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A Review of the Shoreham Nuclear Power Station Probabilistic Risk Assessment

Internal Events and Core Damage Frequency

Prepared by D. Ilberg, K. Shiu, N. Hanan, E. Anavim

Brookhaven National Laboratory

Prepared for
U.S. Nuclear Regulatory Commission

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ABSTRACT

A review of the Probabilistic Risk Assessment of the Shoreham Nuclear Power Station was conducted with the broad objective of evaluating its risks in relation to those identified in the Reactor Safety Study (WASH-1400). The scope of the review was limited to the "front end" part, i.e., to the evaluation of the frequencies of states in which core damage may occur. Furthermore, the review considered only internally generated accidents, consistent with the scope of the PRA. The review included an assessment of the assumptions and methods used in the Shoreham study. It also encompassed a re-evaluation of the main results within the scope and general methodological framework of the Shoreham PRA, including both qualitative and quantitative analyses of accident initiators, data bases, and accident sequences which result in initiation of core damage. Specific comparisons are given between the Shoreham study, the results of the present review, and the WASH-1400 BWR, for the core damage frequency. The effect of modeling uncertainties was considered by a limited sensitivity study so as to show how the results would change if other assumptions were made. This review provides an independently assessed point value estimate of core damage frequency and describes the major contributors, by frontline systems and by accident sequences.

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The work was performed for the Reliability and Risk Assessment Branch (RRAB) of the U. S. Nuclear Regulatory Commission. Mr. Ed Chow of RRAB was the technical monitor of the project. The authors wish to acknowledge Ashok Thadani, Chief, RRAB, and Arthur Buslik and Ed Chow (RRAB) for constructive comments on the preliminary and the final drafts of this report.

Finally, we would like to express our appreciation to Cheryl Conrad, Marguerite Marsch and Nancy Nelson for an excellent job in typing this document.

NOMENCLATURE

A	Large LOCA
ADS	Automatic Depressurization System
A_{out}	Large LOCA outside Containment
ARI	Alternate Rod Insertion
ATWS	Anticipated Transient Without Scram
B_0	LOCA - Induced Loss of Offsite Power
BWR	Boiling Water Reactor
C	Scram
C_A	Alternate Rod Insertion
C_E	Electrical Failure to Scram
C_I	Scram Initiation
C_M	Mechanical Failure to Scram
C_2	One Standby Liquid Control Loop
C_{21}	Second Standby Liquid Control Loop, given C_2
CD	Core Damage
CDFT	Core Damage Fault Tree
CET	Containment Event Tree
CM2B	Common Mode Failure of 2 Batteries (Divisions 1 and 2)
CM3B	Common Mode Failure of 3 Batteries (Divisions 1, 2, and 3)
CM2D	Common Mode Failure of 2 Diesel Generators (Divisions 1 and 2)
CM3D	Common Mode Failure of 3 Diesel Generators
CRD	Control Rod Drive
D	Failure of Diesel Generators and Failure to Recover of Division I or II Diesel in 2 hours
DHR	Decay Heat Removal
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EPG	Emergency Procedure Guidelines
EPS	Electrical Power System
ESF	Engineered Safeguard Features
ESWS	Essential Service Water System
FSAR	Final Safety Analysis Report
FT	Fault Tree
FTA	Fault Tree Analysis

NOMENCLATURE (Continued)

FW	Feedwater
G	Drywell Heat Removal
HEP	Human Error Probability
HPCI	High Pressure Core Injection System
I	Recovery of Offsite power in 30 minutes
II	Recovery of Offsite power in 2 hours
III	Recovery of Offsite Power in 4 hours
IV	Recovery of Offsite power in 10 hours
IORV	Inadvertent Open Relief Valve
L	Level Control and Stable Cooling Established
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
M	Maintain Reactor Pressure
M_S	Manual Shutdown
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
NPCC	Northeast Power Coordinating Council
NPSH	Net Positive Suction Head
P	Safety Relief Valve Reclose
P_A or D	ADS inhibit
P_1	One Stuck Open Relief Valve (SORV)
P_2	Two or more SORV
PCS	Power Conversion System
Q	Feedwater System
R	Redundant Reactivity Control System
RB	Reactor Building
RBCLCW	Reactor Building Closed Loop Cooling Water
RBSVS	Reactor Building Standdby Ventilation System
RCIC	Reactor Core Isolation Cooling
RCIC SC	RCIC in Steam Condensing Mode
RHR	Residual Heat Removal System
RHRHX	RHR Heat Exchanger

NOMENCLATURE (Continued)

RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
S_1	Intermediate LOCA in Drywell
S_2	Small LOCA in Drywell
SDV	Scram Discharge Volume
SJAE	Steam Jet Air Ejector
SNPS	Shoreham Nuclear Power Station
SORV	Stuck Open Relief Valve
SRV	Safety Relief Valve
SWS	Service Water System (or RBSWS = Reactor Building SWS)
TBSWS	Turbine Building Service Water System
T_C	Loss of Condenser
T_D	Loss of a DC bus (Division 1 or 2)
T_E	Loss of Offsite Power
T_F	Loss of Feedwater
T_{FA}	Isolation ATWS
T_I	Inadvertent Open Relief Valve
T_M	MSIV Closure Transient
T_{MT}	Loss of Drywell Coolers
T_R	Loss of a Reference Leg in Reactor Water Level Measurement System
T_{SW}	Loss of Service Water System
T_T	Turbine Trip
TAF	Top of Active Fuel
U	High Pressure Injection Function
U'	Reactor Core Isolation Cooling System
U''	High Pressure Core Injection System
V	Low Pressure Injection Function
V_C	Condensate Injection
V_4	Low Pressure Core Cooling Systems (includes LPCI and LPCS)
X	Depressurization (via Automatic Depressurization System or Manual)
W	Containment Heat Removal Function (includes Residual Heat Removal System and Power Conversion System)
W'	RHR or RCIC in Steam Condensing Mode
W''	Power Conversion System
Z	The function of "MSIV reopened in the long term"

EXECUTIVE SUMMARY

This review of the Probabilistic Risk Assessment of the Shoreham Nuclear Power Station was conducted by Brookhaven National Laboratory under the sponsorship of the U.S. Nuclear Regulatory Commission. The review of the internally generated plant accident sequences which could potentially lead to core damage began in December 1983 and was concluded at the end of October 1984. Two draft versions of this report were published (November and December 1984) with the objective of soliciting comments. This version of the report is the final report incorporating comments from the NRC, the utility staff, and consultants. The broad objective of the review was to evaluate the core damage frequency as calculated in the Shoreham Probabilistic Risk Assessment in relation to that identified in the Reactor Safety Study (WASH-1400). The review by Brookhaven included an assessment of the assumptions and methods used in the Shoreham study. The review also included a re-evaluation of the main results within the scope and general methodological framework of the Shoreham study. This included both qualitative and quantitative analyses of accident initiators, some of the data bases, and accident sequences which result in the initiation of core damage.

The review process included a meeting with the Shoreham owner and its consultants, a site visit, and one formal round of (written) questions and answers. The utility and its consultants were helpful and cooperative throughout the course of the review. The Shoreham PRA package was quite comprehensive as originally submitted, and there was no significant need to augment the information by additional submittals. Finally, comments were received from the NRC and the utility, and they were discussed with BNL in an additional meeting.

The main conclusions of this review are the following:

a. Within its stated scope, the Shoreham study is a good and comprehensive piece of work. The utility produced a study which used the basic approach and techniques of the Reactor Safety Study (event tree/fault tree methodology), but which accounted for plant-specific design differences between Shoreham and the Reactor Safety Study plant and included, in some instances, some additional details beyond those provided in the Reactor Safety Study.

b. The Brookhaven reviewers believe that the Shoreham study can be updated within its present framework and structure, by taking into account the specific recommended changes in modeling and data, as well as comments found in the main body of this review report.

c. The reviewers found that some of the analyses in the Shoreham study were rather non-conservative, i.e., led to an underestimation of core damage frequencies. In several instances, this may have resulted from insufficient justification to support an analysis or quantification of data. In some other parts of the study, the analysis was determined to be conservative, and more realistic alternatives have been put forth. However, the results of this review show that, overall, more assumptions and modeling were judged to be non-conservative than conservative.

d. Item c notwithstanding, BNL found that, in general, the SNPS-PRA approach included considerable detail and was an attempt to address the modeling of the accident sequences and their quantification as realistically as possible, based on the specific Shoreham designs and procedures and on past nuclear power plant experience.

e. Most of the BNL comments on the SNPS-PRA, and most of the BNL modifications, relate to accident sequence quantification. In many instances, error, lack of supportive evidence in the PRA, or new information from LERs or other sources were the reasons for BNL modifications to event tree quantification. Overall the modifications resulted in large changes in the ranking of dominant accident sequences in the BNL revised results. Even though the overall change in core damage frequency is by a factor of 2.5, it includes both increases and decreases in individual sequences, so that a different ranking of dominant sequences was generated in the BNL re-assessment.

f. Within the perspective of the foregoing comments, the Shoreham study constitutes a very useful tool for identifying accident sequences that may lead to initiation of core damage. The PRA, as well as our review, reveals a hierarchy of contributors to the frequency of a variety of core damage states and indicates possible weaknesses that may require additional evaluations. Furthermore, the study could be used in implementing a program aimed at prevention of the important accident sequences. The review did not include an evaluation of the cost-benefit tradeoffs of any strategies or programs in this area, and therefore no conclusion is drawn in this regard.

g. The main quantitative results of the BNL revision along with the results of the Shoreham study are given in Table 0.1, as frequencies per plant-year of operation. The table shows that the BNL revision results are higher by a factor of three than the SNPS-PRA results. The main contributors appear to come from ATWS, LOOP, transients with scram, and internal flood initiators. Interfacing LOCA was determined to be about half an order of magnitude higher than the SNPS-PRA estimate.

h. The difference between the Shoreham and Brookhaven point value estimates for the core damage frequency can be attributed mainly to the following factors:

   1. Based on an updated source of experiential data, the BNL review assessed an increase in the transient initiator frequencies which affected the ATWS sequence contribution and the MSIV closure and turbine trip transients.

   2. The BNL re-assessment of the LOOP frequency is 0.15 per year for the Shoreham site compared with 0.08 per year in the SNPS-PRA. This increase is partly counterbalanced by higher LOOP recovery probabilities derived from a more recent evaluation of LERs used in the BNL re-assessment.

3. Loss of instrumentation indications in the control room also contributed to the increase in the BNL assessed LOOP initiated core damage frequency.

4. BNL calculated a higher frequency for the "excessive release of water in the reactor building" initiator (about a factor of four). A more elaborate time-phased model considering the early failure of HPCI and RCIC also contributed to the increase.

5. A more refined treatment of the level instrumentation reference leg leakage and the various failure modes enabled the identification of several new sequences that were not included in the SNPS-PRA. These new sequences increased the total core damage contribution from this initiator. Since the original submittal of this review, BNL has been informed that additional level of measuring instrumentations are being added at Shoreham. Based on an informal assessment, it is judged that this instrumentation will substantially decrease the frequencies of most of the new sequences identified by BNL.

6. Revised ATWS functional event trees were developed considering Shoreham plant-specific information. The major contribution to the increase comes from the BNL initiator frequencies. Changes from event tree modification resulted in only a smaller increase.

i. Figure 0.1 depicts the SNPS-PRA and BNL results according to the five classes of core damage states considered in the SNPS-PRA. Class III (the class related to LOCA sequences) exhibited only slight changes, and Class IV (related to the ATWS sequences) increased the most, for the reasons h(1) and h(6) given above. The Class I core damage state increased mainly because of the increased contribution estimated by BNL for LOOP frequency, excessive release of water, and transients. Class II core damage frequency was also changed. This class does not lead to core damage* in all cases; in many cases it results in containment failure, with the core continuing to be cooled. The increase in Class II is attributed to the inclusion of additional sequences BNL considered in the loss of service water and the LOOP transients. Loss of condenser vacuum also contributed to the increase in class II frequency. Finally, the interfacing LOCA frequency was based in the BNL review on several precursor events of this type, rather than on LER valve data. This treatment resulted in an increase in the initiating frequency of this event.

*The Shoreham PRA used the term "core vulnerable" rather than "core damage" because damage to the core will not occur for all sequences in Class II. However, they have extended this terminology to all other classes as well. BNL believes that it is more appropriate to retain the terminology "core damage" for all classes (in order to be consistent with terminology in previous PRAs), and to note separately those special cases where core damage may not occur.

j. Figure 0.2 provides an alternative means of presenting the results. It shows that in general the BNL and SNPS-PRA results are in agreement with respect to the main class of contributors. The difference in the relative contribution of ATWS and transients is not as great as might be expected because a significant fraction of the SNPS ATWS core damage frequency was added to the transients rather than included in the "ATWS class IV" part of the pie chart. BNL considered all ATWS sequences that result in core damage to lead eventually to Class IV* core damage.

The BNL review concluded that if improvements are to be considered, the greatest impact may be achieved in the following areas:

a. Since the submittal of the SNPS-PRA, BNL was informed that additional systems have been added to the onsite emergency system. Conceivably this may help to reduce the contribution to core damage from LOOP events and total loss of level instrumentation.

b. Staggered procedures with respect to calibration of the most important sensors. Actuation of the high and low pressure systems from another redundant pair of RPV water level sensors different from the four N091 level sensors currently employed. BNL was informed, after this review was completed, that indeed this design change is being implemented. Thus, if detailed information and a PRA update were submitted for review, the BNL assessment might well change.

c. Treatment of sequences related to the interfacing LOCA and excessive release of water. Review of emergency procedures may be one example.

d. Treatment of the ATWS sequences. Review of emergency ATWS procedures may be one example. However, BNL identified some need for additional generic physical analyses of ATWS if better understanding of operators' response time is desired.

BNL concluded that the results of the SNPS-PRA, taking into account BNL review considerations, provide an effective framework for further studies of the Shoreham plant design and operation and for evaluating modifications in those areas.

A final comment is in order regarding any possible comparison between the results of the SNPS-PRA and results of some other similar PRAs. Superficial numerical differences cannot be relied upon as indicators of relative core damage frequency; some earlier PRAs, in adhering more closely to WASH-1400 thinking, have provided results which are to some extent not as realistic in the non-conservative direction as those of the Shoreham PRA, which has advanced the conceptual basis in the direction of greater realism. Comparisons between PRAs should be made only in light of a clear understanding of where realistic credit has been taken for mitigating systems and where different assumptions have been applied. A major goal of this review has been to indicate where the SNPS-PRA has made advances in this area.

*See Appendix D for details. The basis for the BNL assumption is that there is a lack of time for the operator to inhibit ADS, as level 1 is reached promptly.

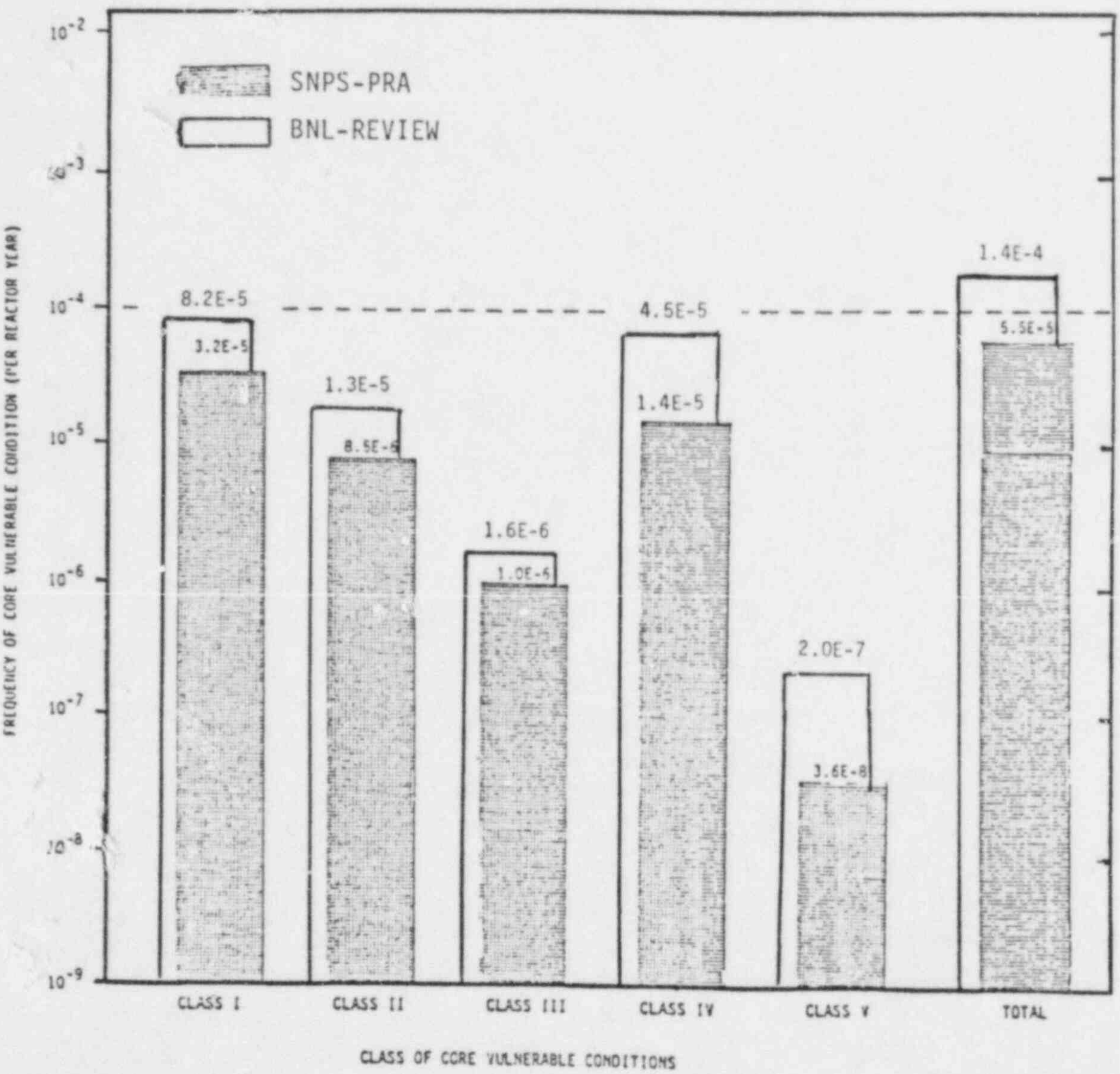

Figure 0.1 Summary of the Results of the Event Tree Quantification Displayed by Class of Postulated Core Damage Condition.

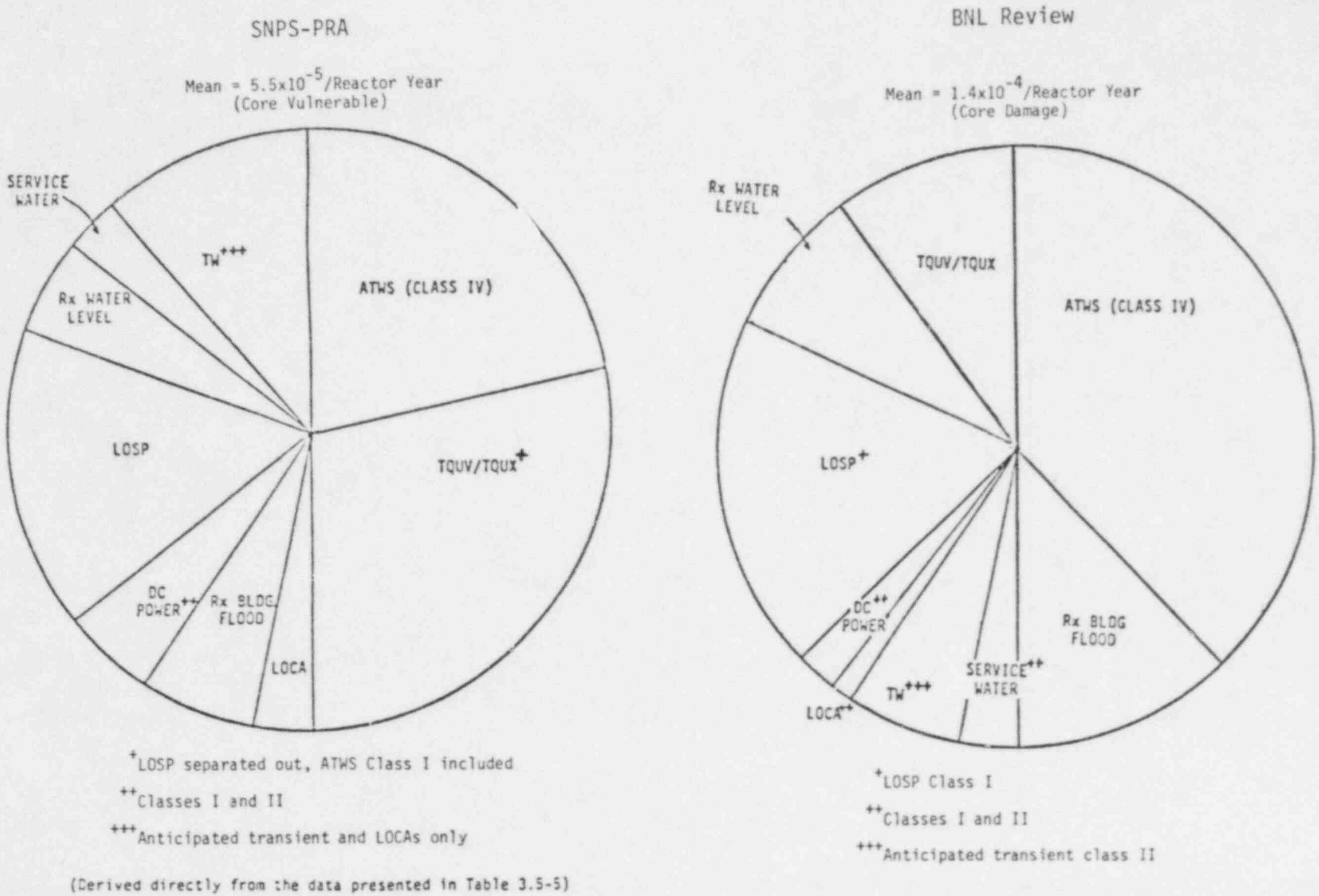

Figure 0.2 Comparison of the SNPS-PRA and the BNL Review Contributing Accident Sequences to the Calculated Core Damage Frequency (per Reactor Year) Due to the Identified Accident Sequence Contributors.

Summary Table 0.1 Comparison of SNPS-PRA and BNL Review Results

Accident Sequence Initiator		Core Damage (CD) Class					
		I	II**	III	IV	V	CD
Loss of Coolant Accidents (LOCA)	SNPS BNL		1.0E-6 5.3E-7	1.0E-6 1.3E-6			2.0E-6 1.8E-6
Anticipated Transient Without Scram (ATWS)	SNPS BNL	4.0E-6 *		2.1E-9 2.8E-8	1.4E-5 4.5E-5		1.8E-5 4.5E-5
Loss of Offsite AC Power (LOOP)	SNPS BNL	9.9E-6 2.9E-5	1.1E-6 1.4E-6				1.1E-5 3.0E-5
Transients (Turbine Trip Manual Shutdown, MSIV and other)	SNPS BNL	8.7E-6 1.5E-5	4.3E-6 6.4E-6				1.3E-5 2.2E-5
Level Instrumentation (Reference leg and drywell cooling)	SNPS BNL	3.8E-6 1.2E-5	1.2E-7 2.5E-8	5.2E-9 1.5E-7			3.9E-6 1.2E-5
Flooding at Elevation 8 of Reactor Bldg.	SNPS BNL	3.1E-6 1.8E-5	7.8E-7 2.0E-6				3.9E-6 2.0E-5
LOCA Outside Drywell	SNPS BNL					3.7E-8 2.0E-7	3.7E-8 2.0E-7
Loss of Service Water, or DC Bus	SNPS BNL	3.0E-6 7.6E-6	7.7E-7 2.4E-6				3.8E-6 1.0E-5
TOTAL	SNPS BNL	3.2E-5 8.2E-5	8.5E-6 1.3E-5	1.0E-6 1.5E-6	1.4E-5 4.5E-5	3.7E-8 4.2E-7	5.5E-5 1.4E-4

*In BNL review all ATWS sequences are assumed to lead to core damage class IV. This is based in part on the judgment that the operator will not be able to inhibit ADS.

**Class II leads in many cases to containment failure without loss of core cooling. Therefore, only a part of Class II results in core damage.

1. INTRODUCTION

This section explains why a probabilistic risk assessment (PRA) was performed for the Shoreham Nuclear Power Station (SNPS), how the review of the PRA was performed by Brookhaven National Laboratory (BNL), and how this report is organized.

1.1 Background

The Shoreham PRA[1,2] is a self-motivated undertaking by the Long Island Lighting Company (LILCO), the owner and operator of the Shoreham facility. LILCO initiated and managed the PRA study in order to provide basic data to its risk management program by evaluating the plant response beyond the normal design basis. LILCO's intention is to make use of PRA methodology to better assess the Shoreham design relative to postulated accident sequences and their resulting public risk. The PRA, in its first revision form, was submitted on June 24, 1983. The NRC contracted with BNL to perform an in-depth review of the PRA, which began in December 1983.

The Shoreham PRA was prepared according to NRC guidelines, and is similar to the Limerick or GESSAR PRAs[3,4] with respect to scope, methodology, and data. Like the two other PRAs reviewed by BNL[5,6], it was carried out with the basic approach and techniques of the RSS[7]. However, plant specific features and design information were used. In many instances, more detailed modeling and recent data such as LER information were incorporated.

The SNPS-PRA study also addressed the comments on RSS made by the Lewis Committee[8], and LILCO reflected these comments in the SNPS-PRA as they thought appropriate.

The BNL review was concluded at the end of October 1984. Some of the minor sequences were reviewed to a lesser depth than the significant ones. For example, in some cases if an in-depth, time consuming review was expected to result in much less than a factor of two change in core damage frequency of a particular sequence, it was not undertaken. On the other hand, based on the SNPS-PRA itself and on reviewers' experience with other PRAs, several additional sequences were found to contribute to the core damage frequency and were included in the BNL re-assessment. In summary, most of the SNPS-PRA sequences were reviewed, and several modifications, additions, or subtractions were made, as shown in the rest of this report.

The current report (May 1985) supersedes two previous drafts issued for soliciting comments (November, December 1984). This final report incorporates comments made on the previous drafts by NRC and by LILCO.

1.2 Objective, Scope, and Approach to Review

The broad objective of the BNL review of the SNPS-PRA was to evaluate qualitatively and quantitatively the assessment of the important accident sequences that are internally generated and lead to core damage initiation. To be consistent with the SNPS-PRA scope, the review excluded internally generated fires, but it included an assessment of the externally generated LOOP accident initiator. To carry out this objective, BNL reviewed the

assumptions and methods of the SNPS-PRA within its stated scope. This review included reevaluation of the important accident sequences that may lead to core damage, their respective frequency of occurrence, the total frequency of core damage initiation, and the impact of several changes made in the assumptions on the total frequency calculated for the baseline case. In particular, the review included evaluations of accident initiators, data, and development and quantification of accident sequences.

This review of the "internally" generated accident sequences with respect to the frequency of core damage constitutes part of the work on the SNPS-PRA done by BNL for the NRC. Other BNL reviews consider the core melt phenomenology and the containment analysis, and will be reported separately.

The review was performed over a one year period in two phases. In Phase I, an overall review was performed and a list of questions was sent to the utility. These were discussed in a meeting held in December 1983 between NRC, BNL, and LILCO. The review process benefitted from this productive meeting. LILCO and its consultants were entirely cooperative in providing the information needed to gain a detailed understanding of the PRA for the in-depth review process.

Responses and additional information were submitted in May 1984[9]. A report[2], "Review of Shoreham Water Level Measurement System" prepared for LILCO by S. Levy, Inc., was also part of the response package. BNL included this report in its PRA review package; whenever the SNPS-PRA is mentioned in this review, this report should be considered part of it.

Phase I of the review included an in-depth re-evaluation of the sequences following a release of excessive water into Elevation 8 of the Shoreham reactor building[10]. The report summarizing this review was submitted to NRC in April 1984. Participating in Phase I were Kelvin Shiu, Yang-Ho Sun, Eshagh Anavim, and Ioannis A. Papazoglou.

Phase II of the review took place from June to October 1984. An in-depth review of the accident sequence modeling and systems, as well as the data used in the SNPS-PRA, was performed. This is summarized in the following chapters of this report. Dan Ilberg, Kelvin Shiu, Nelson Hanan, and Eshagh Anavim participated in this phase.

The most important sequences were reviewed, as mentioned above. Those sequences are reassessed and the results are presented in appendices to Section 5 of this report. The quantification and sequence modification are explained whenever they deviated from the original SNPS-PRA with the intention, of providing sufficient detail to enable others to follow the review considerations. The review of the fault trees was based on comparison with the Limerick fault trees, taking into account the BNL review of the latter and the comments in the BNL Limerick PRA review[5]. The SNPS-PRA included more explicit modeling of functional dependences in the event trees by increasing their detail. Based on the above, and based on the result of a previous review[5] indicating that Core Damage Fault Tree (CDFT) modified the results by about a factor of two, it was determined that this approach if applied to SNPS-PRA would change the net result by a smaller factor. Hence, BNL judged that a CDFT approach was unnecessary for SNPS-PRA. Functional level event trees were utilized by BNL to account for the dependence between the short and

long term PCS functions (Q function vs. W and Z functions), because this seemed to be treated non-realistically (see Appendix 5A) on most event trees.

The scope of this review did not include uncertainty and importance analyses. Nevertheless, in several instances it seemed that, besides the baseline assumption, other assumptions could be made if properly substantiated. The impact of these different assumptions on the results was assessed in a limited sensitivity analysis, summarized in section 5.3, which provides some additional insight on range of core damage frequency values that could potentially be generated for the SNPS-PRA.

The SNPS-PRA should be cited for its comprehensiveness and self-contained nature which facilitated an in-depth peer review.

1.3 Organization of Report

Section 2 provides a description of plant modeling which includes identification of initiating events that result in challenging of the safety systems of the plant, and a discussion of safety functions and systems important to preventing or mitigating core damage events. Section 3 contains a description of accident sequence definition, and a discussion of both the BNL revised and the SNPS-PRA event tree/fault tree approaches. Section 4 is a review of the SNPS data, including the numerical values for the initiating event frequencies used in the SNPS-PRA and the BNL assessment, and the numerical values for some of the parameters necessary for quantification of accident sequences (i.e., for LOOP time phased sequences). Section 5 covers accident sequences quantification, a brief description of the SNPS-PRA approach to quantification, the BNL modifications to the quantification, and the revised core damage frequencies. It also describes a limited sensitivity study checking the influences of a few of the assumptions on the core damage frequencies calculated for the baseline case.

Appendices to Section 5 provide more detailed discussions of the event trees reviewed and include the BNL modifications along with their bases. These appendices should help others to review our considerations.

1.4 References to Section 1

1. "Probabilistic Risk Assessment Shoreham Nuclear Power Station Long Island Lighting Company, Final Report", Science Applications, Inc., June 24, 1983.

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10. Shiu, K., et al., "A Review of the Sequences Following Release of Excessive Water in Elevation 8 of the Reactor Building in the Shoreham Nuclear Power Station", Brookhaven National Laboratory, NUREG/CR-4049, November 1984.

2. PLANT MODELING

The plant modeling part of the SNPS-PRA covers the identification of the initiating events that can lead to core damage, the safety functions important to preventing or mitigating core damage events, and the systems directly performing each of the safety functions, as well as the assessment of the success criteria of the safety functions and the systems. These systems are referred to as frontline systems. In addition, the plant modeling includes the identification of the support systems for each frontline system, i.e., the systems required for the function of the frontline systems.

This section has three parts. Subsection 2.1 describes the safety functions, the corresponding frontline and support systems, and their success criteria and provides a comparison with the Reactor Safety Study[1] and LGS-PRA[2]. Subsection 2.2 discusses the particular initiating events and their partition into groups containing events having the same success criteria for the frontline systems. In both subsections, the SNPS-PRA assumptions are reviewed, evaluated, and compared with those of the Reactor Safety Study (RSS). Subsection 2.3 is a summary of BNL's assessment.

2.1 Safety Functions and Corresponding Systems

2.1.1 Safety Functions and Frontline Systems

The safety functions important to preventing or mitigating the consequences of core damage following an initiating event are given in Table 2.1. These functions can be further subdivided for the SNPS into the functions given in Table 2.2, each of which is directly performed by one or more frontline systems. The frontline systems for the SNPS are given in Table 2.3, and in Table 2.4 they are compared with the corresponding systems of the BWR plant analyzed in the Reactor Safety Study (RSS-BWR) and in the LGS-PRA. A short description of SNPS frontline systems and their differences from those in the RSS-BWR and LGS follows.

Reactor Protection System (RPS) - The SNPS has incorporated several design changes, as recommended by Alternate 3 of NUREG-0460[3], to reduce the probability of a failure to scram:

a) Alternate rod insertion (ARI) - this system is effective in reducing electrical common-mode failure to scram. (Similar to LGS, dissimilar to RSS).

b) Diverse and redundant water level sensors for the Scram Discharge Volume (SDV) - this is expected to reduce the chance of an occurrence similar to that at the Browns Ferry plant. (Similar to LGS, dissimilar to RSS.)

c) MSIV closure on reactor level 1 rather than level 2.

Standby Liquid Control (SLC) - The SNPS system is different from the Alternate 3 described in NUREG-0460, which requires two automatically initiated SLC pumps with 86 GPM (43 GPM per pump). It includes two SLC pumps (43 GPM each) manually initiated, with only one pump working at any time. The RSS-BWR has two similar manually actuated SLC pumps. The LGS has three SLC

pumps having automatic initiation rather than manual, allowing for two pumps injection of 86 GPM.

Reactor Core Isolation Cooling (RCIC) - There are no major differences between the SNPS, the LGS, and RSS-BWR designs. SNPS RCIC flow rate is, however, 400 GPM compared with 600 GPM in the other two BWRs. This is a 10% reduction in flow rate corresponding to the power difference between the reactors.

High Pressure Coolant Injection (HPCI) - The major difference is that, for SNPS and the RSS-BWR, HPCI injects into a feedwater line, whereas for the LGS, HPCI injection is split between the core spray injection line and the feedwater line.

Control Rod Drive (CRD) - There are no major differences between the SNPS, the LGS, and RSS-BWR designs. No credit to this system is given in the PRA or BNL assessment, even though it may provide successful high pressure injection two hours after initiation of several transients. The effect is not very large (see Table 5.15).

Automatic Depressurization System (ADS) - The SNPS-ADS system has three separate compressed gas supplies; these are (1) compressed nitrogen, (2) plant air backup, and (3) accumulators (see Table 2.4). It incorporates the following additional features beyond the RSS-BWR or LGS-PRA*:

a) SNPS has an automatic initiation of ADS upon low level signal (level 1).

b) SNPS has individual accumulators to store pneumatic energy for each SRV operation. Each accumulator is sized to provide five actuations.

c) Each SRV has two solenoid pilot valves.

d) After receipt of the automatic ADS initiation signal, a timer provides two minutes delay to allow operator to inhibit before actual ADS initiation.

Low Pressure Coolant Injection (LPCI)

a) The SNPS and the RSS-BWR LPCI system primary mode is to inject water into the recirculation loops to ensure injection into the intact loop. The LGS LPCI system injects water directly into the core shroud through four separate injection lines.

b) The LGS pumps can pump saturated water. The RSS-BWR LPCI pumps have net positive suction head (NPSH) requirements which may not always be met and could lead to pump failure. This is particularly important if there is excessive containment leakage. The SNPS-PRA states that the LPCI NPSH appears to be marginal at saturated pool temperature and containment atmospheric pressure. However, calculations show the NPSH to be adequate.

*LGS has recently modified its ADS initiation logic.

Low Pressure Core Spray (LPCS) - The SNPS and LGS core spray pumps can pump saturated water. The RSS-BWR pumps have NPSH requirements which may not always be met. All three plants have two redundant loops, but the SNPS utilizes one pump per loop whereas the others have two pumps per loop.

Residual Heat Removal (RHR) - The major differences between the SNPS, RSS-BWR, and LGS RHR systems are: (1) two RHR heat exchangers for LGS and SNPS, compared with four RHR heat exchangers for the RSS-BWR and (2) credit was taken for the steam condensing mode* of RHR only in the SNPS-PRA. The SNPS and LGS pumps cannot pump saturated water. However, if saturation conditions exist in the reactor pressure vessel only, both plants can still pump.

Containment Sprays - All three reactors have a manually actuated containment spray system that can spray either the drywell or the wetwell volumes.

2.1.2 Success Criteria for the Frontline Systems

The SNPS-PRA considers four general classes of initiating events:

1) Loss-of-coolant accidents (LOCAs),

2) Transients with successful scram,

3) Anticipated transients without scram (ATWS),

4) Low frequency transients of special interest.

The choice of initiating events is discussed in detail in Section 2.2.

The success criteria for the systems available to provide successful termination of an initiating event without leading to core damage are summarized in Tables 2.5 and 2.7 (taken from the SNPS-PRA report). They are defined in terms of the minimum number of systems required to prevent excessive fuel clad temperature and to remove decay heat. The success criteria used in the SNPS-PRA represent "realistic" requirements and do not necessarily correspond to Final Safey Analysis Report (FSAR) criteria and/or predictions. The SNPS criteria were developed in part from vendor deterministic analyses[4,5]. Here the SNPS-PRA departs from the Reactor Safety Study, where FSAR criteria were used. In the following three subsections the success criteria assumed in the SNPS-PRA are compared with those in the RSS and the LGS-PRA for the first three major classes of initiating events, and BNL review comments or changes to SNPS success criteria are given. The fourth class (low frequency transients), has the same success criteria as do the anticipated transients and is covered in Section 2.1.2.2.

2.1.2.1 Success Criteria for LOCA Initiators

Table 2.6 compares the success criteria for LOCA initiating events (with successful scram) for the SNPS, LGS, and RSS-BWR. It shows the required

*Shoreham does not regularly use the steam condensing mode. Section 5.3 shows the effect on Class II core damage when no credit is given to the steam condensing mode (see Table 5-15).

systems for both steam and liquid breaks as a function of the break size. Major differences are as follows:

1. The RSS distinguishes between injection and recirculation phases for large breaks in which only low pressure systems are adequate. This results in a stricter requirement for the injection phase for the RSS-BWR than that for SNPS.

2. The RSS-BWR requires operation of four ADS valves for depressurization following small and medium break LOCA vs. three ADS valves for the SNPS.

3. In the small LOCA case, the SNPS takes credit for successful high pressure injection using the feedwater system when the MSIVs remain open or can be reopened within 30 minutes.

4. In the LOCA cases, the RSS-BWR and the LGS-PRA require only one LPCS pump or one LPCI pump to operate for successful low pressure injection. For the SNPS, in addition to the above, injection with one condensate pump is also considered a success. (In the BNL review, the condensate pump is assumed to be a success for medium or small LOCA only).

5. The SNPS analysis takes credit for the PCS as a means of long-term cooling for the small and medium LOCA based on successful reopening of one or more MSIVs. The LGS-PRA also takes credit for PCS in the case of small and medium LOCAs, but the RSS does not.

6. The RSS-BWR analysis takes credit for one CRD pump as a means of injection for steam breaks of less than 1 in. diameter or liquid breaks of less than 0.6 in. diameter. The SNPS-PRA and LGS-PRA took no credit for CRD pump injection.

Table 2.6 shows that the LOCA success criteria for the three plants are in general agreement; use of the PCS for injection and long-term cooling of the core is the most notable difference between the SNPS and the RSS-BWR. Table 2.4 shows that HPCI and LPCI are sized in proportion to each plant's thermal power (smaller by a factor of 0.75 for SNPS than for LGS or RSS-BWR). However, the RCIC is rated 10% less for the SNPS than the equivalent flow rate in LGS or RSS-BWR if their RCIC were scaled down by the 0.75 power ratio factor.

For its re-assessment, BNL in general accepted the SNPS LOCA success criteria given in Table 2.5. One exception is for large LOCA liquid line breaks connecting to the RPV below the top of the core. Due to the lack of supporting results of a best-estimate analysis for core cooling given a large LOCA and a condensate pump injection of 1000 gpm, BNL can only provide a limited assessment of the adequacy of condensate pump injection. Based on engineering judgement, the following success criteria were applied by BNL for the large LOCA case:

(1) Large LOCA break is above the core: Condensate pump injection of 1000 gpm is successful.

(2) Large LOCA break is below the core: Condensate pump injection of 1000 gpm is unsuccessful.

The basis for the judgement of adequate cooling in the first case stems from the assumption that the core will be covered in this case, and only steam will be able to discharge through the break. The steaming rate corresponds to the decay heat of the core which can be replenished by the 1000 gpm injection. The BNL judgement for case (2) is that the makeup capability of 1000 gpm to the hotwell[11] would not be sufficient for compensating the flow out of the break and steaming out of the assumed open ADS.

The success criteria for the different types of LOCA can be defined also in terms of system effectiveness rather than according to break size:

Large LOCA: No ADS is required. High pressure injection, as well as PCS, is inoperable. The condensate pump would be capable of about 1000 gpm for long duration, which is assumed insufficient for large break (Liquid) (e.g., larger than 10"ϕ).

Medium LOCA: ADS is needed as well as HPCI, but RCIC is not an effective injection mode. The effectiveness of PCS is unclear, and two assumptions are used in the sensitivity study (see Section 5.3 in Table 5.15; the impact is seen to be small). The baseline gives credit for PCS in medium LOCA for both injection and long term heat removal.

Small LOCA: ADS is needed, and RCIC is effective as well as the PCS.

The LOCA initiating events were further subdivided to LOCAs inside and outside drywell. The latter include the following:

1. Steamline or main feedwater breaks outside containment (within the reactor building).

2. Breaks in the HPCI/RCIC steam supply or pump discharge lines.

3. Interfacing LOCAs in low pressure systems.

The success criteria for these cases remain unchanged.

2.1.2.2 Success Criteria for Transient Initiators

The success criteria for transient initiating events (with successful scram) for the SNPS-PRA, given in Table 2.5, are similar to those for the LGS and RSS-BWR, with the following exceptions.

1. For transient initiators, the RSS-BWR applies the small LOCA success criteria given in the FSAR. It is noted in RSS, (page I-67) that these criteria were selected in attempt to be conservative. The SNPS and LGS use more realistic analysis (deterministic analysis performed by the vendor) as their basis.

2. The RSS-BWR requires operation of four ADS valves out of five for depressurization following a transient in which low pressure injection systems are required; the LGS requires only two out of seven; the SNPS requires three out of seven. These differences have little impact on ADS unavailability because the dominant contributors are loss of nitrogen supply, maintenance, calibration errors, and other commonalities of all ADS valves.

The more realistic success criteria used in the SNPS-PRA for the transient initiators are considered reasonable on the basis of NEDO-24708[4]. One exception is the assumption that RCIC is capable of supplying adequate vessel water makeup to an isolated reactor with two stuck open relief valves. The validity of this assumption remains to be verified.

The BNL assessment assumes that in the case of a transient with coincident two stuck open relief valves (2 SORVs), RCIC would not be effective, the reactor will depressurize in less than 2 hours, and low pressure injection will be required later on. This is similar to the medium LOCA case. However, relatively more credit to the PCS is given in the transient with 2 SORVs sequences.

2.1.2.3 Success Criteria for ATWS Initiators

This section presents the SNPS-PRA and the BNL reassessed success criteria for ATWS initiators. There are no comparable criteria for the RSS-BWR since ATWS was not evaluated in as much detail.

Table 2.7 gives the Shoreham ATWS success criteria for six initiators, listed in the first column. The other columns indicate the failure of various mitigation functions, with "A" denoting an acceptable condition and "N" an unacceptable one. These success criteria are derived from a GE report[5] and a KMC letter[10].

BNL reviewed these two documents to determine the applicability and the reasonableness of the results as they relate to SNPS. The GE report was prepared on a generic basis, analyzing the BWR-4 Mark I plant, with the assumption of an automatic SLC system that can deliver 86 GPM of boron to the core upon actuation. This is to be compared with the Shoreham design in which SLC initiation is manual and the maximum boron injection rate into the core is 43 GPM. Given the critical nature of the SLC initiation time and the amount of boron that can be injected into the core, the GE report provides only limited insights in the determination of the SNPS ATWS success criteria.

The KMC letter gives the results of an analysis modeling a generic BWR-4 reactor with a Mark I containment. It also includes some sensitivity results on the effects upon suppression pool temperature of 43 GPM versus 86 GPM SLC system injection rate and of the time delay in initiating the SLC system. It discusses the reasoning behind the selection of a maximum suppression pool temperature limit of 285°F. This limit should be contrasted with the 240°F cited in the SNPS-PRA, where 240°F is considered to be an unacceptable plant condition. Both documents assume in their calculation that the RHR system would be operational within a short time, in the range of 3 to 11 minutes.

Because of the lack of detailed results of the ATWS analysis, BNL can provide only limited assessment of the adequacy of the ATWS success criteria. Revisions made to the criteria are based on the two documents used in the SNPS-PRA and on engineering judgment. SNPS plant specific information in these areas and additional information pertinent to the determination of these criteria could potentially affect the results.

The revised set of ATWS success criteria given in Table 2.8 is basically the same as that of the SNPS-PRA except for two areas. The first is the success of the decay heat removal system. The SNPS criteria indicate that since the condenser is available, the operability of the RHR should be optional. BNL is of the opinion that the information in the two referenced documents does not provide enough detail to support the assumption that the condenser with one or no RHR loops is sufficient to maintain suppression pool temperature for a turbine trip event. If there is immediate feedwater runback and the reactor power level is reduced quickly, by lowering the water level, to below the maximum condenser limit without a MSIV closure, the SNPS-PRA criteria appears to be reasonable. If, however, feedwater runback does not occur immediately or if the water level is maintained high, then excessive heat (for which containment heat removal needs to be provided) would be discharged into the suppression pool, making the success of RHR loops critical. In the BNL revised criteria, failure of any RHR loop is assumed to be an unsuccessful sequence.

In a related way, the SNPS loss of feedwater ATWS success criteria stipulated that failure of one RHR loop is considered to be a successful event. In this case, feedwater is automatically terminated by the initiating event, and the reactor power can be accommodated by the condenser only if the water level inside the vessel is further lowered; otherwise, the power level may still be a few percent above the condenser limit. BNL also assumed in the re-assessment of accident sequences that all RHR loops must be operational for containment heat removal purposes.

BNL also considers the results from analyses insufficient to justify the allowed SLC initiation time of 2 to 30 minutes; in fact, evidence appears to indicate the contrary. BNL assumes that if the SLC system is initiated within a 10 minute period, then the accident sequence is considered successful.

A discussion of the physical analysis performed for an ATWS accident sequence appears in Appendix 5D.3.

2.1.3 Support Systems

Each of the main systems supporting the frontline systems in the SNPS-PRA, listed in Table 2.9, is briefly discussed here.

2.1.3.1 Electric Power System (EPS)

Three subsystems of the EPS are considered in accordance with their impact on frontline systems:

1. Offsite Power: SNPS has three incoming offsite transmission lines. It has two separated switchyards.

2. AC emergency power subsystem of the EPS: The SNPS-PRA analysis is based on the availability of three diesel generators and a gas turbine without black start onsite*, available to supply power to three emergency AC bus divisions, but only two divisions supply most of the redundant safety systems as division III basically supplies power to two out of four LPCI, SWS, and RHR pumps.

3. DC-EPS: Three DC divisions with batteries are provided, but division III supplies two out of four RHR or LPCI actuation only.

The EPS for SNPS, LGS, and RSS-BWR are compared in Table 2.10.

2.1.3.2 Emergency Service Water (ESW)

Apparently, the LGS ESWS has more redundancy than the SNPS-PRA SWS, as shown by the partial comparison in Table 2.4; other backup systems are available in the plants such as normal NSW in LGS and TBSWS in SNPS.

2.1.3.3 Plant Air and Compressed Nitrogen Systems

The redundancy of the plant air and nitrogen systems in the SNPS and LGS is comparable with that in the RSS-BWR, as seen in Table 2.4.

2.2 Initiating Events

This discussion of the initiating events that could challenge the safety systems is divided into three parts. The first describes the approach used in the SNPS-PRA, the second compares this with the LGS and RSS-BWR approaches, and the third presents the results of the BNL review with respect to the choice of initiating events.

The SNPS-PRA considers four general classes of initiating events:

a. Loss-of-Coolant Accidents (LOCAs),

b. Transients with successful scram,

c. Anticipated transients without scram (ATWS),

d. Other low frequency accident initiators.

2.2.1 SNPS Initiators' Selection

2.2.1.1 LOCA Initiators

The LOCA initiators are subdivided into three groups according to the equivalent size of the break and the corresponding success criteria for the frontline systems:

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*The onsite AC emergency power subsystem has been upgraded since the SNPS-PRA was prepared. The BNL review refers to the original configuration.

a. Large LOCAs - equivalent break size diameter about 4 in. or more, for liquid or steam breaks.

b. Medium LOCAs - 1 in. < equivalent diameter < 4 in., for liquid break; 1.7 in < equivalent diameter < 4 in., for steam break.

c. Small LOCAs - equivalent break size diameter about 1 in. or less for liquid break and about 1.7 inch or less for steam break.

The LOCA initiators are further subdivided into two groups, by break location: outside drywell within reactor building, and within drywell.

2.2.1.2 Transient with Successful Scram

The transient initiators for which scram is successful are divided into seven groups, where the transients in each group impose the same success requirements on the frontline systems.

1. Transients that result in turbine trip.

2. Transients caused by MSIV closure which lead to isolation of the reactor vessel from the main condenser.

3. Transients following loss of feedwater flow.

4. Transients resulting from loss of condenser.

5. Transients resulting from loss of offsite power.

6. Transients resulting from inadvertent open relief valve (IORV).

7. Orderly and controlled manual shutdown.

The transient initiators in these groups were obtained from an EPRI survey[6] of operating experience with BWRs in which 37 were identified. These are listed in Table 2.11 and categorized into the first six groups.

This categorization of the transient initiators has been reviewed and is considered acceptable. A recent change in the SNPS control logic (for ATWS purposes) helps to show the advantage of the more detailed grouping of the isolation initiators. The MSIV closure set point has been moved from reactor level 2 [10 ft above top of active fuel (TAF)] to reactor level 1 (2 ft above TAF). As a result the frequency of a transient with subsequent MSIV closure on low level may decrease because more time for operator recovery actions would be available. The separation of isolation transients into MSIV closure, loss of feedwater, and loss of condenser events allows a more realistic modeling of feedwater recovery between level 2 and level 1. Credit for such a change in the control logic could hardly impact the plant PRA unless the number of transient groups is increased to differentiate between the various isolation transients.

The MSIV closure transient is a more severe challenge than turbine trip or loss of feedwater flow. On the other hand, as will be seen in Section 5.2,

loss of condenser is more severe than MSIV closure. The groupings resulted in a smaller contribution from the isolation transients, because the more severe loss of condenser transient has only one third the frequency of isolation transients. This grouping allows also for more meaningful feedback from LERs.

2.2.1.3 ATWS: Anticipated Transient Without Scram

If the reactor protection system fails to scram the reactor after an initiating event in any of the first six transient groups, then an ATWS results. Six groups of ATWS initiators were, therefore, considered.

1. Turbine trip ATWS

2. MSIV closure ATWS

3. Loss of feedwater flow ATWS

4. Loss of condenser ATWS

5. Loss of offsite power ATWS

6. IORV ATWS.

For the ATWS sequence evaluation and quantification, initiators 2 and 4 were eventually combined.

The completeness of the list of initiating events considered in the SNPS-PRA was evaluated by comparisons with the Reactor Safety Study[1] and other BWR-PRAs[2,7,8,9].

2.2.2 Comparison with Reactor Safety Study and Other PRAs

2.2.2.1 Comparison with RSS-BWR

In the RSS, all transient initiating events were grouped together and a single event tree was developed. The 15 likely transient initiators considered in the RSS (Table 2.12) are all included in the SNPS-PRA list. Worst case assumptions were made about the required responses and availability of the frontline systems in the single transient event tree of the RSS; the SNPS-PRA approach of creating seven groups of transient initiators is a more realistic approach. Furthermore, in the RSS, a failure to scram leads directly to core damage, whereas, in the SNPS-PRA, each failure to scram is classified into one of the ATWS groups and a detailed plant response is considered. In this regard also, the SNPS-PRA is more realistic than the RSS.

For the LOCA initiators, the SNPS-PRA considers three groups according to the equivalent break size, as does the RSS. Interfacing LOCA is considered in the SNPS in greater detail than in the RSS-BWR. Additional attention is given to the effects of LOCA in the reactor building (see Appendix 5C.2).

The reactor vessel rupture initiator is handled the same way in both studies. That is, large and medium-size ruptures are considered to be among the large and medium LOCA initiators, respectively, and massive reactor vessel

ruptures are considered to be within suppression pool capability in most cases and cause it to breach with a small probability. BNL did not review this initiator frequency.

Thus, overall, the handling of the initiating events in the SNPS-PRA is more detailed and realistic than in the RSS.

2.2.2.2 Comparison with RSSMAP Grand Gulf[7]

The Grand Gulf study considered two transient initiator groups, one consisting of the loss of offsite power and one covering all others. A single event tree was then used to model the plant response to the two transient initiating events considered.

LOCA initiators were first partitioned according to two break sizes and then a single event tree was developed to represent the entire spectrum of break sizes.

It follows that the SNPS-PRA treatment of initiating events is more detailed and realistic than that of the Grand Gulf Study.

2.2.2.3 Comparison with the Big Rock Point (BRP) PRA[8]

In the BRP study, the selection of initiating events was based on a review of plant and industry experience for precursors to significant accident sequences. Failures that would require an active response of the plant were classified as transients, loss-of-coolant accidents, or anticipated transients without scram. External events, although treated in the BRP study, are not included in the comparison in order to be consistent with the scope of the SNPS-PRA. Table 2.13 shows the initiating events for which BRP event trees were developed and their frequencies.

For the initiating events considered in the BRP PRA and not treated separately in many past PRAs, the following remarks are made:

- Loss of instrument air initiator. This was given a frequency of 6×10^{-2}/yr and was found to contribute less than 5% to the total core melt frequency in the BRP PRA. In the SNPS, failures due to loss of compressed air are treated in the system fault trees. The use of accumulators for providing at least five ADS actuations for each SRV valve and the use of backup air supply resulted in system unavailability of $\approx 3\times10^{-4}$, which contributes $\approx$7% to core damage frequency, in both the PRA and the BNL review.

- Steam line break outside containment. According to the RSS, the associated accident sequences leading to core damage are several orders of magnitude smaller than that of the sequences covered in the large LOCA tree. In the BRP PRA, it is 0.2% of the total core damage frequency. In the SNPS-PRA these sequences are studied in detail (see Appendix 5C.2), and they contribute only $\approx$0.02%.

2.2.2.4 Comparison with LGS[2] and GESSAR[9] PRAs

These two PRAs include more detailed selection of initiating events than do the RSS-BWR, Grand Gulf, and BRP-PRAs discussed previously, yet the SNPS-PRA includes all the initiating events of these two PRAs. In particular, the following are considered in greater detail than in the LGS-PRA, and in many cases, also in the GESSAR PRA.

a. LOCAs:

   1. Interfacing system LOCA is treated in detail.

   2. A treatment of steam line or main feedwater breaks outside containment (within the Reactor Building).

   3. A treatment of breaks in the HPCI/RCIC steam supply or pump discharge lines.

b. Transients with Successful Scram

   Isolation transients were separated to:

   1. MSIV closure.

   2. Loss of feedwater flow.

   3. Loss of condenser.

c. Transient without Scram

   1. Loss of feedwater flow was treated separately from other isolation ATWS.

d. Other Low Frequency Accident Initiators:

   1. Loss of a reference leg leading to loss of measured water level.

   2. Loss of drywell cooling.

   3. Loss of a DC bus.

   4. Loss of the service water system.

   5. Reactor building elevation 8 flooding following a postulated release of excessive water.

Like other BWR PRAs, the SNPS-PRA does not discuss the failure of RCP seal following a station blackout.

2.3 BNL Assessment of the SNPS-PRA Initiating Events and Success Criteria

As seen in the preceding section, the SNPS-PRA has gone into great detail in the selection of initiating events. This has resulted in a more realistic analysis that more closely follows the progression of the accident sequence.

It avoids the need to assume mitigating systems failure based on the worst case response to the most severe initiator within a lumped group of initiators. Furthermore, the addition of special treatment of low frequency initiating events improves the insight into the sources of the contributors to core damage frequency in this plant. This last group of separately treated initiators is responsible for one-fourth of the SNPS core damage frequency.

BNL has accepted the list of initiating events and grouping of the SNPS-PRA without significant changes. The increased detail in the initiators required a similar increase in the use of data and modeling to determine the frequencies of the initiating events and their course of progression. The SNPS approaches to accident sequence definition and data assessment are the subject of the next two sections and are given along with the BNL comments and independent assessment.

BNL, in general, accepted the success criteria used by the SNPS-PRA. The same frontline and support systems used by SNPS are also used in BNL's reassessment described below. Note that credit for the CRD system was not taken in either assessment even though it might be shown to be a conservative assumption. However, the impact on core damage frequency is small, as seen from Table 5.15, which shows the impact of credit given to CRD system.

The changes made by BNL with respect to success criteria are the following:

1. RCIC is assumed incapable of preventing core uncovery in case of two stuck open relief valves.

2. HPCI is successful for two hours in the above case, but later only low pressure injection will be effective. However, at that time ADS would not be required.

3. In the original Table 2.5, which is taken from the SNPS-PRA, it is stated that condensate injection or PCS would not be considered for Medium or Large LOCA, but the corresponding SNPS-PRA event trees take some credit for these systems (see note 5 in Table 2.5). This credit results in some decrease of the Class IIIC core damage state (see Appendix 5C.1 and Table 5.15). BNL accepted this success criteria for small, medium, and large break LOCAs where the break is above the core. However, additional analyses need be provided to substantiate credit given in SNPS-PRA for the liquid line large breaks at or below core level. Also, the procedures for replenishing hot well inventory should be provided.

4. Failure of any RHR loop is assumed to be an unsuccessful ATWS sequence for turbine trip and loss of feedwater initiators. Additional SNPS plant specific analyses pertinent to the determination of the increase in suppression pool temperature during ATWS events could potentially affect these criteria.

5. SLC initiation time between 2 and 10 minutes is considered a successful ATWS sequence. Results of analysis are insufficient to justify the allowed time period between 2 and 30 minutes used in the SNPS-PRA.

2.4 References to Section 2

1. Reactor Safety Study--"An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants", WASH-1400, NUREG/75-014, October 1975.

2. "Probabilistic Risk Assessment Limerick Generating Station", Philadelphia Electric Company, Docket No. 50-352, 353, Rev. 5, September 1982.

3. "Anticipated Transient Without Scram for Light Water Reactors", U.S. Nuclear Regulatory Commission, NUREG-0460.

4. Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors, GE Report NEDO-24708, December 1980.

5. "Assessment of BWR Mitigation of ATWS", GE Report NEDE-24222, Vols. 1 & 2, December 1979.

6. "Anticipated Transients Without Scram: A Reappraisal, Part 3--Frequency of Anticipated Transients", EPRI NP-2230, January 1982 (SNPS-PRA used the previous edition of this report--EPRI NP-801, 1978).

7. Hatch, S. W., "Reactor Safety Study Methodology Application Program: Grand Gulf #1 BWR Power Plant", NUREG/CR-1659/4 of 4, October 1981.

8. "Consumer Power Company Probabilistic Risk Assessment of Big Rock Point Plant", October 1981.

9. "Probabilistic Risk Assessment BWR/6 Standard Plant", General Electric Company, Docket 50-447.

10. Knuth (KMC) to Graves (NRC), "Supplement ATWS Evaluations," letter dated December 2, 1982.

11. Private communication with LILCO personnel (1984).

Table 2.1 Safety Functions Required for Initiating Events

1) Render reactor subcritical
2) Protect reactor coolant system from overpressure failure
3) Remove decay and sensible heat from core
4) Protect containment from overpressure
5) Scrub radioactivity from containment atmosphere*

Table 2.2 Safety Functions for Shoreham Nuclear Power Station

1) Render reactor subcritical
2) Protect reactor coolant system from overpressure failure
3) High pressure injection of coolant into core
4) Depressurization
5) Low pressure injection of coolant into core
6) Drywell heat removal
7) Containment heat removal
8) Scrub radioactivity from containment atmosphere*

*Not considered in the review summarized in this report.

Table 2.3 Frontline Systems for Shoreham Nuclear Power Station

Safety Function	Frontline Systems
1) Reactor subcriticality	1) Reactor protection system 2) Recirculation pump trip 3) Alternate rod insertion 4) Standby liquid control
2) Reactor coolant system overpressure protection	5) 11 Safety relief valves (SRV)
3) High pressure injection	6) RCIC 7) HPCI 8) CRD* 9) Feedwater system with power conversion system
4) Depressurization	10) Automatic depressurization system (7 of the 11 SRVs used for this function) 11) Manual depressurization
5) Low pressure injection	12) LPCI 13) LPCS 14) Condensate pumps
6) Drywell heat removal	15) Drywell coolers 16) Containment sprays
7) Containment heat removal	17) RHR 18) PCS 19) Suppression pool
8) Scrub radioactivity from containment atmosphere	20) Suppression pool* 21) Containment spray*

*This system was not considered in the PRA front end analysis.

Table 2.4 Comparison of SNPS, LGS, and RSS-BWR Safety Systems

	SNPS	LGS	RSS-BWR
Power (MWT)	2436	3293	3293
Containment	MK-II (concrete with steel liner)	Mk-II (concrete with steel liner)	MK-I (free standing steel)
# Relief valves	11 SRVs	14 SRVs	11 SRVs
# Safety valves	---	---	2
RCIC	400 gpm	600 gpm	600 gpm
HPCI	4250 gpm	5600 gpm minimum	5000 gpm
LPCI	4 pumps, 10,000 gpm per pump with 2 loops	4 pumps, 10,000 gpm per pump with 4 loops	4 pumps, 10,000 gpm per pump with 2 loops
LPCS	2 loops, 4725 gpm per loop with 1 pump per loop	2 loops, 6350 gpm per loop with 2 pumps per loop	2 loops, 6250 gpm per loop with 2 pumps per loop
ADS Valves	7 SRVs	5 SRVs	5 relief valves
RHRHX	2, cooled by SWS	2, cooled by RHRSW	4, cooled by HPSW
EDG	3	4	4, shared by 2 units
RPS	Has ARI, RPT	Has ARI, RPT	Has RPT
SLC	2 pumps, manual actuation, 43 gpm per pump (one pump at a time)	3 pumps, automatic actuation, 43 gpm per pump (2 pumps at a time)	2 pumps, manual actuation
RHR	2 loops with 2 pumps (100%) per loop. Each provides 7700 gpm. Each loop serves 1 RHRHX.	2 loops with 2 100% pumps per loop. Each loop serves 1 RHRHX for each unit (i.e., shared between units)	---
HPWS	---	---	4 pumps, 100% each no cross-connection with other unit considered
ESW	2 100% loops with 2 50% capacity pumps per loop. Each pump 8000 gpm.	2 100% loops with 2 50% capacity pumps per loop. Shared between units.	1 100% pump per unit
FW and Condensate	2 turbine-driven feed pumps. 2 electric condensate and booster pumps.	3 turbine-driven feed pumps and 3 electric-driven condensate pumps.	3 turbine-driven feed pumps and 3 electric-driven condensate pumps.
Containment Sprays	Manually actuated, sprays either the drywell or wetwell.	Manually actuated, sprays either the drywell or wetwell.	Manually actuated, sprays either the drywell or wetwell.
Plant Air and Compressed Nitrogen	Compressed nitrogen plant air backup and accumulators (allowing five SRV actuations).	Compressed nitrogen plant air backup.	Compressed air and plant air backup and accumulators (allowing five SRV actuations).

Table 2.5[1] Summary of Success Criteria for the Mitigating Systems. Tabulated as a Function of Accident Initiators (LOCAs and Transients with Successful Scram)

	Success Criteria	
Accident Initiator	Coolant Injection	Containment Heat Removal
Large LOCA: Steam Break ≥ 0.08 ft^2 Liquid Break ≥ 0.1 ft^2	1 of 4 LPCI Pumps OR 1 of 2 Core Spray Pumps OR 1 Condensate Pump (5,7)	1 RHR
Medium LOCA: Steam Break 0.016 to 0.08 ft^2 Liquid Break 0.004 to 0.1 ft^2	HPCI OR {1 of 4 LPCI Pumps OR 1 of 2 CS Pumps OR 1 Condensate Pump[5]} and ADS[2]	1 RHR OR PCS[5]
Small LOCA: Steam Break <0.016 ft^2 Liquid Break < 0.004 ft^2	HPCI OR RCIC OR 1 Feedwater Pump OR {1 of 2 CS Pumps OR 1 of 4 LPCI Pumps OR 1 Condensate Pump} and ADS[2]	PCS OR 1 RHR OR RCIC in-Steam condensing mode[6]

Table 2.5 Continued

	Success Criteria	
Accident Initiator	Coolant Injection	Containment Heat Removal
Transient (Including Transient + 1 SORV)	Same as Small LOCA	Same as small LOCA
IORV	Same as Small LOCA	Same as small LOCA
Transient + 2 SORVs[3]	1 of 2 CS Pumps OR 1 of 4 LPCI Pumps OR 1 Condensate Pump } and ADS[4]	1 RHR OR PCS

[1]This is Table 1.5.2 of the SNPS-PRA, but includes corrections made according to their use in the PRA-event trees[5].

[2]ADS requires operation of only three safety/relief valves for adequate depressurization.

[3]This line added by BNL reviewers and is different from SNPS-PRA.

[4]Feedwater or HPCI and the ADS functions are required, in this case, only for the first 2 to 3 hours. After this, RPV pressure is assumed below 100 psi.

[5]These are corrections made to the original SNPS-PRA Table 1.5.2 based on the actual use in the PRA-event trees.

[6]This option, considered in SNPS-PRA, is not regularly used by SNPS. The effect of this change is given in Section 5.3, Table 5.15.

[7]BNL considered condensate pump injection unsuccessful for large LOCA because the replenishing capability of the hotwell is about 1000 gpm, which may not suffice.

Table 2.6 LOCA Success Criteria

Equivalent Break Size Diameter	SNPS Steam	SNPS Liquid	LGS Steam	LGS Liquid	RSS-BWR Steam	RSS-BWR Liquid
13.5 in. - 8.5 in. -	A* 1/4 LPCI or 1/2 CS or Condensate and 1 RHR		A* 1/4 LPCI or 2/4 CS and 1 RHR		A* For Injection 4/4 CS or 3/4 LPCI and 2/4 CS For Recirculation 1/4 CS or 1/4 LPCI and 1 RHR	
4.7 in. -					S1+ HPCI or 1/4 LPCI or 1/4 CS	(4 SRVs) ADS and 1 RHR
4.3 in. - 3.8 in. - 2.5 in. - 1.7 in. -	S1+ HPCI or 1/4 LPCI or 1/2 CS or condensate	ADS (3 SRVs) and 1 RHR or PCS	S1+ HPCI or 1/4 LPCI or 2/4 CS	ADS (2 SRVs) and PCS or 1 RHR	S2** HPCI or RCIC or 1/4 LPCI or 1/4 CS	1 RHR (4 SRVs) ADS and 1 RHR
1.0 in. - 0.85 in. - 0.6 in.-	S2** HPCI or RCIC or FW or 1/4 LPCI or 1/2 CS or Condensate and PCS or 1 RHR	ADS (3 SRVs)	S2** HPCI or RCIC or FW or 1/4 LPCI or 2/4 CS and PCS or 1 RHR	ADS (2 SRVs)	<S2** 1 CRD PUMP	

*A: Large LOCA. +S1: Medium LOCA. **S2: Small LOCA

Table 2.7 SNPS-PRA: Success Criteria for ATWS Accident Sequences Based on Modifications Implemented at Shoreham[a]

TRANSIENT INITIATING EVENT	EFFECT OF POTENTIAL ADDITIONAL FAILURES (In Addition to ARI Failure)								
	REDUCED OR LATE POISON INJECTION (b)	REDUCED COOLANT INJECTION		REDUCED SUPPRESSION POOL COOLING		PRESSURE RELIEF	OTHER ATWS FEATURES		
		FW	FW & HPCI	1 RHR	BOTH RHRs		RPT	ADS INHIBITED	HPCI OR FW SHUTS OFF AT LEVEL 8
MSIV CLOSURE	N	A	N	N	N	N	N	N	N+
TURBINE TRIP (e)	N (c)	A (d)	N	A	A	N	N	N	N+
IORV	N	A	N	N	N	N	A	N	N+
LOSS OF OFF-SITE POWER	N	A	N	N	N	N	A	N	N+
LOSS OF FEEDWATER	N	A	N	A	N	N	N	N	N+
LOSS OF CONDENSER	N	A	N	N	N	N	N	N	N+

A = Acceptable (Successful): acceptable implies no significant fuel damage and suppression pool temperatures less than 240°F.
N - Not Acceptable (Not Successful).
\+ These evaluations neglect operator action to stop the HPCI from overfilling the vessel. If such action were taken in 10 minutes after level recovery was apparent to the operator, successful shutdown would probably be maintained since the excess boron provided would be greater than the potential dilution.

(a) Combinations of failures not shown on the above table as acceptable should be considered unacceptable. These success criteria can be used to evaluate the successful states of the plant following an ATWS from less than 25% power. Note that RPT is not required for sequences from 25% power or less.

(b) SLC initiation is a manual operation which should be performed in the time frame of 2 to 30 minutes.

(c) SLC initiation may be delayed for a relatively long period of time (i.e., hours) if sufficient turbine bypass capability exists (i.e., ~ 25%) and feedwater can be controlled.

(d) Turbine trip is considered acceptable if the feedwater can be controlled at a relatively low flow so that steaming flow rates are below bypass capacity of 25%.

(e) All changes in recirculation flow outside of acceptable limits are treated as leading to a turbine trip as are all increasing feedwater flow transients.

Table 2.8 BNL Review: Success Criteria for ATWS Accident Sequences Based on Modifications Implemented at Shoreham[a]

TRANSIENT INITIATING EVENT	EFFECT OF POTENTIAL ADDITIONAL FAILURES (In Addition to ARI Failure)								
	REDUCED OR LATE POISON INJECTION (b)	REDUCED COOLANT INJECTION		REDUCED SUPPRESSION POOL COOLING		PRESSURE RELIEF	OTHER ATWS FEATURES		
		FW	FW & HPCI	1 RHR	BOTH RHRs		RPT	ADS INHIBITED	HPCI OR FW SHUTS OFF AT LEVEL 8
MSIV CLOSURE	N	A	N	N	N	N	N	N	N+
TURBINE TRIP[e]	N[c]	A[d]	N	N	N	N	N	N	N+
IORV	N	A	N	N	N	N	A	N	N+
LOSS OF OFF-SITE POWER	N	A	N	N	N	N	A	N	N+
LOSS OF FEEDWATER	N	A	N	N	N	N	N	N	N+
LOSS OF CONDENSER	N	A	N	N	N	N	N	N	N+

A = Acceptable (Successful): acceptable implies no significant fuel damage and suppression pool temperatures less than 240°F.
N = Not Acceptable (Not Successful).
\+ These evaluations neglect operator action to stop the HPCI from overfilling the vessel. If such action were taken in 10 minutes after level recovery was apparent to the operator, successful shutdown would probably be maintained since the excess boron provided would be greater than the potential dilution.

(a) Combinations of failures not shown on the above table as acceptable should be considered unacceptable. These success criteria can be used to evaluate the successful states of the plant following an ATWS from less than 25% power. Note that RPT is not required for sequences from 25% power or less.

(b) SLC initiation is a manual operation which should be performed in the time frame of 2 to 10 minutes.

(c) SLC initiation may be delayed for a relatively long period of time ____ if sufficient turbine bypass capability exists (i.e., > 25%) and Feedwater can be controlled.

(d) Turbine trip is considered acceptable if the feedwater can be controlled at a relatively low flow so that steaming flow rates are below bypass capacity of 25%.

(e) All changes in recirculation flow outside of acceptable limits are treated as leading to a turbine trip as are all increasing feedwater flow transients.

Table 2.9 Support Systems for Shoreham Nuclear Power Station

Frontline System	Support Systems
1) Reactor Protection System	1) AC/DC - Electric Power System (EPS)
2) Alternate Rod Insertion	1) AC/DC - Electric Power System
3) Standby Liquid Control	1) AC/DC - Electric Power System
4) Recirculation pump trip	1) AC/DC - Offsite Power
5) High Pressure Core Injection, HPCI	1) DC* - Electric Power System 2) Condensate Storage Tank
6) Reactor Core Isolation Cooling, RCIC	1) DC - Electric Power System 2) Condensate Storage Tank
7) Feedwater System	1) AC/DC - Offsite Power
8) Automatic Depressurization System (7 SRV's used for this function)	1) DC* - Electric Power System 2) Compressed Nitrogen System/Plant Air System
9) Manual Depressurization	1) DC - Electric Power System 2) Compressed Nitrogen System/Plant Air System
10) Low Pressure Core Injection, LPCI	1) AC/DC Electric Power System 2) Service Water System
11) Low Pressure Core Spray, LPCS	1) AC/DC Electric Power System 2) Service Water System
12) Condensate Pumps	1) AC - Offsite Power 2) Condensate Storage Tank
13) Residual Heat Removal, RHR	1) AC - Electric Power System 2) Service Water System
14) Power Conversion System, PCS	1) AC/DC - Offsite Power
15) Room Coolers	1) AC - Offsite Power (Manual Transfer to EPS) 2) Service Water System
16) Suppression Pool	-----
17) Containment Sprays	1) AC - Electric Power System 2) Service Water System

*ADS is dependent on the operation of one LPCI or LPCS pump, which is unavailable upon loss of AC power.

Table 2.10 Electric Power Systems

	SNPS	RSS-BWR	LGS
a)	Three diesel generators - No bus ties	Two diesel generators/unit - Inter unit bus tie	Four diesel generators/unit - no inter unit bus ties
b)	Three load divisions	Two load divisions/unit	Four load division/unit
c)	Three ESF divisions: - Two main divisions - One division for 2/4 LPCI and RHR	Two ESF divisions	Four ESF divisions
d)	Three 125 V DC Class 1E buses. Two of them feeding most ESFs. - No bus ties - One battery charger/battery	Four 125 V DC Class 1E buses between two units. - With bus tie - One battery charger/battery	Four 125 V DC Class 1E buses for each unit. - No bus tie - Two battery chargers/battery
e)	Two 138 KV and one 69 KV incoming lines - Two separate switchyards	One 230 KV and one 13.8 KV incoming lines - One switchyard	Three 500 KV and two 230 KV incoming lines - Two separate switchyards

Table 2.11 Summary of the Categories of BWR Transients Used in SNPS-PRA to Classify Operating Experience Data on Anticipated Transients*

	Transient Initiator	Group**
1.	Electric Load Rejection	T_T
2.	Electric Load Rejection with Turbine Bypass Valve Failure	T_C
3.	Turbine Trip	T_T
4.	Turbine Trip with Turbine Bypass Valve Failure	T_C
5.	Main Steam Isolation Valve Closure	T_M
6.	Inadvertent Closure of One MSIV (Rest Open)	T_T
7.	Partial MSIV Closure	T_T
8.	Loss of normal Condenser Vacuum	T_C
9.	Pressure Regulator Fails Open	T_T
10.	Pressure Regulator Fails Closed	T_T
11.	Inadvertent Opening of a Safety/Relief Valve (Stuck)	T_I
12.	Turbine Bypass Fails Open	T_T
13.	Turbine Bypass or Control Valves Cause Increased Pressure (Closed)	T_T
14.	Recirculation Control Failure -- Increasing Flow	T_T
15.	Recirculation Control Failure -- Decreasing Flow	T_T
16.	Trip of One Recirculation Pump	T_T
17.	Trip of All Recirculation Pumps	T_T
18.	Abnormal Startup of Idle Recirculation Pump	T_T
19.	Recirculation Pump Seizure	T_T
20.	Feedwater -- Increasing Flow at Power	T_T
21.	Loss of Feedwater Heater	T_T

Table 2.11 Continued

	Transient Initiator	Group**
22.	Loss of All Feedwater Flow	T_F
23.	Trip of One Feedwater Pump (or Condensate Pump)	T_T
24.	Feedwater -- Low Flow	T_T
25.	Low Feedwater Flow During Startup or Shutdown	T_T
26.	High Feedwater Flow During Startup or Shutdown	T_T
27.	Rod Withdrawal at Power	T_T
28.	High Flux Due to Rod Withdrawal at Startup	T_T
29.	Inadvertent Insertion of Rod or Rods	T_T
30.	Detected Fault in Reactor Protection System	T_T
31.	Loss of Offsite Power	T_E
32.	Loss of Auxiliary Power (Loss of Auxiliary Transformer)	T_T
33.	Inadvertent Startup of HPCI/HPCS	T_T
34.	Scram due to Plant Occurrences	T_T
35.	Spurious Trip via Instrumentation, RPS Fault	T_T
36.	Manual Scram -- No Out-of-Tolerance Condition	T_T
37.	Cause Unknown	T_T

*From EPRI-SAI Study[6].

**T_T - Turbine Trip

T_C - Loss of Condenser

T_E - Loss of Offsite Power

T_M - MSIV Closure

T_I - Inadvertent Open Relief Valve

T_F - Loss of Feedwater Flow

Table 2.12 BWR Transients (Reactor Safety Study Table I.4-12)

Likely Initiating Events

1. Rod Withdrawal at Power
2. Feedwater Controller Failure - Max. Demand
3. Recirculation Flow Control Failure (Increasing Flow)
4. Startup of Idle Recirculation Pump
5. Loss of Feedwater Heating
6. Inadvertent HPCI Pump Start
7. Loss of Auxiliary Power
8. Loss of Feedwater Flow
9. Electric Load Rejection (Turbine Valve Closure)
10. Turbine Trip (Stop Valve Closure)
11. Main Steam Line Control Valve Closure
12. Recirculation Flow Control Failure (Decreasing Flow)
13. Recirculation Pump Trip (One Pump)
14. Recirculation Pump Seizure
15. T-G Pressure Regulator Failure - Rapid Opening

Table 2.13 Initiating Events for BRP PRA for Which Event Trees Were Developed

Initiating Event	Frequency (per year)
Turbine Trip	1.4
Loss of Main Condenser	6.0×10^{-2}
Spurious Closure of MSIV	6.0×10^{-2}
Loss of Feedwater	1.6×10^{-1}
Loss of Offsite Power	1.3×10^{-1}
Loss of Instrument Air	6.0×10^{-2}
Spurious Opening of Turbine Bypass Valve	1.0×10^{-1}
Spurious Opening of RDS Isolation Valve	1.2×10^{-3}
Spurious Closure of Both Recirculation Line Valves	1.7×10^{-2}
Stuck-Open Safety Valve	2.6×10^{-4}
Interfacing LOCA	1.98×10^{-3}
High Energy Line Break in Recirculation Pump Room	3.9×10^{-7}
High Energy Line Break in Pipe Tunnel	3.8×10^{-6}
Small LOCA	1.0×10^{-3}
Medium LOCA	1.0×10^{-4}
Large LOCA	1.0×10^{-5}
Small Steam Line Break Inside Containment	1.0×10^{-3}
Medium Steam Line Break Inside Containment	1.0×10^{-4}

3. ACCIDENT SEQUENCE DEFINITION

The introduction to this section presents the general methodology used in the SNPS-PRA and an overview of BNL comments. Sections 3.2, 3.3, and 3.4, provide a discussion and the major conclusions of the review on the following topics: the SNPS-PRA accident sequence definition and the qualitative description of the event trees; the system fault trees that were used in the SNPS-PRA; and the various aspects of human performance analysis that entered into the risk assessment.

3.1 Introduction

3.1.1 The General Methodology

To assess the various accident sequences, i.e., the combinations of system failure events that, following the initiating events, lead to core damage, the SNPS-PRA used an approach based on the event tree and fault tree techniques. This approach differs, however, from that utilized in the Reactor Safety Study in the following way. In addition to using functional and systemic event trees and system fault trees, the SNPS-PRA employed three variations of these techniques, namely, the time-phased systemic event trees, the functional fault trees, and the functional-level event trees.

The logic employed in the SNPS-PRA for the definition of the accident sequences is as follows:

a) Twenty one functional event trees were developed for the different accident initiators (see Table 4.1) considered in the SNPS-PRA. A functional event tree depicts combinations of safety functions that can lead to a safe core condition or core damage, or constitute an initiating event for some other kind of potential accident. The SNPS-PRA functional event trees employ a finer safety function decomposition than that of the RSS functional event trees. For example, the coolant makeup function was decomposed into high pressure and low pressure makeup (see also Table 2.2). The combinations of the failed safety functions in these trees (tree paths) that can lead to core damage constitute the accident sequences for the SNPS-PRA. The quantification of each branch point in the functional event trees was done with the help of functional fault trees, functional-level event trees, and system fault trees. Table 5A.2 in Appendix 5A is an example of a functional event tree developed in the SNPS-PRA.

b) For certain functions in the functional event trees, functional fault trees were developed. In these latter trees, the top event is the failure of a particular function and this failure is further decomposed into failures of the frontline system which performs this function. For other functions in the functional event trees, functional level event trees were developed. These trees depict combinations of system successes and failures that can lead to a success or failure of the function in question. Figure 5E.2 is an example of a functional fault tree. Table 5A.1 is an example of a functional level event tree.

c) For some functions in the functional event trees -- those entailing systems that can be recovered (if failed) during the course of the accident

-- time-phased event trees were constructed in the SNPS-PRA. The headings of these event trees are the states of the involved systems at various instants of time, e.g., unavailability of AC power one half hour after initiation of the accident. This approach is equivalent to discretizing the recovery times of the various systems, and it thus allows for incorporation of recovery in the analysis. The main application is in the loss of offsite power event tree, e.g., Table 5B.1. BNL, in addition, used a time-phased event tree in the evaluation of the loss of service water system initiator.

d) Unavailabilities for some systems in the functional event trees, the functional fault trees, and the time-phased systemic event trees were obtained by developing system fault trees.

Functional fault trees, functional-level event trees, and time-phased event trees were employed in the SNPS-PRA to account for dependences among frontline systems (through shared hardware or common support systems) and to account for the possibility of recovery of systems that were unavailable at the initiation of the accident.

The various types of logic trees employed in the SNPS-PRA along with the modeling of human errors and of dependences are further discussed in Section 3.3 below. The functional event trees, in particular, are discussed in Section 3.2. Comments on the modeling of human performance, which has also been extensively used in part of the PRA, appear in Section 3.4.

3.1.2 Functional Event Tree Development

In general the functional event trees start with an initiator, followed by the subcriticality function. If the reactor is not subcritical, the sequence is transferred to the ATWS group of functional event trees. Transfers are made also to LOCA event trees or to other event trees for continuation. For sequences in which there is a successful insertion of control rods, other functions are evaluated, including adequate pressure control, coolant injection, depressurization, containment heat removal, and others. The end points of the functional event trees in the SNPS-PRA can be one of the following:

a) Successful shutdown and cooldown.	
b) Loss of coolant makeup core damage (for transients)	Class I
c) Loss of containment heat removal and drywell failure while coolant makeup is available to the core (All)	Class II
d) Accident sequences following LOCA resulting in core damage (LOCA)	Class III
e) Accident sequences involving failure to insert negative reactivity leading to a containment failure due to high containment pressure (ATWS)	Class IV
f) Unisolated LOCA outside containment resulting in core damage with drywell bypass	Class V

g) Transfer to other sequences which will then result in one of the above six end points.

A successful shutdown and cooldown is defined in the SNPS-PRA as conditions such that the reactor reaches hot stable shutdown. This is characterized by conditions such as: reactor is subcritical; pressure in the reactor is stabilized; temperatures in the fuel and reactor are within all limits; containment and suppression pool cooling are maintained; and reactor pressure vessel level is controlled.

3.1.3 Qualitative Dependence Analysis

This section provides an overview of the dependence modeling used in the SNPS-PRA and of the review comments and modifications by BNL. Detailed discussions on the quantification of these dependences appear in Sections 3.3 (fault-trees) and in the Appendices to Section 5, in which the quantification of the SNPS-PRA accident sequences is discussed.

Papazoglou et al.[1] give details on the various types of dependences, which can be classified as 1) functional, 2) physical, and 3) human induced dependences. Note that these are not mutually exclusive. A finer resolution yields the following six categories: 1) system functional dependences, 2) system physical dependences, 3) system humanly induced dependences, 4) component functional dependences, 5) component physical dependences, and 6) component humanly induced dependences.

3.1.3.1 System Functional Dependences

This type of dependence can be characterized by a functional relationship between two or more systems. Functional dependences due to "process coupling" (i.e., input-output relationships) are best modeled in the functional event trees. These dependences were in general properly addressed in the SNPS-PRA. Most noted examples are:

a) HPCI, RCIC dependence on suppression pool water temperature.

b) HPCI, RCIC, LPCI, and LPCS dependence on the ECCS equipment area temperature in Elevation 8 of the reactor building in case of LOCA outside containment.

c) Water level measurement system dependence on drywell temperature and reactor vessel pressure.

d) Failure of ADS safety relief valves due to excessively high drywell pressure.

No significant omissions were found in the PRA. In one case, however, a "process coupling" was assumed which is not correct in most incidents. The SNPS-PRA assumed that HPCI, LPCI, and LPCS would be initiated automatically by high drywell pressure (1.7 psi) or low reactor water level signal. This is true for LOCA or ATWS situations, but for most transients (all transients apart from loss of offsite power and loss of drywell cooling, which amount to approximately 4% of the total transient frequency) and for manual shutdowns it will take at least one hour after the initiation of the event (see Table 5F.1)

to reach the high drywell pressure setpoint (1.7 psi). Thus, in these events all ECCS injection systems depend only on the reactor four water level transmitters (N091A, B, C, D) for their automatic initiation*. Consequently, the miscalibration of these four transmitters would cause the automatic initiation failure of the high and low pressure systems following a transient (see Appendix 5A.1.4).

Another case of dependence included in the BNL review is the tripping of the drywell coolers. If these coolers are not recovered within 10 to 15 minutes, then the drywell temperature is expected to rise quickly, reaching a drywell pressure of 1.7 psi. This will cause the isolation of all drywell coolers, making recovery more difficult.

In the event that containment heat removal (i.e. RHR cooling of the suppression pool) is delayed for two hours or more, the drywell pressure is also expected to reach 1.7 psi, tripping the drywell coolers, and in about 15 additional minutes the drywell temperature is expected to rise to ≈300°F, which may be sufficient to impact level measurements if RPV should be at low pressure at that time. However, as shown in Appendix 5F, this dependence is of moderate significance.

Functional dependences due to "hardware coupling" were also treated in the SNPS-PRA. These dependences are best treated by combining all the system fault-trees of related systems, and subsequently performing Boolean reduction of the resulting "super tree." This has been done for <u>several functions</u> only in the SNPS-PRA (e.g., RCIC - HPCI - ADS: see Table 3.1 for complete list). The best way, as stated above however, is to combine <u>all the systems fault trees</u> on the same accident sequence leading to core damage, and perform their Boolean reduction. Treatment by this core damage fault trees (CDFT) approach was not done by SNPS-PRA. It was done in BNL past reviews[2,3]. In this BNL review, the CDFT approach was judged to be unnecessary because of the following features of the SNPS-PRA:

a) detailed treatment of functional dependences in the functional level event trees;

b) the Boolean reduction for some of the functions;

c) treatment of frontline system dependences on support systems such as
   - AC Power,
   - DC Power,
   - Service Water System, and
   - Drywell Cooling; and

d) transfer of support system unavailabilities during transients to initiators treating the loss of the support systems.

The most notable examples of functional dependences included in the SNPS-PRA functional event trees are the following:

*Shoreham is currently adding level instrumentation, and isolating the HPCI initiation from the other ECCS equipment. This may reduce significantly the probability of losing automatic initiation of ECCS.

a) Shared hardware between the low pressure coolant injection system (LPCI) and the RHR system.

b) Shared hardware between systems within the same function, such as HPCI with RCIC and LPCI with LPCS (shared water level instrumentation).

c) RCIC in injection phase and in steam condensing mode.

d) A system failure as part of the initiating event and its unavailability as a preventing frontline system. An example is the assumed unavailability of feedwater injection when the initiator was loss of condenser, and an increase in unavailability of PCS for this case.

e) PCS and the condensate pumps injection.

f) HPCI and RCIC failure due to loss of DC power 4 to 10 hours after station blackout.

As in to the case of "process coupling," BNL modifications to the functional event trees were related largely to the quantification of dependence. Most dependences were judged to be taken into account by SNPS-PRA. However, the degradation of the Power Conversion System (PCS or the W" function) due to feedwater system failure in the injection phase (the Q function) was not always treated consistently, or was not sufficiently supported. Because of the large number of transients, in almost any of which the recoveries of PCS and MSIV were evaluated to have somewhat different probabilities, BNL decided to employ functional level event trees using consistent sets of values for their quantification. This BNL approach to the treatment of dependence between Q and W" functions is discussed in Appendix 5A. It has some impact on the frequency of Class II core damage.

3.1.3.2 System Physical Dependences

This type of dependence was treated in the SNPS-PRA in an appropriate way. Important examples are the following:

a) The effect of loss of containment heat removal on drywell temperature and pressure, which affects other systems such as drywell integrity.

b) Loss of room cooling resulting from station blackout or loss of service water system.

c) Effect of flooding on ECCS systems located in reactor building.

No significant omission was found in the review.

