  - INITIATOR-SUPPORT STATE X PLANT DAMAGE STATE

- PLANT DAMAGE STATE BY INITIATOR MATRIX
  - INITIATOR X PLANT DAMAGE STATE

- PLANT DAMAGE STATE VECTOR
  - TOTAL CONTRIBUTION TO EACH PLANT DAMAGE STATE

- TOTAL PLANT DAMAGE

Organization of Results (2)

The presentation of the output of the calculation should include the unavailability matrices, the accident sequence frequency matrix, the plant damage state by initiator, the distribution of total plant damage state frequencies, and the total plant damage state frequency.

The accident sequence frequency matrix actually contains all of the information developed by the risk analysis. It is this data which must be edited in order to cast the results of the analysis into various formats. The identifiers of the accident sequences are:

- the initiator,
- the support state,
- the Front-line Function failure sequence,
- and the plant damage state distribution

The last of these is a series of plant damage state descriptors which describe:

- the time of core damage
- the extent of reactor damage
- the amount of core debris released to the drywell floor,
- the time of containment loss of integrity, and
- the nature of the loss of containment integrity.

This data provides great flexibility in the manner in which summary data can be displayed. It can for example be used to determine the nature and frequency of the contribution to plant damage from specific system failures. It can also be used to perform sensitivity studies. The primary requirement for such activities, if the risk analysis is computer based, is a flexible edit program to collect and organize the appropriate data.

AN IPE APPLICATION

THE INTEGRATED RISK REDUCTION STUDY

PURPOSE OF THE STUDY

THE APPROACH TAKEN

STUDY DELIVERABLES

The Integrated Risk Reduction Study

This view graph outlines the presentation of the integrated risk reduction study. This study is one application of the SSES-IPE. It is reviewed here to demonstrate the utility of the support state method, and also demonstrates PP&L approach to bring closure to the severe accident issues. The purpose of the study, the method employed and the deliverables expected are reviewed.

PURPOSE OF THE INTEGRATED RISK REDUCTION STUDY

TO BRING CLOSURE TO THE SEVERE ACCIDENT ISSUES FOR SUSQUEHANNA

IDENTIFY WEAKNESSES

SPECIFY IMPROVEMENTS

Integrated Risk Reduction

The purpose of the integrated risk reduction study (IRRS) is to systematically identify the initiating events and subsequent failures which contribute the greatest to Susquehanna Plant risk, and then to specify the most promising actions to reduce this risk. Through various other studies and evaluations PP&L is aware of potential weaknesses in Susquehanna's design and procedures. Many of these weaknesses, however, and thus their solutions, are interconnected. Therefore resolution of any severe accident problem requires an integrated approach. The goal of the IRRS study then is to identify severe accident problems at SSES and to integrate proposed solutions into a consistent and coordinated long term resolution of severe accident concerns at SSES, and thus reduce the risk of core damage, and containment failure.

IRRS METHODOLOGY

DEVELOPMENT OF EVALUATION CRITERIA

EVALUATION OF ACCIDENT SEQUENCES AGAINST CRITERIA

EVALUATE PROPOSED SOLUTIONS TO IDENTIFIED WEAKNESSES

IRRS Methodology

As discussed previously, the scope of this study could quickly grow out of control due to real uncertainties that exist in accident sequence evaluation, a potential for a broad spectrum of vulnerabilities, and a multiplicity of solutions to identified problems. However, a methodology has been devised which limits the scope of this effort. This methodology consists of three steps. First, a qualitative evaluation is carried out to identify weaknesses in the plant design or procedures. The second step involves identifying root problems which result in the plant weaknesses identified in the step one. Finally, solutions to these root problems are evaluated to determine the optimal approach to dispositioning the problems. A description of the approach to each step is provided in the following paragraphs.

The first step is structured around qualitative evaluation criteria. This approach has been utilized at PP&L for many years as demonstrated by the Station Blackout and ATWS safety evaluations and more recently by the SSES IPE. The criteria employed in those efforts were formalized as defense-in-depth criteria at the NRC Region I PRA workshop in March 1987. They were again revised for use in this study and appear earlier in this presentation.

Many assumptions concerning plant design and operations, and analytical models used in quantitative analysis, are necessary in a study of this nature. These assumptions must be collected and submitted to Engineering and Operations to ensure they are realistic.

With the evaluation criteria established, the next question to consider is what set of accident sequences should be judged against the criteria. The SSES IPE represents the largest collection of accidents evaluated for SSES and includes those considered in other sources such as the FSAR. Because the evaluation criteria require all accident sequences, including those having low calculated frequencies, to be subject to the defense-in-depth criteria, all SSES IPE sequences are being evaluated.

Each accident sequence is evaluated against the defense-in-depth criteria. The evaluation is performed by constructing a table that for each accident sequence identifies the plant damage sequence, the sequence frequency, and the results of the defense-in-depth evaluation. Each criterion is listed: equipment, procedures, and interface requirements. A discussion of how the accident sequence is evaluated against each criterion is discussed below.

After all accident sequences are reviewed a list of weakness is compiled. Modifications will then be proposed which eliminate the weaknesses. Risk calculations will be performed to determine which set of proposals are most effective of reducing risk.

EQUIPMENT EVALUATION

SEQUENCE	FREQUENCY	CRITERIA 1	2	3	4
SBO.HP.EC	8.3×10^{-8}	HPCI RCIC	E DIESEL	FIRE MAIN INJECTION	E DIESEL

Equipment Evaluation

The equipment criteria state that additional independent failures must occur before the loss of the fission product barriers, that is the fuel clad, and the primary containment. The reactor pressure vessel is considered also to be a barrier to core debris. This barrier is not a fission product barrier, but it does serve the important role of preventing an attack on critical containment components by core debris if integrity of the lower vessel head is lost. Saving this barrier greatly reduces the magnitude of the challenge to containment integrity.

The equipment failures that must occur prior to loss of a barrier are listed in the evaluation table identified. As an example, core damage (loss of the cladding barrier) will occur during station blackout if the HPCI and RCIC pumps also fail. Thus the failures of the diesels or components in Emergency Service Water (ESW) and of HPCI and RCIC are listed in the evaluation table under core damage. BWRSAR calculations indicate that fire main injection within one hour will terminate core damage with the reactor vessel intact. Therefore the fire main is listed under the criteria for vessel failure. Finally, should the core melt breach the vessel, tying in the fifth diesel for decay heat removal and remote manual operation of the two inch vent line for noncondensibles gas control will prevent containment overpressure failure. Thus, these systems are listed under the containment barrier criteria. This process is performed for each accident sequence in the IPE. Instances where defense-in-depth is not satisfied are documented.

PROCEDURAL AND INTERFACE EVALUATION

SEQUENCE	FREQUENCY	PROCEDURAL 1	PROCEDURAL 2	PROCEDURAL 3	INTERFACE 4
SBO.HP.EC	8.3×10^{-8}	NO VIOLATION IDENTIFIED	HPI SUCTION TRANSFER	FIRE MAIN ALIGNMENT DIESEL ALIGNMENT	INSTRUMENTATIC ON DC POWER

Procedural and Interface Evaluation

The method of evaluating the procedural criteria is radically different from that used for the equipment criteria. In this evaluation each procedural step performed by the operator in response to the accident sequence is evaluated against the criteria. The EOP flow chart steps taken by the operator during the particular sequence are highlighted. Steps that satisfy the criteria are labeled with that particular procedure criteria number, while steps that may violate the procedural criteria are marked with an asterisk (*) and the number of the criterion violated, and steps that are not applicable are left unmarked. In the defense-in-depth table, only those steps that may violate the criteria are identified.

The procedural criteria are targeted at ensuring that no deleterious actions are performed, that necessary actions are performed to protect operable equipment, and that the necessary actions are taken to accommodate additional failures. In the accident sequence used to illustrate the equipment defense-in-depth, the first procedural criterion is satisfied. The procedures instruct the operator to prevent HPCI and RCIC suction from the suppression pool if its temperature exceeds 150°F. This action protects the pumps which may be challenged if they should pump water with a temperature that exceed 170°F. That step satisfies procedural criterion No. 2. Finally, the procedures instruct the operator to align the fire main system for RPV injection. This step anticipates the loss of HPCI and RCIC and therefore satisfies procedural criterion No. 3. This is an expected result for station blackout, however, since PP&L developed these procedures using this philosophy.

The third criterion requires that the information and time necessary to make the decisions identified in the procedures exist. Equipment and procedural requirements are thus integrated. The method used to evaluate this criterion is similar to that used for procedural defense. Required instrumentation and sequence timing information must be available to evaluate acceptance.

This type of defense-in-depth evaluation is performed on each sequence in the SSES-IPE. The result of the evaluation will be a list of instances where the equipment or procedures fail to satisfy defense-in-depth. In phase two of the study these deficiencies will be grouped by root problem or by system interaction. Once this information is cataloged integrated solutions can be investigated.

IRRS DELIVERABLES

A TECHNICAL REPORT TO PP&L MANAGEMENT

IDENTIFIES WEAKNESSES IN DEFENSE-IN-DEPTH

EXPLAINS WHY WEAKNESSES EXIST

OUTLINES PROPOSALS TO CORRECT WEAKNESSES

RECOMMENDS PREFERRED MODIFICATIONS

IRRS Deliverables

The output of this study will be a technical report. The purpose of the report will be three fold. First it will identify those instances when SSES has not met our definition of defense-in-depth. Second, it will describe on a conceptual level why defense-in-depth was not met. Finally, a preferred set of modifications and procedures will be proposed to establish defense-in-depth or a justification for any deficiency will be presented. Alternate proposals will also be presented which may not produce the same benefit as the preferred set but may be more practical to implement. Proposals will be sufficiently detailed to allow preliminary cost estimates to be made.

In conclusion, we believe that the approach being taken in IRRS represents a major step toward resolving severe accident concerns at SSES.

HUMAN ERROR TREATMENT
IN
RISK ASSESSMENT

HUMAN ERROR TYPES

- o ERRORS WHICH LEAVE STANDBY EQUIPMENT UNAVAILABLE
  - INCLUDED IN UNAVAILABILITIES
- o ERRORS WHICH CAUSE A TRANSIENT (INITIATING EVENT)
  - INCLUDED IN INITIATING EVENTS
- o ERRORS IN RESPONSE TO AN INITIATING EVENT
  - PP&L HAS ASSUMED OPERATORS FOLLOW PROCEDURES WITHOUT ERROR.
  - FAILURE TO EXECUTE THE PROCEDURE OCCURS ONLY WHEN THE TIME AVAILABLE IS INSUFFICIENT.

Human Error Types

In performing the Susquehanna IPE human error which contributes to an initiating event or to equipment unavailabilities were assumed to be imbedded in the initiating events and equipment unavailabilities used to characterize Susquehanna. Initially generic data was used for this purpose, but, as operating data is gathered for the plant, actual plant data will be used for this purpose. In both cases the actual human error contribution is included in the data used.

This leaves us with the problem of characterizing and quantifying operator error in response to an initiating event which challenges the safety systems of the plant and which could lead to severe accident conditions in the event of sufficiently severe equipment failure or operator error. In performing the Susquehanna IPE we determined that the answers to two questions were necessary to characterize and quantify the operator error in response to an initiating event. These were:

1. Does the operator follow the procedural step as the symptom (cue) which calls for it occurs?

2. How long does execution of the procedural step require?

The first of these questions we resolved simply by stating that we would assume that the operator would follow procedures and that failure to follow procedures would have a negligible contribution to the risk of plant damage. The second we resolved by application of the Human Cognitive Response (HCR) model developed by G. W. (Bill) Hannaman of NUS, San Diego.

The assumption made with respect to the first of the above questions has been characterized by many as unrealistic and extremely optimistic. We did not make this assumption casually, however. We believed that our new symptom based EOPs were comprehensive and would be optimal for any initiating event and additional equipment failures for which operator response could have a favorable impact. This judgement was based on experience gained in detailed studies performed for Susquehanna on Station Blackout and ATWS accident sequences.

We further believed that our operator training in the use of the EOPs was very thorough and emphasized rigorous adherence to procedures. For these reasons we believed that our assumption represented a closer approximation to reality than some arbitrarily assumed probability of a deviation of an undefined type.

It must be recognized that if some rate of procedural deviation were to be considered, several very difficult questions must be answered. The first layer of such questions is:

1. Why was the procedure violated?

2. What is the nature of the violation?

It is clear that the answer to the second question is strongly dependent on the answer to the first. We therefore examined potential answers to the first of these questions.

It is reasonable to postulate that the reasons for deviation on a procedural step could be due to one of the following reasons:

1. The violation was willful and represents the operator's preference in responding to the plant conditions.

2. The operator misunderstood the procedural instruction and executed it improperly as a result.

3. The operator was unaware of the occurrence of the cue for the action.

4. The operator did not have adequate time to successfully perform the action.

The first of these possible answers we reject on the basis that our operators are indoctrinated in the necessity to adhere rigorously to procedures. The second, we believed would be unlikely since considerable care had gone into preparing and formatting the EOPs. The third we believed to also be unlikely as a consequence of our symptom based procedures and the generally slow nature of BWR transient response which allows considerable time to monitor the results of an action and take corrective action if necessary. For the very severe events for which time limited actions are involved, Station Blackout and ATWS, we believed the symptoms of the event to be unmistakable and that operator misinterpretation was simply not credible.

This leaves only the fourth item above as a credible answer, and therefore, the nature of the procedural deviation is simply failure to take the action in time to be effective. The critically important aspect of this type of failure, however, is that the operator is aware of the situation and therefore we must believe that he will attempt to execute the backup actions which would generally be available.

Quantification and characterization for the fourth of these potential answers is therefore straightforward. The HCR method may be used to quantify the failure probability, and the backup actions will characterize those cases where failure occurred. Quantification and characterization of the first three of the potential answers have no currently credible basis for either quantification or characterization.

ERRORS IN RESPONSE
TO AN INITIATING EVENT

- o MOST BWR TRANSIENTS ARE SLOW AND ABUNDANT TIME IS AVAILABLE FOR OPERATOR ACTION AND CORRECTION OF ERRORS.

- o FOR A LIMITED CLASS OF EVENTS THE TIME AVAILABLE FOR OPERATOR ACTION IS LIMITED AND PROMPT ACTION IS CRITICAL.

  - CLARITY OF PROCEDURES IS ESSENTIAL
  - OPERATOR TRAINING IS ESSENTIAL

Error in Response to an Initiating Event

We reiterate here that in most BWR transient events the reactor and plant response is slow and provides the operator with ample time for diagnosis and corrective action.

In those cases when the time for this is limited, it is essential that the EOPs be direct, unambiguous, and complete. Further, it is important that these situations be covered in operator training to assure a awareness of these time critical situations.

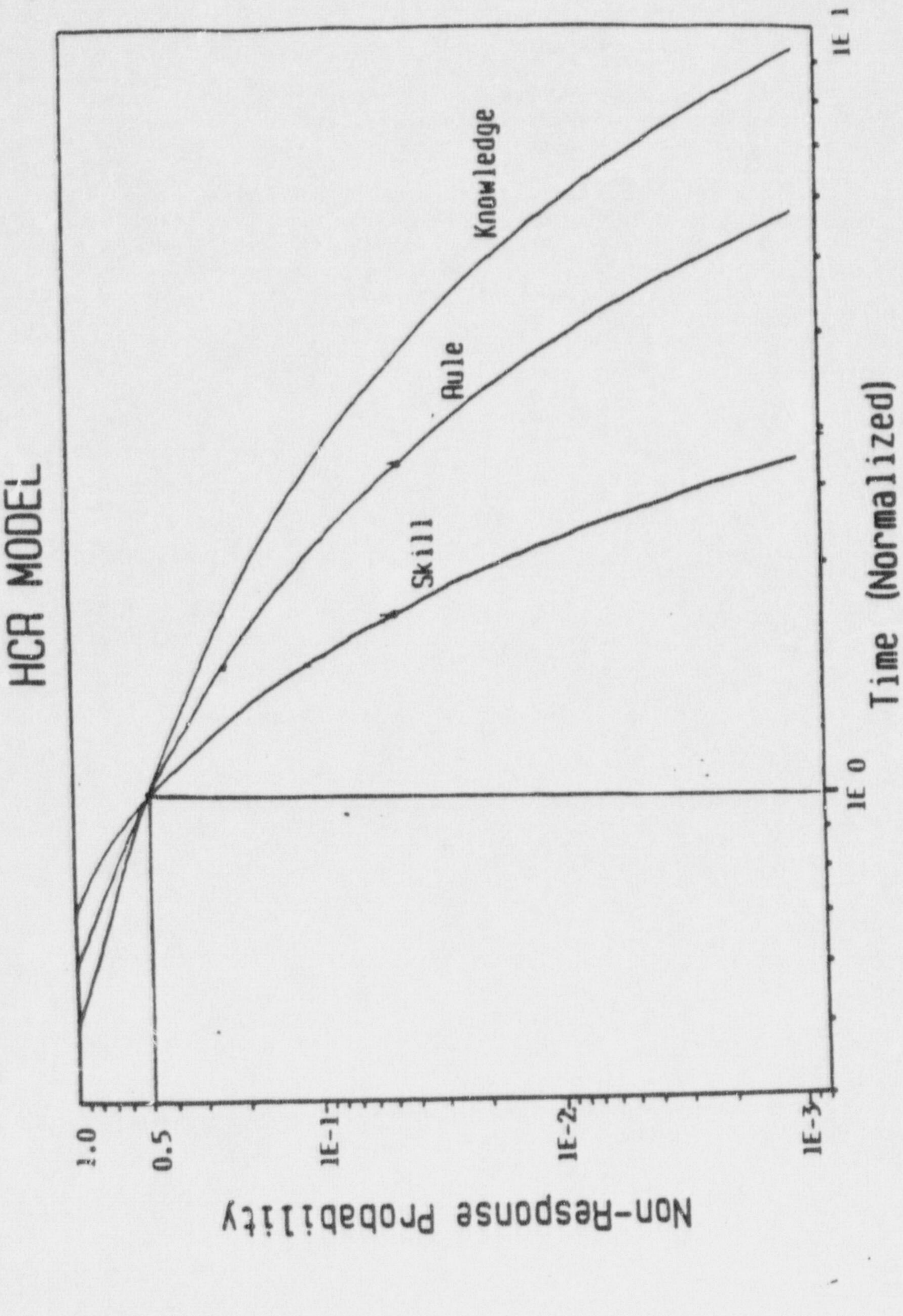
TIME CONSTRAINED ACTIONS

- SLCS INITIATION IN RESPONSE TO ISOLATION ATWS.
- FEEDWATER RUNBACK IN RESPONSE TO NON-ISOLATION ATWS.
- DEPRESSURIZATION IN RESPONSE TO ISOLATION ATWS WITH SLCS FAILURE.
- PREVENTION OF HPCI SUCTION TRANSFER IN ATWS.
- PREVENTION OF HPCI TRIP IN ATWS.
- MANUAL VALVE OPERATION FOR LPECCS INJECTION VALVE FAILURE.
- COMPLETION OF CONNECTION AND ALIGNMENT TO PERMIT FIRE MAIN INJECTION IN STATION BLACKOUT.

Time Constrained Actions

This viewgraph lists the important time constrained actions derived for the Susquehanna IPE. It should be noted, that with one exception, these actions are required in accident sequences which the operator must immediately recognize as a consequence of the symptoms which accompany the event. We do not believe that an error is credible in prompt recognition of these events.

The one exception, failure of the LPECCS injection valves to open, requires that the operators monitor plant response to assure that water level response is as expected with low pressure pump injection. Since operator training emphasizes the importance of monitoring plant response to actions taken, we believe that this symptom is very unlikely to be overlooked and that the failure rate, which is high for this action, will be dominated by the time available limitation.


HCR MODEL
Knowledge
Rule
Skill
Non-Response Probability
1.0
0.5
1E-1
1E-2
1E-3
Time (Normalized)
1E 0
1E 1

HCR Model

This figure represents the time dependence for operator non-response probability in executing an action required. This probability is seen to depend on a normalized time variable and on the nature of the cognitive processing involved; skill, rule, or knowledge. The curves used are Weibull distributions which have a fundamental negative exponential nature.

REQUIRED STEPS

FOR THE

HCR METHOD

- DETERMINE THE TIME AVAILABLE TO TAKE THE ACTION.
  - TRANSIENT ANALYSIS
- DETERMINE THE MEDIAN TIME REQUIRED FOR THE ACTION.
  - USE JUDGEMENT OR MEASURE
- DETERMINE THE PERFORMANCE SHAPING FACTORS.
  - EXPERIENCE, STRESS, INTERFACE QUALITY
- CALCULATE THE ADJUSTED MEDIAN TIME FOR THE ACTION.
- CALCULATE THE NORMALIZED AVAILABLE TIME FOR THE ACTION.
- DETERMINE THE TYPE OF COGNITIVE PROCESSING.
- USE THE CURVE TO DETERMINE THE FAILURE PROBABILITY.

Required Steps
for the
HCR Method

This viewgraph presents the several steps required to apply the HCR method. The first of these steps is not actually a part of the HCR model but only represents an input parameter for the model.

The seven steps are described further in subsequent viewgraphs.

TRANSIENT ANALYSIS

- o DETERMINE THE TIME AT WHICH THE SYMPTOM WHICH WILL TRIGGER THE ACTION OCCURS.

- o DETERMINE THE LAST TIME AT WHICH THE ACTION CAN BE COMPLETED AND BE SUCCESSFUL.

- o THE DIFFERENCE IN THESE TWO VALUES IS THE AVAILABLE TIME.

Transient Analysis

Determination of the time available to take a procedural action is a critical step in the quantification of the non-success probability for the action. The determination must come from a transient analysis of the accident sequence. Two times must be determined. The first is the time at which the symptom occurs which will trigger the action. The second is the latest possible time at which the action can be taken and accomplish its purpose. The difference in these two times is the available time.

For those actions which are found to be time constrained, a realistic determination of the available time is extremely important.

MEDIAN TIME

- JUDGEMENT

  SELECT A TIME FOR WHICH YOU BELIEVE 50% OF ALL INDIVIDUALS OR GROUPS WILL HAVE SUCCESSFULLY COMPLETED THE ACTION AND 50% WILL EXCEED.

- MEASUREMENT

  HAVE INDIVIDUALS, OR GROUPS, PERFORM THE ACTION ONE TIME, INDEPENDENTLY, AND SELECT THE TIME FOR WHICH 50% OF THE TRIALS ARE LESS AND 50% ARE GREATER.

Median Time

An equally important aspect of the HCR method is the determination of a characteristic time required to take an action. For this purpose, the median time is used - the time for which 50% of the actions in a series of independent operations will exceed.

The median action is best determined by execution of a series of experiments in which the actual time required to take the action can be measured. The median time, then, is just the middle value of the rank ordered time measurements. In general such measurements are not available in which case, expert judgement must be utilized. This judgement should be exercised by operators and simulator instructors whenever possible.

PERFORMANCE-SHAPING FACTORS

	Coefficients
OPERATOR EXPERIENCE (K_1)	
1. Expert, well trained	-0.22
2. Average knowledge training	0.00
3. Novice, minimum training	0.44
STRESS LEVEL (K_2)	
1. Situation of grave emergency	0.44
2. Situation of potential emergency	0.28
3. Active, no emergency	0.00
4. Low activity, low vigilance	0.28
QUALITY OF OPERATOR/PLANT INTERFACE (K_3)	
1. Excellent	-0.22
2. Good	0.00
3. Fair	0.44
4. Poor	0.78
5. Extremely poor	0.92

Performance Shaping Factors

In general, it is considered that the median time required for an action will depend on various factors which can favorably or adversely influence the time required. In the original HCR model three such factors were postulated. These were:

1. Operator experience

2. Stress level

3. Quality of the Operator/Plant Interface

The table in the viewgraph indicated a means for characterizing the nature of these factors and quantifying them. Of course, if the median time has been determined by measurement of the actual action, such corrections may not be needed.

PP&L has reservations about the use of performance shaping factors of this type.

PERFORMANCE SHAPING FACTORS

APPLICATION

- ADJUST THE VALUE OF MEDIAN TIME SELECTED BY THE EQUATION:

$$T_M = (1 + K_1)(1 + K_2)(1 + K_3)\, T_M^*$$

T_M^* = THE MEDIAN TIME

T_M = THE ADJUSTED TIME

K_1 = EXPERIENCE FACTOR

K_2 = STRESS FACTOR

K_3 = INTERFACE QUALITY FACTOR

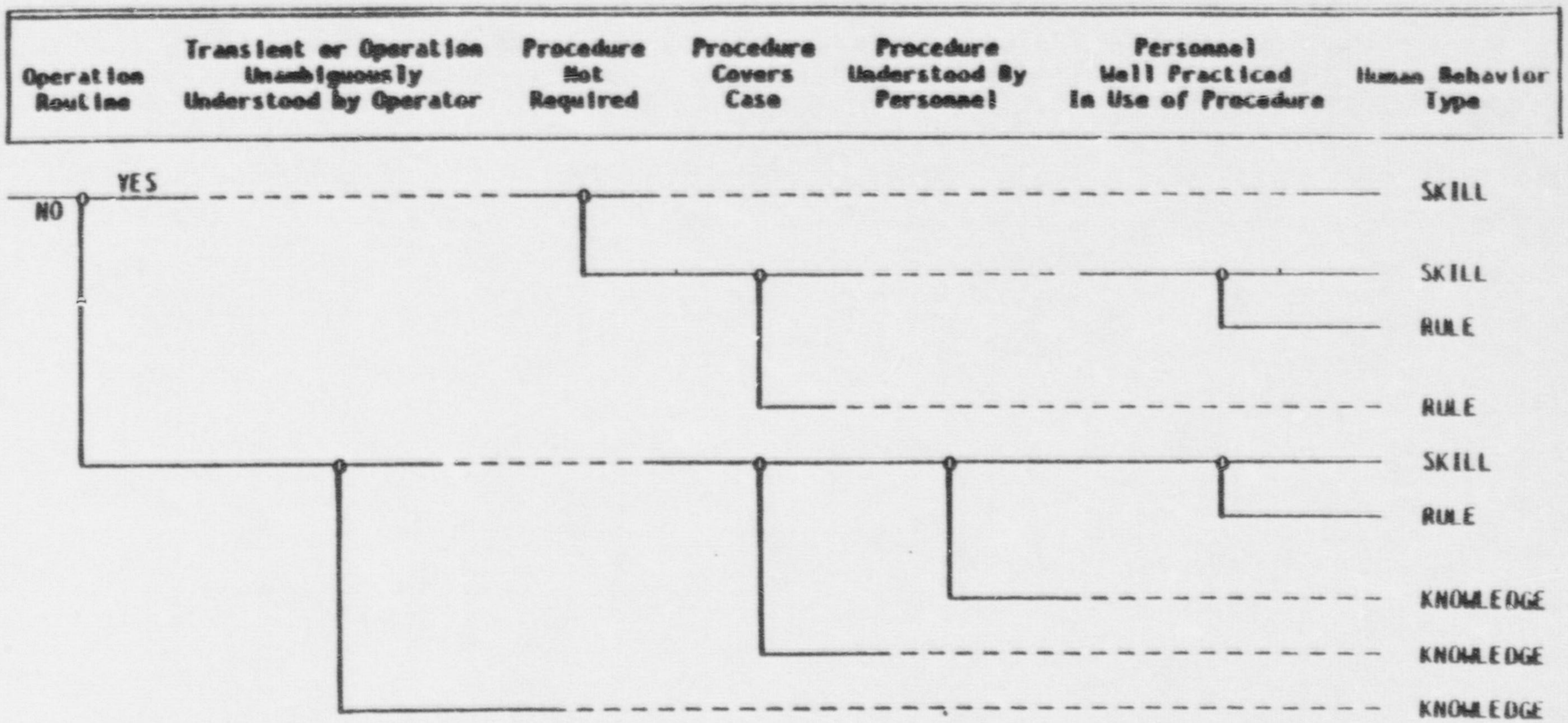
- NORMALIZED TIME

$$T = T_C / T_M^*$$

T_C = AVAILABLE TIME FROM ANALYSIS

Performance Shaping Factor Application

If performance shaping factors are used, they are applied in conformance to the equation shown in the viewgraph. This viewgraph also shows the determination of the dimensionless time parameter by dividing the actual time by the adjusted median time. This normalization is also performed for the available time.



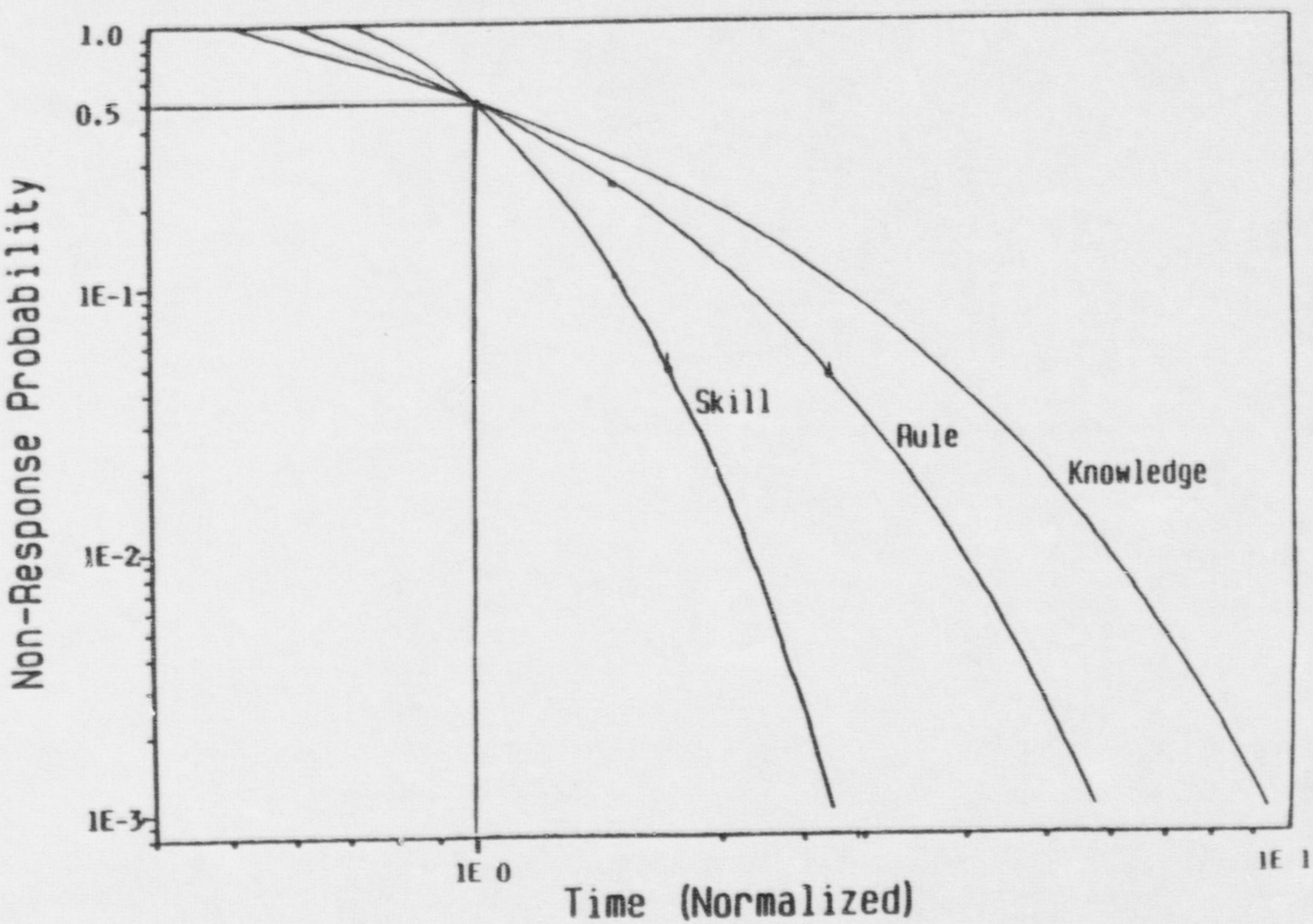
AN APPROACH FOR CHARACTERIZING THE TYPE OF COGNITIVE PROCESSING

An Approach for Characterizing the Type of Cognitive Processing

Use of the HCR curve requires a determination of whether the action is skill, rule, or knowledge based. One approach to this determination is to use the series of questions shown on the viewgraph.

PP&L also has reservations on this characterization of cognitive processing types. First, we would expect all EOP actions to be either skill or rule based. Actual measurements at the Susquehanna simulator have shown Weibull distribution parameters having even sharper time variation than the skill based curve. We believe that the use of the skill based curve is probably conservative for actions where the HCR approach is valid.

USE OF THE HCR MODEL



Use of the HCR Model

In the preceding steps all of the parameters necessary to determine the operator non-success probability for an action have been determined. To determine the non-success probability one simply determines the value on the appropriate curve at the normalized value of the available time for the action.

