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VOLUME 1

A REVIEW OF BWR/6 STANDARD PLANT PROBABILISTIC RISK ASSESSMENT: VOL. 1: INTERNAL EVENTS, CORE DAMAGE FREQUENCY

N. A. HANAN, K. K. SHIU, R. KAROL,
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DEPARTMENT OF NUCLEAR ENERGY, BROOKHAVEN NATIONAL LABORATORY
UPTON, NEW YORK 11973

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ABSTRACT

A review of the Probabilistic Risk Assessment (PRA) of the BWR/6 Standard Plant (GESSAR-II) was conducted with the broad objective of evaluating the contribution of the internally generated accidents to the frequency of core damage. The review included a technical assessment of the assumptions and methods used in the GESSAR-II PRA study. The BNL staff reevaluated the main results of the study within the scope and general methodological framework, including both qualitative and quantitative analyses of accident initiators, and accident sequences which result in core damage. The review assessed the relative importance of various accident sequences as well as systems with regard to their contribution to the core damage frequency. The effect of uncertainties was considered throughout the review process, and the uncertainty bands for the core damage frequency were quantified.

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NOMENCLATURE

A	Large LOCA
ADS	Automatic depressurization system
B_0	LOCA - induced loss of offsite power
C	Scram failure
C_A	Alternate rod insertion failure
C_M	Mechanical failure to scram
C_E	Electrical failure to scram
C_I	Scram initiation failure
C_1	Failure of one standby liquid control loop
C_{21}	Failure of second standby liquid control loop, given C_1
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
CM3D1	Recovery of common mode failure of 3DGs in 30 minutes
CM3D2	Recovery of common mode failure of 3DGs in 2 hours, given CM3D1
DG1F2	Failure of diesel generator No. 1 and failure to recover in 2 hours
DG1F2(IF)	Independent failure
DG1F2(CM)	Common mode failure of DG No. 1 and DG No. 2
EPS	Electrical power system
ESW	Essential service water system
FW	Feedwater system
HPCI	High pressure core injection system
HPCS	High pressure core spray system
I_M	Two or more IORV, given one
L1	Failure to recover offsite power in 30 minutes
L2	Failure to recover offsite power in 2 hours
L_H	Level control failure
LPCI	Low pressure core injection system
LPCS	Low pressure core spray system
M	Failure to maintain reactor pressure
PCS	Power conversion system

P_1	One stuck open relief valve (SORV)
P_2	Two or more SORV, given one SORV - when there is no P_1 in the sequence, P_2 means two or more SORV
P_A	ADS inhibit failure
Q	Feedwater system
R	Redundant reactivity control system
RPV	Reactor presure vessel
RHR	Residual heat removal system
S_1	Intermediate LOCA in drywell
S_{1x}	Intermediate LOCA in containment
S_{1xx}	Intermediate LOCA outside containment
S_2	Small LOCA in drywell
S_{2x}	Small LOCA in containment
S_{2xx}	Small LOCA outside containment
SSWS	Standby service water system
T_{DC}	Loss of two DC buses (Divisions 1 and 2)
T_E	Loss of offsite power
T_F	Isolation
T_{FA}	Isolation ATWS
T_I	Inadvertent open relief valve
T_M	Manual shutdown
T_T	Turbine trip
U_H	High pressure core spray system
U_R	Reactor core isolation cooling system
URHE	Manual control of reactor core isolation cooling system
U_{HR}	High pressure core spray and reactor core isolation cooling systems
V_4	Low pressure core cooling systems (includes LPCI and LPCS)
V_C	Condensate injection
X	Depressurization (via automatic depressurization system or manual)
W	Containment heat removal function (includes residual heat removal system and power conversion system)
W_W	Containment spray system
W_X	External water supply