3.1.3.3 System Human Induced Dependences

These dependences were also addressed to a limited extent. They include operator cognitive errors. Examples of dependences appearing in the SNPS-PRA are the following:

a) Failure to inhibit ADS in an ATWS event.

b) Failure to initiate feedwater runback in an ATWS event.

c) Failure to reduce water level and maintain it above level 1* in the case of ATWS.

d) Failure to depressurize, flood the reactor vessel and maintain level 3 in cases of conflicting water level measurement readings.

e) Delaying depressurization in blackout events. Maintaining pressure consistent with HPCI and RCIC operational pressure, and suppression pool temperature.

f) Failure to control HPCI and RCIC flow when level instrumentation is unavailable during a blackout event.

g) Failure to control condensate flow rate in case of a large LOCA.

It is seen that the SNPS-PRA in many cases included cognitive errors of operators (see also Table 3.2). Errors of commission, that is, the turning off of a system contrary to procedures, were excluded from the SNPS-PRA analysis, as in past PRAs. However, if control room information was unavailable or conflicting, a probability for cognitive error of commission was considered in the SNPS-PRA. An example is the act of early erroneous depressurization by the operator in the event of loss of a reference leg in the water level measurement system (see Ref. 4, page D-14, Figure D-6). BNL judged it reasonable to assume erroneous acts of commission when information is conflicting and when procedures exist which suggest depressurization in a case of other similar conditions. Appendix C of Ref. 4, which deals with core damage frequency contributed by the water level measurement system, deals mainly with quantification of this kind of dependency. BNL accepted many of these treatments (see Section 3.4 and Appendices 5E and 5F), sometimes with quantification changes, and in some instances with a model change.

3.1.3.4 Component Functional Dependences

This type of dependence was implicitly addressed in the SNPS-PRA in that the fault trees were developed up to a point where no functional dependence exists between the basic events (component failures).

3.1.3.5 Component Physical Dependences

The SNPS-PRA has included on some of the fault trees basic events related to physical dependence in the plant. Some examples of physical dependences in the fault tree analysis are the following:

a) Contamination of all SRVs' solenoid valves of the ADS system.

b) Suppression pool water unavailability due to common-mode failure clogged strainers.

*Level 1 (rather than TAF) is suggested in the SNPS-PRA and response #52 to BNL questions.[9]

3.1.3.6 Component Human Interaction Dependences

The SNPS-PRA includes a large number of miscalibration errors and maintenance errors. Miscalibrations appear on almost every fault tree and contribute significantly to system unavailabilities (some examples are shown in Table 3.3).

3.2 Qualitative Description of Functional Event Trees

The functional event trees used in the SNPS-PRA provide a logical method for developing and displaying accident sequences which may follow an initiating event. In the following subsections some of the more important functional event trees of SNPS are discussed qualitatively and the major modifications made by BNL are presented.

3.2.1 Turbine Trip (T_T) (Appendix 5A.1)

This type of transient presents the least challenge to the plant apart from manual shutdown. Both feedwater available and feedwater unavailable cases are considered in the event tree. The turbine trip functional event tree (see Table 5A.2) comprises thirteen safety functions. The failure of the first function, reactor subcriticality (C), results in an ATWS event which is more appropriately addressed in the turbine trip ATWS functional event tree shown and discussed in Appendix 5D. After the reactor has attained subcriticality, failure to accommodate the pressure surge caused by the transient due to failure of safety relief valves (SRVs) to open (M) is conservatively assumed to result in a large LOCA event. The success and the failure of the SRVs to reclose lead to two different, yet similar, sequence paths. Both branches are then evaluated for the high pressure system functions, viz., the feedwater function, Q, and the HPCI or RCIC function, U. The Q function in the SNPS-PRA includes different recovery assumption for each of the cases of the turbine trip--with and without two SORVs. The Q function is evaluated in the BNL review on the basis of the SNPS-PRA general approach and data, with a functional level event tree used to model the recovery of feedwater and the PCS in half an hour (see example in Table 5A.1). If the high pressure functions are successful, core damage may not occur, provided that the containment heat removal function is successful. If it happens that both high pressure functions fail, then, before the timely ADS actuation function, X, is evaluated, the SNPS-PRA provides the operator a "second chance" to recover feedwater. This is also included in the Q function of the BNL functional level event tree, rather than in the functional event tree as done in the SNPS-PRA. The ADS function automatically depressurizes the RPV upon reaching level 1. Next, the low pressure injection, V, is modeled, and can provide successful injection. This function in the SNPS-PRA is given in detail by the separation into LPCS, LPCI, and condensate injection. Failure of the containment heat removal function (W) or the low pressure injection or the timely ADS actuation function leads to core damage. The W function in the SNPS PRA has two subfunctions: RHR with RCIC steam condensing mode, W', and PCS, W". The PCS is included in BNL functional level event tree.

3.2.2 MSIV Closure/Loss of Condenser/Loss of Feedwater Transient (T_M, T_C, T_F) (Appendix 5A.3, 5A.4, 5A.5)

These types of transients lead to a more significant challenge to the plant than do the turbine trip transients. The MSIV functional event tree is identical in structure to that of the turbine trip because of the similarities in the required response of the safety functions of the plant to mitigate the events. The only difference between the two functional event trees resides in the reduced unavailabilities of the feedwater/power conversion system both for high pressure injection and for the long-term containment heat removal functions. This is due to the more significant challenge to the plant from a MSIV closure initiator, as noted earlier. The loss of condenser (T_C) is similar to MSIV closure, but has no recovery of FW in the short term, and low availability of the PCS in the long term. It is more severe than MSIV closure. The differences are more clearly seen from comparison of the functional level event trees of the BNL approach (Tables 5A.5 and 5A.9). Loss of FW (T_F) is the weakest challenge of the three, and the SNPS-PRA treats it also separately.

3.2.3 Inadvertent Open Safety-Relief Valve (T_I) (Appendix 5A.6)

This transient was treated separately because the operator must recognize the event and manually scram the reactor. Additionally, the containment conditions are different from those during other transients because of the higher total heat addition to the suppression pool at the time of plant shutdown, which places a more significant demand on the containment heat removal function. The SNPS-PRA also assumes that MSIV closure occurs in all IORV cases.

The principal distinction of this tree stems from the three branches depicted for the timely scram initiator function, c', c" (see Table 5A.12 in Appendix 5A.6). The top branch represents a successful timely scram in which no additional requirement is placed on the cooling of the suppression pool. The center branch denotes the scenario in which the reactor is scrammed prior to the suppression pool reaching a temperature requiring prompt RHR system operation and PCS recovery. The third branch is equivalent to failure to scram the reactor prior to exceeding the containment heat removal capability and is transferred to the ATWS event tree analysis. The feedwater/PCS system is not evaluated in this tree because operational data indicate that during an IORV event the MSIVs may close, thus causing all decay heat to enter the suppression pool. BNL gave credit to MSIV reopening for long term containment heat removal as in the cases of two SORVs or medium LOCA (see Appendix 5A.6).

3.2.4 Manual Shutdown (M_S) (Appendix 5A.2)

This event tree accounts for challenges to the plant resulting from a controlled manual shutdown -- not a scram but a manual control rod insertion in a slow, orderly manner. Examples of such shutdowns are scheduled or forced maintenance outages and refueling outages.

Operating experience indicates that because of the controlled nature of the transient, the SRVs are not challenged. Therefore, only the high pressure injection function, timely ADS actuation, low pressure injection function, and the containment heat removal functions are evaluated. Failure of the high pressure functions, and failure of the timely ADS actuation function, X, or

the low pressure injection function would lead to core damage. Failure of the containment heat removal function results in the loss of drywell. No changes were made to this event tree. However, a functional level event tree was prepared to treat the dependences between the FW/PCS system in the injection and containment heat removal phases.

3.2.5 Loss of Offsite Power (T_E) (Appendix 5B)

This transient provides unique initial conditions for accident sequences because of the loss of AC power and the resulting demand for the diesel generators. The initial condition of loss of AC power affects the majority of the frontline systems since AC power is needed for most plant systems. This tree has been time phased for the coolant injection and containment heat removal functions to account for recovery of AC power. BNL modified the SNPS-PRA event tree mainly with respect to containment heat removal, which was treated by BNL on the LOOP event tree rather than transferred to the MSIV closure event tree.

3.2.6 Comparison with the Treatment of Transients in RSS and LGS-PRAs

The transient event tree in the RSS (Figure I 4-16 in WASH-1400) was a single tree used by the RSS for all anticipated transients requiring reactor shutdowns from power operation.

The SNPS-PRA approach is considered to be a significant improvement over the one-transient event tree in the RSS. The use of separate event trees for MSIV closure (T_M)/loss of condenser (T_C)/loss of feedwater (T_F) in the SNPS-PRA is an improvement over LGS-PRA. The RSS analyzed the loss of offsite power transient by using the same transient event tree. The SNPS-PRA added more detail over this simplified approach in its loss of offsite power (T_E) event tree. This is considered to be a significant improvement. The use of the T_I tree in the SNPS-PRA is another improvement over the RSS approach. The RSS concluded that these types of transients are insignificant to the frequency of core damage.

3.2.7 LOCA Event Trees (A, S_1, S_2) (Appendix 5C.1)

For the LOCA-initiating events the SNPS-PRA developed three functional event trees corresponding to the three break size categories (large, medium, small) as was done in the RSS. The small LOCA event tree is almost identical to the transient event trees, and in particular to the case of IORV. The medium LOCA is similar to the small LOCA; the only differences are that RCIC is not sufficient to prevent core uncovery, and that high RPV pressure decreases with time. In the case of large LOCA only low pressure injection systems can supply coolant injection.

The LOCA event trees used in the SNPS analysis are slightly different from those used in the RSS. The three event trees model the different effects on the reactor and the different success criteria required as a function of LOCA break size and location (liquid or steam break). The large LOCA event tree handles the breaks that depressurize the reactor, and the two smaller LOCA trees handle the breaks that do not cause immediate reactor depressurization.

The SNPS-PRA large LOCA event tree (shown in Table 5C.1) differs from the one used in the RSS. It contains the same systems and has the same structure as the RSS event tree with the exception of the containment leakage (G), and core cooling functions (F).

The medium LOCA and small LOCA event trees for SNPS-PRA (Appendix 5C, Table 5C.1) also differ from the RSS small LOCA event trees. Vapor suppression (D), and containment leakage (G) were eliminated since their effect is small and treated in the Containment Event Tree (CET). Since the plant's reaction to a small LOCA is similar to a transient, the small LOCA event tree resembles a transient event tree (IORV).

LOCA outside containment was considered in detail in the SNPS-PRA. The event tree is basicaly similar to that for a large LOCA. Only large LOCA was considered to be a significant problem because of the short time available for preventing core damage (Appendix 5C.2).

3.2.8 ATWS Event Trees (T_T, T_M, T_I, T_F, T_E) (Appendix 5D)

The SNPS-ATWS event trees handle those transients which do not result in successful scram. These trees include analysis of the five major transient groups (turbine trip loss of feedwater, MSIV closure, IORV, and loss of offsite power). Thus, there are five ATWS event trees:

1) Turbine trip - In the event of a turbine trip with failure to scram, two scenarios have been developed in the SNPS to model the plant response. The first case assumes that, given the turbine trips, the turbine bypass remains open. The condenser and feedwater are available. However, should the turbine bypass fail, or should feedwater trip off line or the condenser not be available, the SNPS-PRA assumes that the situation is similar to either a total loss of condenser heat sink or a MSIV closure or a loss of feedwater event. These second case events are treated in the respective ATWS functional event tree.

2) MSIV closure/loss of condenser - This group includes those transients that challenge the plant in a manner which results in a closure of all MSIVs or a loss of condenser. Also included are the turbine trips that were shown to result in either MSIV closure or loss of condenser.

3) Loss of feedwater - This initiator includes the events that are characterized by a loss of feedwater with condenser available. The events include loss of feedwater initiators and transfers from turbine trip and MSIV closure.

4) Loss of offsite power - The single initiator is loss of offsite power with ATWS.

5) IORV - The single initiator is inadvertent opening of a SRV with ATWS.

*Small LOCA event tree is similar to IORV with a successful early shutdown (Table 5A.12).

These types of ATWS event trees were not used in the RSS. The SNPS-PRA use of these trees yields a detailed analysis of ATWS mitigating function, and this constitutes a realistic, less conservative approach to the evaluation of the ATWS contribution to the core-damage frequency and to the total risk.

3.2.9 Other Event Trees

The SNPS-PRA studies several low frequency events in separate event trees, some of which were not studied either in the RSS-BWR or in past BWR-PRAs:

a) Loss of a DC bus (Appendix 5G.2)

b) Release of excessive water at elevation 8 (Appendix 5G.1)

c) Loss of service water systems (Appendix 5G.3)

d) Loss of drywell cooling (Appendix 5F)

e) Loss of a reference leg (Appendix 5E).

BNL made no significant changes in its revised trees for case (a). In all other cases the changes were significant and are discussed in the respective appendices to Section 5. The main changes made are listed below.

a) Time phase event tree treatment of the release of water at elevation 8.

b) Addition of functional level event trees for RBSWS and TBSWS recovery in the case of loss of service water transient. The event tree was revised and time-phased, and the "GOLX" function was removed because it is insignificant to this event. All these changes resulted in a simpler event tree.

c) For the case of loss of drywell cooling, the number of event trees was reduced by combining all the contributions from transient without isolation into the loss of drywell cooling initiator event tree. This was similar to the transfer by SNPS-PRA of the contribution from transients to the support system event trees, e.g., loss of the SWS tree.

d) The event tree for loss of drywell cooling was not changed significantly. In the BNL re-assessment the SNPS-PRA "O" function was omitted and only the "L" function preserved. Quantification changes were more significant.

e) The LOOP event tree with loss of drywell cooling was significantly changed. The SNPS-PRA included the G function, which seemed unwarranted for this case (Table 5F.4). This sequence was found to be a very important contributor to SNPS-PRA core damage frequency because of loss of almost all control room level information.

f) For the case of loss of reference leg, again significant changes were made. The event of random failure of an additional level measurement channel was separated into three constituents, which increased significantly the contribution from this branch of the event tree compared with the SNPS-PRA (Table 5E.2).

3.2.10 Summary of the Qualitative Review of Functional Event Trees

The SNPS-PRA presents a very detailed and elaborate study of the various types of accident sequences applicable to the SNPS plant specific conditions which could conceivably occur within the plant. BNL concurs with the overall approach used in the development of the functional event trees, and these trees are basically adopted in BNL's re-assessment of the majority of the sequences. In most cases only minor improvement were made, and basically the same structure was used in the BNL revised trees. On the other hand, quantification of accident sequences by BNL led to modifications in many of these trees, as discussed in Section 5. In a few cases the event trees' structure was revised more significantly, as discussed above and shown in detail in Appendices to Section 5. The most important such cases are the following:

a) ATWS event trees,

b) Release of water at elevation 8,

c) Loss of a reference leg,

d) Loss of service water system.

Comparison with past BWR-PRAs showed that a more detailed functional event tree analysis was performed in the SNPS-PRA for several low frequency events, most notably loss of drywell cooling and loss of a reference leg.

3.3 System Fault Trees

3.3.1 System Fault Trees Analysis in SNPS-PRA

The system level fault trees are compiled in a separate volume (Volume IV) of the Shoreham Nuclear Power Plant (SNPS) PRA. The cutsets for these fault trees are given in Appendix J of the PRA, along with the identification of the most important cutset contributors to each system. The data for the fault trees' quantification are provided in Appendix A.2 (component failure rate data), Appendix A.3 (human error failure rates), and Appendix A.4 (quantification of system unavailabilities due to maintenance). BNL reviewed this information along with the fault trees. The review of these data appears in Section 4.2; here only some more pertinent comments about the analyzed fault trees are presented.

The following system fault trees are given in the PRA:

1. Reactor Core Isolation Cooling (RCIC),

2. High Pressure Coolant Injection (HPCI),

3. Service Water (SW),

4. Standby Liquid Control (SLC),

5. Residual Heat Removal (RHR),

6. Reactor Building Closed Loop Cooling Water (RBCLCW),

7. Electrical Power: Emergency AC and DC,

8. Core Spray (CS),

9. Low Pressure Coolant Injection (LPCI),

10. Automatic Depressurization System (ADS),

11. Reactor Building Standby Ventilation System (RBSVS) and CRAC Chilled Water,

12. Feedwater (FW),

13. RCIC/Steam Condensing Mode (RCICSC),

14. Condensate,

15. Scram System*,

16. Diesel Generator*.

The BNL comments in Ref. 2 were used in the review of the SNPS-PRA fault trees. Hence, the following comments (see Section 3.3.2 below) refer in part to how, SNPS has taken into account the previous comments in the new SNPS trees, and indicate which recommendation of Ref. 2 are still in effect, as well as including comments generated in the present review.

The following systems have not been analyzed in detailed fault trees:

1. Plant Air and Compressed Nitrogen Systems: A subtree for this support system was developed as part of the ADS fault tree. The details are developed to the subsystem level rather than the component level as in the other system fault trees. This will be further discussed as part of the ADS tree.

2. Reactor Protection System*: The unavailability of the scram system is based on NRC studies[5] and the analysis made by SNPS in Appendix A.7 of the PRA. The small tree given in the PRA is not quantified and includes only a part of the RPS.

3. Diesel Generators*: A fault tree was constructed, but the analysis is based on the information gathered from LERs. This information is reviewed in Section 4.2.2, below, and hence this system is not discussed further here.

4. Drywell Coolers: This system was not included separately in the fault tree analysis. A probability of 1.0 for human failure to recover this system after a high drywell pressure isolation(actuated on 1.7 psi) was used by the SNPS-PRA and BNL. A probability of 0.7 for human failure to initiate this system after its isolation (on

*The quantification of these system fault trees was not used in the probabilistic analysis. The trees were constructed and included to provide supplemental information regarding possible systems interactions.

level 1) was considered by BNL whenever credit for this system was given in the PRA. A hardware failure probability of 6.6×10^{-4} was calculated based on a functional fault tree in the SNPS-PRA for use during transients.

5. Suppression Pool and Condensate Storage Tank (Supply of Cooling Water to Safety Injection Systems): The unavailability of these sources of cooling water is analyzed as a subtree mainly on the HPCI fault tree. The failure probability was calculated in the SNPS-PRA as $7.\times10^{-5}$ and $3.\times10^{-4}$ for suppression pool and condensate storage tank, respectively.

6. Containment Spray: The system was not analyzed separately, but it is a part of the RHR/LPCI system, which was modeled and analyzed. A probability of 0.05 for human failure to initiate was used whenever some credit was taken for this system; this is apparently higher than the probability of its hardware unavailability.

7. Turbine Building Service Water System: The system was not analysed separately.

It is stated in the PRA (Appendix J), that the system fault tree models were constructed by SNPS for general application without regard to any specific transient event sequence and therefore do not include transient dependences. The changes made to system unavailabilities due to the impact of the transient initiator or the specific event sequences are discussed in the presentation of the system level event trees in Appendices to Section 5 of this report. The result of the fault tree analysis performed by SNPS is summarized in Tables J.4-1 of the PRA, which is reproduced in columns 1 and 2 of Table 3.1. Column 3 shows BNL review results for the trees that were judged important and were reviewed in detail.

In general, most of the system fault trees appear to be reasonably complete and accurate, but BNL made some additions and modifications. These changes are discussed in the following subsection, and their quantitative effect is summarized in Table 3.1. Their impact on the core damage frequency is small, amounting to a few percent (see Table 3.1 and Section 3.3.3 for further details). The trees are resolved down to the component level. The level of resolution is determined by the availability of data and by the possibility that further resolution will uncover existing dependences. The level of resolution in the trees is consistent with state-of-the-art PRA practice. The fault trees were developed to allow each component either to operate as designed or to fail (no partial failure). This approach is conservative, but it is consistent with the present PRA state-of-the-art. The following items were excluded from the analysis of the failure of a component (or system) as being outside the scope of the PRA:

a) External events,

b) Sabotage,

c) Operator errors of commission,

d) Most location-dependent common-mode failures, such as fires, but location-dependent CMFs due to internal flooding were included[6] and are discussed in Appendix 5G.1.

Manual operation of coolant injection, if required, was assumed to have a 30-minute grace period. This appears to be justified by thermal hydraulic calculations[7]. For large and medium LOCAs and for ATWS events, however, less time is assumed for manual restoration of injection. The failure rates used in the fault trees were point values and were meant to represent the average over the plant lifetime (i.e., wear-in and wear-out rates were averaged into the failure rates). Note that the risk during the first year of plant operation may be higher than the average risk over the plant lifetime because of a higher initiator frequency and higher failure rate during the wear-in period. Failure rates are further discussed in Section 4.2.

The dependences within a system were treated by using the same alphanumeric designator for a component that appears several times in the tree. For systems within the same function, for example, HPCI and RCIC for the function of High Pressure Coolant Injection, this method was also used to allow for Boolean manipulation of functions. The SNPS-PRA, in general, properly used this method. BNL's review, however, found that this designation was not followed consistently in all cases, and changes were made to correct discrepancies as listed in the next section and in Table 3A.1.

In summary, the SNPS-PRA has made a good and detailed systematic fault tree analysis that provides a model of the system (as seen in the next section). The SNPS-PRA has provided analyses of several fault-trees in addition to those done in RSS-BWR and LGS-PRA. It will be shown in the next section that several BNL comments in the LGS-PRA review[2] were taken into account in the fault trees of SNPS.

3.3.2 Summary of BNL Modifications to SNPS System Fault Trees and Their Impact

The following is a list of the main modifications that were made to the SNPS fault trees and resulted in changes of the system unavailabilities. The unavailabilities derived in the SNPS-PRA, along with those suggested by this review, are summarized in Table 3.1. Appendix 3A lists all changes or comments on SNPS system fault trees recommended by the BNL review.

The BNL review of the system fault trees was based on comparisons with the LGS-PRA and information for FSAR. The review did not, however, go to a level of examining specific equipment differences that warrant a change in failure rates; only design features using generic failure rate data were considered.

3.3.2.1 Reactor Core Isolation Cooling (RCIC)

Several improvements were made in the SNPS-PRA fault tree. For example, the turbine subsystem, which is a dominant contributor to RCIC, was treated in some more detail. In doing so, however, sometimes lesser failure rates were used. The lube oil (turbine auxiliaries) in SNPS-PRA is an example of a case in which the failure rate was reduced by a factor of approximately 4, compared with that in the LGS-RCIC tree and SNPS-HPCI turbine auxiliaries subtree.

However, the event "loss of flow through turbine driven pump" remained overall quite similar in all cases. BNL increased the lube oil (turbine auxiliaries) unavailability in RCIC turbine subtree from $1x10^{-3}$ to $3.6x10^{-3}$ to make it consistent with the HPCI tree and LGS tree. To perform a study of whether there were specific equipment considerations to reduce this failure rate by a factor of approximately 4 was not considered to be within the scope of the review. Another distinct change in the SNPS RCIC fault-tree is the increase (approximately tenfold relative to a past PRA) in failure rate of sensor in the "false signal" failure mode that was given a value of $2.6x10^{-3}$ for 10 sensors. Thus, "false steam pipe area high temp signal" constitutes one third of RCIC unavailability. The high failure rate implies a low frequency surveillance test of these sensors, and further implies that a favorable change in this frequency or procedures may be able to decrease the RCIC unavailability. Investigating the exact nature of the difference between the past PRA and SNPS in quantifying the failure rate of these sensors was again considered outside the scope of the review.

Table 3A.1 of Appendix 3A lists the changes or comments on the RCIC fault tree. No one of them causes any significant change to RCIC unavailability. However, some of them are CMF of both HPCI and RCIC:

a. Changed name of miscalibration "too high" of level 8 trip sensors, to properly account for commonality with HPCI level 8 trip (RCIC No. 5 in Table 3A.1).

b. Steam leakage from HPCI or RCIC steamline may cause their isolation. ("HCOMMON" event included in the BNL review, RCIC No. 1). The effect of these common-mode failures of RCIC and HPCI is discussed in Section 3.3.2.4 below.

3.3.2.2 High Pressure Core Injection System (HPCI)

The turbine subsystem is modeled in detail, as is the automatic transfer from CST to suppression pool suction. The overall unavailability of HPCI is within a reasonable range. The SNPS-PRA is more realistic than the LGS-PRA by treating the probability of failure-to-start on subsequent starts as comparable with that on initial start. A factor of 1/3 was used in SNPS-PRA compared with 1/10 in the LGS-PRA. Table 3A.1 of Appendix 3A lists changes and comments on the HPCI fault tree. They do not impact significantly the HPCI unavailability, but have some impact on the HPCI/RCIC CMFs.

a) The failure of the shaft-driven lube oil pump, which is included on the fault tree, was also added to the list of cutsets resulting in a small increase of HPCI unavailability from 0.096 to 0.1 (HPCI No. 5).

b) The high drywell pressure signal to initiate HPCI was deleted for transient initiators (HPCI No. 1). Thus, miscalibration of water level sensor becomes a significant contributor to CMF of HPCI, RCIC and ADS not considered in the SNPS-PRA (see Section 3.3.2.4 for quantification).

c) The name of the miscalibration event of HPCI turbine pressure trip set point was changed to conform to the same RCIC event (HPCI No. 7).

d) "HCOMMON" included. See comment (b) for RCIC (HPCI No. 5).

3.3.2.3 Automatic Depressurization System (ADS)

The three comments of Ref. 2 were taken into account by improvements in the SNPS-ADS fault trees:

a) The common-mode failure of all ADS valve solenoids due to contaminated nitrogen gas supply was included in the SNPS-ADS tree ($1x10^{-4}$).

b) No credit is given to human action to recover nitrogen gas supply if main supply or accumulators were lost.

c) A common-mode miscalibration of all pressure sensors in CS and RHR discharge lines was assumed, but with reduced probability - $5x10^{-5}$ instead of $2x10^{-3}$. The $2x10^{-3}$ is for non-staggered calibration. For staggered calibration of different systems, the value of $5x10^{-5}$ seems to be realistic. In addition, this value is rightly multiplied by operator failure to initiate ADS manually (0.1).

On the basis of these improvements, BNL accepted this unavailability of ADS ($8.4x10^{-4}$).

The CMF miscalibration of level 1 was correctly denoted by the same name in HPCI and RCIC. The operator manual initiation was given a different name from the high pressure injection manual initiation, as expected. No changes were made to the ADS fault tree.

3.3.2.4 Boolean Combination of High Pressure Injection Function (U) and the ADS Function (UX).

The SNPS introduced this feature in its PRA to account for dependences between safety functions. Basically, the "super"-trees of several systems were evaluated in the SNPS-PRA and cutsets for the super-trees were examined. The results of this Boolean reduction were used in the event tree quantification. This diminished the need for the core damage fault tree (CDFT) approach which BNL has used in its past reviews.[2,3] However, the review of the Boolean combination of the U function (HPCI and RCIC) and of the UX function (HPCI, RCIC and ADS) revealed some significant omissions, which are discussed here.

U-Function

The results of the SNPS-PRA analysis are given in Tables J.4-16 of PRA-Appendix J. Only two CMF contributions to U are identified there:

a) Both HPCI and RCIC are unavailable because of maintenance (plant technical specifications require a shutdown within 12 hours). Failure probability = $1.4x10^{-4}$.

b) Failure of a level transmitter or miscalibration (high above level 8 set point), which causes the failure of HPCI and RCIC trip on high water level (L8) and leads to gross moisture carry-over in the steam supply lines, as well as damaging both HPCI and RCIC turbines.

The SNPS-PRA incorrectly estimated the probability of this CMF to be 1.36×10^{-3}. In our review only miscalibration was considered, leading to 0.2×10^{-3} (0.2 taken for operator error rather than 0.1 as in SNPS-PRA).

BNL added the following four commonalities:

c) Common miscalibration of level 2 transmitters leading to the failure of level 2 autoinitiation of HPCI and RCIC. The failure probability is $2 \times 10^{-3} \times 0.1 = 2 \times 10^{-4}$ (where the 0.1 is due to operator failure).

d) Miscalibration of level 8 trip sensors (below the nominal level 8 set point) leading to repetition of turbine pump trips on both HPCI and RCIC: $2 \times 10^{-3} \times 0.5 = 1 \times 10^{-3}$.

e) Miscalibration of turbine pressure trip set points for both RCIC and HPCI: $2 \times 10^{-3} \times 0.5 = 1 \times 10^{-3}$ (suggested by SNPS-PRA, see RCIC FTA, but not caluclated).

f) Steam leakage from HPCI or RCIC steam line causing their isolation - "HCOMMON" = 1×10^{-3}.

The SNPS-PRA summed up the commonalities of HPCI and RCIC to the total of 9×10^{-3} (see Table 3.1). This does not follow from Table J.4-16, where a total of only 7.8×10^{-3} is shown.* According to the six commonalities listed above, the total is 0.01. This is the BNL value for the "U" function.

UX-Function

The results of the SNPS-PRA analysis are given in Table J.4-17 of Appendix J. One CMF contribution of all three systems to UX was identified there, see (a) below, and two additional CMFs of two out of the three systems, see (b, c):

a) Loss of all Division I and II electric power supplies. Failure probability is 3.2×10^{-6}.

b) Combinations of dominant cut sets of HPCI with failures of level instrumentation, and operator actions which defeat both automatic and manual initiation of RCIC and ADS. Failure rate is 4.0×10^{-6}.

c) Combination of dominant cut set of ADS with failure to isolate HPCI and RCIC on level 8 (leading to carryover in the steam lines). Failure rate is 1.3×10^{-6}.

The total of CMF contributions becomes 8.5E-6, which is consistent with the values in the event trees. However, some additional contribution for other ADS cut sets combined with other HPCI and RCIC cut sets (failing independently) was not included.

*The combinations of dominant cutsets of HPCI and RCIC result in 6.3×10^{-3}.

The SNPS-PRA incorrectly estimated, however, the CMF of item (b). In this case HPCI is assumed to be initiated by high drywell pressure signal. This is true only for LOCA or ATWS. For transients* and manual shutdowns no high drywell pressure is expected in less than 1 hour after the incident initiation, and therefore initiation of HPCI will fail manually and automatically too. This increases this commonality (see item (b) above) by a factor of 5 ($2\times10^{-3}\times0.1\times0.1 = 2\times10^{-5}$).

In the judgment of BNL, given proper staggering procedures for level instrumentation, the value of 2×10^{-3} for miscalibration would be too high by a factor of 10 or more. Therefore, BNL did not change the UX quantification on the transient and manual shutdown event trees. The special case of miscalibration is not ignored, however, and is discussed in Appendix 5A.1.4. It is a significant contributor to core damage frequency, but it can be easily eliminated by appropriate procedures.

The calculated commonality of HPCI, RCIC, and ADS in the BNL review becomes:

HPCI/RCIC commonalities with ADS cut sets which are independent:	$7\times10^{-3}\times6\times10^{-4} = 4\times10^{-6}$
Item (a) - loss of all Division I and II electric power:	$= 3\times10^{-6}$
Item (b) - miscalibration of level instrumentation (corrected):	$= 2\times10^{-6}$
	9×10^{-6}

Where 7×10^{-3} and 6×10^{-4} are the unavailabilities of "u" and "x" respectively after items (a) and (b) are subtracted.

The event trees values were not changed to reflect this small increase.

3.3.2.5 Low Pressure Core Spray (LPCS or CS)

The core spray system is, in general, adequately modeled in the SNPS-PRA fault tree for this system. The small number of changes made by BNL tend to have counterbalancing effects, so that the LPCS unavailability remained unchanged in the BNL review (see Table 3.1). The main changes are as follows:

a) The LPCS system will not initiate on high drywell pressure in case of a transient sequence. When this is eliminated from the fault tree a new cut set appears, "HHU720DXI * (LHU500DXI + LHU600DXI)", which probability is 2×10^{-4}. (The LHU500DXI and LHU600DXI should be AHU199DXI, see Table 3A.1 LPCS no. 5.)

b) The probability of the event "suppression pool water unavailability due to clogged strainers" is incorrectly included in the SNPS-PRA analysis as 2.6×10^{-4}, which is correct for a single clogged strainer. In the BNL review, a value of 5×10^{-5} for CMF of all strainers is used, which is consistent with the SNPS-PRA HPCI fault tree. (LPCS No. 2)

c) The SNPS-PRA states that valves LMV05ADPI and LMV05BDPI are tested only during refueling rather than on a quarterly basis. This

*Apart from loss of drywell cooling and loss of offsite power.

increased their failure rate from $4x10^{-3}$ to $9.3x10^{-3}$ by adding $1.6x10^{-6}$/hr x 8760 hr x 3/4 x 1/2 = $5.3x10^{-3}$.

However, in Appendix J the LPCS unavailability was calculated on the basis of $4x10^{-3}$. This was corrected in the BNL review, which resulted in an addition of $1.5x10^{-4}$ to the LPCS unavailability.

Since these changes cause only 4% increase in the BNL re-quantification of the LPCS unavailability, the SNPS-PRA unavailability value was used also in the BNL review. Table 3A.1 in Appendix 3A describes the changes to the SNPS-LPCS fault tree.

3.3.2.6 Low Pressure Coolant Injection (LPCI)

The LPCI is, in general, adequately modeled in the SNPS-PRA fault trees. The small number of changes made by BNL tend to counterbalancing effects. As seen in Table 3.1, the BNL review practically did not change the LPCI unavailability. The main changes are very similar to those in the LPCS fault tree, discussed above:

a) The LPCI will not initiate on high drywell pressure in case of transient sequences (same as item (a) of LPCS).

b) CMF of clogged suppression pool strainers is included (same as item (b) of LPCS).

c) The operator failure to initiate manually the LPCI is assumed to be dominated by the failure of the operator to initiate ADS if it failed to initiate automatically (Table 3A.1, LPCI No. 2).

These changes (see also Table 3A.1 of Appendix 3A) did not result in any significant effect on LPCI unavailability. They do, however, affect significantly the unavailability of the low pressure injection function which combines both LPCI and LPCS, as discussed in the next section.

3.3.2.7 Boolean Combination of LPCI and LPCS (V_4)

The main contributors to the failure of LPCI and LPCS are miscalibration of all reactor vessel pressure transmitters (N097A, B, C, and D) of the LPCI and miscalibration of differential pressure transmitters (DPIs N005A and B) of the LPCS. They are not dependent if these channels are calibrated separately one from the other. However, miscalibration of all N091 level transmitters is a commonality of both systems, at least under conditions prevailing during transient sequences. This commonality was not included in the SNPS-PRA, as explained before. The commonalities of LPCI and LPCS are as follows (most of them are included in the SNPS-PRA list of Appendix J Table J.4-18):

a) Miscalibration of level transmitters and operator failure to initiate manually (mentioned above)	$2x10^{-4}$
b) CMF of clogged suppression pool strainers	$5x10^{-5}$
c) Suppression pool water unavailability due to maintenance (ZTM) or due to high water temperature (ZTK200KWI)	$2x10^{-5}$

d) Combinations of manual system shutoffs on high reactor vessel level with failures subsequently to restart the systems when needed — $3x10^{-4}$

e) Combinations of dominant cut sets of both systems ($3x10^{-3}$ x $2x10^{-3}$) — $6x10^{-6}$

Since these contributions sum up to a value only 7% less than the $6.2x10^{-4}$ used in the SNPS-PRA, the value was not changed in the BNL review.

3.3.2.8 Service Water System (SWS)

There are two service water systems:

a) Reactor Building Service Water System (RBSWS),

b) Turbine Building Service Water System (TBSWS).

Only the RBSWS was modeled in a fault tree. It is discussed here.

The SWS is a safety related system designed as a two-loop system, and the SNPS-PRA fault tree was constructed accordingly. The CMFs of both loops are the main contributors to SWS unavailability. The following main contributions to SWS unavailability were evaluated in the SNPS-PRA:

a) Both service water loops in maintenance	$1.4x10^{-4}$
b) Failure of all four SWS pumps	$3.5x10^{-5}$
c) Combination of excess leakage in one loop with failure to isolate the opposite loop	$0.3x10^{-5}$
d) Combination of one loop in maintenance with two pump failures in the opposite loop	$0.2x10^{-5}$
e) Loss of water supply to screen well	$3x10^{-5}$

These resulted in the unavailability of $2.1x10^{-4}$ for SWS in the SNPS-PRA.

BNL considers this analysis to be realistic apart from item (a), which is conservative (yet is right for inclusion in the initiating event frequency for SWS). The only change in the BNL review was the omission of item (e) because it is due to external events, which are excluded from the PRA scope. (This is recognized in note No. 1 on the SWS fault tree, but not carried out.) However, the fault tree also includes event WFL 480 HEI which was quantified as $5x10^{-5}$ and stands for "All pumps suction clogged." This event is not included in the SNPS-PRA list of cut sets given in Appendix J Table J.4-5, but it is included in the BNL review. Thus, the SWS unavailability in the BNL review is $2.3x10^{-4}$.

LERs[8] include precursors of the event of clogged strainers for all SWS loops' suction. A real event has not occurred in a BWR. The value $5x10^{-5}$ is judged to be conservative. BNL did not change this value because SNPS, being situated on Long Island Sound, is considered more susceptible to this failure mode than an average nuclear power plant.

Table 3A.1 in Appendix 3A shows the two changes to the SWS fault trees discussed above.

3.3.2.9 Residual Heat Removal (RHR) System

Even though a fault tree was separately developed for RHR, the SNPS-PRA does not present its cut sets in Appendix J. Another problem is that Table J.4-1 gives a value of 4.8×10^{-4} for RHR unavailability, which is inconsistent with SNPS-PRA functional event trees. This apparently arose from an error in the RHR fault tree (as explained below) which SNPS-PRA corrected in a later revision of the PRA and did not correct in Appendix J.

BNL review found the following contributors to RHR unavailability, based on the SNPS-PRA fault tree for RHR:

a)	Both pump loops in maintenance	1.4×10^{-4}
b)	Failure of all 4 RHR pumps	3.5×10^{-5}
c)	Suppression pool water is unavailable due to clogged strainers	5×10^{-5}
d)	Combinations of one loop in maintenance with two pump failures in the opposite loop	0.4×10^{-5}
e)	Both heat exchanger bypass valves fail open (valves F048A and B)	1.6×10^{-5}
f)	Both MOVs at RHR heat exchanger outlets fail closed (MOV 34A and B on the SWS side)	2×10^{-5}
g)	Failure of SWS system (maintenance of both loops and failure of SWS pumps are excluded because the turbine building SWS would be able to provide the cooling water)	5.5×10^{-5}

These contributions sum to 3.2×10^{-4} for the RHR unavailability. Using a 20-hour repair time with MTTR = 19 hours results in $3.2\times10^{-4} \times \exp(20/19)$ 1.1×10^{-4}. This value is used in BNL reassessment. The same value is also used by SNPS, but not enough information is included to support its derivation.

Two changes were made by BNL to the SNPS-RHR fault trees. These are detailed in Table 3A.1 of Appendix 3A.

Finally, it should be noted that the above RHR unavailability assumes either that PCS was available for several hours following an accident sequence or that RHR was initiated to cool the suppression pool during the first 10 hours after the initiation of an accident sequence. When these conditions are not met and suppression pool cooling starts 20 hours after a transient or LOCA initiation, the suppression pool temperature will reach temperatures above 200°F and the RBCLCW system would be needed to cool RHR pump seals in order to prevent their failure. This increases the RHR unavailability by 2×10^{-4} (to a value of 3×10^{-4} rather than 1.1×10^{-4}), if the operator is successful in

aligning the system. This dependency was not included because of its small impact on the overall Class II core damage frequency.

3.3.2.10 RCIC in the Steam Condensing Mode and RHR

No changes were made to the SNPS-PRA fault tree of RCIC in the steam condensing mode. The unavailability of this system is evaluated as 0.14. However, in the PRA this system is always used in the same function with the RHR. Thus, the Boolean reduction of the RHR and the RCIC in the steam condensing mode is of interest. This was not presented in the PRA Appendix J. The result of this Boolean reduction is given without its derivation in Table J.4-1. The value of 6.8×10^{-5} seems to be based on an earlier evaluation of RHR unavailability of 4.8×10^{-4}. The conditional failure probability of RCIC in steam condensing mode given RHR has failed is 0.4 [$= 6.8 \times 10^{-5}/(4.8 \times 10^{-4} \times 0.35)$].

The commonalities of RHR and the RCIC in the steam condensing mode are as follows:

a) The unavailability of the SWS (with credit to TBSWS) 5.5×10^{-5}

b) Both MOVs at RHR heat exchanger outlets fail closed (MOV 34A and B) 2.0×10^{-5}

c) Both RHR heat exchanger bypass valves fail open (valves F048A and B) 1.6×10^{-5}

The probability of independent failure of both systems is $1.4 \times 10^{-1} \times 3.2 \times 10^{-4} \approx 4 \times 10^{-5}$; when the 20-hour repair probability of exp(-20/19) is applied to the sum of the values above, the unavailability obtained is 4.5×10^{-5}. This is less by a factor of 0.4 than the RHR unavailability of 1.1×10^{-4}. The SNPS-PRA also applied a factor of 0.4 and used the value 4.4×10^{-5} for the function of RHR with RCIC in the steam condensing mode. The same value was used also in the BNL reassessment, based on the above discussion and derivation.

3.3.2.11 The Electric Power System (EPS)

The fault tree of this system includes two top events:

a) Loss of power from 480 V Bus Division I, II, or III. This was found by SNPS-PRA to be 1.4×10^{-4}.

b) Loss of 125 V DC Bus Division I, II, or III. This was found by SNPS-PRA to be 3.7×10^{-4}.

The unavailability of a DC bus can be estimated from operating experience. NUREG-0666 evaluates the loss of a DC bus as 6×10^{-3} per year, which is about 10^{-6}/hr. Thus, the unavailability of a DC bus evaluated in the SNPS-PRA represents a mission time longer than the 24 hours used in general in fault tree quantification. This is apparently so because the loss of a DC bus does not necessarily cause reactor shutdown in the SNPS, and the plant can continue to operate for a few days. However, the unavailability has very small impact on the fault trees of other systems. The effect of the loss of a DC bus is

evaluated as a separate initiating event in the SNPS-PRA, and this accident sequence is reviewed in Appendix 5G.2.

The BNL review did not change the fault tree for the EPS.

3.3.2.12 Feedwater System

The SNPS-PRA tree of this system was prepared in detail. A review of the tree with respect to previous BNL comments[2] shows that the fault tree has the features BNL considered important, such as the following:

a) Failure of the operator to start the mechanical vacuum pump if the SJAE is unavailable (quantified with 0.1 failure probability)

b) Common-mode miscalibration of both reactor level channels, causing a spurious level 8 trip of the feedwater system (2×10^{-3})

c) Most of the other BNL concerns[2]

The dominant contribution to the failure of the system is failure of the operator to control the system during long-term coolant injection. This was quantified as 2.5×10^{-2}, which amounts to 50% of the feedwater unavailability. The loss of the condenser vacuum is another important contributor (2.5×10^{-3}). On the basis of the above remarks, no significant changes were made to the feedwater system fault tree.

3.3.2.13 Condensate System

The SNPS-PRA developed a separate detailed fault tree for the condensate system. Unlike the feedwater system, the condensate system shows no clear relationship between the list of cut sets (Table J.4-15) and the fault tree. The main contributions to the condensate system unavailability derived from the PRA fault tree and cut sets in Appendix J are listed below, with some examples of inconsistencies:

a) The main contribution comes from the failure of the operator to provide long-term makeup water to the condenser (0.025). This does not appear on the fault tree.

b) Simultaneous failure of both condensate pumps or both condensate booster pumps ($\approx 4 \times 10^{-5}$). This appears on the feedwater system fault tree and is developed in a different way on the condensate system fault tree.

c) Flow control instruments fail to supply signal or supply false signal to train A and B. This contributes $\approx 4 \times 10^{-4}$ to the condensate unavailability. It appears on the fault tree but is not shown in the cut sets list.

d) Event "ERUPT" is considered in the fault tree and stands for "rupture of piping/heat exchanger." This 1.1×10^{-4} contribution is not in the cut sets list.

e) Loss of offsite power during the mission time for the system ($\approx 10^{-3}$). This item appears both in the fault tree and the cut sets list.

It is apparent that the value given in Table J.4-1 of the PRA (0.12 for the condensate unavailability) has an error. The unavailability is about 0.03. This unavailability is dominated by the operator error to provide long-term makeup water to the condenser.

In the BNL re-assessment, the system unavailability is also dominated by operator response. However, different values for the operator error are used for short-term responses. A value of 0.1 is assumed for failure of the operator to:

a) Control the flow rate of the condensate pumps so that it will match the rate of condenser makeup flow rate of about 1000 gpm.

b) Verify the successful initiation and operation of the condenser makeup from the Condensate Storage Tank (CST), which is automatic.*

3.3.2.14 Power Conversion System (PCS)

No fault tree is given for this system in particular. Major parts of this system are included in the feedwater and condensate systems fault trees. The PCS includes also the MSIV, the condenser, the turbine bypass, and the circulating water system. The feedwater and condensate system fault trees represent these additional systems by undeveloped events (which are not resolved to the component level).

The SNPS-PRA based the PCS unavailability on experiential data, which result in an unavailability of 1.1×10^{-2}. Using a recovery probability of 0.45 in 15 hours (repair with MTTR = 19 hours) it derived a value of 0.005 for PCS (see response No. 8 in Ref. 9).

In the BNL re-assessment, the fault trees for the condensate and feedwater were used to estimate hardware unavailabilities for the PCS:

a) MSIV hardware failure	0.0005
b) Circulating Water System hardwa e failure (including failure to run)	0.001
c) Condensate System Control failures contribution	0.0003
d) Condensate system pumps and valves failure contribution (including failure to run)	0.0003
e) Steam Jet Air Ejector or Mechanical Vacuum Pump	0.002
Total	0.004

*Mr. Dick Paccione (LILCO), Private communcation with BNL (1984).

This value is used in BNL functional level event trees for the evaluation of the long-term PCS unavailability (see Apendix 5A.1).

3.3.3 Summary of the Review of Fault Tree Analysis and its Impact on Core Damage Frequency

The BNL review did not result in significant changes to the front or support system unavailabilities. It concentrated on the cut sets of safety functions which combine several front systems. The review, also, did not significantly change the unavailabilities of the safety functions. In the latter case, however, the main contributors to the functions' unavailabilities were modified, i.e., failure modes other than those in the SNPS-PRA were found to be important in the BNL review. The changes are as follows:

a) In SNPS-PRA the "U" function is dominated by miscalibration "high" of level 8 transmitters (high above level 8 set point). In the BNL review this is a minor contributor, and the main contributions come from miscalibrating "low" the level 8 transmitters (below the nominal level 8 set point), and from miscalibration of the turbine pressure trip set points of both HPCI and RCIC.

b) In the SNPS-PRA the "UX" function is dominated by loss of AC power to Divisions I and II electric power supplies, failures of level instrumentation combined with HPCI and operator failures, and level 8 miscalibrated "high." The SNPS-PRA appears to include only some of the contribution to the core damage frequency from the combination of the "U" and "X" functions; proper evaluation of UX would increase the SNPS-PRA result. The "UX" function is seen (Section 3.3.2.4) to be about 50% independent failure of "U" and "X" in the BNL re-assessment, with the other 50% coming from loss of AC power, as in the SNPS-PRA, and from miscalibration of the level 1 instrumentation.

c) In the SNPS-PRA the "V_4" function is dominated by suppression pool failure to supply water. BNL found the miscalibration of level 1 transmitters to be the important contributor.

d) In the case of RHR combined with RCIC in the steam condensing mode, BNL found that, unless the turbine building service water (TBSWS) is given credit, the reactor building service water (unavailability = 2.3×10^{-4}) will dominate the unavailability of this function, and there is little to be gained from the RCIC steam condensing mode. The SNPS-PRA factor of 0.4 was obtained by BNL only with credit given to TBSWS (the SNPS-PRA gave credit to TBSWS in the case of loss of SWS transient, see Appendix 5G.3).

e) The event of miscalibration of level 1 and 2 N091A, B, C, and D transmitters, named "HHU720DXI," appears on the fault trees and affects the "UX" and "UV" functions for transient sequences. This important dependence was not addressed in the SNPS-PRA. Details are discussed in Appendix 5A.1.4.

The impact on core damage frequency of the fault trees modification is small. BNL major modifications affected the contributors to the unavailability of safety functions when combining several system fault trees. However, these changes had impacts that either increased or decreased core damage frequencies, so that the overall result did not change the SNPS-PRA estimation of core damage frequencies.

3.4 Human Performance Analysis

Two types of human errors (cognitive and procedural) can contribute to the unavailability of frontline systems and impact on core damage frequency. These are addressed in the SNPS-PRA[1].

3.4.1 Cognitive Human Errors

The SNPS-PRA explicitly modeled cognitive human errors in the event trees and in the fault trees. These human errors, with a description of the required action and the time available (or assumed) for action, are listed in Tables 3.2 and Table 3.3.

The BNL review in general agreed with the qualitative modeling approach to most cognitive human errors. BNL disagreed with the model, in only a few cases, the most notable being the "GOL" model of the SNPS-PRA, which BNL changed to a "GL" model (see Appendix 5F for details), and loss of a reference leg, for which BNL moved some cognitive errors to an earlier stage in the BNL event tree and thus affected the core damage frequency (see Appendix 5E for details). In many cases, however, BNL disagreed with the quantification of the human errors. Tables 3.2 and 3.3 include BNL quantifications* for comparison with the SNPS-PRA values where significant changes were made. Appendix C of Ref. 4 went into great detail in modeling potential cognitive errors in the analysis of SNPS water level measurement system and is discussed in the detailed review in Appendices 5E and 5F.

3.4.2 Procedural Human Errors

Procedural human errors contribute to system or component unavailabilities through routine procedures such as calibration testing and maintenance or normal plant operation. In most cases the SNPS-PRA followed the techniques recommended in NUREG/CR-1278[10] for their quantification. The BNL review concentrated on determining whether any procedural human errors were omitted in the analysis; their quantification was not part of the review. Tables 3.2 and 3.3 present the most important procedural human errors covered in the SNPS-PRA.

3.5 References to Section 3

1. Papazoglou, I. A., et al., "Probabilistic Safety Analysis Procedure Guide," NUREG/CR-2815, September 1983.

*The quantifications shown are for illustrative purposes. The appendices include the background for these quantifications.

2. Papazoglou, I. A., et al., "A Review of the Limerick Generating Station Probabilistic Risk Assessment," Brookhaven National Laboratory, NUREG/CR-3028, February 1983.

3. Hanan, N., et al., "A Review of BWR/6 Standard Plant Probabilistic Risk Assessment, Vol. 1, Internal Events and Core Damage Frequency," Brookhaven National Laboratory, NUREG/CR-4135P, May 1985

4. "Review of Shoreham Water Level Measurement System, Revision 1," S. Levy, Inc., SLI-8221, November 1982.

5. "Anticipated Transients Without Scram for Light Water Reactors," Nuclear Regulatory Commission, NUREG-0460, 1980.

6. Shiu, K., Sun, Y. Anavim, E., and Papazoglou, I. A., "A Review of the Accident Sequences Following an Excessive Release of Water at Elevation 8 of Reactor Building in the SNPS, Brookhaven National Laboratory, NUREG/CR-4049, April 1984.

7. Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors, GE Report NEDO-24708, December 1980.

8. Haried, J. A., Evaluation of Events Involving SWS in Nuclear Power Plants," Oak Ridge National Laboratory, NUREG/CR-2797, November 1982.

9. LILCO's Response to Questions on Shoreham Probabilistic Risk Assessment, Long Island Lighting Company, SNRC-1021, May 1984.

10. Swain, A. D., and Guttmann, H. E., "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," NUREG/CR-1278, October 1980.

Table 3.1 Point Estimates of SNPS System Unavailability Compared to BNL Review

System(s)	Quantified SNPS-PRA	Unavailabilities BNL Review
RCIC	6.87E-2	7.E-2
HPCI	9.63E-2	1.E-1
SERVICE WATER	2.12E-4	2.3E-4
STANDBY LIQUID CONTROL	1.05E-1	1.05E-1
RHR	4.83E-4	3.2E-4
RBCLCW	3.99E-4	
Electric Power*		
125 V DC	3.66E-4	3.7E-4
480 V AC	1.4E-4	1.4E-4
Core Spray	3.62E-3	3.6E-3
LPCI	2.68E-3	2.7E-3
ADS	8.56E-4	8.4E-4
RBSVS & CHILLERS	2.33E-4	
FEEDWATER	5.46E-2	--***
RCICSC	1.40E-1	1.4E-1
CONDENSATE	1.23E-1	--***
HPCI [A] RCIC**	8.99E-3	1.E-2
LPCI [A] Core Spray**	6.25E-4	6.2E-4
RHR [A] RCICSC**	6.8E-5+	4.4E-5+
HPCI [A] RCIC [A] ADS**	9.5E-6	9.E-6
HPCI [A] RCIC [A] LPCI [A] Core Spray**	4.0E-6	6.25E-6

*Failure of one of the three emergency divisions.
**"[A]" represent a Boolean AND operation denoting the simultaneous failure of two or more systems.
***The fault trees were used to obtain an estimate of the PCS hardware unavailability for long-term containment heat removal. BNL used 0.004 for PCS hardware unavailability and failure to run for ten hours.
+Include repair [exp(-20/19)].

Table 3.2 Human Errors Modeled in Event Trees

Symbol	Description of Required Action	Time Available for Action	Quantification*	
			SNPS PRA	BNL Review
Q	Feedwater Runback (ATWS)	15 minutes	0.3**	0.2**
CLI	Reduce reactor vessel water level during ATWS. The SNL value includes also failure to inhibit ADS.	minutes	---	0.19**
D	ADS inhibit during ATWS	minutes	0.5	(0.2)
C_2	SLC injection initiation (ATWS)	"***	0.11	0.15
C'	Timely manual shutdown of reactor (IORV)	≈ 1/2 hour	0.001	0.01
Q,W"	Recovery of FW and PCS, including reopening MSIV in the short and long term (Transient/LOCA)	- minutes - 1/2 hour - hours	various values**	various values**
V'''	Condensate pumps flow control and verification of proper water makeup to hotwell (Transients/small LOCA): (Large LOCAs or LOCA outside containment):	 1/2 hour minutes	 0.01 0.2	 0.1 0.2
X (Phase I)	Timely ADS actuation when high pressure injection failed (LOOP)	1/2 hour	0.02	0.02
X (Phase II, III)	Operator error in performing early depressurization (LOOP)	hours	0.1**	0.1**
X'	Maintaining reactor in depressurized conditions (LOOP)	hours	0.2	0.2
T	Successful cross tie of turbine building SWS given RBSWS failed (Loss of SWS)	1/2 hour	0.26	0.24+
L	Maintaining water level 3 in reactor vessel (loss of drywell cooling)+: (loss of offsite power-blackout conditions)+:	 hours 1/2 hour	 0.005 0.06	 0.001+ 0.05+
G	Recovery of drywell coolers or initiation of containment sprays (loss of drywell cooling)+	1/2 hour	0.05**	0.05**
X_H	Erroneous actuation of ADS (Loss of reference leg transient)	1/2 hour	0.01	0.01
H	Operator recognizes the need for manual initiation of injection (high and low pressure injection) (Loss of reference leg transient)	1/2 hour	0.062**	0.062**

* Only significant cases are shown. The values are failure probabilities. The quantified values are illustrative, and should not be used without the bases given in the Appendices of Section 5.

** Values are sequence dependent. One example is shown.

*** 30 minutes assumed available in SNPS-PRA; only 5 to 10 minutes in BNL review.

\+ Modeling changes were made in this case which have larger impact than the change in quantification.

Table 3.3 Major* Human Errors Modeled in System Fault Trees

Description of Required Action	Time Available for Action	Quantification in SNPS-PRA
HPCI/RCIC		
1. Manual actuation of HPCI upon failure of auto-start signal	1/2 hour	0.1
2. Miscalibration of all level transmitters	---	0.002**
3. Miscalibration of turbine pump trip exhaust pressure transmitters	---	0.002
4. Failure to control or shutoff RCIC/HPCI before water carryover upon failure of level 8 trip	minutes	0.1
5. Human error failure to transfer HPCI from CST to suppression pool in time, upon failure of auto transfer	1/2 hour	0.1
6. Manual actuation of HPCI upon failure of auto start (including auto starc not reset)	1/2 hour	0.1
ADS		
1. Manual depressurize plant given that automatic depressurization has failed	1/2 hour	0.1
LPCS		
1. Failure to manually start the LPCS pump given that it failed to start automatically	1/2 hour	0.1**
2. Same as LPCI items (3) and (4)		
3. Miscalibration of reactor pressure transmitters	---	0.002

*Only human errors which are included in the major cut sets of the systemic fault trees.

**Modifications were made in BNL review (see Appendices or Section 4.3)

Table 3.3 Continued

Description of Required Action	Time Available for Action	Quantification in SNPS-PRA
LPCI		
1. Manually start the LPCI pump given that it failed to start automatically	1/2 hour	0.1**
2. Manually open pump discharge valves in alternate discharge line (same as RHR)	1/2 hour	0.025
3. Operator fails to restart LPCI as water level decreases	1/2 hour	0.003
4. Operator manually shut off LPCI on high level during an accident	---	0.1
5. Miscalibration of differential pressure channels	---	0.002
Electrical Power		
1. Direct power to 480-V bus is not restored within 2 hours	2 hours	0.8
RHR		
1. Start suppression pool cooling when required, and correct valve misalignments during line-up of the system	hours	4×10^{-5}**
2. Manually open pump discharge valves in alternate discharge lines, given that normal discharge line valves have failed	hours	0.025
SLC		
1. Failure to manually initiate SLC	1/2 hour**	0.1
SWS		
1. Failure to manually initiate SWS pump upon failure of automatic initiation	1/2 hour	0.9

**Modifications were made in BNL review (see Appendices or Section 4.3)

APPENDIX 3A

CHANGES MADE TO SNPS-PRA FAULT TREES

The changes to the SNPS-PRA fault trees suggested by BNL are summarized in Table 3A.1 for each system, in the following order:

1. RCIC - Reactor Core Isolation System
2. HPCI - High Pressure Core Injection System
3. LPCI - Low Pressure Core Injection System
4. LPCS - Low Pressure Core Spray System
5. RHR - Residual Heat Removal System
6. SWS - Service Water System
7. RCICSC - RCIC Steam Condensing Mode
8. EPS - Electrical Power System
9. Feedwater System
10. Condensate System

The SNPS-PRA also includes the following systemic fault trees to which BNL made no modifications:

1. ADS - Automatic Depressurization System
2. SLC - Standby Liquid Control System
3. RBCLCW - Reactor Building Closed Loop Cooling Water
4. RBSVS and CRAC - Chilled Water
5. Feedwater System
6. EPS - Electrical Power System

Table 3A.1 BNL Changes in SNPS-PRA Fault Trees

System	No.	Page	Gate Name	Gate Type	Input Name	Value	Description
RCIC	1	7	RTOP	OR	HCOMMON	10^{-3}	This is a CMF of RCIC and HPCI which appears on both trees, but is ignored in the PRA evaluation without explanation. It can be justified as a steam leakage from HPCI or RCIC steam lines or valves that cause some area temp sensors to isolate these systems. A value of 10^{-3} may not be too high for a small steam leakage. It was considered in the BNL evaluation of HPCI/RCIC CMF.
	2	10	RLTA	OR	RLU002DWI	10^{-3}	This lube oil system unavailability was judged to be too low compared with that in past PRAs, and with the unavailability of the lube oil system of HPCI, which is almost 8 times as high. This event was developed in detail in the HPCI fault tree, but here it remained undeveloped. A value of 3.6×10^{-3} was assumed.
	3	12	HAUTO	OR	HSW001DXI	5.8×10^{-4}	The SNPS-PRA tree designates this manual switch as common between RCIC and HPCI. The BNL review assumed separate switches for HPCI and RCIC.
	4	20	RFTT	OR	RHU100DXI	2×10^{-3}	Note 3 says that a common-mode miscalibration of both RCIC and HPCI exhaust turbine pressure trip/shutoff sensors can conservatively be made. However, on the HPCI tree the designator HHU002DXI (page 28) is used. This was changed to the same designator on both trees, and included as CMF of both systems. It is missing in the cut set resulting from the Boolean combination of HPCI and RCIC trees (see item below).

Table 3A.1 Continued

System	No.	Page	Gate Name	Gate Type	Input Name	Value	Description
RCIC	5	21	RFFT	OR	HHU909DXI	$2x10^{-3}$	This event is a cut set which is missing in the list of Appendix J. It is also a CMF with HPCI; however, there it is designated HHU001DXI. It was changed to HHU001DXI on the RCIC tree, page 21. Note that on RCIC, page 9, and HPCI, page 18, there are two other HHU909DXI miscalibration events of level 8, but these are errors "too low". The HHU001DXI then designates miscalibration error of level 8 "too high".
	6	22	RFLVCI	OR	----	$2x10^{-3}$	BNL added input RHU200DXI to account for miscalibration of low pressure sensors, giving false isolation valve closure.
	7	27	RPMDI	OR	RTRID	10^{-3}	Not appearing on cut set list even though instability in turbine exhaust is a potential trip mechanism in subsequent starts, the same way as it was on initial start.
	8	App. J page J-36					The discussion here implies that, at some time, RCIC had 29 cut sets rather than the 28 shown. This needs correction.
HPCI	1	5	HFTG	AND	HPRES1	$2x10^{-3}$	This is true for LOCA initiators, but not for transients with successful scram, in which it takes at least one hour to reach the 1.7 psi drywell pressure setpoint if RHR is not cooling the suppression pool. It was separated into the above two cases, so that, in case of a transient that does not cause drywell pressure, an event HPR = 1.0 was added to the OR gate HPRESI on page 6.

Table 3A.1 Continued

System	No.	Page	Gate Name	Gate Type	Input Name	Value	Description
HPCI	2	9	YREFIL	OR	YHU100DXI	10^{-3}	A value of 0.05 for failure to replenish water to CST was used in BNL review.
	3	12	HINTMAN	OR	HSW001HWI	2.47×10^{-3}	This failure of manual switch received an hourly failure rate rather than the per demand failure rate of 5.8×10^{-4} given elsewhere for similar events. See HPCI fault tree, page 4, event HSW001DXI.
	4	15	HINTB	OR	HPRBD	---	This event is developed on page 29 under the name HSPH. HPRBD changed to HSPH. Other similar changes should be made on these two pages.
	5	16	HPM	OR	HCOMMON	10^{-3}	Included. For description see RCIC item No. 1.
	6	23	HLUBE	OR	DWI	4.5×10^{-3}	Auxiliary oil pump is used for startup of HPCI turbine and when the turbine gains speed the shaft driven oil pump begins to supply the hydraulic pressure. Should the shaft-driven oil pump malfunction, causing oil pressure to drop, the auxiliary oil pump restarts. The fault tree, nevertheless, assumes both pumps are required and puts them in series. The cut sets of Appendix J ignore DWI for the shaft-driven, i.e. assume they are in parrallel. This should be clarified. Until then, a conservative assumption is that for long-term success of HPCI (10 hrs) both are required.

Table 3A.1 Continued

System	No.	Page	Gate Name	Gate Type	Input Name	Value	Description
HPCI	7	24	HOT	OR	HMV007DQI	1.24×10^{-4}	Typo error: NC-FC should be NO-FC.
	8	28	HSPT	OR	HHU002DXI	2×10^{-4}	Was changed to RHU100DXI. See RCIC item 4 for description.
	9	34	HIND	AND	HCV019DPD	3.33×10^{-5}	1) The data base value of the check valve failure is 10^{-4} per demand. There is no apparent basis to assume 1/2 of its failure rate in subsequent starts. 10^{-4}/d was assumed. 2) Automatic transfer to suppression pool suction precludes use of CST. The analysis assumed the probability of this event to equal 1.0 after 1 hr, when automatic transfer on high suppression pool level was assumed. However, this is not correctly modeled in the fault tree. Event HCV019DPD should be replaced by OR gate with two inputs: HCV019DPD for the first hour and HINAUTS for the case of the probability of high level in suppression pool = 1.0.
	10	App. J pg. J-36					The discussion here implies that at some time HPCI had 40 cut sets rather than the 39 shown. This needs correction.

Table 3A.1 Continued

System	No.	Page	Gate Name	Gate Type	Input Name	Value	Description
LPCS	1	4 and 5	LILOG2 LILOG2	AND	LPRA LPRB LPRC LPRD	2.0×10^{-3} + 2.7×10^{-3}	Value of 1.0 was used for these inputs in the case of transient. For LOCA and ATWS the value of the input remains unchanged.
	2	2	LPCS1 LPCS2	OR	----	----	Added to each of these "OR" gates the event "LSP" which mainly stands for failure of suppression pool due to clogged strainers, and which is included in the SNPS-PRA cut sets list. (ZFL100HEI = 5×10^{-5}). See also LPCI fault tree, page 4, and HPCI fault tree, page 11.
	3	13	LD1DIS	OR	LMV05DPI	4×10^{-3}	Changed to 9.3×10^{-3} to account for less frequent testing as stated in SNPS-PRA note 10 to the LPCS fault tree.
	4	14	LD2DIS	OR	LMV05BDPI	4×10^{-3}	Changed to 9.3×10^{-3} as above.
	5	4	LAUTO	OR	LHU500DXI and LHU600DXI	0.05 + 0.05	Should be changed to event "AHU199DXI" appearing on page 13 of the ADS fault tree. This accounts for the failure of the operator to initiate low pressure injection manually following failure of the high pressure injection. It is assumed that failure of the operator to initiate ADS will result in his failure to initiate the LPCS or LPCI, i.e., these are dependent failures.

Table 3A.1 Continued

System	No.	Page	Gate Name	Gate Type	Input Name	Value	Description
LPCI	1	2	D1IA	AND	L1AUTO	small	The changes made by BNL to the LPCS tree (see LPCS No. 1) will change this entry on the LPCI fault tree to $\approx 2\times10^{-3}$, and it will appear in the cut sets list of the system as "HHU720DXI x AHU199DXI," contributing 2×10^{-4} to the LPCI unavailability.
	2	2	D1IA	AND	DHU111DXI	0.1	Changed to the event "AHU199DXI," which appears on ADS fault tree, page 13. See comment LPCS No. 5.
RHR	1	4	DSTAX	OR	DFLOIAHEI	2.6×10^{-4}	This is a "single strainer blockage/fail ure" of the suppression pool strainers. This should be a CMF of all strainers and be common to both HPCI and RHR. It was changed to the notation "ZFL100HEI" as on the HPCI fault tree (page 11) and quantified as 5×10^{-5}.
	2	5	DHUM	OR	all entries	4×10^{-5}	These are operator and procedure errors that cause failure to align the RHR to the suppression pool. This event can be reasonable for the first few hours following an accident, but the probability of its occurring 20 hours after the accident sequence initiation is assumed to be lower--in the 10^{-6} range. Hence, it is not included in the BNL list of contributors in Section 3.3.2.6.

Table 3A.1 Continued

System	No.	Page	Gate Name	Gate Type	Input Name	Value	Description
SWS	1	3	WEW1A	OR	WATER	$3x10^{-5}$	Deleted. This is an external event and, as such, is not considered in the current scope of the PRA.
	2	3	WEW1A	OR	WFL480HEI	$5x10^{-5}$	Included in the fault tree analysis. Even though this event appears on the SNPS-PRA fault tree, it was excluded from the list of cut sets. BNL included it as a cut set of SWS.
RCIC in Steam Con-densing Mode	1	4	RHXWA	OR	DHXA DXHB	----	The "DHXA" and "DHXB" are inputs transferred from the RHR fault tree. These gates should transfer-in the unavailabilities of Service Water Systems loop A and loop B. The correct gate names on RHR or SWS fault trees are, however, "WEWA" and "WEWB." This was changed by BNL.
	2	4	RHXWA	OR	----	----	A new gate named "WMV34ADPI" (and "WMV34BDPI") had to be added to account for the failure of both MOV34A and MOV34B on the RHR heat exchanger outlet from the SWS side.
	3	6	DHXATSP	AND	several	several	This gate has a subtree which is more accurately developed in the RHR fault tree, pages 11, 12, and 13. A transfer-in from the RHR fault tree was included in the BNL re-assessment. The main difference is that event "DHU471DXI for "operator fails to manually realign flow path discharge to suppression pool" is missing in the RCIC in the steam condensing mode fault tree.

Table 3A.1 Continued

System	No.	Page	Gate Name	Gate Type	Input Name	Value	Description
RCIC in Steam Con-densing Mode	4	4	RHXWA	OR	----	----	Added a new gate to the fault tree named event "DMV48ADWI" (and "DMV48BDWI"), which appear on page 6 of the RHR fault tree. This is the failure of the bypass valves of the RHR heat exchanger, causing flow diversion.
Conden-sate System	1	1	FLPINJ	OR	----	----	A new input was added with the name "FHU212DXI" and a value of 2.5×10^{-2}, similar to page 3 of the feedwater system fault tree. It stands for "Long-term operator actions to control conden-sate flow and makeup during cooldown."
	2	8	FCPA FCPB	OR	----	----	New inputs were added "FCPA" and "FCPB" transferred-in from page 15 of the feedwater system fault tree and also "FCPBA" and "FCPBB" transferred-in from page 17 of the feedwater system fault tree.
	3	14	FSJ	OR	several	several	This gate should be an "and" gate, exact-ly the same as in the feedwater system fault tree, page 16.
	4	21	FLPHBY	OR	several	several	This gate should be an "and" gate.
	5	21	FAVTOBY	AND	several	several	This gate should be an "or" gate.

4. DATA ASSESSMENT

This section reviews the numerical values of the parameters necessary for the quantification of the accident sequences. Subsection 4.1 presents the SNPS-PRA frequencies for the initiating events along with the BNL assessments. Subsection 4.2 discusses the SNPS-PRA data base used in the evaluation of component unavailabilities along with the BNL evaluation. Comparisons with the LGS-PRA are also presented.

4.1 Frequencies of Initiating Events

4.1.1 Initiating Event Frequencies Used in the SNPS-PRA

The SNPS-PRA considered six groups of initiators:

a. Transient initiators excluding loss of offsite power (LOOP) with successful scram,

b. Manual shutdown initiators,

c. Loss of coolant accidents (LOCAs),

d. Transient initators without scram (ATWS),

e. Low frequency transient events,

f. Loss of offsite power initiator.

The frequencies of these initiators are treated separately also in the BNL review as described in the following subsections.

The frequencies of transient initiators used in the SNPS-PRA were based on data included in an EPRI-NP-801 report[5] which summarizes experiential data obtained from twelve operating BWRs and covers plant histories up to 1978.

The frequency of manual shutdown events was taken from an SAI report[7]. LOCA frequencies were based on a 1977 EPRI report[8]. The SNPS-PRA evaluated the frequencies of large, medium, and small LOCAs inside the drywell according to that 1977 EPRI report. It also calculated the frequencies of large LOCAs outside containment, and of interfacing LOCAs. The first was calculated according to failure rates taken from WASH-1400 and pipe length and isolation considerations. The calculation of the latter was different from that in WASH-1400; the data are based on Ref. 15, which summarizes LERs on valve failure, and the analysis is similar to that in an NRC work[16].

Frequencies of initiators coupled with failure to scram were based again on Ref. 5, with use of the same values derived for transients multiplied by the probability of failure to scram.

Low frequency transient events such as loss of DC, containment flooding, loss of service water, loss of reference leg in the water level measuring system, and loss of drywell cooling (see Table 4.1) were considered again on the basis of LER data, or, if the latter were unavailable, on the basis of estimated system failure probabilities.

The frequency of the loss of offsite AC power initiator was given plant specific treatment in the SNPS-PRA with use of LILCO fossil plant LOOP experience gathered since 1965.

Table 4.1 gives the frequencies used in the SNPS-PRA for the six groups of transient initiators, manual shutdown, the LOCA initiators, initiators coupled with a failure to scram, other low frequency transient events, and the LOOP frequency. SNPS-PRA values are compared with results of the BNL review.

4.1.2 BNL Assessment of the Initiator Frequencies

a. Transient Initiators with Successful Scram

An independent assessment was conducted to determine point values and associated distributions for the frequency of each one of the transient initiators used in the study.

The assessment is based on experiential data obtained from sixteen operating BWRs[6] and it includes both generic (i.e., characterizing the whole population) and particular (i.e., plant-specific) evaluations. The technique used is based on the "two-stage" Bayesian approach* first proposed and used by Kaplan[1] in the Zion and Indian Point PRAs[2,3] and as modified by Papazoglou[4]. The basic assumption of this method is that there is an actual variability in the frequency of each initiator within the population, but the characteristics of this variability are not exactly known because of limited information.

The technique calls for the assessment of a prior distribution for certain parameters. This is equivalent to assessing a prior distribution, for the frequency of the initiator, that characterizes the plant population. The prior distributions are then updated by using experiential data. In the present assessment, the prior distribution for the initiator that characterizes the plant population was practically log-uniform in the range of 10^{-4}/yr to 10^{+1}/yr.

The data were obtained from a recent EPRI report[6] that provides information on occurrences of 37 types of transients in BWRs. The data consist of 910 events occurring over 101.5 plant-years at 16 different plants. Means, medians, and five and ninety-five percentiles have been determined for each of the 37 initiators considered and for each of the 16 different plants.

For each initiator, a distribution was also generated to represent the population as a whole. This distribution best characterizes the uncertainties in the frequency of initiators for plants (such as the SNPS) that belong to the population but for which experiential data are not available.

The population distributions were further combined according to the grouping previously described (Section 2, Table 2.11). Table 4.2 summarizes and compares the results of the SNPS-PRA and those of the BNL review. The grouping of the transient initiators is indicated in parenthesis; the numbers

*Because the SNPS has not started power operation, there are no plant specific transient data from the plant, and a one stage Bayesian approach was used by BNL.

show the initiator sequential number as it appears in EPRI NP-2230[6]. The groupings of the SNPS-PRA were not changed in the BNL review, as stated in Section 2.

The first four columns of Table 4.2 show the SNPS-PRA results. The next four columns show the results obtained by applying the same SNPS methodology to the more recent data source[6]. The two last columns present BNL results obtained by using the updated source[6] and the two-stage Bayesian methodology[4]. Most of the increase in BNL initiator frequencies is seen to be derived from the updated experience of BWR-related events[6]. In the BNL independent assessment, the values in the last column of Table 4.2 were used. The basis for this choice is further explained below.

The results in Table 4.2 are generic initiator frequencies. At least in one case there is some plant specific information that suggests a lower initiator frequency for Shoreham. The Shoreham plant utilizes Target Rock two stage SRVs which are more reliable than those SRVs which are included in the data base for the IORV. Thus, a lower IORV frequency can be anticipated for SNPS than used in the PRA or in the BNL review. However, the effect of this transient on the results is very small, and a reduced initiator frequency for IORV would not have any significant effect.

The SNPS-PRA differentiated between the impact of failures during the first year of plant operation and of those in later years. BNL concluded, however, that the data base used[5] was not sufficiently refined for this purpose. The later EPRI-NP-2230 update[6] showed that the impact of ignoring the first year of plant operating experience causes a reduction of about 20% in initiator frequencies (see last two columns of Table 4.2). In addition, BNL considers the "weighted average" approach of the SNPS-PRA to result in small underestimations of the initiator frequencies, due to the lack of experience from aging plants (after 30 to 40 years of operation), which may be comparable with the first-year frequencies in the number of challenges because of increased failure rate (wear-out).

The purpose of subtracting the data for the first year of operation was to obtain transient initiator frequency for the evaluation of risk associated with Shoreham during mature plant operation. The BNL review is aimed at obtaining the average risk associated with Shoreham during the entire lifetime of power operation. This can be obtained by deriving the initiator frequency from the data of EPRI-NP-2230 for all years of operation. Note that this EPRI report includes, on the average, experience from 7 years of a plant operation; thus the first year of operation is weighted 1/7 (and not 1/35 as in the SNPS-PRA). As shown in Table 5.15 of Section 5-3, the difference between these two assumptions amounts to 10% in the total core damage frequency for SNPS. Therefore, it was judged by BNL that the last column of Table 4.2 using the entire data base is at this time (prior to the plant's first year of operation) more appropriate for the assessment.

b. Manual Shutdown Initiators

The frequency of such initiators has a relatively low impact on core damage probability. Considering the limited funds and time allotted to this review, BNL chose not to review it in detail. The value chosen in the

SNPS-PRA, basically taken from Ref. 7, appears to be in the reasonable range, and it was used in the assessment.

c. LOCA Initiators

The LOCA initiator frequencies used in the SNPS-PRA for large, medium, and small LOCA, as well as for LOCA outside containment and pressure vessel failure, appeared to be reasonable when compared with the available data[8] and therefore were not independently assessed. The frequency of interfacing LOCA was evaluated separately in more detail (Appendix F of the SNPS-PRA). The SNPS data and analysis were reviewed by BNL, and the results are compared with the SNPS-PRA data in Table 4.3 . The frequency of core damage in Class V was significantly affected by these changes; that in Classes I through IV was not. The main changes are due to the different approaches used in the SNPS-PRA and the BNL review:

a) The SNPS-PRA used valves failure rate from LERs, whereas BNL used six specific LERs, which are interfacing LOCA precursors.

b) The SNPS-PRA used only leakage and rupture failure rates for MOVs. BNL also considered spurious opening.

Appendix 5C.2 includes further descriptions of the different approaches in the BNL review and the SNPS-PRA.

d. ATWS Initiators

ATWS initiator frequencies were derived basically from the corresponding transient initiator frequencies, with some minor exceptions. In the SNPS-PRA, turbine trip ATWS events were evaluated by using a turbine trip initiator event tree (see Figure 4.1). The tree considered whether feedwater was properly controlled, whether turbine bypass was available, and whether condenser heat sink was available. Failure to balance feedwater, or failure of the turbine bypass or the condenser heat sink, was conservatively assumed to have a plant response similar to that of loss of feedwater, the MSIV closure, or loss of condenser events. Figure 4.1 shows the quantification method by which the turbine trip frequency was calculated, and also the fraction of the turbine trip initiator frequency that was transferred to the other ATWS initiators.

BNL analyzed the sequences following turbine trip and prepared an event tree similar to Figure 4.1 which is shown in Appendix 5D Figure 5D.8. The main difference in the BNL event tree is that BNL considered it more appropriate to treat the feedwater runback on the functional event trees for ATWS. The feedwater runback is one part of a set of procedural actions which the operator has to follow promptly. These actions also include manual actuation of the SLC system, reducing level and maintaining it above TAF, and ADS inhibit. In BNL's judgement, these actions are partially dependent.

The differences in the quantification of Figures 4.1 and 5D.8 result from the use BNL made of the turbine trip transient functional level event tree (Table 5A.1). The same values are used in ATWS Figure 5D.8 as in Table 5A.1 for the transient with successful scram.

The resulting ATWS initiators frequencies for the SNPS-PRA and the BNL review are compared in Table 4.1. For total ATWS frequency, the SNPS-PRA values of 5.49, given all power levels, and 3.87, for power levels above 25%, are compared with BNL values of 9.61 and 7.34 respectively. The difference for power level above 25% is almost 100%. This is because EPRI-NP-801, used by SNPS-PRA, has 60% of the data from the first year of plant operation which includes many cases of low power testing. EPRI-NP-2230 removed the data that belong to the time between first criticality and the start of commercial operation. Thus, in EPRI-NP-2230, only 33% of the data are from the first year of plant operation.

The difference between the values of SNPS-PRA and BNL for the particular initiators is also due in part to the different treatment of feedwater runback which was discussed above. In summary, it can be expected that about a factor of two difference between SNPS-PRA and BNL review results for ATWS core damage frequencies stems from the different sources of data for evaluating the initiator frequencies.

Additional discussion is provided in Appendix 5D, and in particular in Table 5D.2.

e. Low Frequency Transient Events

These events include the following:

a) Loss of DC power bus,

b) Reactor water level measurement system reference line leak,

c) Drywell cooler failure,

d) Loss of service water,

e) Excessive release of water into Elevation 8 of the reactor building (Maintenance and Rupture).

The frequency of loss of DC power bus initiator was based on a NUREG report[21] which takes into account DC bus related LERs and calculates a frequency of 6×10^{-3} per year for a bus failure. A recovery factor of 0.5 was used on the basis of considerations from this report, which SNPS says that it implemented in its design and procedures.

The frequency of loss of a reference leg and of a drywell cooler were based on LERs.

Loss of service water frequency was derived from the experience of no loss of service water in 400 BWR reactor-years, giving 0.0025 as a conservative value. BNL used a frequency of one event in 600 reactor-years.

The frequency of excessive release of water in Elevation 8 was calculated quite differently in the PRA and by BNL. Shiu[25] provides the details of the two approaches. The SNPS-PRA event frequency is given in Table 3.4-25 of the PRA (page 3-263). BNL used Markov modeling and recovery considerations

different from those in the SNPS-PRA, which resulted in an increase of the total flooding initiator frequency (see Table 4.1).

The results of the BNL assessment are listed in Table 4.1 along with the values used in the SNPS-PRA, LGS-PRA, and RSS. Because of its importance, the frequency of the loss of offsite power initiator is discussed in detail here.

4.1.3 Loss of Offsite Power Initiator

The frequency used for the loss of offsite power initiator in the SNPS-PRA was derived from non-nuclear plant experience and reflects only Long Island Lighting Company (LILCO) fossil-plant data.

The data cover the period January 1, 1965, through January 1, 1981, for LILCO plants with three or more circuits emanating from them. The data consist of the following for each plant:

- o Years of operation during the period January 1, 1965, to January 1, 1981,
- o Number of outages,
- o Duration of outages.

Table 4.4 summarizes the LILCO specific grid reliability data. In total, these plants had four occurrences in 61.5 plant-years. The loss of offsite power was calculated as follows:

$$T_E = \frac{\text{occurrences + hypothesized incipient failure}}{\text{years plant experience}} ,$$

$$T_E = \frac{4 + 1}{61.5} = 0.08/\text{year} .$$

The SNPS-PRA methodology for evaluating the frequency of loss of offsite power does not consider any regional nuclear power plant experience. The SNPS-PRA acknowledged that "the specific case applicable to SNPS is the Northeast Power Coordinating Council (NPCC)" (SNPS-PRA, page A-192); however, this effect was not included. The BNL assessment of the frequency of the loss of offsite power initiator and the associated uncertainties were derived from the nuclear plant experience of the NPCC, which includes New York, Massachusetts, Connecticut, Vermont, and Maine. Fossil-plant experience was not included to remain consistent with current nuclear plant PRA practice, which does not include non-nuclear plant experience in the quantitative estimation of the frequency of loss of offsite power, and their recovery probabilities. BNL believes that both the probability of LOOP and the recovery probabilities as a function of time should be calculated from the same data base. This is done in this review as described below and in Section 4.1.4.

The technique applied to assess the frequency of loss of offsite power and the associated uncertainties is described in Subsection 4.1.2 and in more detail in Ref. 4. This technique takes into account the LOOP experience of other nuclear plants in the same electrical reliability council to which SNPS belongs. The methodology and data used by BNL to assess the LOOP frequency

are different from those used in the SNPS-PRA and reflect the difference between the the SNPS-PRA and the BNL LOOP initiator assessed frequency.

The results for the NPCC, to which the SNPS belongs, were used in the BNL review. The data used were taken from Ref. 10, in which the loss of offsite power is categorized into four groups. The first group includes total loss of offsite AC power in nuclear power plants, and this was used by BNL. However, the loss of offsite power during cold shutdown (group four in Ref. 10) was included by BNL in the final evaluation for LOOP frequency (Table 4.5) because the LOOP frequency should be evaluated on a yearly basis, and the mode of plant operation is irrelevant to the LOOP frequency. These events, if caused by maintenance error, are recovered immediately, and this is taken into account in the recovery probability distribution. The results of the analysis are given in Table 4.6.

Since the SNPS is a new plant, not yet in operation, and therefore lacks plant-specific data, the appropriate values are those characteristic of the population of this particular reliability council. That is, the SNPS should be treated as a plant taken randomly from the population of NPCC plants. BNL's judgment is that utilizing merely LILCO fossil-fuel plant experience in calculating LOOP frequency, and using generic nuclear plant data for recovery probabilities rather than the same set of data used for LOOP frequency, is not a consistent and realistic approach.

The mean value of 0.15 occurrences per year (see Table 4.6) was used in the BNL review for the frequency of the LOOP transient initiator.

In the RSSMAP Grand Gulf PRA, this frequency was assumed to be 0.20 occurrences per year and in the Big Rock Point PRA, 0.13 occurrences per year.

In the RSS, nuclear power experience was considered for the year 1972 which included three LOOP events. These events occurred in about 150,000 operating hours, giving a point estimate for the rate of 2×10^{-5} failures per hour or 0.18/yr.

4.1.4 Recovery of Offsite Power

The probability of recovery of offsite power, within a given time, was assessed in the SNPS-PRA by using EPRI-NP-2301 data base[9]. The data representing the entire population of U.S. plants was used in the SNPS evaluation. The recovery probability was simply taken as the percentage of events that were recovered in a particular time interval of interest. The BNL review used updated data taken from Ref. 10, which reconcile many of the differences between Scholl[11] and EPRI-2301[9] data. However, in BNL'sjudgment, events of type IV* should be included in the data base (as discussed in Section 4.1.3 above). Their recovery time was included. The number of events for the NPCC region is sufficiently large to be considered separately rather than the data from the overall U.S. population of nuclear plants.

In the BNL approach, the recovery times were assumed to be lognormally distributed. Next, the two parameters of the lognormal distribution were

*No offsite power available during cold shutdown because of special maintenance conditions that do not occur during or immediately following operation[10].

assumed to be random variables distributed according to given probability density functions. The experiential data for the 10 plants of the NPCC were updated through December 1983 (Table 4.5) and then used for a Bayesian updating of the assumed prior distributions for the two parameters. Finally, by "averaging out" the dependence of the distribution of the recovery time on the two parameters, a "Student t" distribution was obtained to represent the distribution of the recovery times.

The probability of not recovering offsite power within a given time is calculated from the complementary cumulative distribution and is shown in Table 4.7 along with the SNPS-PRA values.

The use of data from Ref. 10 "as is" without modification resulted in a LOOP frequency of 0.13 per year; however, the associated recovery probabilities were lower than in the case discussed before. Table 5.15 compares the results of both cases and shows that they are basically giving the same results. Thus, the inclusion of the LOOP events occurring due to maintenance at plant shutdown does not affect the core damage frequency results.

4.1.5 Conclusion

The frequencies of the initiating events determined by the BNL approach differ, as shown in Table 4.1, from those used in the SNPS-PRA.

The BNL-assessed frequencies of the initiator events were used to quantify the accident sequences. In Section 5, the relative contributions of the initiating event frequencies to the total core damage frequency are reported. It is seen there that the changes in the ATWS frequency, LOOP initiation frequency, MSIV closure frequency, and turbine trip frequency are the most important.

4.2 Component Unavailabilities

4.2.1 SNPS Data Base

The data base used in the SNPS-PRA to quantify component failure rates in the fault tree models comes from four basic sources:

- Licensee event reports (such as Ref. 14 and 15),
- General Electric BWRs operating experience data (such as those in the LGS-PRA),
- The Reactor Safety Study (RSS),
- IEEE reliability data for electrical components (ANSI/IEEE std. 500-1977).

The priority for data selection followed the above listed order. This has resulted in many cases in which NRC LER data was used.

The maintenance and test data used in the SNPS-PRA are, in general, said to be obtained from GE operating experience with BWRs. The technical

specification values and the test frequencies are dervied from SNPS draft technical specifications (February 1983).

The probability of diesel generator failure to "start and run" and the conditional probability that multiple diesels will fail, given the probability of the first diesel failing, which were used in the SNPS-PRA, are evaluated in its Appendix A.5. The values appear to be in the appropriate range. They were further reviewed by BNL (see Section 4.2.2 below), and recovery data of diesel generators were also reviewed and slightly modified.

4.2.2 Data Assessment for Diesel Generator Availability

The SNPS-PRA uses data from nuclear power plant operating experience to characterize diesel generator performance in case of Loss of Offsite Power. The experiential data sources are two EPRI reports (NP-2099 and NP-2433)[22,23] and the NUREG/CR-1362 report[14].

From these data, the SNPS-PRA calculates three sorts of information needed for the event-tree quantification:

1. The probability of a single diesel generator failing to start on demand.

2. The conditional probability of multiple diesel failures given that one diesel failed.

3. Data on the length of time required to restore a diesel to operation (recovery times).

The approach used in the SNPS-PRA to obtain these data, and the BNL review comments and adopted values, are discussed in the following sections.

a. Probability of a Single Diesel Generator Failing to Start

The SNPS-PRA used a value of $2x10^{-2}$ per demand for the failure to start probability of an average diesel generator. This is an average value derived from assessment of LERs of 36 plants, obtained mainly from NUREG/CR-1362[14]. BNL considered the value to be a reasonable choice at the time the PRA was performed. Newer data in NUREG/CR-2989[20], published after the SNPS-PRA was completed, support this average value. The new data include failure to start probabilities (for about 40 plants) ranging from $3x10^{-3}$/d to $6.25x10^{-2}$/d with an average value of $2.2x10^{-2}$/d. If a fraction of the autostart failure is also considered to contribute to the overall failure to start probability, a value of $2.5x10^{-2}$/d could be used, and that is the average value cited in NUREG-CR/2989 (see their Table 9.5.19). However, this NUREG report is aimed at obtaining plant specific diesel generators' unavailability estimates, and provides abundant information for this purpose. Apparently there are plans to modify the SNPS diesel generator design configuration and hardware, but they were not included in the version of the PRA reviewed by BNL. Rather than using plant specific values for an evolving design, BNL decided to replace them by conservative values from the older NUREG/CR-1362[14]. The sensitivity study in Section 5.3 shows the effect of an improved diesel-generator design using NUREG/CR-2989 data as given in Section 4.2.2b below. The value used in this report is the same as that in the SNPS-PRA, $2x10^{-2}$/d. This is based, in

the BNL review, on the NUREG/CR-1362 data base with one week between tests, and includes failure to run during the first hour (Table 20 of Ref. 14). The above discussion is summarized in Table 4.8.

b. Conditional Probabilities of Multiple Diesel Failures

The SNPS-PRA used the data from plant Q (Plant-X in LGS-PRA[17]--see their Table A.5.9), because these were the best single-plant applicable data. The LGS-PRA used a value obtained by averaging plant Q, Cook, and Zion values, the RSS value, and the NUREG/CR-1362 values. All the values are quite close, as seen in Table 4.8. From NUREG/CR-2989, a value for the failure probability of the third diesel given that two have failed P(3/2) can be easily derived, which is also similar to those of the SNPS and LGS-PRAs. To derive a value for P(2/1) from NUREG/CR-2989 a specific design must be assumed. When Table 4.8 is considered in its entirety, the values of SNPS-PRA appear to be on the high side of the spectrum of generic type values. This is thought to be suitable until information on the SNPS specific design for upgrading is submitted. Data from NUREG/CR-2989 could be used in such a case. BNL therefore used the SNPS results, but for sensitivity study purposes, evaluated the following values:

- o Failure to start on demand 1×10^{-2}
- o P(2/1) 0.11
- o P(3/2) 0.40

These are examples of values derived from NUREG/CR-2989 for a design with three dedicated diesels, using average procedures and having service water cooling.

c. Recovery Times for Diesel Generators

The SNPS-PRA used the recovery data from NUREG/CR-1362[14] after comparison with Peach Bottom data. In its Appendix A.5, recovery of diesel generators within the first half hour is argued to be uncertain, and a value of 1.0 for nonrecovery is suggested, but in the LOOP event tree, a value of 0.88 is used. A value of 0.95, which is consistent with Peach Bottom data and with LGS-PRA recovery data, is used in the BNL review. For all other recovery times, BNL used the SNPS-PRA data, which are the same as those in the LGS-PRA.

d. Summary of Data for Diesels

In summary, the data used by BNL are not very different from those used by the SNPS-PRA. Both are generic, consistent with LER data, and quite conservative when a weekly testing interval is assumed. However, the data are not plant specific. BNL recommends that, for a modified SNPS design (if submitted), the unavailability should be evaluated on the basis of data from NUREG/CR-2989[20] or other comprehensive new studies.