SUSQUEHANNA SIMULATOR MEASUREMENTS

BACKGROUND

- A SYSTEM 1 VALIDATION PROGRAM HAD BEEN PLANNED FOR EARLY 1987 TO DEMONSTRATE THE ADEQUACY OF OUR PROCEDURES, OPERATOR TRAINING, AND CONTROL ROOM FACILITIES TO COPE WITH A SEVERE ACCIDENT.
- THE SUSQUEHANNA IPE TOOK A NEW APPROACH TO QUANTIFICATION OF OPERATOR ERROR.
  - MAINTENANCE AND SURVEILLANCE ERRORS ARE IMBEDDED IN HISTORICAL DATA ON UNAVAILABILITY AND INITIATING EVENTS
  - CONTROL ROOM OPERATOR

    ERROR DURING NORMAL OPERATION IS IMBEDDED IN INITIATING EVENT RECORDS

    ALWAYS FOLLOWS PROCEDURES IN RESPONSE TO AN INITIATING EVENT

    ONLY FAILS TO EXECUTE A PROCEDURE WHEN TIME IS LIMITED
- PP&L MANAGEMENT SAW A NEED TO DEVELOP SUPPORTING INFORMATION FOR THE SUSQUEHANNA IPE AND APPROVED USE OF THE SYSTEM 1 VALIDATION FOR THIS PURPOSE.

Susquehanna Simulator Measurements

When the results of the Susquehanna IPE were presented to PP&L management in early 1986, there was considerable skepticism over the treatment of operator error. Many of the managers reviewing the document intuitively believed that control room operators were prone to error. In some cases this attitude was based on considerable experience in plant operations. For this reason a program to perform measurements at the Susquehanna simulator was given high priority. PP&L had planned to perform a series of tests to determine the effectiveness of our EOP procedures and various control room enhancements. This program was titled System 1 Validation.

It was decided to utilize the System 1 Validation program to make measurements of operator performance. The objectives of the two programs were highly compatible since both had the objectives of observing the performance of all Susquehanna operating crews and to subject these crews to the most difficult and challenging accident sequences which would exercise as much of the EOPs as possible. The measurements were scheduled for and performed in early 1987.

SUSQUEHANNA SIMULATOR MEASUREMENT OBJECTIVES

INITIAL MEASUREMENT OBJECTIVES

1. TO DETERMINE THE EXTENT OF CONTROL ROOM OPERATOR FAILURE TO FOLLOW PROCEDURES.

2. TO DETERMINE OPERATOR RESPONSE TIME IN EXECUTING AN ACTION IN RESPONSE TO A SYMPTOM OF AN EMERGENCY SEQUENCE.

3. TO DETERMINE THE ACTUAL DEGREE AND QUALITY OF THE OPERATOR'S UNDERSTANDING OF THE CONTROLLING PHENOMENA IN AN EMERGENCY SEQUENCE.

4. TO DETERMINE THE ADEQUACY OF OUR PROCEDURES.

5. TO DETERMINE THE QUALITY OF OUR SIMULATOR

$1 + 2 \longrightarrow$ HCR MODEL

$3 + 4 + 5 \longrightarrow$ SYSTEM 1 VALIDATION

Susquehanna Simulator Measurement Objective

The objectives of the System 1 Validation program combined with the operator performance measurements are tabulated on the viewgraph. In order to achieve these objectives the actual simulator runs were videotaped and the post test debriefing session was performed by having the crew observe the videotape and explain the reasons for their actions and their thought processes as the event progressed. An experienced facilitator kept the discussion going by use of neutral prompts to obtain further comment when comments become infrequent.

These video tapes and the audio tapes taken during crew debriefing have been used by independent observers to extract times and actions for the scenario. The results of these independent observations have shown almost total agreement with observations taken during the actual measurements.

This observer data, independently verified, when supplemented by data printed and recorded by the simulator computer has provided very reliable information on actual crew simulator performance, and, most important, the crew's reasons for their actions.

SUSQUEHANNA SIMULATOR MEASUREMENT RESULTS

PRELIMINARY OPERATOR PERFORMANCE OBSERVATIONS

- NO CLEAR CASE OF PROCEDURAL ERROR WAS OBSERVED OUT OF APPROXIMATELY 1650 PROCEDURAL STEPS.

- A FEW INSTANCES WERE OBSERVED WHERE PROCEDURAL AMBIGUITY OR LACK OF PRECISION CAUSED QUESTIONABLE RESPONSE ACTIONS.

- OUR GENERAL PERCEPTION OF THE UNANALYZED EXECUTION TIME DATA IS THAT THE SUSQUEHANNA IPE IS PROBABLY CONSERVATIVE.

- THERE MAY BE DEFICIENCIES IN OPERATOR PERFORMANCE IN TAKING ANTICIPATORY ACTIONS.

Susquehanna Simulator Measurement
Results

The results of the System 1 Validation measurements are considered to represent strong support for the operator performance assumption made in the Susquehanna IPE. Only a few apparent deviation from procedures were observed. Subsequent review of the information available revealed two important facts.

1. The procedural step in question was worded ambiguously, and

2. the crews involved had members out of place.

The second of these factors is unavoidable in that Susquehanna has a two unit control room with a single shift supervisor. This means that in one half the crews tested at least one and generally two substitutions are required. As a result of these findings we will give the issue of crew substitution closer attention in future measurements. In the meantime the ambiguities which caused the problems have been corrected. In each instance, however, the crew involved had a very logical interpretation of the procedural instruction in justification of their actual action.

For this reason, we believe that we have seen no instances of procedural error or deviation in measurements which involved well over 1600 procedural steps. We believe that this supports our view that given complete and unambiguous procedures the operators will follow procedures with a very low frequency of errors or deviations. At this time we have no such errors which enable a characterization of such errors. We can only say that the frequency of such errors is probably below 10^{-3}. This rate is small in comparison with equipment failure in general. Since there is no evidence to date of willful deviation, improper execution or failure to observe a symptom, we are inclined to believe that failure to execute the procedural step is dominated by equipment failure and in a small fraction of cases lack of time. In these cases the operator will execute the backup actions, if available, and in most cases avoid the extremely severe consequences which could result from the other causes for error or deviation. At this time we see no evidence that supports the need to consider procedural error involving a failure to understand the plant circumstances and the associated actions required at a level greater than 10^{-3}. We suspect the actual frequency is much below this. We will continue to devote our primary attention to this issue in future simulator measurements.

The measurements of execution time were limited to actions taken within the control room. In general the observed times were shorter than the distributions used in the Susquehanna IPE yielded. Observations of execution times for actions taken outside the control room have not been measured at this time. Based upon the data obtained for control room actions we see much less variance in action time than that implied by the HCR model skill based curve. At this time we do not fully understand the reasons for this, but it may be the result of the small scale tests from which the HCR model parameters were derived are not truly representative of the actions of highly trained and skilled control room operators in following EOPs.

We are concerned over the execution of anticipatory actions. These actions all involve actions outside the control room and cannot be realistically

simulated on the simulator. This may be the result of less attention to these actions in training or more probably a psychological response of the crews to the lack of realism involved in these actions. This area will also receive closer attention in future measurement programs.

THE PP&L VIEW OF OPERATOR ERROR

- o THE PREFERRED SOURCE OF DATA FOR INITIATING EVENTS AND EQUIPMENT UNAVAILABILITIES IS OPERATING PLANT DATA AS OPPOSED TO THEORETICAL MODELS.

- o THE OPERATOR ERROR CONTRIBUTION TO THESE ITEMS IS IMBEDDED IN THE DATA AND NEED NOT BE SEPARATELY CONSIDERED.

- o OPERATOR ERROR IN EXECUTING EOPs IS DOMINATED BY EQUIPMENT FAILURE WITH A SMALL CONTRIBUTION FROM LACK OF TIME FOR ADEQUATE EXECUTION OF THE ACTION REQUIRED IN SOME CASES.

- o THE DOMINANT ISSUE IN ERRORS IN EXECUTING EOPs IS OPERATOR FAILURE TO FOLLOW A PROCEDURAL STEP OR TO EXECUTE IT PROPERLY.

- o PP&L BELIEVES THE DATA ON SUSQUEHANNA OPERATOR PERFORMANCE TO DATE SUPPORTS AN ERROR RATE OF THIS TYPE OF LESS THAN 10^{-3}.

- o WE BELIEVE THAT WHEN SUFFICIENT DATA IS AVAILABLE TO PROPERLY QUANTIFY AND CHARACTERIZE SUCH ERRORS, THEY WILL BE FOUND TO HAVE A VERY MINOR CONTRIBUTION AT MOST.

The PP&L View of Operator Error

In the case of human error which leads to an initiator or failure of equipment, we much prefer actual operating experience as a basis for the data used. In the original Susquehanna IPE this was not possible since little operating experience had been developed in mid-1985 at Susquehanna. for that reason, we used generic data considered to be generally applicable to BWR and plants, in some cases, where generic data was lacking, we used a precursor model which imposes the limited plant experience on a theoretical model to derive a partially theoretical partially operating experience based model. We consider this to be superior to a completely theoretical model. This technique is most useful for very highly reliable systems which rarely experience complete failures, but do experience occasional partial failures (unavailability of one division or channel) which may be considered as "precursors" to a complete failure. This technique was used in the Susquehanna IPE for the RHR suppression pool cooling mode (the limiting safety related mode of RHR which envelops all other contributions to risk). It would also have been very useful to have applied the technique to the scram function as well. Unfortunately we had neither the time nor resources to do so. Instead we simply adopted the NRC values presented in NUREG-0460 as reasonable estimates of the actual value.

It is important when the precursor method is used to examine the model and the data very closely when the precursor technique yields a significantly different value of unavailability than the theoretical model. In general, the source of such a discrepancy is the treatment of common mode failure in the theoretical model. In the case of RHR the pure theoretical model and the precursor model gave comparable results of a very credible nature. Loss of the function was due to one division in maintenance with sufficient equipment failures in the other division to make it unavailable.

We have recently completed a precursor model for the scram function using LER data for the operating experience data base. When the precursor model was applied to the pre-10CFR50.62 design for Susquehanna, we found that the expected probability of failure to scram (due to SDV system failure) was very close to the NUREG-0460 value if a two to one partitioning between electrical and mechanical failure is assumed.

We continue to believe the contribution due to error in following EOPs represents a minor contribution at most. Some care must be taken with interpretation of this assumption, however, since even very low probabilities of operator error here can result in a dramatic increase in the frequency of sequences having severe consequences if the cause is failure of the operator to understand the seriousness of the failure to follow the procedure or failure to be aware that the action is required. This comes about since, in general not only must equipment fail, but a backup mitigating action must also fail due to equipment failure. This means that the operator error rate for which no backup action can be postulated must be in the range of 10^{-4} or greater in order for such errors to be a significant contribution. We believe that our expectation of a low probability below this range is a reasonable expectation for Susquehanna crews based on information available at this time.

A CAUTION

- o THE PP&L VIEW IS CRITICALLY DEPENDENT ON THE EXISTENCE OF:

  - CAREFULLY DERIVED EOPs, EXTENSIVELY TESTED AGAINST A BROAD RANGE OF ACCIDENT SCENARIOS

  - EXTENSIVE TRANSIENT ANALYSIS OF ACCIDENT SEQUENCES TO OPTIMIZE THE EOPs

  - A THOROUGH PROGRAM OF TRAINING OF THE OPERATORS ON EOPs AND INDOCTRINATION IN THE IMPORTANCE OF RIGOROUSLY FOLLOWING THESE PROCEDURES

- o IN THE ABSENCE OF ANY OF THESE REQUIREMENTS, THE PP&L TREATMENT OF CONTROL ROOM OPERATOR PERFORMANCE IN FOLLOWING EOPs IS PROBABLY NOT VALID.

- o WE DO NOT BELIEVE THERE IS ANY CREDIBLE WAY TO CHARACTERIZE AND QUANTIFY OPERATOR ERROR IN THAT CASE.

A Caution

The PP&L attitude in preparation of EOPs and the training of our operators was that the threat of a severe accident is real and our primary defense against such an event would be good procedures and operators well trained in their use. The view that the risk of a severe accident can better be controlled by sufficient attention to good maintenance practices and concentration of operator training on normal operating conditions was shown by the results of the Susquehanna IPE to be largely invalid. These actions primarily are directed at avoiding transients which lead to an initiating event and the Susquehanna IPE results show that the contribution of transients to plant damage frequency is very small even with the extremely conservative transient initiator frequencies used in the analysis. Such occurrences are a major economic factor, however, and therefore deserve close attention in order to avoid poor financial performance in operation of the plant.

If careful attention is not devoted to EOPs and EOP training, we simply do not know how to quantify and characterize the nature of operator errors to be expected. We would expect a much higher frequency of such procedural errors and, further, we would expect a high likelihood of very severe consequences from them. We find the prospect of dealing with this type of problem very negative, and we are convinced that we are much better off by avoiding that type of situation. We believe that in the long run the economics are more favorable if the investment in procedure development and training are made. We believe that closure of the severe accident issue may not be possible otherwise and as a result, a long series of escalating requirements and unduly expensive plant modifications could be expected.

We believe we can obtain closure on the severe accident issue by our approach with a minimum of plant modifications. This does not mean no modifications however.

CONVENTIONAL PRA CONSERVATISMS

CONVENTIONAL PRA CONSERVATISMS

- FAILURE TO:
  - INITIATE SLCS
  - CONTROL WATER LEVEL IN ATWS
  - PREVENT HPCI/RCIC SUCTION TRANSFER AND HIGH BACK PRESSURE TRIP
  - DEPRESSURIZE WHEN THE HEAT CAPACITY TEMPERATURE LIMIT (HCTL) IS REACHED IN ATWS
  - SHUTDOWN BY MANUAL ROD INSERTION FOR SLCS FAILURE IN ATWS
  - PARTITION ATWS EVENTS
  - ACCOUNT FOR SAVING THE REACTOR PRESSURE VESSEL
  - PROVIDE BACKUP WATER INVENTORY FOR THE CONDENSATE STORAGE TANK
  - STRIP NON-ESSENTIAL DC LOADS IN STATION BLACKOUT
  - DEPRESSURIZE EARLY IN STATION BLACKOUT TO ALLOW SUCCESSFUL FIRE MAIN INJECTION
  - MAKE EARLY FIRE MAIN CONNECTION TO RHR SERVICE WATER (RHRSW)
  - USE ALTERNATE WATER INJECTION SOURCES
  - USE ALTERNATE DEPRESSURIZATION METHODS
  - USE REACTOR WATER CLEANUP (RWCU) BLOWDOWN FOR DHR IN TRANSIENTS WITH RHR FAILURE
  - INCREASE THE HEAT CAPACITY TEMPERATURE LIMIT (HCTL) CURVE FOR ATWS
- USE OF UNREALISTIC COMMON MODE FAILURE (CMF) TREATMENT

Conventional PRA Conservatism

In the performance of the Susquehanna IPE a number of potential operator actions using non-safety systems and using safety related equipment in unconventional ways was found to have a dramatic impact on the calculated frequency of plant damage. In order to derive the full benefit of these actions, however, it was necessary to improve some of the conventional models used to represent the plant and to modify the HCTL for ATWS. These improvements also involved,

- common mode failure treatment
- ATWS partitioning
- Arresting core damage progression before reactor vessel failures

Taking more realistic credit for the various actions and modeling assumptions in this list can reduce calculated plant damage frequency by about two decades. More important, to do so has a dramatic impact on the plant damage profile reducing the conditional probability of containment failure by as much as two decades or more also.

ANTICIPATED TRANSIENT WITHOUT SCRAM

(Frequency - yr^{-1})

Plant Damage State	Present IPE	One SLCS Pump	No ARI	No Partitioning	No HPCI Bypass	(1) Procedural Error (0.1)	WASH 1400	NUREG 1150
1. No Damage (2)								
2. Reactivity Transients	$9.3x10^{-9}$	$1.8x10^{-8}$	$2.2x10^{-7}$	$2.2x10^{-6}$	$4.5x10^{-5}$	$4.1x10^{-5}$		
3. Metal-Water Reaction								
4. Core Melt and Vessel Failure								
5. Vent + 1 (2)								
6. Vent + 2								
7. Vent + 3								
8. Vent + 4								
9. 1 + COPF (3),(4)								
10. 2 + COPF	$6.6x10^{-12}$	$6.6x10^{-12}$	$1.9x10^{-11}$	$2.2x10^{-11}$	$9.0x10^{-11}$	$8.1x10^{-11}$		
11. 3 + COPF								
12. 4 + COPF								
13. COPF + 4	$2.2x10^{-11}$	$2.2x10^{-11}$	$6.3x10^{-11}$	$7.2x10^{-11}$	$7.2x10^{-11}$	$2.0x10^{-5}$		
TOTAL	$9.3x10^{-9}$	$1.9x10^{-8}$	$2.2x10^{-7}$	$2.2x10^{-6}$	$4.5x10^{-5}$	$6.1x10^{-5}$	$1.3x10^{-5}$	$1.0x10^{-6}$

(1) The dominant contribution is failure to initiate SLCS or any subsequent mitigating action.
(2) Not included in total because no significant release.
(3) Sequence indicates order of damage occurrence.
(4) COPF = containment overpressure failure.

Anticipated Transients Without Scram

In this table we have shown the results of systematically stepping back from the Susquehanna IPE representation of ATWS to the situation which prevailed at the time of WASH-1400. It is easily seen that the plant and procedural modifications made since that time have a dramatic effect on core damage frequency.

The single assumption of an operator error rate of 10% in failing to initiate SLCS not only has a dramatic effect on core damage frequency, but also increases the frequency of containment overpressure failure by over five decades. In comparing the values derived to WASH-1400, it is important to note that the type of core damage identified here (Plant Damage State 2) would not have been considered to be core damage and so the 4.1×10^{-5} value under Procedural Error would have zero in WASH-1400. On this basis the comparison to WASH-1400 is quite good.

The comparison with NUREG-1150 must consider that NUREG-1150 did give credit for ARI and two pump equivalent SLCS. The "No Partitioning" column most directly compares to the NUREG-1150 models, but without consideration of manual rod insertion to save the containment.

STATION BLACKOUT

(Frequency - yr^{-1})

Plant Damage State	Present IPE	Correction For Deficiencies (1)	No Fire Main Injection	No Early Venting	No DC Load Shedding Credit	No Fifth Diesel	WASH 1400	NUREG 1150
1. No Damage (2)								
2. Reactivity Transients	1.2×10^{-9}	1.3×10^{-9}	1.3×10^{-9}	1.3×10^{-9}	1.3×10^{-9}	1.3×10^{-9}		
3. Metal-Water Reaction	7.6×10^{-8}	1.4×10^{-7}	1.5×10^{-7}	2.3×10^{-7}	5.9×10^{-7}	2.8×10^{-6}		
4. Core Melt and Vessel Failure (2)	3.9×10^{-8}	8.8×10^{-8}	1.0×10^{-7}	2.7×10^{-7}	9.1×10^{-7}	5.0×10^{-6}		
5. Vent + 1	8.5×10^{-6}	3.0×10^{-6}	2.8×10^{-6}	0	0	0		
6. Vent + 2								
7. Vent + 3	(5)	2.1×10^{-8}	7.6×10^{-8}	0	0	0		
8. Vent + 4	1.6×10^{-8}	6.0×10^{-8}	1.6×10^{-7}	0	0	0		
9. 1 + COPF (3),(4)								
10. 2 + COPF								
11. 3 + COPF								
12. 4 + COPF	1.4×10^{-9}	2.1×10^{-9}	2.0×10^{-9}	2.0×10^{-8}	2.0×10^{-8}	1.4×10^{-7}		
13. COPF + 4								
TOTAL	1.3×10^{-7}	3.1×10^{-7}	4.9×10^{-7}	5.2×10^{-7}	1.5×10^{-6}	7.8×10^{-6}	1×10^{-7}	7×10^{-6}

(1) New DC loads plus non-obtuse off-site lines.
(2) Not included in total because no significant release.
(3) Sequence indicates order of damage occurrence.
(4) COPF - containment overpressure failure
(5) Not calculated

Station Blackout

A similar sensitivity study performed for Station Blackout shows similar results. Here the various operator actions called out by the Susquehanna EOPs reduce vessel failure frequency by over one decade with similar improvement for containment failure. The addition of a fifth diesel generator to the plant produced another decade improvement in these plant damage states.

This result then compares well with the recent NUREG-1150 results for Peach Bottom.

INFLUENCE OF THE DIFFERENCES

(FREQUENCY, PER YEAR)

SEQUENCE TYPE	PEACH BOTTOM*		SUSQUEHANNA	
	OCCURS	DAMAGE	OCCURS	DAMAGE
STATION BLACKOUT	2×10^{-4}	8.6×10^{-6}	1.4×10^{-4}	1.3×10^{-7}
LOCA & ATWS	1.1×10^{-7}	-	5.9×10^{-8}	5.9×10^{-8}
ATWS	8.6×10^{-5}	4.1×10^{-6}	2.0×10^{-4}	9.3×10^{-9}
LOCA	3.7×10^{-3}	1.7×10^{-6}	3.3×10^{-3}	8.5×10^{-9}
TRANSIENTS	2.9	6.2×10^{-6}	8.1	5.3×10^{-9}
LOCA OUTSIDE CONTAINMENT	?	3.8×10^{-7}	-	E
RPV RUPTURE	3.1×10^{-7}	3.1×10^{-7}	-	E
SPECIAL INITIATORS	?	6.1×10^{-7}	-	E
TOTALS	-	2.2×10^{-5}	-	2.1×10^{-7}

* FROM THE PEACH BOTTOM IPE.

Influence of the Differences

The IDCOR IPEM Appendix D provides models that are essentially comparable to those used in WASH-1400. This table shows the rather interesting impact that the conservatisms identified can have.

It is seen that the frequency of occurrence of the various initiators is not greatly different between Appendix D and the Susquehanna IPE. The frequency of plant damage is radically different, however. The differences caused by various models and assumptions for ATWS and Station Blackout were shown on the previous tables. Similar transformations could easily be developed for each of the initiators to demonstrate the origin of the difference in results.

REASONS FOR THE DIFFERENCES

SBO

- o FAILURE TO:
  - CONSIDER VENTING
  - EXTEND DC POWER ENDURANCE
  - CONSIDER MASS ADDITION TO POOL
  - PROVIDE BACKUP FOR VESSEL INJECTION
  - DEPRESSURIZE EARLY
  - CONSIDER BACKUP DIESEL (SUSQUEHANNA UNIQUE)
  - CONSIDER BACKUP FOR DRYWELL SPRAYS

ATWS

- o FAILURE TO:
  - PARTITION ATWS EVENTS
  - ASSURE HPCI OPERATION
  - CONTROL LEVEL OR INJECT BORON
  - CONSIDER MANUAL ROD INSERTION
  - TAKE CREDIT FOR POWER REDUCTION BY PRESSURE REDUCTION
  - RAISE HCTL CURVE
  - USE LOW PRESSURE BACKUP INJECTION

TRANSIENTS

- o FAILURE TO:
  - VENT
  - CONSIDER RWCU FOR DECAY HEAT REMOVAL
  - CONSIDER ALTERNATIVE DEPRESSURIZATION METHODS
  - CONSIDER BACKUP VESSEL INJECTION SOURCES
  - CONSIDER BACKUP INVENTORY FOR THE CST
  - CONSIDER BACKUP FOR DRYWELL SPRAYS

Reasons for the Differences

This viewgraph lists the various contributors to the difference in results between the Appendix D and the Susquehanna results. It is true that not all BWR plants may have all of the various plant capabilities exploited by Susquehanna EOPs. On the other hand, many plants may have other capabilities not available at Susquehanna. The plant specific differences can have a very strong influence on calculated values of plant damage frequency and on the dominant contributors to it.

THE REQUIREMENTS FOR ELIMINATION OF CONSERVATISMS

- o THE CONTRIBUTIONS OF THESE CONSERVATISMS MAY BE ELIMINATED ONLY IF:
  - THE ACTIONS ARE SPECIFICALLY INCLUDED IN THE PLANT EOPS
  - ANALYSIS HAS BEEN DONE TO SHOW THAT THE ACTION WILL BE SUCCESSFUL
  - OPERATORS HAVE RECEIVED TRAINING ON THE REASONS FOR THE ACTIONS AND IN EXECUTION OF THE ACTIONS
- o OTHERWISE SOME CONTRIBUTION MUST BE CONSIDERED:
  - WE DO NOT KNOW HOW TO QUANTIFY THE PROPER CONTRIBUTION
  - WE DOUBT THAT THERE IS A CREDIBLE METHOD FOR QUANTIFICATION

The Requirements for Elimination of Conservatisms

If the negative influence of the conventional PRA assumptions identified are to be avoided in a credible manner, it is essential that the plant EOPs call for the necessary actions at the appropriate time and that the operators have been trained and understand the reasons for the actions. Obviously, a credible analysis must also be performed to assure that the actions are effective.

If the procedure and training do not cover an action, we believe that some contribution from failure to take the action must be considered. Since this failure would be based on the operator's lack of knowledge of the need for the action, we do not know how to quantify the failure probability. Our inclination is to assume a 100% probability of failure since we do not know how to quantify it. In the absence of EOP and training guidance we believe, in general, that the failure rate would be high.

CATEGORIES OF UNCERTAINTY

CATEGORIES OF UNCERTAINTY

- o PHENOMENA
  - PROCESS MODELS
- o DATA
  - PHYSICAL DATA
  - EQUIPMENT PERFORMANCE
  - HUMAN PERFORMANCE

Categories of Uncertainty

The factors which can result in uncertainty in the results of a risk analysis are quite complex and diverse in nature. The presentation given here is intended to present a rather simplified view of the issue and to point out the most critical factors to be considered.

We consider uncertainty to have two basic sources, phenomena and data, which are fundamentally different in their nature and their impact on the results of the analysis.

PROCESS MODEL UNCERTAINTIES

- o MAY BE A RESULT OF
  - MATHEMATICAL APPROXIMATIONS
  - USE OF EMPIRICAL METHODS
  - LACK OF A MECHANISTIC MODEL

- o PROCESS MODELS OF CONCERN
  - CORE DAMAGE PROGRESSION
  - REACTOR VESSEL FAILURE
  - CORE DEBRIS ATTACK ON CONTAINMENT COMPONENTS
  - NATURE AND MAGNITUDE OF THE CONTAINMENT FAILURE
  - FISSION PRODUCT RELEASE FROM FUEL
  - FISSION PRODUCT TRANSPORT PROCESSES
  - METEOROLOGICAL PROCESSES
  - VARIOUS RADIATION EXPOSURE PROCESSES
  - HEALTH EFFECTS FROM RADIATION EXPOSURE

Process Model Uncertainty

There are a variety of types of uncertainties in the various process models used in risk analysis. By process model we mean mathematical description of some physical process. The uncertainties may be a result of:

1. mathematical approximations,

2. use of empirical methods, or

3. lack of a comprehensive mechanistic description of the physical processes and their interactions.

Examples of process models for which this source of uncertainty is important are:

1. the core damage progression process,

2. the reactor vessel failure process,

3. the attack of core debris on various containment components,

If we were to decide to use fission product release to the environment to judge adequacy of our plant against severe accidents, we would need to include:

4. the nature and magnitude of the containment failure process,

5. the process of fission product release from the fuel and

6. the nature of transport processes for fission products.

If we were to use off-site consequence as the basis for our judgements we would need to include in addition:

7. meteorological processes,

8. the various exposure processes, and

9. the assessment of health effects resulting from exposure.

If we consider this sequence of nine process issues, it becomes clear that each of them has a very high level of uncertainty that has its basis in our inability to construct a credible mathematical model which can be shown to be consistent or compatible with observation. In some cases we can say that there have been no observations and therefore there is nothing to compare against. In others we can say that the observations are so obscured by interfering phenomena that the statistical level of confidence is very low.

For this reason PP&L prefers to limit consideration to only the first three of these items. These are sufficient to derive an estimate of the frequency of the conditions which could have severe off-site consequences. While the uncertainties in these first three items are very large, we can show by our probabilistic analysis that the frequency of the first is very low, the frequency of the second is a small fraction of the first, and the frequency of

the third is, in our opinion, only a small fraction of the second. This strategy of focusing on the progression from core damage to core debris attack on containment components clearly demonstrates the progressively lower probability as the severity of the plant damage state becomes worse. This permits the impact of the uncertainties in these phenomena to be bounded so that the most severe condition - core melt, vessel failure, and unquenchable core debris causing severe containment challenges - is clearly constrained to a very low contribution to the overall uncertainty in the calculations.

Finally, in the case of the remaining six items, we can say that the mathematical approximations are so poor, that the results are very questionable. For all nine it may be stated with some confidence that all three of the sources of uncertainty cited above influence the final uncertainty in off-site consequence. When an attempt is made to judge adequacy of a plant on the basis of such a performance indicator, there is a virtually unlimited field of opportunity for undisciplined and unfocused controversy. The general approach to avoiding such controversy is to use such conservative models that the room for controversy is limited. Unfortunately, this produces answers which PP&L believes to be unrealistic and negative. Even worse, this practice obscures the actual nature of the early stages involving the failure progression and obscures the nature of the accident sequence and the opportunities for operator intervention.

TREATMENT OF PROCESS MODEL UNCERTAINTIES

o FOR PROCESS MODELS

- THE DOMINANT ISSUE IS THE CHOICE OF ALTERNATIVE MODELS

- THESE MODELS ARE OFTEN CHOSEN ON AN INTUITIVE BASIS RATHER THAN AN EMPIRICAL BASIS

- THE UNCERTAINTIES IN THE PARAMETERS OF THE MODEL CHOSEN ARE OF LESS CONSEQUENCE IN GENERAL

o THE EFFECTS OF AN IMPROPER MODEL CHOICE SHOULD NOT BE TREATED AS AN UNCERTAINTY OR A STATISTICAL VARIABLE, BUT SHOULD ONLY BE LOOKED AT AS A SENSITIVITY.

Treatment of Process Model Uncertainties

The nature of the uncertainty in what we have characterized as process models is generally not a matter of degree but, rather, is a matter of kind. The most immediate example of this is the contrast between the core damage progression model of MAAP-BWR in comparison with that of BWRSAR. In the former, there is early channel blockage resulting in inability to cool the core, in situ, with resulting melt of large quantities of core material. When this molten mass reaches a certain magnitude, it is released to the lower plenum where failure of a penetration occurs immediately regardless of water inventory in the lower plenum.

In contrast, the BWRSAR model sequentially melts and relocates control rods, channel boxes, and fuel clad without channel blockage. The fuel later relocates to the core support plate before melting and causes local failure of the core support plate due to high temperature. The core debris then falls into the lower plenum, unmolten, where it is quenched. Failure of a lower vessel head penetration does not occur until boil off of the lower plenum water is complete and the debris begins to reheat.

PP&L feels that the BWRSAR model is the more credible for the great majority of BWR accident sequences. Only in the case that water level is some how maintained in the lower few feet of the core, and so protects the core support plate, would the MAAP description of core damage progression have credibility. Based on our current information, however, we believe that such sequences are extremely unlikely and that nearly all BWR core damage sequences involve complete core dryout.

If one wishes to address the issue of the consequences of having made the wrong choice, one can do a sensitivity calculation, that is, perform the calculations using the rejected model to observe the consequences. Sometimes this is useful to determine whether or not actions could be taken to accommodate the consequences of either model.

We object, however, to attaching a probability to the validity of each model and using a weighted average of the results to characterize risk. We believe this is an improper use of probabilistic methods.

PHYSICAL DATA UNCERTAINTY

- MEASURED SYSTEM PARAMETERS ARE NOT INCLUDED

- PHYSICAL DATA PARAMETERS INCLUDE:
  - HEAT TRANSFER COEFFICIENTS,
  - MASS OF VESSEL COMPONENTS,
  - VESSEL INTERNAL VOLUMES,
  - SPECIFIC HEAT OF MATERIALS, ETC.

- THESE UNCERTAINTIES ARE REFLECTED AS UNCERTAINTIES IN TRANSIENT TIMES

- THE TIME VARIATION IS STRONGLY CORRELATED THROUGH THE DATA PARAMETER VARIATIONS

Physical Data Uncertainty

Physical data is the data used in transient models that cannot be directly measured from a system test. HPCI flow rate, for example would not be considered as physical data on this basis. Physical data involves such parameters as

1. heat transfer coefficients,

2. mass of vessel components,

3. vessel internal volumes,

4. specific heat of materials, etc.

The primary influence of variability of parameters of this type will be to determine the variation in timing of the various transients. The effect of physical data variability, therefore, must be reflected in uncertainty in the various time constraints used in the various success criteria.

Extreme care must be taken in representing such uncertainty, however, since the variability in the various times derived are strongly correlated through the variation in the physical data. If the time variations are important, treating the uncertainties in the various input times as independent could lead to a serious misrepresentation of the final resulting distributions.

EQUIPMENT PERFORMANCE UNCERTAINTIES

- o THREE CATEGORIES MUST BE CONSIDERED
  - UNAVAILABILITY
  - FAILURE THRESHOLD
  - PERFORMANCE PARAMETERS

- o UNAVAILABILITY AFFECTS FAULT TREE AND EVENT TREE BRANCHING PROBABILITIES AND IS AN INDEPENDENT RANDOM INPUT VARIABLE

- o FAILURE THRESHOLD AND PERFORMANCE PARAMETERS INFLUENCE TIMES IN VARIOUS SUCCESS CRITERIA. THESE VARIATIONS MUST BE TREATED AS CORRELATED VARIABLES IN THE INPUT DATA.

Equipment Performance Uncertainties

Equipment performance uncertainties must be divided into three categories. These are:

1. Unavailability

2. Failure threshold

3. Performance parameters

The unavailability is direct input into the IPE and determines branching probabilities in event trees or fault trees. These uncertainties feed directly into the final uncertainty in plant damage state frequencies.

The other two, however, ultimately determine a time parameter which again will feed into one of the success criteria. The failure threshold is generally a temperature or pressure value beyond which a piece of equipment will not function. The time at which this occurs is determined from a transient calculation and the uncertainty is influenced both by the uncertainty in the transient timing and by the parameter value at which the equipment will actually fail.