EXECUTIVE SUMMARY

This review of the Probabilistic Risk Assessment of the BWR/6 Standard Plant, submitted as part of the General Electric Standard Safety Analysis Report, was conducted by Brookhaven National Laboratory under the sponsorship of the U.S. Nuclear Regulatory Commission. The review of the internally generated plant accidents which lead to core damage, began in July 1982 and involved five members of the Risk Evaluation Group at BNL. A companion review of the core melt phenomenolgy, fission product behavior, and offsite consequences was initiated at the same time by the Accident Analysis group at BNL. The results of that study are being issued in another volume in this series of reports. Thus, the results presented in this volume of the report refer to the delineation and quantification of accident sequences that can lead to core damage and are initiated by internally generated functional events ("internal" initiators) and loss of offsite power. In light of the ongoing review of the consequence analysis and the assessment of the risk from the environmental (or "externally" initiated) events, the results and conclusions reported herein should be considered only as a partial risk assessment. Upon receiving the comments and upon completion of the review of the consequence analysis and the "environmental events" risk assessment, BNL will issue a final report (NUREG/CR-4315, Vol. 4) presenting the results and conclusions of the review.

The broad objective of the review contained in this volume was to evaluate the GESSAR-II PRA with respect to the frequency of core damage, the main contributors to this frequency, the associated uncertainties, and all these for accidents initiated by functional events internal to the plant, as well as loss of offsite power. The review by Brookhaven included a technical assessment of the assumptions and methods used in the GESSAR-II PRA study. The review also included a reevaluation of the main results within the scope and general methodological framework of the GESSAR-II study. This included both qualitative and quantitiative analyses of accident initiators, data bases, and accident sequences which result in core damage.

The review process included seven meetings with General Electric and NRC staff and two formal rounds of (written) questions and answers. The character of the review process was highly interactive and has included (since the beginning of the review), in addition to the meetings, the submittal of a major revision of the PRA and a number of documents with supporting analysis for various modifications. These changes resulted, understandably, in some scheduling difficulties for the reviewers since the original review planning and schedule assumed the review of a single document. The GE staff was helpful and cooperative throughout the course of the review. Overall, the interactive review process was highly beneficial to the reviewers, and the resulting revisions considerably enhanced the value of the information developed in the GESSAR study.

The results and conclusions presented in this report might be altered by insights from and findings of the ongoing review of the consequence analysis and the risk assessment from environmental events generated externally (e.g., seismic) or internally (e.g., fires, floods) to the plant. In that sense, the results presented in this report should be considered preliminary in nature. This remark notwithstanding, the main conclusions of this review are the following:

a. Within the stated scope, the GESSAR-II PRA study is a reasonably good piece of work. GE performed a PRA following the basic approach and techniques of the Reactor Safety Study (event tree/fault tree methodology), but which accounted for some extensions of the state-of-the-art and for design differences between the Reactor Safety Study plant and the GE BWR/6 Standard Plant.

b. The Brookhaven reviewers believe that the GESSAR-II PRA study was reasonably successful in identifying the major failure combinations that can lead to core damage. The reviewers, furthermore, believe that if the GESSAR-II PRA study is modified in accord with the results of this review (presented in the main body of the report) it will more reasonably portray the major characteristics of the BWR/6 Standard Plant design with respect to the risk to the integrity of the core.

c. The reviewers believe that the GESSAR-II PRA study constitutes a very useful tool for the evaluation of the basic characteristics of the present design of the BWR/6 Standard Plant as well as for the evaluation of modifications or of alternative design concepts. The BWR/6 Standard Plant, however, does not include all the aspects of the overall design of the nuclear power station. The design of the Feedwater, the Power Conversion, and the Condensate System, portion fo the essential service water system, as well as the design of the plant air systems, are unspecified. Thus, fault trees were not developed for these systems. Rather, assumptions from extrapolations from other BWR designs and PRAs (e.g., Limerick) were made in the assessment of the accident sequences and their frequencies. The Redundant Reactivity Control System drawings are also not included in the "GESSAR-II BWR/6 Nuclear Island Design." This system was included, however, in the PRA. For this reason it would be rather inappropriate to draw conclusions uncritically from the results of this study as to the actual risk from a real plant that may be built in the future on the basis of this design.