4.3 Human Error Probabilities

As stated in Section 3.4, two different types of human errors--procedural and cognitive--are considered in evaluating the system unavailabilities. The

procedural human errors were based, in most cases, on NUREG/CR-1278[24] and were not part of the BNL review. Major procedural errors affecting the systems' unavailability are shown in Table 3.3 along with the probabilities used in the SNPS-PRA. In most cases BNL used the same values or model (see footnote to Table 3.3); in only one case, the miscalibration of all sensors, did BNL use a different value for a procedural error probability.

The value of $2x10^{-3}$ used for miscalibration of all sensors (event "HHU720DXI") is developed in the SNPS-PRA Appendix A.3. It is derived similarly to the NUREG/CR-1278 Human Error Probability (HEP) tree*, but different quantification of the HEP tree results in a more conservative estimate of the gross miscalibration of all four level sensors. The SNPS-PRA model includes (Appendix A.3 page A-120):

a) Use of a faulty setup such as a wrong scale or connection at an incorrect point. This was conservatively quantified by a probability of 10^{-2}.

b) Technician rechecks the setup and recovers the gross miscalibration in the second sensor with a probability of 0.7.

c) Technician rechecks and corrects the error in his third calibration with probability 0.3.

d) All other sensors would be miscalibrated given the technican failed to detect the error in the first two cases.

This model resulted in a probability of $2x10^{-3}$ for gross miscalibration in the SNPS-PRA. It does not consider staggering of the calibration procedure.

NUREG/CR-1278[24] distinguishes between small and large miscalibrations. For the small miscalibration of all four channels the probability from the the HEP tree is $5x10^{-4}$, mainly because the HEP tree assumes a probability of 0.9 for step (c). This is based on the assumption that a technician may accept a small change in the calibration for one channel, but in 9 out of 10 cases he will realize that something may have gone wrong when he finds a small change in the second channel also.

For large miscalibration, NUREG/CR-1278 assumes that the recheck probability is 0.9 for step (b) and 0.99 for step (c). Thus the HEP tree gives a probability of $5x10^{-6}$ for large miscalibration.

BNL considered the value $5x10^{-6}$ too small if special procedures are not used, but found the value $2x10^{-3}$ unrealistic for the large miscalibration needed to fail the level 2 and level 1 auto start of HPCI, RCIC, ADS, LPCI, and LPCS. BNL considered a value smaller than $2x10^{-4}$ to be realistic when miscalibration procedures are available that guide the technician to recheck his setup whenever he finds a significant change in calibration to be required.

The list of the major cognitive errors introduced in the SNPS-PRA is given in Tables 3.2 and 3.3. The number of quantification changes performed

*NUREG/CR-1278[24], August 1983 Revision, page 10.7.

by BNL in the cognitive human errors is significant. Most of these changes are based on the judgment of the total time available to the operator and the number of additional actions he would be required to perform concurrently. In most cases they involve changes made in the event trees (see Table 3.2), and are explained in the tables depicting the revised event trees in Appendices 5A to 5G. The remaining changes made in the cognitive errors are shown in Table 3.3 and discussed in the next paragraph.

The SNPS-PRA treatment of the manual initiation of ADS, LPCS, and LPCI, given the failure of the auto start of all three, is as follows:

ADS: event "AHU199DXI" = 0.1, which stands for "Operator fails to initiate ADS given auto system failure."

LPCS: events "LHU500DXI or LHU600DXI" = 0.1, which stands for "failure to manually initiate LPCS."

LPCI: event "DHU111DXI" = 0.1, which stands for "failure to manually initiate LPCI."

However, these three events are not independent under all accident sequence conditions. In the case of the failure of high pressure injection systems, an operator error--failing to initiate ADS--will result with high probability in the failure to initiate other safety systems. Furthermore, if the operator fails to manually initiate LPCS or LPCI, depressurization will not occur even if the operator tries to depressurize by the ADS manually. Thus, there are two dependences: (1) functional dependence, (2) human interaction dependence which assumes that failure of the operator to initiate the first system implies that the operator will not respond to initiate the second either. This latter dependence, which was recognized in SNPS-PRA Appendix A.3, was included in the BNL re-assessment. All the above different operator actions were denoted "AHU199DXI" = 0.1 in all three cases. This is also consistent with the NUREG/CR-1278[24] approach.

The two BNL modifications to SNPS-PRA human error treatment discussed in this subsection constitute the event of "miscalibration of all four water level transmitters." The impact on core damage frequency of this event is discussed in Section 5A.1.4 of Appendix 5A.

4.4 References to Section 4

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6. "Anticipated Transients, A Reappraisal," EPRI NP-2230, January 1982.

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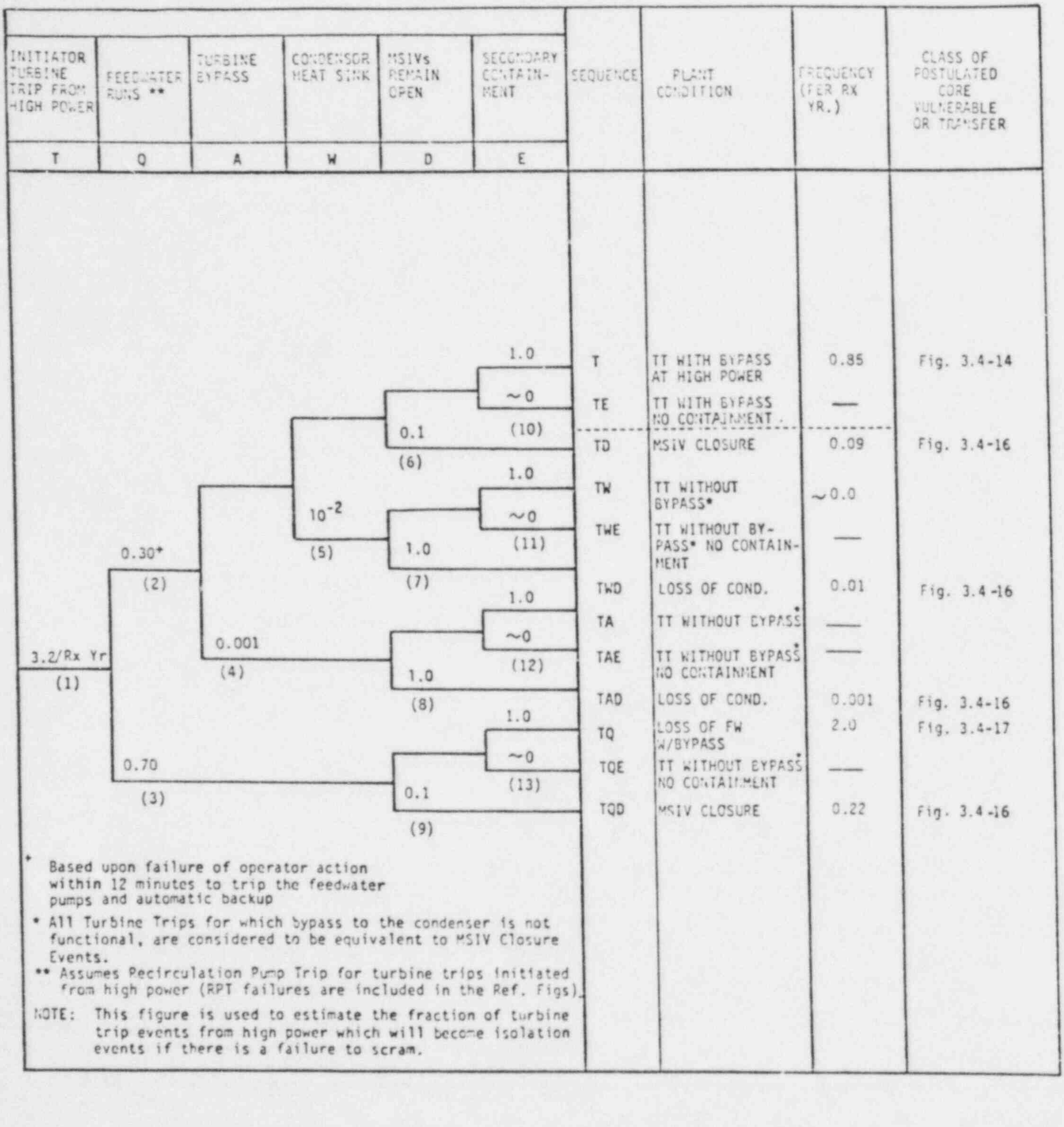

Figure 4.1 Event Tree Diagram of Accident Sequences Following a Turbine Trip Initiator From High Power

Table 4.1 Frequency of Initiating Events (Mean Values/yr)

	SNPS-PRA	LGS-PRA	BNL Review SNPS	BNL Review LGS	WASH-1400
Transients	9.8	9.1	13.95	13.02	11
Turbine trip	4.49	3.98	8.01	8.17	
MSIV Closure	0.24	1.78	0.57	1.23	
Loss of Condenser	0.41	Included in MSIV	0.50	Included in MSIV	
Loss of Feedwater	0.18	Included in MSIV	0.13	Included in MSIV	
Loss of Offsite Power	0.08	0.053	0.15	0.11	
IORV	0.09	0.07	0.25	0.25	
Manual Shutdown	4.3	3.2	4.3	3.2	
LOCAs					
Large	7.0×10^{-4}	4.0×10^{-4}	7×10^{-4}	4×10^{-4}	2.7×10^{-4}
Medium	3.0×10^{-3}	2.0×10^{-3}	3×10^{-3}	2×10^{-3}	8.1×10^{-4}
Small	8.0×10^{-3}	1.0×10^{-2}	8×10^{-3}	1×10^{-2}	2.7×10^{-3}
Breach of the RPV	3.0×10^{-7}	---	3×10^{-7}	---	
Interfacing LOCA (LOCA Outside Containment)	1.8×10^{-7}	---	3×10^{-7}	---	
ATWS	5.49(3.87)*	5.92	9.61(7.34)*	9.82	
Turbine Trip	2.14(.85)*	3.6	7.0(5.3)*	7.39	
MSIV Closure	0.56(0.50)*	2.2	.88(.65)*	2.01	
Loss of Condenser	0.41(0.25)*	Included in MSIV	.57(.46)*	Included in MSIV	
Loss of Feedwater	2.2(2.1)*	Included in MSIV	.77(.59)*	Included in MSIV	
Loss of Offsite Power	0.08	0.053	0.15	0.11	
IORV	0.09	0.07	0.25(.16)*	0.25	

*In parentheses: Initiators frequency, which is at above 25% power (above condenser bypass capability). Without parentheses: Initiator frequency at all power levels (0 to 100%).

Table 4.1 Continued

	SNPS	LGS-PRS	BNL	WASH-1400
Low Frequency Transients				
Excessive Release of Water into Elevation 8 of the Reactor Building (Maintenance & Rupture)	$6x10^{-5}$	---	$5.0x10^{-4}$	---
Loss of a DC Power Bus	$3.0x10^{-3}$	---	$3.0x10^{-3}$	
Loss of all DC Power Buses	$3.0x10^{-6}$		$3.0x10^{-6}$	
Reactor Water Level Measurement System Reference Line Leak	$3.6x10^{-2}$	---	$3.6x10^{-2}$	---
Drywell Cooler Failure	$1.0x10^{-2}$	---	$1.0x10^{-2}$	---
Loss of Service Water	$2.5x10^{-3}$	---	$1.7x10^{-3}$	---

Table 4.2 SNPS-PRA and BNL Results for Initiator Frequency and Sources of Differences

	SNPS-PRA: EPRI-NP-801 Data[5]				BNL Review: EPRI-NP-2230 Data[6]				BNL Review: Two-Stage Bayesian	
Transient	1st Year	Subseq. Years	All Years Average	SNPS-PRA Weighted Average*	1st Year	Subseq. Years	All Years Average	Weighted Average	Subseq. Years	All Years**
Loss of Condenser Vacuum (2,4,8)	1.6	0.38	0.67	0.41	1.0	0.38	0.47	0.40	0.40	0.50
Turbine Trip	16.9	4.14	7.3	4.46	13.4	6.39	7.39	6.59	6.85	7.89
MSIV Closure (5)	2.2	0.19	0.67	0.24	1.67	0.27	0.47	0.31	0.29	0.57
Loss of FW (22)	0.6	0.16	0.27	0.18	0.27	0.11	0.13	0.12	0.11	0.13
LOOP (31)	0.4	0.11	0.16	0.08+	0.13	0.12	0.12	0.08+	0.12	0.15++
IORV (11)	0.7	0.08	0.20	0.09	0.53	0.15	0.21	0.16	0.19	0.25
CRW (27, 28)	0.1	0.03	0.04	0.03	0.13	0.10	0.11	0.10	0.11	0.12
Total	22.5	5.09	9.3	5.49	17.1	7.52	8.9	7.76	8.07	9.65

+Based on SNPS grid data.
++Based on NSAC-80 report[10].
*Used in the PRA.
**Used in the BNL review.

Table 4.3 Summary of Quantification for Exposing the Low Pressure Systems to Primary System Pressure

System	SNPS-PRA: Point Estimate Calculation Per Interface	No. of Interfaces	SNPS-PRA: Frequency Total Calculated (Per Rx Yr)	BNL Review Frequency Total Calculated (Per Rx Yr)
Core Spray (Figure F.2-1)*	4.8×10^{-8}	2	9.6×10^{-8}	
RHR Head Spray (Figure F.2-2)	8.6×10^{-12}	1	8.6×10^{-12}	
LPCI Injection (Figure F.2-3)	4.8×10^{-9}	2	9.6×10^{-9}	
RHR Shutdown Cooling Line (Figure F.2-4)	1.6×10^{-8}	1	1.6×10^{-8}	
Total	---	6	1.2×10^{-7}	3.0×10^{-7}**

*Figures in Appendix F of SNPS-PRA.

**Calculated in Appendix 5C.2 of this report for the entire plant (not system by system).

Table 4.4 Summary of the Historical Data on the LILCO Grid for Loss of Offsite Power Incidents

LILCO-Specific Grid Data Loss of OffSite Power (1/1/65 - 1/1/81)			
Plant	Years of Operation	Occurrences	Duration (minutes)
Barrett	16.0	1	222*
Glenwood	16.0	1	199*
Northport	13.5	0	---
Port Jefferson	16.0	2	58* 15
Total**	61.5	4	

*East Coast Blackout (11/9/65).
**Totals: 61.5 plant-yrs., 4 occurrences + 1 hypothesized incipient failure.

Table 4.5 Experiential Evidence from Plants of the Northeast Power Coordinating Council (NPCC) Loss of Offsite Power

Plant Name/ Date of Accident	Recovery Time	No. of Occurrences EPRI Data NP-2301[9]	No. of Occurrences EPRI Data NSAC/80[10]	No. of Occurrences BNL Review	Years in Operation EPRI NSAC/80	Years in Operation BNL Review	Events** in BNL Review
1. Fitzpatrick		2	0	2	9.2	9.05	3/27/79 (3 min)* 10/4/78 (14 sec)*
2. Ginna							
3/4/71	30 min						
10/21/73	40 min	2	2	2	14.3	14.10	
3. Haddam Neck							
4/27/68	29 min	5	5	5	15.9	16.30	
7/15/69	9 min						
7/19/72	1 min						
1/19/74	20 min						
6/26/76	16 min						
4. Indian Point 2 & 3							
7/20/72	55 min	1	3	3	12.2	10.5	
7/13/77	6:28 hr						
6/3/80	1:45 hr						
5. Main Yankee		0	0	1	11.3	11.10	8/31/78 (1 min)*
6. Millstone 1 & 2							
8/10/76	5 hr	1	2	2	13.2	13.10	
7/21/76	5 min						
7. Nine Mile Point							
11/17/73	10 sec	1	1	1	14.3	14.25	
8. Pilgrim							
5/10/77	2:40 hr						
2/6/78	8:54 hr	2	2	2	11.50	11.45	
9. Vermont Yankee		0	0	0	11.80	11.70	
10. Yankee Rowe							
11/9/65	33 min	1	1	1	22.50	23.30	

*Recovery Time.
**Relative to NSAC/80.

Table 4.6 LOOP Initiator Frequency Considered in SNPS-PRA and BNL Review

	SNPS-PRA	EPRI-NSAC Study[10]		BNL Review			
	Fossil Plant Experience	NSAC/80 Data Base for NPCC	NSAC/80 Data Base for Nat'l Population	NSAC/80 Data Base for NPCC	NSAC/80 Data Base for NPCC + 3 Add'l Events[4]	NSAC/80 Data Base for Nat'l Population	Special Case Plant Specific
Approach	Point Estimate	Point Estimate	Point Estimate	Two-Stage Bayesian	Two-Stage Bayesian	Two-Stage Bayesian	Two-Stage Bayesian
Data Used	5 events[1] in 61.50 Plant Years	16 events in 136.20 Reactor Years	47 events in 532.70 Reactor Years	16 events in 136.20 Reactor Years	19 events in 134.85 Reactor Years	47 events in 532.70 Reactor Years	18 events[2] in 152 Reactor Years
LOOP Frequency	0.08/Rx	0.12/Rx	0.088/Rx	0.13/Rx	0.15/Rx[3]	0.09/Rx	0.12/Rx

1 Four actual events and one hypothetical for some margin.
2 Fossil-Fuel Plant which experienced 2 events in 16.0 Plant Years is included as a hypothetical example of performance.
3 Judged by BNL to be most appropriate for the BNL review reassessment.
4 Three events were judged in BNL Review to be considered as LOOP initiators even though rejected by NSAC/80 evaluation.

Table 4.7 Recovery Time Distributions

	SNPS-PRA		BNL Review					
			National		NPCC		NPCC	
Recovery Time in Hours	No. of Events[1]	Cumulative Probability of Recovery	No. of Events[2]	Cumulative Probability of Recovery	No. of Events[2]	Cumulative Probability of Recovery	No. of Events[3]	Cumulative Probability of Recovery
≤0.5	20	0.48	25	0.55	9	0.55	12	0.63
0.5-1.0	6	0.62	7	0.68	3	0.67	3	0.73
1.0-2.0	4	0.72	7	0.80	1	0.78	1	0.81
2.0-4.0	2	0.77	4	0.88	1	0.86	1	0.88
4.0-8.0	6	0.91	3	0.94	1	0.91	1	0.92
8.0-10.	1	0.93	1	0.95	1	0.93	1	0.93
10.-24.	1	0.96	0	0.98	0	0.96	0	0.97
>24.	2	.00	0	≈1.00	0	≈1.00	0	≈1.00
Total	42		47		16		19	

[1] Based on EPRI-NP-2301[9]; point estimate.
[2] Based on NSAC-80[10]; Student t distribution.
[3] Based on NSAC-80 and three additional events included by BNL; Student t distribution (used in BNL re-assessment).

Table 4.8 Comparison Between SNPS-PRA Diesel Generator Data and Other Evaluations

	SNPS-PRA	LGS-PRA	NUREG/CR-1362[14]	Zion*	Wash 1400	NUREG/CR-2989[20]	BNL Review
Failure of a Diesel Generator to Start Upon Demand	2×10^{-2}	1.7×10^{-2}	2×10^{-2}	1.9×10^{-2}	3×10^{-2}	2.5×10^{-2}**	2×10^{-2}
Probability of Second Diesel Failure Given One Failed - P(2/1)	0.19	0.23	0.42*	0.08	0.03	Plant or Design Specific	0.19
Probability of Third Diesel Failure Given Two Failed - P(3/2)	0.63	0.55	0.17*	0.45	1.0	0.49***	0.63
Failure of a Diesel Generator to Run (Six hours or more)	---	---	---	---	---	2.4×10^{-3}/h	2.4×10^{-3}/h

*Taken from Table A.5.9 of LGS-PRA[17].
**This is an average value, but this report deals mainly with plant or design specific evaluations.
***Derived from Table 9.6.8 of NUREG/CR-2989 (SWS, below average procedures).

5. SUMMARY OF ACCIDENT SEQUENCE QUANTIFICATION AND IDENTIFICATION OF DOMINANT CONTRIBUTORS TO CORE DAMAGE FREQUENCIES

This section describes the SNPS-PRA approach to quantification of the accident sequences and the BNL modifications in this approach, and presents the revised results of the BNL review. Subsection 5.1 presents the SNPS-PRA and the BNL approaches and highlights the main differences; further details are given in Appendices 5A to 5G. Subsection 5.2 presents the BNL revised results compared with the SNPS-PRA results: this is the summary of results of this review study. Subsection 5.3 provides additional insight into the results by presenting a limited sensitivity analysis with regard to some other different assumptions.

The quantification results presented are point estimates of the accident sequence frequencies. Uncertainty analysis was outside the scope of the review.

5.1 Modifications Made by BNL in the Accident Sequences Quantification

Subsection 5.1.1 describes the SNPS-PRA accident sequence quantification approach and presents the resulting accident sequence frequencies and the total frequency of core damage. Subsection 5.1.2 highlights the BNL approach followed in the review of the SNPS-PRA, and refers to the detailed description in the Appendices.

5.1.1 Overview of the SNPS Approach to Accident Sequence Quantification

In the SNPS-PRA, accident sequences were defined in terms of combinations of safety function failures given the occurrence of an initiator. These combinations were generated with the help of the functional event trees (see Section 3.1.2). The branch point probabilities in the event trees were calculated (as probabilities of function failures). To calculate the probability of each accident sequence, the failure probabilities of the functions involved in the sequence were multiplied by the frequencies of the corresponding initiators.

The failure probabilities for the functions were derived on the basis of the system fault trees (Table 3.1) and in some cases with the help of functional fault trees* and/or the functional-level event trees, or on the basis of additional explanations supplied in the PRA.

The unavailabilities of the frontline systems were calculated from the corresponding system fault trees (see Section 3.3). The frontline system fault trees contain failures both of frontline system hardware and of support systems, and these failures were further resolved down to the component level. Hardware, as well as test, maintenance, and human error contributions to the component unavailabilities were considered.

This quantification procedure was followed for all the functions on the event trees that model the plant response to the various initiators (see

*The SNPS-PRA refers to functional fault trees in several places and states that they are developed in detail in Appendix B.10, but Appendix B.10 does not include any functional fault tree.

Section 3.1 and Appendices 5A to 5G). The accident sequences of each event tree were classified into three categories: core damage sequences, non core damage sequences, and transfers (see Section 3.1). The transfer sequences were the ones judged to be more appropriately modeled in a different functional event tree.

In addition, all the core damage sequences were divided into classes according to the nature and scenario of core damage:

a) Class I core damage sequences are characterized by the loss of core coolant inventory makeup and core damage before containment failure.

b) Class II sequences comprise events involving loss of long-term containment heat removal function resulting in containment failure which may be followed by core damage. Only part of this class will result in a core damage state.*

c) Class III core damage sequences are characterized by LOCA in drywell conditions.

d) Class IV are ATWS sequences with containment failure prior to core damage.

e) Class V are sequences of LOCA outside containment, which bypass the suppression pool and drywell.

The total core damage frequency is the sum of the frequencies of all the core damage sequences. Figure 5.1, from the SNPS-PRA, shows the total core damage frequency, as well as the frequency of each class as calculated in the PRA study. The largest contribution to core damage frequency is seen to be from Class I, loss of coolant makeup; it is larger than the sum of the contributions from all the other classes. The total core damage frequency in the SNPS-PRA is estimated at 5.5×10^{-5} per reactor year. Table 5.14 includes a summary of dominant sequences calculated in the SNPS-PRA.

5.1.2 BNL Modifications to the Accident Sequence

BNL comments on the SNPS-PRA approach were given in Sections 3.1 and 3.2 when functional event trees and treatment of dependencies were discussed. In general, BNL found that the SNPS-PRA approach included considerable detail and tried to address the modeling of the accident sequences and its quantification as realistically as possible based on the SNPS specific design and past nuclear power plants' experience. BNL agrees to the general approach used. Most BNL comments and modifications relate to quantification. However, some relate to the specific modeling of certain sequences.

The BNL review of the SNPS-PRA functional event trees had two parts:

a) A case by case review of the functional event tree accident sequence modeling.

*In the SNPS-PRA, it is considered to be a core vulnerable state. In the BNL review, it is considered as a core damage state, even though core damage will not always occur following the containment failure.

b) A case by case review of the functional event tree quantification.

Both parts of the review were based on the information provided in the SNPS-PRA and its appendices, the SNPS-FSAR, the SNPS plant specific emergency procedures, the fault tree analysis of the systems, and the system description and drawings. In addition, realistic calculations of BWR plant response to transients were consulted in several GE, BNL, ORNL and other reports (referenced in Section 5.4 and in the previous sections). This information made it possible to check the validity of the modeling and the quantification of the SNPS-PRA approach. It should be noted that the PRA itself included the needed information in many cases.

Highlights from the results of BNL review of the functional event tree modeling were presented in Section 3.2. Additional detail on modeling changes and the reasoning behind them are presented in the appendices to this section. These appendices provide BNL revised functional event trees, which include the modeling changes that were judged important and also the re-quantification by BNL. Each event tree is accompanied by a table explaining the values used on the event trees and their sources, or the reasoning that led to their choice. All the SNPS-PRA initiators were treated. To facilitate comparisons between the SNPS-PRA and the BNL revised event trees, the appendices are ordered in the same way as the sections of the SNPS-PRA:

Appendix 5A: Deals with all the transient with successful scram discussed in Section 3.4.1 of the SNPS-PRA, except Loss of Offsite AC Power (Section 3.4.1.6 in the PRA), which is dealt with in Appendix 5B.

Appendix 5B: Loss of Offsite Power Event Tree (PRA Section 3.4.1.6).

Appendix 5C: Treats LOCA both inside and outside containment (Section 3.4.2 of the SNPS-PRA).

Appendix 5D: Treats the ATWS sequences and provides BNL revised event trees (Section 3.4.3 of the SNPS-PRA).

Appendix 5E: Reviews the transients initiated by the loss of a reference leg in the water level instrumentation system (Section 3.4.4.3.1 of the SNPS-PRA).

Appendix 5F: Treats the case of loss of drywell cooling for all transients and for the case in which this event is the initiator (Section 3.4.4.3.2 of the PRA).

Appendix 5G.1: Presents the case of the excessive release of water at Elevation 8 of the reactor building. In this case, however, reference is made to the BNL review report[1] of this accident sequence (Section 3.4.4 of the PRA).

Appendix 5G.2: Loss of a DC bus is treated (Section 3.4.4.2 of the PRA).

Appendix 5G.3: Revised tree for the case of loss of the service water system is presented (Section 3.4.4.4 of the PRA).

The Appendices 5E and 5F are an in-depth review of the report[2] "Review of Shoreham Water Level Measurement System", from which Sections 3.4.4.3.1 and 3.4.4.3.2 of the SNPS-PRA are a summary.

In general, the BNL review resulted in modifications related to the quantification of almost all the SNPS-PRA. The reasons behind quantification changes are explained in the tables attached to the revised event trees (see Appendices 5A to 5G). Each appendix provides the background information on the SNPS-PRA approach for the case, the general reasons for BNL modeling changes, and the results obtained.

The next section focuses on the results, and presents the main differences from the SNPS-PRA. The summary of the findings from the appendices is also given in Table 5.1, where it is compared with the summary of SNPS-PRA results.

5.2 Summary of the Results of the BNL Review in Comparison with the SNPS-PRA

The summary tables of this report are presented in Section 5.2.1, along with a discussion of the results for each accident sequence group. Section 5.2.2 provides some additional tables for comparisons such as the list of dominant sequences in each core damage class and SNPS-PRA and BNL dominant sequence lists.

5.2.1 Summary of the Results

Table 5.1 presents a summary of the BNL review and SNPS-PRA results. It is seen that in the BNL review the core damage frequency increased by a factor of 2.5 (1.4E-4 vs. 5.5E-5/yr) as compared with the SNPS-PRA. From Table 5.1 the following comments can be made:

- o The major contributions to the increase in the revised BNL core damage frequency are due to ATWS, LOOP, Transients with Scram, and Internal Flooding initiation.

- o The core damage frequency contribution from LOCA outside drywell is about five times as high in the BNL review as in the SNPS-PRA. Even though its contribution to total core damage frequency is very small ($\approx$ 0.2%), it may be a very important contribution to risk.

- o The contribution from transient initiators is increased by a factor of 1.7, largely because of the revised frequencies of the initiators, discussed in Section 4.1. It is important to point out that, if a common-mode miscalibration of all water level sensors, which are the only signals for the automatic initiation of HPCI, RCIC, ADS, LPCI, and LPCS in the case of a transient, with a probability of 2×10^{-3} as given in the SNPS-PRA (page A-121) were used, the core damage frequency from transient initiation would be about 5.4×10^{-5} instead of 2.2×10^{-5}. However, BNL previously judged that the probability used in the SNPS-PRA for the miscalibration was not realistic, and the modification to these numbers are given in Appendix 5A.1 and in Section 4.3.

- o The contribution from LOCA inside drywell remained practically the same as in the SNPS-PRA.

Figure 5.1 shows the results by core damage classes and compares them with Figure 3.5-3 (Page 3-338) of the PRA. The results summarized by groups of initiators are given in Figure 5.2 in a "pie chart".

Figure 5.3, reproduced from the SNPS-PRA, provides the BWR-RSS and the SNPS-PRA results for comparison.

The main reasons for the higher BNL results are discussed in detail in the appendices to this section. Here a brief summary of the main differences is presented.

5.2.1.1 Loss of Coolant Accidents (LOCA) Inside Drywell

LOCAs are minor contributors to core damage frequency. Large and medium LOCAs were modeled and quantified by BNL in the same way as in the SNPS-PRA. BNL used a more realistic modeling for PCS recovery in the long term which resulted in a small decrease in the Large and Medium LOCA contribution to Class II sequences. In addition, for Large LOCA in liquid lines originating at a low point in the RPV, it was assumed that break discharge flow rates would be higher than the hotwell makeup can replenish. This leads to the small increase in Class III contribution in the BNL results. Pressure vessel failure was not reviewed in detail and its failure frequency remained unchanged. The LOCA-in-drywell initiators are the major contributors to Class III. The results of this review (Table 5.2) show little difference between the BNL and the SNPS values.

5.2.1.2 Anticipated Transients Without Scram (ATWS)

The SNPS-PRA shows that ATWS sequences are a major contributor to core damage frequency. The BNL review found that some of the SNPS-PRA assumptions had additional implications which were not fully addressed in the PRA:

a) Lowering water level below Level 1 has the implication of MSIV closure and is accompanied by a high probability of operator failure to inhibit ADS.

b) Manual feedwater runback was treated in the SNPS-PRA as part of the turbine trip initiator event tree rather than on the functional event tree. However, the large unavailability value used for this function resulted in overestimation of some of the sequence frequencies.

The BNL review identified three areas of concern:

a. The ATWS physical analysis: Because available ATWS thermal hydraulic analysis results directly applicable to a BWR-4 reactor with manual 43 GPM SLC system are limited, it is difficult to establish critical parameters that define the condition of the SNPS and the time available for operator actions. Based on the limited analyses, engineering judgment was used in reviewing the SNPS analysis and changes were made to the SNPS event trees.

b. The SNPS specific ATWS emergency procedures: BNL considers the current emergency procedures to be unsatisfactory in areas of operator

control of RPV water level, ADS inhibit function, and PRV pressure control.

c. The extent of operator action required during an ATWS event to secure the plant to hot shutdown: The SNPS requires manual actions for most of the ATWS mitigation systems. However, the operator has very little time to perform these tasks, which often must be done within 10 minutes after the onset of the event. This is why the Shoreham ATWS core damage frequency is about an order of magnitude larger than that of the Limerick or the GESSAR-II standard plant. It is prudent to recognize that large uncertainties are associated with the estimates of human errors and therefore the ATWS core damage frequency could be very sensitive to changes in the human error probabilities.

Finally, BNL performed a realistic re-assessment of the SNPS ATWS event as shown in Appendix 5D. The results indicate that, given the assumptions used, there is only a small increase due to different assumptions and modifications to the event trees. The ATWS core damage frequency calculated by BNL using the SNPS initiator frequencies is 2.2×10^{-5}, compared with the SNPS value of 1.8×10^{-5} (see Table 5.3). Use of the BNL initiator frequencies raises the total core damage to 4.5×10^{-5}, about a factor of 2.5 higher than the SNPS value. Note that the BNL initiator frequencies, like those in the SNPS-PRA, distinguish whether the plant operated above or below a plant condition of 25% power.

5.2.1.3 Transients with Successful Scram

Apart from loss of offsite AC power, which is treated separately in the next subsection, the SNPS-PRA included separate event trees for loss of feedwater, MSIV closure, and loss of condenser transients. Table 5.4 shows that the main contributors to the core damage frequency are the loss of condenser and the turbine trip transients, and that the increase in core damage frequency in the BNL review is due to the different frequency of transient initiation, described in Section 4.1.

Table 5.4 shows that for Class II, if the effect of initiating event frequencies is not taken into account, the SNPS-PRA and the BNL review obtained the same result: 4.8E-6. However, there are two differences between the SNPS-PRA and BNL review which balanced each other:

a) BNL included a dependence between Q and W functions in the functional level event trees that increased the Class II results.

b) BNL assumed that for a case of successful feedwater injection (Q is successful) throughout the transient, no additional means of containment heat removal are required. (See for example Tables 5.A.1 and 5.A.2.)

The two SORVs case was treated in the SNPS-PRA in great detail without any impact. BNL also found it to be a minor event, but not of negligible effect as in the PRA. BNL developed one case in detail (Table 5A-2 in Appendix 5A.1 for the turbine trip transient) to show that it has some impact and should not be totally ignored, as one may conclude from the SNPS-PRA results. The results for two SORVs are calculated by BNL to be 4×10^{-7} in Class I,

similar to the results of the IORV transient with successful early scram. They also contribute more than the small LOCA sequences.

Finally, the transient results of BNL include the impact of miscalibration assuming the probability of gross miscalibration of all four level sensors to be 1/10 of the value used in the SNPS-PRA (see Section 4.3). If the SNPS-PRA value of $2x10^{-3}$ was used (which is judged by BNL to be unrealistically high--see Appendix 5A.1.4) the transient contributions would become over $5.4x10^{-5}$. The BNL review concluded that the transient group of initiators contribute 15% to core damage frequency.

5.2.1.4 Loss of Offsite AC Power (LOOP)

The SNPS-PRA treated the initiator in a detailed time phased event tree, using fossil plant experiential data for LOOP frequency. Diesel failure frequency and recovery factors were based on nuclear power plants' LERs. The event tree included dependences of RCIC, LPCI, and ADS upon conditions of DC power, suppression pool temperature, and drywell temperature and pressure. BNL did not change this modeling apart from the treatment of the initiator frequencies and of the loss of the containment heat removal function. The latter was transferred to MSIV closure in all cases, omitting the special case of recovery of diesels without recovery of offsite power for over 15 hours. This low frequency event of non-recovery of offsite power has a probability of occurrence of 3%, which make it an additional Class II contributor as seen from Table 5B.2 (sheet 2/5).

The quantification changes made by BNL were mainly a higher deterioration rate of the batteries between 4 and 10 hours, and the assumption of their loss at about 10 hours. Thus, HPCI and RCIC were assumed to be unavailable after 10 hours in the BNL review. The SNPS-PRA did not sufficiently support its assumption of the possibility that the battery will last 24 hours and allow for HPCI or RCIC operation for that long. Furthermore, several calculations of BWR suppression pool and drywell heat-up in blackout situations (or a statement in the SNPS-PRA itself), indicate that the drywell pressure may reach ≈ 60 psi at 13 to 15 hours which may render ADS unoperable, and lead to core damage condition earlier than 24 hours.

The SNPS-PRA described in detail its level measurement system as part of its in-depth model of the effect of loss of a reference leg of the system. This revealed that level instrumentation readings are lost in the control room during blackout with DC power available because they lack DC backup (initiation of HPCI and RCIC or ADS is not lost). This was not appropriately modeled in the SNPS-PRA treatment of the interaction of LOOP and loss of level instrumentation (see Section 3.4.4.3.2 of the PRA). This sequence was included in the BNL re-assessment as shown in Table 5B.1 (sheet 2/5) and discussed in more detail in the event tree of Table 5F-4, branch T_EIDGL. It contributes $1x10^{-5}$ to the core damage frequency because it impairs the operator ability to follow procedures (contradicting procedures) and to control HPCI without level information.

The LOOP frequency evaluated by BNL, based on a new NSAC report (NSAC/80--see Section 4.1.3), was found to be 0.15 per year, and the recovery probabilities were those given in Table 4.7 column 9. BNL judges that the value 0.15 is appropriate for the SNPS, which is part of the NPCC region, a fact not

included in the SNPS evaluation. The BNL LOOP frequency value is twice the SNPS-PRA value. The recovery probabilities of the SNPS-PRA are significantly larger than those used by BNL. These two changes partially balanced each other. Overall, the increase in the BNL results for the core damage frequency is due to the quantification changes and level instrumentation and only to a lesser extent to the re-evaluation of initiating event frequencies.

Diesel generator data used in the PRA were found reasonably conservative and reflect SNPS onsite power conditions when the PRA was submitted. New updated probabilistic evaluations of the onsite AC power were not submitted to BNL during its review, even though some changes were apparently taking place due to other licensing reviews, which have the potential to reduce the impact of the LOOP sequences.

As seen from Table 5.5, the loss of offsite AC power is a major contributor to the core damage frequency, and account for 25% of the total frequency.

5.2.1.5 Excessive Release of Water at Reactor Building Elevation 8

This initiator was treated in depth by BNL in a separate report[1]. As seen in Table 5.6, the BNL results are significantly higher, mainly because of two changes: (1) a higher initiator frequency calculated by BNL from a more up-to-date and elaborate model, and (2) an increase in the condensate injection failure probability (0.1 instead of 0.01 as was used in BNL revised transient event trees). Also, a time phased event tree was utilized to take into account the early failure of HPCI and RCIC at a water level lower than that for the LPCI/RHR or LPCS failure. This resulted in a 20% increase in the core damage frequency. More detailed results are shown in Appendix 5G.1 and in Ref. 1.

It will be seen in Section 5.2.1.7 below that BNL considered interfacing LOCA to be a significant initiator in the SNPS-PRA. This BNL result was obtained from the same considerations that resulted in the high core damage frequency calculated for the excessive release of water initiator. These considerations include the situation that all ECCS equipment may become compromised in a relatively short time that does not provide the operator sufficient time to recover. The conditional probabilities of core damage (given the initiator) for these sequences are higher than those found in other BWR-PRAs reviewed by BNL in the past. When the combined impacts of excessive release of water at elevation 8 and the interfacing LOCA are considered together, their contribution to SNPS risk is expected to be above 20%. Their calculated core damage frequency is, however, only ≈ 13% of the total.

5.2.1.6 Level Instrumentation: Loss of Reference Leg and Loss of Drywell Cooling

These two groups of initiators related to the water level instrumentation system were treated in greater depth than any other group. Past BWR-PRAs did not treat them in detail. The BNL review of these event trees is described in Appendices 5E and 5F, and the results are shown in Table 5.7. Loss of a reference leg is the major contributor to this group of initiators, and in the BNL review its contribution was increased by a factor of 3 compared with SNPS-PRA results. This increase is due to two major BNL changes:

a) The common mode failure due to maintenance of the second reference leg: BNL evaluation of this event was based on the LER data provided in the PRA and on BNL judgment related to probability of a human error. These resulted in an increase of the contribution of this sequence, i.e., loss of both reference legs. This event, loss of both reference legs, defeats the automatic initiation of all ECCS systems and leaves the operator without level information if the water level drops below level 3.

b) The miscalibration of the two sensors on the other leg: In this case the value used in the SNPS-PRA, 2×10^{-3}, for common miscalibration of two level sensors is reasonable and results in significant contribution to core damage frequency due to this group of initiators. This miscalibration, as well as the loss of DP cell, which is similarly important, have not been correctly included in the SNPS-PRA modeling. However, modification made at Shoreham will apparently reduce the impact of this sequence.

Loss of drywell cooling contributes an additional fraction to this group of initiators. The major contributor is the loss of off-site power transient with recovery of the diesels, but without recovery of the drywell cooling. This was not correctly modeled by the SNPS-PRA (see Appendix 5F).

The major contributors discussed above have $\sim 1\times10^{-5}$ contributions to SNPS core damage frequency (~7% of the total core damage frequency).

The increase in the BNL result in this case is due mainly to the BNL modeling, which included sequences not correctly treated in the SNPS-PRA.

The design of the SNPS has only two "safety related" reference legs, and four level sensors supply all ECCS initiation signals. In other plants HPCI and RCIC are initiated, at least in part, by different sets of sensors* and have more reference legs. A GE generic study of water level instrumentations[11] suggested improvements which have been implemented in the SNPS. However, the core damage frequency calculated in the BNL review with these improvements taken into account is comparable with the frequency in other plants before implementation of the recommended improvement.

5.2.1.7 Interfacing LOCA

Despite its low frequency, this is an important initiator. The increase in BNL review results by a factor of 5 above the SNPS-PRA value (Table 5.8), resulted partly from a change in the initiator frequency estimation and partly from BNL's judgment that condensate injection of 1000 gpm will be insufficient for a large interfacing LOCA in the LPCI system. The modeling and quantification of the event tree were only slightly modified by BNL. The "0.2" for condensate unavailability was based in the BNL review on different consideration (Appendix 5C.2). The frequency increase was based on LERs, distributed recently by the NRC[3,25], containing two precursor cases and 5 failures of testable check valves, which increase the probability of such an event. The impact of this sequence was discussed in Section 5.2.1.5 above.

*This is implemented in Shoreham in a recent design change to the Water Level Instrumentation System.

5.2.2 Dominant Sequences in BNL Review

The contributions to core damage frequency grouped according to their initiators were listed in Table 5.1 and summarized in the preceding section. In this section the individual sequences contributing to the SNPS core damage frequency are presented. Tables 5.9 to 5.13 list the dominant sequences contributing to each core damage class. Classes I, II, and IV have large numbers of contributors, but for Classes II and IV more than half the total frequency is attributed to a small number of contributors. Class I has the largest number of small contributors.

Finally, Table 5.14 provides a comparison of the dominant accident sequences in the BNL review and in the SNPS-PRA. The basic pattern in the SNPS-PRA of having no single sequence contributing a large fraction to the total SNPS core damage frequency is seen also in the BNL results. The most dominant contributors in the BNL list consist of accident sequences from each of the initiation groups, ATWS, loss of off-site AC power, and excessive release of water at elevation 8.

The following are some comments on the dominant accident sequences in Table 5.14:

a) 50% of core damage frequency is attributed to thirteen sequences in the SNPS-PRA, but to only ten in the BNL review.

b) The following sequences are dominant in both the SNPS-PRA and the BNL review:

   1) ATWS sequence of MSIV closure.

   2) The excessive water release sequence.

   3) Loss of off-site power sequences; however, several differences are noted and explained below.

   4) The most important single contributor of the water level measurement systems appears roughly in the middle of both lists, but more accident sequence contributors are included in the BNL results.

c) Important differences between the top sequences of the SNPS-PRA and those of the BNL review are as follows:

   1) Loss of condenser contribution T(C)UX and loss of a DC bus contribution T(D) D(I)Q rank much higher in the SNPS-PRA than in the BNL review, but their absolute frequency is almost the same.

   2) Loss of off-site AC power contributions appear in both results but differ in their details. The SNPS-PRA has the time-phase III and IV contributions corresponding to failures at 4 to 10 and 10 to 24 hours into the transients. BNL has only Phase IV ranking high, but it has, in addition, the sequences representing water level blackout conditions; these appear high in the BNL list and are missing from the SNPS-PRA.

3) The turbine trip ATWS sequence ranks high in the BNL review, but very low on the SNPS-PRA list.

4) The Loss of Service water system contribution is higher on the BNL review list.

5.3 A Limited Sensitivity Study

A limited sensitivity study was done to provide insight into the impact of changes in the assumptions used in this PRA review. It focused on the impact on core damage frequency from two types of changes:

a) Changes in a few assumptions that represent modeling uncertainties.

b) Changes in a few assumptions that illustrate the particular importance of these assumptions in this PRA, or the great importance of selected safety systems with regard to core damage frequency.

The following tests of assumptions to represent modeling uncertainties:

a) LOOP frequency and recovery probabilities: The BNL review modified the data recommended in NSAC/80 for deriving LOOP frequency and recovery probabilities, as described in Section 4.1.3 above and shown in Tables 4.6 and 4.7. The results of the BNL review that included three additional LOOP events occurring during shutdown were compared with the results obtained by using the NSAC/80 recommended data (without modification)--see Table 5.15 line 1. The inclusion of the three LOOP events that NSAC/80 did not recommend using in the derivation of LOOP frequency and recovery probabilities had a minimal impact on the result. Hence, BNL concluded that it is better to include these events and obtain a complete data base containing all total LOOP events than to screen out events on a judgmental basis.

b) The BNL review was performed with realistic assumptions that led to a probability of 2×10^{-4} or less for gross miscalibration of all four level sensors (see Sections 4.3 and 5A.1.4). The SNPS-PRA assumed that this probability may be conservatively quantified as 2×10^{-3} (Appendix A.3 of the SNPS-PRA) but did not model the effect of this assumption correctly for the case of transients. The impact of this quantification in the case of transients is shown in Table 5.15 line 2, which shows that the total SNPS-PRA core damage frequency would have been much larger if the conservative value of the PRA had been used with adequate modeling.

c) The BNL review used transient-initiator frequencies based on experiential data from BWR plants averaged over their entire operating period from the date of initial commercial operation. The SNPS-PRA used a "weighted average" approach with the experience from the first year of plant operation weighing 1/35 and the subsequent experience weighing 34/35. This was done in the SNPS-PRA in order to represent the mature plant. However, the SNPS-PRA used an earlier data evaluation (from EPRI report NP-801) rather than the updated one from NP-2230, as described in Section 4.1.2 above. The impact of removing the experience of plant occurrences from the first year of

operation is shown in Table 5.15 line 3 by comparing the results from the use of columns 10 and 11 of Table 4.2. These columns were explained in Section 4.1.2 and are considered a more appropriate modeling of the impact of the transient initiator frequencies. The results are changed by about 25% for the transient initiator frequencies, but by only 10% for the total core damage frequency calculated in this review. BNL considers the approach of using data from the start of commercial operation to be more appropriate and to account more realistically for a possible "wear-out" period later in a plant's lifetime, if the average plant risk from its entire lifetime is desired.

d) Both BNL and the SNPS-PRA included credit for PCS and condensate systems in their analyses of medium and large LOCAs. Such credit was not taken in some past PRAs as seen by comparing the success criteria shown in Table 2.6. Table 2.5 (Table 1.5.2 in the SNPS-PRA) shows that the stated SNPS-PRA success criteria do not include credit for PCS and condensate in all cases of LOCAs (see note 5 to Table 2.5). The impact of the two sets of success criteria (with and without consideration of note 5 in Table 2.5) are shown in Table 5.15 line 4. Comparison of these results with the results in Table 5.2 for LOCAs shows that these assumptions have great impact on the Class III core damage frequency. The BNL review considers the inclusion of the credit for PCS and the condensate system to be more appropriate if realistic PRA results are desired. It is also important to consider this type of change when comparing past PRAs with the SNPS-PRA.

The following tests of assumptions illustrate the importance of selected assumptions made in the PRA or the importance of some safety systems in impacting the core damage frequencies in this particular PRA.

a) BNL basically used the SNPS-PRA data on diesel generator availability and recovery probabilities as discussed in Section 4.2.2. Modifications are being made at SNPS to the on-site AC power supply system. To illustrate the impact of a possible increase in the availability of this system, a case study is suggested in Section 4.2.2 which is compared in Table 5.15 line 5 with the baseline data used by the SNPS-PRA and the BNL review. It is seen that a significant reduction (10 to 15%) in the total core damage frequency can be obtained by increasing the availability of the on-site AC power supply system.

b) Explicit credit to the Turbine Building Service Water System (TBSWS) is given only in the analysis of the loss of service water transients in the SNPS-PRA. However, apparently credit for this system was also considered in determining the availability of the RHR and the RCIC in the steam condensing mode, as partly shown in Section 3.3. BNL found the contribution to core damage frequency of this system to have a large significance, as shown in Table 5.15 line 6. If no credit to this system is given, a 20% increase in the total core damage frequency can be calculated. The impact in Class I is from the loss of service water transient only and in Class II from an increase in the unavailabilities of the RHR and RCIC in steam condensing mode, for all transients including the loss of service water transient.

c) Credit to the condensate system injection as part of the low pressure system injection was not given in all past PRAs, as seen from the comparison in Table 2.6 above. Furthermore, if improvement in the availability of this system could be claimed, significant reduction in the contribution of some important sequences could be obtained. The important impact of this system in the SNPS-PRA is shown in Table 5.15 line 7. The results are quite linear with the availability assumed for this system; in the case of Classes III and V an unavailability of 0.2 was generally assumed, and in the case of Class I, 0.1 and 0.2 were used (with the average being about 0.15). BNL considers that credit to the condensate system should be given if realistic PRA results are desired. However, it is important to consider this credit when comparing different PRA results.

d) Four cases related to ATWS were studied. Like the other cases, they are given for illustration purposes only, and they show some of the different contributions to core damage frequency from ATWS. The case in line 8 shows the impact of operator failure to inhibit ADS when low low level is reached in the RPV. The results are clearly sensitive to the quantification of operator error probability. Improvements to ADS manual inhibit have been suggested by GE and apparently applied to the SNPS recently, but credit for any improvements was not given in either the SNPS-PRA or the BNL review. The case in line 9 shows the effect of more reliable SLC, which seems to be as large as that of ADS inhibit in the BNL model. The case in line 10 is also as important as ADS inhibit.

   An increase in the SLC system flow rate (from 46 to 86 gpm) or an equivalent increase in the boron concentration (by a factor of 2) will tend to allow for somewhat increased time for the operator to respond to the ATWS incident. This results from the assumption that manually putting a double capacity SLC into operation after 15 to 20 minutes instead of 10 minutes, will lead to approximately the same total amount of power being transferred to the suppression pool and drywell.

   Based on the above, the BNL event trees for ATWS were reevaluated assuming operator response required in 20 minutes instead of 10 minutes as assumed in the BNL base case. The results are shown in line 11 of Table 5.15. The estimate assumed that a higher probabilty for feedwater runback will ensue from the larger response time available (0.1/0.9 vs. 0.2/0.8) and that "U_H" will be 10 to 15% smaller (if "U_H" is not equal to 1.0).

e) The cases in lines 12 and 13 are similar and illustrate the impact of protecting one train of coolant injection from the impact of a very large flooding. They show that it may be possible to eliminate the flooding sequences from the main list of core damage contributors. The SNPS-PRA as well as the BNL review conservatively assumes that all injection but the condensate system would be lost in case of a large flooding.

f) The RCIC in the steam condensing mode is not normally allowed in SNPS operation. The emergency procedures do not refer to it. Thus, it is

a cooling mode for the very special cases of severe accidents when all other long-term cooling modes failed. This was not considered in quantifying the failure of the operator to initiate the containment heat removal mode. Line 14 shows the total impact of this system in the BNL review.

g) BNL was informed after the review was completed that the water level instrumentaion system is undergoing modifications to include four additional level transmitters (two on each reference leg) which would be used to initiate HPCI level 2 and level 8 signals separately from RCIC and other ECCS systems. Furthermore, they will be connected to the other DC buses, so that any reference leg side would not coincide with a single DC bus. These changes can potentially remove some of the sequences of Table 5E-2, and a significant reduction in core damage frequency can be obtained as seen in Table 5.15 line 15.

h) One of the NRC comments was that the Control Rod Drive (CRD) system may provide some additional risk reduction. The impact of including CRD is tested in line 16. The system reliability is assumed ideal, and it is assumed that it affects all the "UX" sequences apart from the "UX" belonging to the LOOP transient. However, the system can provide adequate core injection to remove decay heat only after about two hours of a transient initiation. This implies that HPCI and RCIC failure to start, or being in maintenance, are not recovered by CRD. Similarly, the miscalibration with failure of the operator to manually initiate injection and failure of Divisions I and II contributions to HPCI/RCIC failure would fail CRD as well. The estimated impact of CRD on the "UX" sequences is shown in line 16.

Note that this limited sensitivity study was done for illustrative purposes only, to provide another point of view on the results of the SNPS-PRA and the BNL review which are summarized in Sections 5.1 and 5.2.

5.4 References to Section 5 and Appendices

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16. Perkins, K. R., "Success Criterion for PCS with Two or More SORV's," Memorandum to D. Ilberg, Brookhaven National Laboratory, August 21, 1984 (Based on Ref. 8 above and Ref. 4 of Section 2).

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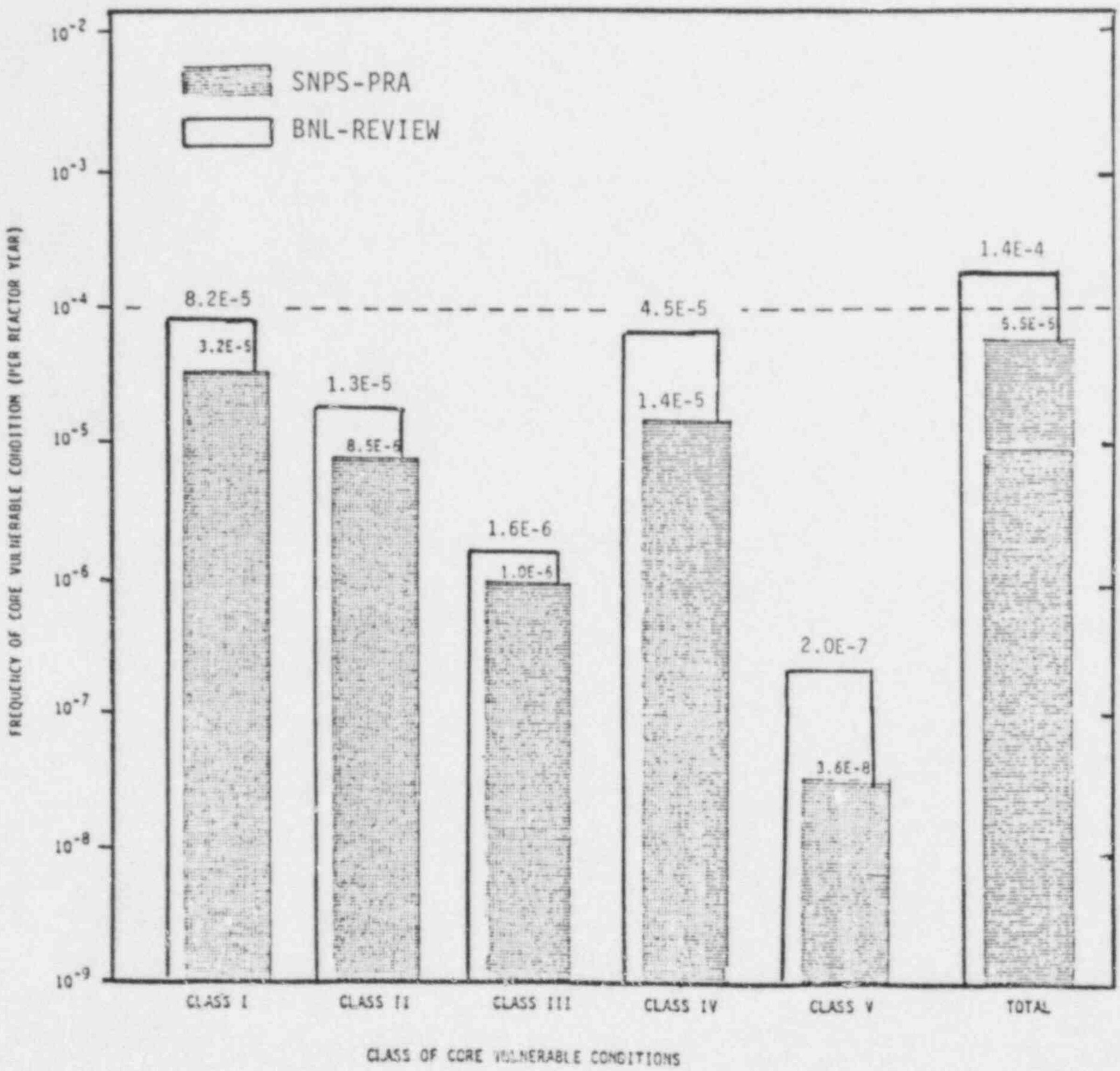


Figure 5.1 Summary of the Results of the Event Tree Quantification Displayed by Class of Postulated Core Damage Condition.

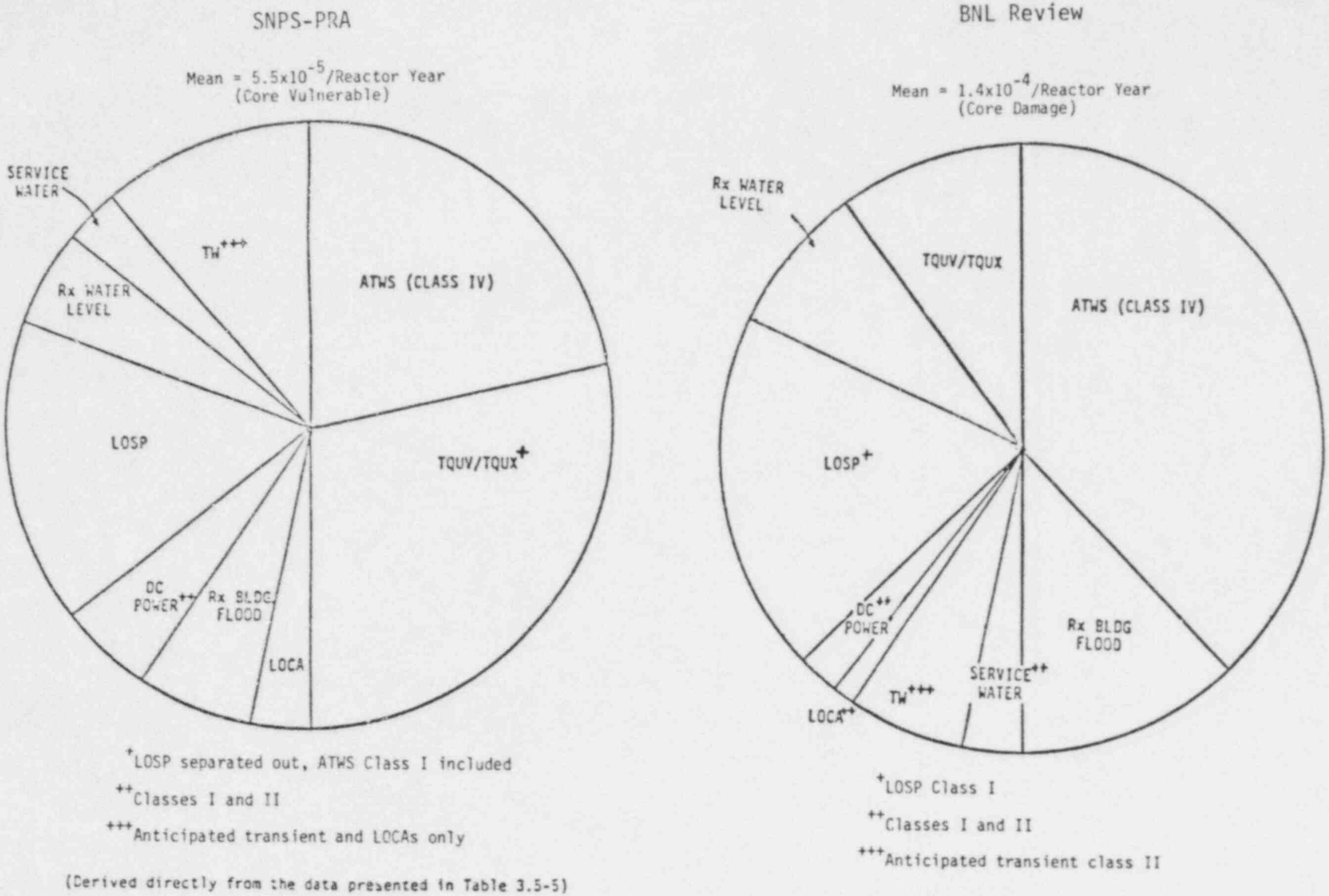
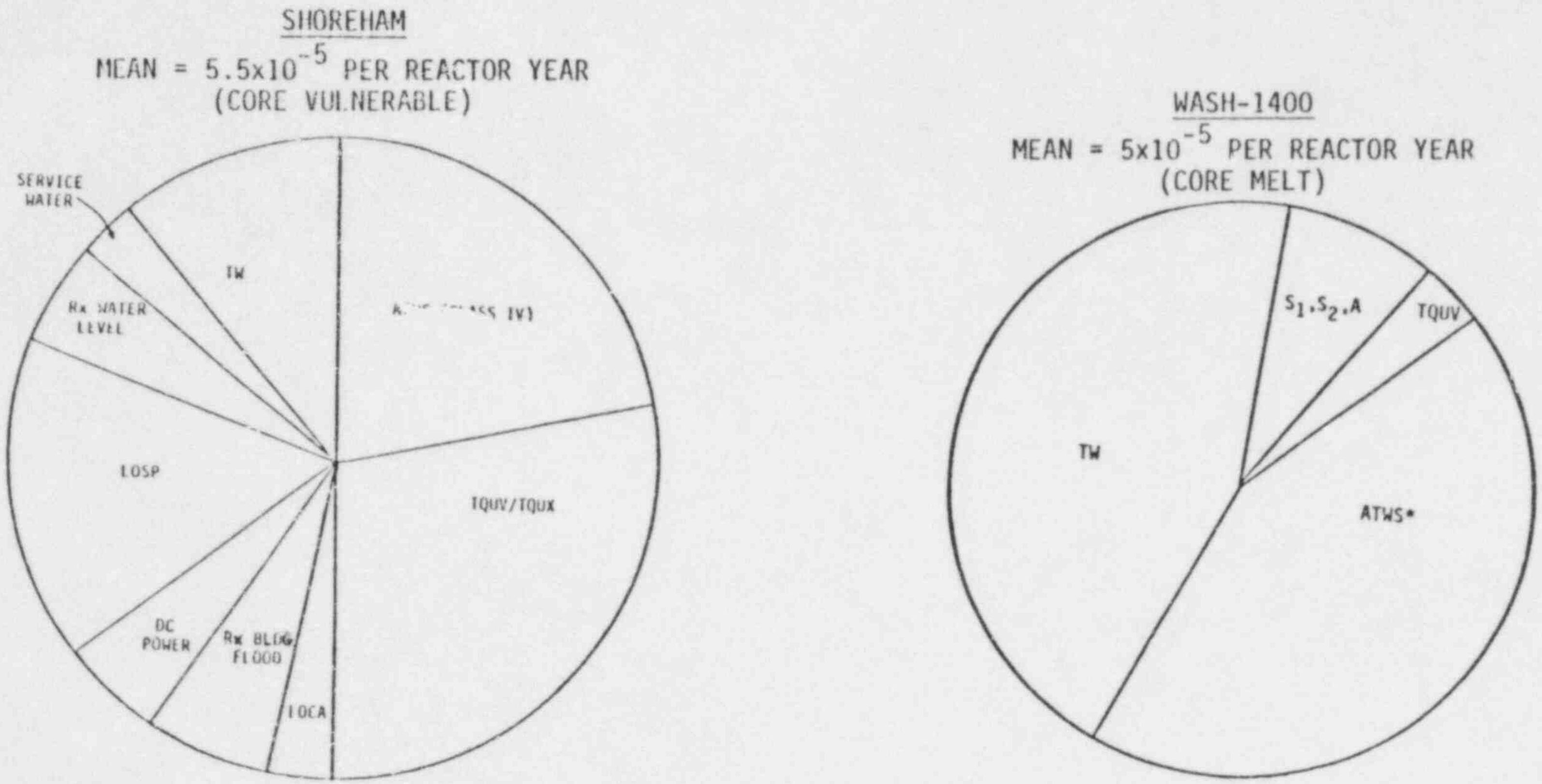


Figure 5.2 Comparison of the SNPS-PRA and the BNL Review Contributing Accident Sequences to the Calculated Core Damage Frequency (per Reactor Year) Due to the Identified Accident Sequence Contributors.



* Subsequent to WASH-1400, NRC evaluations of the potential contribution of ATWS to core melt (e.g., NUREG-0460) placed the frequency of ATWS in a BWR at nearly twenty times that evaluated in WASH-1400. If this were incorporated into the figure, it would be the single dominant contributor to core melt and would be significantly larger than the frequency of core melt calculated for Shoreham.

Figure 5.3 Comparison of the Contributions of Various Accident Sequences to the Calculated Frequency of Core Melt (from WASH-1400) and to the Calculated Frequency of Core Vulnerable Conditions (from the Shoreham Analysis). Area of "Pie Chart" is Proportional to Mean Frequency. Reproduced from SNPS-PRA.

Summary Table 5.1 Comparison of SNPS-PRA and BNL Review Results

Accident Sequence Initiator		Core Damage (CD) Class I	II**	III	IV	V	CD
Loss of Coolant Accidents (LOCA)	SNPS		1.0E-6	1.0E-6			2.0E-6
	BNL		5.3E-7	1.3E-6			1.8E-6
Anticipated Transient Without Scram (ATWS)	SNPS	4.0E-6		2.1E-9	1.4E-5		1.8E-5
	BNL	*		2.8E-8	4.5E-5		4.5E-5
Loss of Offsite AC Power (LOOP)	SNPS	9.9E-6	1.1E-6				1.1E-5
	BNL	2.9E-5	1.4E-6				3.0E-5
Transients (Turbine Trip Manual Shutdown, MSIV and other)	SNPS	8.7E-6	4.8E-6				1.3E-5
	BNL	1.5E-5	6.4E-6				2.2E-5
Level Instrumentation (Reference leg and drywell cooling)	SNPS	3.8E-6	1.2E-7	5.2E-9			3.9E-6
	BNL	1.2E-5	2.5E-8	1.5E-7			1.2E-5
Flooding at Elevation 8 of Reactor Bldg.	SNPS	3.1E-6	7.8E-7				3.9E-6
	BNL	1.8E-5	2.0E-6				2.0E-5
LOCA Outside Drywell	SNPS					3.7E-8	3.7E-8
	BNL					2.0E-7	2.0E-7
Loss of Service Water, or DC Bus	SNPS	3.0E-6	7.7E-7				3.8E-6
	BNL	7.6E-6	2.4E-6				1.0E-5
TOTAL	SNPS	3.2E-5	8.5E-6	1.0E-6	1.4E-5	3.7E-8	5.5E-5
	BNL	8.2E-5	1.3E-5	1.5E-6	4.5E-5	4.2E-7	1.4E-4

*In BNL review all ATWS sequences are assumed to lead to core damage class IV. This is based in part on the judgment that the operator will not be able to inhibit ADS.

**Class II leads in many cases to containment failure without loss of core cooling. Therefore, only a part of Class II results in core damage.

Table 5.2 Core Damage Frequency for LOCA in Drywell Initiators

	Class II Frequency		Class III Frequency		Total Core Damage Frequency	
	SNPS	BNL	SNPS	BNL	SNPS	BNL
Large LOCA	7.0E-7	2.8E-7	1.8E-7	3.7E-7	8.7E-7	6.5E-7
Medium LOCA	2.7E-7	2.1E-7	4.9E-7	6.1E-7	7.6E-7	8.2E-7
Small LOCA	2.4E-8	3.6E-8	1.6E-8	0.8E-8	4.0E-8	4.4E-8
Reactor Pressure Vessel LOCA			3.1E-7	3.1E-7	3.1E-7	3.1E-7
Total	1.0E-6	5.3E-7	1.0E-6	1.3E-6	2.0E-6	1.8E-6

Table 5.3 Core Damage Frequency for ATWS

	Core Damage Frequency		
	SNPS	BNL FT/ET*	BNL ALL
Turbine Trip	3.5E-6	4.7E-6	2.9E-5
MSIV Closure/Loss of Condenser Vacuum	8.2E-6	7.2E-6	1.1E-5
Loss of Feedwater	4.6E-6	9.2E-6	2.6E-6
Loss of AC Offsite Power	7.6E-7	7.9E-7	1.4E-6
Inadvertent Open Relief Valve	3.2E-7	4.3E-7	7.1E-7
Total	1.8E-5	2.2E-5	4.5E-5

*BNL FT/ET denotes the results of the changes in fault trees and event trees made by BNL excluding the changes in the initiating event frequencies.

Table 5.4 Core Damage Frequency for Transient Initiators

	Class I Frequency			Class II Frequency			Total Core Damage Frequency		
	SNPS	BNL FT/ET	BNL ALL	SNPS	BNL FT/ET	BNL ALL	SNPS	BNL FT/ET	BNL ALL
Turbine Trip	2.5E-6	2.9E-6	5.2E-6	1.0E-6	8.4E-7	1.5E-6	3.5E-6	3.7E-6	6.7E-6
Manual Shutdown	1.4E-6	1.8E-6	1.8E-6	1.2E-6	9.0E-7	9.0E-7	2.5E-6	2.7E-6	2.7E-6
MSIV Closure	7.4E-7	1.3E-6	2.7E-6	3.5E-7	2.4E-7	5.0E-7	1.1E-6	1.5E-6	3.2E-6
Loss of Feedwater	2.0E-7	2.5E-7	1.6E-7	4.2E-8	3.7E-8	3.0E-8	2.4E-7	4.1E-7	3.0E-7
Loss of Condenser	3.0E-6	3.8E-6	4.8E-6	2.1E-6	2.8E-6	3.4E-6	5.2E-6	6.6E-6	8.2E-6
Inadvertent Open Relief Valve	6.8E-7	1.2E-7	3.3E-7	9.0E-8	2.4E-8	6.6E-8	7.7E-7	1.4E-7	4.0E-7
Total	8.7E-6	1.0E-5	1.5E-5	4.8E-6	4.9E-6	6.4E-6	1.4E-5	1.5E-5	2.1E-5

Table 5.5 Core Damage Frequency for Loss of Offsite AC Power Initiator

	Class I Frequency			Class II Frequency			Total Core Damage Frequency		
	SNPS	BNL FT/ET	BNL ALL	SNPS	BNL FT/ET	BNL ALL	SNPS	BNL FT/ET	BNL ALL
Loss of Offsite AC Power	9.9E-6	1.6E-5	2.9E-5	1.1E-6	0.7E-6	1.4E-6	1.1E-5	1.7E-5	3.0E-5

Table 5.6 Core Damage for Excessive Release of Water in Reactor Building Elevation 8 Initiator

	Class I Frequency			Class II Frequency			Total Core Damage Frequency		
	SNPS	BNL FT/ET	BNL ALL	SNPS	BNL FT/ET	BNL ALL	SNPS	BNL FT/ET	BNL ALL
Excessive Release of Water at Elevation 8	3.1E-6	~ 4E-6	1.8E-5	7.8E-7	~ 1E-6	2.0E-6	3.9E-6	~ 5E-6	2.0E-5

Table 5.7 Core Damage Frequency for Level Instrumentation and Drywell Cooling Failure Initiators

	Class I Frequency			Class III Frequency			Total Core Damage Frequency	
	SNPS	BNL FT/ET	BNL ALL	SNPS	BNL FT/ET	BNL ALL	SNPS	BNL ALL
Reference leg line leakage	2.4E-6	7.3E-6	7.3E-6				2.4E-6	7.3E-6
Loss of Drywell Cooling (and transient contribution)	3.3E-7	7.4E-7	9.3E-7				3.3E-7	8.7E-7
Isolation Transient Loss of Drywell Cooling	2.5E-8	8.2E-7	1.4E-6				3.3E-7	8.7E-7
Loss of Offsite AC Power with Diesel Recovery	8.4E-7	1.1E-6	1.9E-6				8.4E-7	1.9E-6
Small and Medium LOCA-Loss of Drywell Cooling	2.1E-7	4.0E-7	4.0E-7	5.0E-9	1.5E-7	1.5E-7	2.1E-7	5.5E-7
Total	3.8E-6	1.0E-5	1.2E-5	5.0E-9	1.5E-7	1.5E-7	3.8E-6	1.2E-5

Table 5.8 Core Damage Frequency for LOCA Outside Containment Initiator

	Class V Frequency		
	SNPS	BNL FT/ET	BNL ALL
Interfacing LOCA	2.4E-8	7.2E-8	1.8E-7
Steam Lines Break Outside Containment	1.1E-8	1.1E-8	1.5E-8
Feedwater Line Break Outside Containment	1.7E-9	1.7E-9	3.0E-9
Total	3.7E-8	8.5E-8	2.0E-7

Table 5.9 Class I Dominant Sequences*

1) T_EIDGL	1.0E-5	IB	
2) T_EIVD	6.7E-6	IB	
3) T_TQUX	5.5E-6	IA	
4) T_CUX	4.2E-6	IA	
5) T_E III DUX	3.3E-6	IB	
6) T_{SW}TSUV	2.6E-6	ID	
7) T_{SW}TSUX	2.6E-6	IA	
8) T_MQUX	2.5E-6	IA	
9) T_RRL$_{R1}$QUH	2.4E-6	IA	
10) T_RRL$_{R2}$QUH	2.2E-6	IA	Total Class I = 8.2E-5
11) $T_D D_I$Q	2.2E-6	IA	
12) T_RRO$_R$QUX	2.0E-6	IA	
13) T_EIGL	1.9E-6	IA	
14) T_E III DV	1.7E-6	IB	
15) M_SQUX	1.6E-6	IA	
16) T_EIDUV	1.4E-6	ID	
17) T_E III DX	1.2E-6	IB	
18) T_E III DUV	1.1E-6	IB	
19) T_E III DU'V	1.0E-6	IB	
20) T_EIUV	1.0E-6	ID	

*Without the contributors from excessive release of water at Elevation 8, which would rank high on the list (1.8E-5 is the total contribution of all "flooding" sequences).

Table 5.10 Class II Dominant Sequences

1)	$T_C W$	2.5E-6	
2)	T_E I IV W	1.4E-6	
3)	$T_{SW}TSW$	1.4E-6	
4)	$T_{SW}TW$	5.8E-7	
5)	$T_C U'W$	4.2E-7	Total Class II = 1.3E-5
6)	$T_{SW}TSUV'''W$	4.1E-7	
7)	$T_T QW$	3.8E-7	
8)	$T_M QW$	3.7E-7	
9)	$M_S QW$	3.2E-7	
10)	AV'V"W	2.0E-7	
11)	$S_1 UV'V"W$	1.8E-7	
12)	A few contributors from excessive release of water at Elevation 8, $2x10^{-6}$ in total.		

Table 5.11 Class V Dominant Sequences

1)	$A_{OUT}V$	2.0E-7	Total Class V = 2.0E-7

Table 5.12 Class III Dominant Sequences

1)	S_1UV	3.1E-7	
2)	AV	3.0E-7	Total Class III = 1.3E-6
3)	S_1UX	2.5E-7	
4)	S_1QEL	1.5E-7	

Table 5.13 Class IV Dominant Sequences

1)	T_TC_MKQ	9.1E-6	
2)	$T_MC_MKU_H$	8.3E-6	
3)	$T_TC_MKU_H$	6.5E-6	
4)	$T_TC_MKC_2$	4.2E-6	
5)	$T_TC_MKUU_H$	3.9E-6	
6)	$T_TC_MK_W$	2.4E-6	
7)	$T_TC_MKP_Q$	1.1E-6	
8)	$T_MC_MKUU_H$	1.0E-6	Total Class IV = 4.5E-5
9)	$T_MC_MKPU_H$	9.1E-7	
10)	$T_FC_MKU_H$	9.1E-7	
11)	$T_EC_MKU_H$	1.1E-6	
12)	$T_TC_MKPU_H$	7.5E-7	
13)	$T_FC_MKC_2$	5.7E-7	
14)	$T_FC_MKUU_H$	5.1E-7	
15)	$T_TC_MKPC_2$	4.5E-7	
16)	$T_TC_MKPUU_H$	4.2E-7	

Table 5.14 Summary Table of Dominant Accident Sequences Leading to Core Damage Conditions, Ranked by Frequency (per Reactor Year)

	Shoreham - PRA			% of Total		BNL - Review		
No.	Sequence Designator	Core Damage Frequency	Class/ Subclass	SNPS	BNL	Sequence Designator	Core Damage Frequency	Class/ Subclass
1	T(M2)C(M)C(2)	6.4E-6	IV	12	7	T(T)C(M)K(Q)	1.0E-5	IV
2	T(C)UX	3.1E-6	IA	17	14	T(E)IDGL	1.0E-5	IB
3	T(T)QUX	2.4E-6	IA	22	21	FS(O)QUX	~ 1.0E-5*	IA
4	T(D)D(I)Q	2.2E-6	IA	26	27	T(M)C(M)KU(H)	8.3E-6	IV
5	T(E) IV DUX	2.2E-6	IB	30	32	T(T)C(M)KU(H)	6.7E-6	IV
6	FS(O)QUX	1.7E-6	ID	33	37	T(E) IV D	6.7E-6	IB
7	T(E)III(C)DV	1.5E-6	IB	35	41	T(T)QUX	5.5E-6	IA
8	T(F)C(M)U	1.5E-6	IC	38	44	T(T)C(M)C(2)	4.2E-6	IV
9	T(F)C(M)UD	1.5E-6	IV	41	47	T(C)UX	4.2E-6	IA
10	T(C)W'W"	1.5E-6	II	44	50	T(T)C(M)UU(H)	3.9E-6	IV
11	M(S)QUX	1.3E-6	IA	46	52	T(E) III DUX	3.3E-6	IB
12	T(E)III(A)DUV	1.2E-6	IB	48	54	T(SW)TSUV	2.6E-6	ID
13	T(E)W(D)	1.1E-6	II	50	56	T(SW)TSUX	2.6E-6	IA
14	T(R)RQUX	1.1E-6	IA	52	57	T(M)QUX	2.5E-6	IA
15	T(F)C(M)C(2)	1.0E-6	IV	54	59	T(C)W	2.5E-6	II

*Estimated

Table 5.14 Continued

No.	Shoreham - PRA: Sequence Designator	Shoreham - PRA: Core Damage Frequency	Shoreham - PRA: Class/ Subclass	% of Total: SNPS	% of Total: BNL	BNL - Review: Sequence Designator	BNL - Review: Core Damage Frequency	BNL - Review: Class/ Subclass
16	T(E)IDUV	9.9E-7	ID	56	61	T(T)C(M)W	2.4E-6	IV
17	T(T1)C(M)C(2)	9.9E-7	IV	58	63	T(R)L(R1)QUH	2.4E-6	IA
18	T(T)W'W"	8.9E-7	II	59	64	T(R)L(R2)QUH	2.2E-6	IA
19	T(E)GOL	8.4E-7	ID	60	66	T(D)D(I)Q	2.2E-6	IA
20	M(S)W'W"	8.2E-7	II			T(R)RO(R)QUX	2.0E-6	IA
21	T(E)IUV	7.7E-7	ID			T(E)IGL	1.9E-6	IA
22	T(M)QUX	7.2E-7	IA			T(E) III DV	1.7E-6	IB
23	T(I)QUX	6.7E-7	IA			M(S)QUX	1.6E-6	IA
24	T(E) .I DU"V	6.5E-7	IB			T(SW)TSW	1.4E-6	II
25	T(M2)C(M)C(2)U	6.2E-7	IC			T(E)I IV W	1.4E-6	II
26	T(M2)C(M)CUD	6.2E-7	IV			T(E) III DX	1.2E-6	IB
27	T(E)C(M)C(2)	6.0E-7	IV			T(T)C(M)PC(Q)	1.1E-6	IV
28	FS(C)QUX	5.5E-7	ID			T(E)C(M)U(H)	1.1E-6	IV
29	T(D)QUX	5.3E-7	IA			T(E) III DUV	1.1E-6	I B
30	T(T1)C(M)U	5.3E-7	IC	70	72	T(M)C(M)UU(H)	1.0E-6	IV
	Total Core Damage = 5.5E-5					Total Core Damage = 1.4E-4		

Table 5.15 Results from a Limited Sensitivity Study

(Only the sequences affected by the changes that are studied are included in the results shown.)

No.	Case Studied	CD Class	Core Damage (CD) Frequency	Baseline Case in BNL Review	CD Class	Core Damage (CD) Frequency
1.	LOOP Initiator: NSAC/80 LOOP frequency and recovery probabilities for NPCC	I II Total	2.9E-5 5.4E-6 3.4E-5	LOOP Initiator: BNL Review LOOP and Recovery Probabilties for NPCC	I II Total	2.9E-5 6.0E-6 3.5E-5
2.	Miscalibration: Use of SNPS-PRA value of $2x10^{-3}$ in the "UX" Function	I	4.7E-5	Miscalibration: Use BNL-Review value of $2.x10^{-4}$ in the "UX" Function	I	1.5E-5
3.	Transients + ATWS: EPRI-NP-2230 Data Base Excluding First Year of Plant's Experience	I II IV Total	1.0E-5 4.3E-6 4.9E-5 6.3E-5	Transients + ATWS: EPRI-NP-2230 Data Base from All Years of Plant Operation (BNL Review)	I II IV Total	1.3E-5 5.5E-6 5.9E-5 7.8E-5
4.	Large and Medium LOCAs: No Credit to PCS or to Condensate System	II III Total	- 2.6E-6 2.6E-6	Large and Medium LOCAs: Credit Given to PCS and Condensate	II III Total	5.7E-7 4.1E-7 9.8E-7
5.	LOOP Initiator: Diesel Data = FTS = 0.01/d P(2/1) = 0.11 P(3/2) = 0.40	I II Total	5.1E-6 6.0E-6 1.1E-5	LOOP Initiator: Diesel Data = FTS = 0.02/d; P(2/1) = 0.19 P(3/2) = 0.63	I II Total	2.9E-5 6.0E-6 3.5E-5
6.	No Credit to TBSWS in the PRA	I II Total	2.1E-5 2.7E-5 4.8E-5	Credit Given to TBSWS in the PRA	I II Total	5.2E-6 1.3E-5 1.8E-5
7.	No Credit to Condensate System in the PRA	I III V	1.3E-4 2.1E-6 2.1E-6	Credit Given to Condensate System in the PRA	I III V	2.2E-5 4.1E-7 4.2E-7
8.	ATWS: ADS Inhibit Improved by 50% (Probability of failure decreased by factor of 2)	IV	1.3E-5	ATWS: ADS Inhibit from BNL Review Results	IV	1.9E-5

Table 5.15 Continued

No.	Case Studied	CD Class	Core Damage (CD) Frequency	Baseline Case in BNL Review	CD Class	Core Damage (CD) Frequency
9.	ATWS: SLC failure probability reduced by factor of 10	IV	5.8E-7	ATWS: SLC failure probability same as in BNL review.	IV	5.8E-6
10.	ATWS: Automatic FW Runback Assumed, that may Reduce Failure Probability by factor of 10	IV	1.0E-6	ATWS: Manually Initiated FW Runback	IV	1.0E-5
11.	ATWS: Increased SLC Boron Concentration by a factor of 2 (or alternatively 86 gpm SLC)	IV	~2.9E-5	ATWS: SLC injection of 46 gpm	IV	~3.6E-5
12.	Water Release at Elevation 8: One LPCI Train Protected Against Flooding	I	1.8E-7	Water Release at Elevation 8: BNL Review	I	1.8E-5
13.	LOCA Outside Containment: One LPCI Train Protected Against Flooding	V	4.2E-9	LOCA Outside Containment: BNL Review	V	4.2E-7
14.	No Credit to RCIC in the Steam Condensing Mode	II	9.1E-6	Credit given to Containment Heat Removal by the RCIC Steam Condensing Mode	II	3.7E-6
15.	Level Instrumentation System having additional four level transmitters for independent initiation of HPCI, RCIC, and low pressure ECCS	I	2.1E-6	Level Instrumentation System having four transmitters for initiation of HPCI, RCIC, and low pressure ECCS	I	7.3E-6
16.	Impact of Inclusion of Control Rod Drive System in the High Pressure Injection Function	I	1.8E-5	Control Rod Drive System not included in the PRA	I	2.0E-5

APPENDIX 5A

ANTICIPATED TRANSIENT WITH SUCCESSFUL SCRAM SEQUENCES

This appendix summarizes BNL's review of the contribution of transients with scram to the SNPS frequency of core damage. The review covered material presented in Sections 3.4.1.1 to 3.4.1.5 and Section 3.4.1.7 of the SNPS-PRA. The following transients are reviewed in this appendix:

a) Turbine Trip

b) Manual Shutdowns

c) MSIV Closure

d) Loss of Feedwater

e) Loss of Condenser Vacuum

f) Inadvertent Open Relief Valve (IORV).

The initiator frequency for these transients was reevaluated as discussed in Section 4.1 and summarized in Table 4.2 above. The frequency of manual shutdown in the SNPS-PRA was judged reasonably conservative and was not further reviewed by BNL. The SNPS-PRA value of 4.3 shutdowns per year was used in the BNL reassessment. For all other transients, the new reevaluated frequencies of Table 4.2 were used by BNL.

The SNPS-PRA attempted to take into account more frontline systems' interdependence in the event trees by increasing the event trees' detail. The interdependences between HPCI and RCIC were detailed in the SNPS-PRA event trees and the same was done for LPCI and LPCS. The condensate and feedwater pumps were explicitly treated in the event tree; the Containment Heat removal function was separated into contributions from RHR, PCS, and failure to recover from a MSIV closure. These improvements made the modeling of the transients' contribution to core damage in the SNPS-PRA more realistic.

The support system dependence was also treated in the SNPS-PRA. The treatment chosen was to screen selected support systems dependences and treat them in separate event trees so as to focus their impact better. Three support systems were treated in this way:

a) AC Power: Transient induced Loss of Offsite Power (LOOP), or LOOP occurring during the transient.

b) DC Power: Loss of a DC bus, both transient-induced and in the course of the transient.

c) Service Water: Loss of service water during the recovery from a transient.

BNL found this treatment helpful and added another support system to the list:

d) Drywell Cooler: Loss of drywell coolers following a transient. SNPS-PRA treated this explicitly on the transient event trees rather than by the same screening method they suggested. BNL used the screening method for loss of drywell coolers, but differentiated between transients that lead to MSIV isolation and non-isolation transients (see Appendix 5F).

Other support systems were included in the fault-trees analysis and their impact, if important, was accounted for in the front line systems' unavailabilities. Note, however, that some underestimation of support system contribution may result when the more rigorous CDFT is not used. As stated in Section 1.2, BNL judged this underestimation to be unimportant.

The SNPS-PRA treatment of anticipated transients is innovative in the division of the isolation transients into separately treated initiators. This was discussed in Section 2.2 above.

5A.1 TURBINE TRIP TRANSIENT

5A.1.1 Background

This is the most frequent transient. The frequency of the transient is 5 per year in the SNPS-PRA and 8 per year in the BNL review. This difference was discussed in Section 4.1. Here, the modeling and quantification differences between the SNPS and BNL approaches are discussed.

Following a successful scram, SRVs are opened to relieve the pressure that is rising in the RPV after the closure of the turbine stop valves. If none of the 11 SRVs opens, then the pressure inside the RPV will breach the pressure boundary at a weak point and a LOCA is assumed to occur. This is, however, a small probability relative to the large LOCA frequency, and has no substantial impact. The open SRV may fail to close after pressure is relieved. A single SRV may fail open with as high a probability as 0.1; however, this apparently does not change the course of the transient significantly, because the high pressure injection system can easily maintain pressure in the RPV in spite of the small loss of coolant inventory through the stuck open relief valve. However, given two SORVs, changes in plant behavior are expected, in three ways:

a) The RPV pressure will slowly decrease to a point at which high pressure injection may no longer be successful. This can happen as early as four hours[16] after transient with two SORVs initiation.

b) The suppression pool will heat up slightly faster, and reach 200°F at about 1 hour rather than 2 hours or more.

c) At the beginning of the incident, if there is no high pressure injection, the RPV water level will decrease faster given 2 SORVs than given none; this in turn will reduce the time for recovery of FW during the time period when water level decreases from level 2 to level 1.

The impact on the PCS availability, however, is small in the case of 2 SORVs relative to none.