Since it is likely that the latter contribution is the greater, the uncertainty in failure time would generally be expected to be not correlated very closely to the parameter variations which determine the transient timing.

The performance parameter will be a physically measured parameter of the system, such as a flow rate. In general, these uncertainties will be rather small in comparison with the actual value of the parameter. The influence of the uncertainty is once again a time uncertainty which may either influence the rate of a transient event or which may set a limit in the transient on the time at which an action must be taken.

In some marginal cases, such as CRD pump success in preventing core damage in a transient event, the actual flow rate may be the critical parameter. That is, above a certain flow rate, no damage will occur, but below that flow rate some form of damage must occur. If the actual CRD pump flow is close to this threshold, the uncertainty in the value of flow may directly impact the distribution of plant damage state frequencies.

When equipment performance uncertainties are to be addressed, it is essential that the characteristics outlined above be kept clearly in mind.

HUMAN PERFORMANCE UNCERTAINTIES

- PROCEDURAL ERROR
  - FAILURE TO FOLLOW PROCEDURE
  - IMPROPER EXECUTION OF PROCEDURAL STEP
- LACK OF ADEQUATE TIME FOR EXECUTION

Human Performance Uncertainties

Human error, which causes initiating events or which contributes to equipment unavailability, is taken into account in derivation of the data used for risk analysis. This leaves for consideration only those human errors which are associated with execution of EOPs. PP&L has devoted considerable resources and effort to development of EOPs, and it has been our intention to derive EOPs which always direct the operator to take the optimum action. Cautions have been provided for actions which are sensitive in some way, and all actions are fundamentally symptom based. In addition there has been a considerable investment in training our operating crews in the use of these procedures by classroom and simulator instruction.

For this reason, PP&L has taken the position that control room operators will follow procedures in response to symptoms, as indicated by control room instruments, without error either in conformance to the procedure or in execution. A more precise, but less clear statement of our position would be: "The error rate of operators in executing procedural steps is very small in comparison with the unavailability of the equipment required for execution of the step." This is an extremely important assumption, because if the operator does not follow the procedure we must consider the reason for his deviation if we are to correctly assess the consequences.

If the operator fails to execute the procedure simply because of improper execution or lack of time, we know that he understands what is required, and, if he fails, we can assume with confidence that he will attempt to execute the backup actions that are nearly always available which will limit the consequences of his failure. These backup actions do not, in general, prevent plant damage, but they do avoid an event having potentially severe consequences. These actions, too, are limited in effectiveness as a consequence of the unavailability of the equipment involved, and should the equipment fail, the consequences may be severe. The probability of such compounded failures, however, becomes extremely low and when combined with the probability of the initiators which can result in such circumstances (ATWS, Station Blackout, and severe LOCA). The overall frequency of such events becomes so low as to represent a negligible threat.

This line of reasoning is the basis for requiring an operator to follow procedures with an extremely low probability of misinterpretation. This probability must be as low as the combined unavailabilities of the equipment needed for the procedural step and for the equipment needed for the backup actions. This will typically yield a value in the range of 10^{-4} to 10^{-5} per demand.

PP&L does not yet have sufficient data to demonstrate this low an error rate for operating crews. At present we can argue that it is less than 10^{-3}. We believe, however, that achieving such a low probability of error is quite practical since we can infer error rates, on the part of commercial air craft crews, of 10^{-7} per demand or less.

VULNERABILITIES IN HUMAN ERROR ESTIMATES

- VALIDITY OF EOPs FOR ALL ACCIDENT SEQUENCES

- INCOMPLETE EVALUATION OF INSTRUMENT FAILURE

- THESE ARE NOT PROPERLY CONSIDERED HUMAN ERROR, BUT ARE A POTENTIAL SOURCE OF ERROR IN OUR RESULTS

- THESE ISSUES REQUIRE CONTINUING ATTENTION

Vulnerabilities in Human Error Estimates

If we can assume that our training of operators is effective and that our operators accept the critical nature of adherence to procedures, we are left with two major vulnerabilities.

1. Validity of the EOPs for all accident sequences

2. Instrument failure

With regard to the first of these, we had a considerable background in analytical studies of ATWS and Station Blackout before our current EOPs were developed, and this information was accommodated in the development of the EOPs. In addition, we have conducted a series of tests, our System 1 Validation, which tested as much as our EOP structure as possible using two or three sequences on each Susquehanna operating crew. As might be expected, we discovered some problems with ambiguity in the EOPs which has since been corrected. The sequences chosen were variations of the dominant sequences from our IPE and represented very complex and difficult sequences.

In the case of instrumentation failure, we have not found this contribution to be significant primarily because of the high level of redundancy in BWR instrumentation. We believe, however, that this source of failure deserves more attention, and we will give it a more detailed examination in our future work.

Both of these sources of failure to execute procedures are not properly to be considered human error, however. The possibility of imperfect EOPs is a very serious concern. Our Integrated Risk Reduction Study has evaluated our EOPs against every plant damage sequence identified by the Susquehanna IPE. In this process we have found a number of problem areas which we will resolve in the process of adopting Revision 4 of the BWROG EPGs into our EOPs.

We are unable to be certain, however, that there are not sequences which we have not considered for which our EOPs are not effective and which we have missed as a consequence of incompleteness in our IPE. We believe the probability of this is low. Our only defense is continued review of our analytical and probabilistic models of the plant.

THE PP&L VIEW OF UNCERTAINTY ANALYSIS

- ONLY MEAN VALUES CAN BE PROPAGATED THROUGH THE RISK ANALYSIS CALCULATION

- PROPAGATION OF DISTRIBUTION FUNCTIONS AND STATISTICAL PARAMETERS REQUIRES MULTIPLE CALCULATIONS FOR RANDOM SAMPLES OF THE INPUT DATA

- CORRELATED INPUT DATA MUST BE TREATED AS EQUALLY PROBABLE SETS OF THE VARIABLES INVOLVED WHICH HAVE BEEN DERIVED FROM A CONSISTENT SET OF INPUT

- THE DISTRIBUTION FUNCTION RESULTS OF THE ANALYSIS MUST BE TREATED AS CORRELATED DATA

The PP&L View of Uncertainty Analysis

The only statistical parameter which can be propagated through a risk analysis without repeated trials is the mean value. That is, if the mean values of all input parameters for the analysis are used, the results of the analysis will also be mean values. If it is desired to develop statistical distribution functions for the results, we believe that it can only be accomplished by means of Monte Carlo techniques.

The approach to developing such results would require that the statistical distribution function of every independent input parameter be specified as input data and that a large number of calculations be performed in which random choices of the independent input variables were made from the distribution function characterizing each. In the case of correlated input variables, such as times from a transient analysis, the random sampling technique must be applied to the selection of input data to the transient calculation. The output would be a set of times for that transient calculation. Several sets of such times would have to be developed and then each set would constitute an equally probable correlated set. In the performance of the risk analysis, then, these sets would be randomly chosen by means of a uniform probability distribution.

The approach therefore to developing the distribution functions for the results of a risk analysis is therefore to identify all independent and correlated sets of input data. Distribution functions would be input for each of the independent variables, and a sufficiently large set of equally probable sets of each of the correlated variables would have to be developed. A large number of risk calculations would then be made by selecting random values for each of the input variables and propagating this input through the analysis.

This process would yield a distribution function then for each plant damage state. Some care would be required in interpreting this data, however, since these distribution functions would be very strongly correlated. In the interpretation it would be necessary to treat the set of results for each calculation as a correlated set. The variations in a given plant damage state could not be viewed as variations of an independent variable.

While this approach to uncertainty analysis appears to be very burdensome and complex, we believe it is the only credible approach to the derivation of uncertainties. We believe, that with a computer program of the type described in these presentations, that it is feasible to carry out such investigations.

SUMMARY

SUMMARY

OUR PURPOSE HAS BEEN TO PRESENT:

- o ISSUES IMPORTANT TO RISK ASSESSMENT
- o HOW TO DO RISK ASSESSMENT
- o HOW TO USE RISK ASSESSMENT

Summary

In closing this presentation we will state our views and observations on:

1. important issues in risk assessment,
2. how to do, and
3. how to use risk assessment.

IMPORTANT ISSUES

o HUMAN ERROR:

- THE IMPORTANCE OF EOPs
- THE IMPORTANCE OF TRANSIENT ANALYSIS
- THE IMPORTANCE OF OPERATOR TRAINING

o CONSERVATISM:

- DO NOT COVER UP IGNORANCE WITH CONSERVATISMS
- DO NOT USE CONSERVATISMS TO COVER SHORTCUTS IN ANALYSIS.
- DO NOT FAIL TO TAKE CREDIT FOR ALL PLANT FACILITIES AND CAPABILITIES
- IF NECESSARY MAKE MINOR MODIFICATIONS TO PERMIT USE OF ALL EQUIPMENT

o PLANT DAMAGE FREQUENCY:

- BE SENSITIVE TO THE FACT THAT THE ANALYSIS IS UNAVOIDABLY INCOMPLETE.
- DO NOT ATTEMPT TO ACCOUNT FOR INCOMPLETENESS BY CONSERVATISMS. THIS ONLY INCREASES RISK.

o THE GOAL FOR SAFE OPERATION:

- DEVELOP ASSURANCE THAT MULTIPLE LAYERS OF FAILURES MUST OCCUR FOR ANY KNOWN INITIATOR TO LEAD TO PLANT DAMAGE.
- IMPOSE A REQUIREMENT FOR ADDITIONAL FAILURES BEFORE LOSS OF CONTAINMENT INTEGRITY CANNOT BE AVOIDED.
- TEST EOPs TO ASSURE ABSENCE OF NEGATIVE CONSEQUENCES AND PRESENCE OF APPROPRIATE ANTICIPATORY ACTIONS.

Important Issues

The position that PP&L has taken on human error is critically dependent on high quality procedures and high quality operator training. If we cannot claim that our procedures are effective against all known accident sequences for the plant and that our operators will rigorously and effectively follow them, we do not know how to make a credible case for the adequacy of plant operation relative to public safety.

We believe that it is essential to avoid conservatisms for the sake of reducing the analytical burden without considering the potential consequences of such conservatisms. We believe the analysis must take full credit for use of all plant capability and provide convincing evidence that such capability will be used for all known accident sequences. In some cases, very minor and inexpensive modifications may be needed to assure effective use and we believe these modifications should be made, subject to a careful review for adverse impact.

We know that our analysis is incomplete and that it may contain potentially important inaccuracies. Nevertheless, it represents our best understanding of our plant and it should not be confused or obscured by conservative assumptions. We do not accept low frequency as a demonstration of adequacy.

We attempt to demonstrate that multiple failures must occur before any form of plant damage can occur for any initiator including ATWS and Station Blackout. Further, we attempt to demonstrate that additional failures would be required before containment integrity is lost given core damage. We have examined our EOPs in depth to assure ourselves that we can effectively use plant capability to achieve these objectives.

HOW TO DO

- o EVENT SEQUENCE ANALYSIS
  - DO NOT BIN FRONT-LINE FUNCTION EVENT TREE ENDPOINTS
  - CARRY ALL SEQUENCES OUT TO ALL POSSIBLE FINAL STATES
- o TIME
  - REPRESENT DIRECTLY IN SUCCESS CRITERIA
  - REPRESENT TIME OF CORE DAMAGE EXPLICITLY IN PLANT DAMAGE STATES
  - REPRESENT TIME OF CONTAINMENT LOSS OF INTEGRITY EXPLICITLY IN PLANT DAMAGE STATES
- o ANALYSIS MODELS
  - ACCURATELY REPRESENT WATER INVENTORY AND ITS LOCATION
  - USE SIMPLE HAND CALCULATIONS WHEN POSSIBLE. CHECK AGAINST TRANSIENT CODES.
  - USE PLANT MEASURED DATA WHEN POSSIBLE.
- o PLANT DAMAGE STATES
  - REPRESENT CRITICAL TIMING
  - SEGREGATE SEVERE CONSEQUENCE SEQUENCES
  - SEGREGATE MINOR CONSEQUENCE SEQUENCES
  - SEGREGATE HIGH UNCERTAINTY SEQUENCES
- o UNCERTAINTY
  - TREAT PHENOMENA ISSUES AS SENSITIVITIES
  - PROPAGATE UNCERTAINTIES DIRECTLY THROUGH THE ANALYSIS
  - PRESERVE VARIABLE CORRELATION IN UNCERTAINTY ANALYSIS

How To Do

In order to achieve the objective of explicit demonstration of plant capabilities in avoiding accidents having severe consequences, we believe that each event sequence must be individually followed out to the complete array of possible final plant states. We have demonstrated how this can be done with the support state method.

We also believe that a proper and explicit representation of time in the analysis and its results is also essential. We have shown how this can be done in the structuring of success criteria. This approach permits an explicit representation of the constraints imposed by Technical Specifications.

It is extremely important to properly represent the amount of water in the reactor vessel and elsewhere. For the reactor vessel this means that a proper representation of free area versus elevation be described and that void fractions in various reactor volumes be properly represented, particularly in the upper plenum. We also recommend the use of hand calculations to check against transient analysis codes whenever possible, but we always prefer actual plant measurements when available. This is an extremely important aspect of the analysis since a proper representation of reactor water inventory along with losses and gains, is a dominant influence on the early stages of timing in an accident sequence.

We believe that it is very important to define a spectrum of plant damage states having adequate resolution to permit segregation of specific classes of events and to explicitly represent the time of core damage and loss of containment integrity. This is particularly important in order to isolate sequences involving high uncertainty or controversy over the phenomena involved. This is necessary to avoid confusion over the actual impact of the uncertainty.

We believe that uncertainty analysis should be performed by direct propagation of random variations in input parameters through the analysis. We urge caution, however, since many of the inputs are correlated variables and to treat them as independent could result in a gross distortion of the results. It is also important to understand that the resulting array of plant damage states are also strongly correlated variables. That is, the individual plant damage states cannot be considered to be independent, and the variations in them are correlated. We do not currently have experience on how to deal with this issue or its significance.

HOW TO USE

- o FUNDAMENTAL PURPOSE

  - DEMONSTRATE ACCEPTABLY LOW LEVEL OF RISK TO THE GENERAL PUBLIC

- o CONSTRAINT

  - THE RESULT IS ONLY A "SNAPSHOT"

- o REQUIREMENT

  - THE ANALYSIS PROCESS MUST BE A CONTINUOUS CYCLE
  - PLANT PERFORMANCE MUST BE MONITORED
  - OPERATOR PERFORMANCE MUST BE MONITORED
  - PROCEDURES MUST BE TESTED
  - ALL CHANGES MUST BE INCORPORATED ON A REGULAR CYCLE

- o THE ANALYSIS PROVIDES A DIRECT MEASURE OF THE QUALITY OF PLANT OPERATION RELATIVE TO PUBLIC RISK

- o WE BELIEVE OFF-SITE CONSEQUENCE IS NOT A GOOD MEASURE. WE PREFER TO FOCUS ON EQUIPMENT AND OPERATOR PERFORMANCE.

<u>How To Use</u>

The fundamental purpose of a probabilistic risk assessment is to demonstrate an acceptably low level of risk to the general public from plant operation. This objective cannot realistically be achieved as a one time effort, however, since the plant, its procedures, and its personnel are changing on a continual basis. We believe, therefore, that the plant must be evaluated on a continuing basis, and that performance monitoring of equipment and operators is an important aspect of the process. Only in this way can a credible evaluation of performance be accomplished. This would include recording initiating events, precursor evaluations of close calls, recording equipment unavailability, and measurement of operator performance in execution of and knowledge of EOPs.

Changes in procedures should be incorporated into the risk analysis also, and any improvements to transient analysis methods should be used to reassess any procedures which could be influenced.

Initially, this would involve a considerable effort, but maintenance of such an approach should be quite reasonable. It is expected that performance tracking requirements already in place could be modified to provide the necessary data with reasonable cost.

We believe that it is essential to redirect efforts at demonstration of safe operation of nuclear plants from the off-site consequence arena to focus on performance of plant equipment and operators. Further, we believe this is the only credible approach to demonstrating safe operation.

APPENDIX 1

ERIC HASKIN DECAY HEAT TABLE

ERIC HASKIN TABULAR DECAY HEAT

Time (Seconds)	Power Fraction	Power (MW)	Cum Power (MW-Sec)
0.00000000CE+00	0.599999987E-01	197.579987	0.000000000E+00
1.00000000	0.571499988E-01	188.194931	192.887451
1.50000000	0.555200018E-01	182.827362	285.642822
2.00000000	0.543700010E-01	179.040405	376.109619
3.00000000	0.520399995E-01	171.367706	551.313477
4.00000000	0.504000001E-01	165.967194	719.980713
6.00000000	0.480699986E-01	158.294495	1044.24219
8.00000000	0.461900011E-01	152.103668	1354.64014
10.0000000	0.447300002E-01	147.295883	1654.03955
15.0000000	0.420500003E-01	138.470642	2368.45532
20.0000000	0.401500016E-01	132.213943	3045.16675
30.0000000	0.375600010E-01	123.685074	4324.66016
40.0000000	0.357199982E-01	117.625946	5531.21484
60.0000000	0.331300013E-01	109.097092	7798.44141
80.0000000	0.313699991E-01	103.301407	9922.42578
100.000000	0.300000012E-01	98.7899933	11943.3359
150.000000	0.277700014E-01	91.4466095	16699.2500
200.000000	0.261900015E-01	86.2436676	21141.5039
300.000000	0.242800005E-01	79.9540405	29451.3867
400.000000	0.229300000E-01	75.5084839	37224.5117
600.000000	0.210199989E-01	69.2188416	51697.2422
800.000000	0.196500011E-01	64.7074432	65089.8672
1000.00000	0.185899995E-01	61.2168579	77682.2500
1500.00000	0.166199990E-01	54.7296448	106668.875
2000.00000	0.152200013E-01	50.1194611	132881.125
3000.00000	0.135399997E-01	44.5872040	180234.437
4000.00000	0.123500004E-01	40.6685486	222862.312
6000.00000	0.106699988E-01	35.1362915	298667.125
8000.00000	0.981500000E-02	32.3207855	366124.187
10000.0000	0.915199891E-02	30.1375275	428582.500
15000.0000	0.819300115E-02	26.9795380	571375.125
20000.0000	0.751199946E-02	24.7369995	700666.437
30000.0000	0.676399842E-02	22.2739342	935720.562
40000.0000	0.623200089E-02	20.5219727	1149699.00
60000.0000	0.548399985E-02	18.0588074	1535506.00
80000.0000	0.503899902E-02	16.5934143	1882028.00
100000.000	0.469300151E-02	15.4540539	2202502.00
150000.000	0.414099917E-02	13.6363096	2929761.00
200000.000	0.374899991E-02	12.3454561	3579305.00
300000.000	0.327500002E-02	10.7845745	4735806.00
400000.000	0.294000003E-02	9.68141937	5759105.00
600000.000	0.246599992E-02	8.12053680	7539300.00
800000.000	0.218699989E-02	7.20178986	9071532.00
1000000.00	0.197100011E-02	6.49050331	10440761.0

Note: 1) Power (MW) = Power Fraction x 3293.0 MW
2) Cum Power (MW-Sec) is obtained using Trapezoidal rule

APPENDIX 2

SHUTDOWN MAKEUP FLOW REQUIREMENTS

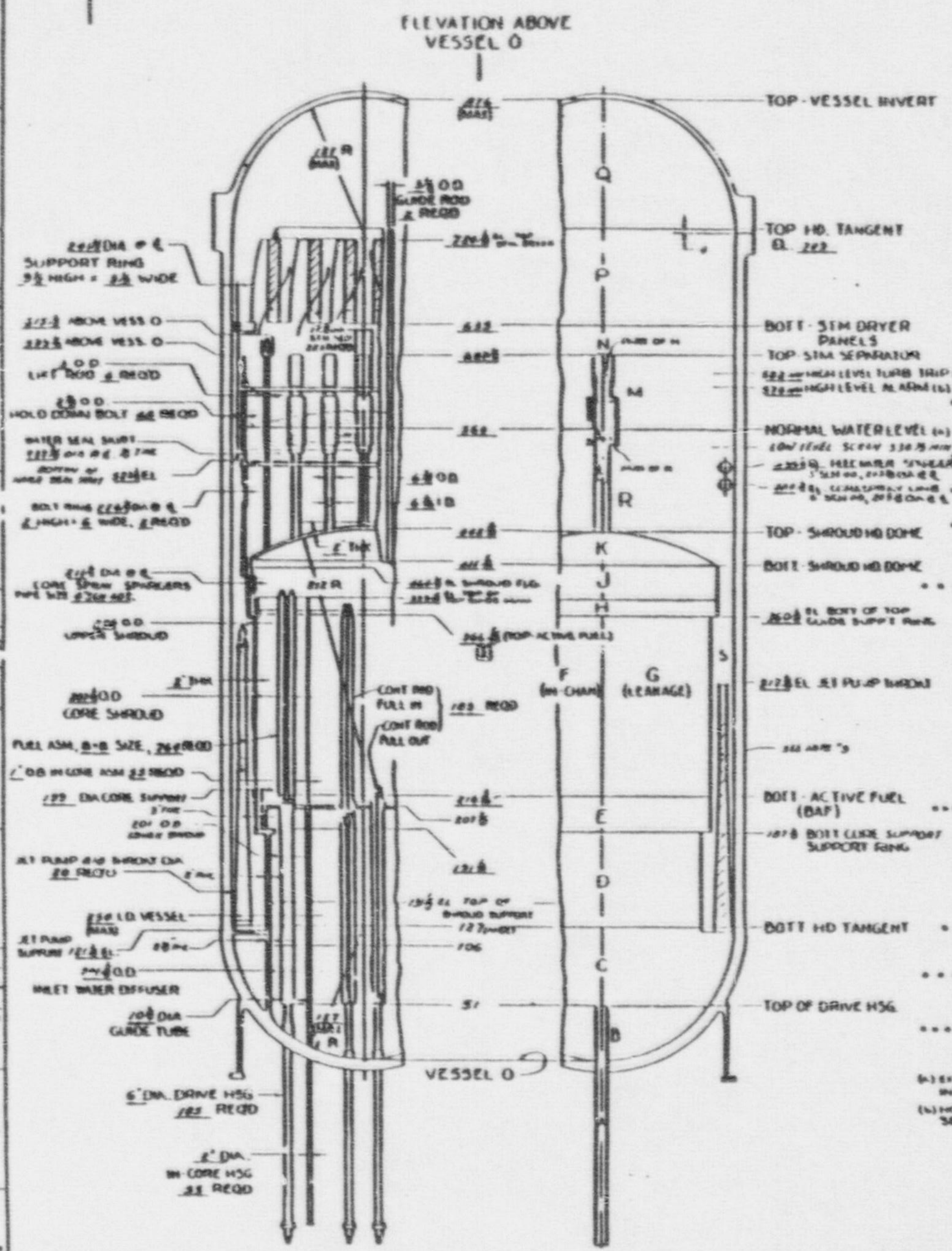
AFTER SHUTDOWN MAKEUP FLOW NEEDED TO KEEP CONSTANT WATER LEVEL
(MINOR PRESSURE DEPENDENCY)

INJECTION ENTHALPY = 58.0 BTU/LBM
INJECTION SPECIFIC VOLUME = .0161 CU. FT./LBM
ENTHALPY OF VAPOR = 1199.19 BTU/LBM

TIME (SEC)	Q(T)-(BTU/SEC)	M(T)-(LBM/SEC)	M(T)-(GPM)
0.000000000E+00	187457.937	164.265320	1187.06348
1.00000000	178553.687	156.462723	1130.67798
1.50000000	173461.062	152.000153	1098.42896
2.00000000	169868.125	148.851746	1075.67725
3.00000000	162588.500	142.472763	1029.57935
4.00000000	157464.687	137.982880	997.133301
6.00000000	150185.062	131.603912	951.035889
8.00000000	144311.375	126.456924	913.841064
10.0000000	139749.875	122.459778	884.955811
15.0000000	131376.750	115.122589	831.933594
20.0000000	125440.625	109.920898	794.343262
30.0000000	117348.687	102.830109	743.101807
40.0000000	111599.937	97.7926025	706.697998
60.0000000	103508.000	90.7018127	655.456543
80.0000000	98009.2500	85.8833771	620.635986
100.000000	93728.9375	82.1326294	593.531494
150.000000	86761.7500	76.0274353	549.412109
200.000000	81825.3750	71.7017975	518.153076
300.000000	75857.9375	66.4726562	480.364502
400.000000	71640.1250	62.7766876	453.655518
600.000000	65672.7500	57.5475922	415.867676
800.000000	61392.4805	53.7968903	388.763184
1000.00000	58080.7187	50.8948669	367.791748
1500.00000	51925.8437	45.5014801	328.816406
2000.00000	47551.8398	41.6686401	301.118408
3000.00000	42303.0000	37.0691986	267.880615
4000.00000	38585.0977	33.8112793	244.337280
6000.00000	33336.2578	29.2118378	211.099396
8000.00000	30664.9922	26.8710632	194.183792
10000.0000	28593.5820	25.0559235	181.066681
15000.0000	25597.3750	22.4304199	162.093475
20000.0000	23469.7227	20.5659943	148.620193
30000.0000	21132.7461	18.5181580	133.821487
40000.0000	19470.6289	17.0616760	123.296219
60000.0000	17071.1562	14.9590836	108.101822
80000.0000	15743.3320	13.7955399	99.6934967
100000.000	14662.3398	12.8482904	92.8481750
150000.000	12937.7187	11.3370419	81.9271393
200000.000	11712.9961	10.2638445	74.1716766
300000.000	10232.0781	8.96614742	64.7938690
400000.000	9185.43750	8.04899883	58.1660767
600000.000	7704.51953	6.75130272	48.7882690
800000.000	6832.83984	5.98746872	43.2684479
1000000.00	6157.99219	5.39611435	38.9950104

APPENDIX 3

BWR VOLUMES AND MASSES



ZONE	ZONE TOTAL VOL (FT^3)	SOLID VOL (FT^3)	FLUID VOLUME (FT^3) SUB CLD	SAT	STM	FLUID WEIGHT (KIPS) HEATED	HOT	COLD
A_1	[illegible]	[illegible]	[illegible]				[illegible]	[illegible]
A_2	[illegible]	[illegible]	[illegible]			[illegible]		
B	[illegible]	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]
C_1	[illegible]	[illegible]	[illegible]				[illegible]	[illegible]
C_2	[illegible]	[illegible]	[illegible]			[illegible]		
D_1	[illegible]	[illegible]	[illegible]				[illegible]	[illegible]
D_2	[illegible]	[illegible]	[illegible]			[illegible]		
E_1	[illegible]	[illegible]	[illegible]				[illegible]	[illegible]
E_2	[illegible]	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]
F	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
G_1	[illegible]	[illegible]	[illegible]			[illegible]		
G_2	[illegible]	[illegible]	[illegible]				[illegible]	
H	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
J	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
K	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
L	[illegible]	[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
M-N	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]	[illegible]
P	[illegible]	190			[illegible]	[illegible]	[illegible]	[illegible]
Q	[illegible]				[illegible]	[illegible]	[illegible]	[illegible]
R	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
S	[illegible]	[illegible]	[illegible]			[illegible]	[illegible]	[illegible]
TOTALS	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

* A, C, D, E, - CONTROL RODS FULL IN

** A, C, D, E, - " " " OUT

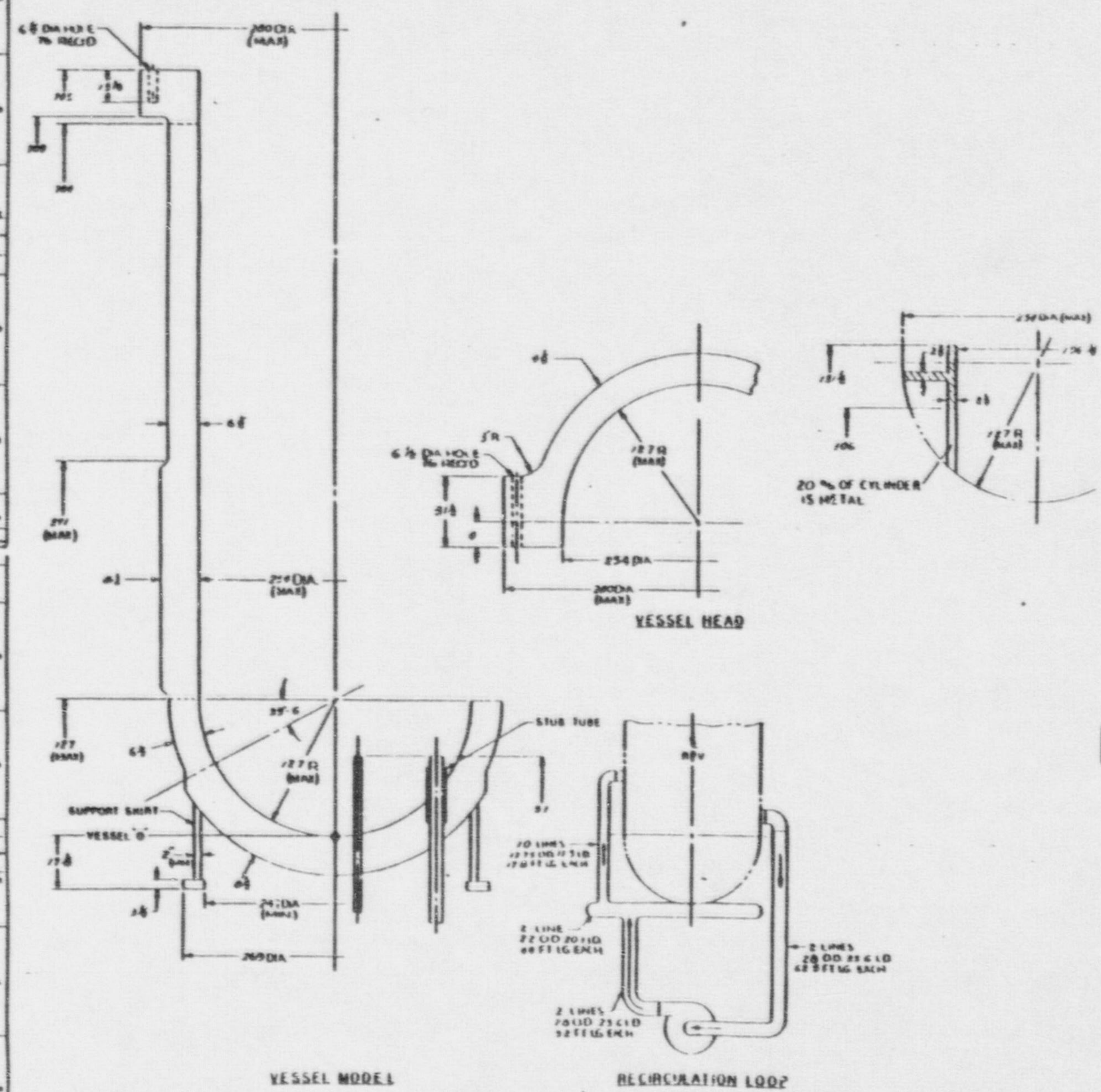
*** G_1 - CONTROL RODS FULL OUT; G_2 - CONTROL RODS FULL IN

**** CONTROL RODS FULL IN

(a) EXPECTED WATER LEVEL AT 100% STEAM FLOW IN PERIPHERAL SEPARATOR REGION (SEE NOTE 2)

(b) NOMINAL WATER LEVEL INSTRUMENT SET POINT (SEE NOTE 2)

NOTES



VESSEL & PIPE MATERIAL DATA

	SOLID VOLUMES (CU FT)	WEIGHTS (LBS)
BOTTOM HEAD	[illegible]	[illegible]
VESSEL CYLINDER	[illegible]	[illegible]
VESSEL FLANGE	220.5	[illegible]
SUPPORT SKIRT	[illegible]	[illegible]
INTERNALS SUPPORT	[illegible]	[illegible]
NOZZLES	[illegible]	[illegible]
CRD HOUSINGS	185.0	[illegible]
STUB TUBES	[illegible]	[illegible]
INCORE HOUSINGS	[illegible]	[illegible]
VESSEL W/OUT HEAD	[illegible]	[illegible]
VESSEL HEAD	360.4	[illegible]
TOTAL VESSEL	[illegible]	[illegible]
RECIRC LOOP PIPE	[illegible]	[illegible]
STEAM LINE PIPE	[illegible]	[illegible]
FEEDWATER PIPE	[illegible]	[illegible]

FEEDWATER LINE PIPE VOL & WTS FOR PORTION INSIDE DRYWELL
STEAM LINE PIPE VOL & WTS FOR PORTION INSIDE DRYWELL

PRIMARY PIPING SYSTEM FLUID VOLUME & WEIGHT

	FLUID VOLUME (FT³)	WEIGHTS (LBS)		
		RATED	HOT STAND	COLD
RECIRC PIPING	[illegible]	[illegible]	[illegible]	[illegible]
STEAM PIPING	[illegible]	[illegible]	—	—
FEEDWATER PIPING	[illegible]	[illegible]	—	—

FEEDWATER FLUID VOL & WTS TO LAST FEEDWATER HEATER
STEAMLINE FLUID VOL & WTS TO TURBINE STOP VALVE

VESSEL NOZZLES

FUNCTION	NUMBER	SIZE IN
INSTRUMENTATION	[illegible]	[illegible]
VENT	[illegible]	[illegible]
STEAM OUTLET	[illegible]	[illegible]
FEEDWATER	6	[illegible]
CORE SPRAY	[illegible]	10
HEAD RETURN	[illegible]	[illegible]
INSTRUMENTATION	[illegible]	6
RECIRC OUTLET	2	[illegible]
RECIRC INLET	10	[illegible]
JET PUMP INST	[illegible]	[illegible]
DRAIN	[illegible]	[illegible]
FLANGE LEAK DET	2	[illegible]
[illegible]	[illegible]	[illegible]