d. The results of this review (as well as those of the GESSAR-II PRA) are based on certain design modifications that were assumed as final but, at the time of the review, were not yet implemented in the Safety Analysis Report (SAR) documents available to the reviewers. These modifications mainly concern the Reactor Shutdown system and are associated with the so-called Alternate-3 ATWS modification proposed by the NRC staff in NUREG-0460. In particular, they include the incorporation of a Redundant Reactivity Control System (RRCS), an Alternate Rod Insertion (ARI) system initiated by RRCS, and an automatic initiation of the Standby Liquid Control System (through a RRCS permissive logic). An additional important design modification assumed in the PRA but not yet implemented in the SAR is the modification of the initiation logic of the depressurization system so that it can also automatically initiate depressurization in the event of a transient.

e. The main quantitative results of the Brookhaven revision are as follows:

	BNL Revision
Point value of the frequency of core damage: (per year of plant operation)	2.2×10^{-5}

This value corresponds to a site that belongs to a grid in the Mid-Atlantic Area Reliability Council (MAAC).

The BNL revised frequency for core damage is to be compared with the 4.4×10^{-6} value presented in the GESSAR-II PRA (for a site in the same region). The frequency of core damage is very sensitive to the assumed frequency of Loss of Offsite Power (LOOP). If the BNL "national average" estimate for the frequency of LOOP is used, the frequency of core damage increases to 3.8×10^{-5}.

The analysis of the LOOP initiator in the GESSAR PRA and in this review was based on the assumption that the Reactor Core Isolation Cooling System (RCIC) can operate without room cooling for two hours. Following the completion of this review and subsequent to the issuance of the draft version of this volume, GE submitted an amendment to the GESSAR-II SAR claiming that the RCIC system can operate without cooling for ten hours. A review of this submittal to substantiate the claim was beyond the scope of the BNL project. If, however, the RCIC can operate without room cooling for ten hours and the batteries can also operate for ten hours, the core damage probability for the MAAC-LOOP frequency will be reduced from 2.2×10^{-5}/yr to 1.2×10^{-5}/yr. This is the maximum reduction possible since the value of 1.2×10^{-5}/yr was derived under the assumption that there is no operator action whatsoever required for the successful operation of RCIC for ten hours.

	BNL Revision
Point value of the frequency of core damage: (per year of reactor operation) assuming that RCIC can operate without cooling for ten hours.	1.2×10^{-5}

This remark notwithstanding, all the review comments, both qualitative and quantitative, presented in this volume are based on the assumption that RCIC can operate without room cooling for two hours only.

In the BNL revision both a point value of the frequency of core damage and the associated uncertainties were assessed. The results are depicted in Figures 0.1 and 0.2 for the MAAC site and the "national average site," respectively. For the MAAC site the 90% probability range for the frequency of core damage of the BNL revision spans an interval slightly higher than an order of magnitude from 3.7×10^{-6} (5% percentile) to 6.0×10^{-5} (95% percentile). The median value is 1.3×10^{-5}. The corresponding values for the "national average" site are 6.8×10^{-6} (5% percentile), 2.5×10^{-5} (median) and 1.1×10^{-4} (95% percentile).

The difference of a factor of 5.0 between the BNL point value (MAAC site) and the GESSAR-II PRA point value for the frequency of core damage can be attributed to two reasons. The first is related to additional dependences between safety functions that the BNL review identified and to certain modeling modifications that were made to the accident sequence (event trees) and system (fault trees) parts of the GESSAR-II PRA. The dependences between the safety functions are due to the use of common support systems for different systems in the GESSAR-II design. Dependences between the initiating events and mitigating systems were also modeled in the BNL analysis. A detailed account of these changes is presented in the main body of the report.