The "Q" function is discussed in detail in the next section. The coolant injection functions and the ADS are modeled next on the event trees. Their quantification is based on the SNPS-PRA system fault trees. The unavailabilities were discussed in Section 3.3, when fault trees were reviewed.

The containment heat removal function includes the following:

1) RHR system unavailability.

2) The RCIC steam condensing mode with RBSWS cooling the RHRHX directly.

3) The PCS.

The RCIC steam condensing mode has a small contribution, considered in the SNPS-PRA to be 0.4. It might not even be this large if the RBSWS, which is common to both, is assumed to fail and no credit is taken for interconnecting the turbine building service water systems. The values of 0.4 or 4.4×10^{-5} are explained in the discussion in Section 3.3 on RHR with RCIC in steam condensing mode.

The PCS is dependent on the availability of offsite power, one circulating pump, the condensate pump, the MSIV, the feedwater discharge valves, and air ejection or mechanical vacuum pump. All these have a hardware failure probability assuming repair of 4.5×10^{-3}. BNL used a value of 0.004, as explained further in Section 3.3.2.14.

5A.1.2 The FW and PCS Availability (Q and W" Functions)

In the event of a Turbine Trip, the SNPS-PRA states that the operator is instructed by procedure and trained to maintain feedwater or recover it immediately. This important feature was taken into account in BNL's reevaluations. The recovery of feedwater function (Q) is an important function in most transients. BNL has therefore followed the approach of past BWR-PRAs[4,5] and constructed a functional level event tree for the Q function and tried to use it in a consistent way for all the transient events. Table 5A-1 gives the description of the functional event tree used for Q and the basis for its quantification in the turbine trip case. The tree includes two phases:

a) The Short-Term Phase: Probability that FW will be available, beginning 30 minutes after initiation of the transient.

b) The Long-Term Phase: Probability that the PCS will be available for containment heat removal, 15 hours after accident initiation.

The BNL functional event trees result in probabilities for Q similar to those found in past BWR-PRAs. However, those PRAs, as well as the SNPS-PRA, assumed that the long-term PCS availability is independent of the unavailability of parts of this system at accident initiation. BNL does not consider this to be realistic.

One anomaly that may arise when considering the long-term PCS recovery to be independent of failure to recover the PCS in the short term is that overall recovery probabilities for the transient duration become unrealistically

high. For example, failure to recover turbine bypass valve in the short term, and then again in the long term, are related, and the probability of late recovery should decrease if early recovery already failed.

The above situation is shown in Table 5A.1, in which both the short- and long-term PCS recovery probabilities are shown on the same tree. The conditional probability of long-term recovery, given that short-term recovery has failed, is of the order of 10^{-2}, which is higher by a factor of 2 than that in the SNPS-PRA, where the short- and long-term phases are assumed to be completely independent. This factor becomes larger in cases of MSIV closure or loss of condenser transient. In fact, the SNPS-PRA assumes some dependence between the short- and long-range recovery in some other transients in a few cases.

The SNPS-PRA considered a dependence between ADS operation and MSIV recovery probability, increasing the non-recovery probability by a factor of two for cases in which depressurization by ADS has occurred. This was insufficiently explained, and it was applied non-uniformly for the transients, with factors ranging from 1.4 to 3.0. The BNL approach, used with its functional level event tree, was to apply a uniform MSIV recovery probability of 0.001 for all cases (but for the case of MSIV closure see Section 5A.3) based on the long period (15 hours) available for the MSIVs to recover. Unlike the SNPS-PRA, BNL did not model the MSIVs as a frontline system, but used them as a part of the PCS.

In summary, the BNL approach used the functional level event tree approach for Q and W" quantification to gain more consistency in their quantification.

Another change made by BNL is the assumption that, when FW injection (Q) is successful, the long-term containment heat removal function is not required because no decay heat would be deposited into the suppression pool throughout the transient.

5A.1.3 The Results of the BNL Revised Event Trees

The revised BNL event trees take all the above considerations into account. They are shown in Table 5A.2, along with additional explanation regarding their quantification.

The result of the BNL re-assessment is about twice the value in the SNPS-PRA, mainly because of the increase in initiator frequency. A small increase in BNL results is obtained from the sequences including two SORVs. Although the contribution of this increase to the total core damage frequency is small, it is much higher than estimated in the SNPS-PRA. The small increase in Class II, apart from the change in initiation frequency, results from the dependence between early and late recovery of the PCS included in the BNL re-assessment.

5A.1.4 The Special Case of Common Mode Miscalibration of Level Instrumentation

The SNPS-PRA considers a miscalibration of all water level transmitters to be an event having a probability of occurrence of $2x10^{-3}$. It does not state that procedures for staggered calibrations are available.

The fault trees of the SNPS-PRA include the miscalibration error of all water level sensors as input to HPCI, RCIC, ADS, LPCI, and LPCS. It is identified on all those fault trees by the same basic event, namely "HHU720DXI". The fault tree model assumes that if miscalibration occurs, no automatic initiation will occur in RCIC and ADS. However, on HPCI, LPCI, and LPCS fault trees, the modeling assumed automatic initiation by high drywell pressure, which is true only for LOCA or ATWS and is incorrect for all transients and manual shutdowns. Therefore, the commonality of miscalibration for high and low pressure injection under transient conditions was not recognized in the cut sets of those fault trees and was not accounted for in the SNPS-PRA transient functional event trees.

The fault trees have included an operator action for manually starting the ECCS subsystems if automatic initiation failed. These include the following:

a) HHU500DXI and HHU600DXI for operator failure to manually actuate HPCI or RCIC,

b) AHU199DXI for operator failure to manually initiate ADS,

c) DHU111DXI for operator failure to manually initiate LPCS,

d) LHU500DXI and LHU600DXI for operator failure to manually initiate LPCI.

Therefore, in the SNPS-PRA fault tree analysis there exists the following cut set (see also Section 4.3):

HHU720DXI * (HHU500DXI + HHU600DXI) * AHU199DXI,

which can lead to Class I core damage if feedwater injection becomes unavailable. This event "T_TQ" for the turbine trip transient is $T_TQ = 8 \times 0.082 = 0.66$ per year.

The core damage probability for turbine trip with miscalibration then becomes

$$T_TQUX = 2 \times 10^{-3} \times 0.1 \times 0.1 \times 0.66 = 1.32 \times 10^{-5},$$

which is double the value for T_TQUX calculated on the event tree of Table 5A.2 (Sheet 2). BNL considers this result conservative for the following reasons:

a) A common-mode miscalibration error rate for a large miscalibration (a miscalibration of level 2 and 1 by over 10 feet is needed to uncover the core without safety system actuation) should be lower than 2×10^{-3}. BNL judges that a value smaller by a factor of 10 would be more realistic if some calibration procedures emphasizing the effect of large calibration changes are used. The Handbook of Human Reliability[7] (NUREG-CR/1278) gives even lower values for similar cases (see Section 4.3).

b) In order for a large miscalibration to be unnoticed, the operators must ignore a white indicator light in the control room for HPCI high level trip transmitters N091 C and D.

c) In order for the operator to perceive that the core is well covered and significantly reduce core injection for a long time, an additional miscalibration must have occurred on the wide or narrow range level transmitters, in a direction that will display high water level in the reactor vessel.

The reviewers consider the situation posed by such a gross miscalibration to be of sufficient importance to warrant calibration procedures that require staggered calibration such that N091 A and C would be calibrated at different times than N091 B and D. Such a procedure may be sufficient to reduce the probability of this event to a fraction of the T_TQUX sequence modeled in the event tree diagram and appropriately represented by the value of $UX = 8.4 \times 10^{-6}$ used from the fault tree analysis.

After the review was completed, BNL was informed that a modification to the level instrumentation system is underway at Shoreham. This modification will potentially reduce the probability of miscalibration by the addition of four new level instruments for HPCI actuation.

Table 5A.1 Functional Level Event Tree Description for FW and PCS Recovery Probability (Turbine Trip) (Sheet 1 of 3)

Function	Probability	Description/Comment
Feedwater system remains on line:	0.1	SNPS probability that the feedwater system fails to rapidly respond to the transient resulting in a level 8 trip or MSIV closure.
Recovery of FW between Level 2 and Level 1:	0.7	The SNPS probability of FW recovery given that HPCI or RCIC does not start. 6-10 minutes are available to the operator before level 1 signal.
Turbine controls and bypass valves available:	0.011	Probability that the main turbine controls and bypass valves are failed or fail during the transient. A factor of 10 was applied since the initiating event involved the turbine.
MSIVs remain open:		Probability that the MSIVs fail to remain open during the transient.
	0.02	If FW system remains on line.
	0.20	If loss of FW system occurs. It may result in MSIV closure on low reactor level or pressure.
MSIVs reopened:	0.1	Probability that the operator fails to reopen the MSIVs within 30 minutes of transient initiation.
Recovery of FW and PCS:	0.01	Probability that the operator fails to recover the FW and PCS within 30 minutes given a failure or turbine trip. The low failure probability is assumed because it is a standard action that the operator is called on to perform normally, and is trained to do on simulators.

Table 5A.1 Functional Level Event Tree Description for FW and PCS Recovery Probability (Turbine Trip) (Sheet 1 of 3 Continued)

Function	Probability	Description/Comment
Turbine Controls and By-pass Valves Available - Long-Term:	0.05	Conditional recovery probability given failure to recover during the early phase. Total failure to recover probability should not exceed 5.5×10^{-4} for the transient duration.
	5.5×10^{-4}	Probability that the turbine controls and bypass valves are not available. System restoration is assumed. (0.5 x Estimated System unavailability).
MSIVs Reopened - Long-Term:	0.001	Probability that the operator fails to reopen the MSIVs during the time available following the initiation of the transient. Assumed to be 15 hours.
MSIVs Reopened - Long-Term:	0.01	Conditional recovery probability given failure to recover during the early phase. Total failure to recover probability during the transient assumed not to exceed 0.001.
Recovery of FW and PCS:	0.001	Probability that the operator fails to recover the FW and PCS during the time available. Assumed to be 15 hours.
	0.1	Conditional recovery probability given failure to recover during the early phase. Total failure to recover probability should not exceed 0.001 for the 15 hour duration.
FW and PCS Hardware:	0.004	Probability that the FW and PCS will not be availabe to provide water to the reactor and remove decay heat to the environment. Value based on the SNPS fault trees (see Section 3.3).

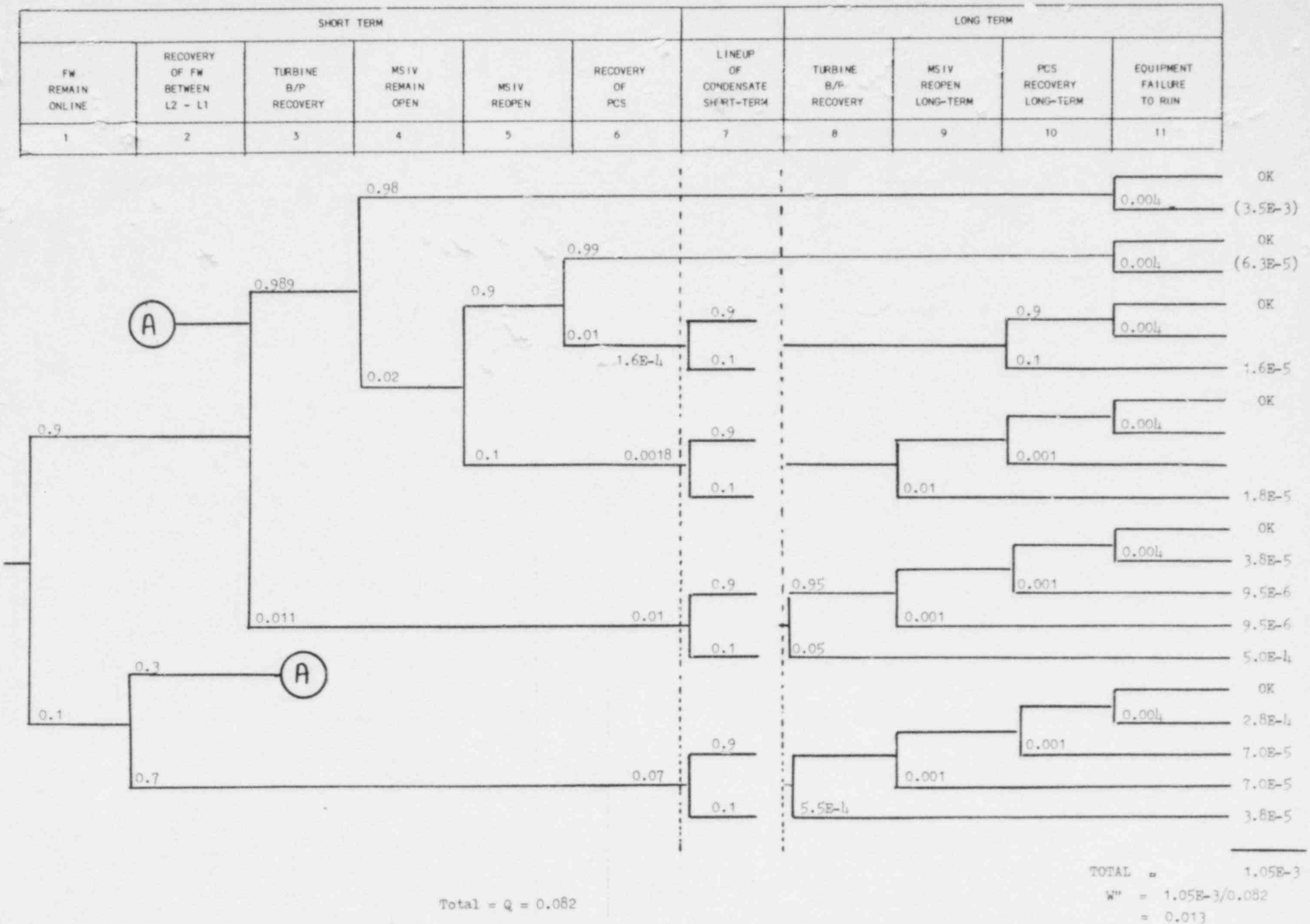
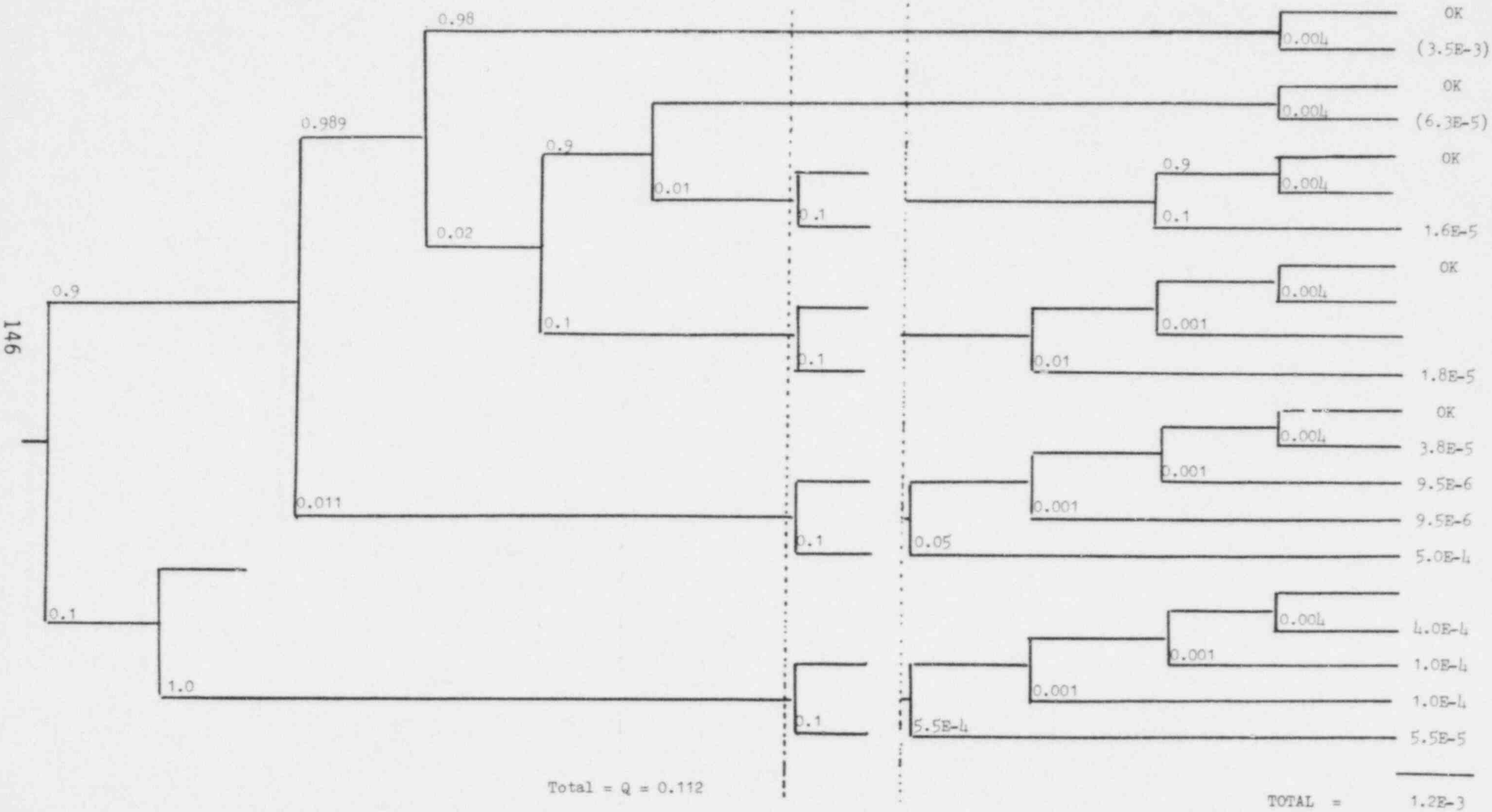


Table 5A.1 Functional Level Event Tree for the Probability of FW and PCS Unavailability Following a Turbine Trip. Short Term and Long Term Recovery Probabilities. Case of No SORVs.
(Sheet 2 of 3)

SHORT TERM							LONG TERM			
FW REMAIN ONLINE	RECOVERY OF FW BETWEEN L2 - L1	TURBINE B/P RECOVERY	MSIV REMAIN OPEN	MSIV REOPEN	RECOVERY OF PCS	LINEUP OF CONDENSATE SHORT-TERM	TURBINE B/P RECOVERY	MSIV REOPEN LONG-TERM	PCS RECOVERY LONG-TERM	EQUIPMENT FAILURE TO RUN
1	2	3	4	5	6	7	8	9	10	11



TOTAL = 1.2E-3

W' = 1.2E-3/0.112

= 0.011

Table 5A.1 Functional Level Event Tree for the Probability of FW and PCS Unavailability Following a Turbine Trip. Short Term and Long Term Recovery Probabilities. Case of 2 SORVs. (Sheet 3 of 3)

Table 5A.2 Event Tree Diagram for Sequences Following a Turbine Trip Initiator (Sheet 1 of 3)

T_T = 8.0:	The frequency of turbine trip transients per year is based on the discussion in Section 4.1.
C = 3.E-5:	This is the scram failure probability (both mechanical and electrical). It is taken from NUREG-0460[23].
M = 1.E-6:	This is the probability assumed for failure of 11 SRVs to open on high reactor pressure exceeding their set point. The failure leads to an unimportant contribution to LOCA frequency.
P = 2.E-3:	The probability that 2 SRVs will be stuck in the open position (stuck open relief valve = SORV). The probability of this failure mode is 3.75×10^{-3}/d. An average of three challenges per valve is assumed for turbine trip transient. The summation of 2 out of 7 combinations results in 2×10^{-3}.
Q = 0.082:	This probability of failure to recover FW is evaluated in Table 5A.1.
Q = 0.11:	This is the feedwater unavailability following turbine trip with 2 SORVs (see Table 5A.1 sheet 3 for derivation).
U' = 0.07:	The unavailability of RCIC based on the fault tree for the RCIC system (see Section 3.3).
U" = 0.1:	The unavailability of HPCI based on fault tree analysis (see Section 3.3 for discussion of the fault trees of the HPCI system).
U = 0.01:	The value of the unavailability of RCIC and HPCI, considering their commonalities (see Section 3.3).
X = 8.4E-4:	The ADS unavailability as derived from SNPS-PRA fault trees.
V'•V" = 6.2E-4:	The unavailability of LPCI and LPCS based on their combined fault tree analysis (Section 3.3).
V''' = 0.1:	The probability that the operator initiates or controls the condensate pump within half an hour or less, following loss of high and low pressure injections.
V''' = 0.02:	This is the probability of aligning the condensate system in the case of 2 SORVs when this system is needed four hours into the accident, after the pressure in the core decreased below high pressure injection reactor pressure requirement.

Table 5A.2 Event Tree Diagram for Sequences Following a Turbine Trip Initiator (Sheet 1 of 3 Continued)

W' = 4.4E-5:	The value of RHR with RCIC in steam condensing mode (see Section 3.3).
1.1E-4:	RCIC assumed unavailable. The value represents RHR reliability with assumed repair for 20 hours (~0.36). It is developed in Section 3.3 based on SNPS-PRA fault tree analysis.
W" = 4.E-3:	Unavailability of PCS if available during the turbine trip transient (see Section 3.3).
0.013:	Conditional unavailability of PCS given it failed to be recovered in the first half hour of the transient (see Table 5A.1).
0.011:	Same as above for 2 SORVs (see Table 5A.1).

INITIATOR	CRITICALITY	PRESSURE CONTROL		COOLANT INJECTION							CONTAINMENT HEAT REMOVAL				
TURBINE TRIP	SCRAM	SRVs OPEN	SRVs RECLOSED	FEEDWATER RECOVERED IMMEDIATELY	RCIC AVAILABLE	HPCI AVAILABLE	TIMELY ADS INITIATION	CS AVAILABLE	LPCI INJECTION	CONDENSATE PUMP INJECTION AVAILABLE	RHR OR RCIC IN STEAM CONDENSING PLUS SW	PCS	SEQUENCE DESIGNATOR	CALCULATED FREQUENCY (PER RX YR)	CALCULATED POSTULATED CORE DAMAGE OR TRANSFER
T_T	C	M	P	Q	U'	U"	X	V'	V"	V'''	W'	W"			

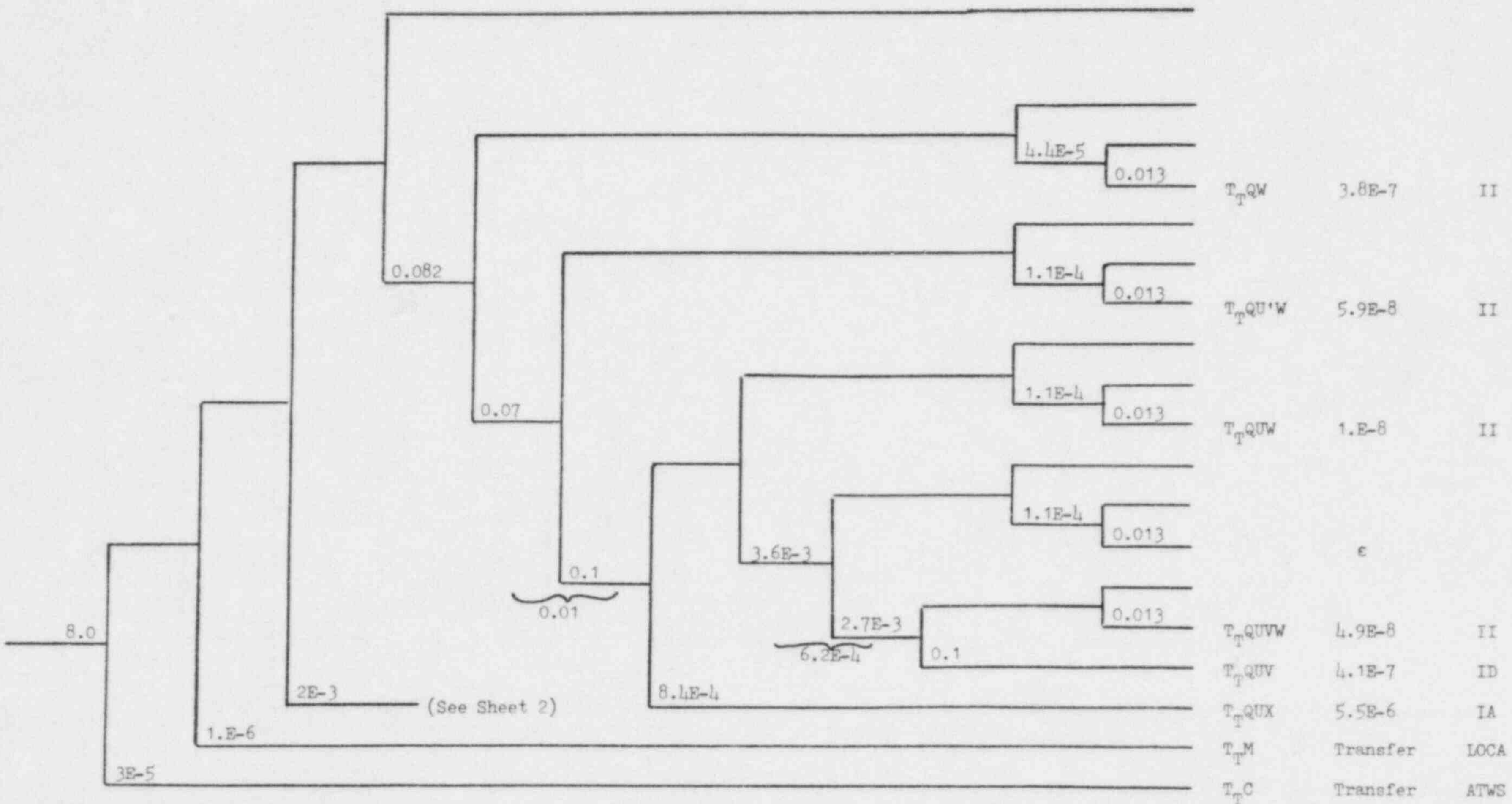
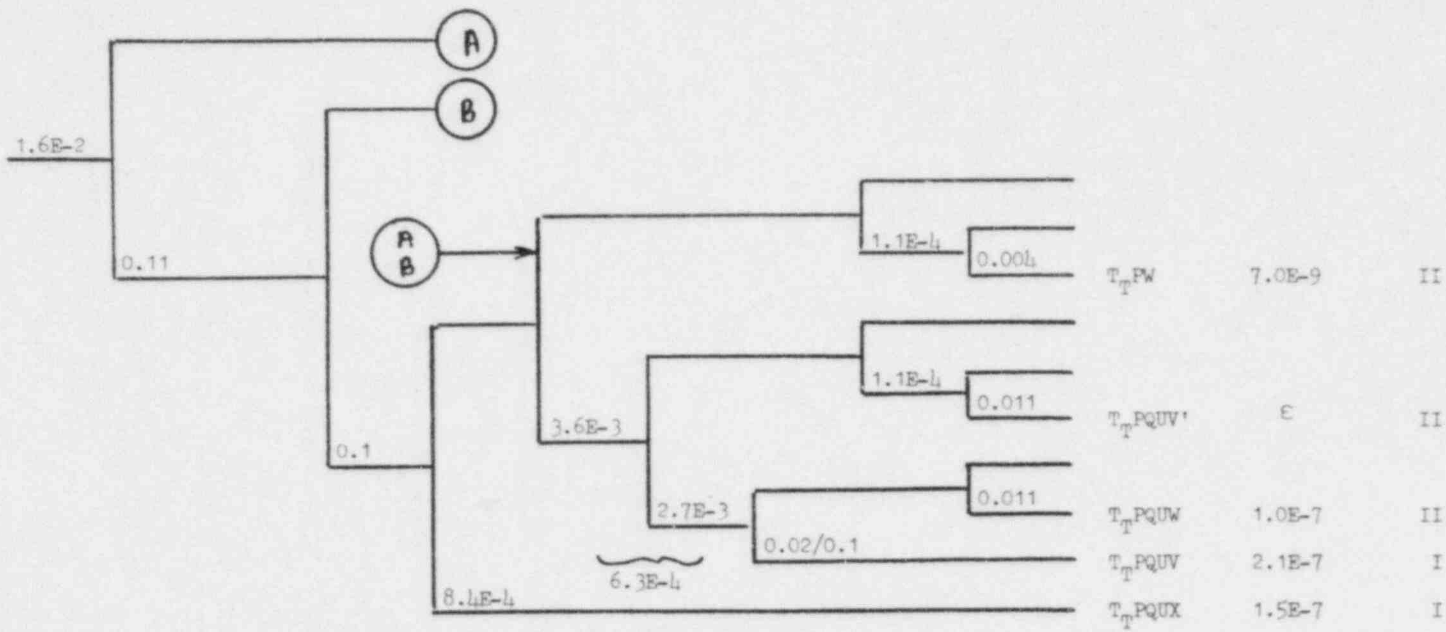


Table 5A.2 Event Tree Diagram for Sequences Following A Turbine Trip Initiator (Sheet 2 of 3)

INITIATOR	PRESSURE CONTROL	COOLANT INJECTION							CONTAINMENT HEAT REMOVAL				
TURBINE TRIP	SRVs CLOSED	FEEDWATER RECOVERED IMMEDI-ATELY	RCIC AVAILABLE	HPCI AVAILABLE	TIMELY REACTOR DEPRES-SURIZATION	CS AVAILABLE	LPCI AVAILABLE	CONDEN-SATE PUMP INJECTION AVAILABLE	RHR OR RCIC IN STEAM CONDENSING PLUS SW	PCS	SEQUENCE DESIGNATOR	CALCULATED FREQUENCY (PER RX YR)	CLASS OF POSTULATED CORE DAMAGE OR TRANSFER
T_T	P	Q	U'	U"	X	V'	V"	V'''	W'	W"			



Legend: 0.02/0.1 = 0.02 for A and B, 0.1 for $\overline{X}$.

Table 5A.2 Event Tree Diagram for Sequence Following a Turbine Trip Initiator. (Sheet 3 of 3)

5A.2 MANUAL SHUTDOWN

Manual shutdowns are gradual controlled reactivity insertion events. They have various reasons. The PRA lists contributors to manual shutdowns (Table 3.4.3 page 3-53). Most of them involve minimal challenges to the plant safety systems because often feedwater and PCS remain available with a very high probability. A few of them result in the challenge and initiation of safety systems.

An elaborate approach could be to treat separately the many possible combinations of manual shutdowns with frontline system unavailabilities, and sum their contributions. The SNPS-PRA chose a more efficient approach even though it may be conservative. It modeled three of the important cases of manual shutdowns with frontline systems unavailability concurrently on the same event tree:

(a) Manual shutdown because of condenser problems.

(b) Manual shutdown because HPCI and RCIC became unavailable.

(c) Manual shutdown because RHR (two loops) became unavailable.

For case (a) the frequency was taken from experience (Table 3.4.3 on the PRA) showing that 4% of manual shutdowns result from condenser problems. For case (b) the SNPS-PRA estimated that 1 of 100 shutdowns will be caused by HPCI and RCIC unavailability. BNL modified the frequency to 1/43 because in Appendix A.4 of the PRA, where maintenance is discussed, the PRA assumes that the same event can occur once in 10 years or 1 in 43 shutdowns. For case (c) the PRA estimated that both RHR systems may become unavailable with a probability less than $8x10^{-4}$ per year or $2x10^{-4}$ per manual shutdown. However, the initiation of this system may be delayed for 20 hours, and therefore a recovery factor of 0.36 accounting for repair was assumed in the BNL revised tree.

Modeling all three cases on the same event tree results in overestimation of the manual shutdown contribution to core damage frequency. The result would most probably be much larger than the sum of the contributions of the many possible sequences of manual shutdown combined with frontline systems unavailability. The revised event tree diagram of the BNL review is shown in Table 5A.4. It shows that, even using the conservative combinations of concurrent system failure and manual shutdowns, the contributions from these sequences are relatively small.

Note also that the SNPS-PRA determined the frequency of manual shutdown, based on experience, to be 4.3 per reactor year. This is on the high side of the range of values used in past PRAs and therefore reinforces the conclusion that the SNPS-PRA results for manual shutdown sequences represent their contribution quite conservatively.

Table 5A.3 Functional Level Event Tree for FW and PCS Recovery Probability (Manual Shutdown) (Sheet 1 of 2)

Feedwater System Remains Online:	0.04	According to Table 3.4-3 (page 3-53) of the SNPS-PRA, in 4% of BWR manual shutdowns the cause is condenser problems.
Recovery of FW before Level 1:	0.7	It is assumed that part of the condenser problems are in the condenser support subsystem and do not interfere with feedwater injection. The value of 0.7 is taken to be the same as in turbine trip.
Turbine Controls and Bypass Valves Available:	0.0011	Same as in the turbine trip case (Table 5A.1), but not multiplied by 10 because initiator event did not occur in this subsystem.
MSIVs Remain Open:	0.02	During manual shutdown operation there is a probability of MSIV closing. The same probability as in the case of turbine trip was used.
MSIV Reopened:	0.1	Same as in the turbine trip case.
Recovery of FW and PCS (short term = 30 minutes):	0.01	Same as in the turbine trip case.
Lineup of Condensate Pumps:	0.1 or 0.2	Probability of operator success to manually control the condensate pumps within less than 30 minutes as well as validating connection of CST to hotwell. This is given as 0.1. However, because a condenser problem is the cause of the shutdown, a 10% probability that the hotwell is involved was added when the condenser has failed.
Turbine Controls and Bypass Valves Available Long-Term:	5.5×10^{-4} or 0.5	The unavailability is assumed to be 0.0011 with 0.5 probability of repair. However, if this system failed in the short term, then it has the 0.5 probability of being repaired in the next 15 hours.
MSIVs Reopened Long-Term:	0.001	This is the probability that the operator fails to reopen MSIV in 15 hours. Taken from SNPS-PRA event "Z" in the case of the manual shutdown tree.
	0.01	If MSIV recovery failed in the short term, it is assumed that the overall failure to recover probability remains 0.001. Therefore a conditional probability is given.

Table 5A.3 Functional Level Event Tree for FW and PCS Recovery Probability (Manual Shutdown) (Sheet 1 of 2 Continued)

Recovery of Condenser Hardware in Long-Term:	0.036 or 0.001	This is based on the SNPS-PRA consideration that 10% of the condenser problems would be hardware malfunctions which have a mean time to repair of 19 hours. Therefore a recovery factor of 0.36 is used for 10% of the hardware. The rest is recovered with a probability of 0.001.
Recovery of FW and PCS Long-Term (before 15 hours):	0.001	The probability that the operator fails to recover the systems in 15 hours during the accident.
	0.1	Conditional probability given recovery has failed during the short term.
FW and PCS Equipment Available:	0.004	This is the hardware availability of the PCS and includes circulating pumps, condensate, air ejector or mechanical vacuum pump, MSIV, instrumentation and control, etc. (see Section 3.3 for PCS unavailability discussion).

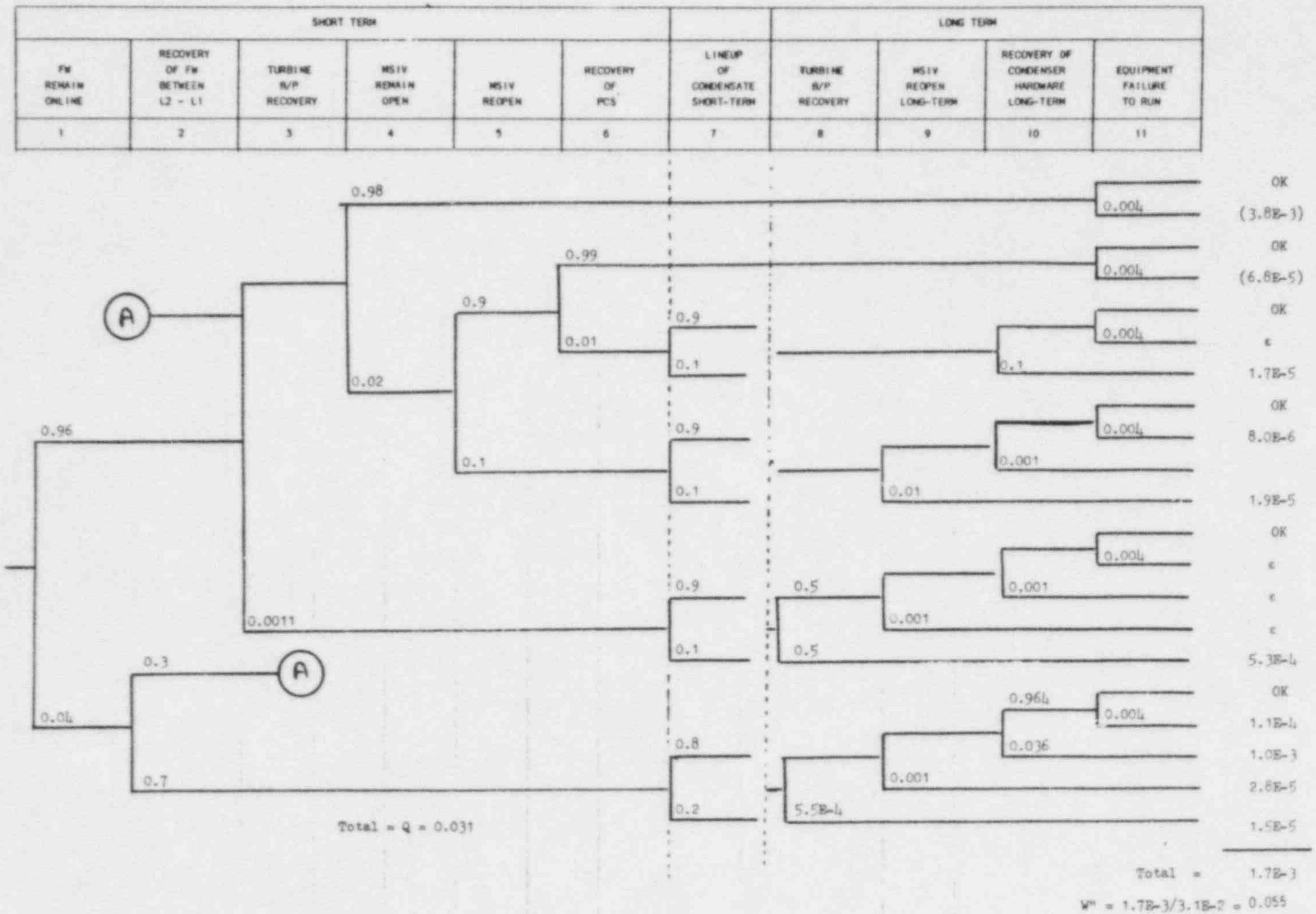


Table 5A.3 Functional Level Event Tree for the Probability of FW and PCS Unavailability Following a Manual Shutdown. Short Term and Long Term Recovery Probabilities. (Sheet 2 of 2)

Table 5A.4 Event Tree Diagram for Sequences Following a Manual Shutdown (Sheet 1 of 2)

M_S = 4.3:	The SNPS-PRA assessed the frequency of manual shutdown based operating experience, and obtained this frequency. It is apparently conservative. BNL used the same value.
P = 2.E-5:	The probability of challenging the SRVs is small for a manual shutdown. If challenged, they would require less valves to lift than in the case of a turbine trip.
Q = 0.03:	This value was developed in Table 5A.3. It represents a case of a condenser problem which required manual shutdown.
U = 0.015:	The normal value for the U function is 0.01. Here it is assumed by BNL, following the rationale of the PRA, that one time in ten years the HPCI and RCIC both will be unavailable and the plant will be manually shut down. This means once in 43 shutdowns or an addition of 1/185 to the U function given 4.3 shutdowns per year.
X = 8.4E-4:	Same as in turbine trip event tree.
V = 6.2E-4:	Same as in turbine trip event tree.
V''' = 0.15:	Because the manual shutdown is assumed to be caused by troubles in the condenser system, it is assumed that in 5% of the cases the problems will be in the hotwell which affects the condensate system.
W' = 7.2E-5:	This value for W' is obtained when a special case of manual shutdown is assumed, in which both RHR loops are unavailable and the plant is manually shut down. The SNPS-PRA estimates the frequency of such an event to be 2×10^{-4} per manual shutdown. To that a recovery factor of 0.36 is applied for 20 hours repair time.
W' = 4.4E-5:	The unavailability of RHR with RCIC steam condensing.
or 1.1E-4:	The unavailability of RHR without RCIC steam condensing.
W" = 0.055:	Developed by BNL in Table 5A.3. It is the conditional probability of having PCS available given manual shutdown was due to condenser problems.
W" = 0.004:	PCS unavailability.

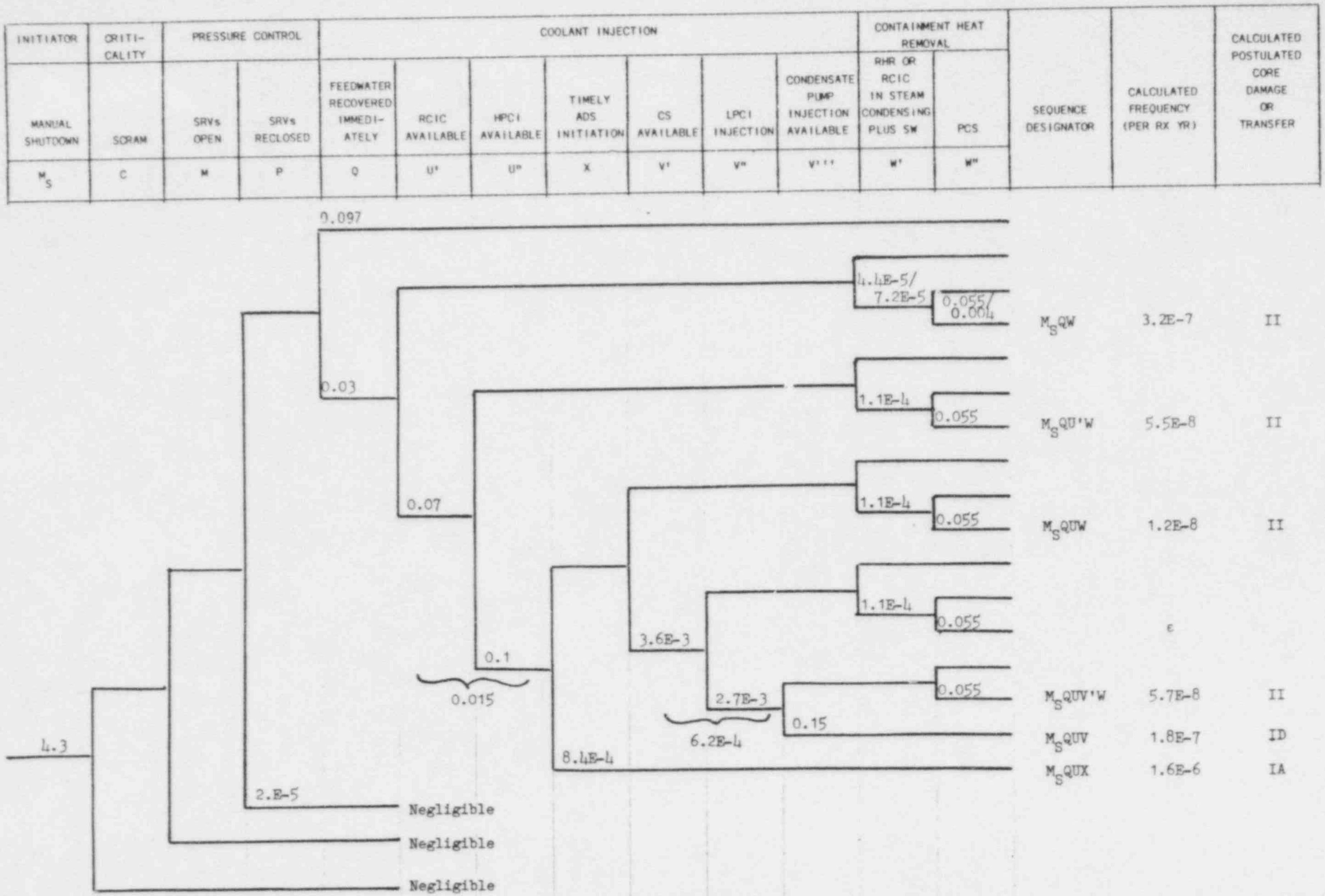


Table 5A.4 Event Tree Diagram for Sequences Following a Manual Shutdown
(Sheet 2 of 2)

5A.3 MSIV CLOSURE TRANSIENT

5A.3.1 Background

The SNPS-PRA MSIV closure transient event trees are reviewed here. Considered as MSIV closure transients are only those events in which the MSIV closure was the initiating event. Cases in which MSIV failed during the transient are dealt with in each respective transient.

The frequency of the initiator, discussed in Section 4.1, is 0.57 per year in the BNL review. To that are added the LOOP cases which are recovered early. The major contribution is 0.15x0.63, where 0.63 is the LOOP recovery probability within half an hour (see Table 4.7). Thus, the total frequency of MSIV closure transients is assumed to be 0.67. (For Class II sequences there is a slight "double counting" because it is also developed in the LOOP event tree).

Some MSIV closure events can be recovered immediately or within half an hour. The recovery probability in the BNL reassessment is evaluated by means of a functional level event tree, as shown in Table 5A.5. It is based on the same functional level event tree structure as in the turbine trip case shown in Table 5A.1. The quantification of that event tree uses the same recovery probabilities shown in the Turbine Trip case. The 0.7 for the early recovery probability of the MSIV is based on the SNPS response to a BNL question[6], and is stated to reflect BWR experience.

The results of this functional event tree are the following:

(a) For MSIV closure without SORV: Q = 0.45: W" = 0.03

(b) For MSIV closure with 2 SORVS: Q = 0.92; W" = 0.018.

The values for item (b) are calculated by using the same functional event tree, with 0.9 instead of 0.3 for failure to recover FW before hitting Level 1, which isolates the MSIVs. The two SORVs case is not further developed because of its small contribution. The case with no SORVs is shown in Table 5A.6 sheet 2 and the values are explained in sheet 1.

5A.3.2 The Results of the BNL Revised Event Tree

The revised event tree shown in Table 5A.6 takes all the above considerations into account. The results of the BNL reassessment are higher by a factor of 3. A factor of 2 is due to the revised initiator frequency and the other 50% to the increase in Q function developed in Table 5A.5.

As in the case of turbine trip, a common miscalibration of all four level transmitters will result in an increase by a factor of 2.5 above the reported BNL results if credit for staggered miscalibration procedures is not given. Otherwise, it would constitute only a small fraction of the 8.4×10^{-6} considered for the "UX" function in Table 5A.6. This miscalibration event was discussed in Sections 5A.1.4 and 4.3.

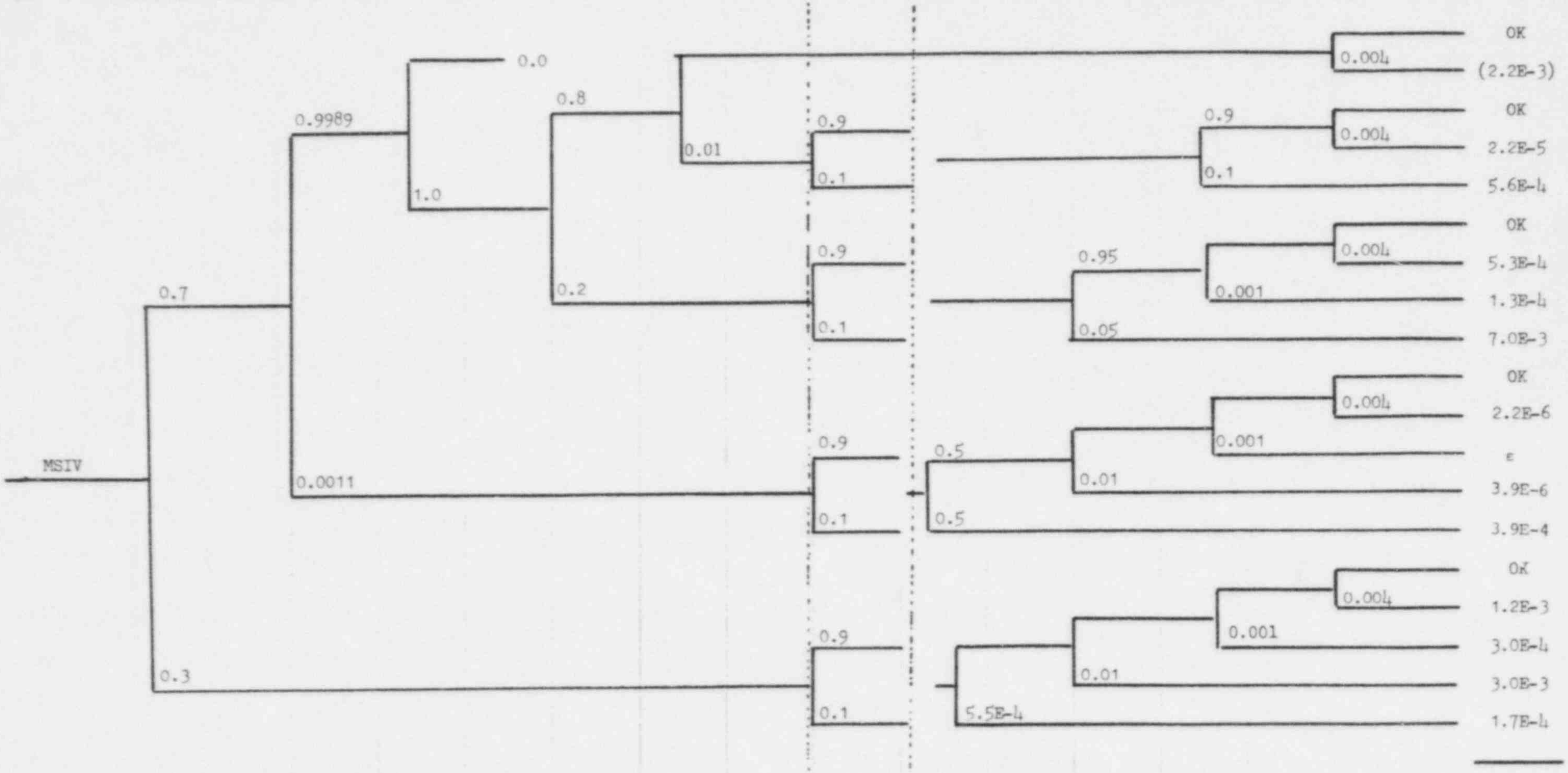
Table 5A.5 Functional Level Event Tree Description for FW and PCS Recovery Probability (MSIV Closure) (Sheet 1 of 2)

Recovery of FW before Level 1:	0.3	The SNPS-PRA event tree for the MSIV initiator uses this value. The basis is given on Page 3-72, and as stated in response #9 to BNL questions[6], comes from operating experience with BWRs.
	0.9	With two SORVs there is a higher rate of level decrease and a shorter time period to recover FW if HPCI and RCIC fail (SNPS-PRA).
Turbine Controls and Bypass Valves Available:	0.0011	Same as Table 5A.1, but not multiplied by 10, because initiator event did not occur in this subsystem.
MSIVs Remain Open:	1.0	Probability that the MSIVs fail to remain open during the transient. Here it fails and initiates the transient.
MSIVs Reopened Short-Term:	0.2	Probability that the operator fails to reopen the MSIVs within 30 minutes. A higher failure probability to recover is assumed (factor of 10) because transient originated in this equipment.
Recover of FW and PCS :	0.01	Same as in Turbine Trip - Table 5A.1.
Lineup of Condensate Pumps:		Same as in Turbine Trip - Table 5A.1.
Turbine Controls and Bypass Valves Available:	5×10^{-4} or 0.5	See comments to Table 5A.1. The probability of recovery in 15 hours is about 0.5, given system is in failed state.
MSIVs Reopened Long-Term:	0.01	SNPS-PRA event tree for MSIV assumes 0.05 for long-term recovery of MSIV. This is because the initiating event originated from this equipment. In the BNL review, a factor of 10 was applied to the MSIV recovery probability used for a Turbine Trip initiator (which is 0.001--see Table 5A.1), to account for this potential dependency.
	0.05	Conditional probability used for long-term recovery of MSIV given failure to recover in the first 1/2 hr.

Table 5A.5 Functional Level Event Tree Description for FW and PCS Recovery Probability (MSIV Closure) (Sheet 1 of 2 Continued)

Recovery of FW and PCS:	0.001	Same as in Table 5A.1.
FW and PCS Equipment Available:	0.004	This has been assumed for the long-term phase. Use is made of the same value as assumed in Turbine Trip, because the initiator did not originate from the PCS.

SHORT TERM							LONG TERM			
FW REMAIN ONLINE	RECOVERY OF FW BETWEEN L2 - L1	TURBINE B/P RECOVERY	MSIV REMAIN OPEN	MSIV REOPEN	RECOVERY OF PCS	LINEUP OF CONDENSATE SHORT-TERM	TURBINE B/P RECOVERY	MSIV REOPEN LONG-TERM	PCS RECOVERY LONG-TERM	EQUIPMENT FAILURE TO RUN
1	2	3	4	5	6	7	8	9	10	11



Total = Q = 0.45

Total = 1.35E-2

W" = 1.35E-2/0.45

= 0.03

Table 5A.5 Functional Level Event Tree for the Probability of FW and PCS Unavailability Following a MSIV Closure. Short Term and Long Term Recovery Probabilities.
Case of No SORVs.
(Sheet 2 of 2)

Table 5A.6 Event Tree Diagram for Sequences Following a MSIV Closure Initiator (Sheet 1 of 2)

T_M = 0.67: Frequency of MSIV closure, which includes the frequency from operating experience as derived in Section 4.1 (Table 4.2) combined with the contribution from LOOP events in which off-site power was recovered early.

M = 2.E-3: Failure of SRV to reclose. The probability is assumed to be the same as in the turbine trip case. The contribution of this sequence is relatively small and is not further developed. It can be evaluated if event tree similar to Table 5A.2 (sheet 3/3) is developed.

Q = 0.45: Developed on Table 5A.5.

U = 0.01: Same as in the turbine trip event tree

X = 8.4E-4: Same as in the turbine trip event tree

V = 6.3E-5: Same as in the turbine trip event tree

W' = 6.4E-5: Same as in the turbine trip event tree

= 1.1E-4: Same as in the turbine trip event tree

W" = 0.004: Same as in the turbine trip event tree

= 0.03 : Conditional probability of PCS recovery given it failed in the short term. Developed in Table 5A.5.

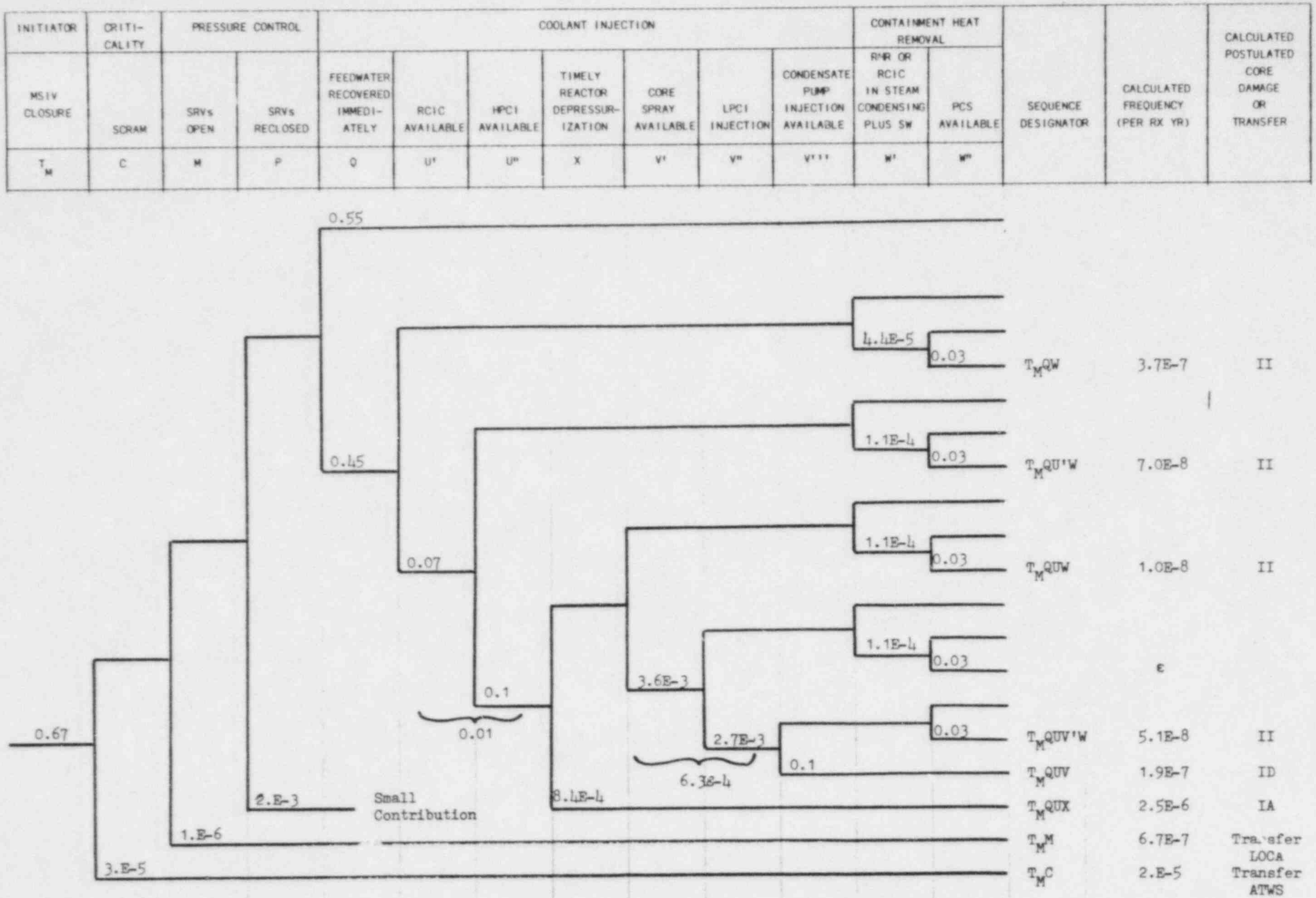


Table 5A.6 Event Tree Diagram for Sequences Following a MSIV Closure Initiator. (Sheet 2 of 2)

5A.4 LOSS OF FEEDWATER TRANSIENT

5A.4.1 Background

This section reviews the loss of FW transient event trees of the SNPS-PRA. Only those loss of feedwater events that initiated the transient are considered. Not considered are cases in which this event occurred subsequent to another initiator such as a turbine trip with Level 8 FW trip. Cases in which FW is lost during the transient are dealt within each respective transient.

The frequency of the initiator, discussed in Section 4.1, is 0.13, which is lower than the SNPS-PRA value--estimated to be 0.18.

Most of the loss of FW events can be recovered in a short time, as BWR experience indicates. This is given credit in the SNPS-PRA event tree and by BNL. The recovery probability is evaluated in the BNL review by means of a functional level event tree, shown in Table 5A.7. This tree is consistent with the other functional level event trees used for MSIV closure or turbine trip. The results of this evaluation are as follows:

(a) For loss of FW without SORV: $Q = 0.12$, $V''' = 0.25$, $W'' = 0.035$.

(b) For loss of FW with 2 SORVs: $Q = 0.51$, $V''' = 0.30$, $W'' = 0.03$.

Values for Item (b) are calculated by using the same functional level event trees, with 0.5 instead of 0.1 for failure to recover FW before hitting Level 1. The case of two SORVs was not further developed here because of its small contribution. The case with no SORV is shown in Table 5A.8 sheet 2, and the values used are explained in sheet 1.

5A.4.2 The Results of the BNL Revised Event Tree

The revised event tree shown in Table 5A.8 takes the above background considerations into account. Most of the values are, however, similar to those in the turbine trip case. The results of the re-evaluation are similar to those of the SNPS-PRA. This is because, based on the functional level event tree, a similar non-recovery probability to that of the SNPS-PRA is predicted by BNL (0.12 compared to 0.14). The similar Class II results are due to the compensating effects in the BNL assumptions: (1) the dependency between W" and Q, and (2) the reduction in BNL frequency for the initiating event, and also by the assumption that after recovery of FW, there is no need for containment heat removal because all decay heat is transferred to the condenser.

This transient as a whole is a small contributor to the SNPS-PRA core damage frequency.

Table 5A.7 Functional Level Event Tree for the Probability of FW and PCS Unavailability Following Loss of FW Transient: Short-Term and Long-Term Recovery Probabilities (Sheet 1 of 2)

Recovery of FW before Level 1:	0.1	BWR operating experience indicates that for about 90% of the loss of FW events, the FW can be recovered.
	0.5	With two SORVs, there is a higher rate of level decrease and a shorter time period to recover FW if HPCI and RCIC fail (SNPS-PRA).
Turbine Control and Bypass Valves Available:	0.0011	Same as Table 5A.3.
MSIVs Remain Open:	0.2	Same as Table 5A.1.
MSIVs Reopened:	0.1	Same as Table 5A.1.
Recovery of FW and PCS:	0.01 or 0.0	Same as Table 5A.1 if MSIV closes. If the FW is recovered and no subsequent failure occurs, no further recovery of PCS is required as in the case of Turbine Trip.
Lineup of Condensate:	0.1 or 0.3	Dominated by operator error to align the CST to condenser hotwell. However, when the recovery of PCS or FW fails, it is assumed that hardware failures in the PCS exist and a conditional probability of 0.3 is used to account for a 1/3 probability that this is in the condensate system. These assumptions result in an increase in condensate unavailability by a factor of about 2 relative to the case of MSIV closure. The SNPS-PRA also used a factor of 2.
Turbine Control Long-Term:	$5x10^{-4}$ or 0.5	Same as Table 5A.3.
MSIVs Reopened:	0.001	Same as Table 5A.1.

Table 5A.7 Functional Level Event Tree for the Probability of FW and PCS Unavailability Following Loss of FW Transient: Short-Term and Long-Term Recovery Probabilities (Sheet 1 of 2 Continued)

Recovery of FW and PCS:	0.001 or 0.3 0.1	Same as in Table 5A.1. However, a factor of 10 is applied to increase the probability of failure to recover, if FW was not recovered in the early phase, because it is considered to result from the original initiating event. In addition, dependences were taken into account and conditional probabilities were calculated so that the 0.001 recovery probability will be preserved in all sequences.
FW and PCS:	0.016	Because the transient originated in the PCS system, a factor of 4 was applied to the PCS equipment unavailability used for the Turbine Trip transient.

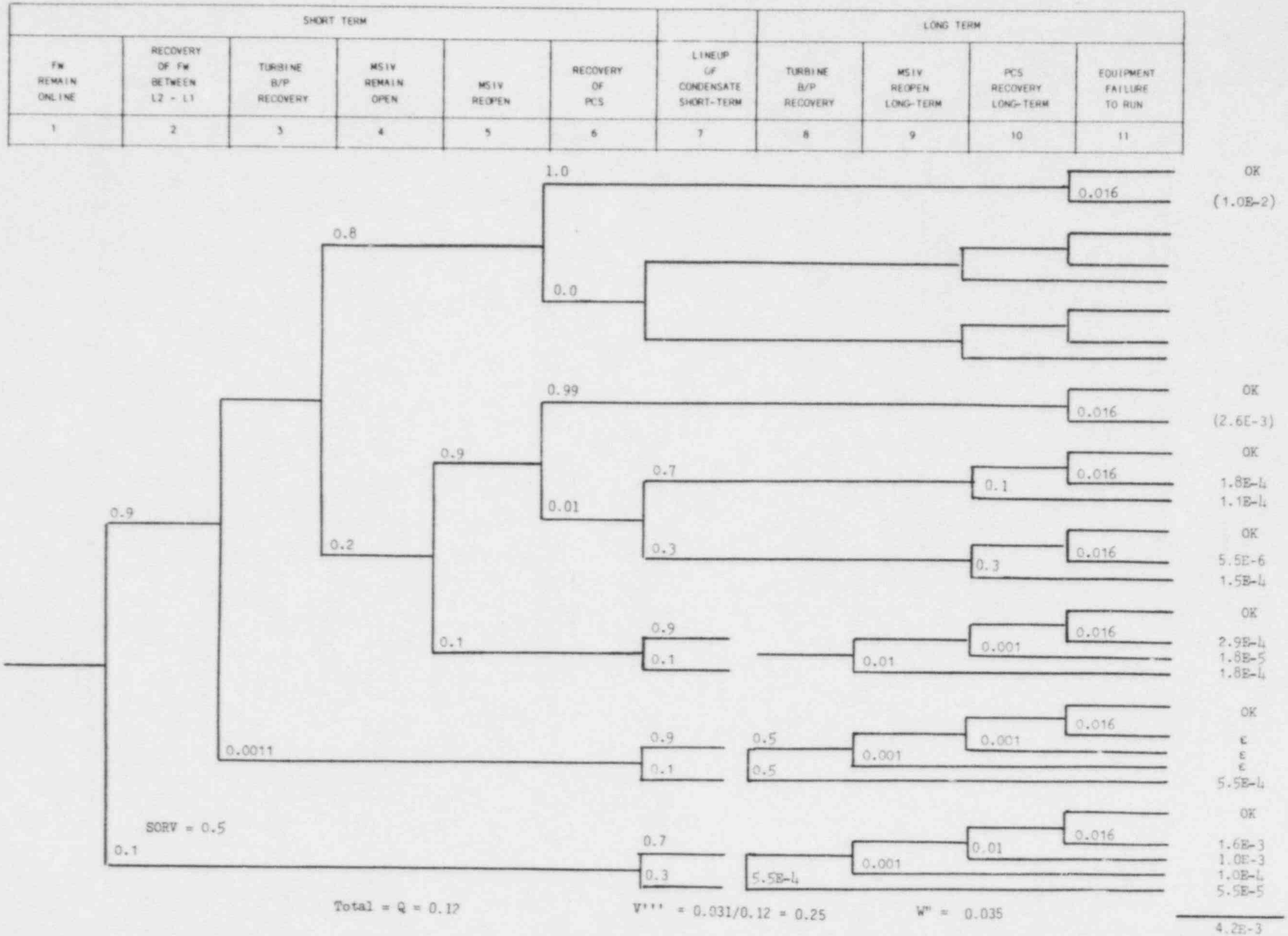


Table 5A.7 Functional Level Event Tree for the Probability of FW and PCS Unavailability Following a Loss of FW Transient. Short and Long Term Recovery Probabilities. (Sheet 2 of 2)

Table 5A.8 Event Tree Diagram for Sequences Following a Loss of Feedwater Initiator (Sheet 1 of 2)

T_F = 0.13:	Frequency of loss of FW derived from operating experience as explained in Section 4.1 of this report. This is 30% smaller than the SNPS-PRA frequency.
C = 3.E-5:	Same as in Table 5A.2.
M = 1.E-6:	Same as in Table 5A.2.
P = 2.E-3:	Same as in Table 5A.2.
Q = 0.12:	Developed in Table 5A.7.
U = 0.01:	Same as in Table 5A.2.
X = 8.4E-4:	Same as in Table 5A.2.
V'•V" = 6.2E-4:	Same as in Table 5A.2.
V''' = 0.25:	Developed in Table 5A.7. It is assumed that part of the initiator frequency is coming from the loss of condensate system.
W' = 4.4E-5 or 1.1E-4:	Same as in Table 5A.2.
W" = 0.035:	Developed in Table 5A.7. A higher non-recovery probability for PCS in the long term is assumed for this initiator.

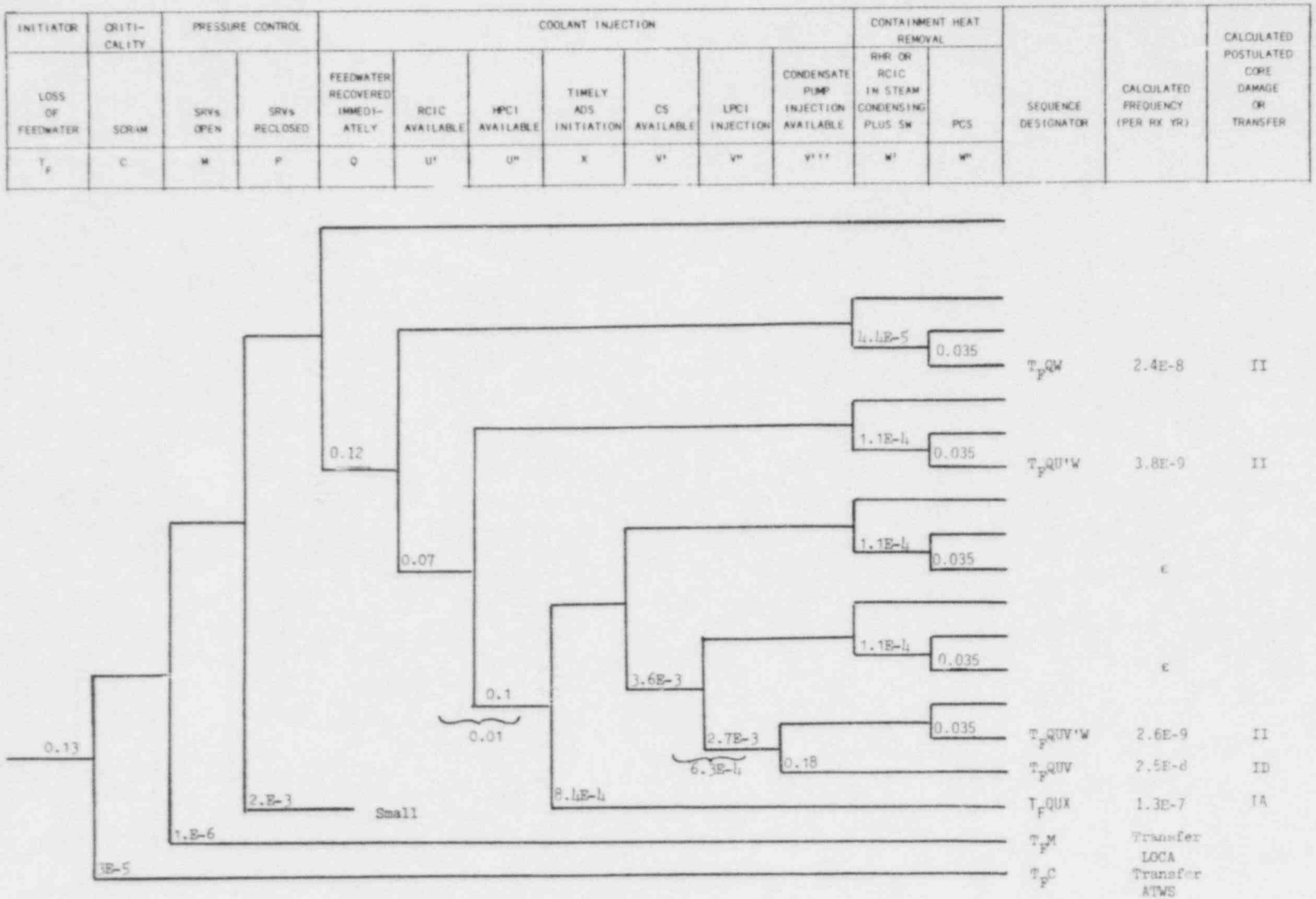


Table 5A.8 Event Tree Diagram for Sequences Following a Loss of Feedwater Initiator.
(Sheet 2 of 2)

5A.5 LOSS OF CONDENSER VACUUM TRANSIENT

5A.5.1 Background

This is an important initiator because it affects both the ability to provide coolant makeup and long-term containment heat removal. Upon loss of condenser, the turbine stop valve will close, the turbine bypass valves will be prevented from opening, a d reactor scram, feedwater pumps trip, and MSIVs closure will be initiated. The pressure buildup will be relieved through the SRVs to the suppression pool. Upon level 2 the HPCI and RCIC will start to maintain level and prevent level 1 MSIV closure and ADS initiation.

The feedwater is assumed to be not recoverable in this event, until vacuum in the condenser is reestablished. Credit, however, is given to the use of the condensate system for low pressure injection. In the case of loss of condenser, it is assumed that 5% of the failure of condenser will affect the hotwell water supply and will fail the condensate system.

Because the PCS is isolated, the suppression pool receives all the decay heat through the SRVs or high pressure inejction steam turbine exhaust. The RHR must be initiated within 20 hours, or the PCS reestablished before 15 hours. The probability of reestablishing condenser vacuum is assumed to be exponentially distributed with 19 hours mean time to repair. This gives, for a 15 hour repair time, a non-recovery probability of 0.45, which is used in Table 5A.9. This is higher than in the SNPS-PRA, where 23 hours were assumed. However, some calculations[8] appear to indicate that at 17 hours without heat removal the drywell pressure will reach ~60 psi, which can fail the SRVs. In addition, the PCS does not cool the suppression pool, but only diverts the decay heat to the condenser. This means that if PCS is initiated at 23 hours, the drywell may remain at conditions close to its failure conditions for several hours, with substantial probability of failure. BNL chose the 15 hours for PCS initiation as a success criterion for this containment heat removal mode. The SNPS-PRA in several other cases also uses 15 hours for PCS initiation rather than the 23 hours used in the case of loss of condenser.

The SNPS-PRA has assumed that only 25% of the cases of loss of condenser require long repair time because of hardware problems. The other 75% are assumed to be easily recoverable within a few hours. BNL used the same value, but did not review it. Table 5A.9 shows that an increase in this number would similarly increase the PCS unavailability for the long-term containment heat removal function, and may increase significantly the Class II contribution.

5A.5.2 The Results of the BNL Revised Event Trees

The revised event trees are given in Table 5A.10. The results of the reassessment are higher by a factor of 1.5 higher than those of the SNPS-PRA in both Class I and Class II. Most of the change is due to the 25% increase in initiating event frequency (see Section 4.1) and some is due to increased failure to recover probabilities given in the BNL review for PCS and condensate pumps. The sequences of loss of condenser are major contributors to Class I and II. They provide about 5% of Class I and 15% of Class II contributions to core damage probability.

Table 5A.9 Functional Level Event Tree for the Probability of FW and PCS Unavailability Following a Loss of Condenser Initiator (Sheet 1 of 2)

Feedwater Remains On Line:	0.0	Loss of condenser event results in feedwater trip.
Feedwater Recovered:	1.0	It is asumed that condenser vacuum is not recovered within one half hour and the feedwater remains in tripped condition.
Lineup of Condensate:	0.15	Loss of condenser does not prevent the condensate system from being realigned or from providing water to the reactor vessel. The probability of the operator failure in this task is assumed to be 0.1 because of the short time available and the stress conditions following the loss of high and low pressure injection. An additional 0.05 is put in because it is assumed that 5% of the events of loss of condenser will involved hotwell unavailablity.
PCS Hardware:	0.25	Following the SNPS-PRA assumption, it is assumed that the fraction of condenser related scrams that could lead to a long term hardware problem is 0.25.
PCS Recovery:	0.45	Failure to recover hardware problems based on MTTR = 19, and 15 hours available to recover the PCS. The SNPS-PRA requires opening of MSIV in 15 hours (page 3-99). MSIV cannot be reopened unless condenser vacuum can be restored.
	0.01	Failure to recover non-hardware problems because of operator errors. A factor of 10 was applied because the initiator originated from this system. Note that hardware reliability is included in the above values and therefore is not modeled separately as in the turbine trip tree.
MSIV Reopened Long-Term:	0.001	See Table 5A.1.

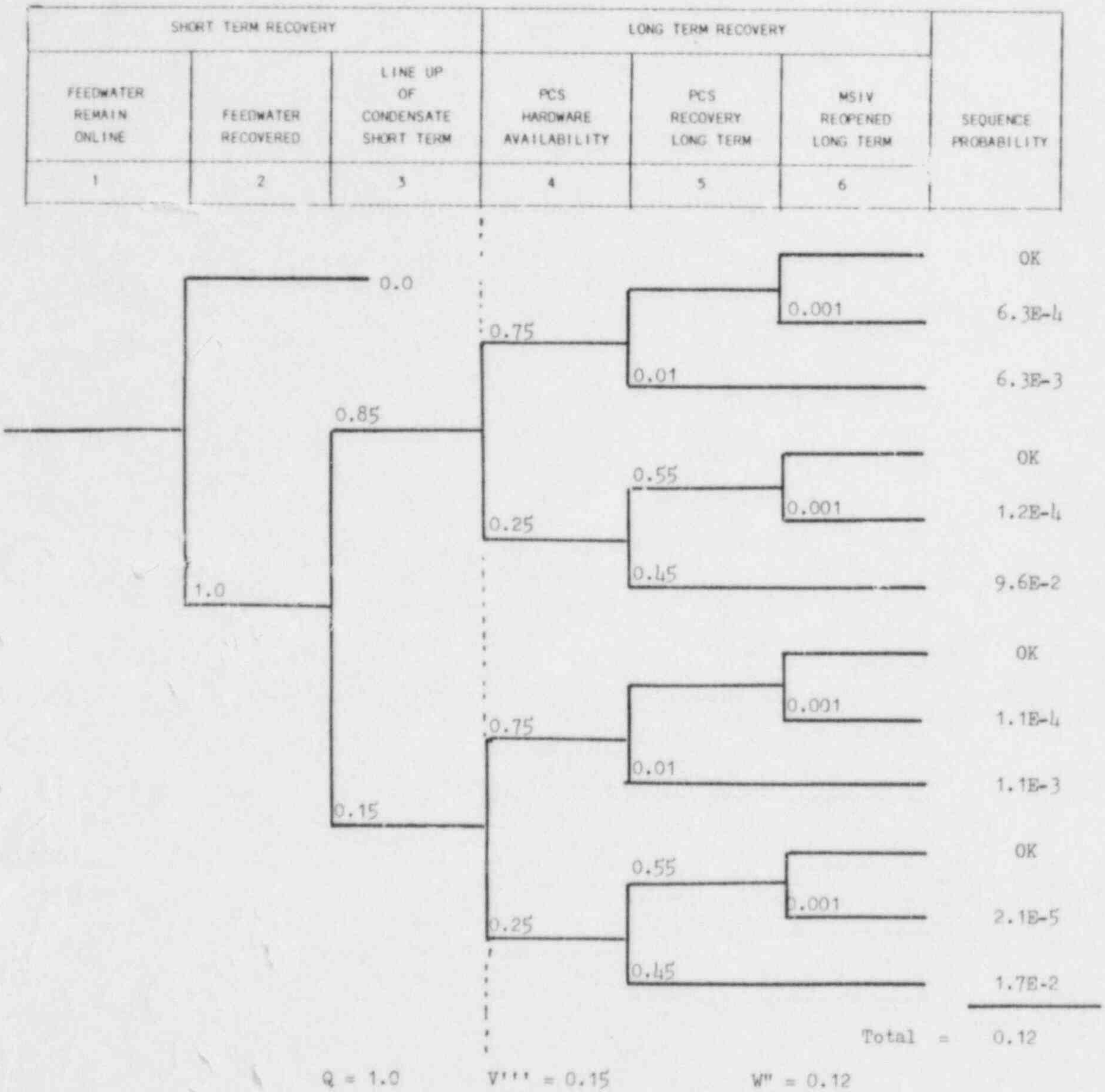


Table 5A.9 Functional Level Event Tree for the Probability of FW and PCS Unavailability Following a Loss of Condenser Initiator. (Sheet 2 of 2)

Table 5A.10 Event Tree Diagram for Sequences Following a Loss of Condenser Vacuum (Sheet 1 of 2)

T_c = 0.5:	Frequency of Loss of Condenser in the BNL re-evaluation as taken from Section 4.1 (Table 4.2). It is slightly higher than in the SNPS-PRA. GE experience apparently shows that recovery is possible in ~50% of the cases, but the SNPS-PRA did not provide the data and did not take credit.
C = 3.E-5:	Same as in turbine trip event tree.
M = 1.E-6:	Same as in turbine trip event tree.
P = 2.E-3:	Same as in turbine trip event tree. The contribution of this sequence was evaluated and the resulting calculated frequencies for two sequences are shown on the event tree. The contributions are small, but they are an order of magnitude higher than in the SNPS-PRA.
U = 0.01:	Same as in turbine trip event tree.
X = 8.4×10^{-4}:	Same as in turbine trip event tree.
V'·V'' = 6.3E-4:	Same as in turbine trip event tree.
V''' = 0.15:	The probability of failure to realign and control the condensate pumps is assumed to include two contributors: a) 0.1 for human error, as in Table 5A.8 or in the turbine trip case. b) 0.05 for a 5% possibility that the loss of condenser is the result of loss of inventory in the hotwell.
W' = 4.4E-5 or 1.1E-4:	Same as in turbine trip event tree.
W" = 0.12:	Developed in the functional level event tree of Table 5A.9.

INITIATOR	CRITICALITY	PRESSURE CONTROL		COOLANT INJECTION							CONTAINMENT HEAT REMOVAL		SEQUENCE DESIGNATOR	CALCULATED FREQUENCY (PER RX YR)	CALCULATED POSTULATED CORE DAMAGE OR TRANSFER
LOSS OF CONDENSER VACUUM	SCRAM	SRVs OPEN	SRVs RECLOSED	FEEDWATER RECOVERED IMMEDIATELY	RCIC AVAILABLE	HPCI AVAILABLE	TIMELY ADS INITIATION	CS AVAILABLE	LPCI INJECTION	CONDENSATE PUMP INJECTION AVAILABLE	RHR OR RCIC IN STEAM CONDENSING PLUS SW	PCS			
T_C	C	M	P	Q	U'	U"	X	V'	V"	V'''	W'	W"			

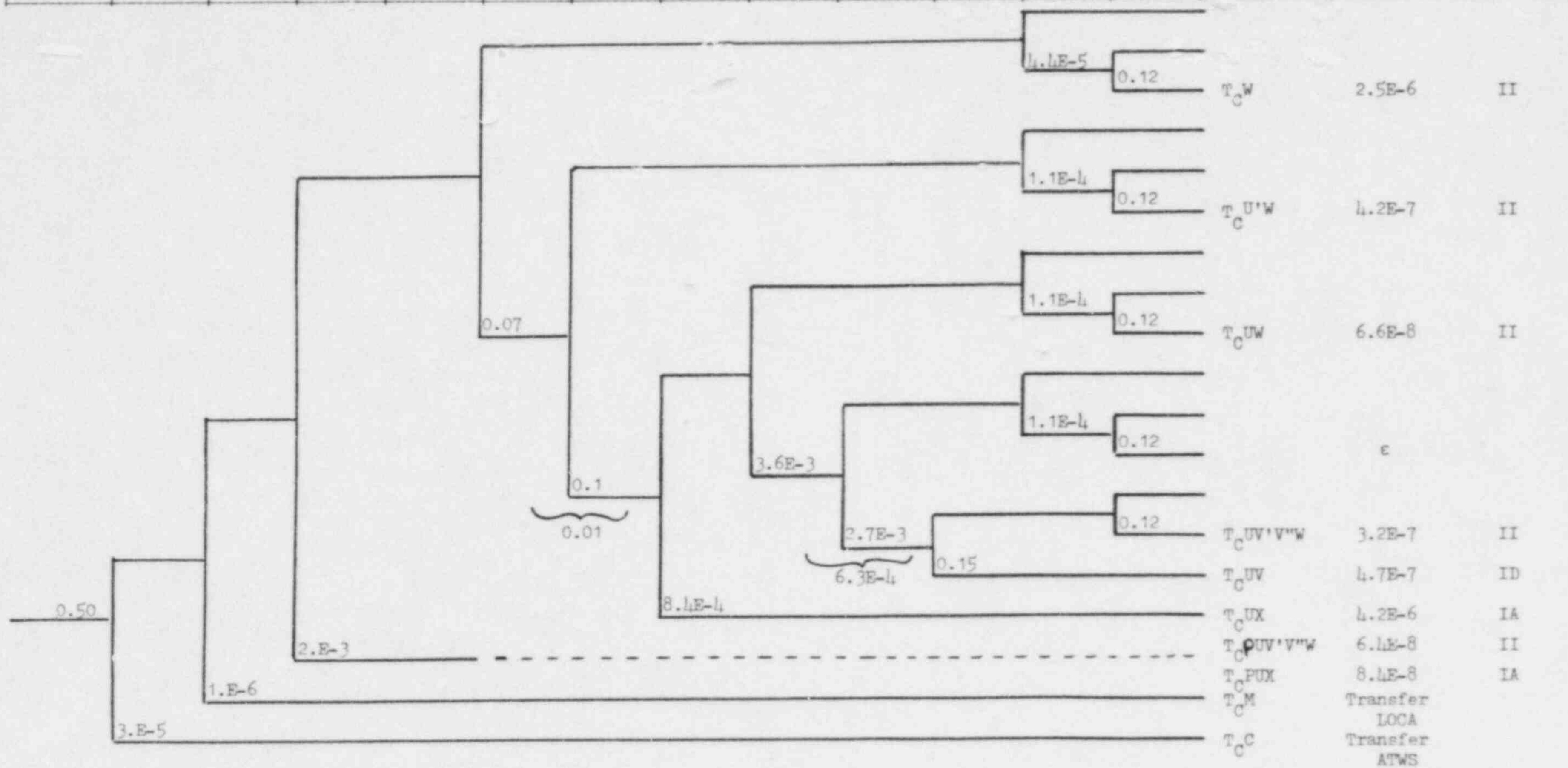


Table 5A.10 Event Tree Diagram for Sequences Following a Loss of Condenser Vacuum (Sheet 2 of 2)

5A.6 INADVERTENT OPEN RELIEF VALVE TRANSIENT (IORV)

5A.6.1 Background

The IORV event includes aspects of both a transient and a small LOCA. It starts like a small LOCA, but discharge is directed to the suppression pool, and ECCS initiation signals may come later than in the LOCA case.

In this case, suppression pool temperature will rise until the reactor is scrammed (first manually and later automatically). The HPCI and RCIC are receiving their lube oil cooling from the coolant flow, and, if suction is taken from the suppression pool, this function would be degraded to some extent. However, RCIC suction can remain on CST for almost the entire duration of the transient.

Another difference from other transients is the low availability of FW and PCS for this event. The SNPS-PRA states that BWR experience shows that in most IORV cases MSIVs closure occurs in the course of the transient. The PRA model for this is very conservative, more conservative than that in past BWR-PRAs, in contrast with the small LOCA event tree. The assumption in the PRA--that for a case of early reactor shutdown PCS will be available in 15 hours, and for a case of shutdown one hour later it would not be available for recovery a few hours later--is apparently too conservative, and was changed in the BNL reassessment to reflect some probability of recovery consistent with small LOCA.

5A.6.2 The Results of the BNL Revised Event Trees

The revised IORV event trees are given in Table 5A.12. Several changes were made by BNL, as explained above and in Table 5A.12, sheet 1. The BNL results are lower than the SNPS-PRA values because of the additional credit given for FW and PCS in the BNL revised quantification, which balanced the increase in the C' and U functions and the increase in the event frequency. The overall contribution is about 4×10^{-7} in the BNL review, which is about half that in the SNPS-PRA. The frequency of this event is calculated generically to be 0.25 per year (see Section 4.1) which apparently overestimates the expected frequency for Shoreham. This is because it does not consider the design change made at Shoreham using two stage Target Rock safety relief valves in order to reduce the frequency of IORV occurrence. As can be seen from the overall low contribution of this sequence to SNPS core damage frequency, the effect of such a frequency change would be relatively small in terms of core damage frequency.

Table 5A.11 Functional Level Event Tree for the Probability of FW and PCS Unavailability Following an IORV (For the Case of Timely Manual Control Rod Insertion) (Sheet 1 of 2)

		<u>Case 1: Timely Control Rod Insertion</u>
FW Remains on Line or Recovery of FW:		FW may remain on line or fail. If it remains on line, it will be lost later when MSIVs close. Operating experience data indicate that the MSIVs will virtually always close during an IORV event (see SNPS-PRA, page 3-134). The BNL functional event tree is based on this premise.
Turbine Controls and Bypass Valves Available:	0.0011	Same as in MSIV closure (Table 5A.5).
MSIV Remains Open:	1.0	See comment above.
MSIV Reopens Short-Term:	0.1	Same as in the turbine trip case (Table 5A.1).
Recovery of FW and PCS Short-Term:	0.01	Same as in the turbine trip case.
All Other Headings and Quantification:		Same as in the manual shutdown case for long term (see Table 5A.3).
		<u>Case 2: Scram is Delayed</u>
MSIV Reopened Short-Term:	1.0	Power operation with IORV is assumed to continue to a point that water level become low and MSIV closes. The SNPS-PRA assumes that MSIV would not be reopened in the short term under conditions of IORV with delayed scram.

Table 5A.11 Functional Level Event Tree for the Probability of FW and PCS Unavailability Following an IORV (For the Case of Timely Manual Control Rod Insertion) (Sheet 1 of 2 Continued)

MSIV Reopened Long-Term:	0.1	The SNPS-PRA conservatively assumed that the MSIV would not be reopened also during the long term. This seems too conservative and difficult to explain. In the cases of 2 SORVs and small LOCA, the MSIV is reopened in the long term. Successful scram is achieved in IORV after 30 minutes at most, on low level or high drywell pressure. From that time on, the transient would be similar to the 2 SORVs or small LOCA case. BNL assumed a 0.1 recovery probability to be consistent with small LOCA. This value is higher than in the 2 SORVs case where 0.001 is assumed; however, in the case of 2 SORVs the heat transferred to the suppression pool is small, and more time is available for recovery, so it is consistent to have higher recovery probability for that case.