VOLUMES & WEIGHTS
VESSEL INTERNAL COMPONENTS

STEAM DRYER ASM

ITEM	COMPONENT	VOL. (FT³)	WT. (KIPS)
1	PANELS	78.00	39.00
2	HOUSING	51.80	25.90
3	SUPPORT RING	14.60	7.30
4	WATER SEAL SKIRT	11.60	5.80
5	TOTAL	156.00	78.00

CORE SHROUD ASM

ITEM	COMPONENT	VOL (FT³)	WT (KIPS)
15	SHROUD	216.22	108.11
16	UPPER CORE GRID	41.39	20.69
17	CORE SPRAY SPARGERS & NOZZLES	8.00	2.00
18	FUEL SUPPORT CASTINGS	11.46	5.73
19	CORE PLATE ASM	62.08	20.50
20	TOTAL	339.14	157.37

STEAM SEPARATOR ASM

ITEM	COMPONENT	VOL (FT³)	WT (KIPS)
6	GUIDE RODS	3.54	1.77
7	HOLD DOWN BOLTS	39.96	19.98
8	LIFT RODS	3.60	1.80
9	BOLT RINGS	9.81	4.90
10	STAND PIPES	63.18	31.59
11	SEPARATORS	158.60	78.30
12	SHROUD HD DOME	35.94	17.97
13	SHROUD HD FLGE	11.47	5.74
14	TOTAL	326.51	162.26

CORE REGION

ITEM	COMPONENT	VOL (FT³)	WT (KIPS)
21	FUEL ASSEMBLIES	920.05	504.84
22	CONTROL RODS (FULL IN)	87.10	43.55
24	IN-CORE ASSEMBLIES	3.60	1.80
25	TOTAL	1010.75	562.41

LOWER PLENUM

ITEM	COMPONENT	VOL (FT³)	WT (KIPS)
26	INLET WATER DIFFUSER	0	0
27	GUIDE TUBES	103.60	31.80
28	CONT ROD DRIVES & THERM SLEEVES	206.46	103.23
29	JET PUMP ASSEMBLIES	44.29	22.14
30	TOTAL	354.35	177.17

ITEM	REACTOR INTERNALS	VOL (FT³)	WT (KIPS)
31	TOTALS	2208.78	1160.18

TOTAL REACTOR INTERNALS VOLUME NOT EQUAL TO TOTAL OF SOLID VOLUMES ON SHEET 1, DUE TO SMALL ACCUMULATIVE ERRORS

Reactor Volume Tables

	HEIGHT ABOVE VESSEL ZERO (INCHES)	CORE FREE VOLUME (CU.FT)	SHROUD FREE VOLUME (CU.FT)	TOTAL FREE VOLUME (CU.FT)
1	0.000	0.000	0.000	0.000
2	6.000	7.950	0.000	7.950
3	11.000	26.360	0.000	26.360
4	16.000	55.020	0.000	55.020
5	21.000	93.480	0.000	93.480
6	26.000	141.300	0.000	141.300
7	32.000	210.420	0.000	210.420
8	37.000	277.290	0.000	277.290
9	42.000	352.100	0.000	352.100
10	48.000	451.740	0.000	451.740
11	51.000	505.380	0.000	505.380
12	58.000	633.480	0.000	633.480
13	64.000	753.110	0.000	753.110
14	69.000	859.140	0.000	859.140
15	74.000	970.430	0.000	970.430
16	79.000	1086.550	0.000	1086.550
17	85.000	1231.610	0.000	1231.610
18	90.000	1356.720	0.000	1356.720
19	95.000	1485.200	0.000	1485.200
20	101.000	1643.170	0.000	1643.170
21	106.000	1777.450	0.000	1777.450
22	111.000	1913.640	0.000	1913.640
23	116.000	2051.280	0.000	2051.280
24	121.500	2203.820	0.000	2203.820
25	127.000	2296.690	58.720	2355.410
26	191.125	3334.980	743.370	4078.350
27	216.312	3712.770	982.480	4695.250
28	317.187	4924.898	1946.080	6870.980
29	360.125	5440.199	2367.920	7808.121
30	379.125	5672.609	2507.090	8179.699
31	411.625	6347.520	2745.150	9092.672
32	414.000	6393.289	2762.740	9156.031
33	417.000	6446.238	2790.400	9236.641
34	420.000	6493.691	2824.230	9317.922
35	423.000	6535.551	2864.320	9399.871
36	426.000	6571.719	2910.770	9482.488
37	429.000	6602.129	2963.680	9565.809
38	431.000	6619.160	3002.590	9621.750
39	434.000	6639.762	3066.480	9706.238
40	437.000	6654.328	3173.920	9828.250
41	440.000	6662.871	3214.520	9877.391
42	442.625	6665.262	3287.950	9953.211
43	510.000	6933.539	4784.512	11718.051
44	533.500	7117.270	5193.379	12310.648
45	564.000	7470.629	5832.891	13303.520
46	577.870	7613.559	6113.609	13727.172
47	607.500	7789.031	6713.301	14502.328
48	635.000	0.000	0.000	15308.719
49	749.000	0.000	0.000	18651.578
50	876.000	0.000	0.000	21134.289

Reactor Volume Tabular Data
from BWRSAR.

1) Core Free Volume = Volume in lower plenum, GRGTs, jet pumps, inside core shroud, and steam separators.

2) Shroud Free Volume = Volume outside core shroud and steam separators.

3) Total Free Volume = Sum of 1) and 2) or steam dome volume above 607.5 inches.

All volumes are accumulative from vessel zero.

APPENDIX 4

SIMPLIFIED DEPRESSURIZATION CALCULATION

TABLE 1

Simplified Depressurization Calculation

$$M_S = [AM_{Li} + B\,M_{Gi} + CM_M + D(Q \quad + Q_S)]/E$$

$$M_{Gf} = [F\,M_{Li} + GM_{Gi} - d_{Gf}(M_M + M_S)]/D$$

$$M_{Lf} = M_{Li} + M_{Gi} + M_M - M_{Gf} - M_S$$

$$A = (h_{Li} - h_{Lf})(d_{Lf} - d_{Gf}) - (h_{Gf} - h_{Lf})(d_{Lf} - d_{Li}) \quad \frac{d_{Gf}}{d_{Li}}$$

$$B = (h_{Gi} - h_{Lf})(d_{Lf} - d_{Gf}d) - (h_{Gf} - h_{Lf})(d_{Lf} - d_{Gi}) \quad \frac{d_{Gf}}{d_{Gi}}$$

$$C = (h_M - h_{Lf})(d_{Lf} - d_{Gf}) + (h_{Gf} - h_{Lf})d_{Gf}$$

$$D = d_{Lf} - d_{Gf}$$

$$E = (h_A - h_{Lf})(d_{Lf} - d_{Gf}) - (h_{Gf} - h_{Lf})d_{Gf}$$

$$F = (d_{Lf} - d_{Li}) \quad \frac{d_{Gf}}{d_{Li}}$$

$$G = (d_{Lf} - d_{Gi}) \quad \frac{d_{Gf}}{d_{Gi}}$$

$$Q_S = [M_V C_V f_V + M_F C_F f_F + M_I C_I f_I](T_i - T_f)$$

TABLE 2

SYMBOL DEFINITIONS

d_G - density of water vapor, #/ft^3

d_L - density of liquid water, #/ft^3

f - subscript denoting final state

f_F - fraction of available sensible heat from fuel released

f_I - fraction of available sensible heat from reactor internals released

f_V - fraction of available sensible heat from reactor vessel released

h_A - average enthalpy of steam released from vessel, BTU/#

h_G - saturation enthalphy of water vapor, BTU/#

h_L - saturation enthalphy of liquid water, BTU/#

h_M - enthalpy of makeup water, BTU/#

i - subscript denoting initial conditions

M_G - mass of water vapor in vessel, #

M_L - mass of liquid in vessel, #

M_M - mass of makeup water added, #

M_S - mass of steam released from vessel, #

$M_F C_F$ - thermal capacitance of fuel, BTU/$^\circ$F

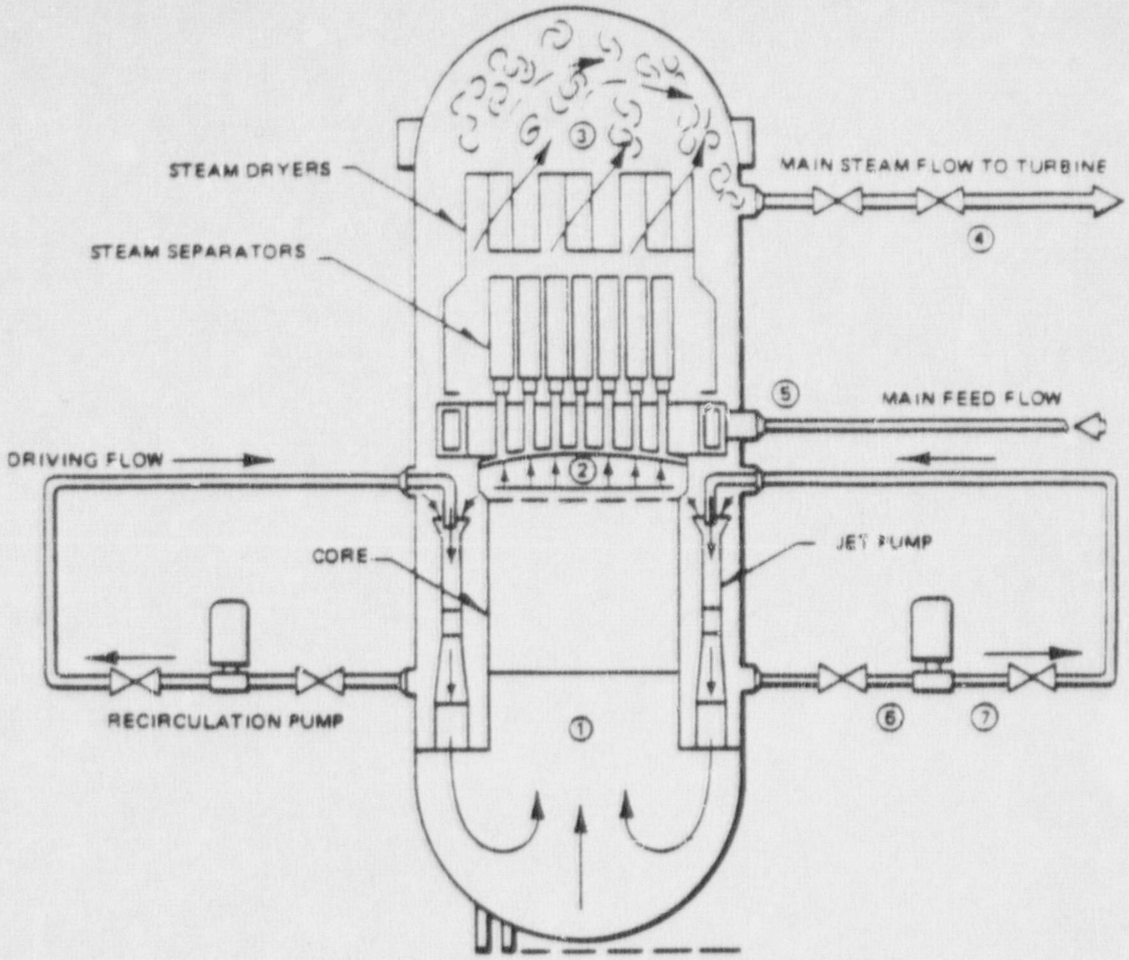
$M_I C_I$ - thermal capacitance of vessel internals, BTU/$^\circ$F

$M_V C_V$ - thermal capacitance of reactor vessel, BTU/$^\circ$F

Q - decay heat released in the time period (t_i, t_f), BTU

APPENDIX 5

USEFUL FSAR DATA



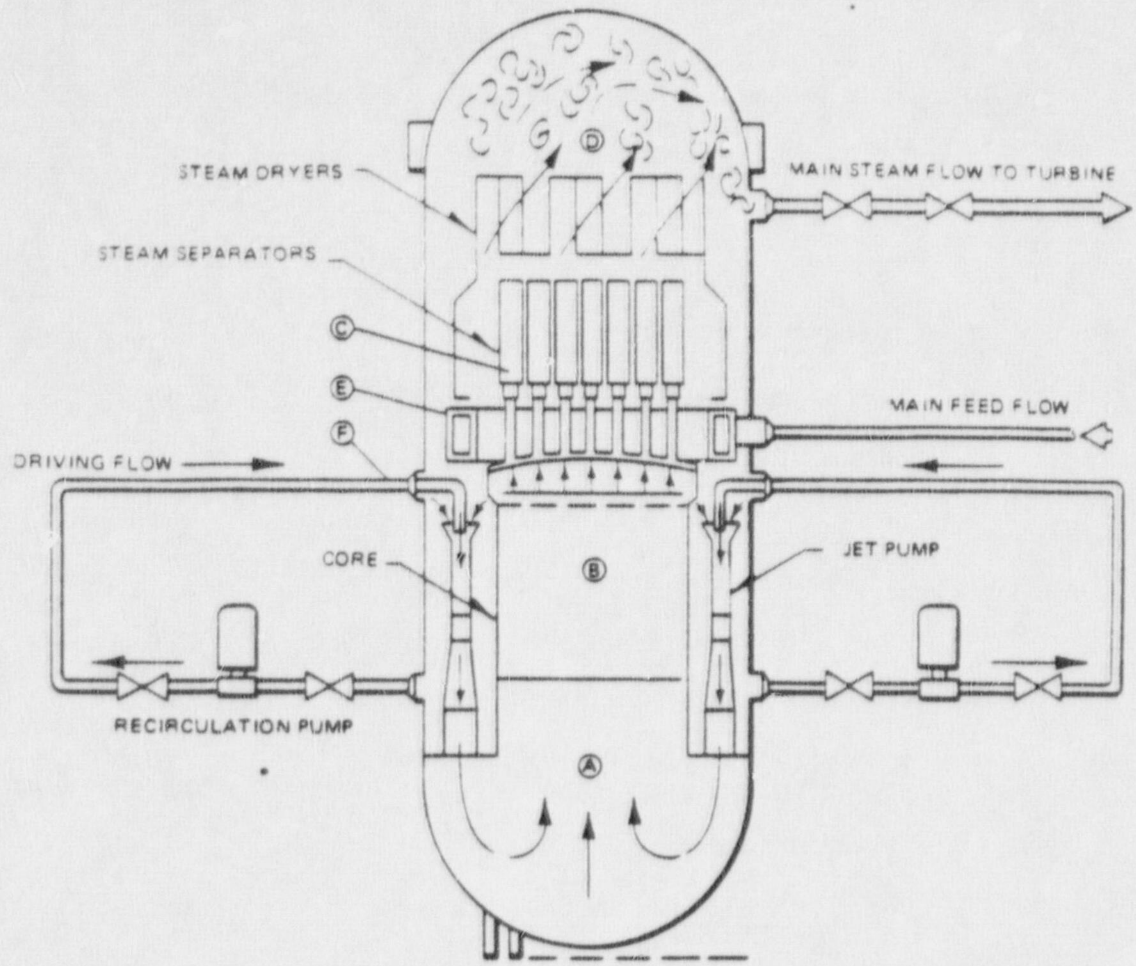
	PRESSURE (psia)	FLOW (lb/hr)	TEMPERATURE (°F)	ENTHALPY (Btu/lb)
1. CORE INLET	1053	100.0×10^6	528	521.8
2. CORE OUTLET	1030	100.0×10^6	548	634.2
3. SEPARATOR OUTLET (STEAM DOME)	1020	13.48×10^6	547	1191.5
4. STEAM LINE (2ND ISOLATION VALVE)	985	13.48×10^6	543	1191.5
5. FEEDWATER INLET (INCLUDES CLEANUP AND DEMINERALIZATION SYSTEM RETURN FLOW)	1045	13.57×10^6	383	357.7
6. RECIRC PUMP SUCTION	1034	34.2×10^6	528	521.6
7. RECIRC PUMP DISCHARGE	1268	34.2×10^6	529	522.6

Rev. 35, 07/84

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

RATED OPERATING CONDITIONS OF THE BOILING WATER REACTOR

FIGURE 5.1-1



	VOLUME OF FLUID (ft^3)
A. LOWER PLENUM	3887
B. CORE	2054
C. UPPER PLENUM AND SEPARATORS	1321
D. DOME (ABOVE NORMAL WATER LEVEL)	7934
E. DOWNCOMER REGION	5830
F RECIRC LOOPS AND JET PUMPS	1398

Rev. 35, 07/84

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

COOLANT VOLUMES OF THE BOILING WATER REACTOR

FIGURE 5.1-2

TABLE 6.2-1

CONTAINMENT DESIGN PARAMETERS

	Drywell	Suppression Chamber
A. DRYWELL AND SUPPRESSION CHAMBER:		
1. Internal Design Pressure, psig	53	53
2. External Design Pressure, psig	5	5
3. Drywell Deck Design Differential Pressure		
a. Download, psid	28	
b. Upload, psid	5.5	
4. Design Temperature, °F	340	220
5. Drywell (including vents) Net Free Volume, ft^3	239,600	-
6. Design Leak Ratio, %/day	0.5	0.5
7. Maximum Allowable Leak Rate, %/day	0.5	0.5
8. Suppression Chamber Free Volume, ft^3		159,130 (low water) 148,590 (high water)
9. Suppression Chamber Water Volume		
Minimum, ft^3		122,410
Maximum, ft^3		131,550
10. Pool Free Cross-sectional Area, ft^2		5,277
11. Pool Depth (normal), ft		23
12. Drywell Free Volume/Pressure Suppression Chamber Free Volume		1.51 to 1.61
13. Primary System Volume/Pressure Suppression Pool Volume		.15

TABLE 6.2-1 (Continued)

B. VENT SYSTEM:

1.	No. of Active Downcomers	82
2.	No. of Capped Downcomers	5
3.	Nominal Downcomer Diameter, ft.	2
4.	Total Downcomer Vent Area, ft^2	242
5.	Downcomer Submergence, ft - high water level	12
	- normal water level	11
	-low water level	10
6.	Downcomer Loss Factor	2.17

TABLE 6.2-3

ACCIDENT ASSUMPTIONS AND INITIAL CONDITIONS FOR LARGE LINE BREAKS

A. Effective Accident Break Area (Total) Recirculation Line Break, ft^2	4.158
B. Effective Accident Break Area, Main Steam Line Break, ft^2	See Figure 6.2-10
C. Components of Effective Break Area (Recirculation Line Break):	
1. Recirculation Line Area, ft^2	3.540
2. Cleanup Line Area, ft^2	.080
3. Jet Pump Area, ft^2	.538
D. Primary System Energy Distribution (1)	
1. Steam Energy, 10^6 Btu	22.1
2. Liquid Energy, 10^6 Btu	333.8
3. Sensible Energy, 10^6 Btu	
a. Reactor Vessel	108.2
b. Reactor Internals (less core)	48.0
c. Primary System Piping	30.5
d. Fuel (2)	6.5
E. Other Assumptions Used in Analysis	
1. Feedwater Valve Closure Time	See Table 6.2-9a
2. MSIV Closure Time (sec)	4 ± 1
3. Scram Time (sec)	1
4. Liquid Carryover, %	100

TABLE 6.2-3 (continued)

NOTES: (1) All energy values except fuel are based on a 32°F datum.

(2) Fuel energy is based on a datum of 285°F.

TABLE 6.2-4

INITIAL CONDITIONS EMPLOYED
IN CONTAINMENT RESPONSE ANALYSES

A. REACTOR COOLANT SYSTEM: (at design overpower of 105% rated steam flow)

1.	Reactor Power Level, MWT	3,434
2.	Average Coolant Pressure, psia	1,055
3.	Average Coolant Temperature, °F	551.14
4.	Mass of Reactor Coolant System Liquid, lbm	610,500
5.	Mass of Reactor Coolant System Steam, lbm	19,900
6.	Liquid Plus Steam Energy, Btu	355.9×10^6
7.	Volume of Water in Vessel, ft^3	11,978
8.	Volume of Steam in Vessel, ft^3	6,990
9.	Volume of Water in Recirculation Loops, ft^3	1,236
10.	Volume of Steam in Steam Lines, ft^3	1,390
11.	Volume of Water in Feedwater Line, ft^3	See Table 6.2-9a
12.	Volume of Water in Miscellaneous Lines, ft^3	0
13.	Total Reactor Coolant Volume, ft^3	21,594

TABLE 6.2-4 (continued)

CONTAINMENT

		Drywell	Suppression Chamber
1.	Pressure, psig	0.1 to 1.5	0.1 to 1.5
2.	Temperature, °F	120 to 150	90 to 150
3.	Outside Temperature, °F	40 to 55	100
4.	Relative Humidity, % Temperature, °F		90.2 max
5.	Water Volume, ft^3	N/A	122410 to 133540
6.	Downcomer Submergence, ft	N/A	10.0 to 12.0

APPENDIX 6

PLANT DAMAGE STATE DISPOSITIONING
WITH
EQUIPMENT RECOVERY

Plant Damage State Dispositioning with Equipment Recovery

Equipment may fail in response to or after an initiating event. Such failures may be due to:

1. An undetected failure in standby equipment,

2. loss of support systems,

3. failure of equipment during operation, or

4. failure caused by consequential plant conditions.

In the event of such failures, however, the operators may initiate recovery actions which will result in restoration of one or more items of failed equipment. Therefore, as the transient progresses in time, the unavailability of equipment must be considered to change with time. The models and information required to determine the availability of equipment, or probability thereof, involve the transient calculation used to track the progress of the event and various equipment recovery and failure models for the various items of equipment important to the event.

The equipment failed may be either front-line systems or support systems. In the case of front-line systems, the capability of these systems to perform the front-line functions, A_i, which represent the top events of the Front-line Function Event Trees are determined by means of appropriate Functional Fault Trees.

The support state for a given event tree specifies what support functions are unavailable. In general, it is true that more than one support function may have the identical effect on front-line function unavailability. In such cases the support functions are grouped together to form a support state. Thus, in general, we consider support state I to be made up of equipment failure combinations S_1, S_2, --- S_m where each S_i is a combination of equipment failures which will produce support state I.

The failure strings S_i each consist of one or more failed components. In the development of the influence of recovery actions presented here, recovery of any one of these failed components is considered to terminate the support state condition and to restore the front-line functions made unavailable by the restored support system.

In order to describe the process of event tree quantification and determination of the frequency contribution to each plant damage state let us first consider the basic elements of the process. First, we select an initiating event and a support state. The support state has the character described above. Second, we must develop a Front-line Function Event Tree (or its logical equivalent) for that initiating event and support state. This Front-line Function Event Tree consists of n top events (A_1, A_2, --- A_n) each of which represents a front-line function needed to assure avoidance or minimization of damage to the plant as a consequence of the initiating event and the associated support system failures corresponding to the support state. In the most general case the outcome of the initiator will be all possible combinations of successes and failures of the set of top events (A_1, --- A_n).

Many of these combinations can usually be eliminated, however, because the outcome of a given top event may have the result that some or all subsequent top events are immaterial to the outcome. The transient analysis of the sequence provides the information necessary to determine the endpoints which can be eliminated (or combined) in this manner.

Given that all of the end points of the event tree are now developed two tasks remain:

1) assignment of each endpoint to a plant damage class, and

2) calculation of the frequency of the end point.

The assignment of endpoints to a plant damage class is once again determined by the analysis of the transient using an accident code such as MAAP or BWRSAR. Certain combinations of top event successes and failures can be assigned to a specific plant damage state based on the transient analysis. The success criteria to avoid a given plant damage state (or a failure criterion which assures its occurrence) must be stated in terms of front-line function failure combinations. Examples of plant damage states which could be defined are:

1. core damage with loss of clad integrity but geometry intact,

2. metal-water reaction resulting in control rod relocation to the core support plate,

3. metal water reaction with channel box relocation to the core support plate,

4. core heat up with fuel relocation to the core support plate,

5. core support plate failure with debris relocation to the lower plenum,

6. lower plenum dryout with lower head failure and debris relocation to the drywell floor, and

7. debris heatup and spreading on the drywell floor with consequential over temperature failure of drywell component.

These definitions are very simple-minded and do not reflect that there are various degrees of material relocation (relocation proceeds locally at differing times) and do not reflect the complications resulting from generation of non-condensible gases. The example given does show the progressive nature of the plant damage progression, however, which is an important and general aspect of the accident progression.

A complication which exists in this process is the fact that in the most general case, more than one combination of the top event successes and failures may result in the same plant damage state. Further, the recovery time constraints for recovering the top event functions may differ among these success and failure combinations due to the nature of the transient trajectory for the different success and failure cases. These differences must be considered in the calculation of the event tree endpoint conditional probabilities. The frequency contribution of the endpoint to the plant damage state is then the product of this conditional probability and the frequency of the initiating event-support state combination.

Define a specific set of successes and failures of event tree top events C_i $(A_1, A_2, ---, A_n)$. If there are n top events in the tree, there are 2^n possible endpoints. As stated above many of these endpoints may be degenerate in the sense that successes or failures subsequent to a given set of successes and failures in the first part of the set can become irrelevant. Thus, in general, we have k possible success/failure combinations where $k<2^n$. Each of these combinations must be assigned to a plant damage state, and let us assume that q plant damage states with associated success criteria have been defined. These are $(D_1, D_2, --- D_q)$. For the first plant damage state 1, a subset of the C_k combinations will be assigned, say $r(r<k)$ of them. In general the time constraint for restoration of a top event in one of the subsets will be different for each member of the subset. Therefore, r times must be defined for $D_1 (T_{11}, T_{12}, ---, T_{1r})$ corresponding to the differing time limits for restoring the top event function. There is then an array of time constraints T_{ij} corresponding to the jth member of the subset of C_k defined for plant damage state D_i. The total number of such times will obviously be just equal to k, the number of success/failure combinations.

While this sounds very complex and formidable, in actual practice it is a much simpler matter than the above discussion would indicate. For example, in a particularly simple case there may only be a need for q times since all times for each plant damage state subset of the C_k are identical. Thus the recovery time constraint is simply associated with the plant damage state. In realistic application this simple example is much nearer the degree of complexity found than the theoretical maximum of the general case.

The time constraints on successful execution of the top events, A_i, and the recovery functions for the various systems which are required for availability of A_i must be used for determining the probabilities of the various event tree end points. The probability of a given endpoint combination is denoted by $P(C_i)$ and is given by:

$$P(C_i) = \sum_{j=1}^{m} P(C_i/S_j)P(S_j/I)$$

where

P(A/B) = the conditional probability of A given the occurrence of B

C_i = the ith top event success/failure combination

S_j = the jth support system component failure combination causing support state I.

m = the maximum value of j.

The term $P(C_i/S_j)$ is given by:

$$P(C_i/S_j) = P(C_i/R_{jt})P(R_{jt}/S_j) + P(C_i/R_{jt})P(R_{jt}/S_j)$$

where

R_{jt} = success in recovery of S_j by time T_t (value 0 or 1).

$\bar{R}_{jt}$ = the complement of R_{jt}, failure to recover S_j by time T_t.

$P(C_i/R_{jt})$ = the conditional probability of C_i given that S_j is recovered at T_t.

$P(C_i/\bar{R}_{jt})$ = the conditional probability of C_i given that S_j is not recovered at time T_t.

We assume that if S_j is recovered at time T_t, it will be available at all times subsequent to T_t. On the other hand, if it is not recovered by time T_t, there is a non-zero probability that it will be recovered prior to the next time in the sequence of time constraints being considered, T_{t+1}. This fact must be accounted for in developing the probabilities for the various C_i of the set C. Induction may be used to develop a general expression for this process.

In the set of success/failure combinations, C_i, consider that a top event function, A_1, is to be recovered before time T_1, where T_1 is a time later than the initial time T_0. We obtain:

$$P(C_i/S_1) = P(C_i/R_{11})P(R_{11}/S_{10}) + P(C_i/\bar{R}_{11})P(\bar{R}_{11}/S_{10})$$

where

R_{11} = the recovery state for support system string, S_1 at time T_1.

S_{10} = the support system string status at the initial time.

A_1 = a member of the set C_i

If two recovery times are to be considered, A_1 must have been recovered at T_1 and

$$\begin{aligned} P(C_i/S_1) = {} & P(C_i/R_{11})P(R_{11}/S_{10}) \\ & + P(A_1/\bar{R}_{11})P(\bar{R}_{11}/S10)[P(C_i/R_{12})P(R_{12}/S_{11}) \\ & + P(C_i/\bar{R}_{12})P(\bar{R}_{12}/S_{11})] \end{aligned}$$

where

R_{12} = the recovery state for support system string S_1 at time T_2.

S_{11} = the support system string status at time T_1.

The terms in the brackets repeat, and, therefore, the expression for s recovery times becomes:

$$P(C_i/S_1) = P(C_i/R_{11})P(R_{11}/S_{10})$$
$$+ P(A_1/\overline{R}_{11})P(\overline{R}_{11}/S_{10})\{P(C_{i-1}/R_{12})P(R_{12}/S_{11})$$
$$+ \ldots + P(C_{i-s-1}/\overline{R}_{1,s-1})[P(C_{i-s}/R_{1s})P(R_{1s}/s_{1,s-1})$$
$$+ P(C_{i-s}/\overline{R}_{1k})P(\overline{R}_{1k}/S_{1s})]\}$$

where

C_{i-1} = the combination C_i with A_1 removed.

C_{i-s} = the combination C_i with A_1, A_2, --- A_s removed.

Therefore, in order to determine the probability of any success/failure combination C_i in general the above process can be executed to derive it.

These relationships have been programmed and incorporated into a computer program which generates support states and quantifies the corresponding event trees using input data which describes the initiating events, success and failure combinations, plant damage states, equipment dependencies and unavailabilities, time constraints, and other information needed to evaluate the final outcome of a given initiating event. This program is currently in the process of being debugged. When this process is complete, input and output routines will be added to create a comprehensive risk analysis calculation package that will generate the type of information described in this document.

APPENDIX 7

AC POWER RECOVERY TABLE

Recovery of Off-Site AC Power

The tables included in this Appendix represent the occurrence and recovery of loss of off-site AC power. The data presented represent the frequency of loss of off-site power for a time equal to or greater than the tabular values of time presented.

For example, the occurrence of loss of off-site power at Susquehanna is calculated to have a frequency of 0.0566 yr^{-1} for losses of very short duration (a few minutes or less), while the frequency a loss which will last one hour or longer is calculated to be 0.01164 yr^{-1}. These values are calculated using the methods and data from NUREG-1032 and have been accepted as valid for Susquehanna by the NRC.

The symbols I(t), GR(t), SR(t), and SS(t) are defined in the NUREG as losses due to:

I(t) = plant related problems,

GR(t) = grid related problems,

SR(t) = severe weather,

SS(t) = extremely severe weather.

For the case of severe weather, different frequencies are derived depending on the routing of the off-site power lines. The lines at Susquehanna are considered to be not obtuse.

The sum of I, GR, SR, and SS then yields to total frequency of loss of off-site power from all causes.

This data must be used in conjunction with a model for loss of all site based AC power and its recovery in order to derive its recovery in order to derive the appropriate curve for recovery from Station Blackout.