The second reason that led to a difference in the GESSAR-II and Brookhaven results was that different values were used for the frequencies of some of the accident initiators.

For the MAAC site, the first reason accounts for an increase of about a factor of 3.8, while a factor of about 1.3 comes from the second reason. For the "national average" site, the total incrase is approximately a factor of 8.6 while a factor of about 2.2 increase is due to the second reason. It is noteworthy that the GESSAR-II PRA did not present a "national average" frequency for LOOP and, hence, the second BNL point value should be regarded with this perspective.

f. The BNL revision is in qualitative agreement with the GESSAR-II study on the relative importance of various initiator groups. The most important sequences are transients caused by a loss of offsite power. The contribution of Loss of Coolant Accidents to the core damage frequency is almost negligible (see Figure 0.3).

The sequences that contribute the most to core damage involve trans-ients caused by a loss of offsite power (LOOP) coupled with failure of the high pressure (U) and low pressure (V) injection functions. These sequences (T_EUV) contribute about 68% to the frequency of core damage for the MAAC site and about 79% for the "national average" site (see Table 0.1). The GESSAR-II PRA found that the LOOP induced transient sequences contributed 90% to the frequency of core damage (see Table 0.1). The BNL review identified station blackout as one of the main modes of inducing core damage following a loss of offsite power (i.e., a LOOP event followed by a common mode failure of the three emergency diesel generators and failure to recover either power source in time to avoid core damage). Another important failure combination consists of a LOOP event followed by failure of two emergency diesel generators coupled with some other failure in the High Pressure Core Spray (HPCS) system.

The BNL review also identified as important contributors to the core damage frequency, sequences involving a transient initiated by an isolation (T_F) followed by loss of the feedwater system (Q) or a transient initiated by a LOOP (T_E), with successful HPCS injection but failure of containment heat removal systems (W). These sequences (T_FQW and T_EW) contribute 8.6% and 3.5%, respectively, to the frequency of core damage for the MAAC site and 5.0% and 4.2%, respectively, for the "national average" site (see Table 0.1). In the GESSAR-II PRA, those sequences have negligible contribution to the core damage frequency.

In addition, the BNL review identified certain ATWS sequences as the next most important contributors to the frequency of core damage (see Table 0.1 and Figure 0.3). In total, ATWS sequences contribute 14.5% to the frequency of core damage (MAAC LOOP frequency). The two most important ATWS sequences are those caused by a transient (T_F) followed by failure of the control rods to insert because of a mechanical failure (C_M) and coupled with a failure of the high pressure injection system (U_H) and failure to inhibit depressurization (P_A) ($T_FC_MU_HP_A$), or failure to control the reactor level (L_H) ($T_FC_ML_H$). These two (as well as other) ATWS sequences become important because of the different probability for the mechanical failure to scram (C_M) assumed in the BNL revision. One additional accident sequence identified by the BNL review is the loss of dc power initiated by a common failure of two dc buses followed by failures of the high pressure and low pressure injection systems.

g. BNL reviewers can identify three areas of disagreement that contribute the most to the difference between the BNL revision and the GESSAR-II PRA, including the recent revisions. These are:

   1. The probability of common mode failure of all three emergency diesel generators: The BWR/6 standard design includes two emergency diesel generators that supply emergency power to the various safety loads in the event of a LOOP, and a third separate and dedicated diesel generator that powers the HPCS system. According to the present design this third diesel will be housed in a different building, will be supplied by a different manufacturer, and, further, all efforts will be made to ensure separation and diversity between this diesel and the other two. On the basis of this information and the analysis presented in NUREG/CR-2989, BNL assessed the probability of common mode failure of the three diesels at $4x10^{-4}$ per demand. This value is to be compared with the initial $6x10^{-4}$ value used in the GESSAR-II PRA. In September 1983, GE submitted an updated c^mmon mode failure analysis for the diesel generators in which the use of the $1x10^{-4}$ (or even smaller) value for this failure is advocated. The BNL reviewers performed a limited review of this submittal and do not believe that the $1x10^{-4}$ value is adequately supported by the submitted analysis. In any event, even if the value of $1x10^{-4}$ were to be used in the assessment the core damage frequency (BNL value) would be reduced by about 33% (for the MAAC site) and the relative ranking of the various accident sequences would not change.