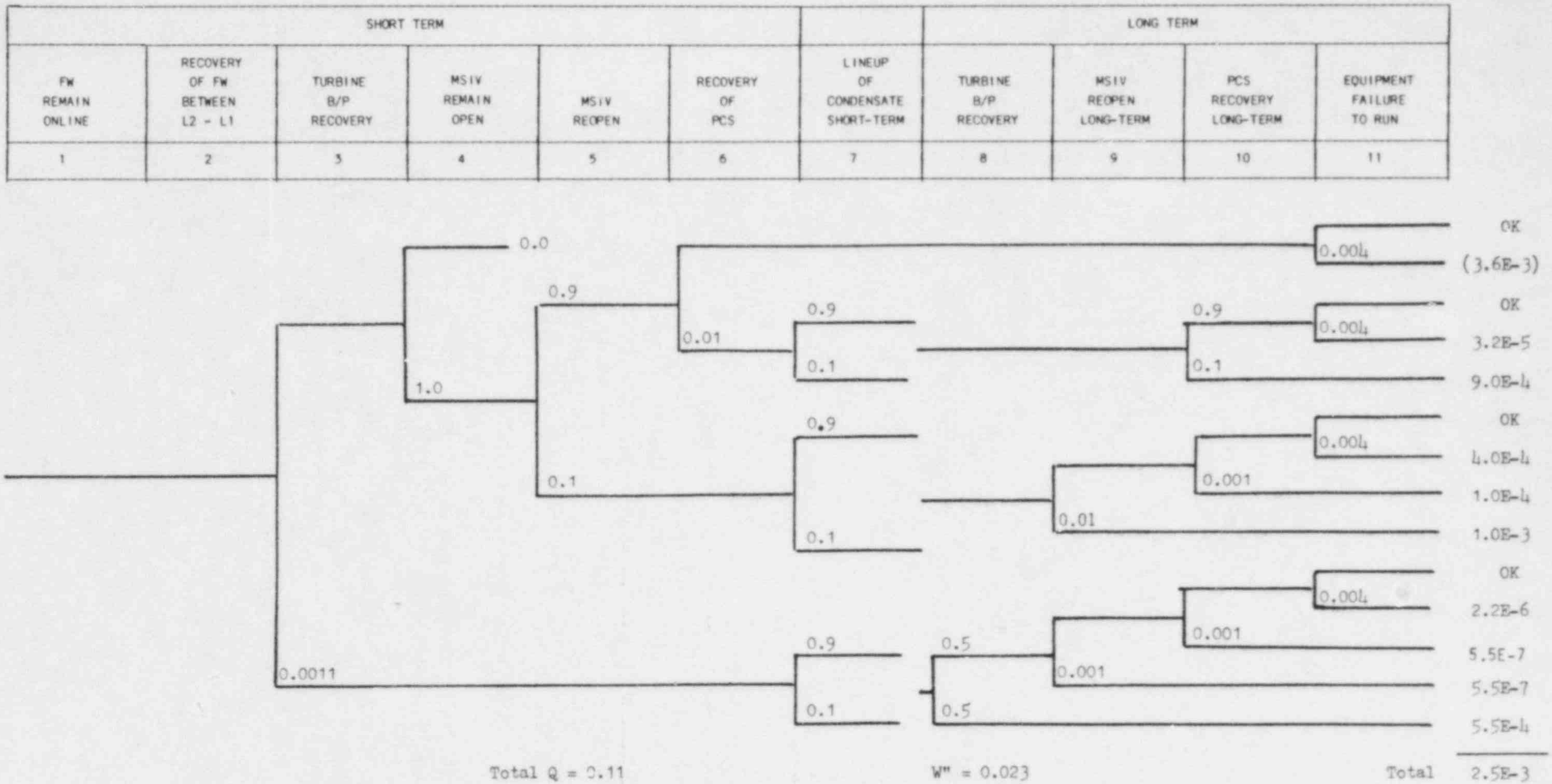


Table 5A.11 Functional Level Event Tree for the Probability of FW and PCS Unavailability Following an IORV. Short and Long Term Recovery Probabilities. (Sheet 2 of 2)

Table 5A.12 Event Tree Diagram for Sequence Following IORV (Sheet 1 of 3)

T_I = 0.25: Taken from Table 4.2. See discussion in Section 4.1. This is three times as high as the SNPS-PRA frequency. It does not consider the Shoreham design change to two stage Target Rock relief valves, which would apparently reduce this frequency of occurrence.

C' = 0.01: Timely manual control rod insertion is a key action in this transient. It is a manual operator action for which several indications and annunciators are available. However, this needs to be completed within a few minutes to prevent suppression pool heat up. BNL used a value taken from past PRAs, supported by functional fault trees, rather than the unsupported SNPS-PRA value. Furthermore, the BNL value is meant to represent a relatively fast operator response, for which feedwater recovery is possible (see next "Q").

Q = 0.11: This is developed in Table 5A.11. BNL gave credit to recovery of feedwater within 30 minutes if manual shutdown was completed early. In cases of early shutdown, BNL assumed that this transient would be similar to small LOCA or turbine trip with 2 SORVs.

Q = 1.0: For late shutdown, it is assumed, as in the SNPS-PRA quantification, that no recovery of MSIV will be successful in the early time frame. Operation at full power for some time before shutdown requires immediate injection after reactor tripped. Operating experience indicates that MSIV would almost always be closed, and therefore the grace time for recovery of FW would be significantly less than the 30 minutes assumed in the turbine trip transient.

U = 0.01: For early shutdown the normal value is used.

0.036: When shutdown is completed later, the suppression pool temperature is assumed to be above 140°F with some impact on HPCI availability (0.3 assumed). For RCIC, however, if the operator does not transfer RCIC to suppression pool suction (0.05 for operator error) then normal availability can be assumed as long as suction continues from CST. (RCIC = 0.07 + 0.05).

X, V, W': Same as in turbine trip event tree (see Table 5A.2).

W" = 0.023: This is explained in the functional level event tree of Table 5A.11 for the case of early shutdown.

Table 5A.12 Event Tree Diagram for Sequence Following IORV
(Sheet 1 of 3 Continued)

W" = 0.1: For late shutdown, BNL used a probability of 0.1 for PCS mainly because of failure to recover the MSIVs. This is made consistent with the small and medium LOCA event tree diagram (Appendix C). The SNPS-PRA assumption of no recovery in several hours is apparently too conservative if reactor shutdown is assumed to be completed in the first hour, before suppression pool temperature exceeds 200°F.

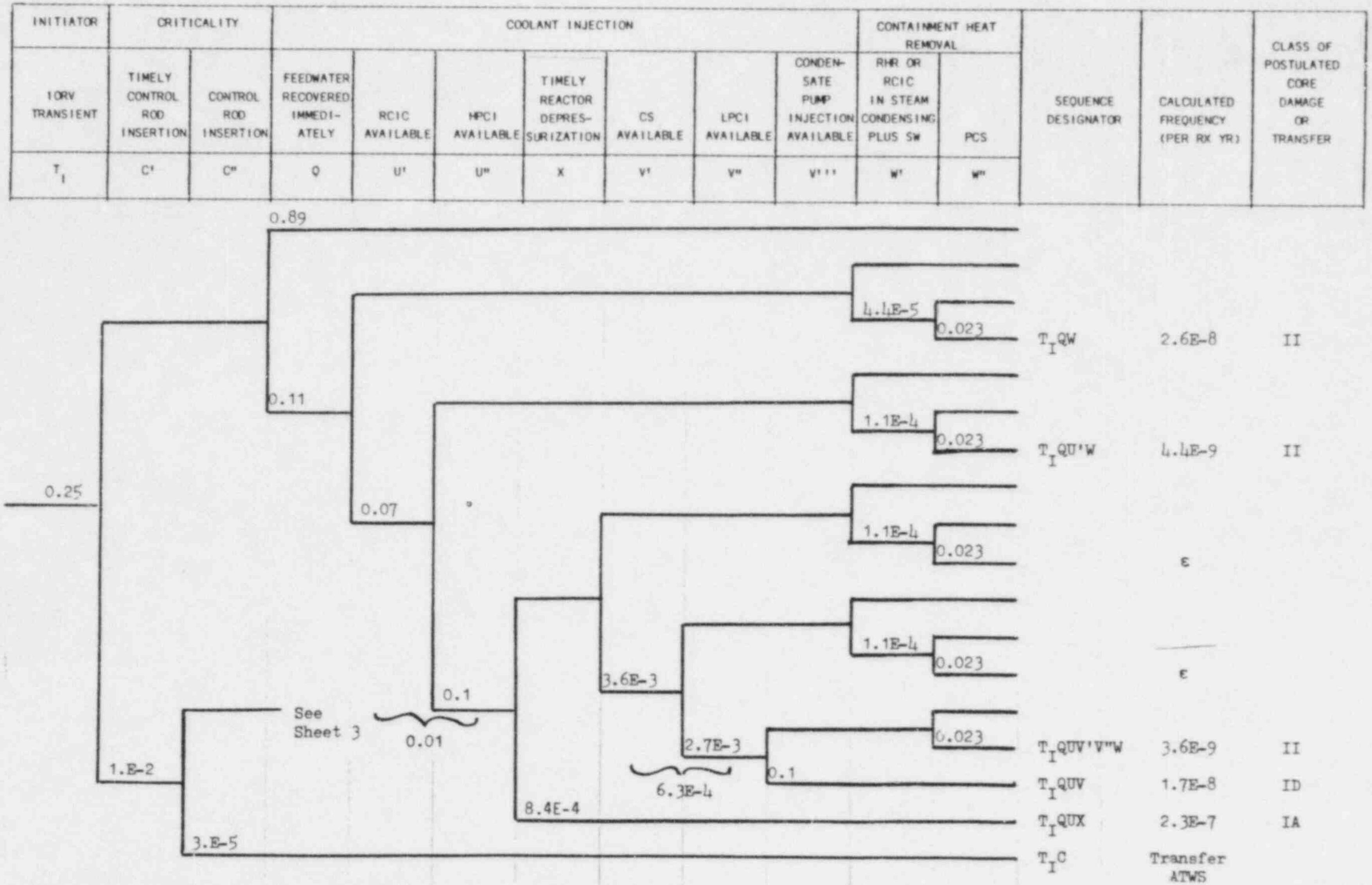


Table 5A.12 Event Tree Diagram for Sequences Following IORV.
(Sheet 2 of 3)

INITIATOR	CRITICALITY		COOLANT INJECTION							CONTAINMENT HEAT REMOVAL		SEQUENCE DESIGNATOR	CALCULATED FREQUENCY (PER RX YR)	CLASS OF POSTULATED CORE DAMAGE OR TRANSFER
IORV TRANSIENT	TIMELY CONTROL ROD INSERTION	CONTROL ROD INSERTION	FEEDWATER RECOVERED IMMEDI-ATELY	RCIC AVAILABLE	HPCI AVAILABLE	TIMELY REACTOR DEPRES-SURIZATION	CS AVAILABLE	LPCI AVAILABLE	CONDEN-SATE PUMP INJECTION AVAILABLE	RHR OR RCIC IN STEAM CONDENSING PLUS SW	PCS			
T_I	C'	C"	Q	U'	U"	X	V'	V"	V'''	W'	W"			

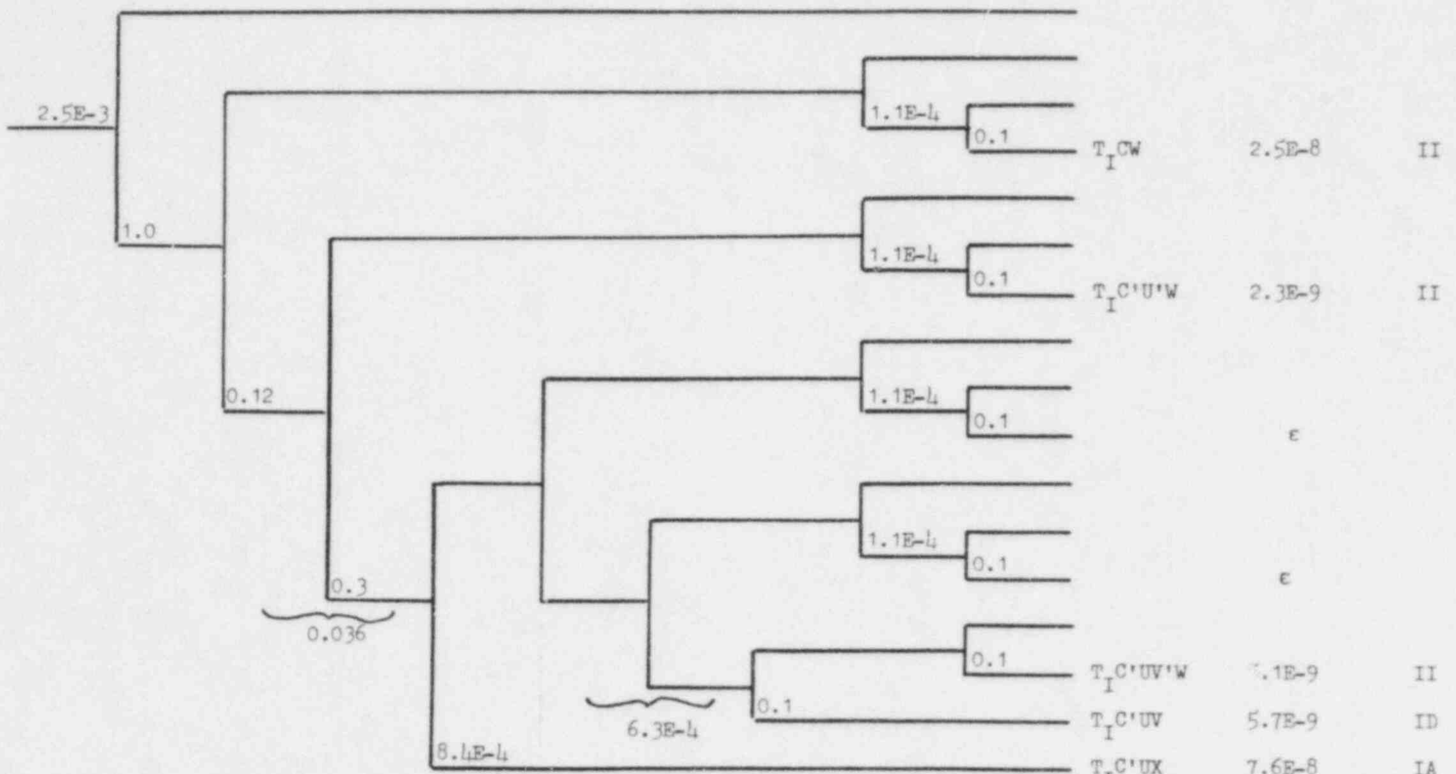


Table 5A.12 Event Tree Diagram for Sequences Following IORV.
(Sheet 3 of 3)

APPENDIX 5B

LOSS OF OFFSITE POWER WITH SUCCESSFUL SCRAM

BNL's review of the contribution of Loss of Offsite Power (LOOP) initiator to the SNPS frequency of core damage is based mainly on Section 3.4.1.6 in the SNPS-PRA.

The LOOP transient is important in the Shoreham PRA because of its high contribution to the frequency of core damage. This is due to of the loss of PCS when LOOP occurs and loss of other frontline systems with the failure of diesels to start or run. The SNPS-PRA analyzed the sequences following LOOP with subsequent loss of diesel generators (blackout) in great detail using four time phases, each assuming blackout conditions not recovered at its start.

Phase I = 0-2 hours: HPCI and RCIC essentially have their normal reliability. Manual ADS is required if HPCI and RCIC fails and diesels are not recovered. The depressurization will allow the use of LPCI with the third diesel train. Suppression pool level and temperature are close to normal.

Phase II = 2-4 hours: Battery is designed to supply DC power for two hours. When operator is successful in shedding out auxiliary loads from the DC power system, the batteries will easily supply the power for this phase. However, HPCI consumes more DC power than RCIC, and therefore two branches for HPCI are modeled:

- HPCI operates from beginning of transient - higher failure probability of batteries.

- HPCI operates only part of Phase II.

At this phase, a switchover of HPCI to suppression pool suction will occur on high suppression pool level. The suppression pool temperature exceeds 140°F at about 2 hours, which is the design temperature of lube oil for HPCI and RCIC. This is more of a problem for HPCI than for RCIC because RCIC can remain or can be returned to CST suction. The drywell temperature is at ≈300°F from the beginning of this phase.

The design of the systems, however, provides sufficient margin to operate reliably during this phase and, in general, failure rates are only moderately above normal.

Phase III = 4-10 hours: The probability of battery failure increases significantly during this phase. Suppression pool temperature exceeds 200°F and may reach 240°F toward the end of this phase. The sustained high temperature in the drywell may degrade the SRVs' solenoid valves and is assumed to result in the failure of ADS in Phase IV if depressurization is not completed in Phase III.

If HPCI started to provide injection before this phase, it is assumed not to survive this phase because of DC depletion and lube oil deterioration.

However, if RCIC operated successfully to Phase III and failed only during this phase, it is assumed that HPCI will be able to complete this phase successfully. In general, in this phase, higher than normal failure rates for these frontline systems are assumed.

Phase IV = 10-24 hours: It is assumed in the BNL review, as in previous BWR-PRAs, that batteries will be depleted during this time. The SNPS-PRA claims small probability for failures of the batteries and possible successful operation of RCIC for the entire time phase. In addition, at times longer than 10 hours the probability of isolation of the RCIC/HPCI steam line due to high area temperatures may become high because of the long time without secondary containment cooling, while in the drywell and the suppression pool the temperatures exceed 250°F, causing a significant amount of heat to be transferred to the secondary containment.

The control room indicators and recorders of the reactor water levels are supplied from RPS and instrumentation buses which have no DC backup. There is apparently one narrow range N004B instrument that is connected to a vital AC bus inverter. Thus, the blackout conditions (even in the case that DC power is available) may result in the loss of level information in the control room. The HPCI in particular and the RCIC systems require level information for their control to prevent level 8 trips. The startup reliability of HPCI and RCIC on subsequent starts is relatively low. Thus, BNL judged that the failure of the injection function during a blackout situation would be about L = 0.05, which makes the sequence T_E I D L one of the most important single sequences of the SNPS. This event is further discussed, and the quantification explained, in Appendix 5F, where the level instrumentation is reviewed. The event is presented in the SNPS-PRA (Figure 3.4-52).

The frequency of the LOOP initiator in the BNL review is 0.15, and it is based on NSAC/80 data as explained in Section 4.1.3 of this report. The SNPS-PRA LOOP frequency is 0.083.

The time phased event trees used by the SNPS-PRA for each of the above time phases were found to be very effective in providing a more detailed and realistic evaluation of the LOOP sequences. However, SNPS used the time phase event trees essentially only for the injection phase. For the containment heat removal phase it used the MSIV closure single-time phase event tree. BNL modeled the containment heat removal function on its Phase I event tree and found a significant contribution to Class II from the LOOP event, which was underestimated in the SNPS approach. This contribution is from a LOOP that is not recovered before 15 hours (≈3%), with recovery of diesel generators followed by their failure to run for the entire decay heat removal mission time. The T_E IV W sequence is the most important to Class II.

BNL's results for the LOOP initiator are significantly higher than those in the SNPS-PRA. These sequences were found to be the most important for Class I states in the SNPS-PRA (PRA page 359). The BNL results are three times as high for Class I and about 1.5 times as high for Class II. The main reasons for these differences were discussed above and can be seen from the event tree diagrams in Table 5B.1. They are summarized in the following list:

a) Loss of all AC could cause loss of water level instrument indications in control room. This can lead to less successful operation of high pressure injections. which require level information for their control.

b) BNL LOOP initiator frequency is twice as high as SNPS-PRA frequency.

c) BNL increased the batteries' failure probabilities for Phases III and IV relative to those in the SNPS-PRA.

d) In the review a Class II sequence is added for unrecovered LOOP, with failure of diesels to run and supply AC power to RHR.

Table 5B.1 LOOP Event Tree Diagram Phase I (0-2 Hours) (Sheet 1 of 5)

Values for Sheet 2 of 5

T_E = 0.15: The probability of LOOP occurrence was discussed in Section 4.1.3 and shown in Table 4.6. The same data base used in deriving the above frequency was also used to generate the LOOP recovery times, which were slightly different from those of the SNPS-PRA (see Table 4.7 in Section 4).

I = 0.37: Offsite power recovery within 30 minutes. The value is derived from Table 4.7. The recovery probabilities in the BNL review are somewhat larger in the short term.

D = 3.6E-3: Appendix A.5 of the SNPS-PRA provides the data and basis for this diesel generator failure probability. These data are discussed in Section 4.2.2 of the BNL review. The data are derived from evaluation of LERs. Even though the data base does not go beyond 1978, it is significant and appropriate. BNL used basically the same values for D in the following way:

$$D = 0.02 \times 0.19 \times 0.95 = 3.6E-3.$$

The first two numbers are from Appendix A.5 (0.02 = single diesel failure to start and run; 0.19 = conditional probability $P[2|1]$). The 0.95 value was used for DG non-recovery within 30 minutes. It is from another BWR-PRA review[4], because the basis for the 0.88 value used in the SNPS-PRA is unexplained. Note that recovery of diesel generator or offsite power is considered successful if only a single diesel or a single offsite line becomes available. This results in a small and insignificant underestimation of failure to run probabilities because, in a fraction of the cases, one train will be available.

U' = 8.E-2 or 7.E-2:
U" = 0.11 or 0.1

These values are used for the RCIC system the same way as in the SNPS-PRA. The additional 0.01 in the first case is included to take into account the possibility of the Division I battery failing during the first two hours of RCIC operation. Similarly, 1.1E-1 and 1.E-1 are used for HPCI, with the 0.01 added for the possibility of Division II battery failure during the first two hour period. The value U = 1.1E-2 seems to account reasonably for a possible CMF of both Division I and II batteries and was used without change in the BNL reassessment.

Table 5B.1 LOOP Event Tree Diagram Phase I (0-2 Hours)
(Sheet 1 of 5 Continued)

Values for Sheet 2 of 5 (Continued)

X = 0.02: A timely ADS failure probability of 0.02 is used in the SNPS-PRA to account for operator failure to initiate ADS manually when injection has failed and automatic ADS initiation is unavailable because of blackout conditions. The same value was used in the BNL review.

V = 2.E-3: The value V = 2E-3 combines the availability of the low pressure systems (6.3E-4) with the failure of the diesel generator to run during the next 10 to 20 hours (≈1.E-3).

V = 0.63: The value V = 0.63 is taken from Appendix A.5 of the SNPS-PRA with no change (see discussion in Section 4.2.2). This is the conditional availability of the Division III diesel generator, given failure of Divisions I and II, which can drive one of the LPCI pumps.

IV = 0.08: Containment heat removal availability is dependent on the availability of offsite power. The SNPS-PRA considered that offsite power will be recovered before 15 hours and transferred all successful injection cases to the MSIV closure event tree. BNL included explicitly the conditional probability of the recovery of offsite power given it was not recovered in half an hour. This is (1-0.97)/0.37 or 0.08.

W' = 3.1E-4: In the case that offsite power is not recovered (= 0.08), the PCS becomes unavailable and only the RHR can be utilized from onsite AC power. The RHR failure probability is then dominated by the failure to run probability of the diesels. It is estimated that three hours of RHR operation would be sufficient to delay containment failure for many hours so that offsite power will be recovered earlier. For a mission time of three hours (say between 15 and 18 hours after the LOOP), BNL obtained 0.0024x3(hours)x0.19x0.23x0.63 = 2.0×10^{-4}. To that was added the RHR unavailability of 1.1E-4.

Phase II (2-4 Hours)
Values for Sheet 3 of 5

T_E: Transfer-in from Phase I. Two were employed:
(1) From RCIC success = 1.8E-4;
(2) From HPCI success = 1.6E-5.

II = 0.51: Conditional probability of recovering offsite power at 2 hours, given failure to recover it at 0.5 hours. This is 0.51 (see Table 4.7 of this review).

Table 5B.1 LOOP Event Tree Diagram Phase II (2-4 Hours)
(Sheet 1 of 5 Continued)

Values for Sheet 3 of 5 (Continued)

D = 0.69: Conditional probability of recovering diesel generators of Division I or II at 2 hours, given failure to do so at 0.5 hours. This is 0.69 (which brings the value back toward the SNPS cumulative value of 0.66 given in Table A.5-8 of the SNPS-PRA).

U' = 0.1: The SNPS-PRA judgment was that the RCIC conditional failure probability during this phase would be 0.05 to account for the following possibilities:

- Batteries depleted as a result of unanticipated drain. The batteries are designed to provide power for 2 hours. Additional time can be obtained only if operator is successful in removing a sufficient number of loads from the DC buses.

U' = 0.1:

- At ≈1.6 hours the suppression pool temperature reaches 140°F, which exceeds the design lube oil cooler inlet temperature. This is a problem, however, only if RCIC is transferred from CST to suppression pool suction (low probability).

- At ≈2.5 hours suppression pool water level exceeds the high level automatic switchover set point for RCIC. RCIC would generally be kept on the CST, but it requires operator intervention.

BNL considered that these events with higher probability will cause RCIC failure, and a value of0.1 was used in the BNL assessment.

U" = 0.22: Two separate cases have to be considered: the HPCI failure probability given either successful operation or failure of RCIC in Phase I. The values given in the SNPS-PRA were used in both cases. The value of 0.22 for the first case was chosen to account for the following considerations:

- At ≈1.6 hours a suppression pool temperature of 140°F will be reached, which is the design temperature of HPCI lube oil.

- At 2.5 hours an automatic switchover of HPCI to suppression pool may be expected. This cannot be easily bypassed.

- The potential for accumulation of water in the HPCI steam line during standby in Phase I.

Table 5B.1 LOOP Event Tree Diagram Phase II (2-4 Hours) (Sheet 1 of 5 Continued)

Values for Sheet 3 of 5 (Continued)

- The start of HPCI has a significant DC power consumption.

U" = 0.3: The value of 0.3 for the second subtree was chosen to account for the above, and for the additional consideration that HPCI operation from the initiation of the accident has a larger potential for draining the batteries because of the higher consumption of DC power required by HPCI operation.

X = 0.02 or = 0.1: Two values are used on the SNPS event tree. Both are used in the BNL assessment. Depressurization is assumed to be required by procedures down to 150 psi, so that HPCI and RCIC can still be in operation if offsite or diesel power is not recovered. The value of 0.02 is the probability for the operator error in failing to depressurize the reactor manually following failure of high pressure injection systems, or in failing to follow depressurization procedures when the suppression pool heats up. The automatic initiation requires AC power, because automatic ADS is conditional upon the running of one LPCI or LPCS pump. The value of 0.1 is the probability for operator error in not performing an early depressurization of the reactor when high pressure injection is successful. This early depressurization is needed because it is considered that deteriorating environmental conditions in the drywell will at later times degrade the SRVs' solenoid valves and prevent depressurization needed at about 10 hours, when the battery may be expected to fail.

V = 0.63: This is the contribution of the Division III diesel and batteries, which can be used to drive one of the low pressure injection pumps. The SNPS-PRA used a value of 0.55. BNL used, for consistency, the value 0.63, which is used in most other cases in the PRA, and is justified in Section 4.2.2.

Values for Sheet 4 of 5

III = 0.63: Recovery of offsite power for this phase. See Table 4.7.

D = 0.71: Recovery of DGs, which is taken from the SNPS-PRA (Appendix A.5).

U'; U": The probability of RCIC failure during this phase is high because of the factors listed above (see Phase II) and the following:

Table 5B.1 LOOP Event Tree Diagram Phase III A or III C
(4-10 Hours)
(Sheet 1 of 5 Continued)

Values for Sheet 4 of 5 (Continued)

- The probability of battery depletion is higher because of the design life, which is less than 10 hours of operation.

- HPCI/RCIC steam line isolation may be caused by high temperature as a result of having insufficient area cooling and by steam leaks and radiative heat transfer from the suppression pool walls. This could be a significant problem between 6 and 13 hours after the accident initiation. A value of 0.25 for RCIC is used in the SNPS-PRA and in the BNL assessment. For HPCI a value of 0.3 is used if RCIC operated for the first 4 hours successfully. However, the SNPS-PRA assumed a failure of HPCI in Phase III if it was running from Phase II throughout Phase III. A CMF of both HPCI and RCIC due to battery depletion is added in the BNL assessment. Its value is assumed to be 50% of the RCIC failure probability used in the SNPS-PRA (0.13).

X' = 0.2 or 0.3: Maintaining the reactor in a depressurized condition is required in case of high pressure injection failure. DC power is required for SRV operation. The failure of the batteries, assumed by BNL to be 0.13, would be a CMF for this function as well. However, when HPCI fails in Phase III C after operating since Phase II, a higher ADS failure probability is used, because the failure of HPCI is caused largely by depletion of the battery due to longer use of the HPCI system.

X' = 0.3 or 0.4: The difference between Phases III B or III D and III A or III C is due to the judgment made in the SNPS-PRA (page 3-116) that failure to depressurize the reactor in a period longer than 2 hours would lead to the following:

a) Accelerated environmental degradation of the solenoid valves in the drywell preventing long-term depressurization.

b) Dynamic oscillation during late blowdown when high suppression pool temperatures prevail.

Table 5B.1 LOOP Event Tree Diagram Phase III A or III C
(4-10 Hours)
(Sheet 1 of 5 Continued)

Values for Sheet 4 of 5 (Continued)

It was taken into account in the SNPS-PRA by increasing "X" by a factor of 2. In the BNL Phase III B and III D sequences, a higher probability of failure was assigned to the "X" function, i.e., 0.4 rather than the 0.3 used in Phases III A and III C. However, in the BNL model the loss of battery is the main factor affecting the results, and not the quantification of the ADS degradation.

UX = 0.13: A CMF of 0.13 is assumed in the BNL assessment. This is chosen to be 50% of the RCIC failure rate. The choice is based on the premise that a large part of the failure probability of RCIC, HPCI, and ADS results from the depletion of the Division I and II batteries up to 10 hours after the accident started. An assumption of a probability of 0.13 for loss of DC within the period from 4 to 10 hours seems reasonable, and is consistent with the assumption that all DC will be lost in the subsequent time period between 10 and 20 hours.

V = 0.63: The value for Division III LPCI operation is taken to be 0.63 because of the dependencies between diesel generator systems (see Section 4.2.2).

Values for Sheet 5 of 5

U = 1.0: BNL gave no credit for RCIC or HPCI after 10 hours. This is based on the SNPS-PRA arguments (pages 3-114 to 3-130) and is consistent with other BWR-PRAs and their reviews, which assumed loss of batteries before 10 hours if no AC recovery was successful. In addition, the SNPS-PRA argues that the RCIC high turbine exhaust pressure trip (40 psi or 26 psi above normal) would be reached at approximately 14 hours, and, similarly, that HPCI/RCIC steam line isolation may be caused by high area temperature (with no area cooling) before 13 hours. Therefore, BNL assumed that, if AC power is not recovered at 10 hours, then a core damage state would be reached. BNL did not distinguish between Phases III - IV E and Phase IV as was done in the SNPS-PRA, and combined them into one single Phase IV sequence.

X = 1.0: In the BNL assessment the probability of maintaining depressurization after 10 hours was assumed to be 1.0, not 0.95.

Table 5B.1 LOOP Event Tree Diagram Phase IV
(10-24 Hours)
(Sheet 1 of 5 Continued)

Values for Sheet 5 of 5 (Continued)

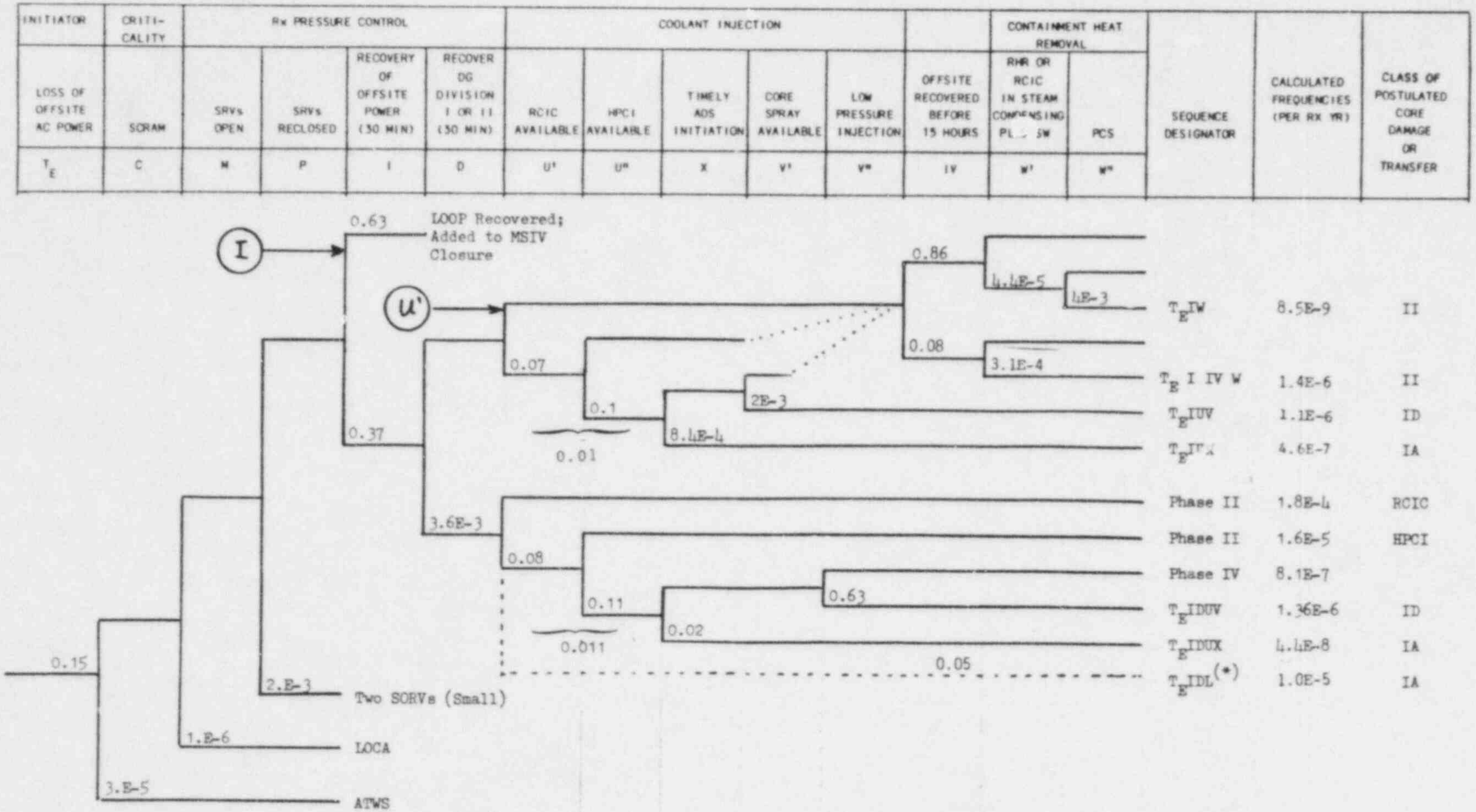
W' = 3.1E-4: When offsite power is not recovered within 15 hours, only the RHR is available for containment heat removal. However, its reliability to complete a mission time of 3 hours (the period from 15 to 18 hours) is basically the reliability of the diesel generators, given by

$0.0024 \times 3 \times 0.19 \times 0.23 \times 0.63 = 2.0 \times 10^{-4}$,

where

- failure to run probability = 0.0024/hr,
- mission time = 3 hrs,
- CMF failure of second diesel $P(2|1) = 0.19$,
- non-recovery probability of diesels within 8 hrs = 0.23, and
- failure probability of Division III train given Division I and II failed = 0.63.

To the 2.0E-4, the RHR unreliability of 1.1E-4 is added, to result in 3.1E-4.



(*)This sequence is discussed in Appendix 5F.

Table 5B.1 Time Phase Event Tree Diagram for LOOP Initiator.
Phase I = 0-2 Hours.
(Sheet 2 of 5)

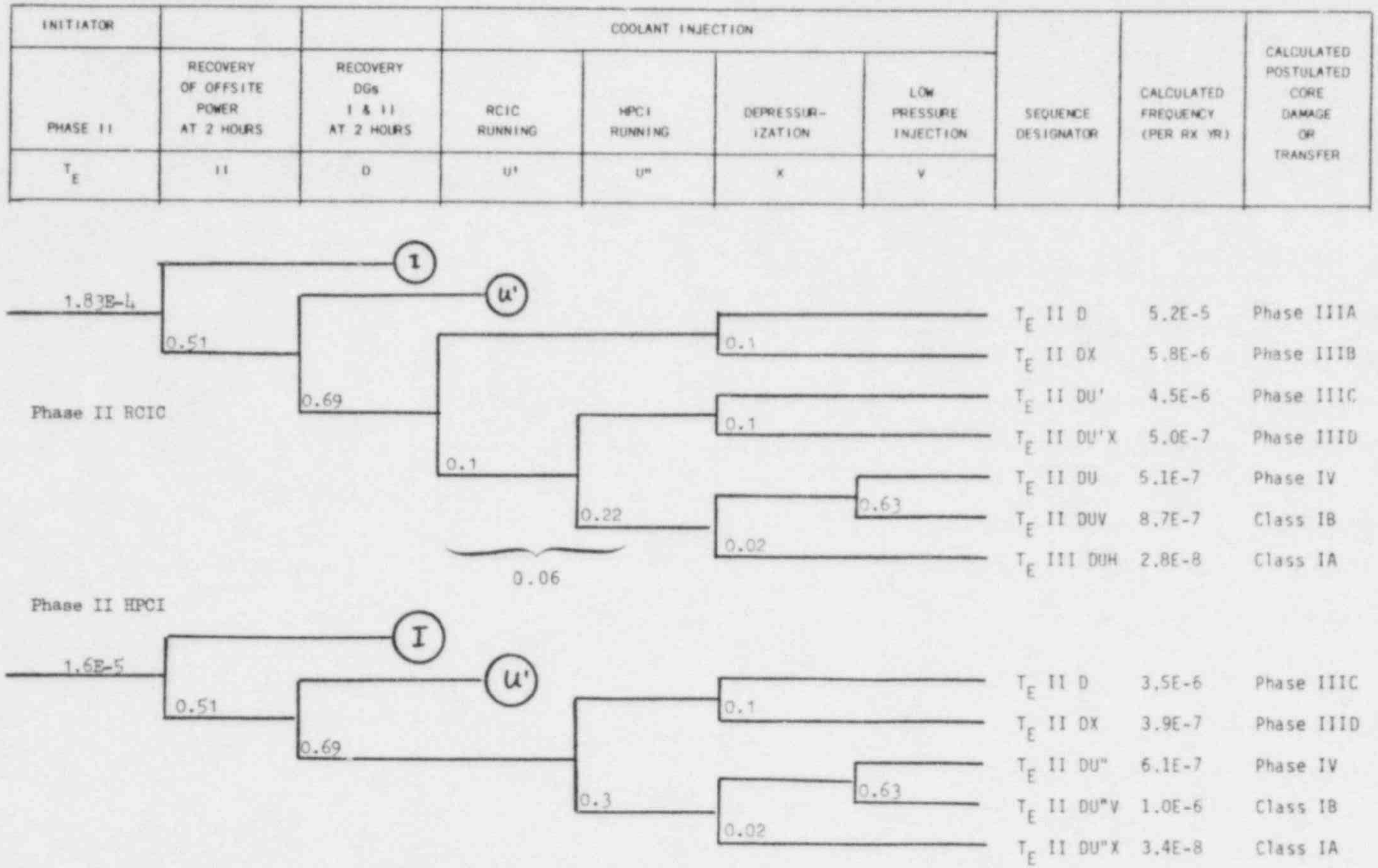
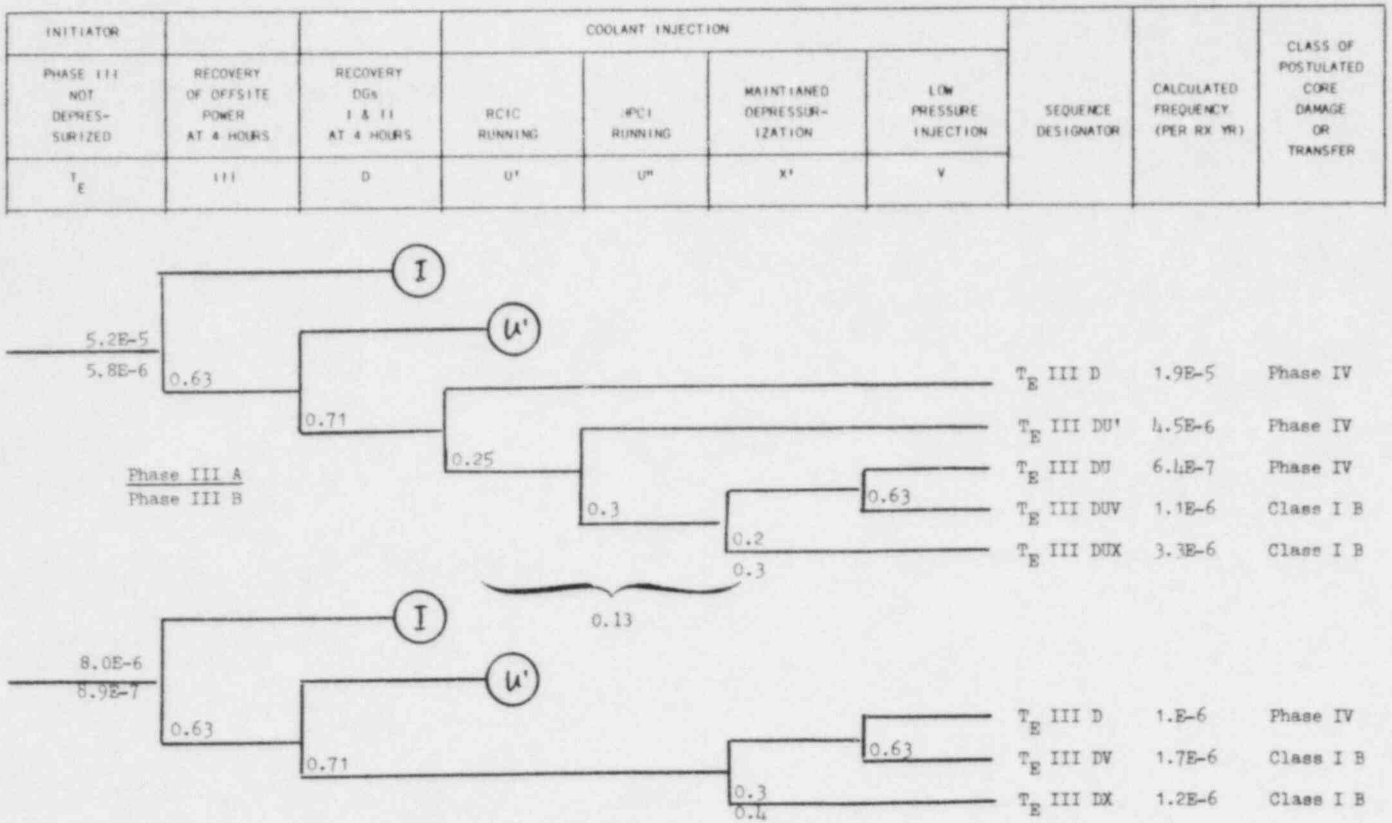


Table 5B.1 Time Phase Event Tree Diagram for LOOP Initiator.
Phase II = 2-4 Hours
(Sheet 3 of 5)



Phase III C
Phase III D

III A: Pressurized and RCIC available
III B: Depressurized, RCIC available.
III C: Pressurized and RCIC unavailable
III D: Depressurized, RCIC unavailable.

Table 5B.1 Time Phase Event Tree Diagram for LOOP Initiator.
Phase III = 4-10 Hours
(Sheet 4 of 5)

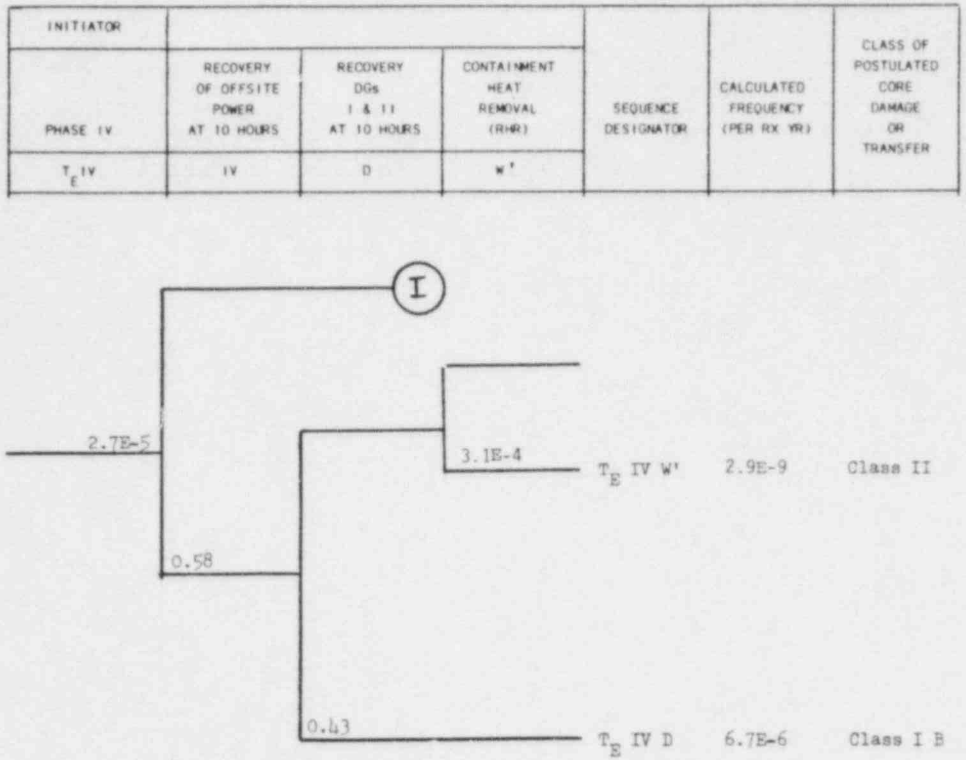


Table 5B.1 Time Phase Event Diagram for LLOP Initiator.
Phase IV = 10-24 Hours
(Sheet 5 of 5)

APPENDIX 5C

LOSS OF COOLANT ACCIDENTS

BNL's review of the contribution of Loss of Coolant Accident (LOCA) initiators to the SNPS frequency of core damage is based mainly on Section 3.4.2 of the SNPS-PRA and on Appendices A.1, A.2, and F.

LOCAs are not important contributors to core damage frequencies. However, their consequences are considered to be greater than those of Class I core damage sequences and therefore their impact on risk might be higher than reflected by the frequencies summarized in this appendix.

The LOCA sequences analyzed are separated into two groups:

a) LOCA inside drywell (Large, Medium, Small and RPV failure)

b) LOCA outside containment (mainly large LOCA in steam lines, water lines, and interfacing system LOCA).

The frequency of group (b) is less than that of group (a), but their consequences are larger because these sequences bypass the primary containment system (drywell and suppression pool). Thus group (b) events, though having lower frequency, are more important with respect to the SNPS risk than group (a).

5C.1 LOCA INSIDE DRYWELL

5C.1.1 Background

The SNPS-PRA approach is very much similar to the RSS-BWR approach and event trees. Two types of breaks are considered, steam line break and recirculation line break. There are differences between the behavior of the reactor vessel pressure and level in the two cases, but both cases can be treated by the same event tree modeling because the differences are in most cases small compared with the impact of the low pressure injection, which in both cases starts within 1 minute after the assumed break, and pumps water in larger amounts than are required to fill the vessel.

The SNPS-PRA chose to model the case of a recirculation line break. It assumes that the line break would render one train of LPCI unavailable. This is modeled on the fault tree, but has a very small effect because low pressure injection is governed by CMF (see Section 3.3.2.7), and the unavailability of one train is not important.

The amount of credit given to PCS is the main difference between the BNL and SNPS-PRA analyses. In the large LOCA case, the SNPS-PRA gives no credit to FW and to PCS even in the long term because of the possibility for radiation isolation of the drywell (MSIV closure). Credit is given only to the condensate pump for injection (even if PCS is unavailable). The value of 0.2 is not explained. BNL uses the same 0.2 for the following reasons:

a) The condensate pumps will remain operating, and will inject ~20000 gpm into the RPV automatically when pressure becomes low as a result

of blowdown. However, at this flow rate the hotwell water inventory will be exhausted in several minutes

b) Therefore, the oeprator is required to control manually the condensate injection to maintain both RPV level and hotwell inventory.

c) The operator will have to replenish water to the hotwell if he failed to control condensate flow in time. Automatic water supply to hotwell from CST is limited to about 1000 gpm[12], and therefore the operator must take control of condensate flow.

In the case of a large LOCA, the 1000 gpm makeup to the condenser hotwell may not be sufficient for all large breaks. It is assumed that for all cases of breaks at an elevation higher than the core, so that steam only will be discharged through the break, the above makeup rate will be sufficient. A flow rate of 500 gpm would be sufficient to remove decay heat by steaming. However, when more than 500 gpm of injection water would be discharged through the break, the makeup of 1000 gpm may become insufficient. Based on some crude estimations, BNL judged that break sizes larger than 10" in diameter may require injection flow larger than the makeup to condenser can provide. Assuming that 50% of the large LOCA breaks would be in this category, we obtain

50% of the breaks: successful condensate injection = 0.2

50% of the breaks: unsuccessful condensate injection = 1.0

Thus, a value of 0.6 was used by BNL for large LOCA.

In the case of a medium LOCA, FW is still assumed unavailable because of MSIV closure on low level or low pressure in the RPV. More credit is given for condensate because a flow rate of 1000 gpm may be sufficient in all cases. In this case credit is given for PCS recovery in the long term, because no radiation from fuel failure is expected.

In the case of a small LOCA, credit is given for FW short-term recovery; however, PCS and condensate are treated the same as in a medium LOCA in the SNPS-PRA. This was slighty changed in the BNL reevaluation, which treated small LOCAs the same way as the IORV transient for the case of early scram (Tables 5A.11 and 5A.12).

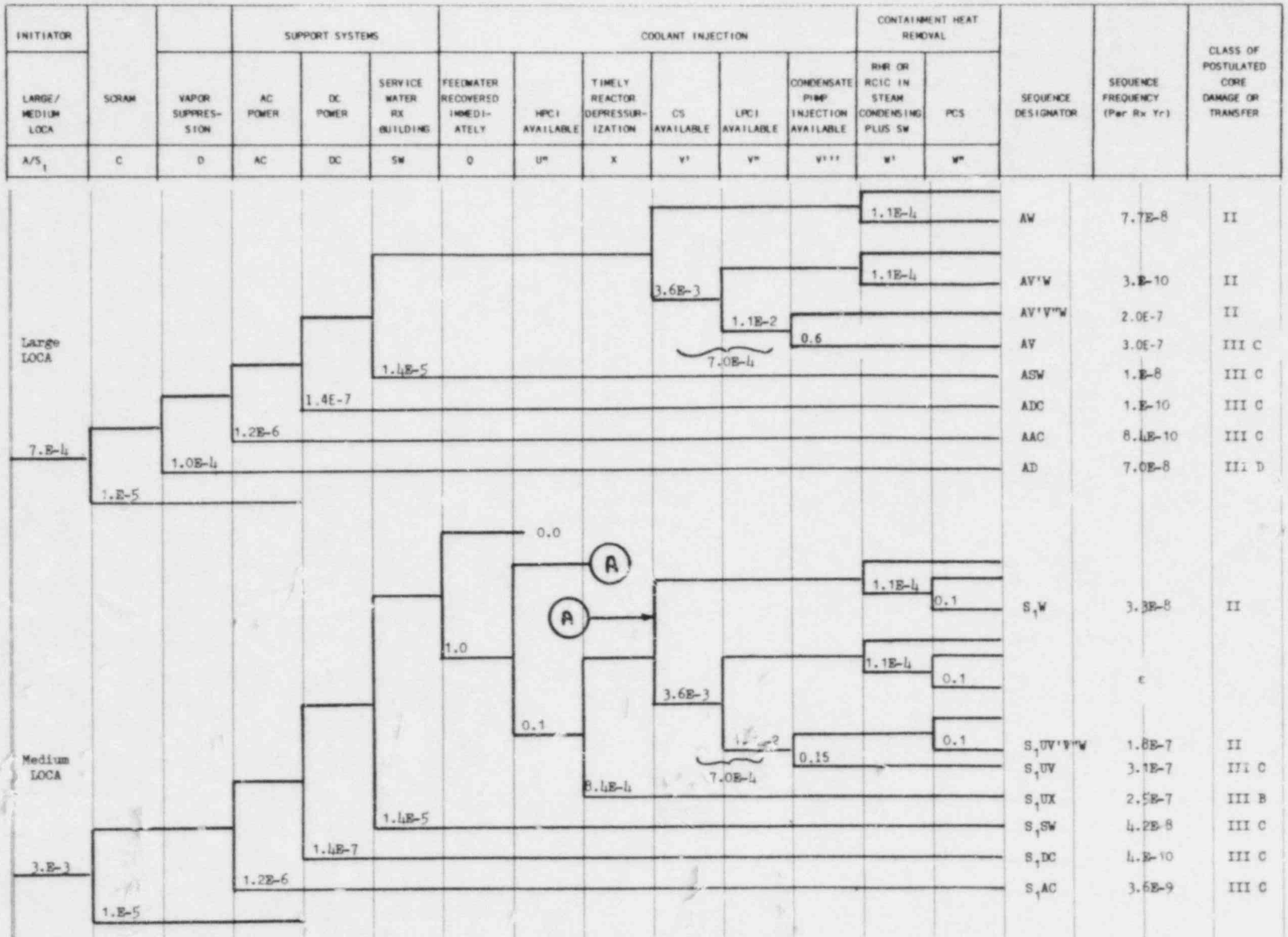


Table 5C.1 Event Tree Diagram for Sequences Following Large and Medium LOCA

5C.1.2 BNL Revised Event Tree

The BNL revised event trees for large and medium LOCAs are shown in Table 5C.1. The revised event tree for a small LOCA is not given; it would be the same as that for IORV with early scram (Table 5A.12 sheets 1 and 2). The frequency of small LOCA is assumed to be same as in the SNPS-PRA, 8×10^{-3}. The effect of small LOCA therefore becomes small, less than 10^{-7}.

The BNL results for the LOCA events are similar to those of the SNPS-PRA. In fact, they are smaller for the Class II contribution and larger for Class III, as seen from Table 5C.2. The reason for the smaller Class II values is that the SNPS-PRA used apparently old values for the "Q"function, which BNL corrected for consistency with the other event trees of the PRA. The cause of somewhat higher Class III contributions is the different quantification of the condensate system injection, which was discussed above.

Table 5C.2 LOCA Contributions to Core Damage Frequencies

		Class		Total Core Damage
		II	III	
Large LOCA	SNPS BNL	7.0E-7 2.8E-7	1.7E-7 3.8E-7	8.7E-7 6.6E-7
Medium LOCA	SNPS BNL	2.7E-7 2.1E-7	4.9E-7 6.1E-7	7.6E-7 8.2E-7
Small LOCA	SNPS BNL	0.24E-7 0.36E-7	0.16E-7 0.08E-7	0.4E-7 0.4E-7
Reactor Pressure Vessel LOCA	SNPS and BNL	———	3.1E-7	3.1E-7
Total	SNPS BNL	1.0E-6 5.3E-7	1.0E-6 1.3E-6	2.0E-6 1.8E-6

5C.2 LOSS OF COOLANT ACCIDENT OUTSIDE CONTAINMENT

A LOCA outside containment has the following adverse characteristics compared with a LOCA inside drywell.

a) In the event of an unisolated break, high environmental stress may be produced on equipment inside the reactor building. This may compromise ECCS operation.

b) In the event of an unisolated break, there may be a flood in the reactor building which may flood high and low pressure injection equipment and compromise their operation.

c) The consequences of core damage in this situation may become significantly different because of the potential direct pathway out of the primary system, bypassing the suppression pool and drywell.

It has a beneficial characteristic in some cases, namely, the possibility of isolating the break in order to limit the release.

The SNPS specific design makes items (b) and (c) of special interest. However, only the core damage probability is evaluated here, not the total risk. The results are assigned a separate core damage class V, for further consequence evaluation.

The SNPS-PRA evaluates the initiator frequency from three sources:

a) Steam line or main feedwater breaks outside containment

b) Breaks in the HPCI/RCIC steam supply or pump discharge lines

c) Interfacing LOCAs in low pressure systems.

Case (c) is the most important contributor to LOCA outside containment. Therefore, larger uncertainties can be tolerated in cases (a) and (b). Most of the uncertainty stems from lack of applicable data for evaluating pipe and valve ruptures.

5C.2.1 Main Steam Line Break Within Reactor Building

The SNPS-PRA assessed the frequency of steam line breaks in the small sections between the inboard isolation valves inside the drywell and the outboard isolation valves inside the reactor building. Breaks downstream of the outboard isolation valves will have two isolation valves between break and RPV, which makes their contribution small. The evaluation in the SNPS-PRA takes the following into considerations:

a) Mean value for pipe rupture taken from the BWR-RSS is 8.6×10^{-10} per hr/section (SNPS-PRA, page A-24).

b) The SNPS pipes in the reactor building steam tunnel are designated as "break exclusion" pipes, which means that they are designed and inspected to even more stringent requirements than the primary system piping. In view of this, the SNPS-PRA applies a factor of 1/10 to the RSS-BWR failure rate. This results in 8.6×10^{-11} per hr/section, which is used for estimating rates of rupture in "break exclusion" pipes.

c) The valves in the subject pipe sections may be subject to external leakage or rupture. The data from RSS-BWR for valve leak or rupture are used. Based on the latest LER review of valves[9], a ratio of 1/18 for rupture/leakage is assumed. In addition, the valves are also

"break exclusion," and an additional factor of 1/10 is taken, which results in 1.5×10^{-10} per hr/valve rupture.

The BNL review notes the lack of a data base for evaluating the rupture probabilities. Considerations (a) and (b) are judged reasonable, but BNL did not review the 1/10 assumption for "break exclusion." Consideration (c) is reasonable, but was judged by BNL to be more appropriate than the LER data from NUREG/CR-1363[9]. Thus the WASH-1400 data used in the SNPS-PRA are also used in the BNL reassessment. The LER data[9] indicate that only a small fraction of the events may be rupture precursors and most of them are leakage that cannot be considered "large LOCAs." Therefore, the factor of 0.05 was judged to be reasonable as well. Additional discussion is given in Ref. 24.

BNL evaluated the annual frequency of steam line breaks by calculating the frequency of pipe or valve breaks in the section outside drywell:

a) 8.6×10^{-11}(rupture/hr) x 24(hr) x 365(days) x 4(pipe) = $3.\times10^{-6}$/yr.

b) 2.7×10^{-8}(leakage/hr) x 0.05(rupture/leakage) x 0.1 (MSIV/MOV) x 24 x 365 x 4 = 5.2×10^{-6}/yr,

where 0.1 is a factor of 10 for assumed better break resistance of the MSIV "break exclusion" valves in the SNPS than of an MOV from the data base (as stated in the PRA).

The inboard isolation valve in the drywell is normally open. It can be isolated and is assumed qualified for this purpose. Its failure rate from NUREG/CR-1363 (Table 23)[9] is, for BWRs,

Failure to close 6×10^{-3}/d.

This value is also used by SNPS-PRA. The probability of unisolated breaks then becomes:

$$(5.2\times10^{-6} + 3\times10^{-6}) \times 6\times10^{-3} = 5.0E\text{-}8/yr.$$

Similarly, from the section, between the outboard MSIV and the Jet Impingement Barrier, we obtain an additional contribution of 6.0×10^{-9}yr. This brings the total calculated frequency for main steamline breaks to 5.6×10^{-8}/yr.

5C.2.2 Feedwater Line Break Contribution

There are two feedwater lines 3 feet long up to the check valve in the reactor building. The failure probability of these is calculated by

8.6×10^{-11}(rupture/hr) x 2(pipe) x 24(hr) x 365(day) = 1.5×10^{-6} per reactor year.

2.7×10^{-8} x 0.055 x 0.1 x 24 x 365 x 2 = 2.6×10^{-6} per reactor year.

The conditional probability for check valve failures is taken by BNL from the Reactor Safety Study to be 3.8×10^{-7}/hr for BWR check valve internal leakage

(mean value). This gives $3.8 \times 10^{-7} \times 24 \times 365 = 3.3 \times 10^{-3}$, which is smaller than the value used in the SNPS-PRA (5.8×10^{-3}).

The contribution of FW line breaks then becomes 1.4×10^{-8}. This value is considered conservative because not all leakages through the inboard check valve are large enough to be the size of a large LOCA.

5C.2.3 HPCI/RCIC Steam Line Break Contribution

RCIC lines are 4" and 3" in diameter and are considered to be too small to cause a large LOCA outside containment. Furthermore, because steam blowdown through the 4" line break will be relatively slow, the time until it will impact equipment in the containment will be relatively large. Hence, there is a significant probability that the operator will successfully follow procedures and will depressurize the reactor by ADS, routing the steam blowdown to the suppression pool rather than to the reactor building atmosphere. In Ref. 24 it is shown that the conditional probability of core damage given medium LOCA is, by a factor of 10, smaller than the conditional probability in the case of large LOCA. Therefore, the contribution from RCIC lines will be small relative to the contribution to core damage frequency from the HPCI lines.

The HPCI has one 10" line, and in response No. 17 to BNL questions[6] it is stated that the HPCI pipe section to the first outboard valve is of "break exclusion" pipe. Therefore, the contribution may become

8.6×10^{-11} (rupture/hr) x 24(hr) x 365(day) = 7.5×10^{-7}/yr.

$0.1 \times 2.7 \times 10^{-8} \times 0.055$(valve rupture/hr) $\times 24 \times 365 = 1.3 \times 10^{-6}$

$2.0 \times 10^{-6} \times 8 \times 10^{-3} = 1.6 \times 10^{-8}$

where 8×10^{-3}/d is the failure of the inboard valve including failure of its command.[9] BNL assumes that this valve will be closed upon demand, because it was designed* to isolate upon sensing the conditions of a steam line break.

Downstream of the outboard isolation valve, which is normally closed, 4 challenges per year of 24 hours each may be assumed. However, piping is non-break-exclusion in this part. Therefore, the contribution from these sections will become

6 (sections) x ($8.6 \times 10^{-10} + 2.7 \times 10^{-8} \times 0.055$) x 4 (challenges) x 24 = 1.4×10^{-6}/yr

$1.4 \times 10^{-6} \times 2 \times 10^{-3}$ (two isolations valves fail by CMF) = 2.8×10^{-9}.

The total frequency of a HPCI steam line break becomes 1.9×10^{-8}/yr.

*The review did not address the question of the adequacy of isolation valve qualification. However, Section 5C.2.5 below compares the contribution of HPCI steam line break to the impact from interfacing LOCA for the assumption of isolation valve failure.

It should be noted here that the SNPS has the outboard isolation valve of HPCI normally closed. In LGS, for example, the inboard and outboard valves are both normally open which increase the contribution from the downstream piping of the HPCI system.

In addition to the HPCI/RCIC lines there are other lines that can potentially cause a LOCA outside drywell if their isolation valves fail:

(1) Reactor Water Cleanup (RWCU) system supply lines

These are 3" to 6" lines having, in addition to the inboard and outboard isolation vlaves, two remote operation valve-arrangements that can be used to isolate the RWCU if a break outside drywell occurs.

(2) Main Steam Line Drain (Inboard)

These are 3" lines. They are not considered for the same reason that is given above for RCIC lines.

(3) Main Steam Line Drain (Outboard) and MSIV Leakage Control

These are 2"-3" lines and are isolated by the inboard MSIV.

(4) Other small lines of size less than 2" in diameter.

All these lines were not further considered by BNL on the basis of the assumption that their isolation valves will close as designed. In such a case, the core damage frequency estimated for these lines (see Ref. 24) is about an order of magnitude smaller than that estimated for the large steam line break in the last three subsections.

5C.2.4 Interfacing LOCA Frequency

If a set of multiple failures should occur, a LOCA could be induced outside containment in piping systems that are rated for low pressure. This is referred to as interfacing LOCA. This section reviews Appendices F and A.2 of the SNPS-PRA, which consider the frequency of interfacing LOCA. It has two parts:

a) Review of SNPS-PRA approach;

b) The BNL reassessment.

The specific pipes of low pressure systems which are potentially sources of an interfacing LOCA are the following:

a) RHR/LPCI loops A and B. Each loop has a testable check valve and two electrically interlocked motor-operated valves (MOVs) in its injection lines. The inboard MOV 37A or B (F015--normally closed) will not be cycled untill the plant is entering cold shutdown. The outermost of the two MOVs--MOV 36A or B (F017 normally open) will be cycled on a 3-month frequency. However, the BNL review considered this second MOV to be unqualified as an isolation valve.

b) RHR reactor head spray line. This has a check valve and two MOVs in series. The MOVs are interlocked to prevent opening at pressure above 135 psi.

c) RHR shutdown cooling mode line, which has two MOVs in series. The MOVs are interlocked to prevent opening at pressure above 135 psi.

d) LPCS loop A and B. Each loop has a testable check valve and MOV in series in its injection lines. The MOV will be checked only during outages.

SNPS procedures state that the testable check valves will be tested during refueling outages only.

A. A Review of the SNPS-PRA Approach

The SNPS-PRA approach to quantification of the frequency of interfacing LOCA follows NUREG-0677[10] with some modifications. The data are valve failure rates taken from NUREG/CR-1363[9]. An analysis of operator errors led to the conclusion that the probability of MOV inadvertent opening by the operator with subsequent failure to isolate is a small contributor.

The SNPS-PRA produced a small reduced fault-tree for each of the four configurations of low pressure systems listed above. The top event is "Large LOCA in Low Pressure System Given Exposure to High Primary System Pressure." These fault trees do not allow for spurious opening of MOVs due to false signals. In one case, credit is given on the tree for MOVs which are not qualified for isolation. This has the effect of doubling the result of the calculations so that both LPCS and LPCI loop A and B contribute similarly (rather than the LPCS alone, as presented in the SNPS-PRA).

The data used for the quantification of the fault trees are taken from the NUREG/CR-1363 with needed modifications. Because MOV or check valve large ruptures did not occur, and the data available are for leakage only, a modifying factor had to be estimated for the fraction of large leakages or ruptures in the entire leakage data. The SNPS-PRA assumes that this factor is 5%. BNL was not able to validate this value. Based on a review of LERs in NUREG/CR-1363, BNL judges that this factor may range from 0.01 to 0.15.

Nothwithstanding, BNL found the SNPS-PRA values too difficult to reproduce. If NUREG/CR-1363 data for BWR valves are used, then by applying the SNPS-PRA approach one may derive the following values:

a) Check valve internal leakage: 1×10^{-6}/hr x 8760(hrs) = 8.8×10^{-3}/yr. Applying the 0.05 factor for large leakages gives 4.5×10^{-4}/yr, which is 1.5 times the SNPS-PRA value appearing on the fault trees.

b) Check valve or MOV rupture: 7×10^{-8}/hr x 8760(hrs) x 0.05 = 3×10^{-5}/yr. The value used in SNPS-PRA is 6 to 7 times as high.

If Reactor Safety Study data is used, then one may derive:

a) Check valve internal leakage: 3.8×10^{-7}/hr x 8760(hrs) = 3.3×10^{-3}/yr

b) Check vaive or MOV rupture: 2.7×10^{-8}/hr x8760(hrs) x 0.05 = 1.3×10^{-5}/yr.

B. BNL Reassessment Approach

The reassessment is based on 6 LERs circulated recently by the Office of Analysis and Evaluation of Operational Data[3,25] of the NRC. These events are precursors events, in which a failure of the boundary between high and low pressure systems has occurred at least temporarily. The data cover events that occurred over more than 15 years. BNL assumed that they are relevant to the BWR reactor operating experience of 250 reactor years. Table 5C.3 provides a short description of the LERs. The following is concluded from the LERs:

a) At least two cases of pressurization of the low pressure systems have occurred for a few minutes (Browns Ferry, Vermont Yankee LERs).

b) Five events are relevant to testable check valves unavailability. If one assumes 250 reactor years, then 0.02/yr is this estimation of frequency.

c) During the two cases of overpressurization, the pipes did not breach or fail. Plants returned to normal operation.

(d) The events were all isolated or recovered shortly.

An additional MOV has to fail in order to challenge the low pressure system. For quantification of the MOV failure probability the following was considered:

a) The conditional probability of spurious opening of an MOV is assumed to be 10^{-3}. This includes mainly the effect of spurious control signals. This value is taken from Table A.2-1 of the SNPS-PRA ($\lambda = 1.6 \times 10^{-7}$/hr). The human contribution during functional testing is assumed to be small because the operator will immediately isolate the MOV when an alarm is received, as occurred in the Browns Ferry LER (Table 5C.3). Furthermore, it is assumed that functional tsting will be performed only during cold shutdown (as specified in SNPS procedures).

b) The data for MOV ruptures or gross leakage seem to be 1.3×10^{-5}/yr. The LER data for MOV failed open (for normally closed MOVs) was evaluated by the SNPS-PRA to be 1.24×10^{-4}.

c) Shoreham has an interlock logic of the injection MOV and the primary system pressure. This interlock is considered to reduce the probability of spurious openings by a factor of 10.

Based on the above consideration, a value of 1.5×10^{-4} was used by BNL for the MOV failure to the open position. This value is considered to include the effects of operator recovery and SNPS specific procedures that require testing of the testable check valves and the MOVs, during cold shutdown.

Table 5C.3 LER Summaries for Interfacing LOCA Events

No.	Plant	Date/LER	Description of Event
1	Browns Ferry 1	08/14/84 LER-84-32	A combination of improper assembly of testable check valve with operator error (failure to electrically disarm the MOV injection valves) caused the check valve to be open for a long period of time (since December 1983) and the MOV to open while testing, compromising high/low pressure boundary. The pressurization of the LPCS above its 500 psi design continued 13 minutes without significant damage. The seal of one pump burst and sprayed steam. This is probably due to substantial design margin. Plant continued power operation. Note: SNPS procedures do not allow for testing the outboard LPCS MOV during power operation.
2	Pilgrim	09/29/83 LER-83-048	During functional testing of HPCI system logic, personnel error occurred causing opening of both injection MOVs. A testable check valve was partially open because of rusted stem to actuator linkage. This caused overpressurization of HPCI (150 psi design pressure). This caused no LOCA, but ruptured the gland seal condenser gasket on the HPCI turbine. The overpressurization caused the testable check valve to close after a short time.
3	Hatch	06/07/83 to 10/28/83 LER 83-112	The testable check valve of the LPCI/RHR was stuck open for about 4 months. This resulted from maintenance errors.
4	LaSalle	09/14/83 LER-83-105 Also: LER 83-066 LER 82-115	Stuck open LPCI testable check valve. The operator opened one LPCI injection valve during routine testing, and leakage into the suppression pool occurred. The plant was in cold shutdown.

Table 5C.3 LER Summaries for Interfacing LOCA Events (Continued)

No.	Plant	Date/LER	Description of Event
5	Cooper	01/21/77	During steady state operation, while the HPCI system logic was being tested, HPCI testable check valve failed to close allowing feedwater backflow into HPCI injection line. HPCI system was isolated. A loose part was found wedged under the edge of the check valve disc preventing the valve from seating.
6	Vermont	12/12/75	During monthly testing of LPCI pump and MOV, one MOV failed to respond. This was because a testable check valve was leaking past its seat causing an excessive dp across the MOV. Another isolation valve was closed before the test but did not shut fully. Its light indicated it was shut. Since this MOV was thought to be shut, the second MOV was cycled open, and a flow pass existed from the RPV to the LPCI loop. This caused the LPCI to be overpressurized past its 450 psi design pressure. Three LPCI relief valves discharged steam and water mixture and a gasket in the RHR heat exchanger leaked.

For a LOCA to occur, the piping must break and the break must be large. The SNPS-PRA states that the low pressure system piping is designed to 500 psi by ASME code standards, with ~100% margin. It assumed that break probability will be 0.5 given high pressure. BNL estimated this probability to be 0.1, on the basis of the following arguments:

(1) The LERs already show two cases of a low pressure system sustaining the high pressure without significant damage, for a significant time period. (This by itself gives a factor of about 1/3.)

(2) The low pressure piping is designed to meet the ASME code, which includes large margins. This indicates that the two cases in which the low pressure system was pressurized for some time and did not breach are apparently typical and not mere chance. Ref. 26 assumes that the large margins may be evaluated as failure probability of 10^{-2} to 10^{-4}. However, it is also stated there that this evaluation has not yet been completed.

Note that Ref. 26 predicts higher LOCA frequencies. However, SNPS procedures do not allow for testing the outboard MOVs during power operation. This can reduce the frequency of the initiating event considerably because five of the six LERs were cases of testing performed on MOVs during plant operation, and therefore may not fully apply to the SNPS.

The BNL approach is summarized as follows:

2×10^{-2} (testable check valves unavailability) x 1.5×10^{-4} (MOV opening) x 0.1 (rupture probability) = 3×10^{-7}/yr.

5C.2.5 Comparison of the Contribution from Steam Line Breaks and from Interfacing LOCA

The frequency of an unisolated HPCI steam line break was estimated in Section 5C.2.3 above to be 1.9×10^{-8} per year. This frequency includes the assumption that the inboard isolation valve on the HPCI steam line can be closed if available upon sensing the conditions of a steam line break. However, if it is postulated conservatively that this isolation valve would fail to close against the pressure conditions of the steam blowdown through the valve into the downstream break, the unisolated HPCI steam line break frequency will become 3.5×10^{-6} per year (see Section 5C.2.3). The frequency calculated in Section 5C.2.4 for interfacing LOCA in the SNPS is 3×10^{-7} per year. It is lower by a factor of ten if no credit is given to the inboard isolation valve closure following HPCI steam line break. Thus, the SNPS-PRA results are sensitive to assumptions on HPCI isolation valve qualifications. Ref. 24 discusses the case of unisolated LOCA outside containment.

5C.2.6 Core Damage Frequency for Large LOCA Outside Containment

The initiators of this sequence were discussed in the previous sections. The results are:

Interfacing LOCA frequency	= $3x10^{-7}$/year
Feedwater and Steam Line Breaks	= $1x10^{-7}$/year
Total	= $4x10^{-7}$/year

The BNL review considers the main impacts of the LOCA outside containment to be the following.

a) Adverse environmental conditions leading to degradation of motor control centers and other electrical equipment.

b) Flooding of the reactor building which has the potential to flood ECCS pumps. The flooding of this systems can, in some cases, happen within less than 10 minutes.

c) Depletion of water from the condenser hotwell leading to insufficient water at the condensate pumps suction.

The SNPS-PRA considers the main impact to be somewhat different:

a) Item (a) above

b) Depletion of water from the primary containment and suppression pool leading to insufficient water at the ECCS suction.

The event tree diagram for this incident in the BNL review is the same as that in the SNPS-PRA. However, in some cases the consideration behind the quantification is different. The event tree is shown in Table 5C.4. The ECCS pumps are considered to be failed because of adverse environmental conditions and flooding.

The condensate system is the main frontline system remaining in this case. BNL assumed a failure probability of 0.2 for the condensate system for the following reasons:

a) The operator needs to control the condensate flow promptly in order to reduce flooding rate, but mainly to conserve the hotwell inventory and thus avoid condensate pump failure or trip upon low hotwell level.

b) The operator should validate that automatic transfer for hotwell makeup from the CST is working.

c) It is assumed that a condensate flow of 1000 gpm to the RPV, which is consistent with the CST makeup to the hotwell, is sufficient to keep the core covered even without line break isolation only for break size smaller than 10" in diameter or for breaks in pipes connecting at a high point of the RPV. (The BNL reviewers failed to find physical calculations showing that for very large break LOCA [such as in the case of interfacing LOCA] the core could be successfully cooled by 1000 gpm.) Therefore, the following was assumed in BNL quantification of the condensate system injection:

For steam line breaks: $v''' = 0.2$

For feedwater line breaks $v''' = 0.2$

For LPCI interfacing LOCA $v''' = 1.0$

For LPCS interfacing LOCA $v''' = 0.2$

$$\overline{v'''} = (1x10^{-7}[0.2] + 1.5x10^{-7}[1.0 + 0.2)]/(4x10^{-7}) = 0.5$$

This value was used in Fig. 5C.4.

The event-tree diagram shows that break isolation is dependent upon re-establishing the PCS and opening an MSIV to allow the containment heat removal function. The W' and W" functions are the following:

W' - Unisolation of the break, with decay heat being removed through the break into the reactor building.

W" - Isolated break and PCS established for containment heat removal of decay heat.

The BNL results are 5-fold higher than the SNPS-PRA results mainly because of the use of the LER occurrences and some differences in failure rate assumed for valves and for the condensate injection.

The BNL review determined that the condensate system is not affected by a flood or adverse environmental conditions in the reactor building. Furthermore, the outboard valves on the feedwater injection line (valves F032A and F032B) through which the condensate pumps transfer cooling water into the RPV are operated from MCCs at elevation 112'9" (40' above the main steam lines in the reactor building) which are located in separate environmentally controlled cubicles isolated from the remainder of the reactor building[12]. The two valves are operated by two separate MCCs located on opposite sides of the containment.

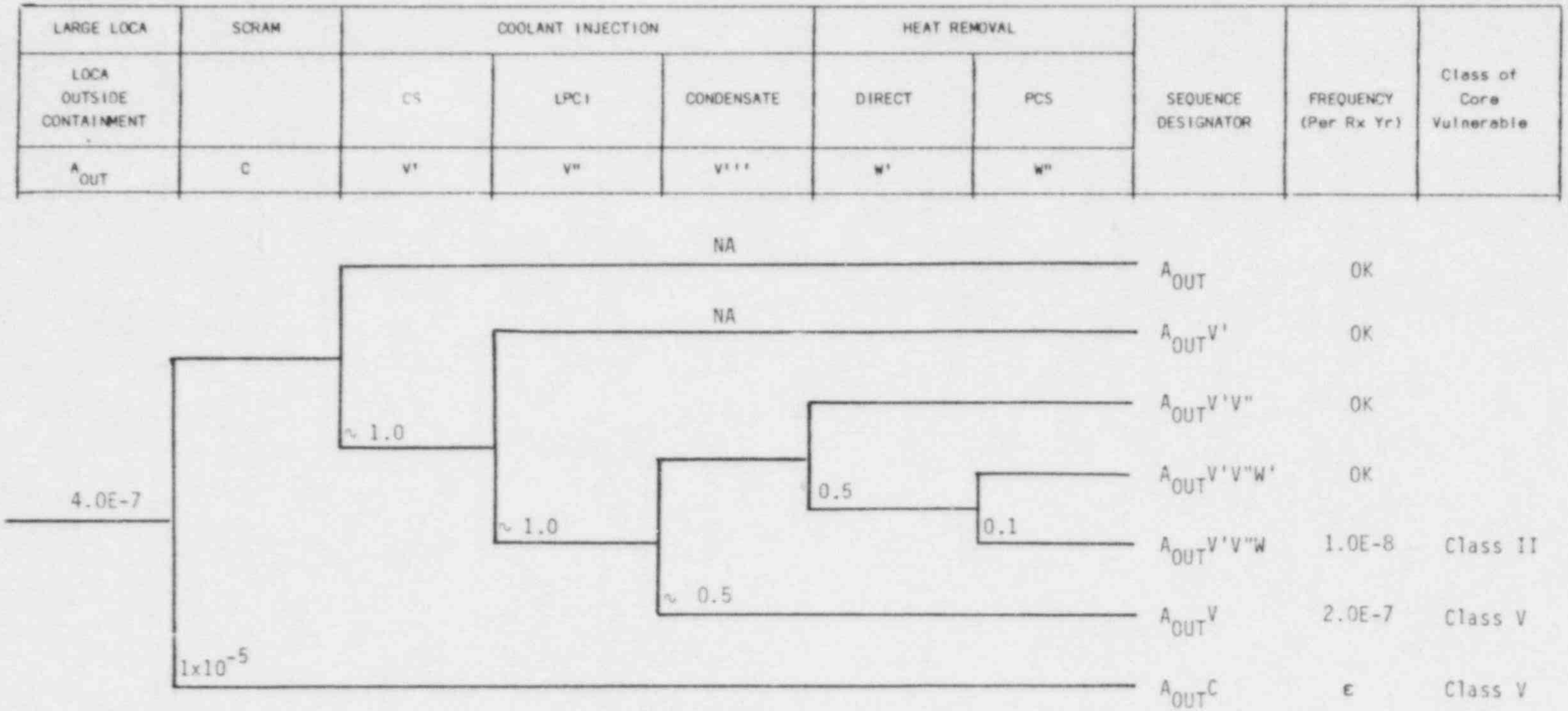


Table 5C.4 Event Tree Diagram for Sequences Following Large LOCA Outside Containment

APPENDIX 5D

ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

5D.1 SUMMARY OF S OREHAM ATWS EVENT TREES

The ATWS event trees developed in the SNPS-PRA are described here, with emphasis on their special features and important aspects; they are discussed in detail in Section 3.4.3 of the SNPS-PRA.

A total of five ATWS functional event trees were developed for the SNPS-PRA: turbine trip, MSIV closure, Loss of Feedwater, Loss of Offsite Power (LOOP), and IORV. A special event tree was developed for the turbine trip ATWS initiator (Figure 5D.1). Since the purpose of this event tree is to identify properly those turbine trip initiator events that eventually result in either a loss of feedwater, a loss of condenser, or a MSIV closure, the event tree evaluates the availability of the following functions: feedwater runback, loss of turbine bypass valves, loss of condenser heat sink, and MSIV closure. The outputs from this event tree are scenarios that can be characterized as a turbine trip with bypass available, a loss of feedwater event, a loss of condenser event, or a MSIV closure event. On the basis of these results, the respective ATWS initiator frequencies are reevaluated. For instance, the ATWS turbine trip initiator frequency becomes 0.85/year instead of 3.2/year, and the loss of feedwater ATWS frequency is 0.08/year rather than 2.10/year.

Figure 5D.2 shows the SNPS ATWS turbine trip event tree. A major departure in the SNPS-PRA treatment of ATWS events from that in other BWR PRAs is that it separates the initiator frequency of a particular ATWS event into that above the 25% power level and that below the 25% power level. A case in point is the turbine trip event presented in Figure 5D.2. The SNPS-PRA reported that only 0.85 event/year can be considered to be turbine trip with bypass available and restore power level above 25%, and the balance constitutes 1.3 events/year for which the reactor is operating below the 25% power level. The rationale for selecting the value ?5% is based on the condenser's capability to remove heat. The probability of an ATWS event occurrence is based on the NUREG/0460[23] values of $1x10^{-5}$ for reactor protection system (RPS) mechanical failure and $2x10^{-5}$ for RPS electrical failure. The recirculation pump trip function is implemented in the SNPS and is actuated given a high reactor vessel pressure or a low reactor water level condition. Alternate rod insertion (ARI) is also installed in the SNPS to provide a redundant means of inserting the control rods, should the RPS electrical system experience malfunction. If indeed an ATWS event is imminent, then the tree evaluates the pressure control functions: namely, the proper opening or reclosing of safety relief valves. The reactivity control function used in the SNPS-PRA entails 4 different tasks: manual initiation of the SLC system, manual feedwater trip to minimize cold water injection into the core, lowering the reactor water level to slightly above level 1, and lastly, re-establishing water level and boron mixing when the SLC tank is empty. The SNPS analysis assumes that the operator will have 25 minutes to perform these tasks. The high pressure injection function, U, is then evaluated. ADS inhibit, D, and water level control, U_H, are also included in the event tree to model the need to preserve the boron concentration inside the reactor vessel. Finally,

the event tree considers the success of the heat removal function through the condenser and the RHR heat exchangers.

The combination of success or failure of these functions, shown in Figure 5D.2, gives rise to the definition of an ATWS accident sequence. For instance, based on the success criteria defined for a turbine trip ATWS event, failure of Recirculation Pump Trip (RPT), given RPS electrical failure, results in a core damage condition. Also, with successful RPT, failure of the ARI and the reactivity control function would still result in core damage. Part A of Figure 5D.2 shows these accident sequences, which are related to RPS electrical failure, and Part B shows sequences related to RPS mechanical failures. Subsequent to the reactivity control function, the tree evaluates the coolant injection and ADS inhibit functions, and finally the maintenance of level and containment heat removal functions.

Figure 5D.3 is an ATWS event tree, similar to that for the turbine trip initiator for MSIV closure events. This tree is also divided into two parts, for mechanical and electrical RPS failure sequences. The initiator frequency is classified into two groups according to whether the power level at the time of reactor scram is above or below 25%. The MSIV ATWS event tree is identical to the turbine trip tree except that the unavailabilities of the various functions are different. Included in this MSIV ATWS initiator frequency is the contribution from loss of condenser ATWS events. These are grouped together and treated in the same event tree because both initiators result in a similar plant response of losing the capability for heat removal to the heat sink.

The loss of feedwater ATWS event tree is shown in Figure 5D.4. The SNPS-PRA considered two power levels, below 25% or above 25%, for this event. The main difference between this event and the turbine trip ATWS is the unavailability of feedwater. In this case, feedwater runback is not necessary. Similarly, the availability of the condenser for the loss of feedwater event distinguishes this event from an MSIV ATWS event, in which the condenser is not available. Otherwise, the ATWS event tree is identical to the other two trees.

The loss of offsite power ATWS event tree (Figure 5D.5) is essentially the same as the MSIV ATWS tree. Given the onset of a LOOP event, the MSIV will close and the response of the plant to the initiator is similar to a MSIV event. However, a LOOP event does, in certain cases, present a more notable challenge to the system availability than the other ATWS discussed thus far because of the loss of offsite AC power. This is noted in the unavailability of the heat removal function; otherwise the two trees are identical.

The last ATWS event tree developed in the SNPS-PRA is that for an IORV event. It is similar to the others, described above, but it contains one additional function that models the failure of the high drywell pressure or high suppression pool temperature signal (Figure 5D.6). Given the onset of an IORV transient, at the initial stage the reactor operator is instructed by the procedures to manually shut down the reactor; however, failure to do so does not necessarily preclude a scram since at high drywell pressure ~2 psi, an automatic scram signal is generated. The SNPS-PRA determined that failure of the suppression pool temperature and the drywell pressure instrumentation would result in the equivalent of an ATWS sequence. This is reflected in the SNPS IORV ATWS event tree.

5D.2 QUALITATIVE REVIEW OF THE SNPS ATWS EVENT TREES

This discussion of the results of the BNL qualitative review of the SNPS ATWS functional event trees is focused on several topical items rather than on each ATWS initiator.

Turbine Trip Initiator Event Tree

BNL's review of the SNPS turbine trip initiator event tree (Figure 5D.1) indicates that the function "feedwater runs" consists of feedwater runback action by the operator in 12 minutes, so as to preserve an orderly shutdown with low suppression pool temperatures. It is considered to have a high likelihood of failure in the SNPS-PRA. Failure of this function leads to either a loss of feedwater or a MSIV closure. However, this appears to contradict the definition of plant condition given for each sequence. For instance, the sequence T is characterized by the success of feedwater runback, turbine bypass, condenser heat sink, and MSIV open. But, if feedwater runback is successful, then the T sequence should behave more like a loss of feedwater than like turbine trip with bypass available. A similar example can be noted in the TQ sequence, where failure to runback, implying that feedwater is available, results in the loss of feedwater events. One possible explanation is that the upper branch of the feedwater run function should be interpreted as no feedwater runback. The SNPS-PRA states that a 0.4 operator error probability is assumed for failure to manually runback feedwater and a 0.75 failure probability is used for the automatic backup feedwater trip on Level 8.

High Power Initiator Frequency

In the SNPS-PRA, the ATWS initiator frequency is separated into two parts: that at high power plant condition, greater than 25% power, and that at 25% power or lower. The basis for this division is existing plant data from BWRs.

BNL did reassess the initiator data base to determine the relative contributions from such a grouping (see Table 5D.2). BNL considered that during the normal operation of a plant, i.e., not including the initial period of commercial operation, some percentage of plant transients would be initiated at low power, and credit should be given to reflect this situation where the condenser is adequate in removing heat from the reactor vessel, thus allowing additional time for the operator to initiate the SLC system. Depending on the nature of the data base, if, during the initial period of plant operation, there tend to be more scram events at plant condition of 25% power or less, then the estimation of this percentage can be potentially biased toward the low power events, and may not be representative of the plant over its averaged life. For the BNL reassessed core damage frequency, all ATWS events are assumed to occur at power greater than 25%, similar to the SNPS-PRA.

Water Level

As described in the preceding section, the SNPS design provides a number of means for reactivity control in an ATWS event. These include injection of boron into the reactor vessel by the SLC system, manual feedwater runback, and lowering of reactor water level to slightly above level 1. BNL concurs that these are important measures which can serve to reduce the reactor power.

With regard to the task of lowering water level, the SNPS-PRA suggested in one place that the water level be maintained slightly above level 1 and in another place that the water level be maintained near the top of active fuel (TAF), and the SNPS ATWS emergency procedure guide[13] offers no insight into this apparent discrepancy, stating that the water level should be kept above TAF. In a broad sense, these statements are not contradictory, but it is left to the reader to interpret the true intent of the procedure. Furthermore, based on the physical analysis performed to support this action, an 8% power level was cited in the SNPS-PRA. This power level corresponds to the water level at TAF. Hence, there is, at best, an uncertainty as to the level at which the reactor water must be maintained. The effects of this operator action are discussed further in the next section.

SLC System Initiation

The SNPS design has two SLC loops, each with the capacity to inject 43 GPM of sodium pentaborate into the reactor vessel, but the maximum injection rate is 43 GPM, so that only one loop can be injecting at any one time. The system is manually actuated. A 25-minute action time is allowed by SNPS-PRA for this task. BNL reviewed the GE report NEDE 24222[17] and the KMC letter[18], and concluded that the maximum action time allowed for the reactor operator appears more likely to be about 15 minutes.

ADS Inhibit

Since the initial submittal of the SNPS-PRA, a modification to the ADS function (including a preliminary conceptual design drawing) was conveyed to BNL via responses to the BNL questions[6]. This modification entails a manual inhibit switch for use during an ATWS event, should the reactor vessel water level drop below level 1, and is designed to eliminate the need for the operator to repeatedly reset the ADS timer. BNL has assessed the impact of this modification by a sensitivity analysis given in Section 5.3. The effect of a manual inhibit switch upon the success of low pressure ECCS in transient events warrants more thorough investigation, since inadvertent operation of the switch would disable all low pressure ECCS. With regard to ATWS considerations, this appears to be a useful design with the benefit of reducing the probability of failure of the operator to achieve timely inhibition of the ADS, as shown in Table 5.15 of Section 5.3.