******************************** SSES DATA GENERATED CURVES **********************************

T (HRS)	I(T)	GR(T)	SR(T) OBTUSE LINES	SS(T)	SUM(T) OBTUSE LINES	SR(T) NO OBTUSE LINES	SUM(T) NO OBTUSE LINES	T (HRS)
0.00	0.4000E-01	0.8000E-02	0.6320E-02	0.5333E-03	0.5485E-01	0.8070E-02	0.5660E-01	0.00
0.10	0.2164E-01	0.6966E-02	0.6196E-02	0.5311E-03	0.3533E-01	0.7912E-02	0.3705E-01	0.10
0.20	0.1337E-01	0.6106E-02	0.6075E-02	0.5289E-03	0.2608E-01	0.7757E-02	0.2776E-01	0.20
0.30	0.9226E-02	0.5388E-02	0.5956E-02	0.5267E-03	0.2110E-01	0.7605E-02	0.2275E-01	0.30
0.40	0.6855E-02	0.4788E-02	0.5839E-02	0.5245E-03	0.1801E-01	0.7456E-02	0.1962E-01	0.40
0.50	0.5315E-02	0.4285E-02	0.5724E-02	0.5223E-03	0.1585E-01	0.7309E-02	0.1743E-01	0.50
0.60	0.4215E-02	0.3862E-02	0.5612E-02	0.5201E-03	0.1421E-01	0.7166E-02	0.1576E-01	0.60
0.70	0.3381E-02	0.3505E-02	0.5502E-02	0.5180E-03	0.1291E-01	0.7026E-02	0.1443E-01	0.70
0.80	0.2726E-02	0.3202E-02	0.5394E-02	0.5158E-03	0.1184E-01	0.6888E-02	0.1333E-01	0.80
0.90	0.2205E-02	0.2944E-02	0.5288E-02	0.5137E-03	0.1095E-01	0.6753E-02	0.1241E-01	0.90
1.00	0.1785E-02	0.2723E-02	0.5185E-02	0.5115E-03	0.1020E-01	0.6620E-02	0.1164E-01	1.00
1.10	0.1446E-02	0.2533E-02	0.5083E-02	0.5094E-03	0.9571E-02	0.6491E-02	0.1098E-01	1.10
1.20	0.1172E-02	0.2368E-02	0.4983E-02	0.5073E-03	0.9031E-02	0.6363E-02	0.1041E-01	1.20
1.30	0.9497E-03	0.2225E-02	0.4886E-02	0.5052E-03	0.8565E-02	0.6239E-02	0.9918E-02	1.30
1.40	0.7698E-03	0.2099E-02	0.4790E-02	0.5031E-03	0.8162E-02	0.6116E-02	0.9488E-02	1.40
1.50	0.6239E-03	0.1988E-02	0.4696E-02	0.5010E-03	0.7809E-02	0.5996E-02	0.9109E-02	1.50
1.60	0.5058E-03	0.1889E-02	0.4604E-02	0.4989E-03	0.7498E-02	0.5879E-02	0.8773E-02	1.60
1.70	0.4100E-03	0.1801E-02	0.4514E-02	0.4968E-03	0.7221E-02	0.5764E-02	0.8471E-02	1.70
1.80	0.3323E-03	0.1721E-02	0.4425E-02	0.4948E-03	0.6973E-02	0.5651E-02	0.8199E-02	1.80
1.90	0.2694E-03	0.1649E-02	0.4338E-02	0.4927E-03	0.6749E-02	0.5540E-02	0.7950E-02	1.90
2.00	0.2183E-03	0.1582E-02	0.4253E-02	0.4907E-03	0.6545E-02	0.5431E-02	0.7723E-02	2.00
2.10	0.1770E-03	0.1521E-02	0.4170E-02	0.4886E-03	0.6357E-02	0.5325E-02	0.7512E-02	2.10
2.20	0.1435E-03	0.1465E-02	0.4088E-02	0.4866E-03	0.6183E-02	0.5220E-02	0.7315E-02	2.20
2.30	0.1163E-03	0.1413E-02	0.4008E-02	0.4846E-03	0.6022E-02	0.5118E-02	0.7131E-02	2.30
2.40	0.9426E-04	0.1364E-02	0.3930E-02	0.4825E-03	0.5870E-02	0.5018E-02	0.6958E-02	2.40
2.50	0.7640E-04	0.1318E-02	0.3852E-02	0.4805E-03	0.5727E-02	0.4919E-02	0.6794E-02	2.50
2.60	0.6193E-04	0.1274E-02	0.3777E-02	0.4785E-03	0.5591E-02	0.4823E-02	0.6637E-02	2.60
2.70	0.5020E-04	0.1233E-02	0.3703E-02	0.4766E-03	0.5462E-02	0.4728E-02	0.6488E-02	2.70
2.80	0.4069E-04	0.1194E-02	0.3630E-02	0.4746E-03	0.5339E-02	0.4636E-02	0.6344E-02	2.80
2.90	0.3298E-04	0.1156E-02	0.3559E-02	0.4726E-03	0.5221E-02	0.4545E-02	0.6207E-02	2.90
3.00	0.2674E-04	0.1121E-02	0.3489E-02	0.4706E-03	0.5107E-02	0.4456E-02	0.6074E-02	3.00
3.10	0.2167E-04	0.1086E-02	0.3421E-02	0.4687E-03	0.4998E-02	0.4368E-02	0.5945E-02	3.10
3.20	0.1757E-04	0.1054E-02	0.3354E-02	0.4667E-03	0.4892E-02	0.4283E-02	0.5820E-02	3.20
3.30	0.1424E-04	0.1022E-02	0.3288E-02	0.4648E-03	0.4789E-02	0.4199E-02	0.5700E-02	3.30
3.40	0.1154E-04	0.9915E-03	0.3224E-02	0.4629E-03	0.4690E-02	0.4116E-02	0.5582E-02	3.40
3.50	0.9356E-05	0.9622E-03	0.3160E-02	0.4609E-03	0.4593E-02	0.4036E-02	0.5468E-02	3.50
3.60	0.7584E-05	0.9339E-03	0.3099E-02	0.4590E-03	0.4499E-02	0.3956E-02	0.5357E-02	3.60
3.70	0.6147E-05	0.9065E-03	0.3038E-02	0.4571E-03	0.4408E-02	0.3879E-02	0.5249E-02	3.70
3.80	0.4983E-05	0.8800E-03	0.2978E-02	0.4552E-03	0.4318E-02	0.3803E-02	0.5143E-02	3.80
3.90	0.4039E-05	0.8544E-03	0.2920E-02	0.4533E-03	0.4232E-02	0.3728E-02	0.5040E-02	3.90
4.00	0.3274E-05	0.8296E-03	0.2863E-02	0.4514E-03	0.4147E-02	0.3655E-02	0.4940E-02	4.00
4.10	0.2654E-05	0.8056E-03	0.2806E-02	0.4496E-03	0.4064E-02	0.3584E-02	0.4841E-02	4.10
4.20	0.2151E-05	0.7823E-03	0.2751E-02	0.4477E-03	0.3984E-02	0.3513E-02	0.4745E-02	4.20
4.30	0.1744E-05	0.7598E-03	0.2697E-02	0.4458E-03	0.3905E-02	0.3444E-02	0.4652E-02	4.30
4.40	0.1413E-05	0.7379E-03	0.2645E-02	0.4440E-03	0.3828E-02	0.3377E-02	0.4560E-02	4.40
4.50	0.1146E-05	0.7167E-03	0.2593E-02	0.4421E-03	0.3753E-02	0.3311E-02	0.4471E-02	4.50
4.60	0.9287E-06	0.6961E-03	0.2542E-02	0.4403E-03	0.3679E-02	0.3246E-02	0.4383E-02	4.60
4.70	0.7528E-06	0.6761E-03	0.2492E-02	0.4385E-03	0.3607E-02	0.3182E-02	0.4297E-02	4.70
4.80	0.6102E-06	0.6567E-03	0.2443E-02	0.4366E-03	0.3537E-02	0.3120E-02	0.4214E-02	4.80
4.90	0.4946E-06	0.6379E-03	0.2395E-02	0.4348E-03	0.3468E-02	0.3059E-02	0.4132E-02	4.90
5.00	0.4009E-06	0.6195E-03	0.2348E-02	0.4330E-03	0.3401E-02	0.2999E-02	0.4052E-02	5.00
5.10	0.3250E-06	0.6018E-03	0.2302E-02	0.4312E-03	0.3336E-02	0.2940E-02	0.3973E-02	5.10
5.20	0.2634E-06	0.5846E-03	0.2257E-02	0.4294E-03	0.3271E-02	0.2882E-02	0.3896E-02	5.20
5.30	0.2135E-06	0.5678E-03	0.2213E-02	0.4276E-03	0.3209E-02	0.2826E-02	0.3821E-02	5.30
5.40	0.1731E-06	0.5516E-03	0.2170E-02	0.4258E-03	0.3147E-02	0.2770E-02	0.3748E-02	5.40
5.50	0.1403E-06	0.5358E-03	0.2127E-02	0.4241E-03	0.3087E-02	0.2716E-02	0.3676E-02	5.50
5.60	0.1137E-06	0.5205E-03	0.2085E-02	0.4223E-03	0.3028E-02	0.2663E-02	0.3606E-02	5.60
5.70	0.9218E-07	0.5056E-03	0.2044E-02	0.4206E-03	0.2971E-02	0.2611E-02	0.3537E-02	5.70
5.80	0.7472E-07	0.4911E-03	0.2004E-02	0.4188E-03	0.2914E-02	0.2559E-02	0.3469E-02	5.80
5.90	0.6057E-07	0.4771E-03	0.1965E-02	0.4171E-03	0.2859E-02	0.2509E-02	0.3403E-02	5.90
6.00	0.4910E-07	0.4634E-03	0.1927E-02	0.4153E-03	0.2805E-02	0.2460E-02	0.3339E-02	6.00

************************************ SSES DATA GENERATED CURVES **************************************

T (HRS)	I(T)	GR(T)	SR(T) OBTUSE LINES	SS(T)	SUM(T) OBTUSE LINES	SR(T) NO OBTUSE LINES	SUM(T) NO OBTUSE LINES	T (HRS)
6.10	0.3980E-07	0.4502E-03	0.1889E-02	0.4136E-03	0.2753E-02	0.2412E-02	0.3276E-02	6.10
6.20	0.3226E-07	0.4373E-03	0.1852E-02	0.4119E-03	0.2701E-02	0.2364E-02	0.3214E-02	6.20
6.30	0.2615E-07	0.4248E-03	0.1815E-02	0.4102E-03	0.2650E-02	0.2318E-02	0.3153E-02	6.30
6.40	0.2120E-07	0.4126E-03	0.1780E-02	0.4085E-03	0.2601E-02	0.2273E-02	0.3094E-02	6.40
6.50	0.1718E-07	0.4008E-03	0.1745E-02	0.4068E-03	0.2553E-02	0.2228E-02	0.3036E-02	6.50
6.60	0.1393E-07	0.3894E-03	0.1711E-02	0.4051E-03	0.2505E-02	0.2184E-02	0.2979E-02	6.60
6.70	0.1129E-07	0.3783E-03	0.1677E-02	0.4034E-03	0.2459E-02	0.2142E-02	0.2923E-02	6.70
6.80	0.9150E-08	0.3674E-03	0.1644E-02	0.4017E-03	0.2413E-02	0.2100E-02	0.2869E-02	6.80
6.90	0.7417E-08	0.3569E-03	0.1612E-02	0.4000E-03	0.2369E-02	0.2058E-02	0.2815E-02	6.90
7.00	0.6012E-08	0.3467E-03	0.1580E-02	0.3984E-03	0.2326E-02	0.2018E-02	0.2763E-02	7.00
7.10	0.4873E-08	0.3368E-03	0.1549E-02	0.3967E-03	0.2283E-02	0.1979E-02	0.2712E-02	7.10
7.20	0.3950E-08	0.3272E-03	0.1519E-02	0.3951E-03	0.2241E-02	0.1940E-02	0.2662E-02	7.20
7.30	0.3202E-08	0.3178E-03	0.1489E-02	0.3934E-03	0.2201E-02	0.1902E-02	0.2613E-02	7.30
7.40	0.2596E-08	0.3088E-03	0.1460E-02	0.3918E-03	0.2161E-02	0.1864E-02	0.2565E-02	7.40
7.50	0.2104E-08	0.2999E-03	0.1431E-02	0.3902E-03	0.2122E-02	0.1828E-02	0.2518E-02	7.50
7.60	0.1705E-08	0.2914E-03	0.1403E-02	0.3885E-03	0.2083E-02	0.1792E-02	0.2472E-02	7.60
7.70	0.1382E-08	0.2830E-03	0.1376E-02	0.3869E-03	0.2046E-02	0.1757E-02	0.2427E-02	7.70
7.80	0.1121E-08	0.2749E-03	0.1349E-02	0.3853E-03	0.2009E-02	0.1722E-02	0.2383E-02	7.80
7.90	0.9083E-09	0.2671E-03	0.1322E-02	0.3837E-03	0.1973E-02	0.1689E-02	0.2339E-02	7.90
8.00	0.7362E-09	0.2594E-03	0.1297E-02	0.3821E-03	0.1938E-02	0.1656E-02	0.2297E-02	8.00
8.10	0.5968E-09	0.2520E-03	0.1271E-02	0.3805E-03	0.1904E-02	0.1623E-02	0.2256E-02	8.10
8.20	0.4837E-09	0.2448E-03	0.1246E-02	0.3790E-03	0.1870E-02	0.1591E-02	0.2215E-02	8.20
8.30	0.3921E-09	0.2378E-03	0.1222E-02	0.3774E-03	0.1837E-02	0.1560E-02	0.2175E-02	8.30
8.40	0.3178E-09	0.2310E-03	0.1198E-02	0.3758E-03	0.1805E-02	0.1530E-02	0.2136E-02	8.40
8.50	0.2576E-09	0.2244E-03	0.1174E-02	0.3742E-03	0.1773E-02	0.1500E-02	0.2098E-02	8.50
8.60	0.2088E-09	0.2180E-03	0.1151E-02	0.3727E-03	0.1742E-02	0.1470E-02	0.2061E-02	8.60
8.70	0.1693E-09	0.2118E-03	0.1129E-02	0.3711E-03	0.1712E-02	0.1441E-02	0.2024E-02	8.70
8.80	0.1372E-09	0.2057E-03	0.1107E-02	0.3696E-03	0.1682E-02	0.1413E-02	0.1988E-02	8.80
8.90	0.1112E-09	0.1998E-03	0.1085E-02	0.3681E-03	0.1653E-02	0.1385E-02	0.1953E-02	8.90
9.00	0.9016E-10	0.1941E-03	0.1064E-02	0.3665E-03	0.1624E-02	0.1358E-02	0.1919E-02	9.00
9.10	0.7308E-10	0.1886E-03	0.1043E-02	0.3650E-03	0.1596E-02	0.1332E-02	0.1885E-02	9.10
9.20	0.5924E-10	0.1832E-03	0.1022E-02	0.3635E-03	0.1569E-02	0.1305E-02	0.1852E-02	9.20
9.30	0.4802E-10	0.1780E-03	0.1002E-02	0.3620E-03	0.1542E-02	0.1280E-02	0.1820E-02	9.30
9.40	0.3892E-10	0.1729E-03	0.9827E-03	0.3605E-03	0.1516E-02	0.1255E-02	0.1788E-02	9.40
9.50	0.3155E-10	0.1679E-03	0.9634E-03	0.3590E-03	0.1490E-02	0.1230E-02	0.1757E-02	9.50
9.60	0.2557E-10	0.1631E-03	0.9445E-03	0.3575E-03	0.1465E-02	0.1206E-02	0.1727E-02	9.60
9.70	0.2073E-10	0.1585E-03	0.9260E-03	0.3560E-03	0.1440E-02	0.1182E-02	0.1697E-02	9.70
9.80	0.1680E-10	0.1539E-03	0.9078E-03	0.3545E-03	0.1416E-02	0.1159E-02	0.1668E-02	9.80
9.90	0.1362E-10	0.1495E-03	0.8900E-03	0.3530E-03	0.1393E-02	0.1136E-02	0.1639E-02	9.90
10.00	0.1104E-10	0.1453E-03	0.8726E-03	0.3516E-03	0.1369E-02	0.1114E-02	0.1611E-02	10.00
10.10	0.8949E-11	0.1411E-03	0.8555E-03	0.3501E-03	0.1347E-02	0.1092E-02	0.1584E-02	10.10
10.20	0.7254E-11	0.1371E-03	0.8387E-03	0.3487E-03	0.1324E-02	0.1071E-02	0.1557E-02	10.20
10.30	0.5880E-11	0.1332E-03	0.8223E-03	0.3472E-03	0.1303E-02	0.1050E-02	0.1530E-02	10.30
10.40	0.4766E-11	0.1294E-03	0.8062E-03	0.3458E-03	0.1281E-02	0.1029E-02	0.1504E-02	10.40
10.50	0.3863E-11	0.1257E-03	0.7903E-03	0.3443E-03	0.1260E-02	0.1009E-02	0.1479E-02	10.50
10.60	0.3132E-11	0.1221E-03	0.7749E-03	0.3429E-03	0.1240E-02	0.9894E-03	0.1454E-02	10.60
10.70	0.2538E-11	0.1186E-03	0.7597E-03	0.3415E-03	0.1220E-02	0.9700E-03	0.1430E-02	10.70
10.80	0.2058E-11	0.1152E-03	0.7448E-03	0.3400E-03	0.1200E-02	0.9510E-03	0.1406E-02	10.80
10.90	0.1668E-11	0.1119E-03	0.7302E-03	0.3386E-03	0.1181E-02	0.9323E-03	0.1383E-02	10.90
11.00	0.1352E-11	0.1087E-03	0.7159E-03	0.3372E-03	0.1162E-02	0.9141E-03	0.1360E-02	11.00
11.10	0.1096E-11	0.1056E-03	0.7018E-03	0.3358E-03	0.1143E-02	0.8962E-03	0.1338E-02	11.10
11.20	0.8883E-12	0.1026E-03	0.6881E-03	0.3344E-03	0.1125E-02	0.8786E-03	0.1316E-02	11.20
11.30	0.7200E-12	0.9964E-04	0.6746E-03	0.3330E-03	0.1107E-02	0.8614E-03	0.1294E-02	11.30
11.40	0.5836E-12	0.9679E-04	0.6613E-03	0.3317E-03	0.1090E-02	0.8445E-03	0.1273E-02	11.40
11.50	0.4731E-12	0.9402E-04	0.6484E-03	0.3303E-03	0.1073E-02	0.8279E-03	0.1252E-02	11.50
11.60	0.3835E-12	0.9133E-04	0.6357E-03	0.3289E-03	0.1056E-02	0.8117E-03	0.1232E-02	11.60
11.70	0.3108E-12	0.8872E-04	0.6232E-03	0.3275E-03	0.1039E-02	0.7958E-03	0.1212E-02	11.70
11.80	0.2520E-12	0.8619E-04	0.6110E-03	0.3262E-03	0.1023E-02	0.7802E-03	0.1193E-02	11.80
11.90	0.2042E-12	0.8372E-04	0.5990E-03	0.3248E-03	0.1008E-02	0.7649E-03	0.1173E-02	11.90
12.00	0.1656E-12	0.8133E-04	0.5873E-03	0.3235E-03	0.9921E-03	0.7499E-03	0.1155E-02	12.00

******************************** SSES DATA GENERATED CURVES ********************************

T (HRS)	I(T)	GR(T)	SR(T) OBTUSE LINES	SS(T)	SUM(T) OBTUSE LINES	SR(T) NO OBTUSE LINES	SUM(T) NO OBTUSE LINES	T (HRS)
12.10	0.1342E-12	0.7901E-04	0.5758E-03	0.3221E-03	0.9769E-03	0.7352E-03	0.1136E-02	12.10
12.20	0.1088E-12	0.7675E-04	0.5645E-03	0.3208E-03	0.9620E-03	0.7208E-03	0.1118E-02	12.20
12.30	0.8817E-13	0.7455E-04	0.5534E-03	0.3194E-03	0.9474E-03	0.7066E-03	0.1101E-02	12.30
12.40	0.7147E-13	0.7242E-04	0.5425E-03	0.3181E-03	0.9331E-03	0.6928E-03	0.1083E-02	12.40
12.50	0.5793E-13	0.7035E-04	0.5319E-03	0.3168E-03	0.9191E-03	0.6792E-03	0.1066E-02	12.50
12.60	0.4696E-13	0.6834E-04	0.5215E-03	0.3155E-03	0.9053E-03	0.6659E-03	0.1050E-02	12.60
12.70	0.3806E-13	0.6639E-04	0.5113E-03	0.3142E-03	0.8918E-03	0.6528E-03	0.1033E-02	12.70
12.80	0.3085E-13	0.6449E-04	0.5012E-03	0.3129E-03	0.8786E-03	0.6400E-03	0.1017E-02	12.80
12.90	0.2501E-13	0.6265E-04	0.4914E-03	0.3116E-03	0.8656E-03	0.6275E-03	0.1002E-02	12.90
13.00	0.2027E-13	0.6086E-04	0.4818E-03	0.3103E-03	0.8529E-03	0.6152E-03	0.9863E-03	13.00
13.10	0.1643E-13	0.5912E-04	0.4723E-03	0.3090E-03	0.8404E-03	0.6031E-03	0.9712E-03	13.10
13.20	0.1332E-13	0.5743E-04	0.4631E-03	0.3077E-03	0.8282E-03	0.5913E-03	0.9564E-03	13.20
13.30	0.1080E-13	0.5579E-04	0.4540E-03	0.3064E-03	0.8162E-03	0.5797E-03	0.9419E-03	13.30
13.40	0.8752E-14	0.5419E-04	0.4451E-03	0.3051E-03	0.8044E-03	0.5683E-03	0.9277E-03	13.40
13.50	0.7094E-14	0.5264E-04	0.4364E-03	0.3039E-03	0.7929E-03	0.5572E-03	0.9137E-03	13.50
13.60	0.5751E-14	0.5114E-04	0.4278E-03	0.3026E-03	0.7815E-03	0.5463E-03	0.9000E-03	13.60
13.70	0.4661E-14	0.4968E-04	0.4194E-03	0.3013E-03	0.7704E-03	0.5356E-03	0.8866E-03	13.70
13.80	0.3778E-14	0.4826E-04	0.4112E-03	0.3001E-03	0.7595E-03	0.5251E-03	0.8734E-03	13.80
13.90	0.3063E-14	0.4688E-04	0.4031E-03	0.2988E-03	0.7489E-03	0.5148E-03	0.8605E-03	13.90
14.00	0.2483E-14	0.4554E-04	0.3952E-03	0.2976E-03	0.7384E-03	0.5047E-03	0.8478E-03	14.00
14.10	0.2012E-14	0.4424E-04	0.3875E-03	0.2964E-03	0.7281E-03	0.4948E-03	0.8354E-03	14.10
14.20	0.1631E-14	0.4297E-04	0.3799E-03	0.2951E-03	0.7180E-03	0.4851E-03	0.8232E-03	14.20
14.30	0.1322E-14	0.4174E-04	0.3724E-03	0.2939E-03	0.7081E-03	0.4756E-03	0.8112E-03	14.30
14.40	0.1072E-14	0.4055E-04	0.3651E-03	0.2927E-03	0.6984E-03	0.4662E-03	0.7995E-03	14.40
14.50	0.8687E-15	0.3939E-04	0.3580E-03	0.2915E-03	0.6888E-03	0.4571E-03	0.7880E-03	14.50
14.60	0.7042E-15	0.3826E-04	0.3510E-03	0.2903E-03	0.6795E-03	0.4481E-03	0.7767E-03	14.60
14.70	0.5708E-15	0.3717E-04	0.3441E-03	0.2890E-03	0.6703E-03	0.4394E-03	0.7656E-03	14.70
14.80	0.4627E-15	0.3611E-04	0.3373E-03	0.2878E-03	0.6613E-03	0.4307E-03	0.7547E-03	14.80
14.90	0.3750E-15	0.3508E-04	0.3307E-03	0.2866E-03	0.6524E-03	0.4223E-03	0.7440E-03	14.90
15.00	0.3040E-15	0.3407E-04	0.3242E-03	0.2855E-03	0.6438E-03	0.4140E-03	0.7335E-03	15.00
15.10	0.2464E-15	0.3310E-04	0.3179E-03	0.2843E-03	0.6352E-03	0.4059E-03	0.7233E-03	15.10
15.20	0.1997E-15	0.3215E-04	0.3116E-03	0.2831E-03	0.6269E-03	0.3979E-03	0.7132E-03	15.20
15.30	0.1619E-15	0.3123E-04	0.3055E-03	0.2819E-03	0.6187E-03	0.3901E-03	0.7033E-03	15.30
15.40	0.1312E-15	0.3034E-04	0.2995E-03	0.2807E-03	0.6106E-03	0.3825E-03	0.6936E-03	15.40
15.50	0.1064E-15	0.2947E-04	0.2937E-03	0.2796E-03	0.6027E-03	0.3750E-03	0.6840E-03	15.50
15.60	0.8623E-16	0.2863E-04	0.2879E-03	0.2784E-03	0.5950E-03	0.3676E-03	0.6747E-03	15.60
15.70	0.6990E-16	0.2781E-04	0.2823E-03	0.2772E-03	0.5873E-03	0.3604E-03	0.6655E-03	15.70
15.80	0.5666E-16	0.2702E-04	0.2767E-03	0.2761E-03	0.5799E-03	0.3534E-03	0.6565E-03	15.80
15.90	0.4593E-16	0.2625E-04	0.2713E-03	0.2749E-03	0.5725E-03	0.3464E-03	0.6476E-03	15.90
16.00	0.3723E-16	0.2550E-04	0.2660E-03	0.2738E-03	0.5653E-03	0.3396E-03	0.6389E-03	16.00
16.10	0.3018E-16	0.2477E-04	0.2608E-03	0.2727E-03	0.5582E-03	0.3330E-03	0.6304E-03	16.10
16.20	0.2446E-16	0.2406E-04	0.2557E-03	0.2715E-03	0.5513E-03	0.3265E-03	0.6221E-03	16.20
16.30	0.1983E-16	0.2337E-04	0.2507E-03	0.2704E-03	0.5444E-03	0.3201E-03	0.6138E-03	16.30
16.40	0.1607E-16	0.2270E-04	0.2457E-03	0.2693E-03	0.5377E-03	0.3138E-03	0.6058E-03	16.40
16.50	0.1303E-16	0.2205E-04	0.2409E-03	0.2682E-03	0.5311E-03	0.3076E-03	0.5978E-03	16.50
16.60	0.1056E-16	0.2142E-04	0.2362E-03	0.2670E-03	0.5247E-03	0.3016E-03	0.5901E-03	16.60
16.70	0.8560E-17	0.2081E-04	0.2316E-03	0.2659E-03	0.5183E-03	0.2957E-03	0.5824E-03	16.70
16.80	0.6938E-17	0.2022E-04	0.2270E-03	0.2648E-03	0.5121E-03	0.2899E-03	0.5749E-03	16.80
16.90	0.5624E-17	0.1964E-04	0.2226E-03	0.2637E-03	0.5059E-03	0.2842E-03	0.5676E-03	16.90
17.00	0.4559E-17	0.1908E-04	0.2182E-03	0.2626E-03	0.4999E-03	0.2786E-03	0.5603E-03	17.00
17.10	0.3695E-17	0.1853E-04	0.2139E-03	0.2615E-03	0.4940E-03	0.2732E-03	0.5532E-03	17.10
17.20	0.2995E-17	0.1800E-04	0.2097E-03	0.2605E-03	0.4882E-03	0.2678E-03	0.5463E-03	17.20
17.30	0.2428E-17	0.1749E-04	0.2056E-03	0.2594E-03	0.4825E-03	0.2626E-03	0.5394E-03	17.30
17.40	0.1968E-17	0.1699E-04	0.2016E-03	0.2583E-03	0.4769E-03	0.2574E-03	0.5327E-03	17.40
17.50	0.1595E-17	0.1650E-04	0.1976E-03	0.2572E-03	0.4714E-03	0.2524E-03	0.5261E-03	17.50
17.60	0.1293E-17	0.1603E-04	0.1938E-03	0.2561E-03	0.4659E-03	0.2474E-03	0.5196E-03	17.60
17.70	0.1048E-17	0.1557E-04	0.1900E-03	0.2551E-03	0.4606E-03	0.2426E-03	0.5132E-03	17.70
17.80	0.8497E-18	0.1513E-04	0.1862E-03	0.2540E-03	0.4554E-03	0.2378E-03	0.5070E-03	17.80
17.90	0.6887E-18	0.1470E-04	0.1826E-03	0.2530E-03	0.4503E-03	0.2332E-03	0.5008E-03	17.90
18.00	0.5583E-18	0.1428E-04	0.1790E-03	0.2519E-03	0.4452E-03	0.2286E-03	0.4948E-03	18.00

******************************** SSES DATA GENERATED CURVES ********************************

T (HRS)	I(T)	GR(T)	SR(T) OBTUSE LINES	SS(T)	SUM(T) OBTUSE LINES	SR(T) NO OBTUSE LINES	SUM(T) NO OBTUSE LINES	T (HRS)
18.10	0.4525E-18	0.1387E-04	0.1755E-03	0.2509E-03	0.4402E-03	0.2241E-03	0.4888E-03	18.10
18.20	0.3668E-18	0.1347E-04	0.1721E-03	0.2498E-03	0.4354E-03	0.2197E-03	0.4830E-03	18.20
18.30	0.2973E-18	0.1309E-04	0.1687E-03	0.2488E-03	0.4306E-03	0.2154E-03	0.4773E-03	18.30
18.40	0.2410E-18	0.1271E-04	0.1654E-03	0.2477E-03	0.4258E-03	0.2112E-03	0.4716E-03	18.40
18.50	0.1954E-18	0.1235E-04	0.1621E-03	0.2467E-03	0.4212E-03	0.2070E-03	0.4661E-03	18.50
18.60	0.1584E-18	0.1200E-04	0.1590E-03	0.2457E-03	0.4167E-03	0.2030E-03	0.4607E-03	18.60
18.70	0.1284E-18	0.1165E-04	0.1558E-03	0.2447E-03	0.4122E-03	0.1990E-03	0.4553E-03	18.70
18.80	0.1040E-18	0.1132E-04	0.1528E-03	0.2437E-03	0.4078E-03	0.1951E-03	0.4501E-03	18.80
18.90	0.8434E-19	0.1100E-04	0.1498E-03	0.2426E-03	0.4034E-03	0.1913E-03	0.4449E-03	18.90
19.00	0.6836E-19	0.1068E-04	0.1469E-03	0.2416E-03	0.3992E-03	0.1875E-03	0.4398E-03	19.00
19.10	0.5541E-19	0.1038E-04	0.1440E-03	0.2406E-03	0.3950E-03	0.1838E-03	0.4349E-03	19.10
19.20	0.4492E-19	0.1008E-04	0.1412E-03	0.2396E-03	0.3909E-03	0.1802E-03	0.4300E-03	19.20
19.30	0.3641E-19	0.9792E-05	0.1384E-03	0.2386E-03	0.3868E-03	0.1767E-03	0.4251E-03	19.30
19.40	0.2951E-19	0.9512E-05	0.1357E-03	0.2376E-03	0.3828E-03	0.1732E-03	0.4204E-03	19.40
19.50	0.2392E-19	0.9240E-05	0.1330E-03	0.2367E-03	0.3789E-03	0.1698E-03	0.4157E-03	19.50
19.60	0.1939E-19	0.8976E-05	0.1304E-03	0.2357E-03	0.3751E-03	0.1665E-03	0.4112E-03	19.60
19.70	0.1572E-19	0.8719E-05	0.1279E-03	0.2347E-03	0.3713E-03	0.1633E-03	0.4067E-03	19.70
19.80	0.1274E-19	0.8470E-05	0.1253E-03	0.2337E-03	0.3675E-03	0.1601E-03	0.4022E-03	19.80
19.90	0.1033E-19	0.8228E-05	0.1229E-03	0.2327E-03	0.3639E-03	0.1569E-03	0.3979E-03	19.90
20.00	0.8372E-20	0.7993E-05	0.1205E-03	0.2318E-03	0.3602E-03	0.1538E-03	0.3936E-03	20.00
20.10	0.6786E-20	0.7764E-05	0.1181E-03	0.2308E-03	0.3567E-03	0.1508E-03	0.3894E-03	20.10
20.20	0.5500E-20	0.7542E-05	0.1158E-03	0.2298E-03	0.3532E-03	0.1479E-03	0.3853E-03	20.20
20.30	0.4459E-20	0.7327E-05	0.1135E-03	0.2289E-03	0.3498E-03	0.1450E-03	0.3812E-03	20.30
20.40	0.3614E-20	0.7117E-05	0.1113E-03	0.2279E-03	0.3464E-03	0.1421E-03	0.3772E-03	20.40
20.50	0.2930E-20	0.6914E-05	0.1091E-03	0.2270E-03	0.3430E-03	0.1393E-03	0.3732E-03	20.50
20.60	0.2375E-20	0.6716E-05	0.1070E-03	0.2260E-03	0.3397E-03	0.1366E-03	0.3694E-03	20.60
20.70	0.1925E-20	0.6524E-05	0.1049E-03	0.2251E-03	0.3365E-03	0.1339E-03	0.3656E-03	20.70
20.80	0.1560E-20	0.6338E-05	0.1028E-03	0.2242E-03	0.3333E-03	0.1313E-03	0.3618E-03	20.80
20.90	0.1265E-20	0.6157E-05	0.1008E-03	0.2232E-03	0.3302E-03	0.1287E-03	0.3581E-03	20.90
21.00	0.1025E-20	0.5981E-05	0.9884E-04	0.2223E-03	0.3271E-03	0.1262E-03	0.3545E-03	21.00
21.10	0.8310E-21	0.5810E-05	0.9690E-04	0.2214E-03	0.3241E-03	0.1237E-03	0.3509E-03	21.10
21.20	0.6736E-21	0.5644E-05	0.9500E-04	0.2205E-03	0.3211E-03	0.1213E-03	0.3474E-03	21.20
21.30	0.5460E-21	0.5482E-05	0.9314E-04	0.2196E-03	0.3182E-03	0.1189E-03	0.3440E-03	21.30
21.40	0.4426E-21	0.5328E-05	0.9131E-04	0.2186E-03	0.3153E-03	0.1166E-03	0.3406E-03	21.40
21.50	0.3587E-21	0.5173E-05	0.8952E-04	0.2177E-03	0.3124E-03	0.1143E-03	0.3372E-03	21.50
21.60	0.2908E-21	0.5026E-05	0.8777E-04	0.2168E-03	0.3096E-03	0.1121E-03	0.3339E-03	21.60
21.70	0.2357E-21	0.4882E-05	0.8605E-04	0.2159E-03	0.3068E-03	0.1099E-03	0.3307E-03	21.70
21.80	0.1911E-21	0.4742E-05	0.8436E-04	0.2150E-03	0.3041E-03	0.1077E-03	0.3275E-03	21.80
21.90	0.1549E-21	0.4607E-05	0.8270E-04	0.2141E-03	0.3014E-03	0.1056E-03	0.3243E-03	21.90
22.00	0.1255E-21	0.4475E-05	0.8108E-04	0.2132E-03	0.2988E-03	0.1035E-03	0.3213E-03	22.00
22.10	0.1018E-21	0.4347E-05	0.7949E-04	0.2124E-03	0.2962E-03	0.1015E-03	0.3182E-03	22.10
22.20	0.8248E-22	0.4223E-05	0.7793E-04	0.2115E-03	0.2936E-03	0.9952E-04	0.3152E-03	22.20
22.30	0.6686E-22	0.4102E-05	0.7641E-04	0.2106E-03	0.2911E-03	0.9756E-04	0.3123E-03	22.30
22.40	0.5419E-22	0.3985E-05	0.7491E-04	0.2097E-03	0.2886E-03	0.9565E-04	0.3094E-03	22.40
22.50	0.4393E-22	0.3871E-05	0.7344E-04	0.2088E-03	0.2862E-03	0.9378E-04	0.3065E-03	22.50
22.60	0.3561E-22	0.3760E-05	0.7200E-04	0.2080E-03	0.2837E-03	0.9194E-04	0.3037E-03	22.60
22.70	0.2886E-22	0.3653E-05	0.7059E-04	0.2071E-03	0.2814E-03	0.9014E-04	0.3009E-03	22.70
22.80	0.2340E-22	0.3549E-05	0.6921E-04	0.2062E-03	0.2790E-03	0.8837E-04	0.2982E-03	22.80
22.90	0.1896E-22	0.3447E-05	0.6785E-04	0.2054E-03	0.2767E-03	0.8664E-04	0.2955E-03	22.90
23.00	0.1537E-22	0.3349E-05	0.6652E-04	0.2045E-03	0.2744E-03	0.8494E-04	0.2928E-03	23.00
23.10	0.1246E-22	0.3253E-05	0.6521E-04	0.2037E-03	0.2722E-03	0.8327E-04	0.2902E-03	23.10
23.20	0.1010E-22	0.3160E-05	0.6394E-04	0.2028E-03	0.2699E-03	0.8164E-04	0.2876E-03	23.20
23.30	0.8187E-23	0.3070E-05	0.6268E-04	0.2020E-03	0.2677E-03	0.8004E-04	0.2851E-03	23.30
23.40	0.6636E-23	0.2982E-05	0.6145E-04	0.2012E-03	0.2656E-03	0.7847E-04	0.2826E-03	23.40
23.50	0.5379E-23	0.2897E-05	0.6025E-04	0.2003E-03	0.2635E-03	0.7693E-04	0.2801E-03	23.50
23.60	0.4360E-23	0.2814E-05	0.5907E-04	0.1995E-03	0.2614E-03	0.7542E-04	0.2777E-03	23.60
23.70	0.3534E-23	0.2733E-05	0.5791E-04	0.1987E-03	0.2593E-03	0.7394E-04	0.2753E-03	23.70
23.80	0.2865E-23	0.2655E-05	0.5677E-04	0.1978E-03	0.2573E-03	0.7249E-04	0.2730E-03	23.80
23.90	0.2322E-23	0.2579E-05	0.5566E-04	0.1970E-03	0.2552E-03	0.7107E-04	0.2707E-03	23.90
24.00	0.1882E-23	0.2506E-05	0.5457E-04	0.1962E-03	0.2533E-03	0.6968E-04	0.2684E-03	24.00