   2. The frequency of failure of the Reactor Protection System: The BNL review used a failure of $3x10^{-5}$ per demand ($1x10^{-5}$ for the mechanical subsystem and $2x10^{-5}$ for the electrical subsystem), as proposed in NUREG-0460. The GESSAR-II PRA used a failure probability of $1x10^{-7}$ per demand for failure to scram (this failure probability includes the failure of the Alternate Rod Injection). The BNL reviewer: believe that such a low failure probability for a system cannot be supported by analysis alone, and, since there is no experiential evidence that can support such a low value, they do not consider it meaningful. If

the 10^{-7} value is used, the frequency of core damage would decrease by about 15% for the MAAC site case.

3. The frequency of transients: There is a difference of about a factor of 2.5 in the frequency of turbine trip and loss of feedwater transients between the BNL revision and the GESSAR-II PRA. The BNL values are based solely on the operating experience of the various Boiling Water Reactor (BWR) power plants. The GESSAR-II PRA values are also based on the operating experience of BWRs but modified to include the effect of design changes in the BWR/6 standard plant, as well as the reduction in frequency of transients in operating plants with operating time ("burn in" effect). The BNL reviewers believe that there is merit in both arguments. They did not, however, agree with the details of the analysis that supported the GESSAR-II PRA arguments, and, since the frequency of these transients has a small effect on the frequency of core damage as well as on the relative ranking of the various accident sequences the decision was made not to allocate substantial review resources to further pursue it at this point.

If the GE values were to be used in all these three areas, the frequency of core damage would decrease by approximately 55% to the value of 1×10^{-5}, for the MAAC site.

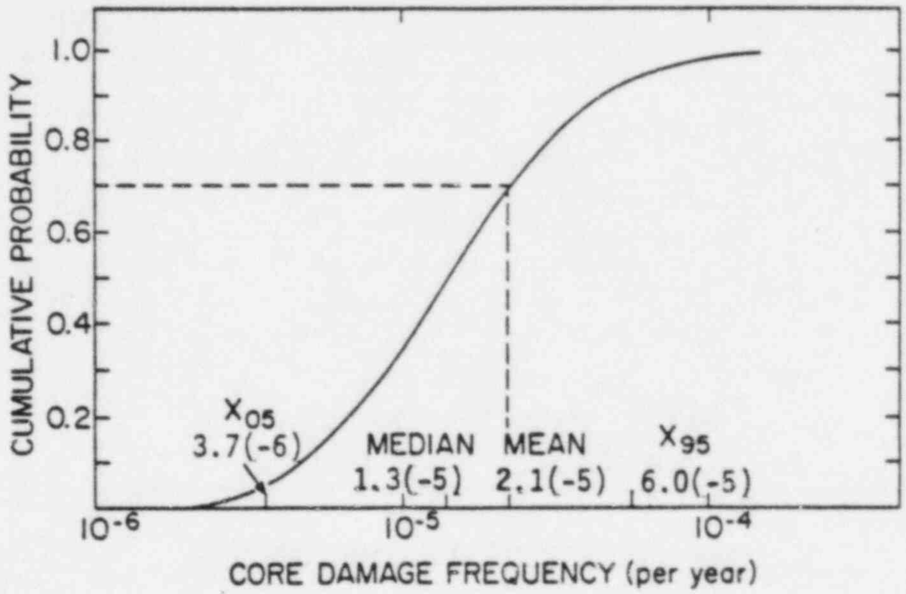


Figure 0.1 Cumulative probability for the frequency of core damage (MAAC site).