BNL found that the SNPS ATWS procedures were not clear in a few areas as to what the operator must accomplish upon the onset of an ATWS. A case in point is the ADS inhibit function. In the procedure, the operator is instructed to initiate either the A or B SLC loop given a range of plant conditions (see Table 5D.1, item 3.6.1). The operator is further required to terminate all injection into the RPV except the CRD and HPCI or the CRD and RCIC maintaining reactor water level above TAF (item 3.6.1.2). At this point, two scenarios are possible. The first is quite benign, in that the reactor water level falls to a point where the operator, by controlling high pressure makeup, is able to maintain the water level above level 1 at all times. In the second scenario, the reactor water falls quickly even with rated high pressure injection systems, and the water level drops below level 1. The procedure does not appear to provide the instruction necessary to guide the operator to identify the critical parameter that must be closely monitored in reducing the water level, and to perform the inhibit function. In item 4.2 of

Table 5D.1, the operator is only directed to manually open enough SRVs to reduce reactor pressure to between 800 and 960 psig when there is cycling of the SRVs. Given the critical nature of this function, failure of which is assumed to lead to core damage, perhaps this operator action warrants more attention than it has been given in the procedure guide thus far.

5D.3 SUMMARY OF PHYSICAL ANALYSIS RESULTS

A few of the ATWS analyses performed on BWRs, and their results, are discussed here, with an emphasis on areas having more direct effects on the assumptions as well as the ground rules and conduct of the ATWS portion of the PRA. In reviewing the SNPS-PRS ATWS analysis, BNL found either a lack of detailed information on some aspects, or information insufficient for reasonable establishment of assumptions. This deficiency will become more apparent as the discussion continues.

Section 5D.3.1 provides a chronology of the ATWS accident sequence, and Section 5D.3.2 focuses on specific areas considered to have more substantial impacts on the ATWS PRA review.

5D.3.1 ATWS Accident Chronology

Given the onset of a plant transient, the MSIV closure event is recognized to impose by far the most severe requirements, compared with other events on the safety systems needed for mitigation. Therefore, for this discussion MSIV closure is selected as the initiating event, and departures from the MSIV discussion will be addressed separately. This discussion will be further confined to BWR-4 reactors.

Upon closure of all MSIVs and failure of the scram system to insert the control rods, an ATWS event is in progress. The reactor pressure rises rapidly causing the safety relief valves to open. Consequently, a substantial amount of heat is being discharged into the suppression pool. Also, the pressure increase in the reactor vessel initiates the recirculation pump trip. Success of such a pump trip will reduce the core power to about 50%. Because the initiating event is a MSIV closure, feedwater will also not be available. Given the large amount of reactor power still being generated, the reactor water level drops rapidly, and at Level 2 both the HPCI and RCIC systems receive a signal to inject from the CST. It is predicted in the GE ATWS report[17] that all of these events occur within a minute after the initial RPS trip signal. At two minutes, the GE analysis assumes that the automatic SLC actuation timer is timed out and the SLC system begins injection into the core. A time trace of the reactor water level (Figure 4.1.3 of NEDE-24222) shows that, after the water level drops below Level 2, the HPCI and RCIC flow reduces the rate at which the level decreases until a point when the boron from the SLC injection begins to take effect. The water level reaches a minimum at about 5 minutes and begins to rise again. This minimum is just short of level 1.

A similar situation occurs with a turbine trip with bypass available event. A time trace of the reactor water level (Figure 4.1.7 of NEDE-24222) shows that the time at which the water level drops to level 2 is about 1.5 minutes. Feedwater is assumed in the GE analysis to be run back within 1 minute after the onset of the event. As in the MSIV case, the SLC is assumed

to begin injection at 2 minutes. Figure 4.1.7 of NEDE-24222 shows that the water level decreases at a lower rate than in the MSIV case. The analysis predicts that at about 5 minute the reactor water level reaches a minimum, which is approximately 1.2 feet above level 1.

The results of the two different calculations indicate that little time, about 5 minutes, is available for the operator to take any action to secure the reactor. According to the SNPS specific ATWS emergency procedures (Table 5D.1), a series of operator actions is to take place. These are of two types: immediate and subsequent. The immediate actions include manually scramming the reactor, tripping the recirculation pumps, initiating RHR suppression pool cooling, initiating SLC, controlling water level, and, if manpower is available, re-scram of the reactor with operation of scram discharge high level bypass and other vent valves and logics. Subsequent actions deal with SRV cycling and plant shutdown procedures.

The SNPS specific ATWS procedures make it obvious that the GE analysis is no longer applicable to the SNPS beyond the 5 minute time frame. An ATWS analysis of Brown Ferry Unit One[19] provides some insights as to the response of the plant given that the operator follows the ATWS emergency procedures guidelines (EPG) for BWR.[20] This ATWS EPG differs in certain areas from the SNPS specific EPG.[13] For instance, the BWR EPG recommends lowering the RPV water level to TAF; it also allows depressurization and use of low pressure systems. In the SNPS EPG, pressure is supposed to be maintained between 800 to 900 psi and no credit is given for low pressure systems. Therefore, the ORNL[19] analysis results are not directly applicable to the SNPS ATWS situation.

The purpose of maintaining the water level below the normal water level is to minimize the amount of heat produced in the core. This, in turn, has two related effects. The first effect is reduction of the amount of heat discharged into the suppression pool, and this allows the second effect: additional time for the operator to actuate the SLC to inject boron into the vessel. The reactor power will eventually diminish because of the boron, and the shutdown procedure can continue.

5D.3.2 Discussion

This section provides a discussion of the pertinent areas that affect the SNPS ATWS PRA.

Water Level Control

According to the SNPS specific procedure, operator control of the water level is important in minimizing reactor power. The SNPS-PRA states that, when the water level is at TAF, the power level is about 8%. This value is referenced in the KMC letter[18], which cites information from GE that ". . . the reactor power level when the water is at TAF should have been 8% rather than 15%." The level to power curve included in the document[18] shows the 16% value (see Figure 5D.7). Figure 5D.7 also shows curves obtained by NSAC[21] and BNL.[22] The range of power level at TAF is between 15 and 20%. If the water level is maintained at Level 1 rather than TAF, then the power level ranges from about 18 to 23%. If the intention is to avoid initiating the ADS function, and to maintain the water level above Level 1, then the power level is

more like 20 to 25%. Because of the significant increase in the slope of the curve near the TAF region, changes in water level in this region have large effects on reactor power.

Suppression Pool Temperature Limit

The SNPS-PRA reports the suppression pool temperature limit to be 240°F; above this point, the plant condition is considered not to be acceptable. Subsequently, the KMC document[18] suggested that, on the basis of GE data on minimum subcooling required for efficient steam condensation, the suppression pool temperature limit may be about 285°F. BNL did not assess the validity of either value, but it is prudent to point out that a 45° increase in the temperature limit provides significant benefit in terms of added allowable time for the operator to perform his task.

SLC System Actuation

The KMC calculations include a sensitivity analysis to model a BWR-4 reactor with a manually initiated 43-GPM SLC system. Three different delay initiation times were assumed in addition to the base case, which is injection in 2 minutes after the onset of an ATWS. The reactor water level was assumed to be at TAF and the power level at about 8%. The maximum suppression pool temperature estimated for a 10-minute delay of initiation is between 260° to 270°F for the SNPS. If the delay is around 2 minutes, the maximum pool temperature is about 220°F.

The above information on water level versus reactor power indicates that, if the water level is at TAF, the power level is more likely to be 18%. This could have a substantive impact on the suppression pool temperature. If it is further assumed that the water level is above Level 1, the time taken to reach the suppression pool temperature limit is even shorter. As a result, the operator will have only a few minutes to initiate the SLC, thus making it a highly likely to fail event.

Summary

In the process of establishing a basis for the SNPS ATWS success criteria and ATWS assumptions, a limited number of documents were reviewed to determine the applicability of their results to the SNPS and the reasonableness of the analyses. A lack of detailed information was found in certain aspects of these analyses; even though these are generic studies, they do not provide a basis broad enough to account for the range of operator actions specific to the SNPS.

The areas of suppression pool temperature limit, boron mixing in the reactor plenum and its impact on delay in plant shutdown, and human action to lower reactor water level are each addressed separately. There is a lack of integrated analysis that could be used to support the SNPS specific situation and the SNPS specific EPG.

It is assumed in the BNL reassessment of the ATWS accident sequence that the water level is to be maintained between Level 1 and Level 2, and that the suppression pool temperature limit is 240°F.

5D.4 QUANTITATIVE REVIEW

The BNL revised ATWS event trees and the ATWS core damage frequency quantification of these trees are discussed here.

Turbine Trip Initiator Tree

As noted in the qualitative review of the SNPS turbine trip initiator event tree, BNL made minimum changes to this tree. The unavailability, 0.7, used by the SNPS-PRA on the feedwater runback function was found to be high. BNL thought that, given the onset of a turbine trip, regardless of whether it is an ATWS event or not, some portion of this event will result in either a MSIV closure or a loss of feedwater or loss of condenser, and developed a revised turbine trip initiator event tree accordingly (Figure 5D.8). The basic structure of this tree is similar to that in the SNPS-PRA. It has a total of four functions: feedwater trip due to high level, turbine bypass, condenser heat sink, and MSIVs remain open. Consistent with that used in the transient event analysis, a 10% probability is assumed for feedwater loss given a turbine trip initiator. In order to further distinguish loss of feedwater events from MSIV closure, a 20% probability is used for failure of the MSIVs to remain open. Loss of turbine bypass or condenser heat sink results in MSIV closure and loss of condenser events, respectively. Given the availability of the feedwater, the bypass and the condenser, the probability that the MSIVs will not remain open is assumed to be 0.02. The end states of this initiator event tree can be clarified into four groups: turbine trip, MSIV closure, loss of condenser, and loss of feedwater. Each of these is transferred to the respective ATWS functional event tree.

Turbine Trip

In the review of the SNPS ATWS turbine trip event tree, BNL found the reactivity control function unavailabilities, namely, RPS electrical or mechanical failures C_E and C_M, recirculation pump trip R, ARI function, and K, to be reasonable. The RPS failure values are derived from NUREG-0460.[23] The R function value reflects sensor failures, and the 10^{-2} value for the K function represents the failure of the diverse logic to scram the reactor.

With regard to SRVs open to control pressure, M, and SRVs reclose, P, the values used are also considered to be reasonable. In general, BNL concurs with the unavailability used for the coolant injection function, U. BNL in the re-quantification revised the values of the remaining 4 functions (C_2, D, U_H, W), and reconstructed the event tree (Table 5D.6).

The first part of the ATWS event tree is identical to that in the SNPS-PRA. Subsequent to the SRV reclose function, the question of feedwater runback is evaluated. Note that the initiator is a turbine trip event with bypass available; the feedwater system continues to provide feedwater flow into the reactor and to maintain the water around the normal level. As discussed above, with regard to the effect of water level on reactor power, if feedwater is not runback, the power level with recirculation pump trip is around 50%. This certainly far exceeds the capability of the condenser.

Therefore, it is important to runback feedwater in a timely manner. The probability of failure to runback feedwater is evaluated to be 0.2, based on the SNPS human error curves.

If feedwater runback is successful in a timely manner, however, then the RPV water level will fall below Level 2 and the probability of failure of the HPCI is assessed to be the same as that used in the transient event trees. RCIC is not considered to be an adequate means of providing coolant injection. In the event that HPCI is successful, the event tree evaluates the "Control Level 1" function and the SLC function. Actually, because of the rapid progression of an ATWS event, the feedwater function, HPCI function, Control level 1 function, and SLC function should be considered to take place concurrently.

It is estimated in the KMC letter that using the EPG no blowdown case with a 10 minute delay in SLC initiation and water level at TAF (8% core power), the suppression pool temperature is calculated to be 221°F. BNL estimated that if core power is at 18%, the 240°F pool limit will be reached. Therefore, BNL assumed that the operator is required to initiate SLC and feedwater runback within 10 minutes. Moreover, if it is above 200°F, the reliability of the HPCI will be significantly degraded because of inadequate lube oil cooling.

As noted in the preceding section, without feedwater the reactor water level will quickly fall below level 2; the SNPS EPG (Table 5D.1) instructs the operator to take control of water level by terminating injection and to maintain it above TAF. Since the MSIV closure and ADS initiation is at level 1 and the EPG contains no explicit instruction for the operator to inhibit ADS, BNL assumed for this study that the water level is to be maintained between level 1 and level 2. The unavailability of the Control Level 1 function is derived from a functional level event tree (Table 5D.7). The tree first evaluates the likelihood that the water level will fall below level 1. A probability of 0.5 is chosen, based on review of a number of documents. The GE[17] report indicates that, even with automatic SLC at 2 minutes, the water level falls to within 1.2 feet of the level 1 setpoint, but the ORNL[19] report indicates that, for a turbine trip event, water does not reach level 1.

The ADS inhibit value of 0.2 is selected on the basis of engineering judgment aided by human error curves (Figure A.3-3 of SNPS-PRA). Finally, failure of the operator to maintain water level above level 1 and below level 2 is assigned a value of 0.1. Failure to control water level will result in core damage.

The SLC manual initiation failure and the RHR initiation failure are given probabilities of 0.15 and 0.1, respectively. Failure of these functions also leads directly to core damage.

Should the HPCI fail to inject given a successful feedwater runback, the RPV water will reach level 1 within a couple of minutes, causing closure of the MSIV and actuation of the ADS if the operator fails to inhibit. For all practical purposes, no successful operator actions can be assumed in these short times, and therefore the control level function is assumed to fail.

MSIV and Loss of Condenser

The MSIV closure and the loss of condenser ATWS are grouped together and treated in one functional event tree (Table 5D.8). The basis for this grouping is the similar plant response of these two types of events. In both cases, the MSIVs are closed and the feedwater injection is lost.

A major difference between this ATWS event tree and the turbine trip tree resides with the feedwater runback function (Table 5D.8). Since the initiator in this case has already caused the loss of feedwater, the runback function is not required and it is represented in the tree with zero failure probability.

Another area of difference is the level control function. A functional level event tree similar to the one developed for the turbine trip ATWS is shown in Table 5D.7-case II. As discussed in the preceding section, the water level following a MSIV or a loss of condenser ATWS initiator reaches level 1 within one or two minutes. Hence, the "water level below level 1" function (Table 5D.7) is chosen to be unity. In light of the situation, the ADS inhibit function is given a 0.9 probability of failure. Given the success of the inhibit function, failure to maintain level is assumed to be 0.2.

For the SLC initiation function, it is assumed in the reassessment that 5 to 10 minutes are available for actuating the system, and it is given an unavailability of 0.25. Similarly, the RHR suppression pool cooling function is assignd a failure probability of 0.2.

LOOP

The loss of offsite power ATWS event tree (Table 5D.9) is developed in a similar way, as the MSIV event tree. The plant and system response are considered identical.

Loss of Feedwater

This event tree also is similar to the MSIV event tree except in the control level function, the SLC function, and the W function (see Table 5D.10). Because of the initiator, feedwater is automatically runback. Despite the loss of feedwater, the MSIV remains open and the condenser is still available. Therefore, the control level function, given the success of coolant injection, is similar to that for turbine trip; the unavailability is 0.19 (Table 5D.7). However, in the event that coolant injection is not successful, the control-level 1 functional level event tree evaluation shows an unavailability value of almost unity (Table 5D.7-case III shows this tree). Since, without injection, level 1 is reached within 1 to 2 minutes, the ADS inhibit function is also assigned a high probability of failure: 0.9. Even in the event inhibit is successful, without injection, level cannot be maintained.

IORV

Table 5D.11 is the BNL revised IORV ATWS event tree. Operation data indicate that the onset of an IORV event often precipitates a loss of feedwater; therefore, the runback function is assumed succesful. The SRV reclose function is assigned unity failure probability because of the initiator. All

the functions other than these are also the same as in the turbine trip event tree.

5D.5 DISCUSSION OF RESULTS

This section presents a discussion of the ATWS results based on the quantification of the BNL revised ATWS event trees and comparisons between the BNL reassessed values and the SNPS-PRA values.

Table 5D.2 lists the BNL ATWS initiator frequencies for six initiators. The first column gives the SNPS ATWS initiator frequencies at 25% power or above, and the third column gives the initiator frequencies with power level larger than 25% used in the transient analysis of this review. Transfers from the turbine trip initiator event tree are identified and listed in column 4; they are made to MSIV, loss of condenser, or loss of feedwater initiators. The last column shows the ATWS initiator frequencies used in the BNL re-quantification.

To illustrate the effects of the BNL modifications to the event trees without the initiators, Table 5D.3 compares the conditional frequency of core damage based on the SNPS and BNL ATWS event trees. Only five initiators are listed; the loss of condenser and MSIV events are consolidated into one group. The increase in conditional frequency is seen to be relatively small; no initiator shows more than a factor of 2 increase, and the MSIV case even shows a slight reduction. Based on this information, it appears that, even though BNL introduced major revisions to the ATWS event trees, the final results (without the contributions from initiator frequencies and from the feedwater runback function) do not change significantly. The results should be interpreted with the understanding that there is a lack of information from physical analysis to fully support the BNL assumptions; they are often derived on the basis of engineering judgment. The final results are also thought to be sensitive to these assumptions made in the reassessment. Moreover, the current EPG can be improved to provide added assurance concerning the operator's role in successfully mitigating an ATWS event. In the SNPS design, the operator is greatly relied on to mitigate such an event, and his failure to follow procedure or to perform a particular task in time is the major contributor to the ATWS core damage sequences. Almost all ATWS accident sequences are related to some form or another of operator error.

Table 5D.4 lists core damage frequencies for the five different types of initiators, obtained by using BNL revised ATWS event trees and SNPS initiator frequencies. The first column shows the SNPS core damage frequencies for comparison. The second and the third columns give the core damage frequencies for Class IV and for the ATWS induced LOCAs based on the BNL revised event trees. The last column is the sum of the second and third, and it gives the total core damage frequencies based on BNL event trees and SNPS initiator frequency. The increase in core damage frequency for most of the initiators is small, less than a factor of two, and there is a slight decrease for the MSIV initiator, from 8.3(-6) to 7.2(-6). The overall increase in core damage frequency is less than a factor of two.

Table 5D.5 lists the core damage frequencies calculated on the basis of BNL revised ATWS event trees and BNL initiator frequencies. It is similar to Table 5D.4, and includes the SNPS core damage values for reference. The Class

IV contribution and ATWS LOCA contribution from the BNL calculation are presented. Note that there are no Class 1 ATWS accident sequences in the BNL quantification. This is because BNL judged that insufficient time is available for the operator to inhibit ADS and prevent a Class IV sequence. BNL judged that most Class IV will result in a core damage due to loss of suppression pool water. The major contributor to core damage comes from turbine trip followed by MSIV events. This is to be contrasted with the SNPS case, where MSIV is the most dominant contributor followed by turbine trip. Note that the BNL MSIV core damage frequency, though contributing less than turbine trip, is still higher than the SNPS MSIV core damage frequency, 1.1(-5) versus 8.3(-6). The major reason for the increase is ascribed to the difference in ATWS initiator frequency. The BNL ATWS core damage frequency is a factor of 2.5 higher than the SNPS value.

5D.6 SUMMARY

BNL reviewed the SNPS-PRA ATWS evaluations, both qualitatively and quantitatively. The assumptions and physical analysis results used in the SNPS-ATWS analysis, as well as the SNPS specific EPG, were reviewed. In general, the SNPS ATWS PRA attempted to model the events as realisticly as possible; areas of conservatism in previous PRAs were explored to provide a realistic picture of the ATWS induced core damage risk. This includes the availability of the condenser heat sink for turbine trip and loss of feedwater events and low power ATWS events. In general, the SNPS analysis was considered to be reasonable and useful in providing an estimate of ATWS core damage risk.

In the course of the review, BNL identified three areas that warrant some discussion here. The first relates to the ATWS physical analysis. There appears to be only a limited amount of ATWS data that are directly applicable to a BWR-4 reactor with a manual 43-GPM SLC system. Consequently, it is difficult to establish critical parameters that define the condition of the Shoreham plant and the time available to the operator for particular actions. Based on the limited analyses, engineering judgment was used in reviewing the SNPS analysis, and changes were made to the SNPS event trees. For instance, these changes affect the time at which RPV water reaches level 1, the suppression pool temperature limit, the effects of 43-GPM SLC on water level, and the effects of delay in actuating the SLC. BNL judges that changes to these physical parameters could have significant major impact in the assessment of core damage frequency.

The second area concerns the SNPS specific ATWS EPG. It is BNL's opinion that improvements in the EPG would be very beneficial in the areas of operator control of RPV water level, ADS inhibit function, and RPV pressure control. More details are needed to assist and guide the operator in responding to the accident at hand.

The last area relates to the extent of operator action required during an ATWS event to secure the plant to hot shutdown. The SNPS requires manual actions for most of the ATWS mitigation systems. However, very little time is available to the operator to perform these tasks; in many cases they must be done within 10 to 15 minutes after the onset of the event. This is why the Shoreham ATWS core damage frequency is about an order of magnitude larger than that of the Limerick or the GESSAR-II standard plant. It is prudent to recognize that there are large uncertainties associated with the estimates of human

errors, and for this reason the ATWS core damage frequency could be very sensitive to changes in the human error probabilities.

Finally, BNL performed a realistic reassessment of the SNPS ATWS event. The results indicate that, given the assumptions used, the increase due to different assumptions and modifications to the event trees is far less than a factor of 2. The ATWS core damage frequency calculated by BNL using the SNPS initiator frequency is 2.2(-5), compared with the SNPS-PRA value of 1.8(-5). Use of the BNL initiator frequency increases the total core damage to 4.5(-5), which is about 2.5 times the SNPS-PRA value.

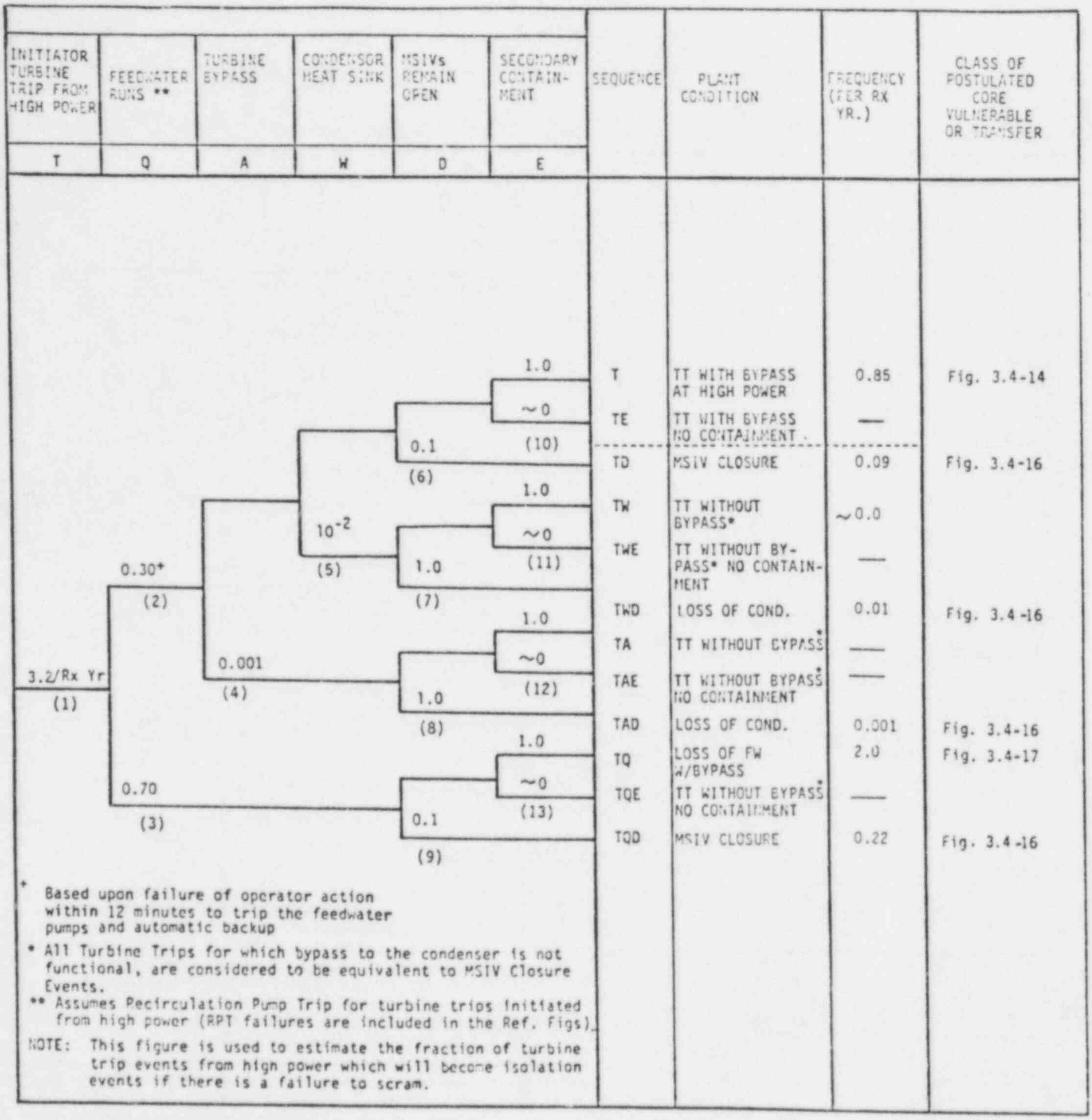


Figure 5D.1 Event Tree Diagram of Accident Sequences Following a Turbine Trip Initiator From High Power

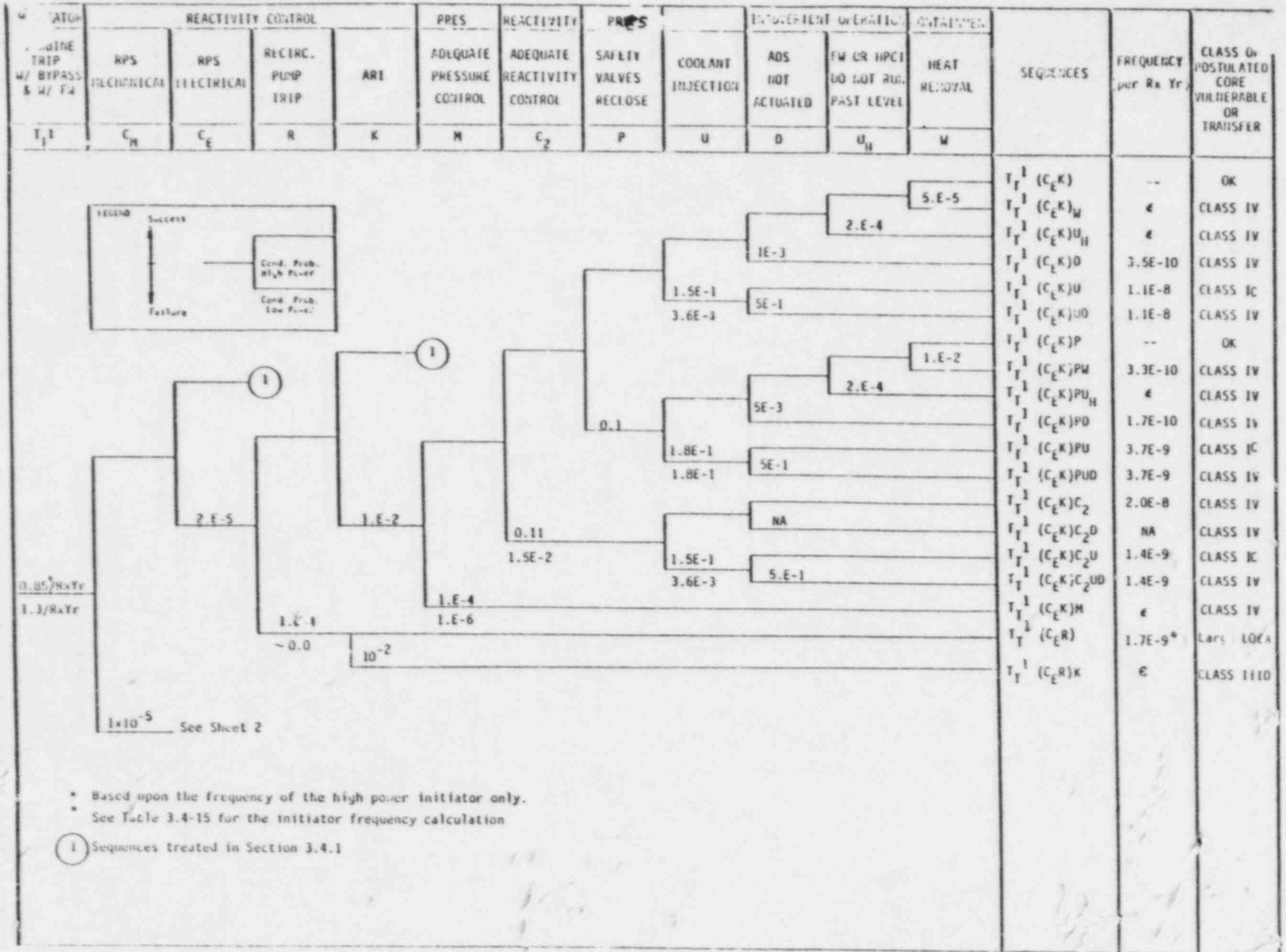


Figure 5D.2 Event Tree Diagram for Postulated ATWS Accident Sequence Following a Turbine Trip w/Bypass Available (Sheet 1 of 2)

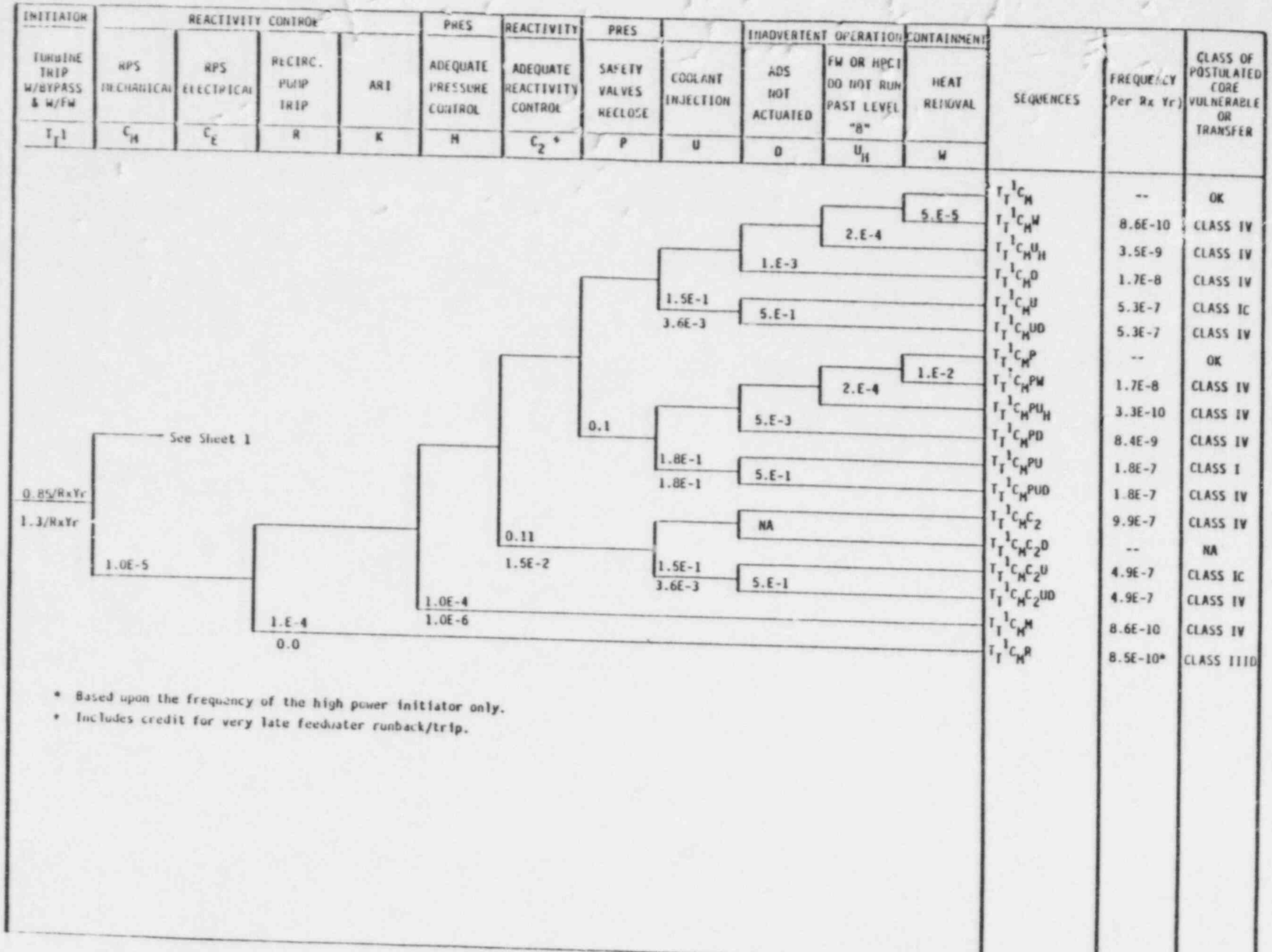


Figure 5D.2 Event Tree Diagram for Postulated ATWS Accident Sequence Following a Turbine Trip w/Bypass Available (Sheet 2 of 2)

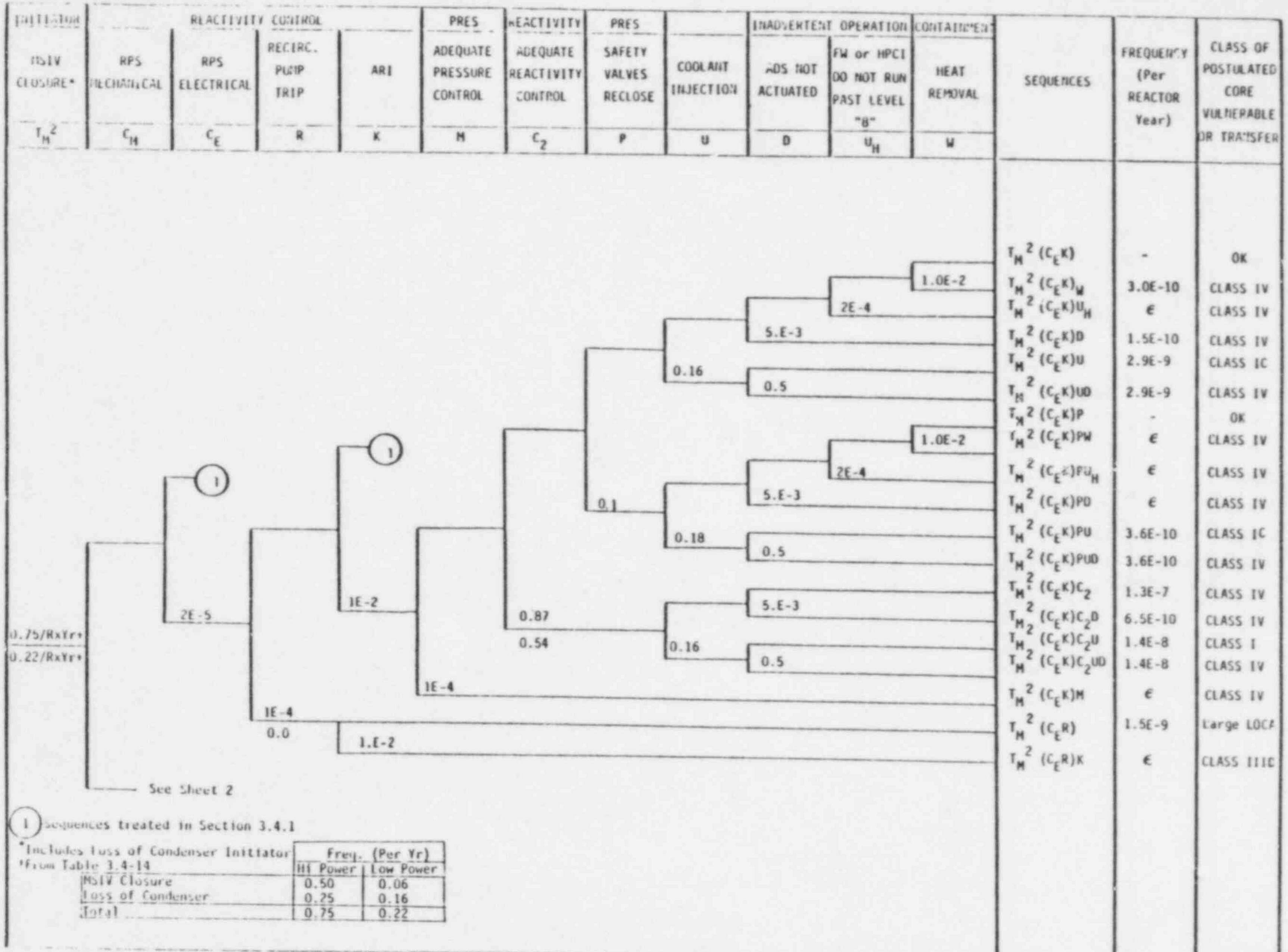


Figure 5D.3 Event Tree Diagram for Postulated ATWS Accident Sequence Following a MSIV Closure (All Initial Power Levels) (Sheet 1 of 2)

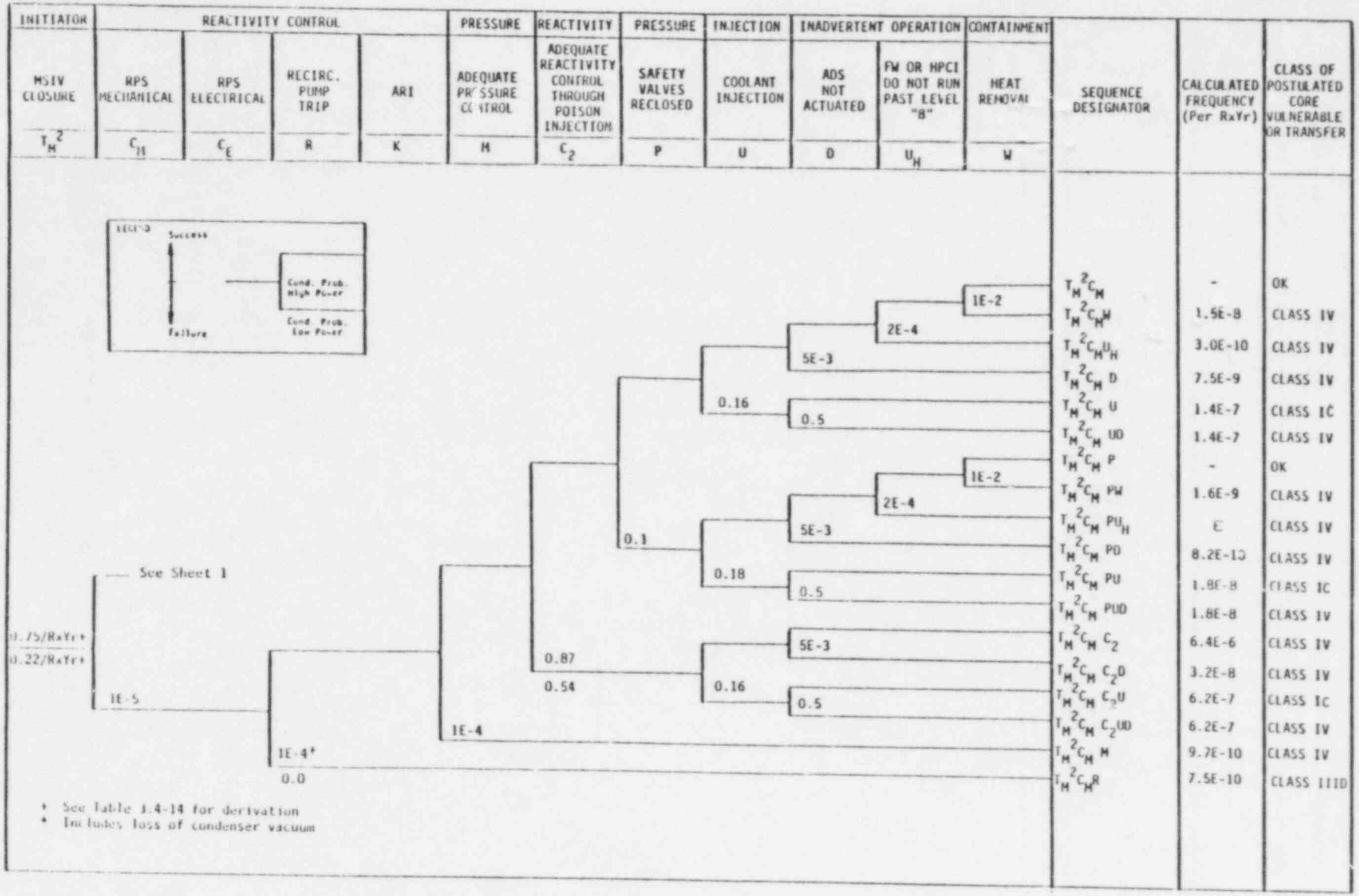


Figure 5D.3 Event Tree Diagram for Postulated ATWS Accident Sequence Following a MSIV Closure (All Initial Power Levels) (Sheet 2 of 2)

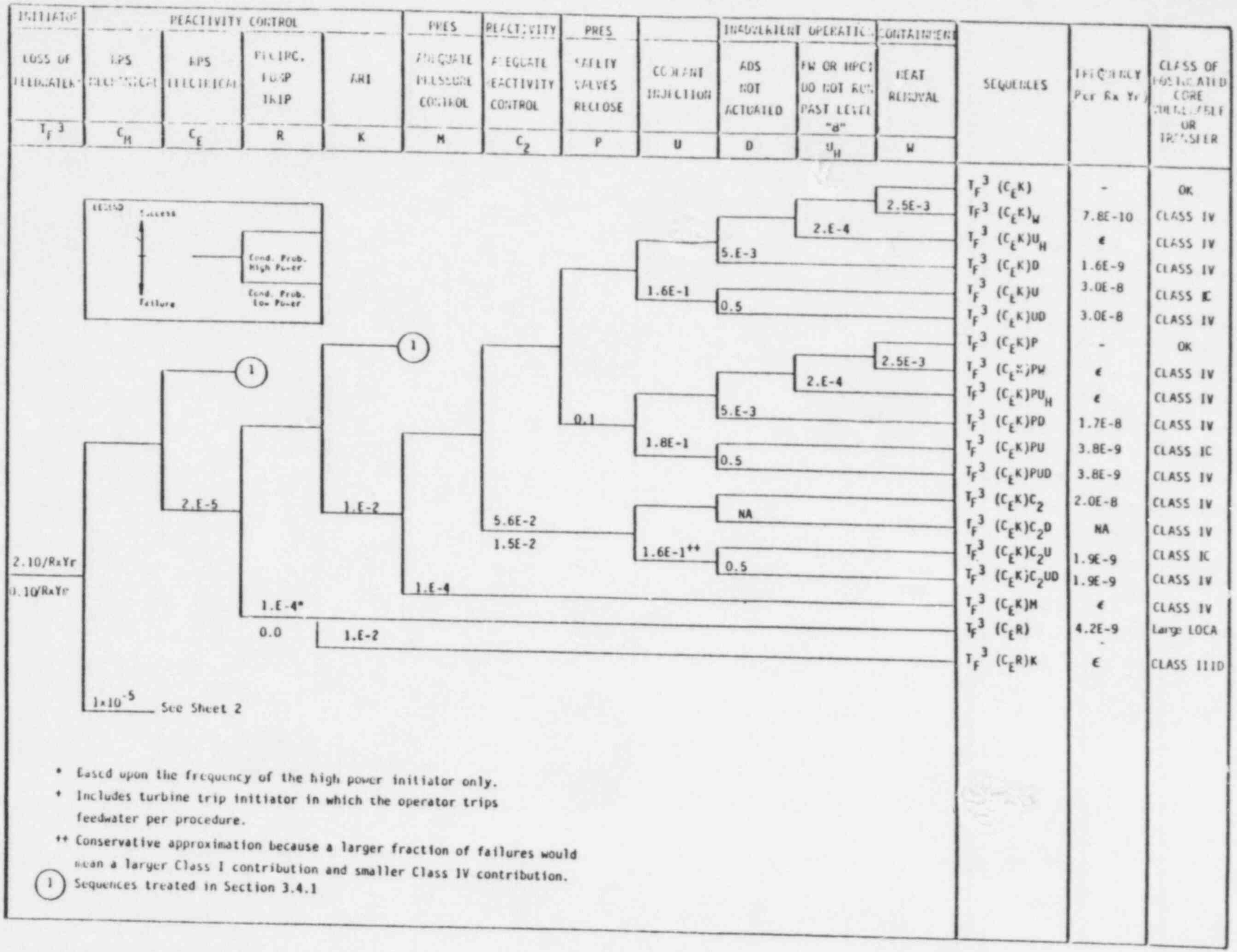


Figure 5D.4 Event Tree Diagram for Postulated ATWS Accident Sequence Following a Loss of Feedwater (Sheet 1 of 2)

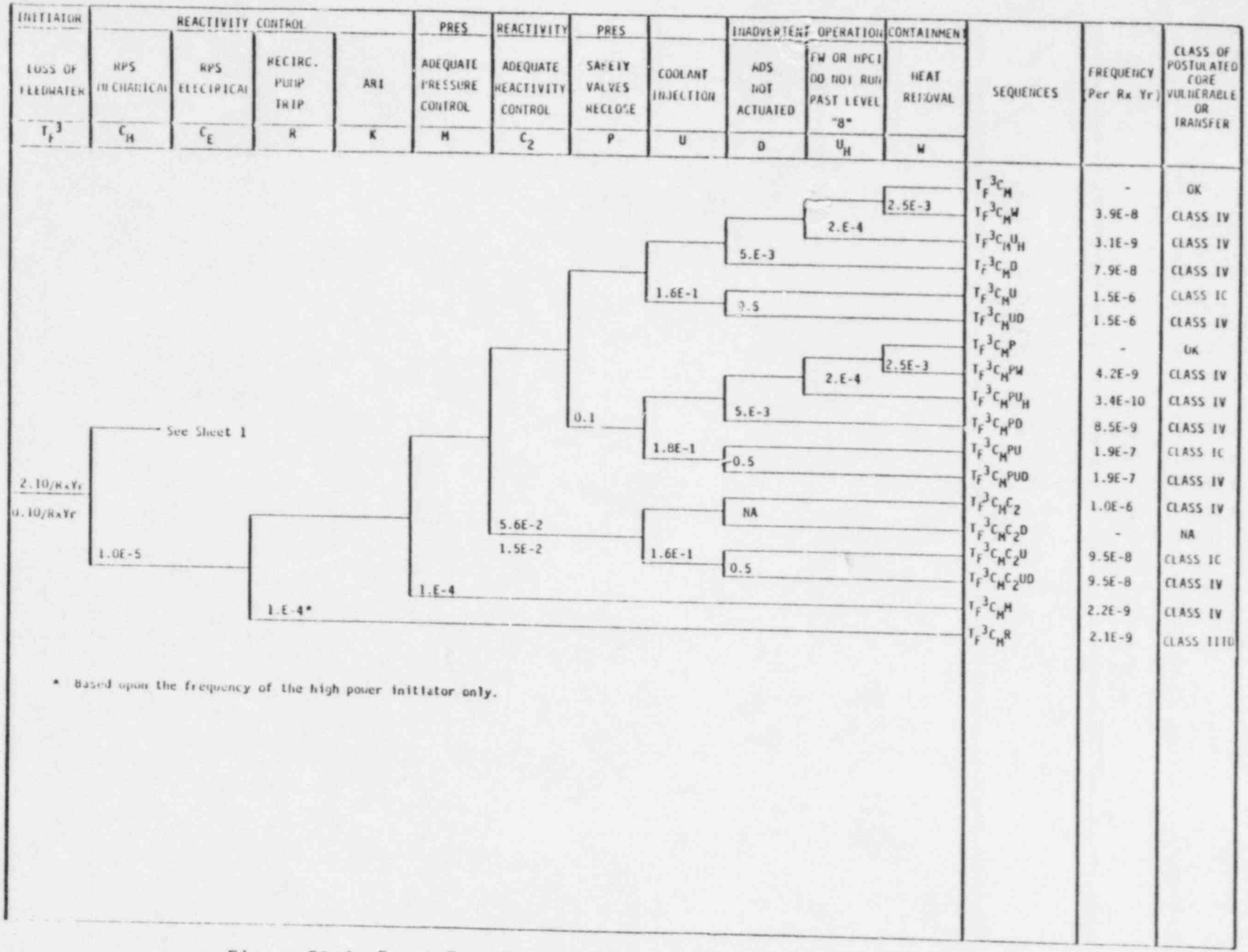


Figure 5D.4 Event Tree Diagram for Postulated ATWS Accident Sequence Following a Loss of Feedwater (Sheet 2 of 2)

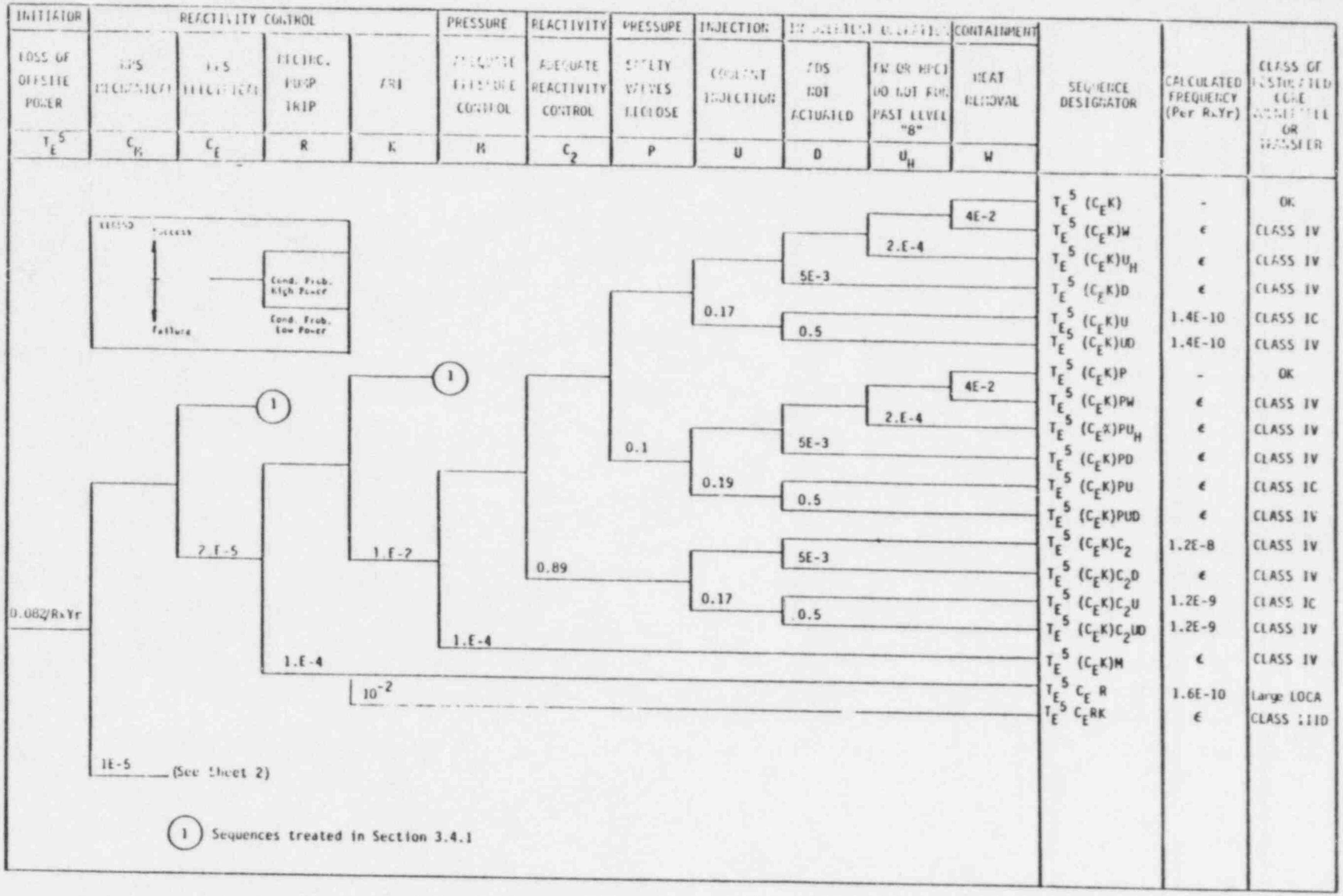


Figure 5D.5 Event Tree Diagram for Postulated ATWS Accident Sequence Following a Loss of Off-Site Power (Sheet 1 of 2)

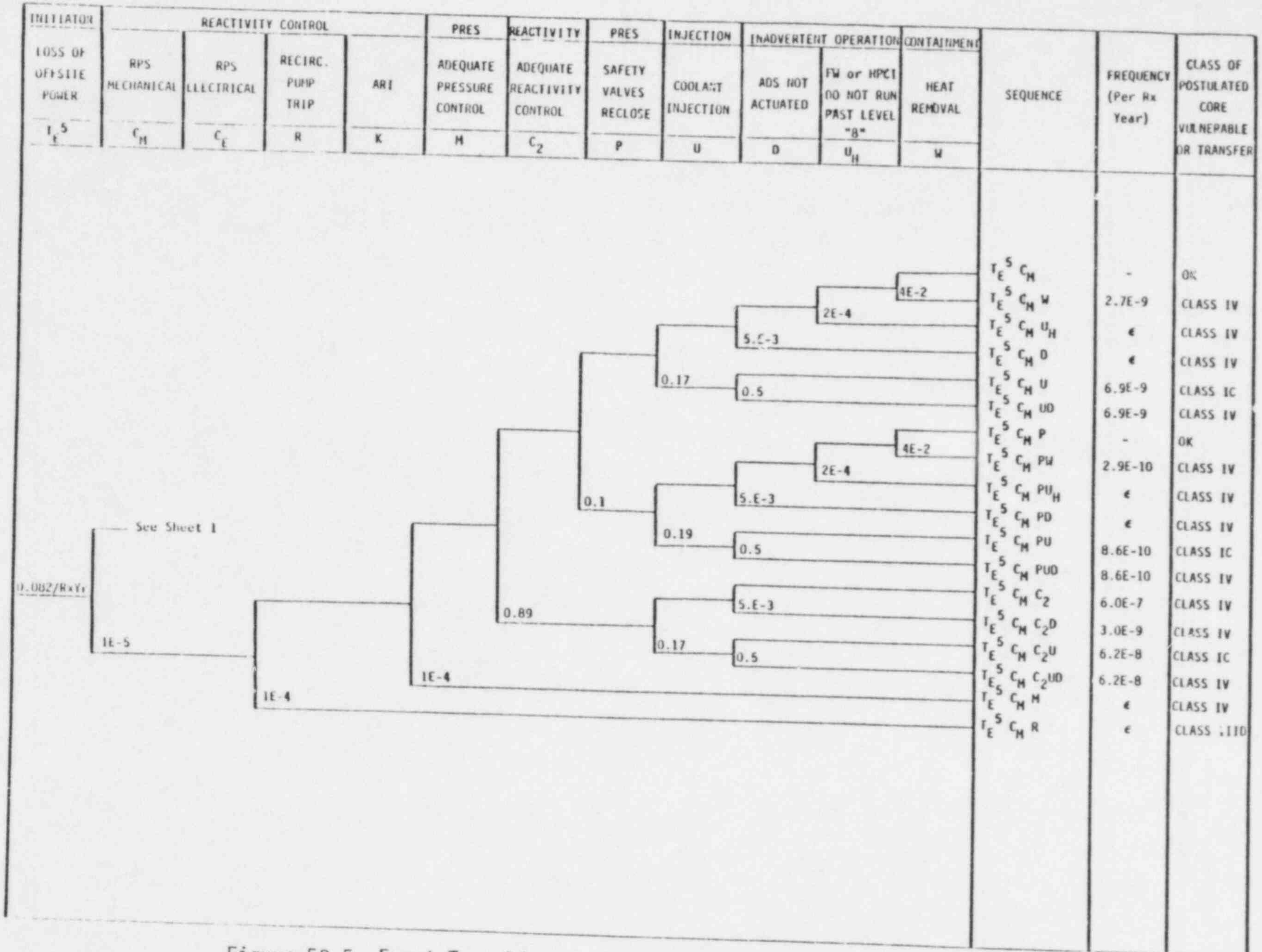


Figure 5D.5 Event Tree Diagram for Postulated ATWS Accident Sequence Following a Loss of Off-site Power (Sheet 2 of 2)

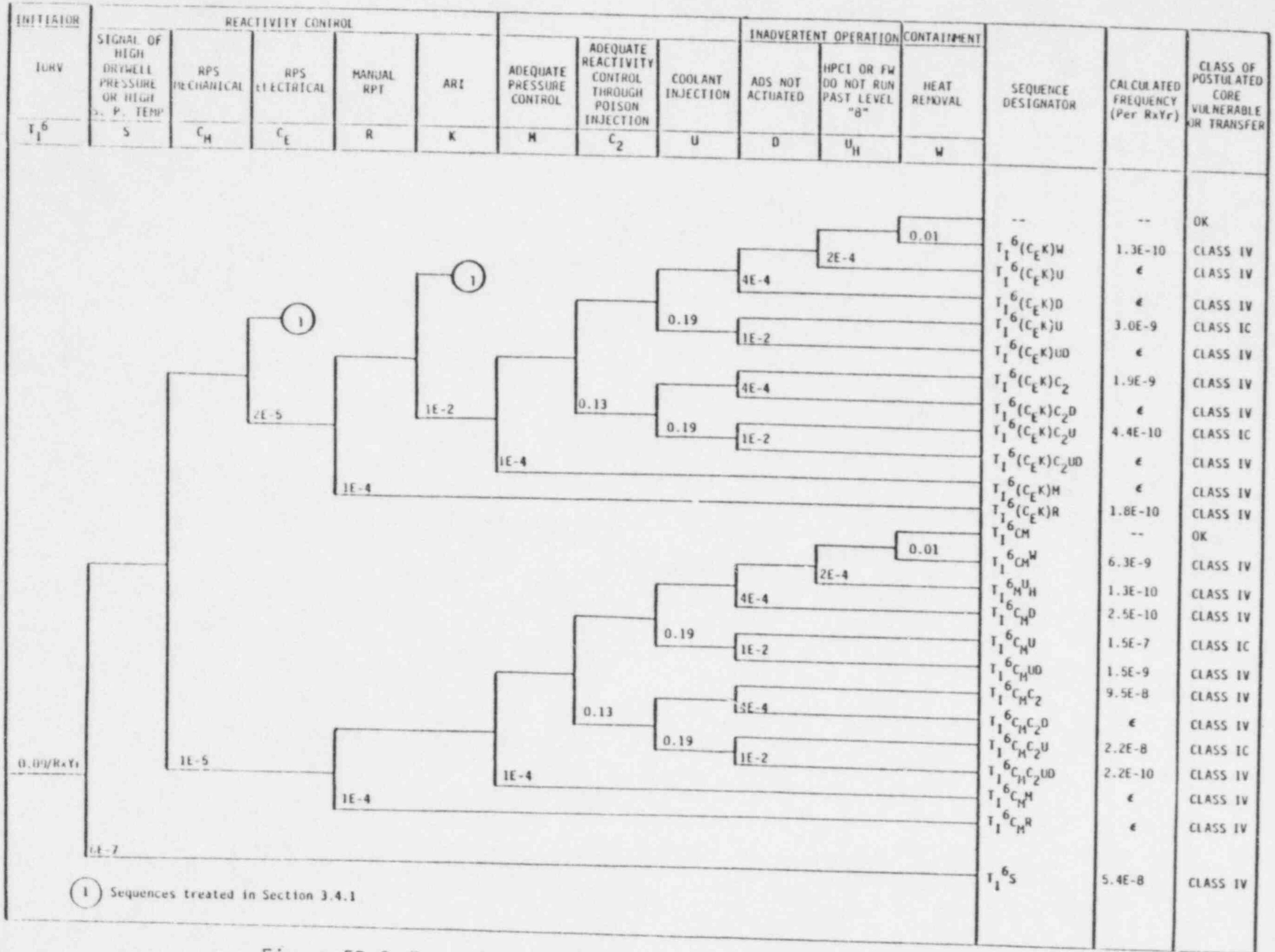


Figure 5D.6 Event Tree Diagram of Postulated ATWS Accident Sequence Following an Inadvertent Open Relief Valve

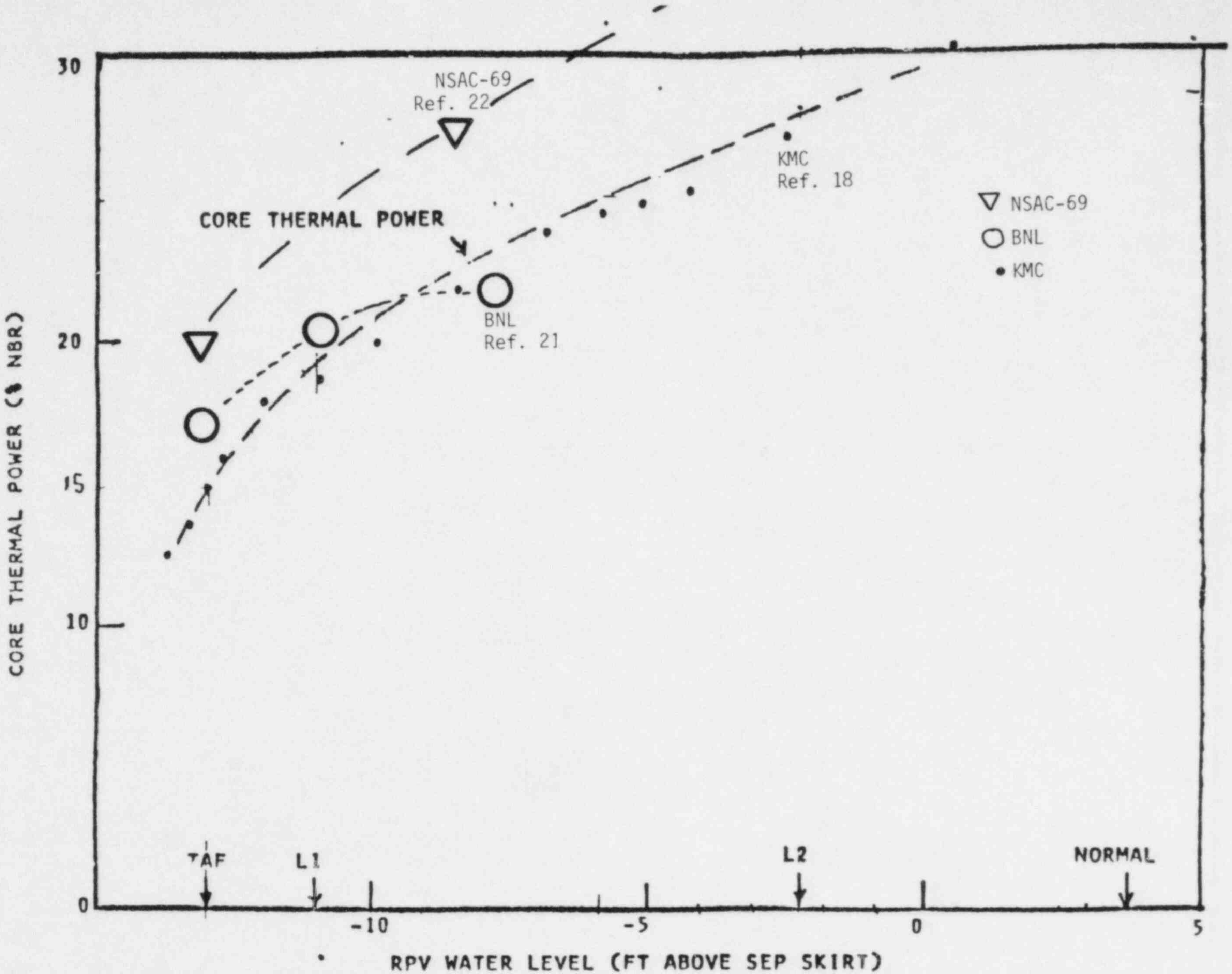


Figure 5D.7 Reactor Core Thermal Power vs RPV Water Level - Redy Estimate

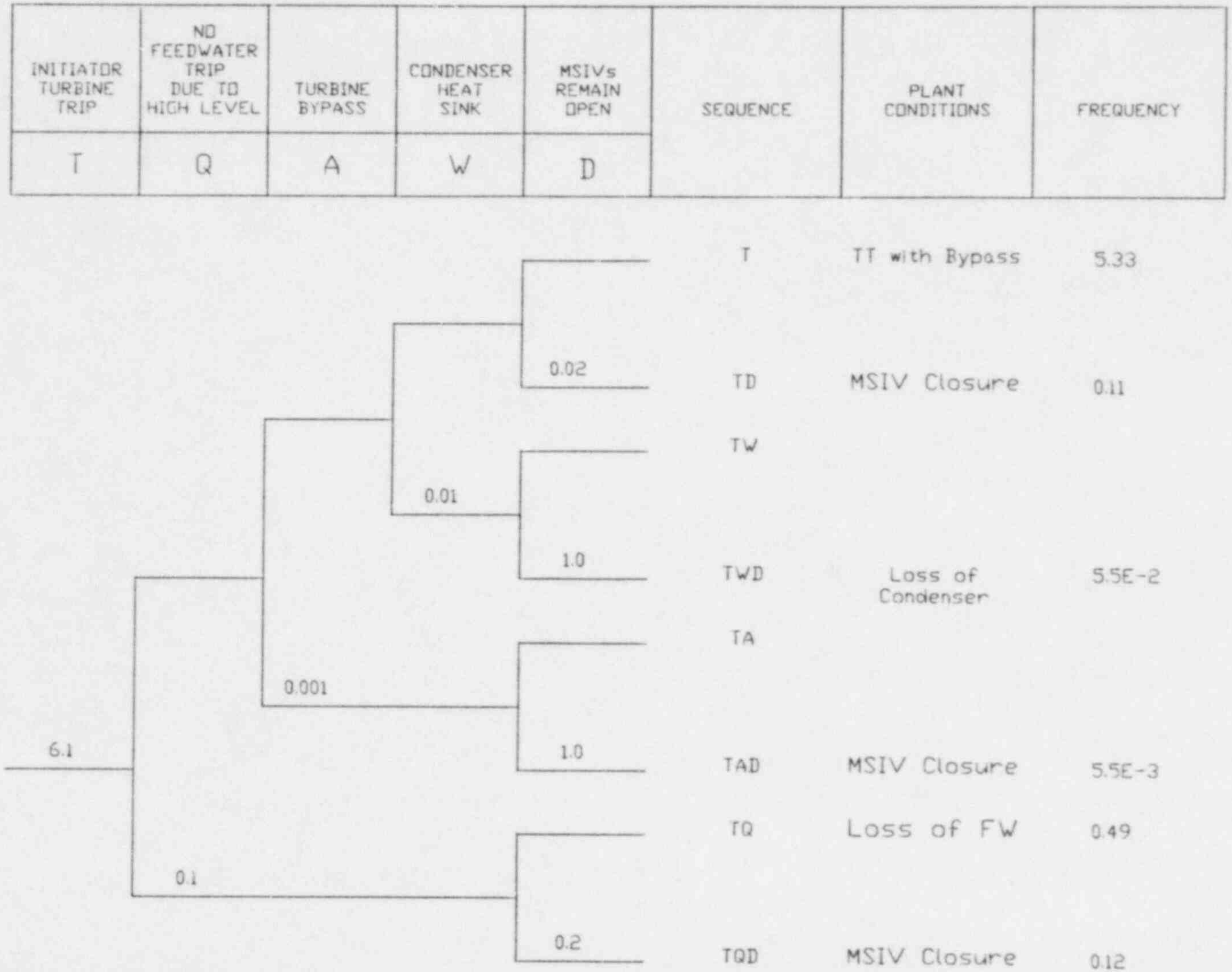


Figure 5D.8 Event Tree Diagram of Accident Sequences Following a Turbine Trip (BNL Review).

Table 5D.1 Transient with Failure to Scram Emergency Procedure

1. SYMPTOMS

1.1 A valid scram signal or condition due to a reactor transient is alarmed or indicated, and all control rods do not fully insert, as indicated on the full core display, the rod position printout on the computer, or the four-rod display.

1.2 Reactor pressure and/or neutron flux indication increases abruptly and may go off-scale on recorders and meters.

1.3 Safety relief valves may lift.

2. AUTOMATIC ACTIONS

2.1 1115 psig reactor vessel pressure and above actuates various safety relief valves. ________

2.2 1120 reactor vessel pressure TRIPS the reactor recirculation pumps. ________

3. IMMEDIATE OPERATOR ACTIONS

3.1 Manually scram reactor per SP 29.010.01 (Emergency Shutdown). ________

3.1.1 Arm and depress manual scram pushbutton. ________

3.1.2 Place the Mode switch in shutdown. ________

3.3.3 Verify all rods are inserted. ________

3.2 IF the reactor scrams AND all rods insert, AND power is decaying, THEN continue in SP 29.010.01. ________

3.3 Trip the recirculation pumps. ________

3.4 Commence suppression pool cooling per SP 23.121.01 (Residual Heat Removal (RHR) System). ________

3.5 The following attempts to scram the reactor are to be performed concurrently if manpower is available. ________

3.5.1 Insert those rods not fully inserted with the reactor manual control system as the Rod Sequence Control System (RSCS) permits. ________

Table 5D.1 Transient with Failure to Scram Emergency Procedure (Continued)

3.5.2 Bypass the scram discharge volume high level scram switches, reset the RPS trip, and verify the vent and drain valves open. ________

3.5.2.1 Alternately RESET the Reactor Protective System and SCRAM the reactor until all rods are fully inserted. ________

3.5.3 Confirm all scram valves are open by observation of scram valve position lights. IF not, THEN perform the following: ________

3.5.3.1 DE-ENERGIZE RP's subchannel logic by opening the following breakers on 1C71*PNL-001 in the relay room: ________

a) CB2A ________

b) CB2B ________

c) CB7A ________

d) CB7B ________

3.5.3.2 Vent air from the scram air system by closing valve C11-02V-0704 and opening vent valve downstream of C11-01V0-7104. ________

3.5.3.3 Restore the breakers and air valves to normal when all scram valves are open. ________

3.5.4 Bypass the scram discharge volume (SDV) high level scram switches, reset the RPS trip, and verify the vent and drain valves open. ________

3.5.4.1 INDIVIDUALLY SCRAM Control Rods at Local Hydraulic Control Units (HCU's) by placing both NORM-TEST-S.R.I. switches to the TEST position. ________

3.6 IF reactor power is above 6% OR RPV level cannot be maintained OR suppression pool temperature reaches 110°F, THEN perform the following. ________

3.6.1 Start either A or B standby liquid control pump and inject the entire contents of the tank. ________

3.6.1.1 IF RWCU automatic isolation did not occur, THEN manually isolate RWCU. ________

Table 5D.1 Transient with Failure to Scram Emergency Procedure (Continued)

3.6.1.2 Terminate all injection into the RPV with the exception of CRD and RCIC or HPCI to maintain RPV water level above the top of active fuel (TAF). ________

4. <u>SUBSEQUENT OPERATOR ACTION</u>

4.1 Verify immediate operator actions. ________

4.2 IF reactor pressure is causing the safety relief valves (SRV's) to cycle, THEN perform the following. ________

4.2.1 Manually open enough SRV's to reduce reactor pressure to between 800 and 960 psig. ________

4.2.2 For subsequent SRV operation, the valves should be cycled in order to minimize local heat loading of the suppression pool. ________

4.2.3 If the HPCI system is not in service, it may be placed in full flow test to minimize SRV cycling. ________

Table 5D.2 BNL ATWS Initiator Frequency

ATWS Initiator	SNPS+	0-100% Power Transient Frequency	25-100% Power Transient Frequency	Transfer*	BNL ATWS Frequency++
Turbine Trip	.85	8.02	6.11	--	5.33
MSIV	.50	.57	0.41	.24	.65
Loss of Condenser	.25	.50	0.41	.05	.46
Loss of Feedwater	2.10	.13	0.10	.49	.59
LOOP	.082	.15	0.15	0	.15
IORV	.09	.25	0.16	0	.16
Total	3.87	9.62	7.34	0.78	7.34

*Transfer from turbine trip initiator event tree, see Figure 5D.8.
+Shoreham PRA ATWS frequency from high power level, i.e., 25% power or more (after transfer from turbine trip initiator event tree, see Figure 5D.1).
++BNL review ATWS frequency from high power level, i.e., 25% power or more (after transfer from turbine trip initiator event tree, Figure 5D.8).

Table 5D.3 Comparison of Conditional Frequency of Core Damage Based on BNL and SNPS ATWS Event Trees

	SNPS-PRA	BNL
Turbine Trip	4.1(-6)	5.5(-6)
MSIV and Loss of Condenser	1.1(-5)	9.7(-6)
Loss of Feedwater	2.3(-6)	4.4(-6)
LOOP	9.3(-6)	9.6(-6)
IORV	3.7(-6)	4.8(-6)

Table 5D.4 Core Damage Frequency of BNL Revised ATWS Event Trees with SNPS Initiator Frequency

	Core Damage with SNPS Initiator Frequency			
	SNPS	BNL		
	Total	Class IV	ATWS LOCA	Total
Turbine Trip	3.5(-6)	4.7(-6)	3.4(-9)	4.7(-6)
MSIV and Loss of Condenser	8.3(-6)	7.2(-6)	2.9(-9)	7.2(-6)
Loss of Feedwater	4.8(-6)	9.3(-6)	8.2(-9)	9.3(-6)
LOOP	7.6(-7)	7.7(-7)	3.2(-10)	7.7(-7)
IORV	3.3(-7)	4.3(-7)	ε	4.3(-7)
	1.8(-5)			2.2(-5)

Table 5D.5 Core Damage Frequency Based on BNL Revised ATWS Event Tree with BNL Initiator Frequency

	Core Damage with BNL Initiator Frequency			
	SNPS	BNL		
	Total	Class IV	ATWS LOCA	Total
Turbine Trip	3.5(-6)	2.9(-5)	2.1(-8)	2.9(-5)
MSIV and Loss of Condenser	8.3(-6)	1.1(-5)	4.4(-9)	1.1(-5)
Loss of Feedwater	4.8(-6)	2.6(-6)	2.0(-9)	2.6(-6)
LOOP	7.6(-7)	1.4(-6)	5.8(-10)	1.4(-6)
IORV	3.3(-7)	7.7(-7)	ε	7.1(-7)
	1.8(-5)			4.5(-5)

ε = Less than 10(-10)

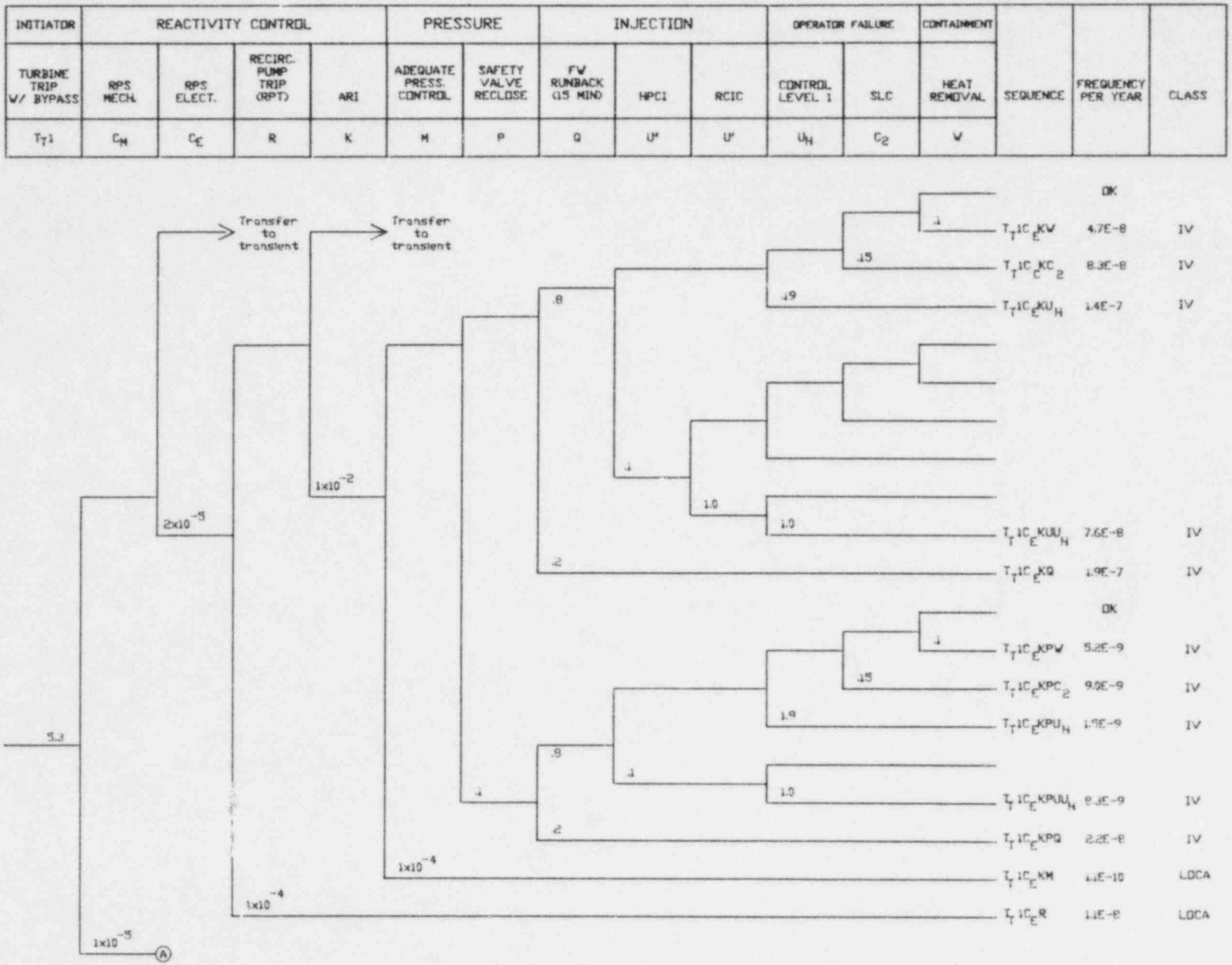


Table 5D.6 Event Tree Diagram for Postulated ATWS Sequence Following Turbine Trip (Sheet 1 of 2)

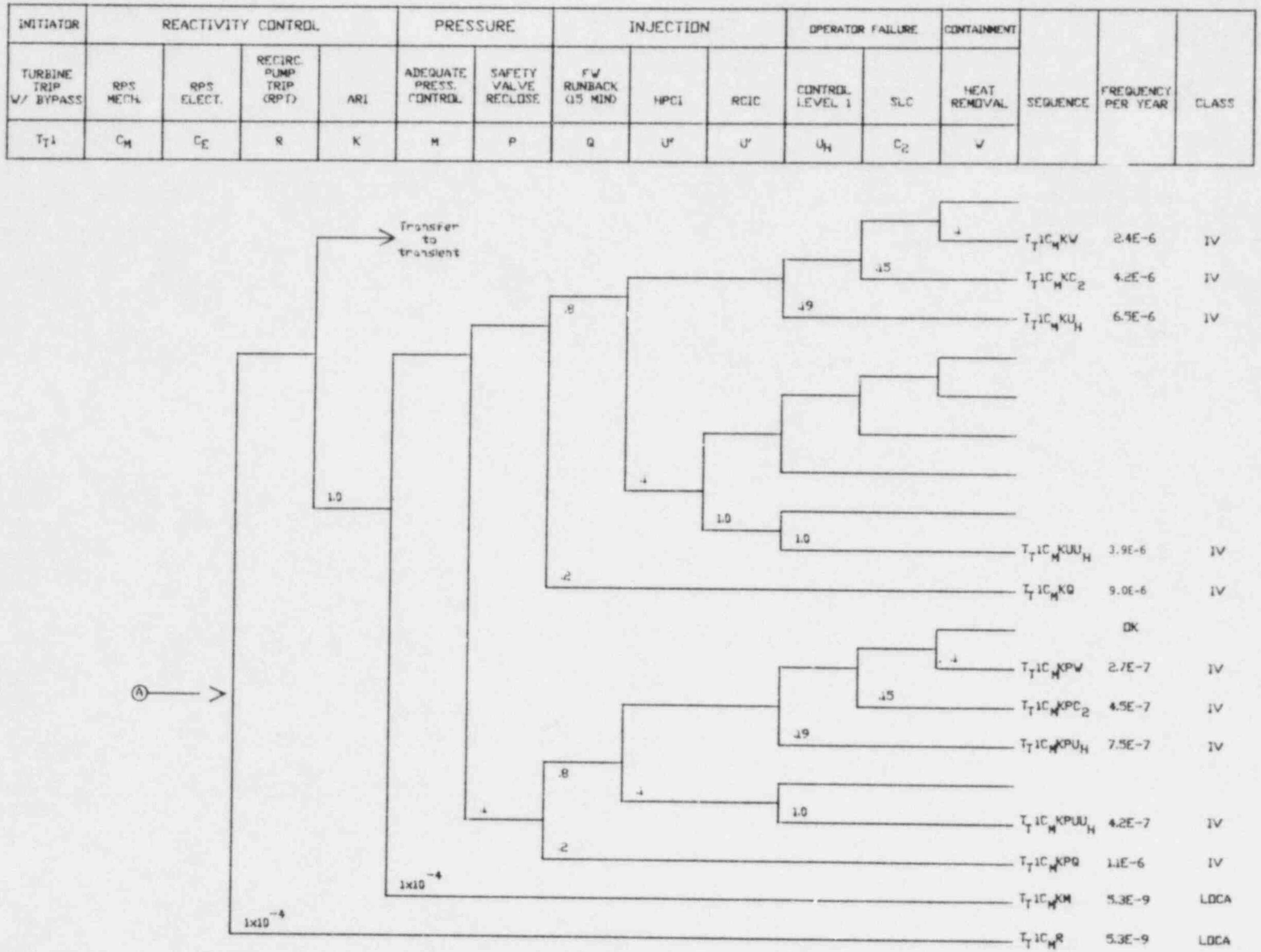


Table 5D.6 Event Tree Diagram for Postulated ATWS Sequence Following Turbine Trip (Sheet 2 of 2)

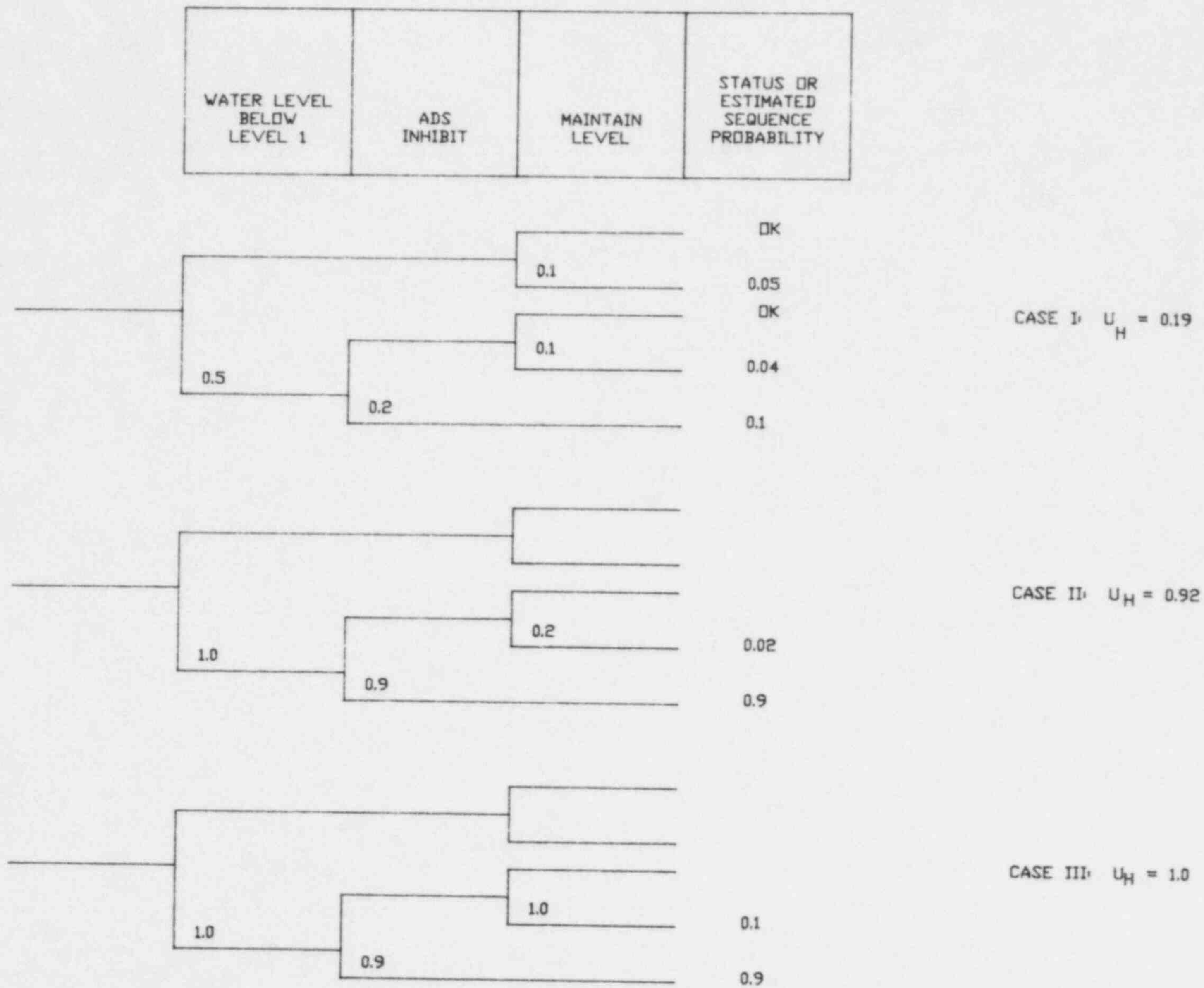


Table 5D.7 Functional Level Event Tree for the Control of RPV Level-1

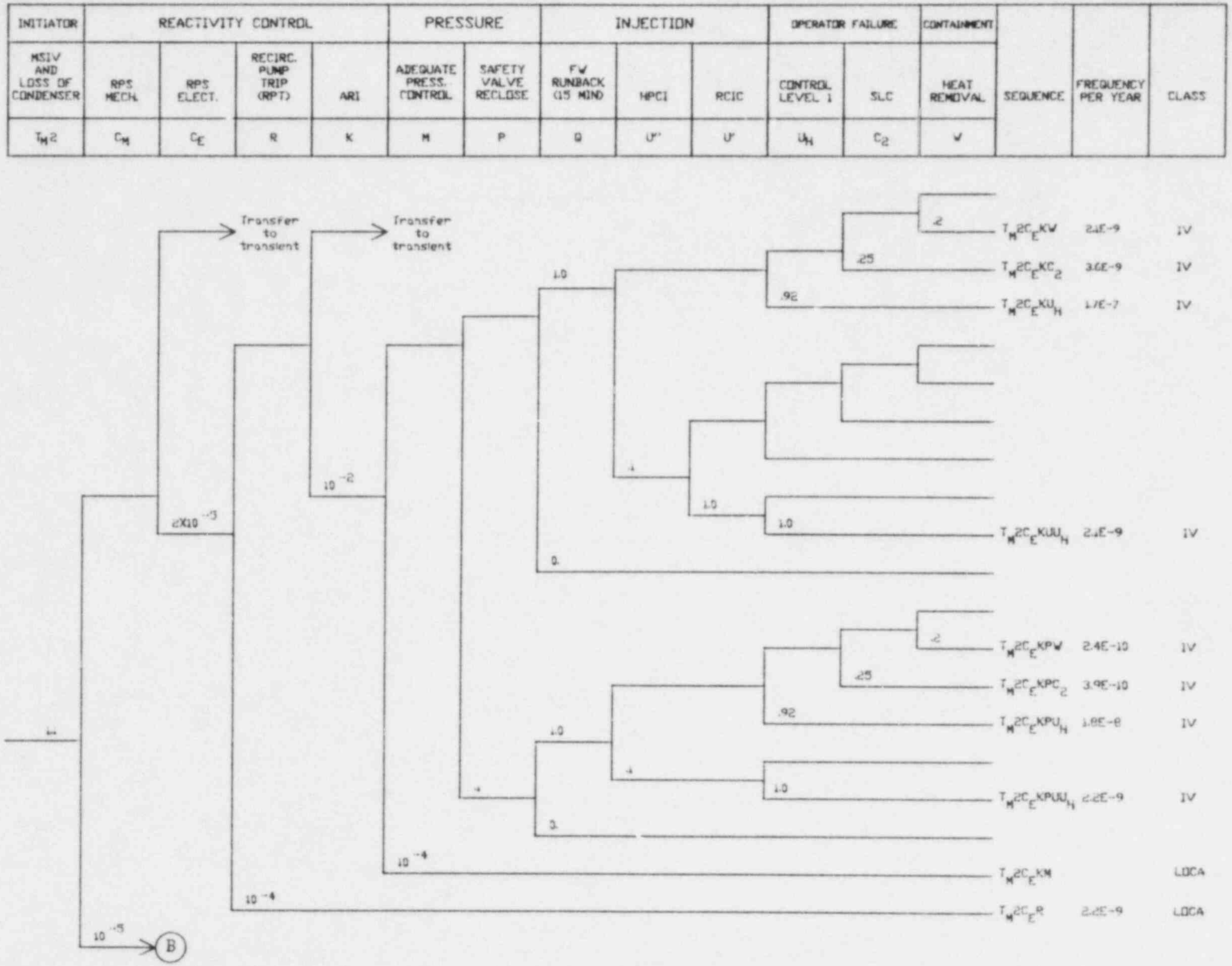


Table 5D.8 Event Tree Diagram for Postulated ATWS Sequences Following MSIV Closure (Sheet 1 of 2)