SSES DATA GENERATED CURVES

T (HRS)	I(T)	GR(T)	SR(T) OBTUSE LINES	SS(T)	SUM(T) OBTUSE LINES	SR(T) NO OBTUSE LINES	SUM(T) NO OBTUSE LINES	T (HRS)
24.10	0.1526E-23	0.2434E-05	0.5350E-04	0.1954E-03	0.2513E-03	0.6831E-04	0.2661E-03	24.10
24.20	0.1237E-23	0.2364E-05	0.5245E-04	0.1946E-03	0.2494E-03	0.6697E-04	0.2639E-03	24.20
24.30	0.1003E-23	0.2297E-05	0.5142E-04	0.1938E-03	0.2475E-03	0.6566E-04	0.2617E-03	24.30
24.40	0.8127E-24	0.2231E-05	0.5041E-04	0.1929E-03	0.2456E-03	0.6437E-04	0.2596E-03	24.40
24.50	0.6587E-24	0.2167E-05	0.4943E-04	0.1921E-03	0.2437E-03	0.6311E-04	0.2574E-03	24.50
24.60	0.5340E-24	0.2105E-05	0.4846E-04	0.1913E-03	0.2419E-03	0.6187E-04	0.2553E-03	24.60
24.70	0.4328E-24	0.2045E-05	0.4751E-04	0.1906E-03	0.2401E-03	0.6066E-04	0.2533E-03	24.70
24.80	0.3508E-24	0.1987E-05	0.4658E-04	0.1898E-03	0.2383E-03	0.5947E-04	0.2512E-03	24.80
24.90	0.2844E-24	0.1930E-05	0.4566E-04	0.1890E-03	0.2366E-03	0.5831E-04	0.2492E-03	24.90
25.00	0.2305E-24	0.1875E-05	0.4477E-04	0.1882E-03	0.2348E-03	0.5716E-04	0.2472E-03	25.00
25.10	0.1869E-24	0.1821E-05	0.4389E-04	0.1874E-03	0.2331E-03	0.5604E-04	0.2453E-03	25.10
25.20	0.1515E-24	0.1769E-05	0.4303E-04	0.1866E-03	0.2314E-03	0.5494E-04	0.2433E-03	25.20
25.30	0.1228E-24	0.1719E-05	0.4219E-04	0.1858E-03	0.2298E-03	0.5387E-04	0.2414E-03	25.30
25.40	0.9952E-25	0.1670E-05	0.4136E-04	0.1851E-03	0.2281E-03	0.5281E-04	0.2396E-03	25.40
25.50	0.8067E-25	0.1622E-05	0.4055E-04	0.1843E-03	0.2265E-03	0.5178E-04	0.2377E-03	25.50
25.60	0.6539E-25	0.1575E-05	0.3975E-04	0.1835E-03	0.2249E-03	0.5076E-04	0.2359E-03	25.60
25.70	0.5300E-25	0.1530E-05	0.3897E-04	0.1828E-03	0.2233E-03	0.4976E-04	0.2341E-03	25.70
25.80	0.4296E-25	0.1487E-05	0.3821E-04	0.1820E-03	0.2217E-03	0.4879E-04	0.2323E-03	25.80
25.90	0.3482E-25	0.1444E-05	0.3746E-04	0.1813E-03	0.2202E-03	0.4783E-04	0.2305E-03	25.90
26.00	0.2823E-25	0.1403E-05	0.3673E-04	0.1805E-03	0.2186E-03	0.4689E-04	0.2288E-03	26.00
26.10	0.2288E-25	0.1363E-05	0.3601E-04	0.1798E-03	0.2171E-03	0.4598E-04	0.2271E-03	26.10
26.20	0.1855E-25	0.1324E-05	0.3530E-04	0.1790E-03	0.2156E-03	0.4507E-04	0.2254E-03	26.20
26.30	0.1503E-25	0.1286E-05	0.3461E-04	0.1783E-03	0.2142E-03	0.4419E-04	0.2237E-03	26.30
26.40	0.1219E-25	0.1249E-05	0.3393E-04	0.1775E-03	0.2127E-03	0.4332E-04	0.2221E-03	26.40
26.50	0.9876E-26	0.1214E-05	0.3326E-04	0.1768E-03	0.2113E-03	0.4247E-04	0.2205E-03	26.50
26.60	0.8007E-26	0.1179E-05	0.3261E-04	0.1760E-03	0.2098E-03	0.4164E-04	0.2189E-03	26.60
26.70	0.6490E-26	0.1145E-05	0.3197E-04	0.1753E-03	0.2084E-03	0.4083E-04	0.2173E-03	26.70
26.80	0.5261E-26	0.1112E-05	0.3135E-04	0.1746E-03	0.2070E-03	0.4003E-04	0.2157E-03	26.80
26.90	0.4264E-26	0.1081E-05	0.3073E-04	0.1739E-03	0.2057E-03	0.3924E-04	0.2142E-03	26.90
27.00	0.3457E-26	0.1050E-05	0.3013E-04	0.1731E-03	0.2043E-03	0.3847E-04	0.2127E-03	27.00
27.10	0.2802E-26	0.1020E-05	0.2954E-04	0.1724E-03	0.2030E-03	0.3772E-04	0.2112E-03	27.10
27.20	0.2271E-26	0.9906E-06	0.2896E-04	0.1717E-03	0.2016E-03	0.3698E-04	0.2097E-03	27.20
27.30	0.1841E-26	0.9623E-06	0.2839E-04	0.1710E-03	0.2003E-03	0.3625E-04	0.2082E-03	27.30
27.40	0.1492E-26	0.9348E-06	0.2783E-04	0.1703E-03	0.1990E-03	0.3554E-04	0.2068E-03	27.40
27.50	0.1210E-26	0.9080E-06	0.2729E-04	0.1696E-03	0.1978E-03	0.3484E-04	0.2053E-03	27.50
27.60	0.9805E-27	0.8821E-06	0.2675E-04	0.1689E-03	0.1965E-03	0.3416E-04	0.2039E-03	27.60
27.70	0.7948E-27	0.8569E-06	0.2623E-04	0.1682E-03	0.1952E-03	0.3349E-04	0.2025E-03	27.70
27.80	0.6443E-27	0.8324E-06	0.2572E-04	0.1675E-03	0.1940E-03	0.3284E-04	0.2011E-03	27.80
27.90	0.5222E-27	0.8086E-06	0.2521E-04	0.1668E-03	0.1928E-03	0.3219E-04	0.1998E-03	27.90
28.00	0.4233E-27	0.7855E-06	0.2472E-04	0.1661E-03	0.1916E-03	0.3156E-04	0.1984E-03	28.00
28.10	0.3431E-27	0.7630E-06	0.2423E-04	0.1654E-03	0.1904E-03	0.3094E-04	0.1971E-03	28.10
28.20	0.2781E-27	0.7412E-06	0.2376E-04	0.1647E-03	0.1892E-03	0.3034E-04	0.1958E-03	28.20
28.30	0.2254E-27	0.7200E-06	0.2329E-04	0.1640E-03	0.1880E-03	0.2974E-04	0.1945E-03	28.30
28.40	0.1827E-27	0.6994E-06	0.2283E-04	0.1633E-03	0.1869E-03	0.2916E-04	0.1932E-03	28.40
28.50	0.1481E-27	0.6795E-06	0.2239E-04	0.1626E-03	0.1857E-03	0.2859E-04	0.1919E-03	28.50
28.60	0.1201E-27	0.6600E-06	0.2195E-04	0.1620E-03	0.1846E-03	0.2803E-04	0.1907E-03	28.60
28.70	0.9733E-28	0.6412E-06	0.2152E-04	0.1613E-03	0.1835E-03	0.2748E-04	0.1894E-03	28.70
28.80	0.7889E-28	0.6228E-06	0.2110E-04	0.1606E-03	0.1823E-03	0.2694E-04	0.1882E-03	28.80
28.90	0.6395E-28	0.6050E-06	0.2068E-04	0.1600E-03	0.1812E-03	0.2641E-04	0.1870E-03	28.90
29.00	0.5183E-28	0.5877E-06	0.2028E-04	0.1593E-03	0.1802E-03	0.2589E-04	0.1858E-03	29.00
29.10	0.4202E-28	0.5709E-06	0.1988E-04	0.1586E-03	0.1791E-03	0.2538E-04	0.1846E-03	29.10
29.20	0.3406E-28	0.5546E-06	0.1949E-04	0.1580E-03	0.1780E-03	0.2489E-04	0.1834E-03	29.20
29.30	0.2761E-28	0.5388E-06	0.1911E-04	0.1573E-03	0.1770E-03	0.2440E-04	0.1823E-03	29.30
29.40	0.2238E-28	0.5234E-06	0.1873E-04	0.1567E-03	0.1759E-03	0.2392E-04	0.1811E-03	29.40
29.50	0.1814E-28	0.5084E-06	0.1837E-04	0.1560E-03	0.1749E-03	0.2345E-04	0.1800E-03	29.50
29.60	0.1470E-28	0.4939E-06	0.1801E-04	0.1554E-03	0.1739E-03	0.2299E-04	0.1788E-03	29.60
29.70	0.1192E-28	0.4798E-06	0.1765E-04	0.1547E-03	0.1728E-03	0.2254E-04	0.1777E-03	29.70
29.80	0.9661E-29	0.4660E-06	0.1731E-04	0.1541E-03	0.1718E-03	0.2210E-04	0.1766E-03	29.80
29.90	0.7831E-29	0.4527E-06	0.1697E-04	0.1534E-03	0.1709E-03	0.2167E-04	0.1755E-03	29.90
30.00	0.6348E-29	0.4398E-06	0.1653E-04	0.1528E-03	0.1699E-03	0.2124E-04	0.1745E-03	30.00

******************************** SSES DATA GENERATED CURVES ********************************

T (HRS)	I(T)	GR(T)	SR(T) OBTUSE LINES	SS(T)	SUM(T) OBTUSE LINES	SR(T) NO OBTUSE LINES	SUM(T) NO OBTUSE LINES	T (HRS)
30.10	0.5145E-29	0.4272E-06	0.1631E-04	0.1522E-03	0.1689E-03	0.2082E-04	0.1734E-03	30.10
30.20	0.4171E-29	0.4150E-06	0.1599E-04	0.1515E-03	0.1679E-03	0.2042E-04	0.1724E-03	30.20
30.30	0.3381E-29	0.4031E-06	0.1568E-04	0.1509E-03	0.1670E-03	0.2002E-04	0.1713E-03	30.30
30.40	0.2740E-29	0.3916E-06	0.1537E-04	0.1503E-03	0.1660E-03	0.1962E-04	0.1703E-03	30.40
30.50	0.2221E-29	0.3804E-06	0.1507E-04	0.1496E-03	0.1651E-03	0.1924E-04	0.1693E-03	30.50
30.60	0.1801E-29	0.3696E-06	0.1477E-04	0.1490E-03	0.1642E-03	0.1886E-04	0.1683E-03	30.60
30.70	0.1459E-29	0.3590E-06	0.1448E-04	0.1484E-03	0.1632E-03	0.1849E-04	0.1673E-03	30.70
30.80	0.1183E-29	0.3487E-06	0.1420E-04	0.1478E-03	0.1623E-03	0.1813E-04	0.1663E-03	30.80
30.90	0.9589E-30	0.3388E-06	0.1392E-04	0.1472E-03	0.1614E-03	0.1777E-04	0.1653E-03	30.90
31.00	0.7773E-30	0.3291E-06	0.1365E-04	0.1466E-03	0.1605E-03	0.1743E-04	0.1643E-03	31.00
31.10	0.6301E-30	0.3197E-06	0.1338E-04	0.1459E-03	0.1596E-03	0.1708E-04	0.1634E-03	31.10
31.20	0.5107E-30	0.3105E-06	0.1312E-04	0.1453E-03	0.1588E-03	0.1675E-04	0.1624E-03	31.20
31.30	0.4140E-30	0.3017E-06	0.1286E-04	0.1447E-03	0.1579E-03	0.1642E-04	0.1615E-03	31.30
31.40	0.3356E-30	0.2930E-06	0.1261E-04	0.1441E-03	0.1570E-03	0.1610E-04	0.1605E-03	31.40
31.50	0.2720E-30	0.2847E-06	0.1236E-04	0.1435E-03	0.1562E-03	0.1578E-04	0.1596E-03	31.50
31.60	0.2205E-30	0.2765E-06	0.1212E-04	0.1429E-03	0.1553E-03	0.1547E-04	0.1587E-03	31.60
31.70	0.1787E-30	0.2686E-06	0.1188E-04	0.1423E-03	0.1545E-03	0.1517E-04	0.1578E-03	31.70
31.80	0.1449E-30	0.2609E-06	0.1165E-04	0.1418E-03	0.1537E-03	0.1487E-04	0.1569E-03	31.80
31.90	0.1174E-30	0.2535E-06	0.1142E-04	0.1412E-03	0.1528E-03	0.1458E-04	0.1560E-03	31.90
32.00	0.9518E-31	0.2462E-06	0.1120E-04	0.1406E-03	0.1520E-03	0.1429E-04	0.1551E-03	32.00
32.10	0.7716E-31	0.2392E-06	0.1098E-04	0.1400E-03	0.1512E-03	0.1401E-04	0.1542E-03	32.10
32.20	0.6254E-31	0.2324E-06	0.1076E-04	0.1394E-03	0.1504E-03	0.1374E-04	0.1534E-03	32.20
32.30	0.5070E-31	0.2257E-06	0.1055E-04	0.1388E-03	0.1496E-03	0.1347E-04	0.1525E-03	32.30
32.40	0.4109E-31	0.2193E-06	0.1034E-04	0.1383E-03	0.1488E-03	0.1321E-04	0.1517E-03	32.40
32.50	0.3331E-31	0.2130E-06	0.1014E-04	0.1377E-03	0.1480E-03	0.1295E-04	0.1508E-03	32.50
32.60	0.2700E-31	0.2069E-06	0.9941E-05	0.1371E-03	0.1473E-03	0.1269E-04	0.1500E-03	32.60
32.70	0.2189E-31	0.2010E-06	0.9746E-05	0.1365E-03	0.1465E-03	0.1244E-04	0.1492E-03	32.70
32.80	0.1774E-31	0.1953E-06	0.9555E-05	0.1360E-03	0.1457E-03	0.1220E-04	0.1484E-03	32.80
32.90	0.1438E-31	0.1897E-06	0.9368E-05	0.1354E-03	0.1450E-03	0.1196E-04	0.1476E-03	32.90
33.00	0.1166E-31	0.1842E-06	0.9184E-05	0.1348E-03	0.1442E-03	0.1173E-04	0.1468E-03	33.00
33.10	0.9448E-32	0.1790E-06	0.9004E-05	0.1343E-03	0.1435E-03	0.1150E-04	0.1460E-03	33.10
33.20	0.7659E-32	0.1739E-06	0.8828E-05	0.1337E-03	0.1427E-03	0.1127E-04	0.1452E-03	33.20
33.30	0.6208E-32	0.1689E-06	0.8654E-05	0.1332E-03	0.1420E-03	0.1105E-04	0.1444E-03	33.30
33.40	0.5032E-32	0.1641E-06	0.8485E-05	0.1326E-03	0.1413E-03	0.1083E-04	0.1436E-03	33.40
33.50	0.4079E-32	0.1594E-06	0.8318E-05	0.1321E-03	0.1405E-03	0.1062E-04	0.1428E-03	33.50
33.60	0.3306E-32	0.1548E-06	0.8155E-05	0.1315E-03	0.1398E-03	0.1041E-04	0.1421E-03	33.60
33.70	0.2680E-32	0.1504E-06	0.7995E-05	0.1310E-03	0.1391E-03	0.1021E-04	0.1413E-03	33.70
33.80	0.2172E-32	0.1461E-06	0.7839E-05	0.1304E-03	0.1384E-03	0.1001E-04	0.1406E-03	33.80
33.90	0.1761E-32	0.1419E-06	0.7685E-05	0.1299E-03	0.1377E-03	0.9813E-05	0.1398E-03	33.90
34.00	0.1427E-32	0.1379E-06	0.7534E-05	0.1293E-03	0.1370E-03	0.9621E-05	0.1391E-03	34.00
34.10	0.1157E-32	0.1339E-06	0.7387E-05	0.1288E-03	0.1363E-03	0.9432E-05	0.1384E-03	34.10
34.20	0.9378E-33	0.1301E-06	0.7242E-05	0.1283E-03	0.1356E-03	0.9247E-05	0.1376E-03	34.20
34.30	0.7602E-33	0.1264E-06	0.7100E-05	0.1277E-03	0.1350E-03	0.9066E-05	0.1369E-03	34.30
34.40	0.6162E-33	0.1228E-06	0.6961E-05	0.1272E-03	0.1343E-03	0.8888E-05	0.1362E-03	34.40
34.50	0.4995E-33	0.1193E-06	0.6824E-05	0.1267E-03	0.1336E-03	0.8714E-05	0.1355E-03	34.50
34.60	0.4049E-33	0.1158E-06	0.6690E-05	0.1261E-03	0.1329E-03	0.8543E-05	0.1348E-03	34.60
34.70	0.3282E-33	0.1125E-06	0.6559E-05	0.1256E-03	0.1323E-03	0.8375E-05	0.1341E-03	34.70
34.80	0.2660E-33	0.1093E-06	0.6431E-05	0.1251E-03	0.1316E-03	0.8211E-05	0.1334E-03	34.80
34.90	0.2156E-33	0.1062E-06	0.6305E-05	0.1246E-03	0.1310E-03	0.8050E-05	0.1327E-03	34.90
35.00	0.1748E-33	0.1032E-06	0.6181E-05	0.1241E-03	0.1303E-03	0.7892E-05	0.1321E-03	35.00
35.10	0.1417E-33	0.1002E-06	0.6060E-05	0.1235E-03	0.1297E-[illegible]	0.7738E-05	0.1314E-03	35.10
35.20	0.1148E-33	0.9735E-07	0.5941E-05	0.1230E-03	0.1291E-03	0.7586E-05	0.1307E-03	35.20
35.30	0.9309E-34	0.9457E-07	0.5825E-05	0.1225E-03	0.1284E-03	0.7437E-05	0.1300E-03	35.30
35.40	0.7546E-34	0.9186E-07	0.5710E-05	0.1220E-03	0.1278E-03	0.7292E-05	0.1294E-03	35.40
35.50	0.6116E-34	0.8924E-07	0.5598E-05	0.1215E-03	0.1272E-03	0.7149E-05	0.1287E-03	35.50
35.60	0.4958E-34	0.8669E-07	0.5489E-05	0.1210E-03	0.1266E-03	0.7008E-05	0.1281E-03	35.60
35.70	0.4019E-34	0.8421E-07	0.5381E-05	0.1205E-03	0.1260E-03	0.6871E-05	0.1274E-03	35.70
35.80	0.3258E-34	0.8180E-07	0.5276E-05	0.1200E-03	0.1253E-03	0.6736E-05	0.1268E-03	35.80
35.90	0.2641E-34	0.7946E-07	0.5172E-05	0.1195E-03	0.1247E-03	0.6604E-05	0.1262E-03	35.90
36.00	0.2140E-34	0.7719E-07	0.5071E-05	0.1190E-03	0.1241E-03	0.6475E-05	0.1255E-03	36.00

SSES DATA GENERATED CURVES

T (HRS)	I(T)	GR(T)	SR(T) OBTUSE LINES	SS(T)	SUM(T) OBTUSE LINES	SR(T) NO OBTUSE LINES	SUM(T) NO OBTUSE LINES	T (HRS)
36.10	0.1735E-34	0.7499E-07	0.4971E-05	0.1185E-03	0.1235E-03	0.6348E-05	0.1249E-03	36.10
36.20	0.1406E-34	0.7284E-07	0.4874E-05	0.1180E-03	0.1230E-03	0.6223E-05	0.1243E-03	36.20
36.30	0.1140E-34	0.7076E-07	0.4778E-05	0.1175E-03	0.1224E-03	0.6101E-05	0.1237E-03	36.30
36.40	0.9240E-35	0.6874E-07	0.4685E-05	0.1170E-03	0.1218E-03	0.5982E-05	0.1231E-03	36.40
36.50	0.7490E-35	0.6677E-07	0.4593E-05	0.1165E-03	0.1212E-03	0.5864E-05	0.1225E-03	36.50
36.60	0.6071E-35	0.6486E-07	0.4503E-05	0.1161E-03	0.1206E-03	0.5749E-05	0.1219E-03	36.60
36.70	0.4921E-35	0.6301E-07	0.4414E-05	0.1156E-03	0.1201E-03	0.5637E-05	0.1213E-03	36.70
36.80	0.3989E-35	0.6121E-07	0.4328E-05	0.1151E-03	0.1195E-03	0.5526E-05	0.1207E-03	36.80
36.90	0.3234E-35	0.5946E-07	0.4243E-05	0.1146E-03	0.1189E-03	0.5418E-05	0.1201E-03	36.90
37.00	0.2621E-35	0.5776E-07	0.4160E-05	0.1141E-03	0.1184E-03	0.5312E-05	0.1195E-03	37.00
37.10	0.2125E-35	0.5611E-07	0.4078E-05	0.1137E-03	0.1178E-03	0.5208E-05	0.1189E-03	37.10
37.20	0.1722E-35	0.5450E-07	0.3998E-05	0.1132E-03	0.1172E-03	0.5105E-05	0.1184E-03	37.20
37.30	0.1396E-35	0.5295E-07	0.3920E-05	0.1127E-03	0.1167E-03	0.5005E-05	0.1178E-03	37.30
37.40	0.1132E-35	0.5143E-07	0.3843E-05	0.1123E-03	0.1161E-03	0.4907E-05	0.1172E-03	37.40
37.50	0.9172E-36	0.4996E-07	0.3768E-05	0.1118E-03	0.1156E-03	0.4811E-05	0.1166E-03	37.50
37.60	0.7435E-36	0.4854E-07	0.3694E-05	0.1113E-03	0.1151E-03	0.4717E-05	0.1161E-03	37.60
37.70	0.6026E-36	0.4715E-07	0.3621E-05	0.1109E-03	0.1145E-03	0.4624E-05	0.1155E-03	37.70
37.80	0.4885E-36	0.4580E-07	0.3550E-05	0.1104E-03	0.1140E-03	0.4534E-05	0.1150E-03	37.80
37.90	0.3960E-36	0.4449E-07	0.3481E-05	0.1099E-03	0.1135E-03	0.4445E-05	0.1144E-03	37.90
38.00	0.3210E-36	0.4322E-07	0.3413E-05	0.1095E-03	0.1129E-03	0.4358E-05	0.1139E-03	38.00
38.10	0.2602E-36	0.4198E-07	0.3346E-05	0.1090E-03	0.1124E-03	0.4272E-05	0.1133E-03	38.10
38.20	0.2109E-36	0.4078E-07	0.3280E-05	0.1086E-03	0.1119E-03	0.4188E-05	0.1128E-03	38.20
38.30	0.1709E-36	0.3962E-07	0.3216E-05	0.1081E-03	0.1114E-03	0.4106E-05	0.1123E-03	38.30
38.40	0.1386E-36	0.3849E-07	0.3153E-05	0.1077E-03	0.1109E-03	0.4026E-05	0.1117E-03	38.40
38.50	0.1123E-36	0.3739E-07	0.3091E-05	0.1072E-03	0.1104E-03	0.3947E-05	0.1112E-03	38.50
38.60	0.9104E-37	0.3632E-07	0.3030E-05	0.1068E-03	0.1098E-03	0.3869E-05	0.1107E-03	38.60
38.70	0.7380E-37	0.3528E-07	0.2971E-05	0.1063E-03	0.1093E-03	0.3794E-05	0.1102E-03	38.70
38.80	0.5982E-37	0.3427E-07	0.2913E-05	0.1059E-03	0.1088E-03	0.3719E-05	0.1096E-03	38.80
38.90	0.4849E-37	0.3329E-07	0.2856E-05	0.1055E-03	0.1083E-03	0.3646E-05	0.1091E-03	38.90
39.00	0.3930E-37	0.3234E-07	0.2800E-05	0.1050E-03	0.1078E-03	0.3575E-05	0.1086E-03	39.00
39.10	0.3186E-37	0.3142E-07	0.2745E-05	0.1046E-03	0.1074E-03	0.3505E-05	0.1081E-03	39.10
39.20	0.2582E-37	0.3052E-07	0.2691E-05	0.1041E-03	0.1069E-03	0.3436E-05	0.1076E-03	39.20
39.30	0.2093E-37	0.2964E-07	0.2638E-05	0.1037E-03	0.1064E-03	0.3369E-05	0.1071E-03	39.30
39.40	0.1697E-37	0.2880E-07	0.2586E-05	0.1033E-03	0.1059E-03	0.3303E-05	0.1066E-03	39.40
39.50	0.1375E-37	0.2797E-07	0.2536E-05	0.1028E-03	0.1054E-03	0.3238E-05	0.1061E-03	39.50
39.60	0.1115E-37	0.2717E-07	0.2486E-05	0.1024E-03	0.1049E-03	0.3174E-05	0.1056E-03	39.60
39.70	0.9037E-38	0.2640E-07	0.2437E-05	0.1020E-03	0.1045E-03	0.3112E-05	0.1051E-03	39.70
39.80	0.7325E-38	0.2564E-07	0.2389E-05	0.1016E-03	0.1040E-03	0.3051E-05	0.1046E-03	39.80
39.90	0.5938E-38	0.2491E-07	0.2343E-05	0.1011E-03	0.1035E-03	0.2991E-05	0.1042E-03	39.90
40.00	0.4813E-38	0.2420E-07	0.2297E-05	0.1007E-03	0.1030E-03	0.2933E-05	0.1037E-03	40.00
40.10	0.3901E-38	0.2351E-07	0.2252E-05	0.1003E-03	0.1026E-03	0.2875E-05	0.1032E-03	40.10
40.20	0.3162E-38	0.2283E-07	0.2208E-05	0.9989E-04	0.1021E-03	0.2819E-05	0.1027E-03	40.20
40.30	0.2563E-38	0.2218E-07	0.2164E-05	0.9948E-04	0.1017E-03	0.2764E-05	0.1023E-03	40.30
40.40	0.2078E-38	0.2155E-07	0.2122E-05	0.9906E-04	0.1012E-03	0.2709E-05	0.1018E-03	40.40
40.50	0.1684E-38	0.2093E-07	0.2080E-05	0.9865E-04	0.1008E-03	0.2656E-05	0.1013E-03	40.50
40.60	0.1365E-38	0.2033E-07	0.2039E-05	0.9824E-04	0.1003E-03	0.2604E-05	0.1009E-03	40.60
40.70	0.1107E-38	0.1975E-07	0.1999E-05	0.9783E-04	0.9985E-04	0.2553E-05	0.1004E-03	40.70
40.80	0.8970E-39	0.1919E-07	0.1960E-05	0.9743E-04	0.9940E-04	0.2503E-05	0.9995E-04	40.80
40.90	0.7271E-39	0.1864E-07	0.1922E-05	0.9702E-04	0.9896E-04	0.2454E-05	0.9949E-04	40.90
41.00	0.5894E-39	0.1811E-07	0.1884E-05	0.9662E-04	0.9852E-04	0.2406E-05	0.9904E-04	41.00
41.10	0.4778E-39	0.1759E-07	0.1847E-05	0.9621E-04	0.9808E-04	0.2359E-05	0.9859E-04	41.10
41.20	0.3873E-39	0.1709E-07	0.1811E-05	0.9581E-04	0.9764E-04	0.2312E-05	0.9814E-04	41.20
41.30	0.3139E-39	0.1660E-07	0.1775E-05	0.9542E-04	0.9721E-04	0.2267E-05	0.9770E-04	41.30
41.40	0.2544E-39	0.1612E-07	0.1741E-05	0.9502E-04	0.9678E-04	0.2223E-05	0.9726E-04	41.40
41.50	0.2062E-39	0.1566E-07	0.1707E-05	0.9462E-04	0.9635E-04	0.2179E-05	0.9682E-04	41.50
41.60	0.1672E-39	0.1522E-07	0.1673E-05	0.9423E-04	0.9592E-04	0.2136E-05	0.9638E-04	41.60
41.70	0.1355E-39	0.1478E-07	0.1640E-05	0.9384E-04	0.9549E-04	0.2094E-05	0.9595E-04	41.70
41.80	0.1098E-39	0.1436E-07	0.1608E-05	0.9345E-04	0.9507E-04	0.2053E-05	0.9552E-04	41.80
41.90	0.8904E-40	0.1395E-07	0.1577E-05	0.9306E-04	0.9465E-04	0.2013E-05	0.9509E-04	41.90
42.00	0.7217E-40	0.1355E-07	0.1546E-05	0.9267E-04	0.9423E-04	0.1974E-05	0.9466E-04	42.00