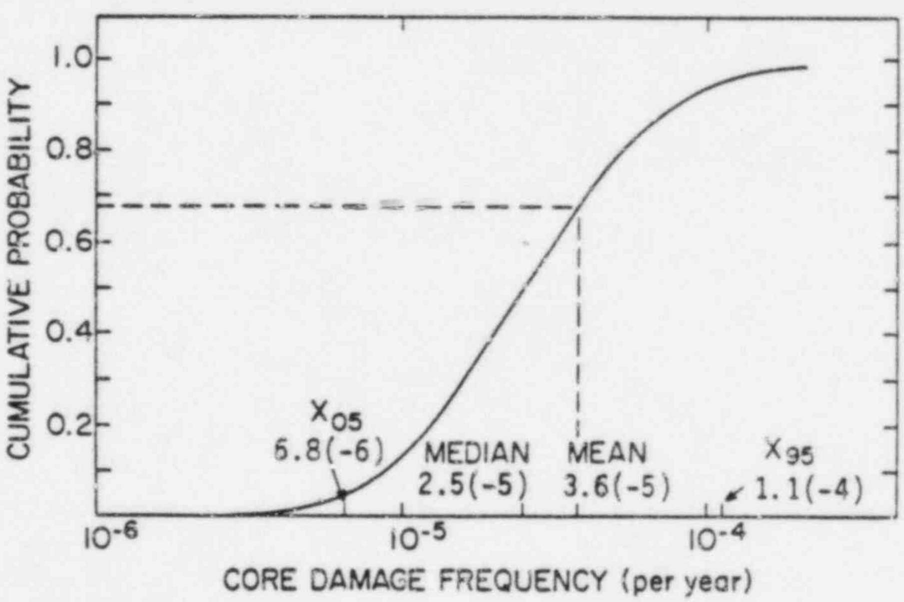


Figure 0.2 Cumulative probability for the frequency of core damage (national average site).