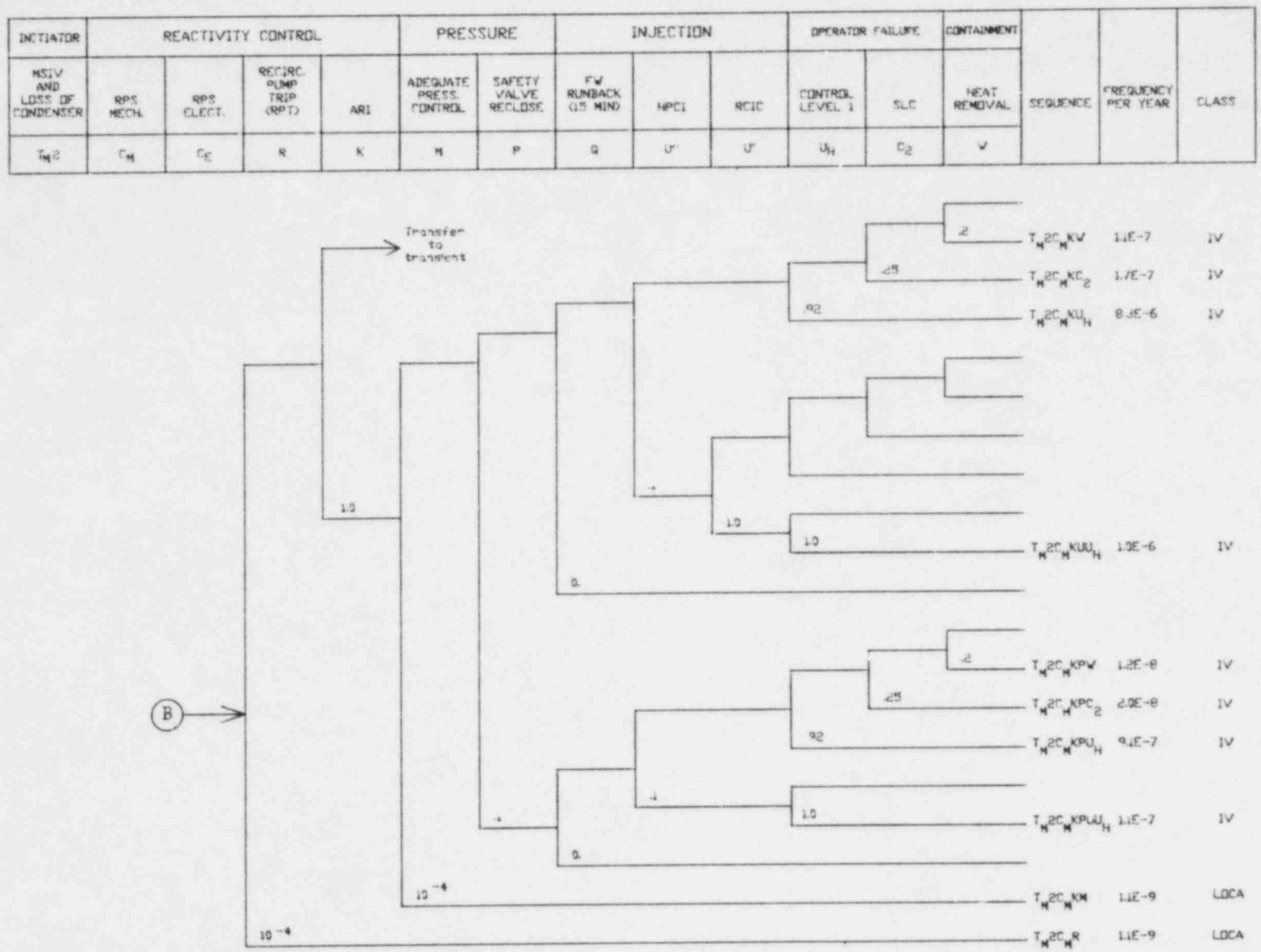


Table 5D.8 Event Tree Diagram for Postulated ATWS Sequences Following MSIV Closure (Sheet 2 of 2)

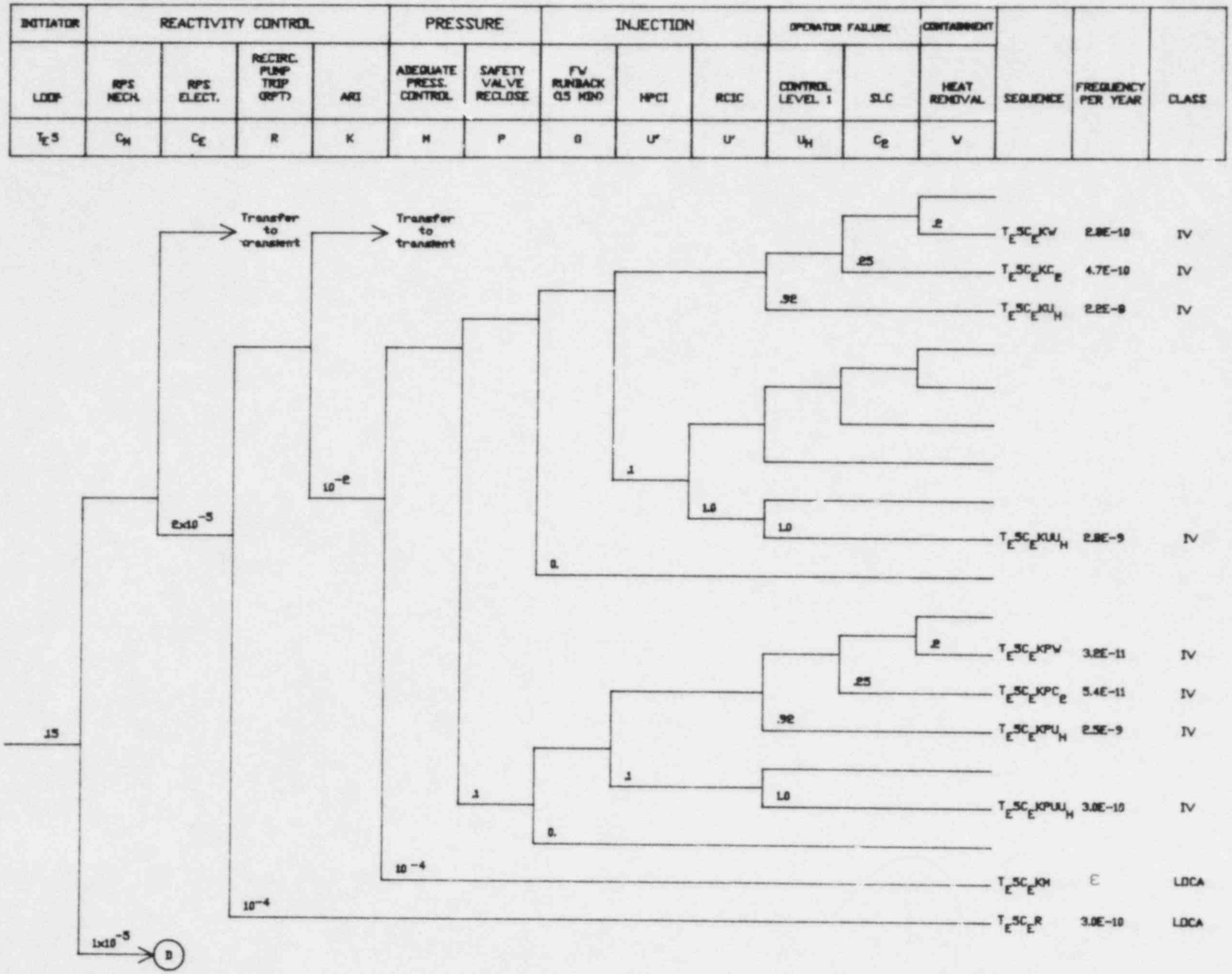


Table 5D.9 Event Tree Diagram for Postulated ATWS Sequences Following LOOP (Sheet 1 of 2)

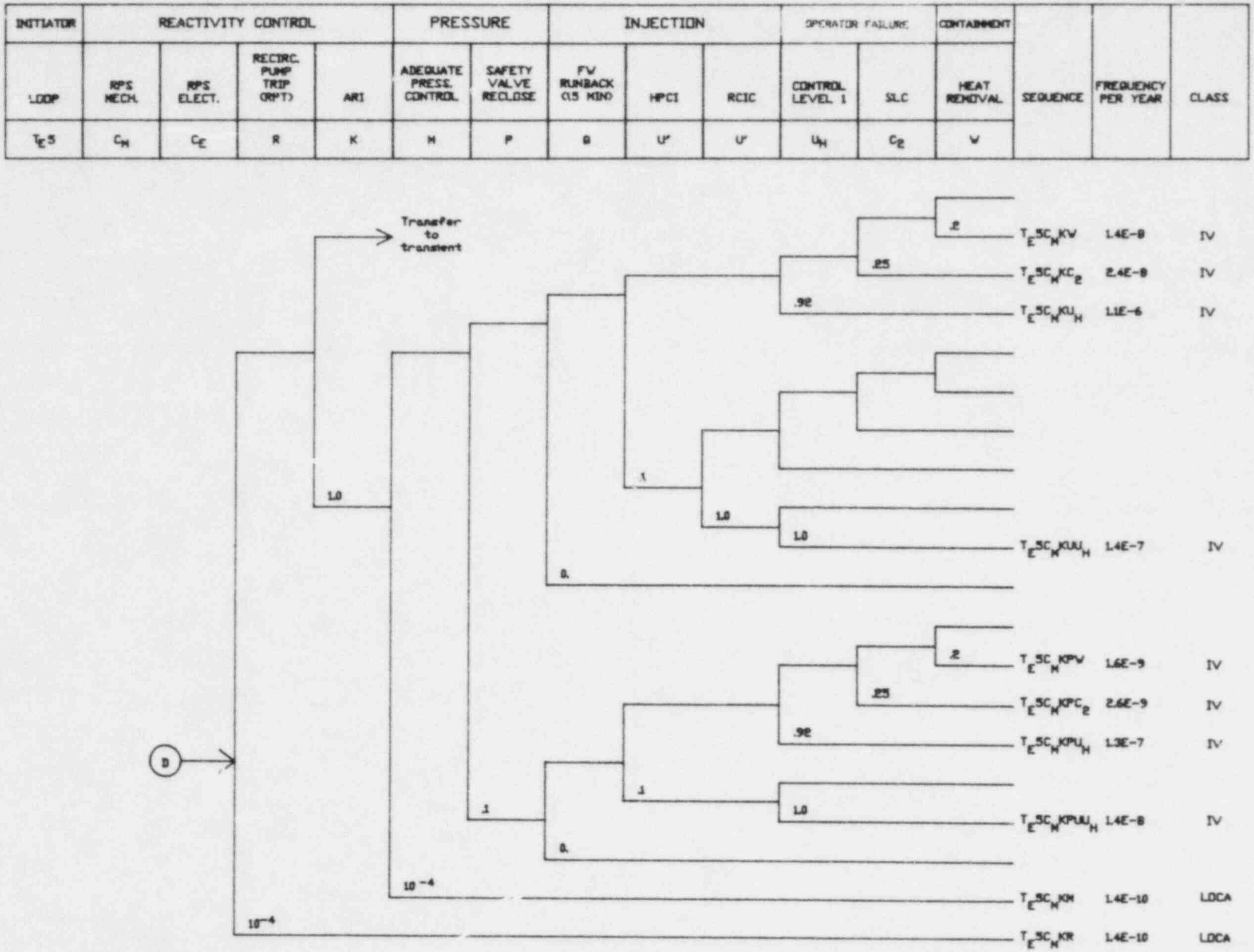


Table 5D.9 Event Tree Diagram for Postulated ATWS Sequences Following LOOP (Sheet 2 of 2)

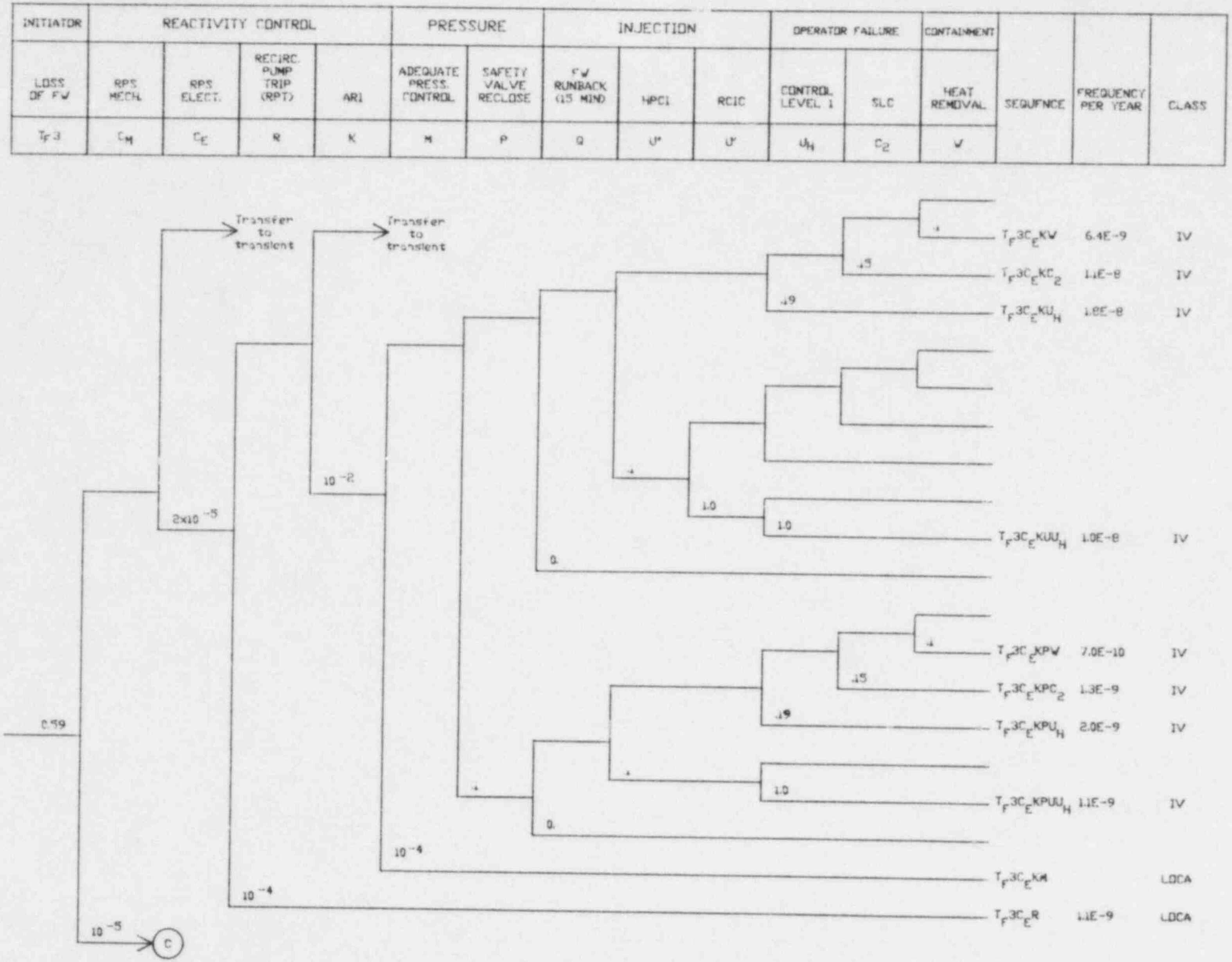


Table 5D.10 Event Tree Diagram for Postulated ATWS Sequences Following Loss of FW (Sheet 1 of 2)

INITIATOR	REACTIVITY CONTROL				PRESSURE		INJECTION			OPERATOR FAILURE		CONTAINMENT	SEQUENCE	FREQUENCY PER YEAR	CLASS
LOSS OF FW	RPS MECH.	RPS ELECT.	RECIRC. PUMP TRIP (RPT)	ARI	ADEQUATE PRESS. CONTROL	SAFETY VALVE RECLOSE	FW RUNBACK (15 MIN)	HPCI	RCIC	CONTROL LEVEL 1	SLC	HEAT REMOVAL			
T_F3	C_M	C_E	R	K	M	P	Q	U'	U'	U_H	C_2	W			

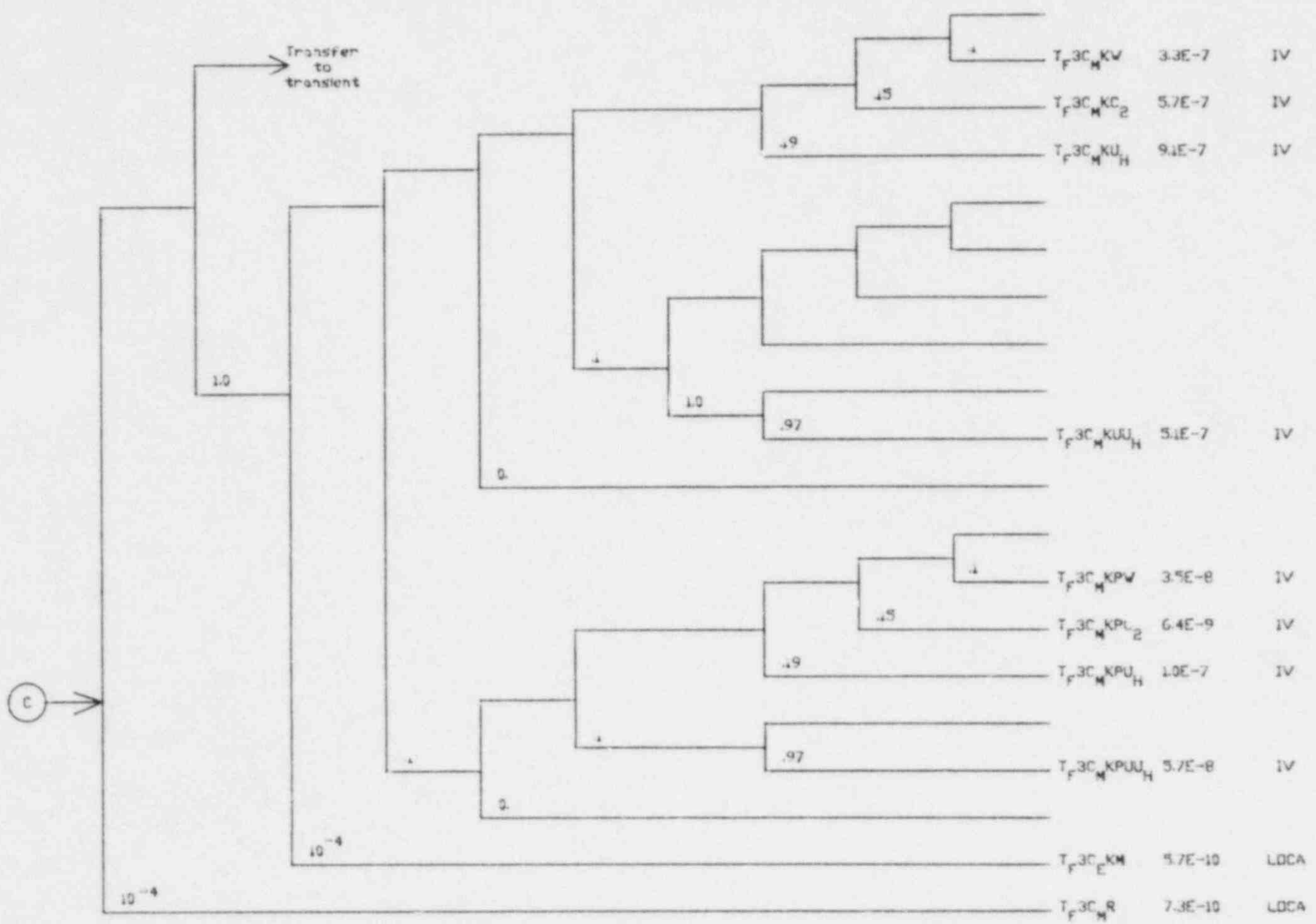


Table 5D.10 Event Tree Diagram for Postulated ATWS Sequences Following Loss of FW (Sheet 2 of 2)

INITIATOR	REACTIVITY CONTROL					PRESSURE		INJECTION			OPERATOR FAILURE		CONTAINMENT			
IORV	SIGNAL HI DW PRESS. HI S.P. TEMP.	RPS MECH.	RPS ELECT.	RECIRC. PUMP TRIP (RPT)	ARI	ADEQUATE PRESS. CONTROL	SAFETY VALVE RECLOSE	FW RUNBACK (15 MIN)	HPCI	RCIC	CONTROL LEVEL 1	SLC	HEAT REMOVAL	SEQUENCE	FREQUENCY PER YEAR	CLASS
$T_1 6$	S	C_M	C_E	R	K	M	P	Q	U'	U'	U_H	C_2	W			

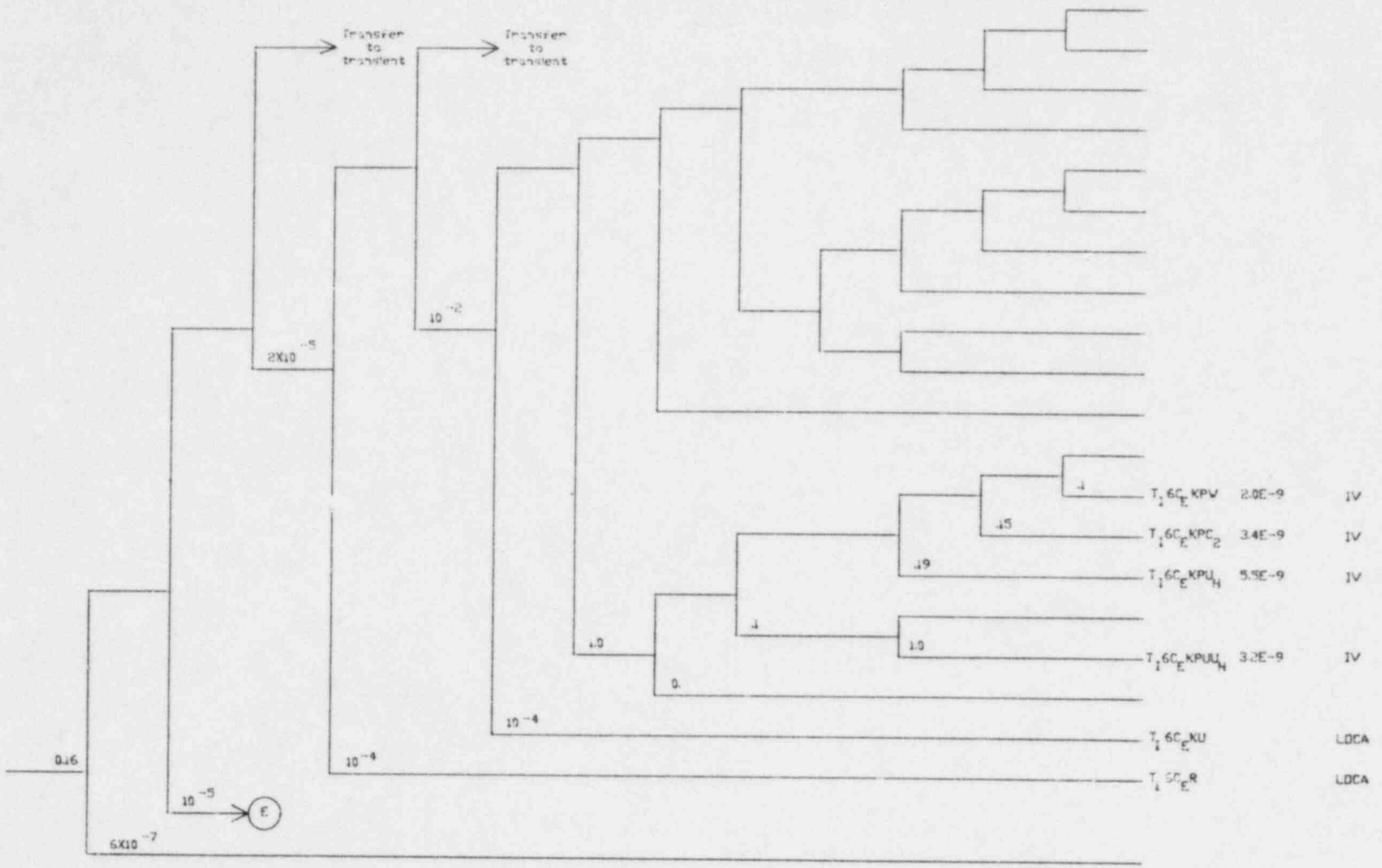


Table 5D.11 Event Tree Diagram for Postulated ATWS Sequences Following IORV (Sheet 1 of 2)

INITIATOR	REACTIVITY CONTROL					PRESSURE		INJECTION			OPERATOR FAILURE		CONTAINMENT	SEQUENCE	FREQUENCY PER YEAR	CLASS
IORV	SIGNAL HI DW PRESS. HI S.P. TEMP.	RPS MECH.	RPS ELECT.	RECIRC. PUMP TRIP (RPT)	ARI	ADEQUATE PRESS. CONTROL	SAFETY VALVE RECLOSE	FW RUNBACK (15 MIN)	HPCI	RCIC	CONTROL LEVEL 1	SLC	HEAT REMOVAL			
T_I6	S	C_M	C_E	R	K	M	P	Q	U"	U'	U_H	C_2	W			

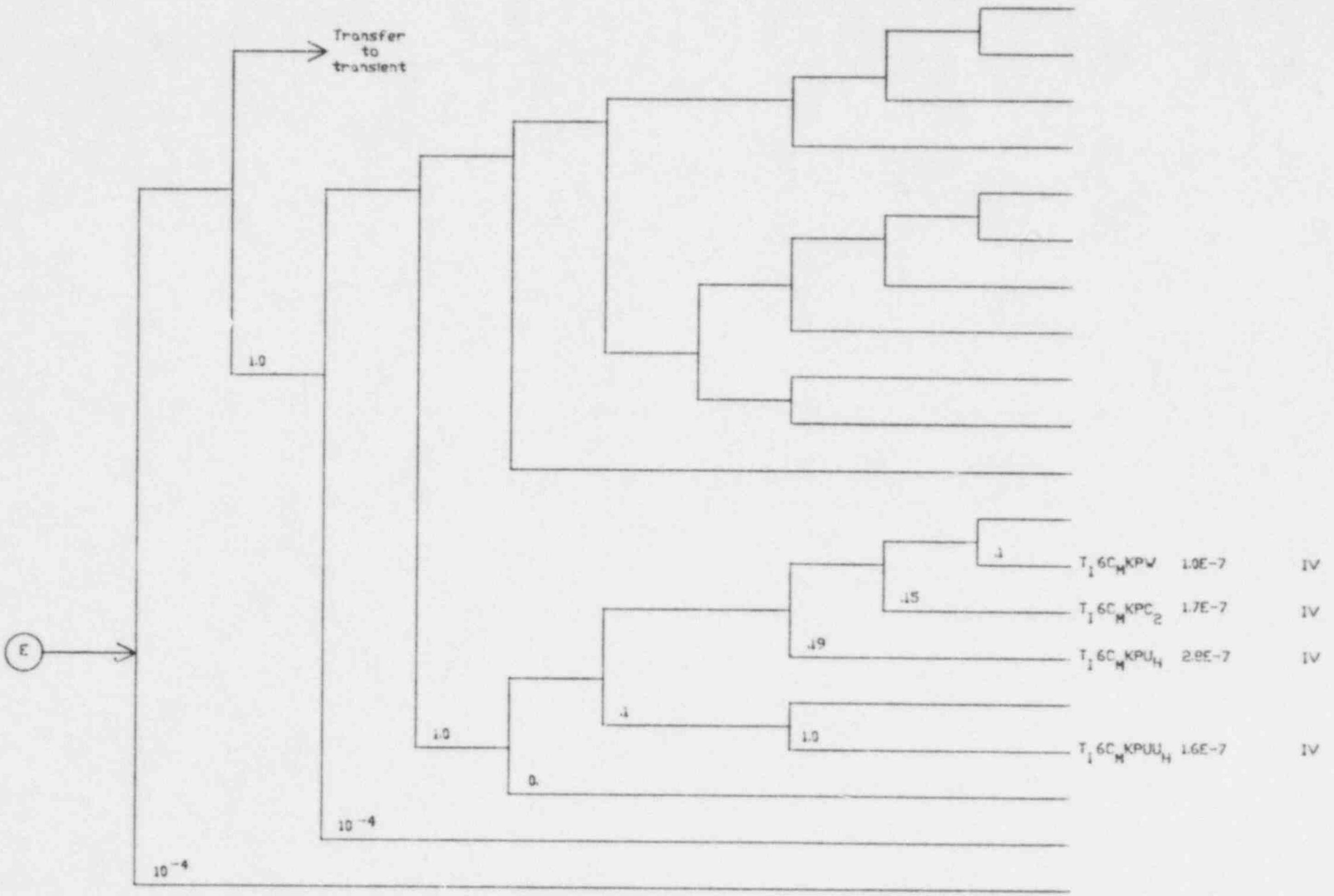


Table 5D.11 Event Tree Diagram for Postulated ATWS Sequences Following IORV (Sheet 2 of 2)

APPENDIX 5E

REACTOR WATER LEVEL INSTRUMENT LINE FAILURE

5E.1 BACKGROUND

The BNL review of the contribution of reactor water level instrument failure to SNPS core damage frequency was an evaluation of the SNPS study SLI 8221[2]. The accident sequences progressing from a loss of a reference leg initiator were found to be an important contribution to SNPS core damage frequency.

The SNPS study consisted of the following:

a) A detailed description of the water level measurement system.

b) Measurement error possibilities due to variation in plant condition or because of instrument line flashing.

c) Vulnerability of the water level measurement system to combinations of several modes of failure, such as the failure of a reference leg by leakage with either subsequent loss of a DC bus, or additional random failure of a level measurement channel, or an additional maintenance error.

d) Description of the sequences leading to core damage and their quantification.

e) For item (d), LERs and human error studies specific to reference leg leakage sequences in the SNPS, taking into account pertinent SNPS information on control room display and annunciators.

The design of the water measurement system* includes the following unique features (see Figure 5E-1 and Table 5E-1):

a) Two reference legs, side A and side B, for all safety instrumentation.

b) Two DC buses (Division I and II), each feeding one side** of the instrumentations (side A - Division I and side B - Division II).

*After the BNL review was completed, BNL was informed that Shoreham is going through a set of modifications to the Reactor Water Level Instrumentation system. The most important one is the addition of four new level transmitters No. xx-A, B, C, D. The initiation for HPCI will be separated from the initiation of RCIC, ADS, and LPCI/LPCS. In addition, HPCI will be associated with DC bus B only and RCIC only with DC-A.

**With the inclusion of the additional four level transmitters, their feed from the DC buses was rearranged so that DC-A and DC-B feed transmitters on both side A and side B.

c) RCIC and HPCI share* their automatic actuation on level 2, originating from the wide-range instrumentations N091A, N091B, N091C, N091D, having a 1 out of 2 twice logic. (A and C are on side A, B and D are on side B).

d) ADS, LPCI, and LPCS share their automatic initiation on level 1, coming from the same four instrumentations providing for level 2.

e) Control room level indications are received from several other level transmitters (N081A, C, D - Wide Range; N004A, B, C - Narrow Range; N037A, B - Fuel Zone Range). However, the transmitters are fed from AC buses, and apparently only N004A narrow-range information is fed from a vital bus having a DC backup.

f) Feedwater control is normally on level transmitter N004A (leg A) and occasionally (10%) on transmitter N004B (leg B).

g) Turbine trip is received when 2 out of 3 transmitters reach level 8. The transmitters are N004A and C on side A and N004B on side B. Therefore, turbine trip can result when leg A fails, but not when leg B fails. However, in case of leg B feedwater control when leg B fails, a runback to shutdown will apparently occur. In the case that feedwater control is on reference leg A and failure of reference leg B occurs, the reactor may continue its power operation.

The SNPS study analyzed a number of transients, all resulting from a reference leg leakage, with an additional failure caused by one of the following:

a) Operator error, causing the second reference leg to leak due to a maintenance error (significant contributor).

b) Loss of a single DC bus (affecting the other leg) (small contributor).

c) Miscalibration of instrumentation on the other leg (significant contributor).

d) Random failure of additional instrument channels (three different cases, one significant and two minor contributors).

Each of the above sequences is discussed separately, and developed on a separate event tree, as shown in Table 5E.2 Sheets 1 through 9.

The major differences between the SNPS analysis and the BNL reassessment are the following:

a) Several events are treated explicitly on the BNL event trees, rather than on functional fault trees as in the SNPS-PRA.

b) The LER failure data on the loss of one reference leg are used to calculate the probability of a second reference leg failure due to maintenance error.

*See comment on previous page.

c) Additional cases are treated separately, such as miscalibration and the mechanical failure of a differential pressure cell.

These changes provided a more realistic analysis, in BNL's view, and resulted in a higher core damage frequency.

5E.2 OPERATOR ERROR CAUSING LEAK ON THE SECOND REFERENCE LEG

This is shown in the BNL event tree (Table 5E.2 Sheet 4). The frequency used in the BNL assessment is based on the revised functional fault tree (Figure 5E.2). This tree is based on the available LER data given in the PRA which includes two events of reference leg failure during maintenance when the reactor was at power operations. Appendix B of the SNPS study of water level instrumentation[2] evaluates the value for the "error rate for maintenance errors during power operation" on the basis of two LER events; the result is 0.000985 event/maintenance. This value is assumed by BNL to account for recovery, because reported LERs are generally events which have resulted in failure and were not immediately recovered.

Using the value derived above, the SNPS-PRA functional fault tree (Figure D.2 of Ref. 2) was revised by BNL as shown in Figure 5E.2. In addition, the value for operator error utilizing repeatedly faulty procedures was judged low and was increased by BNL by a factor of 3 above the SNPS-PRA value. The result $O_R = 1.9 \times 10^{-4}$, which is sevenfold higher than that calculated by the SNPS-PRA, is used in the BNL event tree (Table 5E.2 Sheet 4). The other details of this event tree are similar to those in the SNPS-PRA. This sequence ($T_R R O_R QUX = 2 \times 10^{-6}$ in Class I) became a significant contributor in the BNL reassessment, in contrast to the SNPS-PRA, because of the change described above. This sequence is associated with the use of two reference legs in the SNPS design.

5E.3 LOSS OF A SINGLE DC BUS

The following changes were made in these sequences:

a) A contribution from the loss of a DC bus during power operation with failed reference leg B was added by BNL (compare SNPS' Fig. 3.4-45 Sheet 2 with BNL's Table 5E.2 Sheet 3, branch "B_R"). The amount transferred from the "power operation subtree" to the loss of a DC bus branch is dependent on the time allowed for continuous operation, which is assumed to be 24 hours (should be part of technical specifications). However, the results are not sensitive to the time length assumed.

b) The event Timely Reactor Depressurization X is shown on Fig. 3.4-45 Sheet 4 of the SNPS-PRA. BNL assessed that this should appear before the injection function, because the latter cannot start automatically and thus requires an operator decision at a previous stage. If the operator recognizes the situation correctly when he realizes that high pressure injection did not start, he will manually start HPCI or RCIC and proceed to depressurization only if the high pressure injection fails. However, if the operator fails to recognize the situation and chooses the wrong procedure, or waits too long, then core uncovery will occur before the injection phase starts. SNPS treated

this failure as conditional on HPCI or RCIC failure, which underestimates the sequence potential contribution to core damage frequency by a factor of 10. The operator action tree used by the SNPS study was used, unchanged, in the BNL reassessment (Figure 5E.3).

Even with the two changes described above, this sequence remains unimportant. The modifications to SNPS Water Level Instrumentation will pote ially eliminate this sequence.

5E.4 MISCALIBRATION OF WATER LEVEL INSTRUMENTATION ON HE ALTERNATE LEG

The progression of this event is similar to loss of a DC bus, in creating a situation in which no automatic initiation will occur following the transient. Thus, this sequence is described in a similar way as that for the case of the loss of a DC bus. This sequence is not explicitly modeled in the SNPS-PRA, where a failure of instrument channel is treated as one lumped case and much significant detail is lost. The SNPS-PRA lumped failure rate of 0.016 is comprised of four contributors:

$$\text{Miscalibration error} = 2\text{x}10^{-3}$$

$$\text{Two differential pressure cells} = 2\text{x}(4.4\text{x}10^{-3})$$

$$\text{Two relays and slaves for level 1} = 2\text{x}(1.2\text{x}10^{-3})$$

$$\text{Two relays and slaves for level 2} = \underline{2\text{x}(1.2\text{x}10^{-3})}$$

$$\text{Total} \quad 1.56\text{x}10^{-2} = 0.016$$

The SNPS-PRA treated these together in Fig. 3.4-45 Sheet 5, but BNL treated each element separately, which resulted in an increase in core damage frequency, as shown in Table 5E.2 Sheets 6-9. However, the use of the same operator error probability from Fig. 5E.3 is somewhat conservative, because in the case of miscalibration, live indications of level are not lost as in the case of loss of a DC bus.

Note that the miscalibration error of $2\text{x}10^{-3}$ is used here for a case of the miscalibration of two channels (rather than all four channels as discussed in Section 5A.1.4) and is considered reasonable for this case. Note that the proposed modification will practically eliminate this sequence.

5E.5 FAILURE OF DIFFERENTIAL PRESSURE CELL

BNL modeled this separately in Table 5E.2 Sheet 7 and found a significant contribution. The result is due to the BNL approach, in which the operator faces a decision immediately before injection because injection does not occur automatically. In this case, however, the operator apparently has a higher probability of success in choosing the right response because, in addition to the level information from one wide-range recorder and two narrow-range indicators, he will receive alarms from level 2 and 1 annunciators of the remaining channel, which may alert him to the correct situation. Therefore, the error probability in this case is decreased to 0.013. This is based on the operator action decision tree (Figure C-9 in the SNPS study[2] and Figure 5E.3 of this appendix).

5E.6 FAILURE OF LEVEL 1 OR 2 RELAYS AND SLAVES

In this case, only part of the automatic initiation is lost, and therefore the plant response is not dependent on operator action, and the results are very low (Table 5E.2 Sheets 8-9).

5E.7 CONCLUDING REMARKS

a) The BNL reassessment found the failure of a water level reference leg to be an important contributor to core damage, amounting to 10% of Class I frequency (in the unmodified design).

b) The BNL model is more detailed in some aspects, but all the information appears in the PRA and an SLI report[2], which made the BNL insight and reassessment possible.

c) The SNPS study presents a reasonable human error analysis, taking detailed account of the information available to the operator in each sequence. BNL finds this analysis acceptable and sometimes even conservative (e.g., miscalibration = $2x10^{-3}$ - see Ref. 2, Appendix C, Page C-6 and Section 4.3 of this report). BNL used almost all these human error quantifications without change (apart from those in the drywell cooling analysis--see Appendix 5F).

d) Miscalibration error: throughout the SNPS-PRA the value $2x10^{-3}$ is consistently used for miscalibration of two or more transmitters or actuation channels. Lowering this value by using staggering procedures to prevent concurrent miscalibration of more than two channels can reduce the above failure rate significantly.

e) The modifications which are in progress at Shoreham tend to reduce the contribution of the sequences discussed apart from those related to the use of two reference legs.

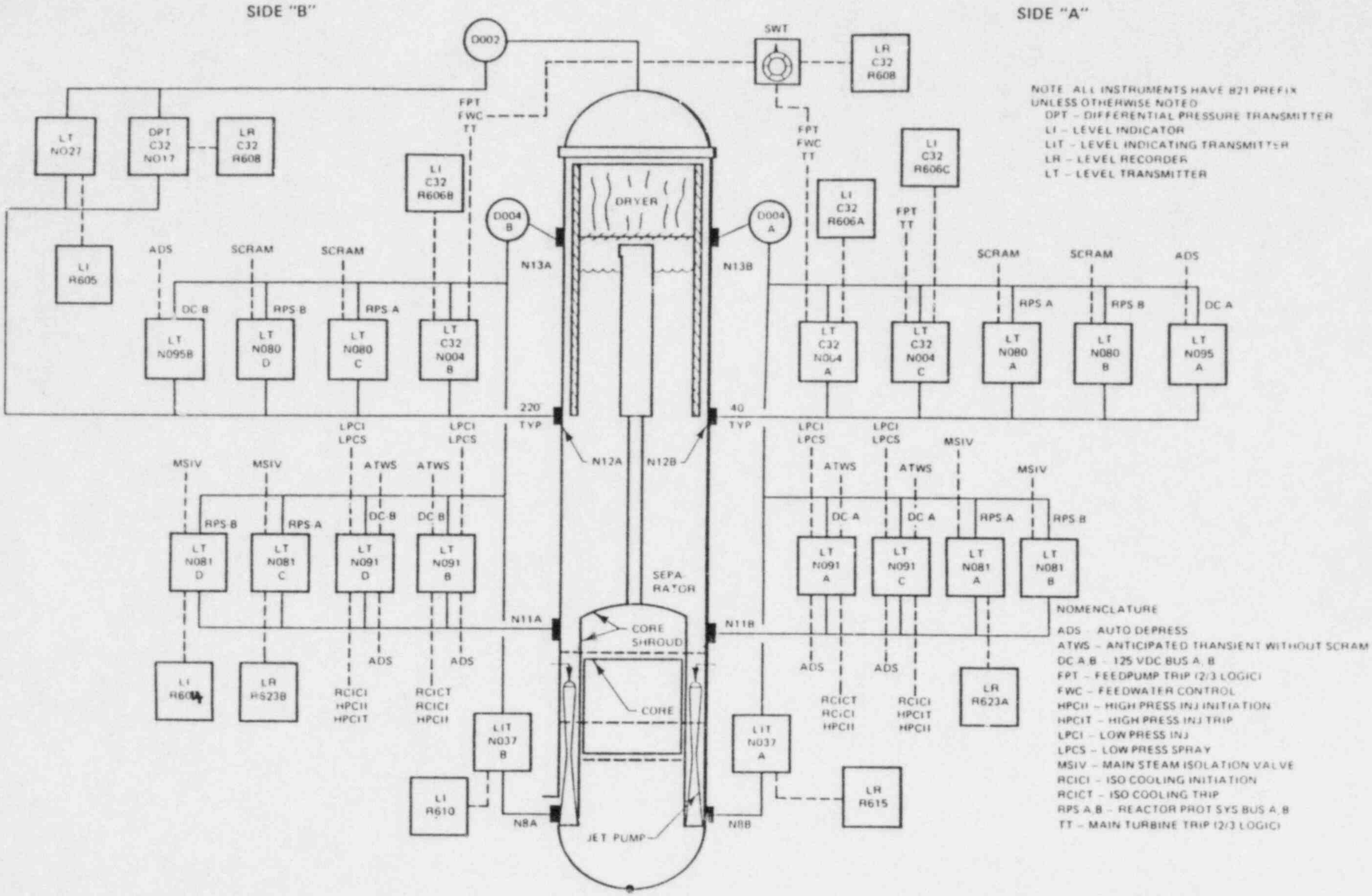


Figure 5E.1 Reactor Vessel Level Instrumentation Orientation

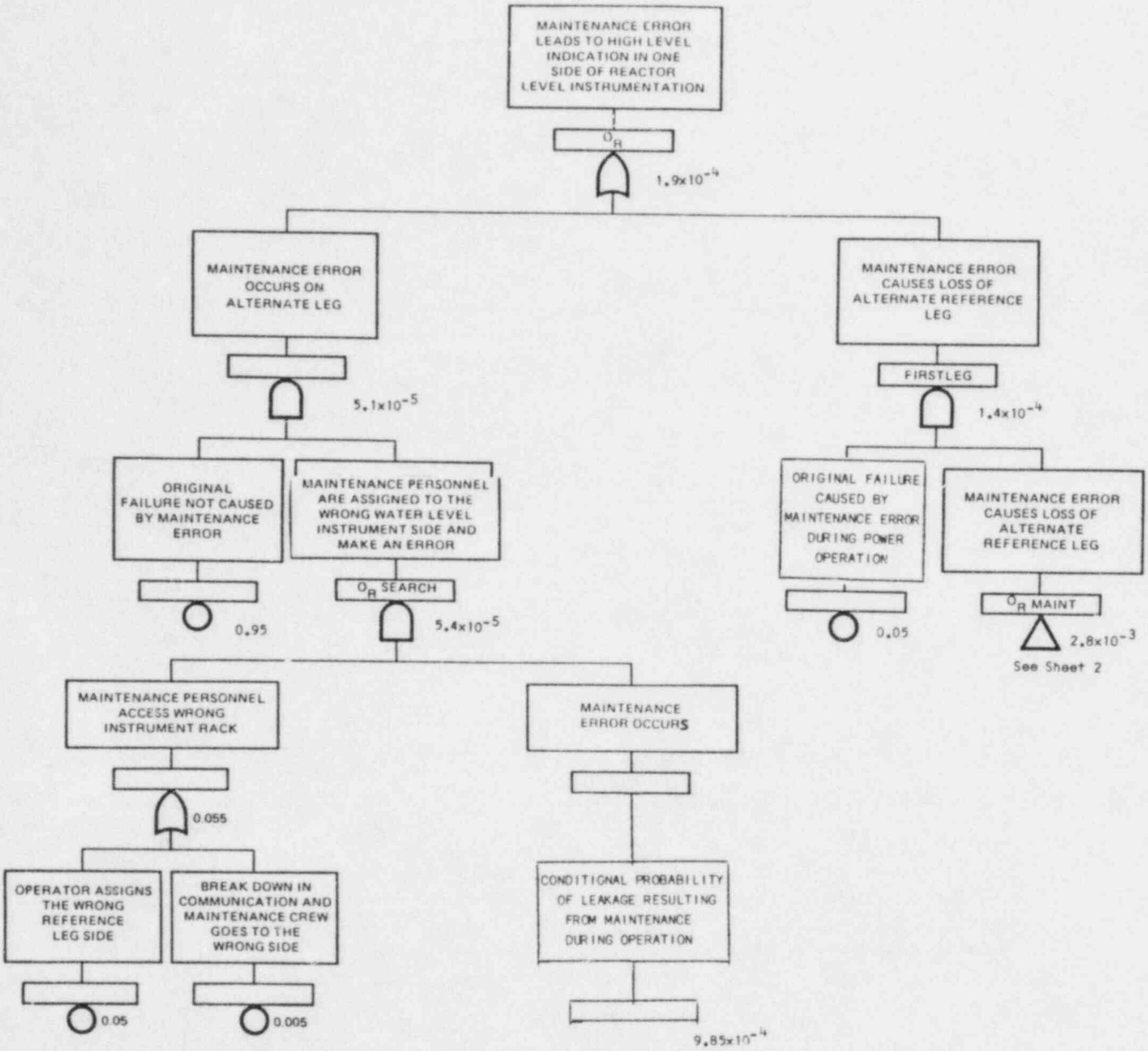


Figure 5E.2 Fault Tree for Operator Error Causes Failure of Alternate Reference Leg (Sheet 1 of 2)

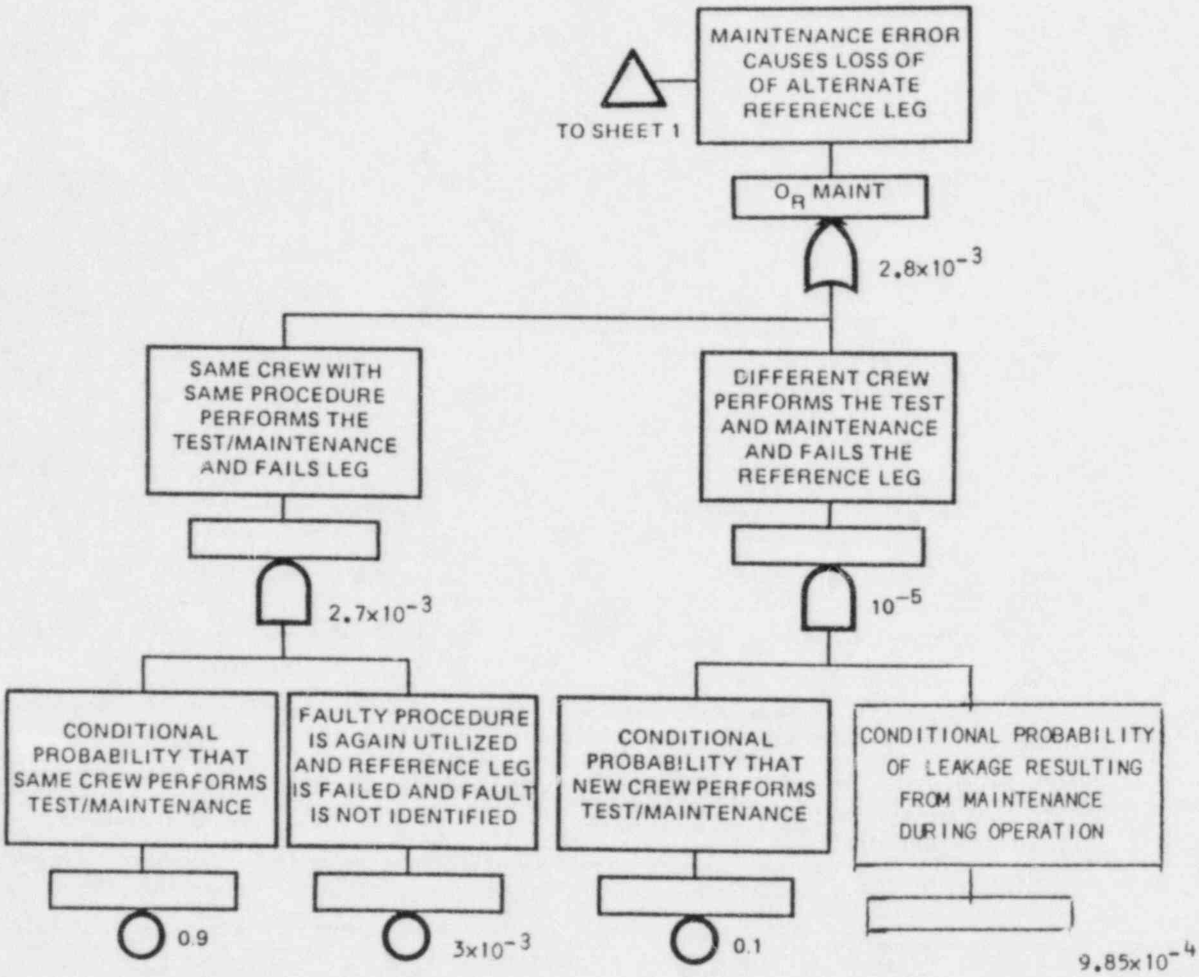


Figure 5E.2 Fault Tree for Operator Error Causes Failure of Alternate Reference Leg. (Sheet 2 of 2)

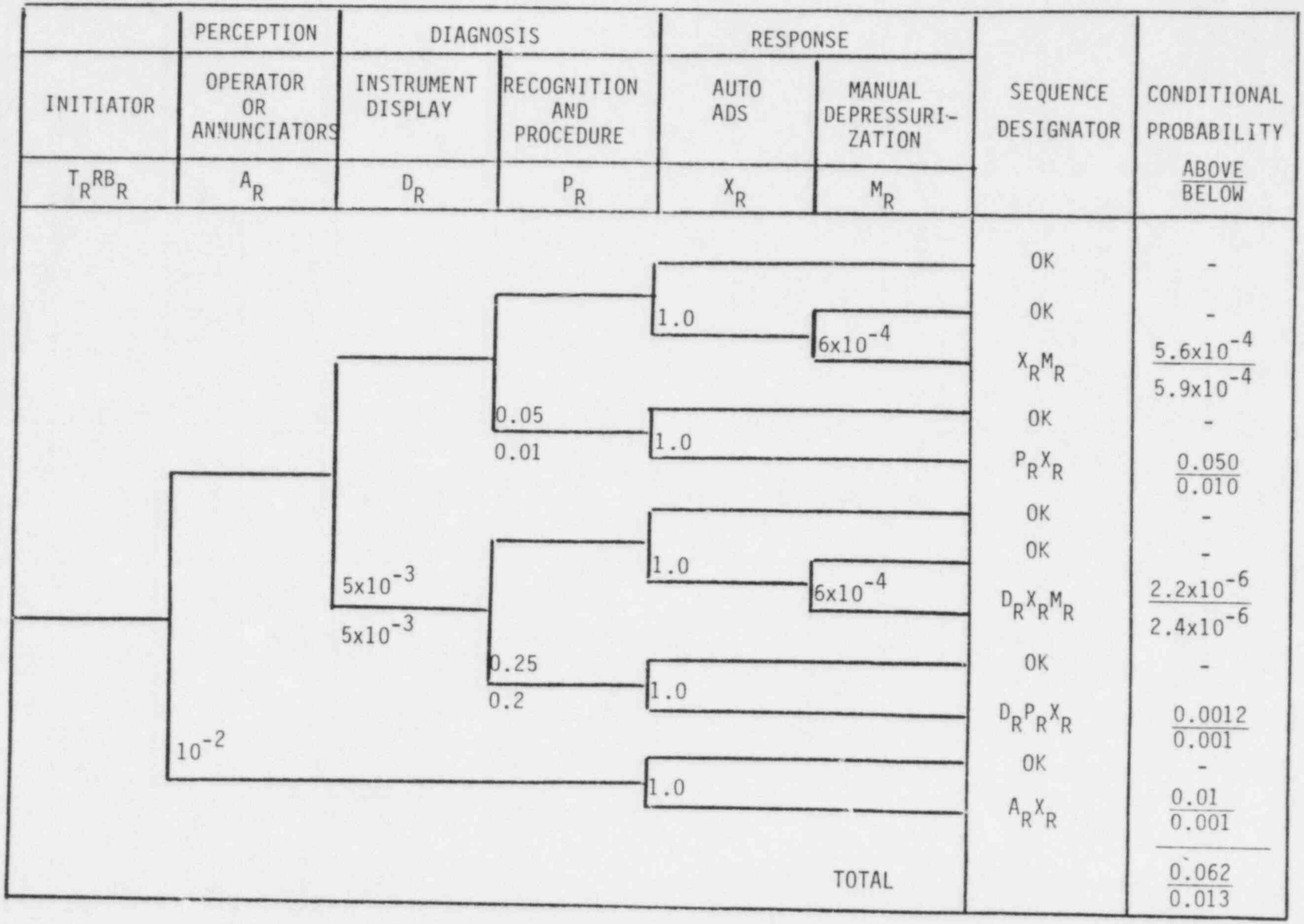


Figure 5E.3 Above: Leak in a Single Reference Leg Coupled with a DC Bus Failure or Miscalibration.
Below: Leak in a Single Reference Leg Coupled with a Failure of Differential Pressure Cell.

Table 5E.1 Level Instrument Assignments

Function	Side A		Side B	
	Instrument	Power	Instrument	Power
Scram & RHR ISO	LT B21-N080A(L3) LT B21-N080B(L3)	RPS A RPS B	LT B21-N080C(L3) LT B21-N080D(L3)	RPS A RPS B
HPCI Trip	LT B21-N091C(L8)	DC-A	LT B21-N091D(L8)	DC-B
HPCI Initiate	LT B21-N091A(L2) LT B21-N091C(L2)	DC-A DC-A	LT B21-N091B(L2) LT B21-N091D(L2)	DC-B DC-B
RCIC Trip	LT B21-N091A(L8)	DC-A	LT B21-N091D(L8)	DC-B
RCIC Initiate	LT B21-N091A(L2) LT B21-N091C(L2)	DC-A DC-A	LT B21-N091B(L2) LT B21-N091D(L2)	DC-B DC-B
MSIV	LT-B21-N081A(L2) LT-B21-N081B(L2)	RPS A RPS B	LT-B21-N081C(L2) LT-B21-N081D(L2)	RPS A RPS B
ATWS RPT	LT B21-N091A(L2) LT B21-N091C(L2)	DC-A DC-A	LT B21-N091B(L2) LT B21-N091D(L2)	DC-B DC-B
ATWS ARI	LT B21-N091A(L2) LT B21-N091C(L2)	DC-A DC-A	LT B21-N091B(L2) LT B21-N091D(L2)	DC-B DC-B
LPCI LPCS	LT B21-N091A(L1) LT B21-N091C(L1)	DC-A DC-A	LT B21-N091B(L1) LT B21-N091D(L1)	DC-B DC-B
ADS	LT B21-N095A(L3) LT B21-N091A(L1) LT B21-N091C(L1)	DC-A DC-A DC-A	LT B21-N095B(L3) LT B21-N091B(L1) LT B21-N091D(L1)	DC-B DC-B DC-B
Feed and Main TT	LT C32-N004A(L8) LT C32-N004C(L8)	Vital AC INST A	LT C32-N004B(L8)	INST B
Narrow Range Display*	LT C32-N004A(IND) LT C32-N004C(IND) LT C32-N004A(REC)+	Vital AC INST A Vital AC	LT C32-N004B(IND) LT C32-N004B(REC)+	INST B INST B
WR Display*	LT B21-N081A(REC)	RPS-A	LT B21-N081C(REC) LT B21-N081D(IND)	RPSB RPSB
Shutdown			LT B21-N027(IND)	INST A
Upset			LT C32-N017(REC)	INST A
Fuel Zone	LT B21-N037A(REC)	INST A	LT B21-N037B(IND)	INST B

* REC = Recorder; IND = Indicator.

\+ Recorder switched between sensors.

Table 5E.2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 1 of 9)

Values for Sheet 2 of 9

T_R = 0.018: Frequency of leakage of a single reference leg based on LERs. Same in SNPS-PRA and BNL assessment.

R_R = 1.0: When control is on side A and the reference leg on this side leaks, turbine trip occurs.

= 0.1: When control is on side A and the reference leg on side B leaks, turbine trip generally does not occur, and power operation may continue. It is assumed that 90% of the time side A control is used. Based on SNPS-PRA functional fault-tree D-1, which is acceptable.

O_R = 1.9E-4: The LER data for the conditional probability of inflicting a reference leg leak given maintenance during operation is given in the SNPS study (Appendix B, page B-12) as 9.85×10^{-4} events per maintenance. It should be noted that LERs include, in many instances, those events that were not recovered. This value was used by BNL as shown on the functional fault tree in Figure 5E.2. This model was used by BNL in place of the SNPS study functional fault tree in Figure D-2. The BNL model results are 7-fold higher than the SNPS value. The impact of this change is moderate.

B_R = 1.4E-4: The SNPS-PRA uses this value in evaluating transient induced loss of DC during other transients (e.g., see SNPS-PRA page 3-37). This value was used by BNL. It still seems to be conservative. The value used by SNPS is based on the fault tree for the DC power system and can be used for the case of continued operation. However, this change has low impact.

L_R = 0.016: This is developed in SNPS-PRA functional fault tree D-3, considered to be consistent, and used also by us. It represents a random failure of additional instrument channels.

C = 1.E-5: Scram probability as in other event trees, including ARI.

Q = 1.0: Leakage from a reference leg results in full scale reading with an unrecovered turbine trip at least for the short-term period.

Table 5E.2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 1 of 9 Continued)

Values for Sheet 2 of 9

$Q = 0.21$: If control is on leg B (≈10% of the time) and leg B failure occurred, then a feedwater runback may result. The value for recovery of feedwater is taken from the loss of FW event tree in Appendix 5A. The value used by SNPS apparently was not updated for the change of MSIV trip logic to level 1. (See Appendix 5A Table 5A.7 for an update of BNL value to 0.12.)

$X_H = 0.01$: This function was added by BNL to the event tree and means "Operator Erroneously Initiates ADS". This is taken from the functional fault trees of the SNPS study (Figures D.5 and D.6), and was explicitly included in the BNL assessment on the event tree. This value is the same as in the SNPS study.

$U' = 0.07$: Fault tree calculated RCIC unavailability.

$U'' = 0.1$: Fault tree calculated HPCI unavailability.

$U = 0.01$: Fault tree calculated RCIC and HPCI unavailability.

$X = 6.E-4$: Fault tree calculated ADS unavailability is 8.4×10^{-4}, which includes miscalibration error of level 1. This contribution to ADS unavailability was deducted, since it is treated explicitly in sheet 6.

$V = 6.2E-5$: Low pressure injection unavailability for LPCI, LPCS, condensate.

$W' = 4.4E-5$ /= 1.1E-4: RHR with/without RCIC steam condensing unavailabilty. Same as in the loss of FW event tree.

$W'' = 0.016$ 0.027: PCS unavailability taken from loss of FW event tree (Table 5A.7), consistent with the $Q = 0.21$. (See Table 5A.7 for an update of these values to 0.12 and 0.035.)

Values for Sheet 3 of 9
(Only if different from Sheet 2 of 9)

$R_R = 0.0162$: Frequency of the case in which failure occurred in leg A with control in leg B, also including a small contribution from failure of leg B with control on leg B and operator successfully transferring control to leg A, avoiding plant shutdown. Therefore, this is the frequency of continued plant power operation. BNL assumed that manual shutdown will be required within 12 to 24 hours.

Table 5E.2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 1 of 9 Continued)

Values for Sheet 3 of 9 (Continued)
(Only if different from Sheet 2 of 9)

B_R = 3.7E-4: Because continued operation is assumed for a day, the probability of failing DC power is taken from the fault tree. The unavailability is assumed to be conservative for this time period. However, the impact of this sequence is low.

Q = 0.04: The SNPS-PRA value is used. There, it is reasonably assumed that a spurious trip may occur with this probability during one day of continued operations. Note that the success branch means no challenge and that the problem was rectified, because all the possible challenges have been transferred out in the previous branches.

W" = 0.027: These values may be conservative because under conditions of spurious trip lower values for PCS unavailability could be used than in the case of loss of FW (e.g., PCS unavailability from turbine trip event tree). See discussion in Appendix 5A. However, the impact is negligible.

Values for Sheet 4 of 9
(Only if different from Sheet 2 of 9)

O_R = 3.4E-6:
3.4E-7
3.1E-6

Transfer-in totals to 6.8E-6. This is the frequency for leakage of two reference legs. Found to be 7-fold higher than in the SNPS-PRA, as discussed in connection with Figure 5E.2.

Q = 1.0: No recovery of FW possible as long as all channels read full scale.

U = 1.0: No high pressure automtic initiation. If initiated manually it is assumed to trip off on level 8. This results from the RCIC and HPCI sharing the same low level initiation and high level trip signals.

X = 0.3: ADS will not initiate automatically because it shares the level 1 initiation signals. The functional event tree in Figure C.7 of the SNPS-PRA provides a reasonably conservative representation of the probability of the operator recognizing the situation and taking actions under the prevailing conditions of very little water level information. This is a contributing sequence.

Table 5E.2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 1 of 9 Continued)

Values for Sheet 4 of 9 (Continued)
(Only if different from Sheet 2 of 9)

V = 5E-3: The failure of the low pressure injection is assumed by the SNPS-PRA to be dominated by an operator error--namely, the operator believes that the water level is high as indicated--followed by his terminating effective low pressure injection.

Values for Sheet 5 of 9
(Only if different from Sheet 2 of 9)

B_R = 2.5E-6/ 6.2E-6: Loss of DC contribution for cases of leg A and leg B failures respectively.

Q = 1.0: Automatic feedwater control cannot be assumed if two channels fail because of a reference leg leakage and the other two channels because of the loss of a DC bus.

H = 0.062: Operator fails to recognize the need for manual initiation of injection (high or low pressure). The functional event tree for this operator action is given in Figure C.8 of the SNPS study and reproduced in Figure 5E.3. It is a reasonably conservative approach. The dominating factor is the recognition of the right procedure and its performance, for which a value of 0.05 is given, consistent with that used elsewhere in the SNPS-PRA. This function is included in the BNL review in a manner different from that in the SNPS-PRA, as described in Section 5E.3. This is an important change in the BNL reassessment, and it resulted in a significant increase in the core damage contribution from loss of a reference leg coincident with miscalibration (see next sheet).

U' = 0.07/1.0: If DC-A fails, RCIC is unavailable.

U" = 1.0/0.1: If DC-B fails, HPCI becomes unavailable.

V = 0.001: Similar to the V value on Sheet 4. However, more information is available to the operator, and therefore the failure probability is reduced. BNL used the value from the SNPS study.

Table 5E.2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 1 of 9 Continued)

Values for Sheet 6 of 9

L_R = 3.6E-5/ 3.6E-6	The upper line is for the case of feedwater trip (Transfer from Sheet 2)
0.0/3.2E-5:	The lower line is for the case of continued operation. (Transfer from Sheet 3). It is assumed that a miscalibration of the instrumentation on the other leg existed while the reference leg leakage occurred. This was assumed in the SNPS study, but treated as affecting one additional channel rather than two.
H = 0.062:	As in the case of the loss of a DC bus, the miscalibration prevents any automatic initiation of HPCI, RCIC, ADS, LPCI, or LPCS. If the operator fails to recognize the need to initiate core cooling injection, the core would eventually uncover, and no automatic prevention is available.

Values for Sheet 7 of 9

L_R = 1.6E-4/ 1.6E-5:	The upper line is for the case of feedwater trip.
0.0/1.4E-4:	The lower line is for the case of continued operation. As in SNPS-PRA, the failure of an ΔP cell is assumed.
H = 0.013:	The operator action tree is similar to that for the loss of a DC bus or miscalibration. In this case, however, the operator will be alerted by the level 2 and level 1 annunciators of the remaining unfailed channel. This was given a factor of 0.2 (one-fifth) relative to the miscalibration case (see Figure 5E-3), i.e., the probability of operator failure to choose the right procedure is assumed to be 5 times lower. The available water level information on the control room indicators and recorders remains the same in all three cases.

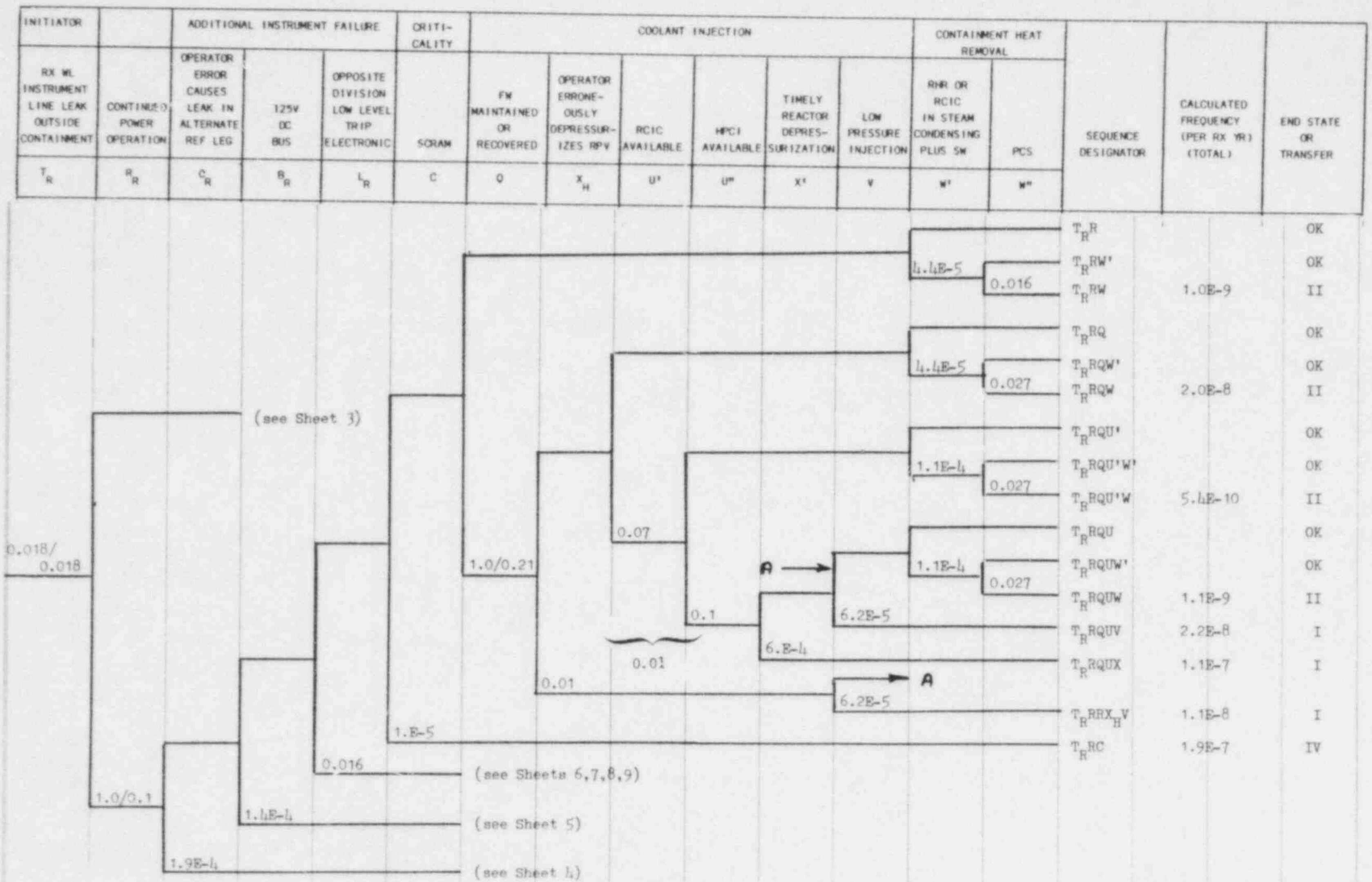
Table 5E.2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 1 of 9 Continued)

Values for Sheet 8 of 9

L_R = 4.3E-5/ 4.3E-6: 0.0/3.9E-5:	This is the frequency of one reference leg leakage and subsequent failure of a relay or a slave, causing level 1 initiation to fail (1 out of 2 twice logic). Level 2 initiation is available.
X_H = 0.01:	The SNPS-PRA assumes the possibility that the operator will erroneously actuate ADS. It is the same as on Sheets 2 and 3.
X' = 0.1:	The same value as in the SNPS-PRA ADS fault tree is used for failure to manually actuate ADS given automatic actuation failure.
V = 0.001:	Same as on Sheet 5.

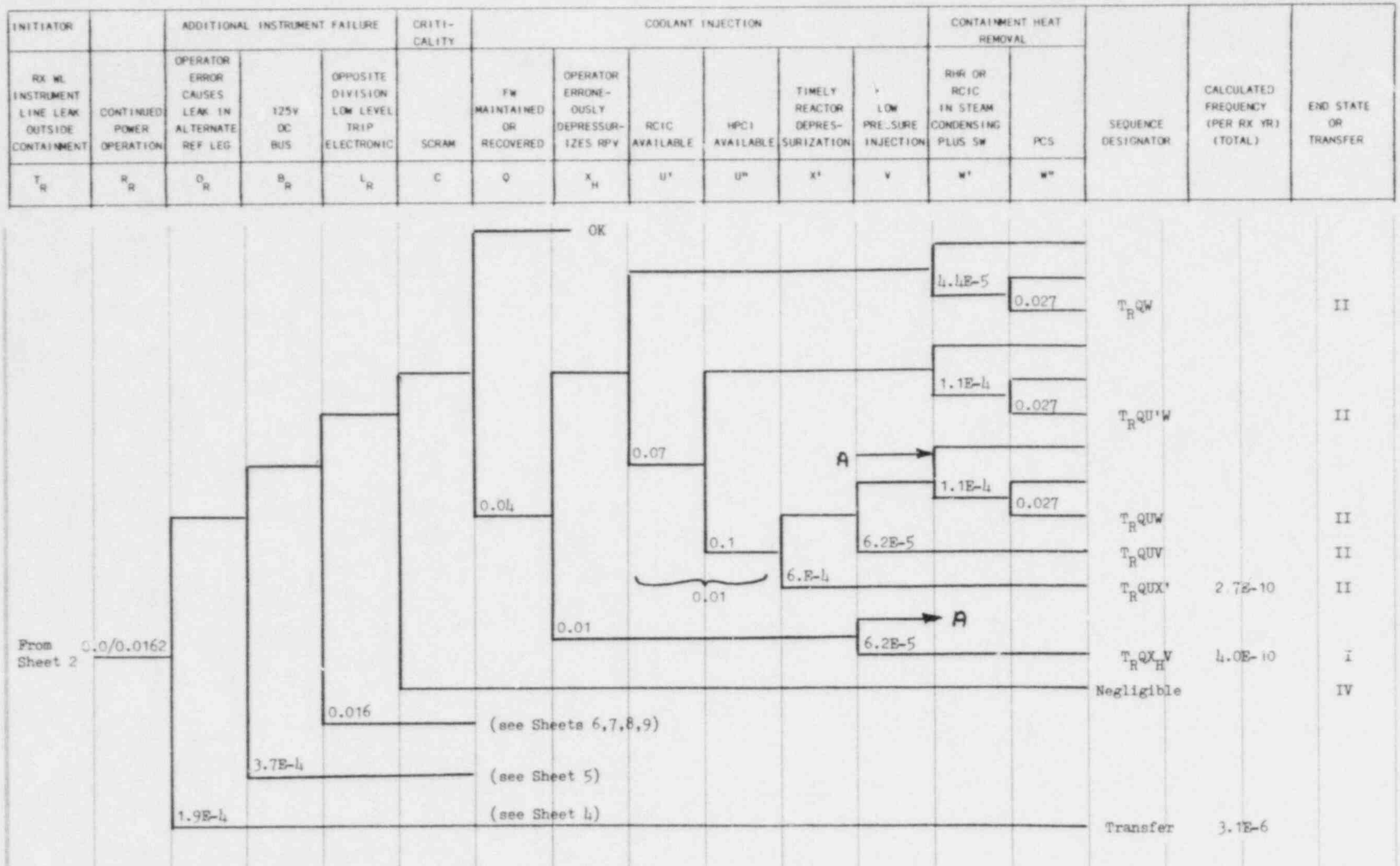
Values for Sheet 9 of 9

L_R = 4.3E-5/ 4.3E-6 0.0/3.9E-5:	This is the frequency of one reference leg leakage and subsequent failure of a relay or a slave, causing level 2 initiation failure. Level 1 is available to initiate ADS.
H = 0.1:	This value is consistent with the fault trees for HPCI and RCIC which assume 0.1 probability of failure to manually initiate high injection given the failure of automatic initiation. ADS, however, will be actuated automatically on level 1, which is available.



Legend: 1.0/0.1 = Leg A failure/Leg B failure

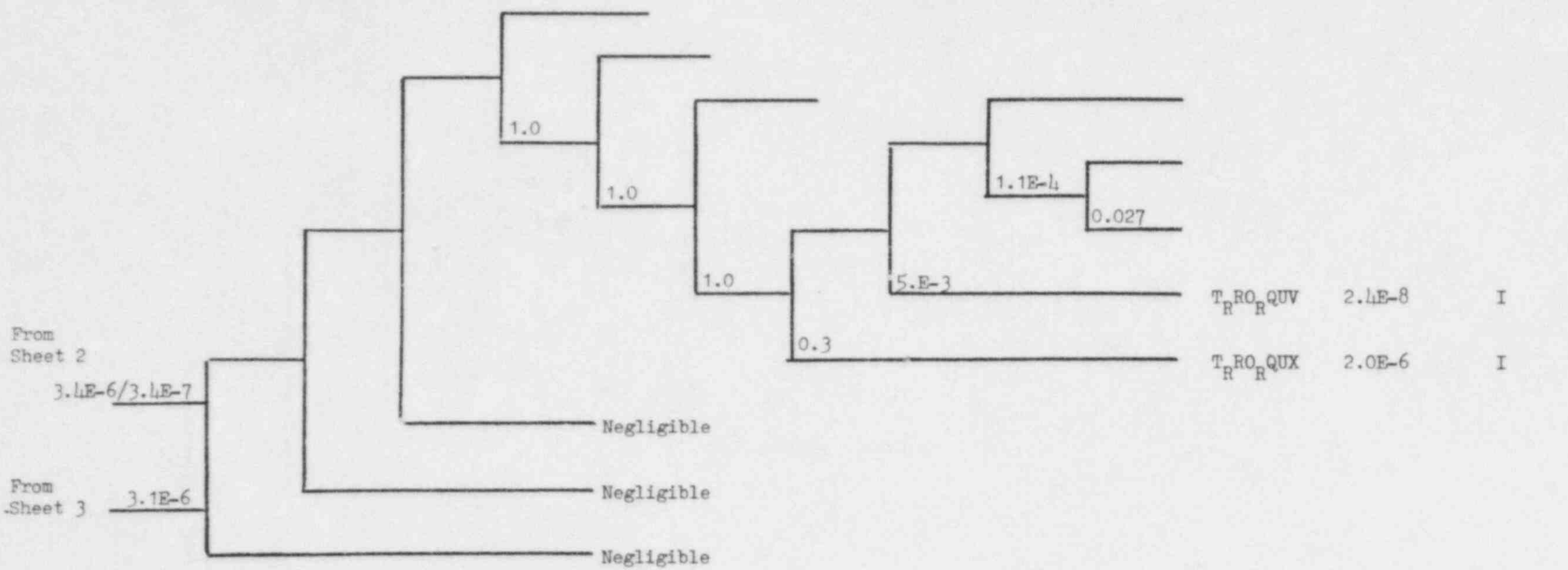
Table 5E.2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 2 of 9)



"Continued Operation Branch"

Table 5E.2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 3 of 9)

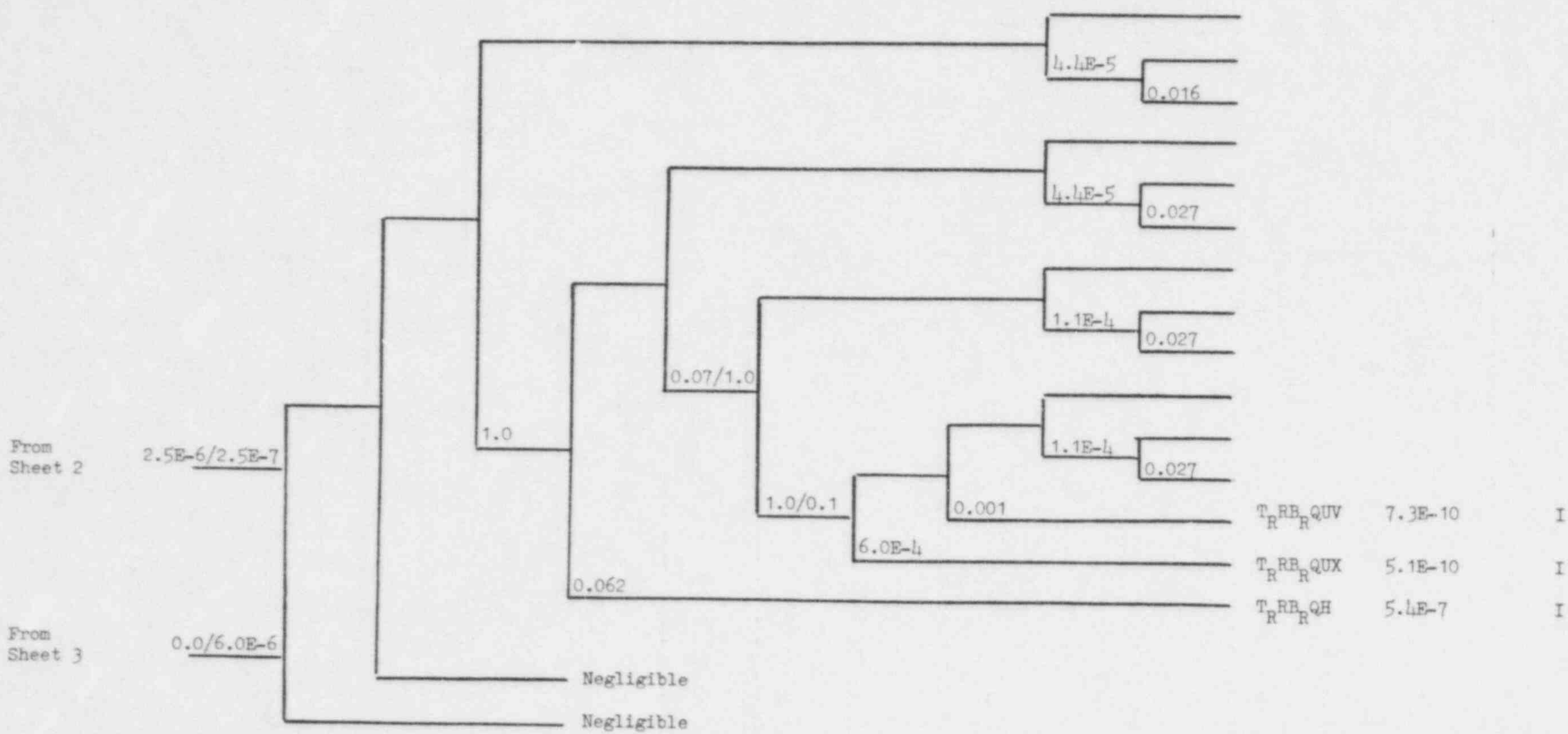
INITIATOR	ADDITIONAL INSTRUMENT FAILURE			CRITICALITY	COOLANT INJECTION					CONTAINMENT HEAT REMOVAL		SEQUENCE DESIGNATOR	CALCULATED FREQUENCY (PER RX YR) (TOTAL)	END STATE OR TRANSFER
RX WL INSTRUMENT LINE LEAK OUTSIDE CONTAINMENT	OPERATOR ERROR CAUSES LEAK IN ALTERNATE REF LEG	125V DC BUS	OPPOSITE DIVISION LOW LEVEL TRIP ELECTRONIC	SCRAM	FW MAINTAINED OR RECOVERED	RCIC AVAILABLE	HPCI AVAILABLE	TIMELY REACTOR DEPRESSURIZATION	LOW PRESSURE INJECTION	RHR OR RCIC IN STEAM CONDENSING PLUS SW	PCS			
$T_R R$	O_R	B_R	L_R	C	Q	U'	U"	X	V	W'	W"			



Maintenance Error on "Second Leg" Branch

Table 5E:2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 4 of 9)

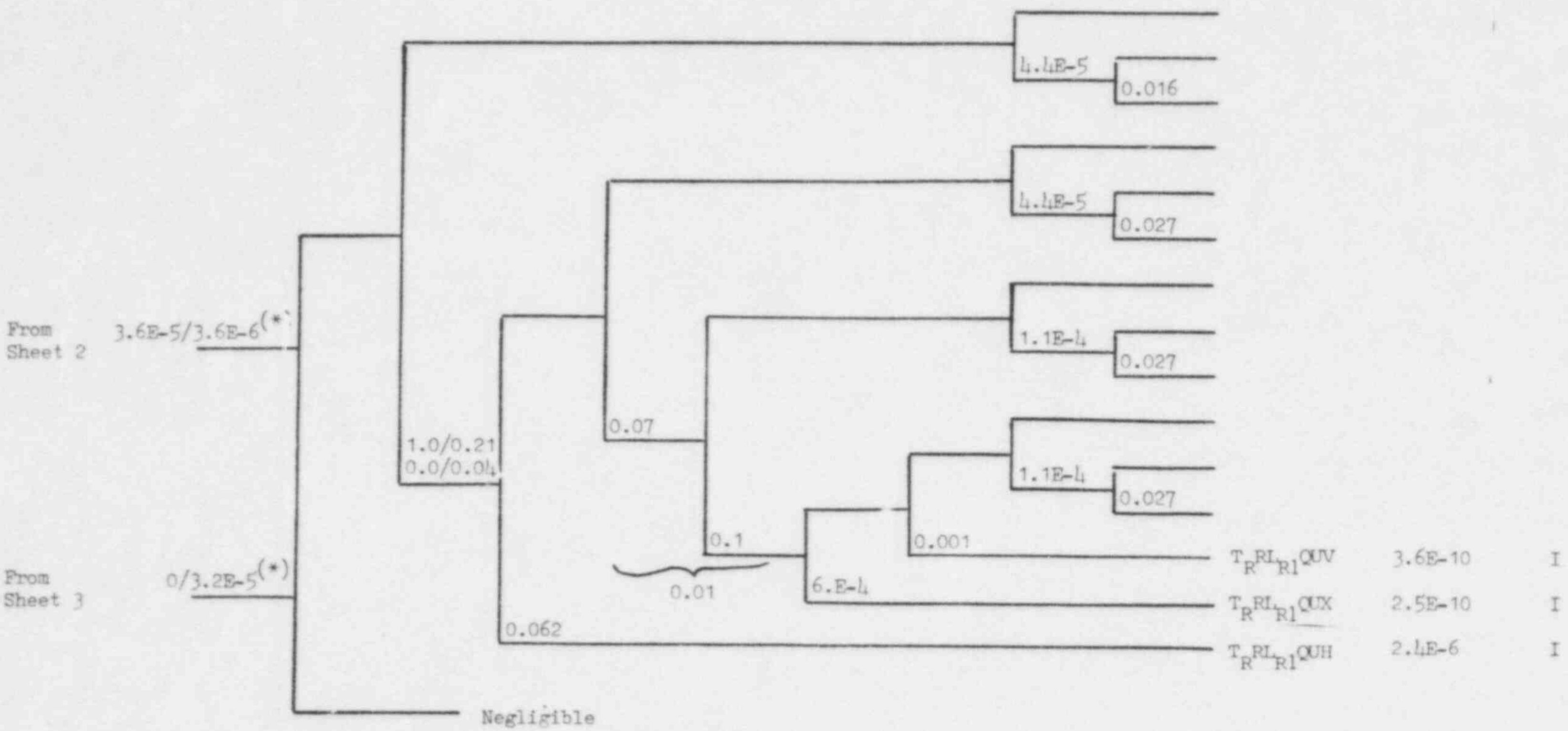
INITIATOR	ADDITIONAL INSTRUMENT FAILURE			CRITICALITY	COOLANT INJECTION						CONTAINMENT HEAT REMOVAL		SEQUENCE DESIGNATOR	CALCULATED FREQUENCY (PER RX YR) (TOTAL)	END STATE OR TRANSFER
RX WL INSTRUMENT LINE LEAK OUTSIDE CONTAINMENT	OPERATOR ERROR CAUSES LEAK IN ALTERNATE REF LEG	125V DC BUS	OPPOSITE DIVISION LOW LEVEL TRIP ELECTRONIC	SCRAM	FW MAINTAINED OR RECOVERED	OPERATOR RECOGNIZES NEED FOR INJECTION	RCIC AVAILABLE	HPCI AVAILABLE	TIMELY REACTOR DEPRESSURIZATION	LOW PRESSURE INJECTION	RHR OR RCIC IN STEAM CONDENSING PLUS SW	PCS			
T_R	O_R	B_R	L_R	C	Q	H	U'	U"	X'	V	W'	W"			



"Coincident Loss of a DC Bus" Branch -

Table 5E.2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 5 of 9)

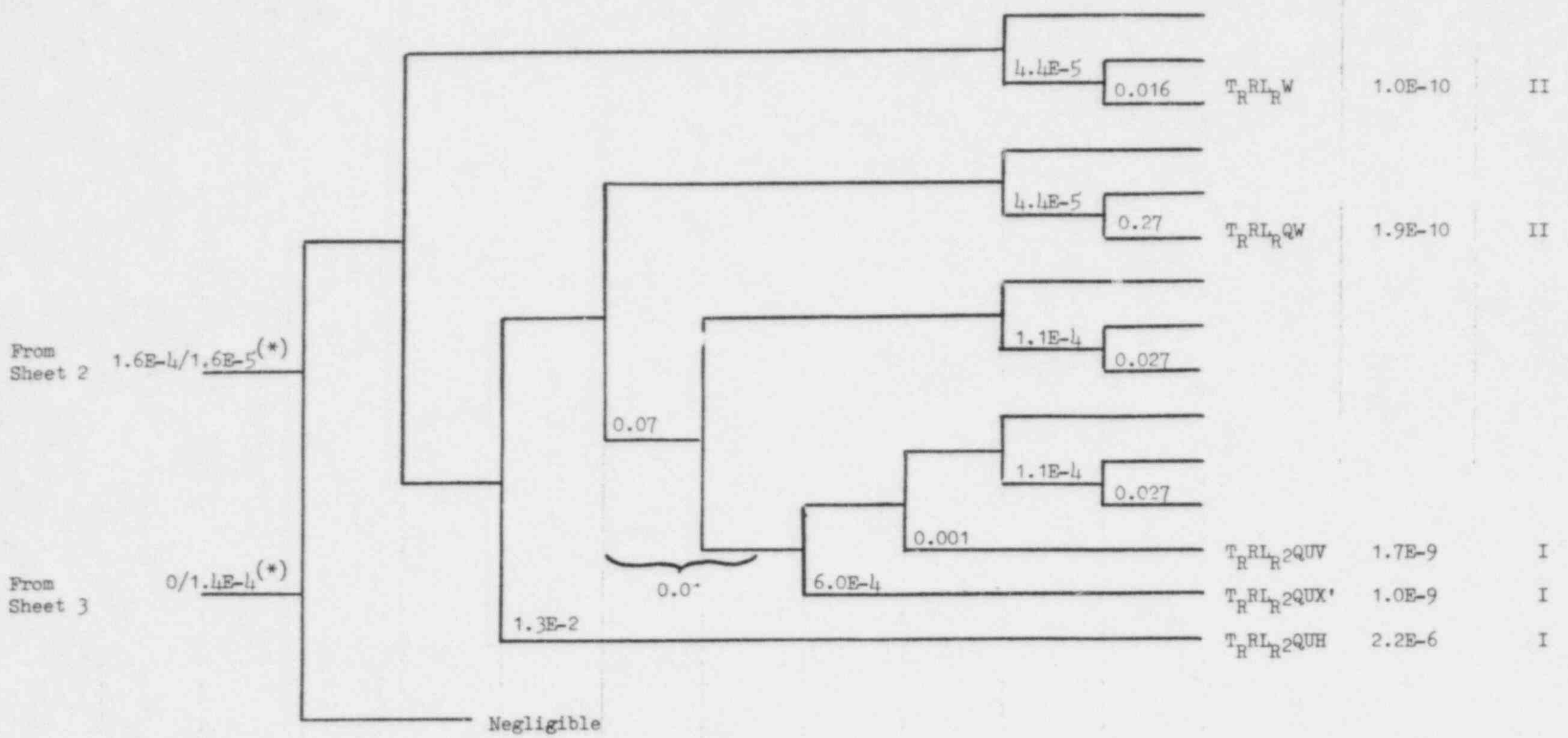
INITIATOR	ADDITIONAL INSTRUMENT FAILURE			CRITICALITY	COOLANT INJECTION						CONTAINMENT HEAT REMOVAL		SEQUENCE DESIGNATOR	CALCULATED FREQUENCY (PER RX YR) (TOTAL)	END STATE OR TRANSFER
RX WL INSTRUMENT LINE LEAK OUTSIDE CONTAINMENT	OPERATOR ERROR CAUSES LEAK IN ALTERNATE REF LEG	125V DC BUS	OPPOSITE DIVISION LOW LEVEL TRIP ELECTRONIC	SCRAM	FW MAINTAINED OR RECOVERED	OPERATOR RECOGNIZES NEED FOR INJECTION	RCIC AVAILABLE	HPCI AVAILABLE	TIMELY REACTOR DEPRESSURIZATION	LOW PRESSURE INJECTION	RHR OR RCIC IN STEAM CONDENSING PLUS SW	PCS			
T_R	O_R	B_R	L_{R1}	C	Q	H	U'	U"	X'	V	W'	W"			



(*) Miscalibration Contribution: 0.018 * 2.0E-3 = 3.6E-5
0.0162 * 2.0E-3 = 3.2E-5

Table 5E.2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 6 of 9)

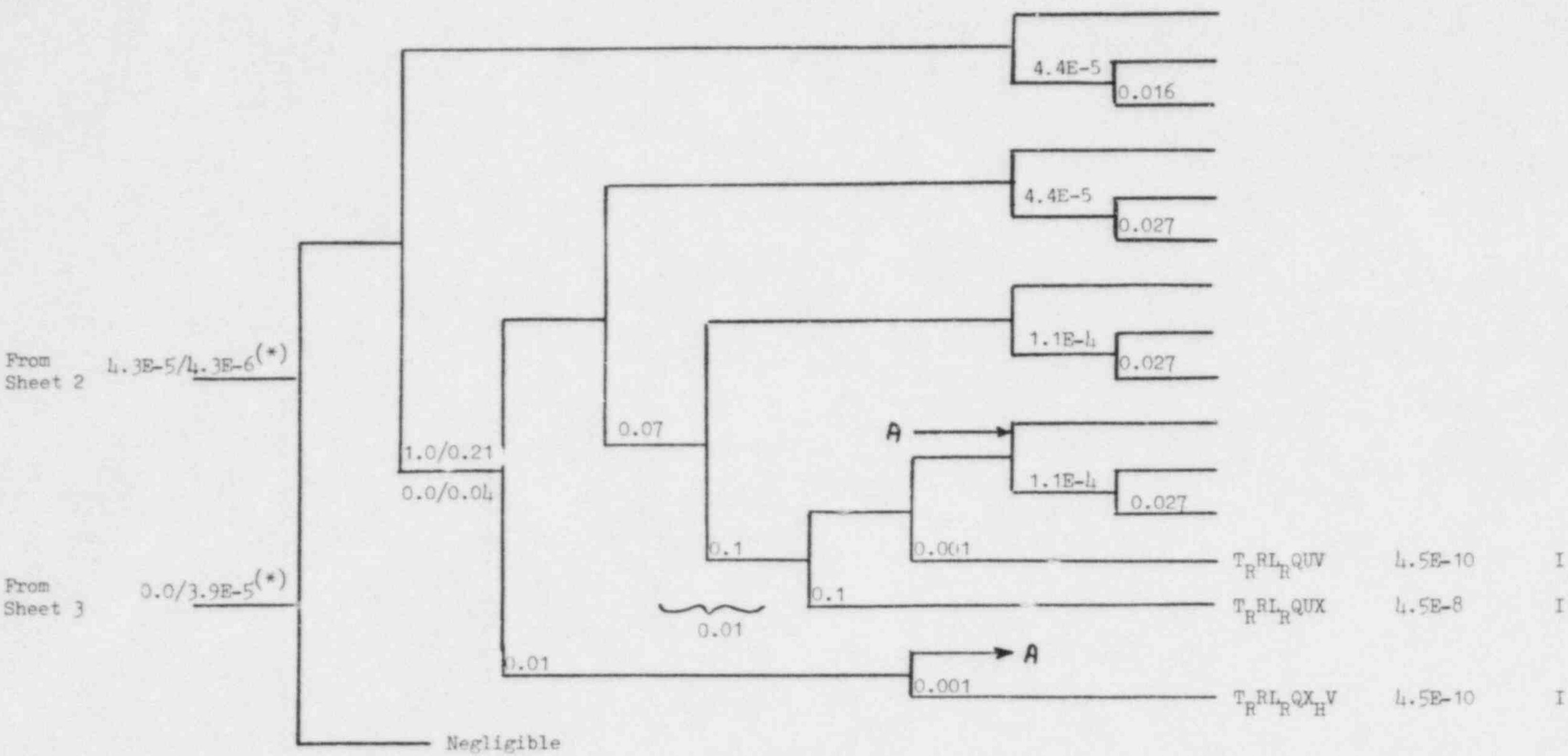
INITIATOR	ADDITIONAL INSTRUMENT FAILURE			CRITICALITY	COOLANT INJECTION						CONTAINMENT HEAT REMOVAL				
RX WL INSTRUMENT LINE LEAK OUTSIDE CONTAINMENT	OPERATOR ERROR CAUSES LEAK IN ALTERNATE REF LEG	125V DC BUS	OPPOSITE DIVISION LOW LEVEL TRIP ELECTRONIC	SCRAM	FW MAINTAINED OR RECOVERED	OPERATOR RECOGNIZES NEED FOR INJECTION	RCIC AVAILABLE	HPCI AVAILABLE	TIMELY REACTOR DEPRESSURIZATION	LOW PRESSURE INJECTION	RHR OR RCIC IN STEAM CONDENSING PLUS SW	PCS	SEQUENCE DESIGNATOR	CALCULATED FREQUENCY (PER RX YR) (TOTAL)	END STATE OR TRANSFER
$T_R R$	O_R	B_R	L_{R2}	C	Q	H	U'	U"	X'	V	W'	W"			



(*) Mechanical Failure of ΔP Cells: 0.018 * 8.8E-3 = 1.6E-4
0.0162 * 8.8E-3 = 1.4E-4

Table 5E.2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 7 of 9)

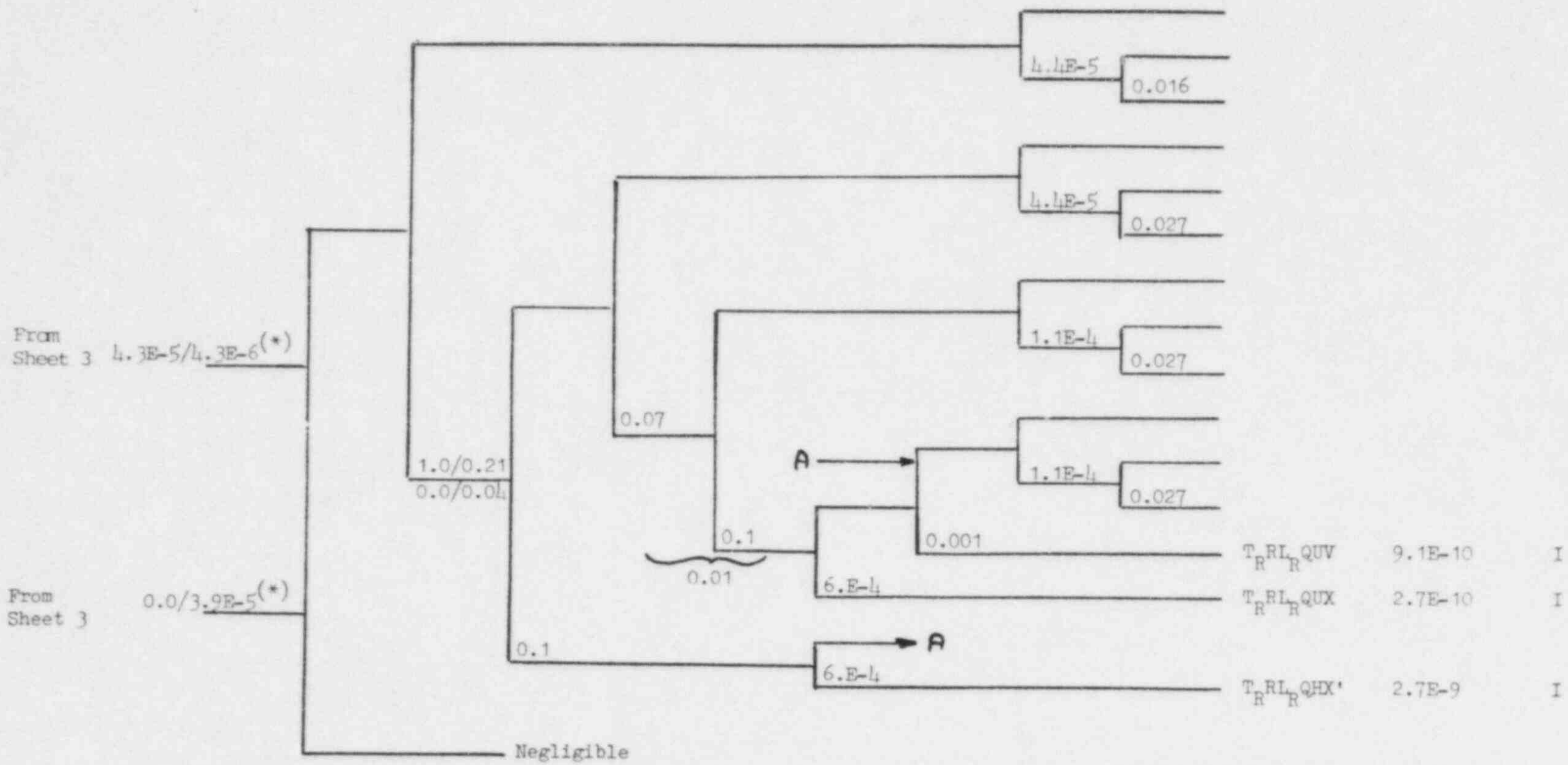
INITIATOR	ADDITIONAL INSTRUMENT FAILURE			CRITICALITY	COOLANT INJECTION						CONTAINMENT HEAT REMOVAL		SEQUENCE DESIGNATOR	CALCULATED FREQUENCY (PER RX YR) (TOTAL)	END STATE OR TRANSFER
RX WL INSTRUMENT LINE LEAK OUTSIDE CONTAINMENT	OPERATOR ERROR CAUSES LEAK IN ALTERNATE REF LEG	125V DC BUS	OPPOSITE DIVISION LOW LEVEL TRIP ELECTRONIC	SCRAM	FW MAINTAINED OR RECOVERED	OPERATOR ERRONEOUSLY DEPRESSURIZES RPV	RCIC AVAILABLE	HPCI AVAILABLE	TIMELY REACTOR DEPRESSURIZATION	LOW PRESSURE INJECTION	RHR OR RCIC IN STEAM CONDENSING PLUS SW	PCS			
T_R	O_R	B_R	L_R	C	Q	X_H	U'	U"	X'	V	W'	W"			



(*)Level 1 Initiation Fails Elec ronically and Level 2 Initiation is Available: 0.018 * (1.2E-3) * 2 = 4.3E-5
0.0162 * (1.2E-3) * 2 = 3.9E-5

Table 5E.2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 8 of 9)

INITIATOR	ADDITIONAL INSTRUMENT FAILURE			CRITICALITY	COOLANT INJECTION						CONTAINMENT HEAT REMOVAL		SEQUENCE DESIGNATOR	CALCULATED FREQUENCY (PER RX YR) (TOTAL)	END STATE OR TRANSFER
RX WL INSTRUMENT LINE LEAK OUTSIDE CONTAINMENT	OPERATOR ERROR CAUSES LEAK IN ALTERNATE REF LEG	125V DC BUS	OPPOSITE DIVISION LOW LEVEL TRIP ELECTRONIC	SCRAM	FW MAINTAINED OR RECOVERED	OPERATOR RECOGNIZES NEED FOR INJECTION	RCIC AVAILABLE	HPCI AVAILABLE	TIMELY REACTOR DEPRESSURIZATION	LOW PRESSURE INJECTION	RHR OR RCIC IN STEAM CONDENSING PLUS SW	PCS			
T_R	O_R	B_R	L_R	C	Q	H	U'	U"	X'	V	W'	W"			



(*)Level 1 Initiation is Available and
Level 2 Initiation Failed Electronically: 0.018 * (1.2E-3) * 2 = 4.3E-5
0.0162 * (1.2E-3) * 2 = 3.9E-5

Table 5E.2 Event Tree Diagram for Sequences Following Reactor Water Level Instrument Line Leak (Sheet 9 of 9)

APPENDIX 5F

IMPACT OF HIGH DRYWELL TEMPERATURE SEQUENCES

BNL's review of the contribution of loss of drywell cooling to the SNPS frequency of core damage is based on the SNPS study "Review of Shoreham Water Level Measurement System" (Report SLI-8221, Nov. 1982)[2].