SSES DATA GENERATED CURVES

T (HRS)	I(T)	GR(T)	SR(T) OBTUSE LINES	SS(T)	SUM(T) OBTUSE LINES	SR(T) NO OBTUSE LINES	SUM(T) NO OBTUSE LINES	T (HRS)
42.10	0.5850E-40	0.1316E-07	0.1515E-05	0.9229E-04	0.9382E-04	0.1935E-05	0.9424E-04	42.10
42.20	0.4742E-40	0.1279E-07	0.1486E-05	0.9190E-04	0.9340E-04	0.1897E-05	0.9381E-04	42.20
42.30	0.3844E-40	0.1242E-07	0.1457E-05	0.9152E-04	0.9299E-04	0.1860E-05	0.9339E-04	42.30
42.40	0.3116E-40	0.1206E-07	0.1428E-05	0.9114E-04	0.9258E-04	0.1823E-05	0.9298E-04	42.40
42.50	0.2526E-40	0.1172E-07	0.1400E-05	0.9076E-04	0.9217E-04	0.1788E-05	0.9256E-04	42.50
42.60	0.2047E-40	0.1138E-07	0.1373E-05	0.9039E-04	0.9177E-04	0.1753E-05	0.9215E-04	42.60
42.70	0.1659E-40	0.1106E-07	0.1346E-05	0.9001E-04	0.9137E-04	0.1718E-05	0.9174E-04	42.70
42.80	0.1345E-40	0.1074E-07	0.1319E-05	0.8964E-04	0.9097E-04	0.1685E-05	0.9133E-04	42.80
42.90	0.1090E-40	0.1044E-07	0.1293E-05	0.8926E-04	0.9057E-04	0.1652E-05	0.9092E-04	42.90
43.00	0.8838E-41	0.1014E-07	0.1268E-05	0.8889E-04	0.9017E-04	0.1619E-05	0.9052E-04	43.00
43.10	0.7164E-41	0.9848E-08	0.1243E-05	0.8852E-04	0.8978E-04	0.1587E-05	0.9012E-04	43.10
43.20	0.5807E-41	0.9567E-08	0.1219E-05	0.8815E-04	0.8938E-04	0.1556E-05	0.8972E-04	43.20
43.30	0.4707E-41	0.9293E-08	0.1195E-05	0.8779E-04	0.8899E-04	0.1526E-05	0.8932E-04	43.30
43.40	0.3816E-41	0.9028E-08	0.1171E-05	0.8742E-04	0.8860E-04	0.1496E-05	0.8893E-04	43.40
43.50	0.3093E-41	0.8770E-08	0.1149E-05	0.8706E-04	0.8822E-04	0.1467E-05	0.8853E-04	43.50
43.60	0.2507E-41	0.8519E-08	0.1126E-05	0.8670E-04	0.8783E-04	0.1438E-05	0.8814E-04	43.60
43.70	0.2032E-41	0.8275E-08	0.1104E-05	0.8634E-04	0.8745E-04	0.1410E-05	0.8775E-04	43.70
43.80	0.1647E-41	0.8039E-08	0.1082E-05	0.8598E-04	0.8707E-04	0.1382E-05	0.8737E-04	43.80
43.90	0.1335E-41	0.7809E-08	0.1061E-05	0.8562E-04	0.8669E-04	0.1355E-05	0.8698E-04	43.90
44.00	0.1082E-41	0.7586E-08	0.1040E-05	0.8526E-04	0.8631E-04	0.1328E-05	0.8660E-04	44.00
44.10	0.8773E-42	0.7369E-08	0.1020E-05	0.8491E-04	0.8594E-04	0.1302E-05	0.8622E-04	44.10
44.20	0.7111E-42	0.7158E-08	0.9999E-06	0.8456E-04	0.8556E-04	0.1277E-05	0.8584E-04	44.20
44.30	0.5764E-42	0.6954E-08	0.9803E-06	0.8420E-04	0.8519E-04	0.1252E-05	0.8546E-04	44.30
44.40	0.4672E-42	0.6755E-08	0.9611E-06	0.8385E-04	0.8482E-04	0.1227E-05	0.8509E-04	44.40
44.50	0.3787E-42	0.6562E-08	0.9422E-06	0.8351E-04	0.8445E-04	0.1203E-05	0.8472E-04	44.50
44.60	0.3070E-42	0.6374E-08	0.9237E-06	0.8316E-04	0.8409E-04	0.1180E-05	0.8434E-04	44.60
44.70	0.2488E-42	0.6192E-08	0.9056E-06	0.8281E-04	0.8372E-04	0.1156E-05	0.8398E-04	44.70
44.80	0.2017E-42	0.6015E-08	0.8879E-06	0.8247E-04	0.8336E-04	0.1134E-05	0.8361E-04	44.80
44.90	0.1635E-42	0.5843E-08	0.8705E-06	0.8213E-04	0.8300E-04	0.1111E-05	0.8324E-04	44.90
45.00	0.1325E-42	0.5676E-08	0.8534E-06	0.8178E-04	0.8264E-04	0.1090E-05	0.8288E-04	45.00
45.10	0.1074E-42	0.5514E-08	0.8367E-06	0.8144E-04	0.8229E-04	0.1068E-05	0.8252E-04	45.10
45.20	0.8708E-43	0.5356E-08	0.8203E-06	0.8111E-04	0.8193E-04	0.1047E-05	0.8216E-04	45.20
45.30	0.7059E-43	0.5203E-08	0.8042E-06	0.8077E-04	0.8158E-04	0.1027E-05	0.8180E-04	45.30
45.40	0.5722E-43	0.5055E-08	0.7884E-06	0.8043E-04	0.8123E-04	0.1007E-05	0.8144E-04	45.40
45.50	0.4638E-43	0.4910E-08	0.7730E-06	0.8010E-04	0.8088E-04	0.9870E-06	0.8109E-04	45.50
45.60	0.3759E-43	0.4770E-08	0.7578E-06	0.7976E-04	0.8053E-04	0.9676E-06	0.8074E-04	45.60
45.70	0.3047E-43	0.4633E-08	0.7430E-06	0.7943E-04	0.8018E-04	0.9487E-06	0.8039E-04	45.70
45.80	0.2470E-43	0.4501E-08	0.7284E-06	0.7910E-04	0.7984E-04	0.9301E-06	0.8004E-04	45.80
45.90	0.2002E-43	0.4372E-08	0.7141E-06	0.7877E-04	0.7949E-04	0.9118E-06	0.7969E-04	45.90
46.00	0.1623E-43	0.4247E-08	0.7001E-06	0.7845E-04	0.7915E-04	0.8940E-06	0.7934E-04	46.00
46.10	0.1316E-43	0.4126E-08	0.6864E-06	0.7812E-04	0.7881E-04	0.8764E-06	0.7900E-04	46.10
46.20	0.1066E-43	0.4008E-08	0.6729E-06	0.7780E-04	0.7847E-04	0.8593E-06	0.7866E-04	46.20
46.30	0.8644E-44	0.3893E-08	0.6597E-06	0.7747E-04	0.7814E-04	0.8424E-06	0.7832E-04	46.30
46.40	0.7007E-44	0.3782E-08	0.6468E-06	0.7715E-04	0.7780E-04	0.8259E-06	0.7798E-04	46.40
46.50	0.5679E-44	0.3674E-08	0.6341E-06	0.7683E-04	0.7747E-04	0.8097E-06	0.7764E-04	46.50
46.60	0.4604E-44	0.3569E-08	0.6217E-06	0.7651E-04	0.7713E-04	0.7938E-06	0.7731E-04	46.60
46.70	0.3732E-44	0.3467E-08	0.6095E-06	0.7619E-04	0.7680E-04	0.7783E-06	0.7697E-04	46.70
46.80	0.3025E-44	0.3368E-08	0.5975E-06	0.7587E-04	0.7648E-04	0.7630E-06	0.7664E-04	46.80
46.90	0.2452E-44	0.3272E-08	0.5858E-06	0.7556E-04	0.7615E-04	0.7480E-06	0.7631E-04	46.90
47.00	0.1987E-44	0.3178E-08	0.5743E-06	0.7525E-04	0.7582E-04	0.7334E-06	0.7598E-04	47.00
47.10	0.1611E-44	0.3087E-08	0.5631E-06	0.7493E-04	0.7550E-04	0.7190E-06	0.7565E-04	47.10
47.20	0.1306E-44	0.2999E-08	0.5520E-06	0.7462E-04	0.7518E-04	0.7049E-06	0.7533E-04	47.20
47.30	0.1058E-44	0.2913E-08	0.5412E-06	0.7431E-04	0.7485E-04	0.6911E-06	0.7500E-04	47.30
47.40	0.8580E-45	0.2830E-08	0.5306E-06	0.7400E-04	0.7453E-04	0.6775E-06	0.7468E-04	47.40
47.50	0.6955E-45	0.2749E-08	0.5202E-06	0.7369E-04	0.7422E-04	0.6643E-06	0.7436E-04	47.50
47.60	0.5637E-45	0.2671E-08	0.5100E-06	0.7339E-04	0.7390E-04	0.6512E-06	0.7404E-04	47.60
47.70	0.4570E-45	0.2594E-08	0.5000E-06	0.7308E-04	0.7358E-04	0.6385E-06	0.7372E-04	47.70
47.80	0.3704E-45	0.2520E-08	0.4902E-06	0.7278E-04	0.7327E-04	0.6259E-06	0.7341E-04	47.80
47.90	0.3002E-45	0.2448E-08	0.4806E-06	0.7248E-04	0.7296E-04	0.6137E-06	0.7309E-04	47.90
48.00	0.2434E-45	0.2378E-08	0.4712E-06	0.7217E-04	0.7265E-04	0.6016E-06	0.7278E-04	48.00

SSES DATA GENERATED CURVES

T (HRS)	I(T)	GR(T)	SR(T) OBTUSE LINES	SS(T)	SUM(T) OBTUSE LINES	SR(T) NO OBTUSE LINES	SUM(T) NO OBTUSE LINES	T (HRS)
48.10	0.1973E-45	0.2310E-08	0.4619E-06	0.7187E-04	0.7234E-04	0.5898E-06	0.7247E-04	48.10
48.20	0.1599E-45	0.2244E-08	0.4529E-06	0.7158E-04	0.7203E-04	0.5783E-06	0.7216E-04	48.20
48.30	0.1296E-45	0.2180E-08	0.4440E-06	0.7128E-04	0.7172E-04	0.5669E-06	0.7185E-04	48.30
48.40	0.1051E-45	0.2118E-08	0.4353E-06	0.7098E-04	0.7142E-04	0.5558E-06	0.7154E-04	48.40
48.50	0.8516E-46	0.2057E-08	0.4268E-06	0.7069E-04	0.7112E-04	0.5449E-06	0.7123E-04	48.50
48.60	0.6903E-46	0.1998E-08	0.4184E-06	0.7039E-04	0.7081E-04	0.5343E-06	0.7093E-04	48.60
48.70	0.5596E-46	0.1941E-08	0.4102E-06	0.7010E-04	0.7051E-04	0.5238E-06	0.7063E-04	48.70
48.80	0.4536E-46	0.1886E-08	0.4022E-06	0.6981E-04	0.7021E-04	0.5135E-06	0.7032E-04	48.80
48.90	0.3677E-46	0.1832E-08	0.3943E-06	0.6952E-04	0.6991E-04	0.5034E-06	0.7002E-04	48.90
49.00	0.2980E-46	0.1779E-08	0.3865E-06	0.6923E-04	0.6962E-04	0.4936E-06	0.6972E-04	49.00
49.10	0.2416E-46	0.1729E-08	0.3790E-06	0.6894E-04	0.6932E-04	0.4839E-06	0.6943E-04	49.10
49.20	0.1958E-46	0.1679E-08	0.3715E-06	0.6865E-04	0.6903E-04	0.4744E-06	0.6913E-04	49.20
49.30	0.1587E-46	0.1631E-08	0.3642E-06	0.6837E-04	0.6873E-04	0.4651E-06	0.6884E-04	49.30
49.40	0.1287E-46	0.1585E-08	0.3571E-06	0.6808E-04	0.6844E-04	0.4560E-06	0.6854E-04	49.40
49.50	0.1043E-46	0.1539E-08	0.3501E-06	0.6780E-04	0.6815E-04	0.4470E-06	0.6825E-04	49.50
49.60	0.8454E-47	0.1495E-08	0.3432E-06	0.6752E-04	0.6786E-04	0.4383E-06	0.6796E-04	49.60
49.70	0.6852E-47	0.1453E-08	0.3365E-06	0.6724E-04	0.6758E-04	0.4297E-06	0.6767E-04	49.70
49.80	0.5554E-47	0.1411E-08	0.3299E-06	0.6696E-04	0.6729E-04	0.4213E-06	0.6738E-04	49.80
49.90	0.4502E-47	0.1371E-08	0.3234E-06	0.6668E-04	0.6701E-04	0.4130E-06	0.6710E-04	49.90
50.00	0.3649E-47	0.1331E-08	0.3171E-06	0.6640E-04	0.6672E-04	0.4049E-06	0.6681E-04	50.00
50.10	0.2958E-47	0.1293E-08	0.3109E-06	0.6613E-04	0.6644E-04	0.3970E-06	0.6653E-04	50.10
50.20	0.2398E-47	0.1256E-08	0.3048E-06	0.6585E-04	0.6616E-04	0.3892E-06	0.6624E-04	50.20
50.30	0.1944E-47	0.1221E-08	0.2988E-06	0.6558E-04	0.6588E-04	0.3816E-06	0.6596E-04	50.30
50.40	0.1576E-47	0.1186E-08	0.2930E-06	0.6531E-04	0.6560E-04	0.3741E-06	0.6568E-04	50.40
50.50	0.1277E-47	0.1152E-08	0.2872E-06	0.6503E-04	0.6532E-04	0.3667E-06	0.6540E-04	50.50
50.60	0.1035E-47	0.1119E-08	0.2816E-06	0.6476E-04	0.6505E-04	0.3596E-06	0.6512E-04	50.60
50.70	0.8391E-48	0.1087E-08	0.2761E-06	0.6449E-04	0.6477E-04	0.3525E-06	0.6485E-04	50.70
50.80	0.6802E-48	0.1056E-08	0.2707E-06	0.6423E-04	0.6450E-04	0.3456E-C6	0.6457E-04	50.80
50.90	0.5513E-48	0.1026E-08	0.2653E-06	0.6396E-04	0.6423E-04	0.3388E-06	0.6430E-04	50.90
51.00	0.4469E-48	0.9963E-09	0.2601E-06	0.6369E-04	0.6395E-04	0.3322E-06	0.6403E-04	51.00
51.10	0.3622E-48	0.9678E-09	0.2550E-06	0.6343E-04	0.6368E-04	0.3257E-06	0.6376E-04	51.10
51.20	0.2936E-48	0.9402E-09	0.2500E-06	0.6317E-04	0.6342E-04	0.3193E-06	0.6349E-04	51.20
51.30	0.2380E-48	0.9133E-09	0.2451E-06	0.6290E-04	0.6315E-04	0.3130E-06	0.6322E-04	51.30
51.40	0.1929E-48	0.8872E-09	0.2403E-06	0.6264E-04	0.6288E-04	0.3069E-06	0.6295E-04	51.40
51.50	0.1564E-48	0.8618E-09	0.2356E-06	0.6238E-04	0.6262E-04	0.3009E-06	0.6268E-04	51.50
51.60	0.1268E-48	0.8372E-09	0.2310E-06	0.6212E-04	0.6235E-04	0.2950E-06	0.6242E-04	51.60
51.70	0.1028E-48	0.8133E-09	0.2265E-06	0.6186E-04	0.6209E-04	0.2892E-06	0.6215E-04	51.70
51.80	0.8329E-49	0.7900E-09	0.2220E-06	0.6161E-04	0.6183E-04	0.2835E-06	0.6189E-04	51.80
51.90	0.6752E-49	0.7674E-09	0.2177E-06	0.6135E-04	0.6157E-04	0.2780E-06	0.6163E-04	51.90
52.00	0.5473E-49	0.7455E-09	0.2134E-06	0.6109E-04	0.6131E-04	0.2725E-06	0.6137E-04	52.00
52.10	0.4436E-49	0.7242E-09	0.2092E-06	0.6084E-04	0.6105E-04	0.2672E-06	0.6111E-04	52.10
52.20	0.3596E-49	0.7035E-09	0.2051E-06	0.6059E-04	0.6079E-04	0.2619E-06	0.6085E-04	52.20
52.30	0.2915E-49	0.6834E-09	0.2011E-06	0.6034E-04	0.6054E-04	0.2568E-06	0.6059E-04	52.30
52.40	0.2363E-49	0.6638E-09	0.1972E-06	0.6008E-04	0.6028E-04	0.2518E-06	0.6034E-04	52.40
52.50	0.1915E-49	0.6449E-09	0.1933E-06	0.5983E-04	0.6003E-04	0.2468E-06	0.6008E-04	52.50
52.60	0.1552E-49	0.6264E-09	0.1895E-06	0.5959E-04	0.5978E-04	0.2420E-06	0.5983E-04	52.60
52.70	0.1258E-49	0.6085E-09	0.1858E-06	0.5934E-04	0.5952E-04	0.2372E-06	0.5958E-04	52.70
52.80	0.1020E-49	0.5911E-09	0.1821E-06	0.5909E-04	0.5927E-04	0.2326E-06	0.5932E-04	52.80
52.90	0.8268E-50	0.5742E-09	0.1786E-06	0.5885E-04	0.5902E-04	0.2280E-06	0.5907E-04	52.90
53.00	0.6702E-50	0.5578E-09	0.1751E-06	0.5860E-04	0.5878E-04	0.2236E-06	0.5883E-04	53.00
53.10	0.5432E-50	0.5419E-09	0.1716E-06	0.5836E-04	0.5853E-04	0.2192E-06	0.5858E-04	53.10
53.20	0.4403E-50	0.5264E-09	0.1683E-06	0.5811E-04	0.582CE-04	0.2149E-06	0.5833E-04	53.20
53.30	0.3569E-50	0.5113E-09	0.1650E-06	0.5787E-04	0.5804E-04	0.2107E-06	0.5808E-04	53.30
53.40	0.2893E-50	0.4967E-09	0.1617E-06	0.5763E-04	0.5779E-04	0.2065E-06	0.5784E-04	53.40
53.50	0.2345E-50	0.4825E-09	0.1586E-06	0.5739E-04	0.5755E-04	0.2025E-06	0.5760E-04	53.50
53.60	0.1901E-50	0.4687E-09	0.1555E-06	0.5715E-04	0.5731E-04	0.1985E-06	0.5735E-04	53.60
53.70	0.1541E-50	0.4553E-09	0.1524E-06	0.5692E-04	0.5707E-04	0.1946E-06	0.5711E-04	53.70
53.80	0.1249E-50	0.4423E-09	0.1494E-06	0.5668E-04	0.5683E-04	0.1908E-06	0.5687E-04	53.80
53.90	0.1012E-50	0.4297E-09	0.1465E-06	0.5644E-04	0.5659E-04	0.1871E-06	0.5663E-04	53.90
54.00	0.8207E-51	0.4174E-09	0.1436E-06	0.5621E-04	0.5635E-04	0.1834E-06	0.5639E-04	54.00

T (HRS)	I(T)	GR(T)	SR(T) OBTUSE LINES	SS(T)	SUM(T) OBTUSE LINES	SR(T) NO OBTUSE LINES	SUM(T) NO OBTUSE LINES	T (HRS)
54.10	0.6652E-51	0.4055E-09	0.1408E-06	0.5598E-04	0.5612E-04	0.1798E-06	0.5616E-04	54.10
54.20	0.5392E-51	0.3939E-09	0.1381E-06	0.5574E-04	0.5588E-04	0.1763E-06	0.5592E-04	54.20
54.30	0.4371E-51	0.3826E-09	0.1353E-06	0.5551E-04	0.5565E-04	0.1728E-06	0.5568E-04	54.30
54.40	0.3543E-51	0.3717E-09	0.1327E-06	0.5528E-04	0.5541E-04	0.1694E-06	0.5545E-04	54.40
54.50	0.2872E-51	0.3611E-09	0.1301E-06	0.5505E-04	0.5518E-04	0.1661E-06	0.5522E-04	54.50
54.60	0.2328E-51	0.3507E-09	0.1275E-06	0.5482E-04	0.5495E-04	0.1629E-06	0.5498E-04	54.60
54.70	0.1887E-51	0.3407E-09	0.1250E-06	0.5459E-04	0.5472E-04	0.1597E-06	0.5475E-04	54.70
54.80	0.1529E-51	0.3310E-09	0.1226E-06	0.5437E-04	0.5449E-04	0.1565E-06	0.5452E-04	54.80
54.90	0.1240E-51	0.3215E-09	0.1202E-06	0.5414E-04	0.5426E-04	0.1535E-06	0.5429E-04	54.90
55.00	0.1005E-51	0.3123E-09	0.1178E-06	0.5392E-04	0.5403E-04	0.1505E-06	0.5407E-04	55.00
55.10	0.8146E-52	0.3034E-09	0.1155E-06	0.5369E-04	0.5381E-04	0.1475E-06	0.5384E-04	55.10
55.20	0.6603E-52	0.2947E-09	0.1133E-06	0.5347E-04	0.5358E-04	0.1446E-06	0.5361E-04	55.20
55.30	0.5352E-52	0.2863E-09	0.1110E-06	0.5325E-04	0.5336E-04	0.1418E-06	0.5339E-04	55.30
55.40	0.4338E-52	0.2781E-09	0.1089E-06	0.5302E-04	0.5313E-04	0.1390E-06	0.5316E-04	55.40
55.50	0.3517E-52	0.2702E-09	0.1067E-06	0.5280E-04	0.5291E-04	0.1363E-06	0.5294E-04	55.50
55.60	0.2851E-52	0.2624E-09	0.1046E-06	0.5258E-04	0.5269E-04	0.1336E-06	0.5272E-04	55.60
55.70	0.2311E-52	0.2549E-09	0.1026E-06	0.5237E-04	0.5247E-04	0.1310E-06	0.5250E-04	55.70
55.80	0.1873E-52	0.2477E-09	0.1006E-06	0.5215E-04	0.5225E-04	0.1284E-06	0.5228E-04	55.80
55.90	0.1518E-52	0.2406E-09	0.9860E-07	0.5193E-04	0.5203E-04	0.1259E-06	0.5206E-04	55.90
56.00	0.1231E-52	0.2337E-09	0.9666E-07	0.5172E-04	0.5181E-04	0.1234E-06	0.5184E-04	56.00
56.10	0.9975E-53	0.2270E-09	0.9477E-07	0.5150E-04	0.5160E-04	0.1210E-06	0.5162E-04	56.10
56.20	0.8086E-53	0.2205E-09	0.9291E-07	0.5129E-04	0.5138E-04	0.1186E-06	0.5140E-04	56.20
56.30	0.6554E-53	0.2142E-09	0.9109E-07	0.5107E-04	0.5116E-04	0.1163E-06	0.5119E-04	56.30
56.40	0.5313E-53	0.2081E-09	0.8930E-07	0.5086E-04	0.5095E-04	0.1140E-06	0.5097E-04	56.40
56.50	0.4306E-53	0.2022E-09	0.8755E-07	0.5065E-04	0.5074E-04	0.1118E-06	0.5076E-04	56.50
56.60	0.3491E-53	0.1964E-09	0.8584E-07	0.5044E-04	0.5052E-04	0.1096E-06	0.5055E-04	56.60
56.70	0.2830E-53	0.1908E-09	0.8415E-07	0.5023E-04	0.5031E-04	0.1075E-06	0.5034E-04	56.70
56.80	0.2294E-53	0.1853E-09	0.8250E-07	0.5002E-04	0.5010E-04	0.1053E-06	0.5013E-04	56.80
56.90	0.1859E-53	0.1800E-09	0.8089E-07	0.4981E-04	0.4989E-04	0.1033E-06	0.4992E-04	56.90
57.00	0.1507E-53	0.1749E-09	0.7930E-07	0.4960E-04	0.4968E-04	0.1013E-06	0.4971E-04	57.00
57.10	0.1222E-53	0.1699E-09	0.7774E-07	0.4940E-04	0.4948E-04	0.9927E-07	0.4950E-04	57.10
57.20	0.9902E-54	0.1650E-09	0.7622E-07	0.4919E-04	0.4927E-04	0.9733E-07	0.4929E-04	57.20
57.30	0.8026E-54	0.1603E-09	0.7473E-07	0.4899E-04	0.4906E-04	0.9542E-07	0.4908E-04	57.30
57.40	0.6506E-54	0.1557E-09	0.7326E-07	0.4878E-04	0.4886E-04	0.9355E-07	0.4888E-04	57.40
57.50	0.5273E-54	0.1513E-09	0.7182E-07	0.4858E-04	0.4865E-04	0.9171E-07	0.4867E-04	57.50
57.60	0.4275E-54	0.1469E-09	0.7042E-07	0.4838E-04	0.4845E-04	0.8991E-07	0.4847E-04	57.60
57.70	0.3465E-54	0.1427E-09	0.6904E-07	0.4818E-04	0.4825E-04	0.8815E-07	0.4827E-04	57.70
57.80	0.2809E-54	0.1387E-09	0.6768E-07	0.4798E-04	0.4805E-04	0.8642E-07	0.4806E-04	57.80
57.90	0.2277E-54	0.1347E-09	0.6636E-07	0.4778E-04	0.4785E-04	0.8473E-07	0.4786E-04	57.90
58.00	0.1845E-54	0.1308E-09	0.6505E-07	0.4758E-04	0.4765E-04	0.8307E-07	0.4766E-04	58.00
58.10	0.1496E-54	0.1271E-09	0.6378E-07	0.4738E-04	0.4745E-04	0.8144E-07	0.4746E-04	58.10
58.20	0.1213E-54	0.1235E-09	0.6253E-07	0.4719E-04	0.4725E-04	0.7984E-07	0.4727E-04	58.20
58.30	0.9828E-55	0.1199E-09	0.6130E-07	0.4699E-04	0.4705E-04	0.7828E-07	0.4707E-04	58.30
58.40	0.7967E-55	0.1165E-09	0.6010E-07	0.4679E-04	0.4685E-04	0.7674E-07	0.4687E-04	58.40
58.50	0.6458E-55	0.1132E-09	0.5892E-07	0.4660E-04	0.4666E-04	0.7524E-07	0.4667E-04	58.50
58.60	0.5234E-55	0.1100E-09	0.5777E-07	0.4641E-04	0.4646E-04	0.7376E-07	0.4648E-04	58.60
58.70	0.4243E-55	0.1068E-09	0.5664E-07	0.4621E-04	0.4627E-04	0.7232E-07	0.4628E-04	58.70
58.80	0.3439E-55	0.1038E-09	0.5552E-07	0.4602E-04	0.4608E-04	0.7090E-07	0.4609E-04	58.80
58.90	0.2788E-55	0.1008E-09	0.5444E-07	0.4583E-04	0.4588E-04	0.6951E-07	0.4590E-04	58.90
59.00	0.2260E-55	0.9791E-10	0.5337E-07	0.4564E-04	0.4569E-04	0.6815E-07	0.4571E-04	59.00
59.10	0.1832E-55	0.9511E-10	0.5232E-07	0.4545E-04	0.4550E-04	0.6681E-07	0.4552E-04	59.10
59.20	0.1485E-55	0.9239E-10	0.5130E-07	0.4526E-04	0.4531E-04	0.6550E-07	0.4533E-04	59.20
59.30	0.1204E-55	0.8975E-10	0.5029E-07	0.4507E-04	0.4512E-04	0.6422E-07	0.4514E-04	59.30
59.40	0.9756E-56	0.8719E-10	0.4931E-07	0.4488E-04	0.4493E-04	0.6296E-07	0.4495E-04	59.40
59.50	0.7908E-56	0.8469E-10	0.4834E-07	0.4470E-04	0.4475E-04	0.6172E-07	0.4476E-04	59.50
59.60	0.6410E-56	0.8227E-10	0.4739E-07	0.4451E-04	0.4456E-04	0.6051E-07	0.4457E-04	59.60
59.70	0.5196E-56	0.7992E-10	0.4646E-07	0.4433E-04	0.4437E-04	0.5933E-07	0.4439E-04	59.70
59.80	0.4212E-56	0.7764E-10	0.4555E-07	0.4414E-04	0.4419E-04	0.5816E-07	0.4420E-04	59.80
59.90	0.3414E-56	0.7542E-10	0.4466E-07	0.4396E-04	0.4400E-04	0.5702E-07	0.4402E-04	59.90
60.00	0.2767E-56	0.7326E-10	0.4378E-07	0.4378E-04	0.4382E-04	0.5591E-07	0.4383E-04	60.00

******************************** SSES DATA GENERATED CURVES ********************************

T (HRS)	I(T)	GR(T)	SR(T) OBTUSE LINES	SS(T)	SUM(T) OBTUSE LINES	SR(T) NO OBTUSE LINES	SUM(T) NO OBTUSE LINES	T (HRS)
60.10	0.2243E-56	0.7117E-10	0.4292E-07	0.4359E-04	0.4364E-C4	0.5481E-07	0.4365E-04	60.10
60.20	0.1818E-56	0.6913E-10	0.4208E-07	0.4341E-04	0.4345E-04	0.5373E-07	0.4347E-04	60.20
60.30	0.1474E-56	0.6716E-10	0.4126E-07	0.4323E-04	0.4327E-04	0.5268E-07	0.4328E-04	60.30
60.40	0.1195E-56	0.6524E-10	0.4045E-07	0.4305E-04	0.4309E-04	0.5165E-07	0.4310E-04	60.40
60.50	0.9684E-57	0.6337E-10	0.3966E-07	0.4287E-04	0.4291E-04	0.5064E-07	0.4292E-04	60.50
60.60	0.7849E-57	0.6156E-10	0.3888E-07	0.4270E-04	0.4273E-04	0.4964E-07	0.4274E-04	60.60
60.70	0.6363E-57	0.5980E-10	0.3812E-07	0.4252E-04	0.4256E-04	0.4867E-07	0.4257E-04	60.70
60.80	0.5157E-57	0.5809E-10	0.3737E-07	0.4234E-04	0.4238E-04	0.4772E-07	0.4239E-04	60.80
60.90	0.4181E-57	0.5643E-10	0.3664E-07	0.4216E-04	0.4220E-04	0.4678E-07	0.4221E-04	60.90
61.00	0.3389E-57	0.5482E-10	0.3592E-07	0.4199E-04	0.4203E-04	0.4586E-07	0.4204E-04	61.00
61.10	0.2747E-57	0.5325E-10	0.3521E-G7	0.4181E-04	0.4185E-04	0.4496E-07	0.4186E-04	61.10
61.20	0.2227E-57	0.5173E-10	G.3452E-07	0.4164E-04	0.4168E-04	0.4408E-07	0.4169E-04	61.20
61.30	0.1805E-57	0.5025E-10	0.3385E-07	0.4147E-04	0.4150E-04	0.4322E-07	0.4151E-04	61.30
61.40	0.1463E-57	0.4882E-10	0.3318E-07	0.4130E-04	0.4133E-04	0.4237E-07	0.4134E-04	61.40
61.50	0.1186E-57	0.4742E-10	0.3253E-07	0.4112E-04	0.4116E-04	0.4154E-07	0.4117E-04	61.50
61.60	0.9612E-58	0.4606E-10	0.3189E-07	0.4095E-04	0.4098E-04	0.4073E-07	0.4099E-04	61.60
61.70	0.7791E-58	0.4475E-10	0.3127E-07	0.4078E-04	0.4081E-04	0.3993E-07	0.4082E-04	61.70
61.80	0.6316E-58	0.4347E-10	0.3066E-07	0.4061E-04	0.4064E-04	0.3914E-07	0.4065E-04	61.80
61.90	0.5119E-58	0.4223E-10	0.3006E-07	0.4044E-04	0.4047E-04	0.3838E-07	0.4048E-04	61.90
62.00	0.4150E-58	0.4102E-10	0.2947E-07	0.4028E-04	0.4031E-04	0.3762E-07	0.4031E-04	62.00
62.10	0.3364E-58	0.3985E-10	0.2889E-07	0.4011E-04	0.4014E-04	0.3689E-07	0.4015E-04	62.10
62.20	0.2727E-58	0.3871E-10	0.2832E-07	0.3994E-04	0.3997E-04	0.3616E-07	0.3998E-04	62.20
62.30	0.2210E-58	0.3760E-10	0.2777E-07	0.3978E-04	0.3980E-04	0.3545E-07	0.3981E-04	62.30
62.40	0.1791E-58	0.3653E-10	0.2722E-07	0.3961E-04	0.3964E-04	0.3476E-07	0.3964E-04	62.40
62.50	0.1452E-58	0.3548E-10	0.2669E-07	0.3945E-04	0.3947E-04	0.3408E-07	0.3948E-04	62.50
62.60	0.1177E-58	0.3447E-10	0.2617E-07	0.3928E-04	0.3931E-04	0.3341E-07	0.3931E-04	62.60
62.70	0.9541E-59	0.3348E-10	0.2565E-07	0.3912E-04	0.3914E-04	0.3276E-07	0.3915E-04	62.70
62.80	0.7734E-59	0.3253E-10	0.2515E-07	0.3896E-04	0.3898E-04	0.3211E-07	0.3899E-04	62.80
62.90	0.6269E-59	0.3160E-10	0.2466E-07	0.3879E-04	0.3882E-04	0.3148E-07	0.3882E-04	62.90
63.00	0.5081E-59	0.3069E-10	0.2417E-07	0.3863E-04	0.3866E-04	0.3087E-07	0.3866E-04	63.00
63.10	0.4119E-59	0.2982E-10	0.2370E-07	0.3847E-04	0.3850E-04	0.3026E-07	0.3850E-04	53.10
63.20	0.3339E-59	0.2896E-10	0.2323E-07	0.3831E-04	0.3833E-04	0.2967E-07	0.3834E-04	63.20
63.30	0.2706E-59	0.2814E-10	0.2278E-07	0.3815E-04	0.3818E-04	0.2909E-07	0.3818E-04	63.30
63.40	0.2194E-59	0.2733E-10	0.2233E-07	0.3799E-04	0.3802E-04	0.2852E-07	0.3802E-04	63.40
63.50	0.1778E-59	0.2655E-10	0.2189E-07	0.3784E-04	0.3786E-04	0.2796E-07	0.3786E-04	63.50
63.60	0.1441E-59	0.2579E-10	0.2147E-07	0.3768E-04	0.3770E-04	0.2741E-07	0.3771E-04	63.60
63.70	0.1168E-59	0.2505E-10	0.2104E-07	0.3752E-04	0.3754E-04	0.2687E-07	0.3755E-04	63.70
63.80	0.9471E-60	0.2434E-10	0.2063E-07	0.3737E-04	0.3739E-04	0.2634E-07	0.3739E-04	63.80
63.90	0.7677E-60	0.2364E-10	0.2023E-07	0.3721E-04	0.3723E-04	0.2583E-07	0.3724E-04	63.90
64.00	0.6223E-60	0.2297E-10	0.1983E-07	0.3706E-04	0.3708E-04	0.2532E-07	0.3708E-04	64.00
64.10	0.5044E-60	0.2231E-10	0.1944E-07	0.3690E-04	0.3692E-04	0.2483E-07	0.3693E-04	64.10
64.20	0.4089E-60	0.2167E-10	0.1906E-07	0.3675E-04	0.3677E-04	0.2434E-07	0.3677E-04	64.20
64.30	0.3314E-60	0.2105E-10	0.1869E-07	0.3660E-04	0.3661E-04	0.2386E-07	0.3662E-04	64.30
64.40	0.2686E-60	0.2045E-10	0.1832E-07	0.3644E-04	0.3646E-04	0.2339E-07	0.3647E-04	64.40
64.50	0.2178E-60	0.1987E-10	0.1796E-07	0.3629E-04	0.3631E-04	0.2293E-07	0.3631E-04	64.50
64.60	0.1765E-60	0.1930E-10	0.1761E-07	0.3614E-04	0.3616E-04	0.2249E-07	0.3616E-04	64.60
64.70	0.1431E-60	0.1875E-10	0.1726E-07	0.3599E-04	0.3601E-04	0.2204E-07	0.3601E-04	64.70
64.80	0.1160E-60	0.1821E-10	0.1693E-07	0.3584E-04	0.3586E-04	0.2161E-07	0.3586E-04	64.80
64.90	0.9400E-61	0.1769E-10	0.1659E-07	0.3569E-04	0.3571E-04	0.2119E-07	0.3571E-04	64.90
65.00	0.7620E-61	0.1719E-10	0.1627E-07	0.3554E-04	0.3556E-04	0.2077E-07	0.3556E-04	65.00
65.10	0.6177E-61	0.1669E-10	0.1595E-07	0.3540E-04	0.3541E-04	0.2037E-07	0.3542E-04	65.10
65.20	0.5007E-61	0.1622E-10	0.1564E-07	0.3525E-04	0.3526E-04	0.1997E-07	0.3527E-04	65.20
65.30	0.4058E-61	0.1575E-10	0.1533E-07	0.3510E-04	0.3512E-04	0.1958E-07	0.3512E-04	65.30
65.40	0.3290E-61	0.1530E-10	0.1503E-07	0.3496E-04	0.3497E-04	0.1919E-07	0.3497E-04	65.40
65.50	0.2667E-61	0.1487E-10	0.1474E-07	0.3481E-04	0.3483E-04	0.1882E-07	0.3483E-04	65.50
65.60	0.2161E-61	0.1444E-10	0.1445E-07	0.3467E-04	0.3468E-04	0.1845E-07	0.3468E-04	65.60
65.70	0.1752E-61	0.1403E-10	0.1416E-07	0.3452E-04	0.3454E-04	0.1808E-07	0.3454E-04	65.70
65.80	0.1420E-61	0.1363E-10	0.1389E-07	0.3438E-04	0.3439E-04	0.1773E-07	0.3440E-04	65.80
65.90	0.1151E-61	0.1324E-10	0.1361E-07	0.3424E-04	0.3425E-04	0.1738E-07	0.3425E-04	65.90
66.00	0.9331E-62	0.1286E-10	0.1335E-07	0.3409E-04	0.3411E-04	0.1704E-07	0.3411E-04	66.00