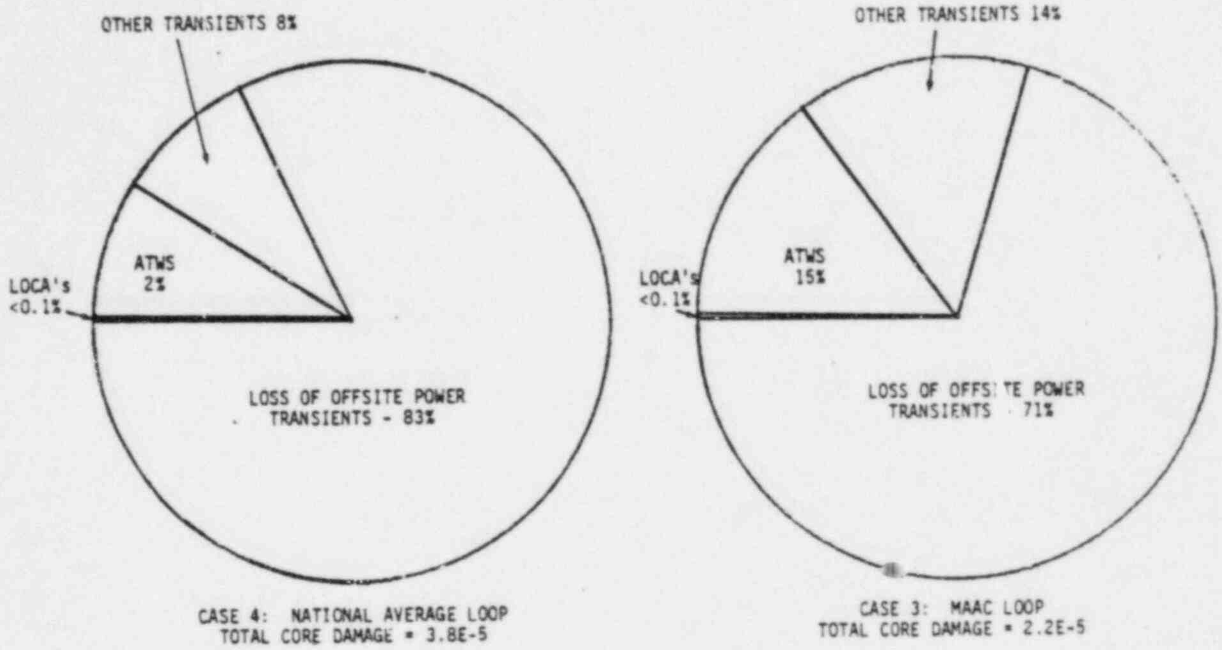


Figure 0.3 Contribution of accident initiators to the frequency of core damage (per reactor year).

Table 0.1 Ranking of BNL and GESSAR-II PRA Sequences by Core Damage Frequency

	BNL Dominant Sequences	
	MAAC Site (Total = 2.2E-5)	National Average Site (Total = 3.8E-5)
1. T_EUV	1.5E-5 (68%)	3.0E-5 (79.0%)
2. T_FQW	1.9E-6 (8.6%)	1.9E-6 (5.0%)
3. T_EW	7.8E-7 (3.5%)	1.6E-6 (4.2%)
4. $T_FC_MU_HP_A$	8.7E-7 (3.9%)	8.7E-7 (2.3%)
5. $T_FC_ML_H$	7.6E-7 (3.4%)	7.6E-7 (2.0%)
6. T_FQUX	5.3E-7 (2.4%)	5.3E-7 (1.4%)
7. $T_{DC}UV$	4.3E-7 (1.9%)	4.3E-7 (1.1%)
8. $T_FC_MC_1PA$	3.9E-7 (1.8%)	3.9E-7 (1.0%)
9. $T_FC_MUP_A$	1.8E-7 (0.8%)	1.8E-7 (0.5%)
10. $T_FC_MP_1U_MP_A$	1.1E-7 (0.5%)	1.1E-7 (0.3%)
11. $T_FC_MC_1C_{21}$	1.1E-7 (0.5%)	1.1E-7 (0.3%)
12. T_FC_MU	1.0E-7 (0.4%)	1.0E-7 (0.3%)
13. T_IUX	1.0E-7 (0.4%)	1.0E-7 (0.3%)

GESSAR-II Dominant Sequences (Total = 4.4E-6)

1. T_EUV	4.0E-6	90.0%
2. T_EUX	1.3E-7	2.1%
3. T_EP_1UV	1.2E-7	2.6%
4. T_FP_1UV	1.6E-8	0.4%
5. T_FQUX	1.3E-8	
6. T_IUV	1.2E-8	

1. INTRODUCTION

This section contains a discussion on how the review of the PRA was performed by Brookhaven National Laboratory (BNL), and how this report is organized.

1.1 Objective, Scope, and Approach to the Review

The broad objective of the BNL review of the GESSAR-II PRA study was to evaluate qualitatively and quantitatively the study's assessment of the important accident sequences that are internally generated and lead to core damage. In addition, the review included an assessment of the externally generated LOOP accident initiator. To carry out this objective, BNL reviewed the assumptions and methods of the GESSAR-II PRA within its stated scope. Within this scope and within the basic methodological framework of the GESSAR-II PRA, BNL reevaluated the important accident sequences that lead to core damage, their respective frequency of occurrence, the total frequency of core damage, and the associated uncertainties. The review included evaluations of accident initiators, data, and accident sequence development and quantification. In addition, importance and sensitivity analyses were performed.

The review of the "internally" generated accident sequences, with respect to the frequency of core damage, constitutes one part of the overall review of the GESSAR-II PRA that is being performed by BNL for the Nuclear Regulatory Commission. Other parts of the review include the core melt phenomenology, and containment analysis (NUREG/CR-4315, Vol. 3). Furthermore, a review is underway for both the system analysis and consequence evaluation of accident sequences initiated by other events such as earthquakes (NUREG/CR-4315, Vol. 2), internal floods, and in-plant fires. The results of these parts of the review together with NRC inputs on off-site consequences will be reported in NUREG/CR-4315, Vol. 4 and will be integrated to provide an overall assessment of the risks of a BWR/6 Standard Plant.