5F.1 LOSS OF DRYWELL COOLING INITIATOR

The initiating event is a loss of drywell cooling followed by a rise in drywell temperature well above the technical specification limit of 135°F. Operating experience provided by the SNPS-PRA shows that the frequency of these events can be about 0.01 per year. Given the conditions of very high drywell temperature, the SNPS emergency procedures[13] require a plant shutdown (if high drywell pressure did not initiate automatic scram) and initiation of the containment (drywell) sprays if the temperature reaches 296°F, stated to be the drywell design temperature. This temperature is expected to be reached after approximately 25 minutes with no drywell cooling (see Table 5F.1) because of heat transferred from the RPV.

The level measurement displayed in the control room is based on level instrumentation that compares the water level difference of two instrument lines: the variable leg and the reference leg, both within the drywell. Changes in the density of the process fluid in the RPV, and of the fluid in the instrument lines, as a function of process temperature and drywell temperature, cause changes in the sensed level. Extreme combinations of process pressure and drywell temperature can cause flashing in the instrument lines. After a short transient of erratic reading, the variable leg will be refilled with fluid at RPV conditions and the reference leg will have some amount of its fluid lost. The analysis of water level measurement error due to this phenomena reported in the SNPS-PRA shows that if the reactor vessel is depressurized while drywell high temperature prevails, the following errors are possible:

a) Narrow-range maximum calculated flashing error:
   Side A - 8.5 feet
   Side B - 4.5 feet

b) Wide-range maximum calculated flashing error:
   Side A - 8.5 feet
   Side B - 4.5 feet

c) Fuel zone range maximum calculated flashing error:
   Side A - 6 feet
   Side B - 3 feet

The height difference between level 3 and level 1 is 12 feet, which is larger than the above errors. Therefore, a procedure to keep the pressure vessel level, under normal conditions, in the range between level 3 and level 5 could be effective in preventing core uncovery even if the calculated maximum flashing should occur. Such a procedure is part of the SNPS emergency procedures[13] for low pressure injection, and BNL gave much credit to its

utilization, assigning a low probability of 10^{-3} for not following it. The possible reason for not following the above procedure (i.e., premature reduction in injection flow) is an operator error due to the different water level indications if flashing occurred in one or more of the legs resulting in large differences in their readings.

In the SNPS model, the focus was on the operator diagnosis of the flashing situation, and the conditional probability of maintaining the core cooled and covered changed by two orders of magnitude between the case that the operator diagnosed this correctly and the case that he failed to diagnose it. BNL based its model on the successful performance of the basic procedure to flood the core and maintain RPV water level above level 3. BNL considered it uncertain that the operator will monitor the level information continuously during the incident to detect the onset of flashing, which may come when it is not expected. Therefore, BNL used only a single value for the probability of maintaining the core cooled and covered, regardless of the diagnosis of flashing.

Table 5F.2 Sheet 2 is the event tree for the loss of drywell cooling initiator, and Table 5F.2 Sheet 1 describes each branch and includes BNL changes or comments. Note that the frequency of the initiator was increased by 8×10^{-3} to account for failure of the drywell coolers subsequent to transients without isolation or manual shutdowns. This is discussed in the next section.

Loss of drywell cooling events (Table 5F.2) contribute 8.7×10^{-7} to core damage frequency. This is not a significant contributor to SNPS core damage frequency. BNL results are higher by a factor of 10 than the original SNPS results, however, because of the higher probability of failure assumed by BNL given the situation of level instrumentation flashing.

5F.2 TRANSIENTS OR LOCAs WITH SUBSEQUENT LOSS OF DRYWELL COOLING

5F.2.1 Transients

During transients or LOCA sequences, loss of drywell cooling may occur and cause a need to follow procedures requiring vessel depressurization, as a result of high drywell temperature, and require flooding of core to above level 3. This procedure is needed to prevent instrument line flashings and their effects.

For most of the transients, FW and PCS are not lost. In such cases there is a very low probability of decreasing water level in the core and of reaching level 1, or increasing suppression pool temperature up to the point corresponding to the 1.7 psi, at which isolation of drywell coolers occurs.

When a non-isolation transient occurs, there is a probability of drywell cooling failing during the course of the incident. This was conservatively calculated by the SNPS study (see Ref. 2 Appendix D) to be 6.6×10^{-4} given an incident. The main contribution is miscalibration of RBCCW level sensors not uncovered during the normal operation before the incident. If this is multiplied by about 12 transients and manual shutdowns per year (see Table 4.1), it results in about 0.008. This has been added to the initiator frequency of loss of drywell cooler in the preceding section.

In the case of isolation transients such as MSIV closure, loss of condenser, loss of FW and IORV*, the PCS is lost, and MSIVs are closed with some small probability of fast recovery. In these cases, the decay heat is discharged to the suppression pool with a slow rise in its temperature. If RHR is not routed to suppression pool cooling within the first two hours, or if there is a single SORV (0.1 conditional probability), a 1.7 psi pressure in the drywell may be reached in less than 2 hours (see Table 5F.1). BNL assumed a probability of 0.003 for delayed RHR initiation. In addition, in these isolation transients the probability of getting level 1 isolation is also significant. Only the latter was taken into account in the SNPS-PRA analysis of isolation transients.

The case of isolation transients was treated in the BNL review as a lumping of MSIV, loss of condenser, loss of feedwater, and IORV transients. It also includes LOOP with recovery within half an hour, which is considered similar to MSIV closure. BNL event trees differentiate between the branches having high pressure injection available and those having low pressure injection (see Table 5F.3 Sheet 2).

When a drywell cooler isolation signal occurs, the temperature rises in the drywell very rapidly, as shown in Table 5F.1, and the operator does not have much time available before he is required to depressurize the reactor according to the increase in drywell temperature. When a 1.7 psi signal is the cause of isolation, it does not disappear unless this logic is bypassed before restart of the coolers. However, at 10 minutes time, a temperature of 200°F may be reached, for which these coolers are apparently not qualified. BNL did not give credit to recovery of the coolers because of these three reasons, namely:

a) 10 minutes available,

b) Bypass needed,

c) Drywell conditions beyond coolers design.

In the case of level 1 isolation, no bypass is required after LPCI/LPCS start to raise the core water level following the ADS. Recovery of the coolers within 10 minutes may be assumed with a probability of success of 0.3.

The second drywell cooling option is the containment sprays. The procedures suggest initiating them at 296°F. The SNPS-PRA probability of 0.05 of failure to initiate containment sprays was also used by BNL. The high error rate was chosen by SNPS-PRA because containment spray actuation is unusual, and operators tend to avoid it.

Failure of containment sprays will require the operator to follow the emergency procedure that requires him to depressurize the reactor and then flood the core and maintain water at level 3. The SNPS-PRA sequence does not consider this procedure explicitly. Its model concentrates on the operator diagnosing the flashing. The value for operator failure to follow procedures in the BNL case (i.e., 0.001) is quite similar to the value used in the

*Loss of offsite power is treated separately in Section 5F.2.2.

SNPS-PRA. It is higher, however, because it includes the following considerations:

a) Failure to follow procedures = $0.01 \times 0.5 \times 0.15 = 7.5\times10^{-4}$, which is for two operators and a shift supervisor at low stress, moderate dependence.

b) Failure of ADS hardware = $\approx 6\times10^{-4}$ (from ADS fault tree).

c) Included in item (b) above is an adverse environment condition inside drywell, which may affect cable/solenoid coils.

d) If operator supposedly failed to follow procedures for RHR and drywell cooling initiation, then conditions exist that prevent him from doing so, and a higher probability (by a factor of 3) is assumed for operator failure to follow the next set of procedures.

It is assumed by BNL that, if the operator does not follow procedures and does not depressurize the RPV as required, then, sooner or later, the reactor would be depressurized and flashing conditions might be encountered.

If the high pressure injection fails and the reactor is depressurized, the high temperature in the drywell may still cause conditions of flashing. Flooding and keeping level 3 may still be required.

The event tree given in Table 5F.3 summarizes the case of isolation transients' contribution to core damage frequency associated with a situation of loss of drywell cooling.

The contribution of loss of drywell cooler during isolation transients to the core damage probability is small relative to the contribution of the isolation transients event analyzed in Appendix 5A. It amounts to 1.4×10^{-6} or $\approx$10% of the overall isolation transients contribution. This amount, however, is greater than calculated in SNPS-PRA, because of the higher probability of failure assumed by BNL for keeping water level at level 3, and because of an additional contribution from the possibility of delayed initiation of RHR, not considered in the SNPS-PRA.

5F.2.2 Water Level Measurement Implication of Losing Offsite and Onsite AC Power

The loss of offsite power event was discussed in great detail in Appendix 5B. Imposing additional conditions on those trees could make them quite difficult to prepare or follow. Therefore, this section provides a separate discussion focused on the special implication to the water measurement system if AC power is lost. To the event tree branches and the results of Appendix 5B, the results of this discussion should be added.

Two implications with respect to water measurement are considered:

a) Loss of water level indications in control room following LOOP with the loss of Division I and II diesel generators. In this case, as a result of inadequate control room information, the procedures required to maintain water level would be very difficult to follow.

b) Loss of drywell cooling and containment sprays resulting from the unavailability of Division I and II AC power, or in the case of losing offsite power alone. In these cases, drywell temperature may rise, and, if drywell cooling is not restored, flashing conditions may be possible when the reactor is depressurized. The rise in drywell temperature may prompt the operator to depressurize the plant in accordance with the procedures which, in turn, may cause the instrument legs to flash and accurate water level indication to be lost. The procedures then direct the operator to flood the primary system with low pressure coolant injection and maintain water level above level 3.

Water Level Indications in the Control Room

Water level measurement failures were discussed in Appendix 5E, and the system was described. Figure 5E.1 summarizes all water level instrumentation. Table 5E.1 provides a summary of the power supply for each of the indicators in the control room based on Ref. 2, the review of the water level measurement of the SNPS system, and on the SNPS-FSAR information.

According to the above information, only one level indicator is available which has DC supply backup (Narrow-Range LT-N004A with indicator LI-R606A) to show level information when LOOP with unavailability of Division I and II diesel generators occurs. This situation was recognized in the SNPS-PRA (page D-34 of Ref. 2), and the value of 0.1 for successfully controlling water level used in Ref. 2 was changed to 0.06 in the revision of this sequence in the revised SNPS-PRA (see page 3-306 of the SNPS-PRA).

Description of the Sequences Following LOOP

The event tree for LOOP, focusing on the functions related to level measurement, is given in Table 5F.4. It considers recovery of offsite power within 30 minutes (0.63 success probability) because after recovery the containment sprays become available and the sequence will progress similarly to the two dominant branches of the isolation transient (Table 5F.3). Therefore, given recovery of offsite power, the sequence is transferred to the isolation transient. If at least one diesel generator is available, containment sprays can be actuated. The drywell cooler actuations require several actions by the operator:

a) To transfer the fan coolers to the operating emergency bus.

b) To bypass drywell coolers isolation signal if drywell coolers down time exceeded approximately 10 minutes, in which 200°F was exceeded in the drywell and high drywell pressure signal may have occurred.

Based on these considerations, a value of 0.05 for containment spray and 0.7 for drywell coolers was used. For this case, there is sufficient information on core water level in the control room, and the value 0.001 was used for failing to maintain level 3 as the procedure requires. This is the same value used in the previous event trees. This branch was not considered in the SNPS-PRA (T_EIGL = 1.9×10^{-6}).

The most severe sequence is the loss of the two emergency AC buses following the LOOP. In this case, there is no AC power to drive any drywell cooling, either containment spray (having one pump on Division III but the drywell spray isolation valves F016 and F021 can be opened by AC power from Division I and II only) or drywell coolers. Credit for the containment sprays under these conditions, given by SNPS-PRA in Figure 3.4-52 (PRA page 3-306) is not justified.

When the "L" function of this sequence is reached, two questions are encountered which the SNPS-PRA does not address in sufficient detail:

a) The advisability, in this particular case, of following the emergency procedures that require the operator to depressurize the reactor and flood it when level information is insufficient or conflicting.

b) The alternatives the operator has in controlling core water level under the condition of successful high pressure injection.

In case (a), depressurizing the reactor will require use of Division III LPCI and Division III diesel generator, for which a very low availability (0.37) is assumed (based on LERs) in the SNPS-PRA under blackout conditions. Therefore, it seems that a special emergency procedure warning the operator away from taking this route is missing for the particular case of blackout conditions.

In case (b), if the operator chooses not to drpressurize the reactor, the level information for controlling high pressure injection is seemingly very poor and several level 8 trips and level 2 starts may occur, challenging the system on each start. If the operator decides to take manual control of the injection, the probability of reaching level 1 is increased. In addition, the failure rate of the single water level indicator available (N004A) must also be considered.

In choosing a value of 0.05, BNL also took into account that the RCIC will be easier to control than HPCI, which has a flow rate 10-fold higher than RCIC. However, the value of 0.05 assumes that the operator procedures would instruct him to avoid depressurization in the case of loss of instrumentation reading due to loss of all AC.

The contribution of level measurement unavailability due to LOOP is seen to be higher than calculated in the SNPS-PRA. This results mainly from the unavailability of the containment sprays, which were given credit in the SNPS-PRA. This sequence is one of the largest single contributors to core damage frequency calculated by BNL and amounts to 1.0×10^{-5}. The total contribution from the sequences discussed here is 1.2×10^{-5}, which is about 15% of the Class I frequency.

5F.2.3 Loss of Coolant Accidents with Loss of Drywell Cooling

The LOCA event trees were discussed in Appendix 5C. The considerations of high drywell temperature are added to them here. These considerations are similar to those in the preceding sections with the following differences specific for LOCAs:

a) Drywell coolers are isolated and ineffective.

b) In case of larger breaks, such as medium LOCA, the reactor will depressurize some time into the incident.

c) The procedure of flooding the core and keeping the water level at level 3 is even more vital in LOCA situations. Thus it is assumed that it will be followed with higher reliability.

d) If containment sprays are not used or failed, the conditions of high temperature in the drywell and low reactor vessel pressure may prevail for longer times than in the case of transients in which the reactor can be repressurized to ≈150 psi and thus avoid flashing conditions. This situation may be reached mainly in cases of medium LOCA.

The event tree diagram of Table 5F.5 takes the above into consideration. Because of the compensating effects of items (c) and (d), the value of 10^{-3} for the probability of failure to follow the procedure (L function) was not changed relative to the previous cases. This resulted in a relative significant contribution from the medium LOCA sequence to the frequency of the Class III category, which is about 10% of the total frequency in this class.

Large LOCA was not considered in the analysis of the effects of the loss of drywell cooling. The reason given in the SNPS study, that the frequency of LOCA is small, is difficult to accept because it is a significant contributor to Class III, larger in fact, than small LOCA, for which an event tree was prepared and discussed. Furthermore, flashing will occur due to the rapid depressurization of the large LOCA, and flashing conditions may be sustained, because of high temperature and low pressure in the drywell, which are characteristic of the consequences of large LOCA.

In the case of large LOCA, the procedures are relatively simple, requiring flooding of the reactor vessel with LPCI or LPCS. The control of level is not important as in the case of small LOCA when the system can be repressurized if inflow is much larger than the outflow through the break. Therefore, it is assumed the reliance on level measurement is much less important in this case than in the small or medium LOCA.

Table 5F.1 Drywell and Suppression Pool Temperature Following a Shutdown from Transient Without DHR or PCS (Approximate) Reference: ORNL[8] and Fig. 5-3[2]

Time (hours)	Drywell Temperature		Suppression Pool Temperature (°F)	Drywell Pressure	
	Loss of Coolers (°F)	Drywell Coolers Available (°F)		Loss of Coolers (psig)	Drywell Coolers Available (psig)
0	120	120	95	0	0
0.1	200	120	95	2	0
0.2	250	120	95	3	0
0.3	280	120	95	4	0
0.5	300	120	95	4 1/2	0
1	310	120	110	5	1
2	310	120	140	6	2**
5	320	120	190	7	3
10	320	130	220	10	7
15	320	160*	250	30	20
20	320	260	280	45	35
25	320	320	300	70	60

*Some time later at ≈200°F the coolers are assumed to be failed because of conditions beyond their qualifications.

**Drywell cooler isolated as a result of 2 psi in drywell. It is assumed in this column that operator bypasses isolation signal and returns coolers to operation.

Table 5F.2 Loss of Drywell Cooling Event Tree Quantification Description (Sheet 1 of 2)

T_{MT} = 0.0173:	0.0093 of this represents two LERs in which drywell cooling was lost and temperature exceeded 212°F. These two LERs occurred in 215 reactor years. An additional 0.008 is from drywell cooler failure during a transient.
C = $\approx 10^{-6}$:	If automatic scram does not occur (with its very low probability) operator can manually scram the reactor within the next 10 minutes with no impact.
Q = 0.082:	The value for FW recovery from the turbine trip event tree (Table 5A.1) is used in the BNL assessment, which is different from the SNPS value.
U = 0.01:	This is the BNL value (see Table 5A.2).
X = 8.4E-4:	SNPS-ADS unavailability.
V = 6.2E-5:	Low pressure injection unavailability (LPCI, LPCS, condensate).
G = 0.05:	Drywell heat removal function. Only containment spray is available for the operator to initiate. The high value for operator error is due to the consideration that this system is rarely used and the operator will, in general, tend to defer its initiation; BNL and SNPS use the same value.
L = 0.001:	BNL assumed a different model, as discussed in this appendix, Section 5F-1.
W' = 4.4E-5:	The value for RHR with RCIC steam condensing is used from the turbine trip event tree.
W" = 1.3E-2:	The value for PCS unavailability from the turbine trip event tree is used.

INITIATOR	CRITICALITY	COOLANT INJECTION				WATER LEVEL INSTRUMENTATION		CONTAINMENT HEAT REMOVAL				
MANUAL SHUTDOWN DUE TO HIGH DW TEMP.	SCRAM	FEEDWATER RECOVERED IMMEDIATELY	RCIC or HPCI AVAILABLE	TIMELY REACTOR DEPRES-SURIZATION	LOW PRESSURE INJECTION AVAILABLE	DRYWELL HEAT REMOVAL	STABLE COOLING ESTABLISHED	RHR OR RCIC IN STEAM CONDENSING PLUS SW	PCS	SEQUENCE DESIGNATOR	CALCULATED FREQUENCY (PER Rx Yr)	CLASS OF POSTULATED CORE DAMAGE OR TRANSFER
T_{MT}	C	Q	U	X	V	G	L	W'	W"			

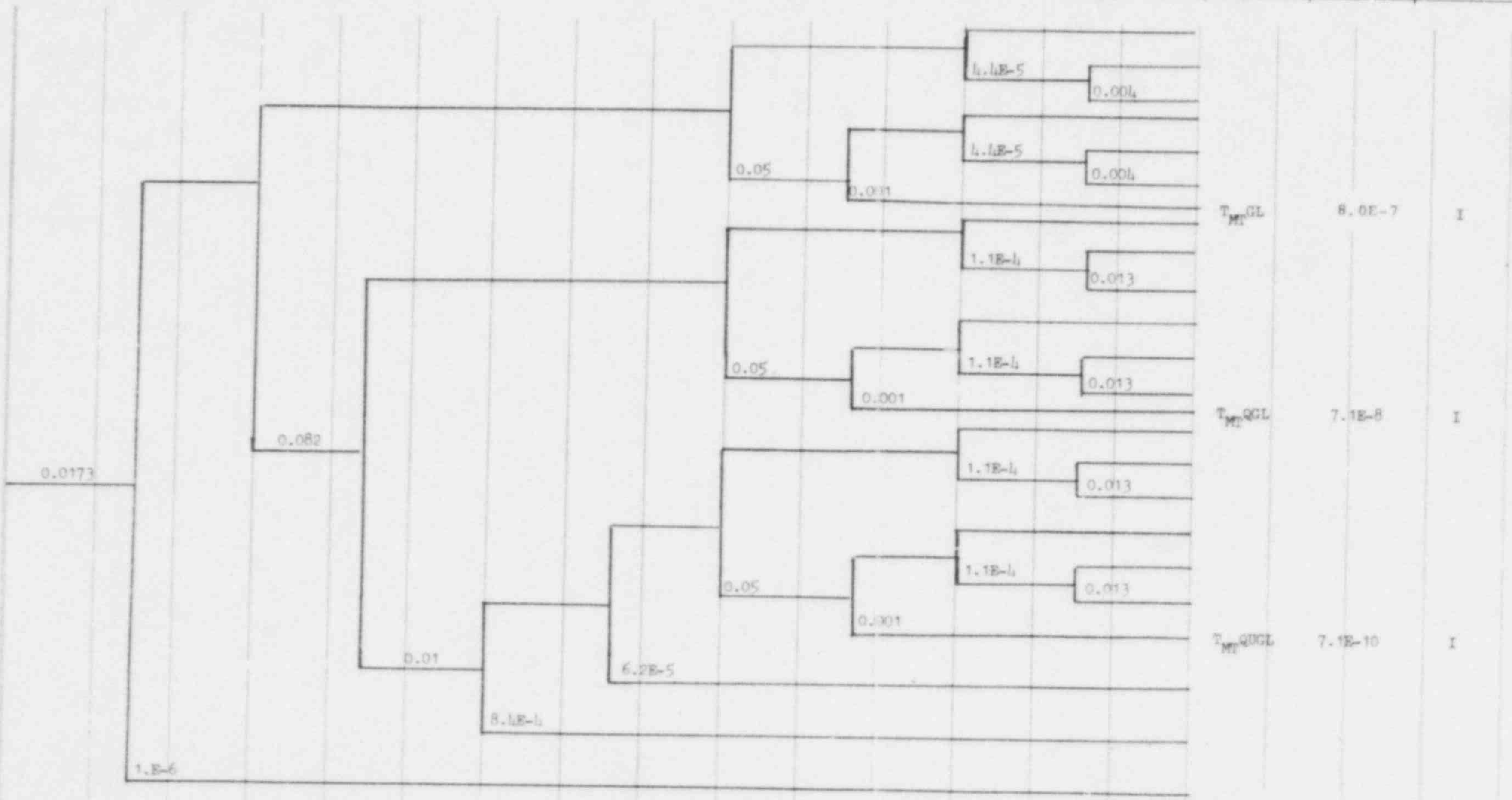


Table 5F.2 Loss of Drywell Cooling Event Tree
(Sheet 2 of 2)

Table 5F.3 Event Tree Diagram for Isolation Transients with Loss of Drywell Cooling (Sheet 1 of 2)

$T_{(M)}$ = 1.73: This is the sum of the frequencies of all transients with loss of FW or MSIV closure in which PCS is not available for at least one hour. Most of these cases will have several hours without PCS. They include MSIV closure without recovery, IORV, loss of condenser, loss of FW, and TT with subsequent loss of FW. Also the case of recovery from LOOP event is included.

U = 0.01: This is BNL's value (see Table 5A.2).

H = 0.003: This is BNL's assumption of the probability of delay in suppression pool cooling initiation with RHR, following a transient. It assumes two hours delay. However 10% of the cases may have SORV, in which case only one hour RHR delay will result in 1.7 psi in drywell.

X = $8x10^{-4}$: BNL value (see Table 5A.2)*

V = $6.2x10^{-5}$: BNL value (see Table 5A-2)*. Includes LPCI, LPCS, and condensate. This value may be low because condensate pumps are available only for a part of the isolation transients.

G = $3.3x10^{-5}$: Drywell cooling failure probability during the transient. Includes ($5x10^{-4}$ + $1.6x10^{-4}$) x 0.05 or (miscalibration + drywell coolers failure probability in 5 to 10 hrs) x Failure to initiate containment spray.

= 0.05: Failure to initiate containment spray. Drywell coolers isolated on high drywell pressure signal.

= 0.035 Same as above times 0.7 for failure to recover drywell coolers before level 1.

L = 0.001: Operator failure probability to maintain level 3 as required under circumstances of potential for flashing in instrumentation lines. For the case that the operator failed to follow required prior procedures, such as initiating RHR when suppression pool temperature reached 90°F and initiation containment spray, the value for L was increased to L = 0.003.

W', W": These values are given just for completeness. All unquantified sequences are duplicates of sequences appearing in Tables of Appendix 5A.

*Value given for approximation only. See particular sequence for correct sequence quantification.

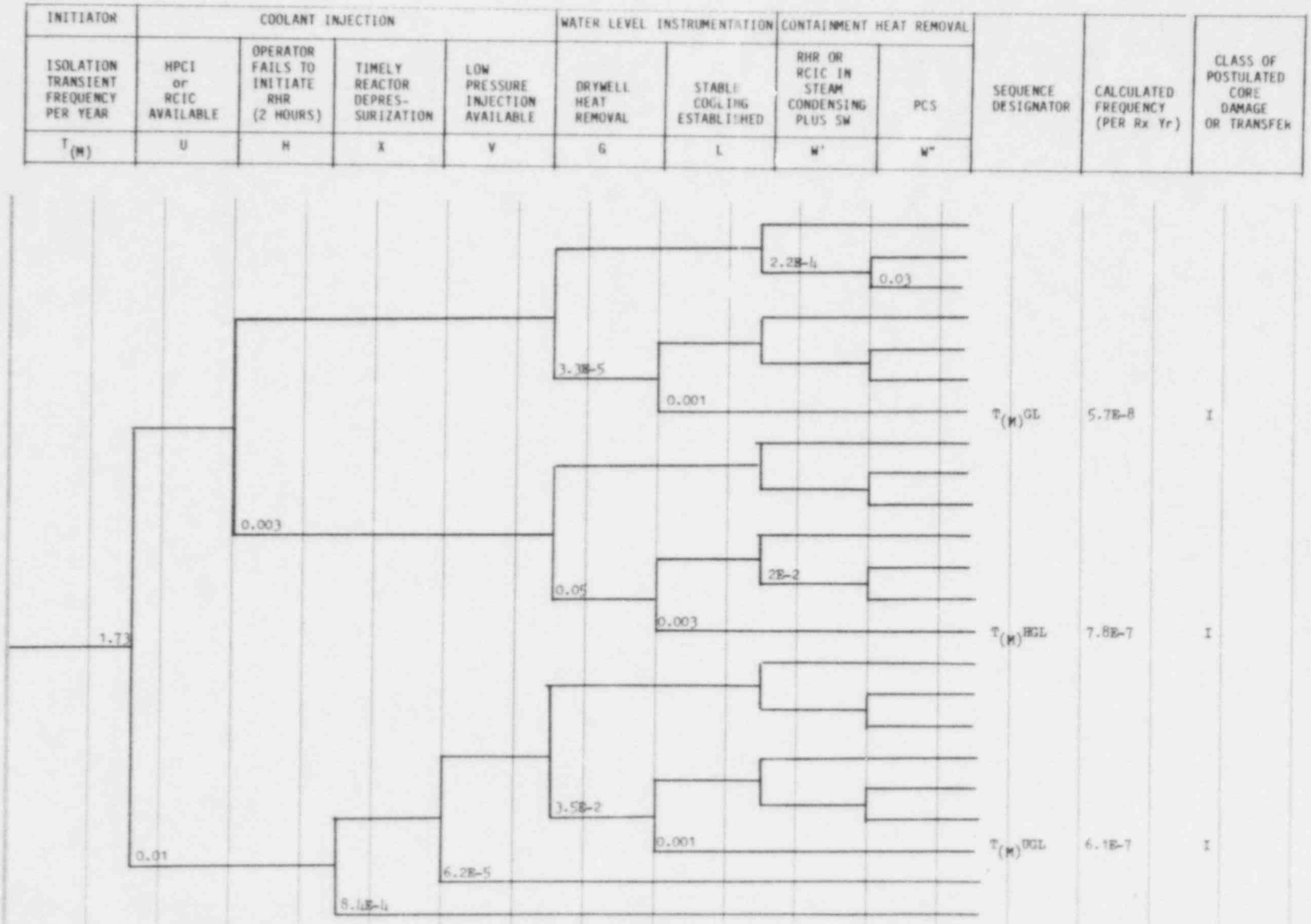


Table SP.3 Event Tree Diagram for Isolation Transients with Loss of Drywell Cooling (Sheet 2 of 2)

Table 5F.4 Loss of Offsite Power Event Tree with Water Level Measurement Implications (Sheet 1 of 2)

T_E = 0.15: LOOP initiator frequency for SNPS. Derived by BNL in Section 4.1.3.

I = 0.37: Recovery of offsite power within 0 to 2 hours. The value for recovery before 1/2 hour is used for this time phase. Derived in Section 4.1.3.

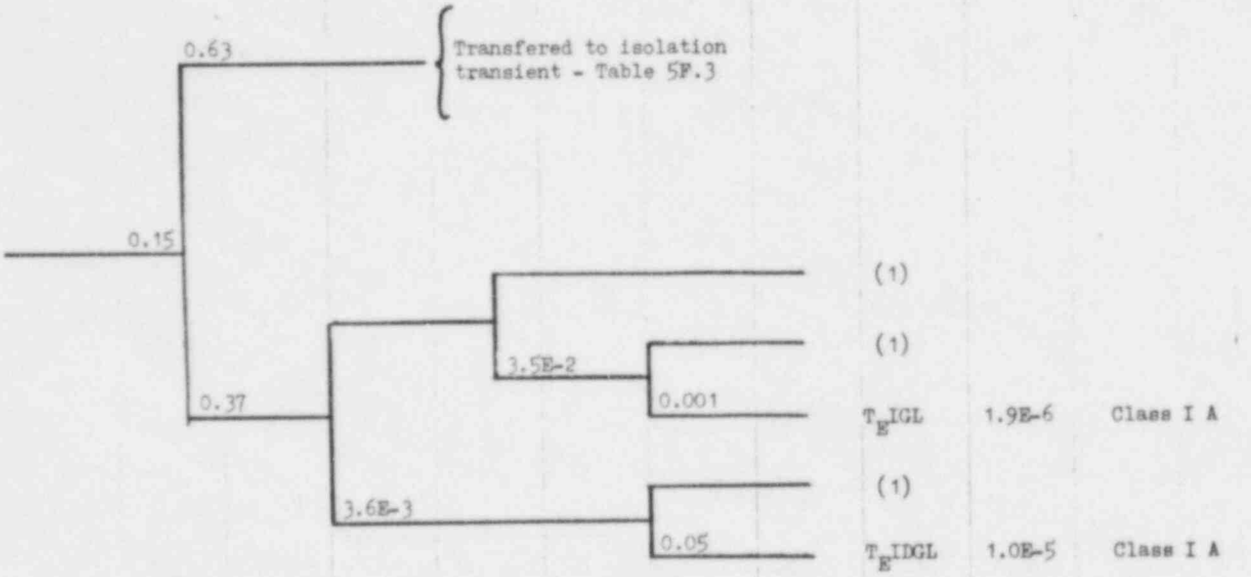
D = 3.6E-3: Common-mode failure of Divsion I and II diesel generators. Derived in Section 4.2.2. Includes recovery of diesels within 1/2 hour.

G = 0.035: Reestablishing drywell cooling following diesel generator initiation. This is 0.05x0.7, where 0.05 is for containment spray initiation and 0.7 is for manual transfer of coolers to emergency bus and their start before high pressure in drywell isolate the coolers.

L = 0.001: Same as in event tree of Table 5F.2.

= 0.05: Failure to maintain the core covered due to loss of water level measurement indications in the control room. The SNPS-PRA used a similar value--see discussion in Section 5F.2.2.

INITIATOR			WATER LEVEL INSTRUMENTATION				
LOSS OF OFFSITE POWER FREQUENCY	RECOVERY OF OFFSITE POWER AT 1/2 HOUR	DIESEL AVAILABILITY DIVISION I or DIVISION II	DRYWELL HEAT REMOVAL	STABLE COOLING ESTABLISHED	SEQUENCE DESIGNATOR	CALCULATED FREQUENCY (PER Rx Yr	CLASS OF POSTULATED CORE DAMAGE OR TRANSFER
T_E	I	D	G	L			



(1) These are developed in Appendix 5B event trees.

Table 5F.4 Loss of Offsite Power Event Tree with Water Level Measurement Implications (Sheet 2 of 2)

Table 5F.5 Event Tree Diagram for Loss of Coolant and Loss of Drywell Cooling (Sheet 1 of 2)

Small LOCA

S_2 = 8E-3: Frequency of small LOCA as evaluated in the SNPS-PRA.

C = 1.E-5: Failure to scram probability.

Q = 0.11: Taken from event P of the turbine trip event tree (Table 5A.2) i.e., two SORVs. This is suggested by the SNPS-PRA because two SORVs are similar to small LOCA.

U = 0.01: Unavailability of HPCI and RCIC according to their combined fault tree.

X = 8.4E-4: Unavailability of ADS.

V = 6.2E-5: Unavailability of low pressure injection (LPCI, LPCS, condensate).

G = 0.05: High drywell pressure isolation under small LOCA conditions leaves the containment sprays as the only drywell cooling option.

L = 0.001: Failure to follow procedures of maintaining level 3.

W = 0.1: Values for W taken from small LOCA event tree diagram (same as IORV - Appendix 5A, Table 5A.12).

Medium LOCA (only when different from the above)

S_1 = 3E-3: Frequency of medium LOCA as evaluated in the SNPS-PRA.

Q = 1.0: No recovery of FW or PCS is assumed for the short term for medium LOCA.

U = 0.1: Only HPCI is assumed capable of supplying sufficient water to maintain core covered with sufficient water level (see Section 2.1.2).

W = 0.12: See medium LOCA event tree diagram (Appendix 5C).

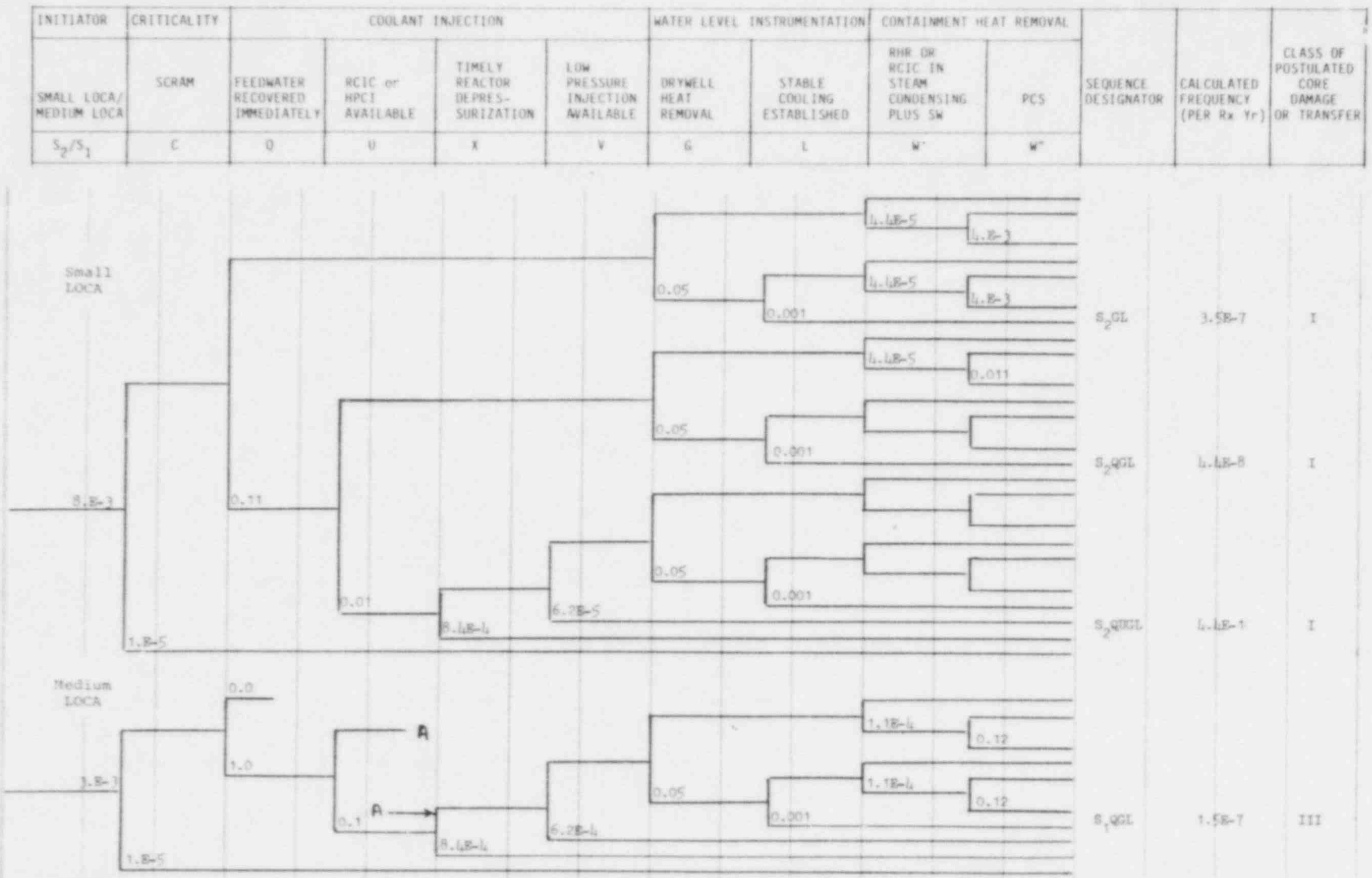


Table SP.5 Event Tree Diagram for Loss of Coolant and Loss of Drywell Cooling
(Sheet 2 of 2)

APPENDIX 5G

EVENT TREE ANALYSIS OF OTHER POSTULATED ACCIDENT INITIATORS

The SNPS-PRA analyzed low frequency transients of special interest to the SNPS:

a) Release of excessive water into the reactor building,

b) Loss of DC power to a bus,

c) Loss of reactor building service water system,

d) Failures in the reactor water level measurement system,

e) Loss of drywell cooling.

Items (d) and (e) were analyzed in Appendices 5E and 5F, and the first three items are addressed in this section. Item (a) was reviewed by BNL in great detail and discussed in a separate report[1]; therefore, only a summary of its event trees' description and the review findings will be given here. The intention is to keep this report as complete as possible in covering sequences of SNPS which may contribute to core damage, but the detailed basis for the summarized results will be found in the reference.

Items (b) and (c) are reviewed in this appendix and their revised event tree diagrams are presented, with the basis for any apparent differences between the BNL and the SNPS approaches or quantifications.

5G.1 EVENT TREE EVALUATION OF SEQUENCES FOLLOWING A POSTULATED RELEASE OF EXCESSIVE WATER IN ELEVATION 8 OF THE SNPS REACTOR BUILDING

The SNPS reactor building surrounds the Mark II containment structure. Most of the safety-related equipment is located throughout the reactor building, with the largest concentration in Elevation 8, the lowest level. All the ECCS pumps are located in the building at Elevation 8 in a large cylindrical compartment. Such an arrangement has the benefits of good maintenance and natural circulation capability on loss of equipment compartment area cooling or ventilation. However, it entails a remote possibility of a common-mode failure disabling all the equipment in the Elevation 8 compartment, such as flooding. This was studied in detail in the SNPS-PRA Appendix G and Section 3.4.4.1.

The critical height of water assumed for HPCI and RCIC failure is ~2 ft. above the floor, and for low pressure ECCS failure ~4 ft. The sump pump capacity is sufficient for flooding rates of about 500 gpm or less. If the flooding rate is larger, the water level may increase rapidly.

The BNL review, which is summarized in a report[1], comprises two parts: the first concerns the likelihood of occurrence of a flood event inside the SNPS Reactor Building, and the second, evaluates the effects of flood events on core damage frequency.

5G.1.1 Flood Initiation Frequency

The assessment of the flood initiator frequency is based on the consideration of experiential data for the estimation of various component failure rates and on Markov models, which describe the stochastic behavior of these components. Two types of flood initiators are considered: maintenance-induced and ruptured-induced.

A flood can be initiated during maintenance of the Emergency Core Cooling System (ECCS) or any other system for which the maintenance process requires dismantling of a component, while an isolation valve is inadvertently opened. The components that contribute to these are valves, pumps, and heat exchangers. Six states are defined in the Markov model, characterized as follows: (1) the component is available; (2) it is unavailable; (3) it is in maintenance, while the reactor is in operation with isolation valve breaker left in place; (4) the component is in maintenance while the reactor is in operation with isolation valve breaker out; (5) the component is in maintenance while reactor is shut down; and, lastly, (6) the flood initiated state.

Transition rates between states, either failure rate or repair rate, are derived from experiential data. The maximum likelihood estimators for the failure rates are evaluated to be 5.7×10^{-5}/hr for turbine-driven pumps and 3.3×10^{-6}/hr for motor-driven pumps based on LER data. The mean times to repair were assumed to be 100 hours and 50 hours for the turbine-driven pumps and motor-driven pumps, respectively. Tests are assumed to be performed on a quarterly basis. The allowable outage times are 14 days and 7 days for turbine-driven pumps and motor-driven pumps, respectively. The probability that the isolation valve breaker is not racked out is assessed to be 10^{-2}, and the rate for its inadvertent operation in the control room is 10^{-4}/hr.

Floods induced by rupture of pressure boundary of water systems are also evaluated in the BNL reassessment. Three types of components are considered: piping, valves, and pumps. It is assumed that when a component fails, it fails in such a way that a catastrophic rupture will occur; that is, the component transits first to a rupture-vulnerable state and then, when a demand occurs, it ruptures. A Markov model that describes this stochastic behavior has been developed. The model includes four transition paths through which a rupture-vulnerable state can transit to a flood initiated state. Three of the paths lead to either turbine trip, main steam isolation valve (MSIV) closure, or manual shutdown. Testing of the system will also result in a flood initiated state with a manual shutdown.

The rate of catastrophic rupture failure in a system is calculated as the summation of piping failure rate weighted by the length of system piping, the failure rate of valves weighted by the number of valves, and the failure rate of pumps weighted by the number of pumps. The rupture failure rate was estimated to be 5.3×10^{-7}/hr, based on the examination of LER data. As in the SNPS-PRA it was assumed that one out of twenty ruptures will produce flooding of the size that is of concern to this analysis. The valve rupture and pump rupture failure rates are based on the WASH-1400 values. A total of 13.7 transients per year were assumed for a BWR.

When the maintenance-induced flood Markov model was quantified, the total flood frequency resulting from the maintenance of the reactor core isolation

cooling (RCIC) system, the high pressure coolant injection (HPCI) system, the low pressure core spray (LPCS) system, the low pressure coolant injection (LPCI) system, and the service water system was found to be in the order of $4x10^{-4}$. Similarly, when the rupture-induced flood Markov model was quantified, the total frequency of a plant transient or a manual shutdown resulting from rupture-induced flooding was found to be on the order of $6x10^{-4}$.

5G.1.2 Evaluation of Core Damage Frequency

The flooding initiators' frequencies may appear to be relatively small compared with the plant transient frequency of more than 13 transients/year. However, flooding events in the SNPS are assumed potentially to affect redundant systems simultaneously.

The review identified the suppression pool, the condensate storage tank, the reactor primary system, the fire protection system storage tank, and the ultimate heat sink as potential water sources of flood events. Various flow rates were postulated for the different flood initiators. A time-phased event tree approach was used to model the impact of the flood progression and to include the contribution from flooding below the 3'-10" level (this is shown in Ref. 1). Four time-phase periods were defined. Phase I is the period during which, if the flood is arrested, no damage has been incurred upon any ECCS equipment and a manual shutdown will follow. Phase II is the period when potential electrical shorts would occur, possibly leading to the failure of motor control centers. Phase III assumes the failure of both HPCI and the RCIC. Finally, Phase IV assumes the failure of all ECCS equipment. The analysis considered the ability of the operator, and the time available, to respond to the flood alarms and to identify and arrest the flood before it reaches Phase IV, when all equipment is assumed lost.

The results of the time-phased event trees are presented in Table 5G.1 for each of the initiators considered, along with definitions of these initiators.

5G.1.3 Summary of the Results

The BNL review found the SNPS-PRA assumptions, methodology, and results to be reasonable. The analyses for the internal flood postulated much more severe scenarios than those of the Shoreham FSAR. BNL re-evaluated the flood precursor frequency[1], using recent LER data and a more detailed quantification methodology (Markov models). This methodology avoids some of the conservatism in the SNPS-PRA approach but it does not compensate the large increase in the flood precursor frequency, calculated by BNL, which resulted in the increase in the BNL calculated core damage frequency.

Similarly, on the basis of the PRA Procedure Guide[14], BNL reviewed the HEP analysis of the SNPS-PRA, and made only minimal changes. The time-phased approach utilized by BNL modeled the progression of the flood events better than did the SNPS-PRA functional event trees. This facilitated the inclusion of more operator recovery actions in the BNL model. However, it was found that 80% of the core damage frequency results from exceeding the higher water level critical height, i.e., 4 feet.

The results are summarized in Table 5G.2. Part A of this table provides a comparison of SNPS and BNL results. Part B shows the effect of the time-phased event tree approach. It is seen that the first three phases have the effect of increasing the results by about 20%. Part C shows the origin of the core damage frequency contributions by (a) "maintenance" or (b) "rupture".

Finally, the importance of the different initiators causing the excessive water release is shown in Table 5G.1 by their systemic origin. The rupture of LPCI lines is found to be most significant.

The BNL results are higher by a factor of ≈5 than those in the SNPS-PRA. This is attributable mainly to the increase in flood precursor frequencies, but also to the quantification of the condensate system injection as 0.1 rather than 0.01 in the SNPS-PRA (e.g., see Table 5A.2).

Table 5G.1 Summary of the Postulated Sequence Initiators Associated with the Potential Release of Water in the Reactor Building Elevation 8

Initiator		
Designator	Description	Core Damage Frequency Contribution*
Maintenance Related Initiators		
T_{FL1}	RCIC in maintenance	1.4E-8
T_{FL2}	HPCI in maintenance	3.7E-6
T_{FL3}	CS in maintenance	6.2E-7
T_{FL4}	LPCI in maintenance	2.7E-6
T_{FL5}	SW in maintenance	3.7E-7
Pipe Failure Related Initiators		
T_{FL6}	HPCI discharge break	7.0E-8
T_{FL7}	CS discharge break	8.6E-7
T_{FL8}	LPCI discharge break	7.3E-6
T_{FL9}	SW discharge break	1.4E-6
T_{FL10}	WFPS break	2.2E-8
T_{FL11}	RCIC suction break (maximum flow rate)	1.7E-8
T_{FL12}	HPCI suction failure (maximum flow flow rate)	5.0E-7
T_{FL13}	HPCI suction failure (large break)	3.1E-8
T_{FL14}	CS suction failure (maximum flow rate)	7.0E-7
T_{FL15}	CS suction failure (large break)	2.8E-7
T_{FL16}	LPCI suction failure (maximum flow rate)	1.1E-6
T_{FL17}	LPCI suction failure (large break)	1.2E-6
	Total =	2.08E-5

*From Table 3.3.3 of Ref. 1.

Table 5G.2 Core Damage Frequencies for Flooding Initiators (Reproduced from NUREG/CR-4049[1])

	Transient	Core Damage Class	SNPS-PRA	BNL Review
PART A	Manual Shutdown	I II	3.7E-8 8.4E-8	4.9E-7
	Isolation (MSIV Closure)	I II	2.5E-6 4.9E-7	1.8E-5
	Turbine Trip	I II	5.7E-7 2.0E-7	2.0E-6
	Total	I II	3.1E-6 7.8E-7	2.0E-5
PART B	Manual Shutdown	I II	3.7E-8 8.4E-8	4.6E-7
	Isolation (MSIV Closure)	I II	2.5E-6 4.9E-7	1.5E-5
	Turbine Trip	I II	5.7E-7 2.0E-7	1.7E-6
	Total	I II	3.1E-6 7.8E-7	1.7E-5
PART C	Manual Shutdown	Maintenance Rupture	3.9E-8 1.6E-7	4.1E-7 7.0E-8
	Isolation (MSIV Closure)	Maintenance Rupture	1.5E-6 1.4E-6	6.9E-6 1.1E-5
	Turbine Trip	Maintenance Rupture	0 6.7E-7	0 2.0E-6
	Total	Maintenance Rupture	1.6E-6 2.3E-6	7.3E-6 1.3E-5

5G.2 LOSS OF 125 V DC EMERGENCY BUS DIVISION I (II)

The SNPS has three DC buses but only two of them (Divisions I and II) can cause a transient if failed. The Division III bus is supplying only a small part of the safety system, namely the LPCI and the RBSWS.

The level 1, 2, and 8 initiation signals are fed from either DC Division I or II. The RCIC is supplied only from DC Bus I, and the HPCI is supplied solely from the Division II bus.

The MSIV valves are fed by two solenoid valves, each connected to one division. Thus they will remain open on loss of DC, but with a higher probability of failure because of the decrease in redundancy.

The LPCI pumps are fed as follows: one pump from Division I or III, another pump from Division II or III, and the third and fourth pumps from Division III. Because of this logic the four pumps of LPCI remain available upon loss of DC Division I or II. If Division III is lost, a transient will not occur in SNPS. Only two LPCI pumps will become unavailable but all other ECCS pumps and initiation logic will remain effective, which makes this case a minor sequence. The two LPCS pumps are each fed by one of the Division I or II buses.

The modeling of the above plant features was adequately performed in the SNPS-PRA and presented in an event tree diagram for loss of a DC bus (Figures 3.4-44 page 3-277 of the PRA). BNL found this reasonable and did not make changes beyond those needed for consistency with other event tree diagrams (see Table 5G.3).

The frequency of the initiating event was based in the PRA on LER information derived from NUREG-0666[15]. This is a reasonably conservative approach. To the frequency of $3x10^{-3}$ derived from that report, an additional $1.4x10^{-3}$ was added to cover the possibility of DC bus induced failures upon another transient or manual shutdown. The SNPS-PRA estimated the induced failure frequency as $1.4x10^{-4}$/transient. This is reasonable considering that loss of a DC bus may not cause a plant transient or immediate shutdown. When this was multiplied by 10 transients per year, the above value was obtained. The resulting frequency of $4.4x10^{-3}$ per year for loss of a DC bus was accepted by BNL.

The analysis by SNPS shows that the CMF of two buses is more important than the loss of a single bus. The conditional probability of a second bus failure was calculated to be 10^{-3} and discussed in Appendix A.7 of the PRA. However, the final value given there has a numerical error.

The considerations given there and the additional information provided by SNPS in its response to a BNL question[6] are judged to be adequate. These considerations include the following:

(a) SNPS has no inter-tie between the DC buses.

(b) SNPS has backup battery charger.

(c) SNPS adopted the policy of no scheduled DC bus maintainance during power operations.

In summary, BNL considered the modeling of this sequence in the PRA to represent adequately its contribution to the SNPS core damage frequency. The results show that loss of DC sequences contribute almost exclusively to Class I. The core damage frequency is moderate and amounts to ~2.5E-6 in both the SNPS-PRA and the BNL review.

Table 5G.3 Event Tree Diagram for Loss of 125 V DC Bus
(Sheet 1 of 2)

T_D = 4.4E-3:	Frequency of initiator having two contributions: (1) 3.E-3 from NUREG-0666[15] based on LER with 0.5 recovery. (2) 1.4E-3 derived from estimated frequency of transient induced loss of DC bus, considering 10 transients per reactor year.
D_I = 0.001:	Probability of second bus failing given first has failed. Based on NUREG-0666 with modifications to reflect better design and improvements of the SNPS DC system.
Q = 0.082:	Source of scram, if it occurs, would most probably be a turbine trip. Thus, values from turbine trip event tree diagram in Appendix 5A.1 are used.
U = 0.1:	A failure of Division I is assumed, which makes RCIC unavailable. The HPCI unavailability is 0.1 as evaluated from the HPCI fault tree.
X = 3.6E-3:	One ADS actuation channel is lost on loss of a DC bus, which increases ADS failure probability slightly.
UX = 6.4×10^{-4}:	The combination of failures of the "UX" function as calculated by SNPS-PRA.
V'•V" = 0.001:	The loss of one DC bus causes an increase in LPCI and LPCS failure probability which is estimated, from the fault trees, to result in about 10^{-3} instead of 6.2×10^{-4}
V''' = 0.1:	Failure of operator to control condensate pump injection.
W' = 1.1 10^{-4}:	Loss of DC is assumed to have no impact on RHR. RCIC steam condensing mode is assumed unavailable.
W" = 0.013:	The values for PCS unavailability are taken from the turbine trip event tree diagram.

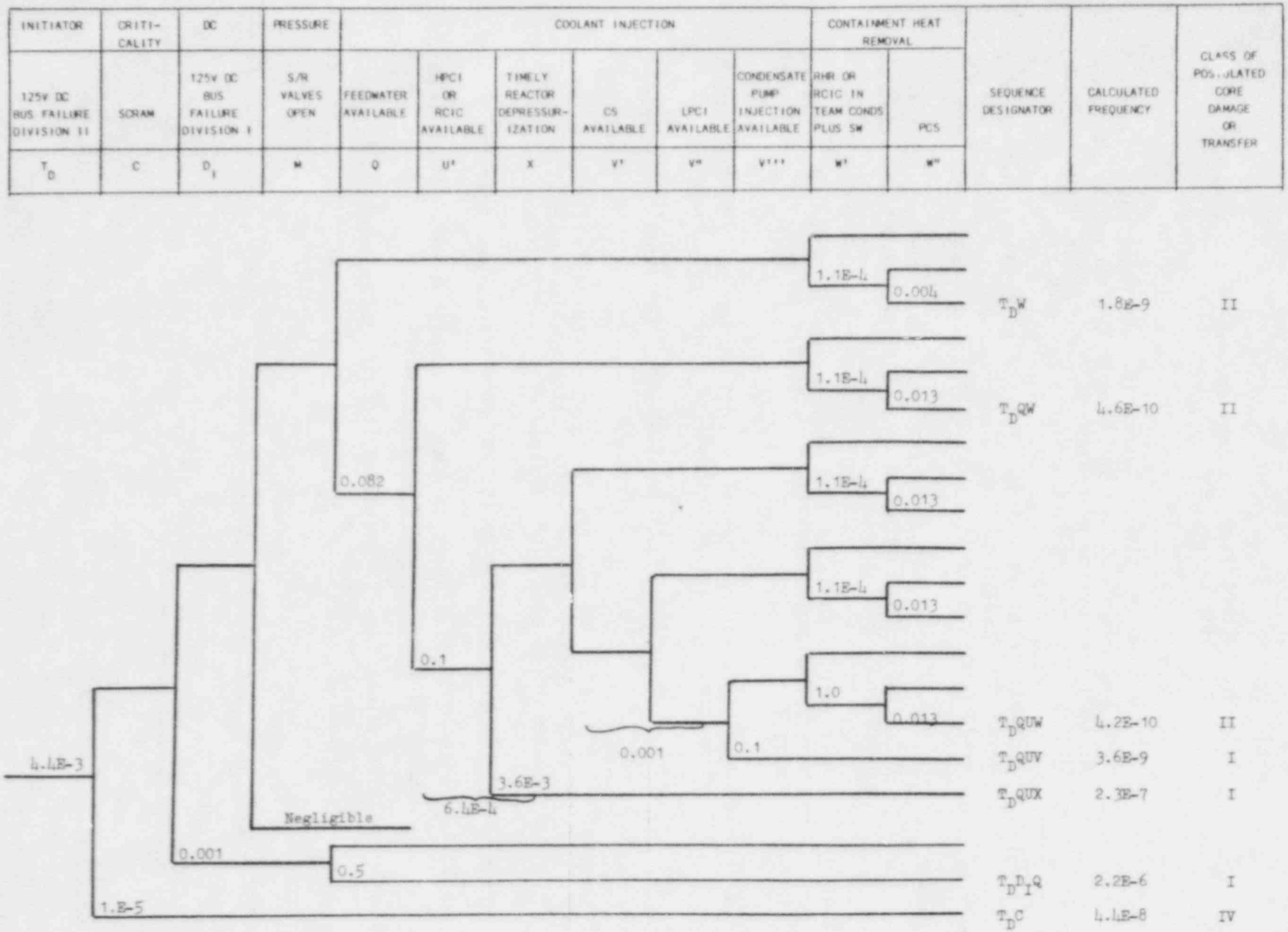


Table 5G.3 Event Tree Diagram for Loss of a 125V DC Bus
(Sheet 2 of 2)

5G.3 LOSS OF REACTOR BUILDING SERVICE WATER INITIATOR

The reactor building service water system (RBSWS) is a four-pump, two-loop normal operation system that assumes emergency cooling after shutdown, to provide cooling for safety systems. It provides raw cooling water to the following support or frontline systems:

(a) RHR heat exchange.

(b) Drywell coolers.

(c) Diesel engine coolers.

(d) Reactor Building Closed Loop Cooling Water (RBCLCW) heat exchanger. This system cools the RHR pump seal coolers and RCP seal coolers.

(e) Reactor Building Standby Ventilation System (RBSVS) chilled water condenser. The RBSVS cools the reactor building (secondary containment) and cools the ECCS pump area at Elevation 8 by four local unit coolers.

A cross connect having two MOVs (apparently locked closed--see FSAR) which can be operated from the control room connects the three-pump one-loop turbine building service water system (TBSWS) to the RBSWS header. The SNPS-PRA assumed that this system can be effective in 75% of the cases of loss of RBSWS because it uses other pumps having another intake. It also assumed that the operator will successfully complete the transfer within 0.5 hours with less than 0.01 failure probability. The SNPS-PRA assumed the TBSWS unavailability to be 0.01 and the RBSWS unavailability to be 2.1×10^{-4}.

The frequency of loss of RBSWS is calculated by the SNPS-PRA as a sum of two contributors:

(a) Assumed initiator frequency of 1/400 considering an incipient event of loss of SWS suction has occurred in the past due to contamination. Several events have been experienced in which partial loss of SWS occurred.

(b) Failure of RBSWS during a shutdown, caused by another transient initiator--mainly turbine trip or manual shutdown. This is calculated to contribute $2.1\times10^{-3} = 10\times2.1\times10^{-4}$, where it is assumed that the probability of transient-induced loss of RBSWS is 2.1×10^{-4} in about 10 transients per year.

BNL considers the above approach to be conservative because no such event has, in fact, occurred in a BWR plant. Furthermore, the use of the value of 2.1×10^{-4} for RBSWS induced failure is not explained in the PRA and may have been used for screening purposes. Also unexplained is a value for RBSWS immediate recovery of 0.5, which may be an underestimation of the time needed to recover from SWS suction problems.

In estimating RBSWS initiator frequency, BNL used the same two contributors with reduced frequencies:

(a) BNL assumed 1/600 (600 being reactor years of BWRs and PWRs) as SWS initiator frequency. Recovery of this is assumed to be difficult because according to LERs it may include mainly suction clogging. Furthermore, it is assumed by BNL that 25% of this initiator can affect the TBSWS intake as well (see Table 5G.4).

(b) BNL estimated that the RBSWS transient-induced failures, or the RBSWS failure probability during an incident, would be smaller than assumed in the SNPS-PRA. This is based on the following arguments:

(1) Only failures lasting for 10 hours or more after the initiation of a transient are assumed to be able to cause high temperature in containment and thus considerably degrade the ECCS. This can be seen from the relatively low contribution to core damage from event tree diagram Table 5G.5 Sheet 2 compared with that from Sheet 3. The first treats the time phase 0 to 10 hours after the initiation of the incident and the latter, 10 to 20 hours.

(2) The probability of transient-induced RBSWS is considered low, because turbine trip or manual shutdown does not cause this system to be in maintenance or cause SWS suction problems, which are the main reasons for RBSWS unavailability. This means that only a small part of the factors leading to SWS unavailability may contribute to the possibility that the RBSWS will become unavailable during a transient.

Based on the above, BNL used one quarter of the SNPS-PRA value, i.e., $0.25 \times 2.3\times10^{-4} \times 13.7 = 7.8\times10^{-4}$, where 2.3×10^{-4} is the unavailability calculated by use of the SWS fault tree (Table 3.1 and Section 3.3.2.5) and 13.7 is the number of transients in the BNL reassessment (Section 4.1).

The overall BNL frequency becomes 2.4×10^{-3}, which is practically the same as calculated by the SNPS-PRA (with the 0.5 recovery) from different considerations. The recovery probability of 0.5 used in the PRA was included in the BNL review in a different way, as shown in Table 5G.4.

The SNPS-PRA treated the loss of SWS as a separate low frequency transient taking into account the impact of the dependency of the above mentioned list of subsystems on the supply of SW cooling water. The following are the main occurrences following loss of SWS which were considered in the SNPS-PRA analysis:

(a) The operator opens the cross-connect valves (035 A and B) between the turbine building service water pumps and the SWS pipe, distributing the cold water to the different coolers and heat exchangers. This can be done from the control room.

(b) Without service water, drywell coolers will be unavailable, and the temperature in the drywell will increase rapidly (15 minutes--see Table 5.F.1) to 300°F and cause high drywell pressure reactor trip.

(c) Without service water the room coolers would be unavailable and RBSVS will be isolated so that the temperature in areas where steam-driven or motor-operated ECCS pumps are working will rise. It is assumed that HPCI, LPCI, and LPCS will not be available for injection after a long time without room cooling, but RCIC is assumed to be available with lower failure probability.

(d) The RBSVS would be operating in the recirculation mode discharging 1160 cfm through the filtered stack and recirculating more than 45,000 cfm in secondary containment. This RBSVS operation redistributing the heat loads in the containment is assumed to limit the secondary containment temperature to 120° to 150°F, which is 15°F below the MSIV closure set point on high steam tunnel temperature. Thus, there is a probability of MSIVs remaining open without the need for the operator to bypass actuation logic. This may allow feedwater injection and PCS cooling.

(e) The SNPS emergency procedure for the case of loss of SWS instructs the operator to restore room cooling via the cross-connect valves from the turbine building SWS and to maintain MSIVs open and PCS available.

(f) For cases with the MSIVs closed, the suppression pool and drywell heatup require the operator to follow procedures and depressurize the reactor vessel slowly. It is assumed that the failure probability of the ADS will be above normal because of high drywell temperatures and after ~10 hours without any cooling, due to high pressure in drywell.

BNL used a time-phased event tree approach (see Table 5G.5) to take into better account the containment heat-up after 10 hours if containment cooling is not recovered before that time.

The results of the BNL review are contrary to those of the SNPS-PRA. BNL finds this event to be of significance. The results of the SNPS-PRA had numerical errors, or inconsistencies with the explanations of the values used, which made the core damage frequency small. If these errors are corrected, then the sequence becomes of significance also in the SNPS-PRA ($\sim 1 \times 10^{-6}$ for Class I and $\sim 3 \times 10^{-6}$ for Class II).

The results of the BNL review are higher by a factor of 20 than those given (uncorrected) for Class I in the PRA. The BNL approach to the analysis is different, as seen from the event trees. Because the RBSWS in the SNPS has moderate availability and the TBSWS can be used as backup only in a part of the situation because of possible commonalities, the contribution of this event becomes significant and warrants an emergency procedure which will clearly state the operator actions required to recover from the situation. As recognized by the SNPS-PRA, keeping the MSIVs open and the PCS available, following the depressurization procedures approximately to pressures which allow RCIC, FW, and condensate injection, as well as maintaining RCIC suction on CST, are key conditions for recovery.

INITIATOR FREQUENCY	DURING TRANSIENT OR INITIATOR	MAINTENANCE OR SUCTION	RECOVER RBSWS	TBSWS EFFECTIVE-NESS	SEQUENCE SUCCESS	CONDITIONAL PROBABILITY
T_{SW}	A	MHS	R_S	T_S		
	0.5	0.75	0.5		OK	
				0.5	OK	
				0.5	--	0.09
		0.25		0.75	OK	
				0.25	--	0.03
	0.5	0.25	0.5		OK	
			0.5	0.5	OK	
				0.5	--	0.03
		0.75		0.75	OK	
				0.25	--	0.09
						0.24

Table 5G.4 Conditional Probability "T" the RBSWS or TBSWS Would be Available Following Loss of RBSWS Initiator

Table 5G.5 Event Tree Diagram for Loss of Reactor Building Service Water Initiator (Sheet 1 of 3)

$T_{SW} = 2.4\ 10^{-3}$: This is the frequency of loss of SWS, which includes two contributors: (1) Loss of SWS initiator frequency based on estimated frequency of occurrence, and (2) Loss of SWS during any other transient (13.7 transients per year). See Section 5G.3 for discussion.

P = 2.E-3: SORVs failure to close has a small contribution, and is not developed further.

T = 0.24: Successful cross-tie of turbine building SWS to the reactor building system. A separate functional event tree was modeled (Table 5G.4) to evaluate this probability.

S = 0.3: The availability of the PCS as both a coolant injection function and a heat sink is dependent upon the prevention of MSIV closure, and the ability of FW to inject at low reactor vessel pressure. The proposed value of 0.7 for MSIV availability takes into account:

(a) A probability that the MSIVs will not close on high temperature in steam tunnel in view of loss of reactor building coolers but successful mixing by RBSVS.

(b) A probability that the operators open MSIVs as required by emergency procedures for this case.

(c) A probability that condenser/circulating water pumps will be made available by the turbine building SWS cooling their seals.

(d) A probability that feedwater will be able to provide injection at low pressure consistent with the depressurization procedures on high drywell and suppression pool temperatures.

(e) A small probability that MSIV solenoids and cables may degrade and fail to remain open during the extended time of their exposure to 300°F in the drywell. (They are qualified to 300°F for several hours.)

Therefore, the unavailability value BNL chose to use is twice the SNPS-PRA value.

Table 5G.5 Event Tree Diagram for Loss of Reactor Building Service Water Initiator (Sheet 1 of 3 Continued)

U' = 0.08: The normal RCIC unavailability is 0.07. The increase is due to the possibility that RCIC will be isolated on high area temperature in the reactor building as a result of 10 hours of operation without unit coolers.

U' = 0.2: After 10 hours of operation, if another 10 hours of successful injection are required with RCIC, it is thought that the success probability of RCIC will decline because of the high temperature in the containment secondary building, which can cause high area temperature RCIC isolation. Furthermore, a need may arise to transfer RCIC from CST to suppression pool suction, which at this time will be at 240°F or more. Should this occur, RCIC will fail shortly.

U" = 0.2: For the first 10 hours, the area temperature isolation may be a common-cause failure of HPCI and RCIC. HPCI, in addition, will start to take suction from the suppression pool after 2 to 2.5 hours, and with temperature as high as 200°F the probability of HPCI lube oil to fail increases.

U" = 1.0: The HPCI is assumed to fail because of high suppression pool temperature after 10 hours.

X = 0.002: This is twice the normal ADS unavailability, and accounts for possible failure of the solenoids due to high drywell temperatures.

X = 0.1: The ADS is assumed to be degraded and to have a high probability of failure after 10 hours without drywell cooling. (This is consistent with the LOOP event for 4 to 10 hours.)

V = 0.002: During the first 10 hours, high temperature in the ECCS area may cause LPCI or LPCS isolation. The unavailability of this system was increased by a factor of 3.

V = 1.0: After 10 hours, i.e., in the time phase from 10 to 20 hours, the area temperature is assumed to be high enough to isolate the systems. Furthermore, the LPCI pumps' seals are cooled either by suppression pool water or by RBCLCWS. The first will have excessive temperature, and the latter is unavailable, therefore, no LPCI is assumed after 10 hours.

Table 5G.5 Event Tree Diagram for Loss of Reactor Building Service Water Initiator
(Sheet 1 of 3 Continued)

$V''' = 0.1$: The condensate system is assumed to be available with 0.1 probability. The 0.1 has two contributors: (1) The result of a significant number of operator actions required to establish this mode, and to control condensate flow rate and hotwell coolant replenish; however, more time is available to the operator action in the BNL case. (2) An increase in unavailabiity due to the probability of TBSWS being unavailable for the cooling of the condensate pump lube-oil heat exchangers.

$R_S = 0.6$: This is the recovery probability of RBSWS calculated as exp(-10/19) = 0.6. It is applied twice, after 10 and 20 hours.

$W' = 1.1 \times 10^{-4}$: Normal RHR unavailability.

$W' - 0.36$: Recovery of RHR in 20 hours.

$W' = 0.6$: The value of R_S is given here.

$W'' = 0.03$: The value of PCS recovery after MSIV closure is taken from the MSIV closure transient, discussed in Appendix 5A.3.

$W'' = 4 \times 10^{-3}$: Normal PCS unavailability.

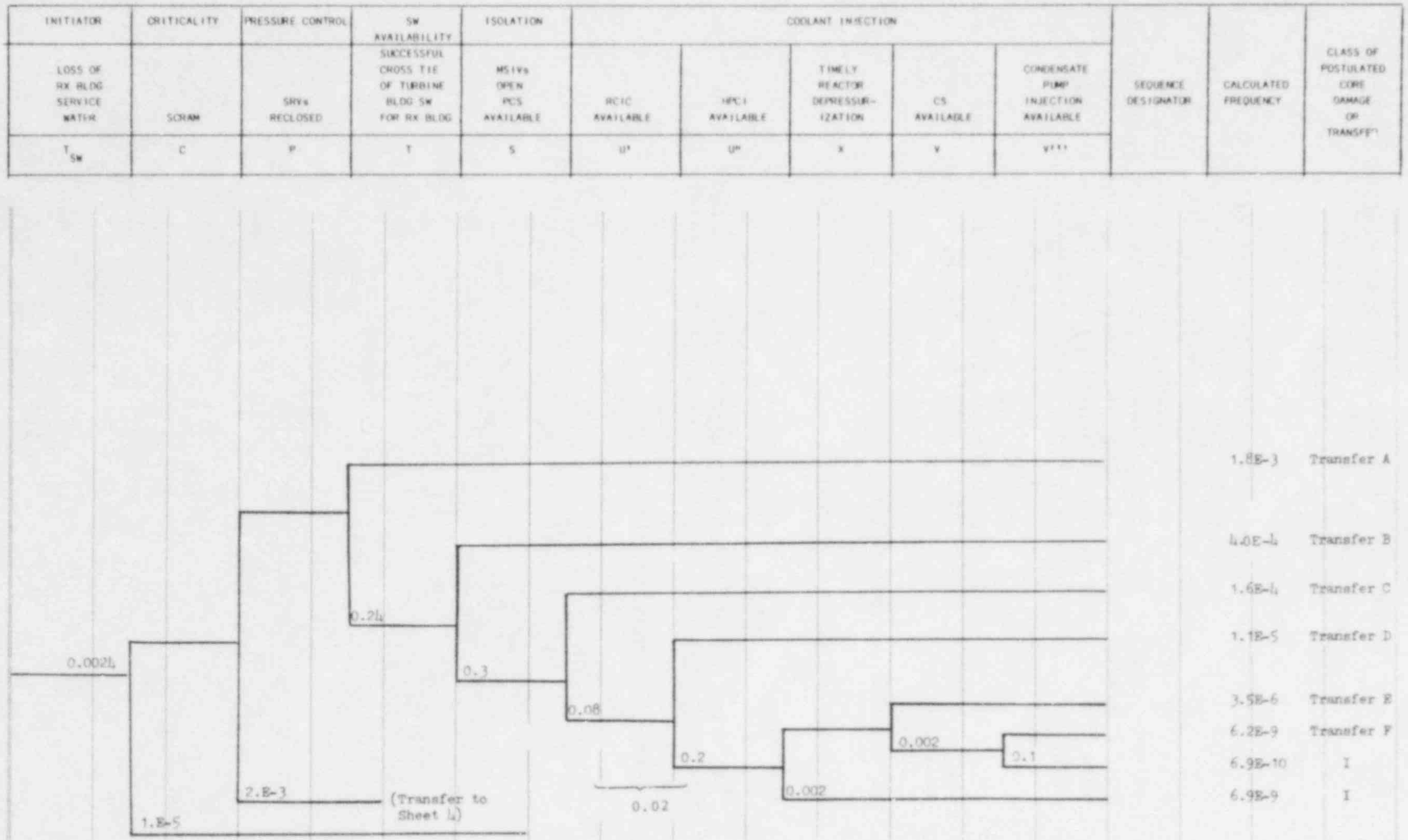


Table 5G.5 Event Tree Diagram for Sequences Following a Loss of Service Water Initiator (Sheet 2 of 3)

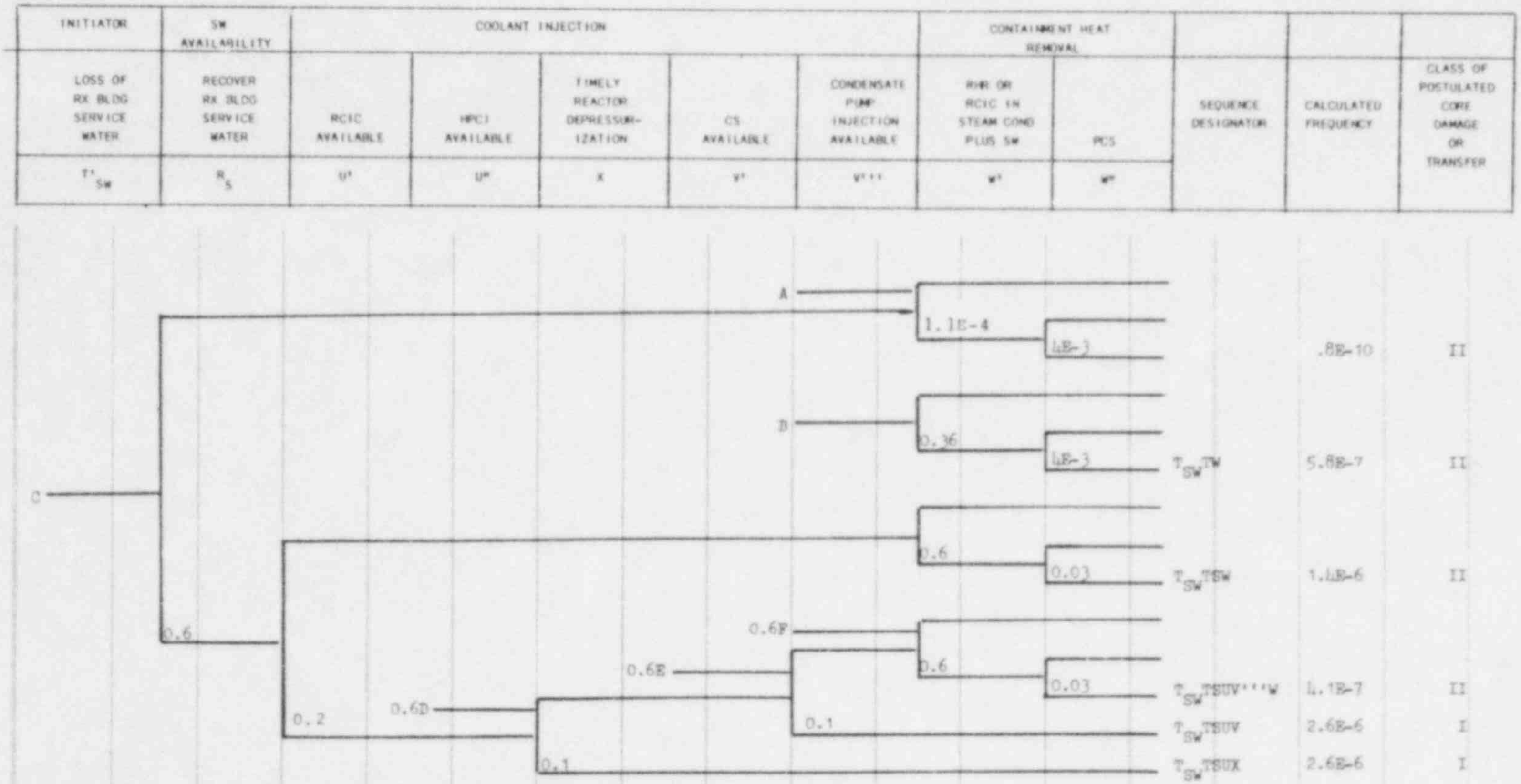


Table 5G.5 Event Tree Diagram for Sequences Following a Loss of Service Water Initiator (Sheet 3 of 3)

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A review of the Probabilistic Risk Assessment of the Shoreham Nuclear Power Station was conducted with the broad objective of evaluating its risks in relation to those identified in the Reactor Safety Study (WASH-1400). The scope of the review was limited to the "front end" part, i.e., to the evaluation of the frequencies of states in which core damage may occur. Furthermore, the review considered only internally generated accidents, consistent with the scope of the PRA. The review included an assessment of the assumptions and methods used in the Shoreham study. It also encompassed a re-evaluation of the main results within the scope and general methodological framework of the Shoreham PRA, including both qualitative and quantitative analyses of accident initiators, data bases, and accident sequences which result in initiation of core damage. Specific comparisons are given between the Shoreham study, the results of the present review, and the WASH-1400 BWR, for the core damage frequency. The effect of modeling uncertainties was considered by a limited sensitivity study so as to show how the results would change if other assumptions were made. This review provides an independently assessed point value estimate of core damage frequency and describes the major contributors, by frontline systems and by accident sequences.

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