********************************* SSES DATA GENERATED CURVES *********************************

T (HRS)	I(T)	GR(T)	SR(T) OBTUSE LINES	SS(T)	SUM(T) OBTUSE LINES	SR(T) NO OBTUSE LINES	SUM(T) NO OBTUSE LINES	T (HRS)
66.10	0.7564E-62	0.1249E-10	0.1308E-07	0.3395E-04	0.3386E-04	0.1671E-07	0.3397E-04	66.10
66.20	0.6131E-62	0.1213E-10	0.1283E-07	0.3381E-04	0.3382E-04	0.1638E-07	0.3383E-04	66.20
66.30	0.4970E-62	0.1179E-10	0.1258E-07	0.3367E-04	0.3368E-04	0.1606E-07	0.3369E-04	66.30
66.40	0.4028E-62	0.1145E-10	0.1233E-07	0.3353E-04	0.3354E-04	0.1574E-07	0.3354E-04	66.40
66.50	0.3265E-62	0.1112E-10	0.1209E-07	0.3339E-04	0.3340E-04	0.1544E-07	0.3341E-04	66.50
66.60	0.2647E-62	0.1081E-10	0.1185E-07	0.3325E-04	0.3326E-04	0.1513E-07	0.3327E-04	66.60
66.70	0.2145E-62	0.1050E-10	0.1162E-07	0.3311E-04	0.3312E-04	0.1484E-07	0.3313E-04	66.70
66.80	0.1739E-62	0.1020E-10	0.1139E-07	0.3298E-04	0.3299E-04	0.1455E-07	0.3299E-04	66.80
66.90	0.1410E-62	0.9905E-11	0.1117E-07	0.3284E-04	0.3285E-04	0.1426E-07	0.3285E-04	66.90
67.00	0.1143E-62	0.9622E-11	0.1095E-07	0.3270E-04	0.3271E-04	0.1398E-07	0.3272E-04	67.00
67.10	0.9262E-63	0.9347E-11	0.1073E-07	0.3257E-04	0.3258E-04	0.1371E-07	0.3258E-04	67.10
67.20	0.7506E-63	0.9080E-11	0.1052E-07	0.3243E-04	0.3244E-04	0.1344E-07	0.3244E-04	67.20
67.30	0.6086E-63	0.8820E-11	0.1032E-07	0.3230E-04	0.3231E-04	0.1317E-07	0.3231E-04	67.30
67.40	0.4933E-63	0.8568E-11	0.1012E-07	0.3216E-04	0.3217E-04	0.1292E-07	0.3217E-04	67.40
67.50	0.3999E-63	0.8323E-11	0.9917E-08	0.3203E-04	0.3204E-04	0.1266E-07	0.3204E-04	67.50
67.60	0.3241E-63	0.8085E-11	0.9722E-08	0.3189E-04	0.3190E-04	0.1241E-07	0.3191E-04	67.60
67.70	0.2627E-63	0.7854E-11	0.9532E-08	0.3176E-04	0.3177E-04	0.1217E-07	0.3177E-04	67.70
67.80	0.2130E-63	0.7630E-11	0.9345E-08	0.3163E-04	0.3164E-04	0.1193E-07	0.3164E-04	67.80
67.90	0.1726E-63	0.7412E-11	0.9162E-08	0.3150E-04	0.3151E-04	0.1170E-07	0.3151E-04	67.90
68.00	0.1399E-63	0.7200E-11	0.8982E-08	0.3137E-04	0.3138E-04	0.1147E-07	0.3138E-04	68.00
68.10	0.1134E-63	0.6994E-11	0.8806E-08	0.3124E-04	0.3125E-04	0.1124E-07	0.3125E-04	68.10
68.20	0.9194E-64	0.6794E-11	0.8633E-08	0.3111E-04	0.3112E-04	0.1102E-07	0.3112E-04	68.20
68.30	0.7452E-64	0.6600E-11	0.8464E-08	0.3098E-04	0.3099E-04	0.1081E-07	0.3099E-04	68.30
68.40	0.6041E-64	0.6411E-11	0.8298E-08	0.3085E-04	0.3086E-04	0.1060E-07	0.3086E-04	68.40
68.50	0.4897E-64	0.6228E-11	0.8135E-08	0.3072E-04	0.3073E-04	0.1039E-07	0.3073E-04	68.50
68.60	0.3969E-64	0.6050E-11	0.7976E-08	0.3059E-04	0.3060E-04	0.1018E-07	0.3060E-04	68.60
68.70	0.3217E-64	0.5877E-11	0.7820E-08	0.3047E-04	0.3047E-04	0.9985E-08	0.3048E-04	68.70
68.80	0.2608E-64	0.5709E-11	0.7666E-08	0.3034E-04	0.3035E-04	0.9789E-08	0.3035E-04	68.80
68.90	0.2114E-64	0.5546E-11	0.7516E-08	0.3021E-04	0.3022E-04	0.9597E-08	0.3022E-04	68.90
69.00	0.1713E-64	0.5387E-11	0.7369E-08	0.3009E-04	0.3009E-04	0.9409E-08	0.3010E-04	69.00
69.10	0.1389E-64	0.5233E-11	0.7224E-08	0.2996E-04	0.2997E-04	0.9224E-08	0.2997E-04	69.10
69.20	0.1126E-64	0.5084E-11	0.7083E-08	0.2984E-04	0.2984E-04	0.9044E-08	0.2985E-04	69.20
69.30	0.9126E-65	0.4938E-11	0.6944E-08	0.2971E-04	0.2972E-04	0.8866E-08	0.2972E-04	69.30
69.40	0.7397E-65	0.4797E-11	0.6808E-08	0.2959E-04	0.2960E-04	0.8693E-08	0.2960E-04	69.40
69.50	0.5996E-65	0.4660E-11	0.6674E-08	0.2947E-04	0.2947E-04	0.8522E-08	0.2947E-04	69.50
69.60	0.4860E-65	0.4527E-11	0.6543E-08	0.2934E-04	0.2935E-04	0.8355E-08	0.2935E-04	69.60
69.70	0.3940E-65	0.4398E-11	0.6415E-08	0.2922E-04	0.2923E-04	0.8191E-08	0.2923E-04	69.70
69.80	0.3193E-65	0.4272E-11	0.6289E-08	0.2910E-04	0.2911E-04	0.8031E-08	0.2911E-04	69.80
69.90	0.2589E-65	0.4150E-11	0.6166E-08	0.2898E-04	0.2899E-04	0.7873E-08	0.2899E-04	69.90
70.00	0.2098E-65	0.4031E-11	0.6045E-08	0.2886E-04	0.2886E-04	0.7719E-08	0.2887E-04	70.00
70.10	0.1701E-65	0.3916E-11	0.5926E-08	0.2874E-04	0.2874E-04	0.7567E-08	0.2875E-04	70.10
70.20	0.1379E-65	0.3804E-11	0.5810E-08	0.2862E-04	0.2863E-04	0.7419E-08	0.2863E-04	70.20
70.30	0.1118E-65	0.3695E-11	0.5696E-08	0.2850E-04	0.2851E-04	0.7274E-08	0.2851E-04	70.30
70.40	0.9058E-66	0.3590E-11	0.5585E-08	0.2838E-04	0.2839E-04	0.7131E-08	0.2839E-04	70.40
70.50	0.7343E-66	0.3487E-11	0.5475E-08	0.2826E-04	0.2827E-04	0.6991E-08	0.2827E-04	70.50
70.60	0.5952E-66	0.3387E-11	0.5368E-08	0.2815E-04	0.2815E-04	0.6854E-08	0.2815E-04	70.60
70.70	0.4824E-66	0.3291E-11	0.5263E-08	0.2803E-04	0.2803E-04	0.6720E-08	0.2804E-04	70.70
70.80	0.3911E-66	0.3196E-11	0.5159E-08	0.2791E-04	0.2792E-04	0.6588E-08	0.2792E-04	70.80
70.90	0.3170E-66	0.3105E-11	0.5058E-08	0.2780E-04	0.2780E-04	0.6459E-08	0.2780E-04	70.90
71.00	0.2569E-66	0.3016E-11	0.4959E-08	0.2768E-04	0.2769E-04	0.6332E-08	0.2769E-04	71.00
71.10	0.2083E-66	0.2930E-11	0.4862E-08	0.2757E-04	0.2757E-04	0.6208E-08	0.2757E-04	71.10
71.20	0.1688E-66	0.2846E-11	0.4767E-08	0.2745E-04	0.2746E-04	0.6086E-08	0.2746E-04	71.20
71.30	0.1368E-66	0.2765E-11	0.4673E-08	0.2734E-04	0.2734E-04	0.5967E-08	0.2734E-04	71.30
71.40	0.1109E-66	0.2686E-11	0.4582E-08	0.2722E-04	0.2723E-04	0.5850E-08	0.2723E-04	71.40
71.50	0.8992E-67	0.2609E-11	0.4492E-08	0.2711E-04	0.2711E-04	0.5735E-08	0.2712E-04	71.50
71.60	0.7288E-67	0.2535E-11	0.4404E-08	0.2700E-04	0.2700E-04	0.5623E-08	0.2700E-04	71.60
71.70	0.5908E-67	0.2462E-11	0.4317E-08	0.2689E-04	0.2689E-04	0.5513E-08	0.2689E-04	71.70
71.80	0.4789E-67	0.2392E-11	0.4233E-08	0.2677E-04	0.2678E-04	0.5405E-08	0.2678E-04	71.80
71.90	0.3882E-67	0.2323E-11	0.4150E-08	0.2666E-04	0.2667E-04	0.5299E-08	0.2667E-04	71.90
72.00	0.3146E-67	0.2257E-11	0.4068E-08	0.2655E-04	0.2656E-04	0.5195E-08	0.2656E-04	72.00
T (HRS)	I(T)	GR(T)	SR(T) OBTUSE LINES	SS(T)	SUM(T) OBTUSE LINES	SR(T) NO OBTUSE LINES	SUM(T) NO OBTUSE LINES	T (HRS)

APPENDIX 8

PP&L DEFINITIONS

DEFINITIONS(1)

(PP&L USAGE)

BACK END ANALYSIS

CORE DAMAGE

CORE STABILIZATION

DEBRIS STABILIZATION

DEFENSE IN DEPTH

EMERGENCY PROCEDURES GUIDELINES

EMERGENCY OPERATING PROCEDURES

EVENT SEQUENCE FAULT TREE METHOD

Definitions(1)

(PP&L Usage)

Back End Analysis

The analysis of the specific nature of loss of containment integrity (venting through catastrophic), the details of the fission product transport processes, and the nature of off-site consequences of the fission product release.

Core Damage

Damage to the fuel elements involving mechanical rupture of the fuel clad or high temperature oxidation of the clad by steam. The threshold for damage is defined as a level at which total noble gas release exceeds some selected fraction of the total core inventory of noble gas, for example 1%.

Core Stabilization

This phrase refers to termination of a core damage progression sequence prior to reactor vessel failure with no further damage to fuel or the reactor vessel as a consequence of the core damage at the time the progression is terminated.

Debris Stabilization

This phrase refers to terminating the attack of core debris on the drywell floor or other containment components. Termination is usually considered to occur as a result of cooling the debris by means of water.

Defense-in-Depth

The PP&L criteria to assure a multiple layer of defenses against severe accident consequences, both in terms of equipment and procedures.

Emergency Procedures Guidelines (EPGs)

A set of generic procedures for response to an initiating event intended to bring the plant to stable, controllable conditions with minimum damage. These procedures were developed by the BWR Owners' Group.

Emergency Operating Procedures (EOPs)

The EOPs are derived from the EPGs for a specific plant and reflect the specific features of the plant design. The PP&L EOPs take the form of flow charts which reference supplemental procedures as necessary.

Event Sequence Fault Tree Method

An alternative to the Support State method for performing a PRA. The event tree top events require quantification by fault trees and involve Boolean algebraic combinations of successes and failures of prior top events.

DEFINITIONS(2)

FRONT END ANALYSIS

FRONT-LINE FUNCTION

INDIVIDUAL PLANT EVALUATION METHOD

INITIATING EVENT

LEVEL 1 PRA

LEVEL 2 PRA

LEVEL 3 PRA

LIMIT CYCLE OPERATION

LOSS OF CONTAINMENT INTEGRITY

Definitions(2)

Front-End Analysis

The analysis used to determine core damage frequency. As applied by PP&L this analysis is extended to define all forms of plant damage out to stable or uncontrolled plant conditions.

Front-Line Function

A function of plant systems which is required to avoid or limit plant damage as a result of an initiating event and independent equipment failures.

Individual Plant Evaluation Methods (IPEM)

A PRA methodology developed by IDCOR to allow a simplified evaluation of core damage frequency. The BWR methodology is based on the Support State approach. PP&L has enhanced the IDCOR IPEM to provide more detail in Support State and Plant Damage States.

Initiating Event

Any deviation in plant operation that requires the operation of front-line systems to restore the plant to an undamaged stable condition.

Level 1 PRA

A risk assessment that does not carry the analysis beyond core damage, or if core damage had not occurred, the sequence is carried to containment failure or successful termination of the event.

Level 2 PRA

A risk assessment which determines the final containment status and the fission product release to the environment.

Level 3 PRA

A risk assessment which determines the off-site consequences of the severe accident contributions for a plant.

Limit Cycle Operation

A form of oscillatory BWR instability in which the oscillation magnitude is limited by the system non-linearities. This behavior is only expected in situations of high power to flow ratio.

Loss of Containment Integrity

Any situation in which fission products are released from the containment exceeding that corresponding to the design basis leak rate. This includes containment venting actions to reduce pressure with large fission product inventories in the containment.

DEFINITIONS(3)

MECHANICAL CLAD DAMAGE

PERFORMANCE MONITORING

PLANT DAMAGE STATE

PROBABILISTIC RISK ASSESSMENT

PROMPT CRITICAL

PROMPT CRITICAL ACCIDENT

RISK

RISK ASSESSMENT

Mechanical Clad Damage

Loss of clad integrity resulting from pellet clad interactions resulting from power cycling or transients.

Performance Monitoring

The process of recording and tracking equipment failures in the plant for the purpose of tracking the risk associated with plant operation. This term also applies to a continuing program of measuring control room operator performance for the same purpose.

Plant Damage State

A specific level of damage to the plant resulting from an accident sequence. In general the state specifies the nature and extent of core damage, reactor vessel damage, core debris release to the containment, and the nature of containment failure.

Probabilistic Risk Assessment

A probabilistic determination of the various kinds of accidents, damage, and consequence for a nuclear plant. The IPE is a form of PRA.

Prompt Critical

An excess reactivity insertion above just critical plus the magnitude of the delayed neutron fraction. In such a condition reactor power rises at a rate characteristic of the neutron lifetime and the excess reactivity above prompt critical. Damaging power excursions can occur when the prompt critical reactivity lower limit is exceeded.

Prompt Critical Accident

An event in which the prompt critical reactivity lower limit has been exceeded. The only credible threats of this in the BWR are ATWS sequences with depressurization and core damage events with control rod relocation and subsequent reflood.

Risk

PP&L uses this term in a much broader sense than is commonly used in PRA work. It is used to refer to the likelihood of any form of plant damage or off-site consequence without distinction. The precise meaning must be determined from context.

Risk Assessment

An abbreviated phrase for probabilistic risk assessment.

DEFINITIONS(4)

RISK MANAGEMENT

SEVERE ACCIDENT COPING

SEVERE ACCIDENT MANAGEMENT

SEVERE ACCIDENT RECOVERY

STABILITY

SUPPORT FUNCTIONS

SUPPORT STATE METHOD

SUPPRESSION POOL BYPASS

Definitions(4)

Risk Management

The process of monitoring performance, risk assessment, feedback, and modifications intended to assure very low plant damage frequency and optimal use of plant facilities to prevent or mitigate damage from an initiating event and any combination of equipment failures.

Severe Accident Coping

This phrase refers to actions taken after control room ability to terminate an accident sequence has been lost which are required to bring the plant to a stabilized and controllable condition without further fission product release.

Severe Accident Management

Actions taken to avoid or minimize plant damage resulting from an initiating event and any combination of additional equipment failures. All such actions are defined by the EOPs and supplemental procedures referenced by the EOPs.

Severe Accident Recovery

Actions taken after termination (control and stabilization) of a severe accident to place the plant in a long term safe condition.

Stability

Relative to ATWS, the absence of oscillatory or excursive behavior of the reactor. Relative to severe accidents, plant parameters holding and controlled at non-threatening steady values or trending, under control, in a safe direction.

Support Functions

The function of plant systems which are required for the operation of one or more front-line systems of the plant. In some cases a system which provides a support function may also, under certain conditions, provide a front-line function.

Support State Method

A method of probabilistic risk assessment which partitions support system failures into groups having equivalent effects and treating these groups separately in the event trees.

Suppression Pool Bypass

A circumstances in which steam produced by the fuel may enter the wetwell airspace without first passing through the suppression pool.

APPENDIX 9

DISCUSSION OF ASSUMPTION 6

This appendix develops the argument that j outages of length T_j can be treated as one outage who's length is the sum of the T_j's. Mathematically this question becomes:

When is

$$\underbrace{\sum_{i=1}^{j} \int_{o}^{T_i} \lambda dt\, pdf(t)}_{\text{(LHS)}} = \underbrace{\int_{o}^{\sum_{i=1}^{j} T_i} \lambda dt\, pdf(t)}_{\text{(RHS)}}$$

a good approximation?

Solve the RHS

$$T_T \overset{def}{=} \sum_{i=1}^{j} T_i$$

Assume $pdf(t) = e^{-\lambda t}$

$$\int_{o}^{T_T} dt \lambda e^{-\lambda t} = (1-e^{-\lambda T_T})$$

Solve the LHS

$$\sum_{i=1}^{j} \int_{o}^{T_i} dt \lambda e^{-\lambda t} = \sum_{i=1}^{j} (1-e^{-\lambda T_i})$$

Recall that

$$e^{-\lambda t} = 1-\lambda t + \frac{(\lambda t)^2}{2!} - \frac{(\lambda t)^3}{3!} + \frac{(\lambda t)^4}{4!} \quad \ldots$$

For the RHS

$$(1-e^{-\lambda T_T}) = 1-1 + \lambda T_T - \frac{(\lambda T_T)^2}{2!} + \frac{(\lambda T_T)^3}{3!} - \frac{(\lambda T_T)^4}{4!} \ldots$$

For the LHS

$$\sum_{i=1}^{j} (1-e^{-\lambda T_i}) = \sum_{i=1}^{j} 1-1 + \lambda T_i - \frac{(\lambda T_i)^2}{2!} = \frac{(\lambda T_i)^3}{3!} - \frac{(\lambda T_i)^4}{4!} \ldots$$

The summation is a linear operator therefore thie expression can be written as follows.

$$\sum_{i=1}^{j} (1-e^{-\lambda T_i}) = \sum_{i=1}^{j} T_i - \frac{\lambda^2 T_i^{\,2}}{2!} + \frac{(\lambda T_i)^3}{3!} - \frac{(\lambda T_i)^4}{4!} \ldots$$

$$\lambda T_T - \frac{\lambda^2}{2!} \sum_{i=1}^{j} T_i^{\,2} + \frac{\lambda^3}{3!} \sum_{i=1}^{j} (T_i^3) - \frac{\lambda^4}{4!} \sum_{i=1}^{j} T_i^4 \ldots$$

A good approximation is guaranteed by ensuring that the differences between the two equations is small therefore,

$$\text{RHS-LHS} = - \frac{\lambda^2}{2!}\left[\left(\sum_{i=1}^{j} T_i^{\,2}\right) - T_T^{\,2}\right] + \frac{\lambda^3}{3!}\left[\left(\sum_{i=1}^{j} T_i^{\,3}\right) - T_T^{\,3}\right] - \frac{\lambda^4}{4!}\left[\left(\sum_{i=1}^{j} T_i^{\,4}\right) - T_T^{\,4}\right]$$

We consider only the square term since the effect of the higher order terms rapidly becomes insignificant.

Recall the identity

$$T_T = \sum_{i=1}^{j} T_i$$

Therefore for the square term

$$T_T^{\,2} = \sum_{i=1}^{j} T_i^{\,2} + 2 \sum_{i=1}^{j-1} T_i \sum_{k=i+1}^{j} T_k$$

$$- \frac{\lambda^2}{2!}\left[\left(\sum_{i=1}^{j} T_i^{\,2}\right) - T_T^{\,2}\right] = - \frac{\lambda^2}{2!}\left\{\left(\sum_{i=1}^{j} T_i^{\,2}\right) - \left[\sum_{i=1}^{j} T_i^{\,2} + 2 \sum_{i=1}^{j-1} T_i \left(\sum_{k=i+1}^{j} T_k\right)\right]\right\}$$

$$= - \frac{\lambda^2}{2!}\left[2 \sum_{i=1}^{j-1} T_i \left(\sum_{k=i+1}^{j} T_k\right)\right]$$

let $T_i = f_i T_T$

$T_k = f_k T_T$

The error term then becomes.

$$\text{Quadratic Error} = - \frac{\lambda^2}{2!} \, 2T_T^{\,2} \left[\sum_{i=1}^{j-1} f_i \left(\sum_{k=i+1}^{j} f_k\right)\right]$$

Recall that

$$T_T = \sum_{i=1}^{j} T_i = T_T \sum_{i=1}^{j} f_i$$

now $\sum_{i=1}^{j} f_i = 1$. Therefore $\sum_{k=i+1}^{j} f_k < 1$

note that

$\sum_{i=2}^{j} f_i = (1-f_1)$ and $\sum_{i=3}^{j} f_i = (1-f_1-f_2)$ so that the error

term becomes

$$\sum_{i=1}^{j} f_i \left(1- \sum_{k=1}^{j} f_i\right)$$

Expanding this formulation yields

$$\sum_{i=1}^{j-1} f_i - \sum_{i=1}^{j-1} f_i^2 - \sum_{i=1}^{j-1} f_i \left(\sum_{k=1+i}^{j-1} f_k\right)$$

The quadratic error term becomes

$$-\frac{\lambda^2}{2!} \; 2\, T_T^2 \left[\sum_{i=1}^{j-1} f_i - \sum_{i=1}^{j-1} f_i^2 - \sum_{i=1}^{j-1} f_i \left(\sum_{k=1+i}^{j-1} f_k\right)\right]$$

It is obvious that the term in brackets is always less than 1. Therefore the

$$\text{Error} < -\lambda^2 \, T_T^2$$

Examining the quadradic error term shows that this approximation always over-estimate the true failure probability by a small amount.

For a λT of 0.1 the error will be less than 1%. This is an insignificant error for this application. One must realize that λT is usually much less than 0.1.

APPENDIX 10

TREATMENT OF EXTERNAL EVENTS

Dependencies exist between plant equipment and the initiating event.

The occurrence of certain initiating event can result in equipment failure. As an example rupture of the recirculation system piping fails the injecting path of 1 loop of LPCI.

These dependencies between the initiator and the front line equipment must be captured in the systems analysis.

They are captured in a manner analogous to that used to identify dependencies between front-line systems and support systems.

The application of matrices to identify dependencies between IE and equipment is a logical extension of the mathematical formation thus far developed.

A layer of complexity is added, however since the initiator can fail front-line systems, support systems or both.

We will first discuss the method in general terms and then give a specific example of its application.

We 1st identify the dependencies between the initiating events and the plant equipment using two matrices: one for front-line systems, and one for support systems. The initiating events are identified by the row and the plant systems are identified by the column. If plant system J depends upon initiating event I then a 1 appears in column J of row I. If no dependency exist then a zero appears in that location.

This information is used when computing the value of a given functional fault. As shown in the functional fault tree on page 204 a basic event is labeled "initiator switch". Suppose a functional fault tree contains a system J. When evaluating this functional fault tree for event I, the initiating event vs. plant equipment matrices are examined. If a one is present in location I,J then a one is passed up through the functional fault tree for test system, otherwise a zero will be passed up.

If the dependency exist between the initiator and a front-line system the influence of the initiator is treated directly in the FFT. If the dependency exist between the initiator and a support system, the influence on the functional fault tree is treated indirectly through the front-line system vs. support system matrix. This is performed by first identify the support system effected. All higher order support dependencies are the identified using the support vs. support matrix. Finally the front-line system which depend on the affected support systems are identified using the front-line system support system matrix. This information is then feed into the functional fault trees.

This process is summarized below.

1. Identify dependencies between the initiating event and the plant equipment in a set of matrices.

2. If front-line systems are effected by the initiating event, change the the initiator switch in the functional fault tree from the effected system from o to 1.

3. If support systems are effected by the initiating event then perform the following steps.

   3.1 Ensure all dependencies between the initiator and the support systems are identified using the support system vs. support system dependency matrix.

   3.2 Identify what front-line system depend upon the support system identified in step 3.1.

   3.3 Change the initiator switch from zero to one for all front-line system identified in 3.2.

4. Once the initiator influence is accounted for compute value of functional fault tree and compute accident sequence frequency.

Consider the following example to illustrate this point. This example concerns several earthquake initiator.

Earthquake represent a convenient event to illustrate this algorith, however, it can be equally applied to other events such as floods and fires.

First we review how earthquakes induce plant failures.

Earthquakes are the result of ground motion. This motion is transferred to structures and equipment in the plant. The resulting acceleration from this motion results is a force or load on the plant structure and equipment in proportion to the acceleration and the equipment mass.

When the loads imposed exceed the equipment capability, the equipment is assumed to fail. Thus, for a given piece of equipment, earthquake imposed failure can be related to ground acceleration.

An example of such information is identified in Table 1. This data is presented for illustration purposes only and is not deemed reliable.

This information is used to develop matrices that relate the earthquake initiating events to plant equipment failure. Five earthquake initiating events are chosen for this example. They are earthquakes with the ground accelerations in the following ranges:

1	$0.35 < x \leq 0.35$
2	$0.35 < x \leq 0.43$
3	$0.42 < x \leq 0.55$
4	$0.55 < x \leq 0.9$
5	$0.9 < x$ 1.3

These ranges are referenced to the gravitational acceleration of the earth. The initiator vs. plant equipment dependency matrices are identified in Tables 2 and 3, for front-line and support systems respectively. This information is used in computing value of the functional fault trees.

Refer to the functional fault tree for high pressure make up found on page 204. The value of this fault tree depends upon the status of the input labeled "initiators switch". We first consider the influence of the earthquake on the front-line systems.

This influence is identified in Table 2. Earthquake 1 through 4 have no influence on the HPCI system; therefore the value of the initiating event switch for these earthquakes is '0'. Table 2 shows, however that earthquake number 5 will produce ground accelerations sufficient to fail the HPCI system. Therefore, the initiating event switch for earthquake number 5 is set to '1'.

Next the earthquake influence on the high pressure makeup function due to support system failures must be considered. Table 3 shows earthquake number 1 fails offsite power. However, reviewing the support system dependency matrix on page 220 and the front-line system support system depending matrix on page 218 shows no dependency between earthquake number 1 and high pressure makeup through the HPCI system. This result is also true for earthquakes 2 and 3. The occurrence of earthquake numbers 4 and 5, however, results in the failure of the 125 volt DC power switch gear. A dependency between 125V DC power and HPCI is identified in the front-line system support system dependency table on page 218. Therefore the initiating event switch for the HPCI system is set to '1' for those earthquakes.

The value of the functional fault tree is computed for each initiator. These values are used when estimating the frequency of plant damage as a result of the initiating event.

TABLE 1

Significant Earthquake-Induced Failures

This data is for information only and is not deemed reliable.

Comp. No.	Component Name	Component Failed or Failure Mode	Median Ground Acceleration Capacity (g)	β_r	β_u
1	480-V AC breakers	Cabinet structure	0.74	0.28	0.22
2	Reactor internals	Core-support aligners	0.78	0.21	0.28
3	Reactor internals	Shroud support	0.68	0.30	0.36
4	Reactor-scram system	CRD guide tubes and fuel bundles	0.66	0.25	0.26
5	500- and 230-kV switchyard	Ceramic insulators	0.20	0.20	0.25
6	230/13.8-kV transformer	Ceramic insulators	0.20	0.20	0.25
7	13.8/4.16-kV transformer	Slide off foundation	0.30	0.20	0.25
8	ESWS and RHR-SW	Soil liquefaction	0.42	0.15	0.10
9	HPCI turbine steam-supply valve	Yoke bending	0.78	0.24	0.38
10	HPCI turbine auxiliary	Structural failure	0.95	0.12	0.40
11	HPCI pump-suction valve	Yoke bending	0.93	0.24	0.38
12	Condensate storage tank	Anchor-bolt	0.79	0.22	0.31
13	RCIC turbine steam supply valve	Yoke bending	0.54	0.25	0.38
14	Core-spray pumps	Foundation bolting	0.80	0.17	0.27
15	RHR pump	Support failure	0.78	0.23	0.21
16	RHR outboard injec-	Bonnet studs	0.87	0.25	0.38
17	RHR heat exchangers	Lower support bolts	0.87	0.32	0.34
18	DG jacket water	Mounting bolts	1.06	0.18	0.21
19	DG control panel	Function	1.21	0.32	0.43
20	125/250-V DC switchgear	Welds at base	0.70	0.27	0.22
21	Reactor vessel	Sacrificial shield wall	1.10	0.32	0.21
22	Reactor and Control building	Shear wall between Units 1 and 2	0.70	0.31	0.24
23	Recirculation pumps	Motor-support lugs	0.79	0.30	0.28
24	Refueling water storage tank	Tank wall	0.16	0.28	0.29

TABLE 2

The Impact of Earthquakes upon front-line systems

Earthquake Initiator	RCIC	HPCI	LPCI DIV I	LPCI DIV II	CS DIV I	CS DIV II	ADS DIV I	ADS DIV II
1	0	0	0	0	0	0	0	0
2	0	0	0	0	0	0	0	0
3	1	0	0	0	0	0	0	0
4	1	0	1	1	1	1	0	0
5	1	1	1	1	1	1	0	0

Front-line Systems (column group heading over RCIC through ADS)

TABLE 3

The Impact of Earthquakes upon Support Systems

Earthquakes Initiator	125V A	125V B	125V C	125V D	4160V A	4160V B	4160V C	4160V D	480V LC A	480V LC B	480V LC C	480V LC D	RHR A	RHR B	RHR C	RHR D	ESW 1	ESW 2	Offsite Power
1	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1
2	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	1
3	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	1	1	1
4	1	1	1	1	0	0	0	0	1	1	1	1	1	1	1	1	1	1	1
5	1	1	1	1	0	0	0	0	1	1	1	1	1	1	1	1	1	1	1