The review of the "internally" generated accident initiator reported in this NUREG/CR was performed over the period of one and a half years. It involved five people at BNL (three on a regular basis and two on a partial basis) and a contribution from Intermountain Technologies which was subcontracted by BNL to assist in the initial review of the PRA. Ioannis A. Papazoglou was the principal investigator for the project and directed the review of the core damage frequency assessment. Nelson Hanan and Kelvin Shiu of the Risk Evaluation Group were the main project engineers; they reviewed the accident sequence and system modeling, and the quantification of the event and fault trees, and they performed the importance and uncertainty analyses. Ray Karol contributed to the qualitative review of both the system and the accident sequence modeling. Eshagh Anavim provided the frequencies of the accident initiators. Finally, Intermountain Technologies performed a qualitative review of the GESSAR-II PRA during the initial phrase and provided the necessary input for the round-one questions submitted to GE through NRC.

The project monitors were David Yue and Mark Rubin of the Reliability and Risk Assessment Branch.

The review process benefited from the several productive meetings held between NRC, BNL, and GE. GE staff were entirely cooperative in providing the information and discussion that were needed to gain a detailed understanding of the PRA for the in-depth review process. The various resubmittals, up dates, and supporting analyses, with which GE responded to the various BNL comments in meetings and/or letter reports, always constituted a technical improvement of the PRA and were always responsive to the BNL comments. This is universally true regardless of whether BNL agrees with each and every point in the responses.

1.2 Organization of Report

Section 2 gives a description of the plant modeling which includes identification of initiating events that can lead to core damage, and a discussion of safety functions and systems important to preventing or mitigating core damage events. Section 3 presents a description of accident sequence definition and a discussion of the event tree/fault tree approach used in the GESSAR-II PRA. Section 4 provides accident sequence quantification, reviews the numerical values of the parameters necessary for this quantification, gives a brief description of the GESSAR PRA approach to quantification, and presents the BNL modifications to the quantification process and the revised core damage frequencies. This section also contains an analysis of the uncertainties in the core damage frequency and an importance analysis. Details of the BNL modifications to the system fault trees, the uncertainty analysis, and the evaluation of the containment event trees, are given in the Appendices.

2. PLANT MODELING

This section reviews the modeling of the plant in the GESSAR-II PRA. The plant modeling includes the identification of safety functions important to prevent or to mitigate core damage events, systems that directly perform these functions (frontline systems), systems that support the frontline systems (support systems), success criteria of the safety functions and the systems, and initiating events that can lead to core damage. Subsection 2.1 describes the safety functions, the corresponding frontline and support systems, and their success criteria. Subsection 2.2 presents the initiating events and their grouping according to the success criteria for the frontline systems. In both subsections, the GESSAR-II PRA assumptions are reviewed, evaluated, and compared to those of the Peach Bottom PRA (Reactor Safety Study, WASH-1400)[1] and those of the Grand Gulf PRA (RSSMAP, NUREG/CR-1659).[2]

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 GESSAR-II into the functions given in Table 2.2. Each of the functions in Table 2.2 is directly performed by one or more frontline systems. The frontline systems for GESSAR-II are given in Table 2.3, and in Table 2.4 they are compared with the corresponding systems of Peach Bottom and Grand Gulf No.1. A short description of the differences in the frontline systems for GESSAR-II, Peach Bottom, and Grand Gulf follows.

Reactor Protection System (RPS) - These systems are similar for GESSAR-II, Peach Bottom, and Grand Gulf, but there are differences in the methods used to render the reactor subcritical and/or to reduce power during an ATWS. The GESSAR-II PRA assumes a separate and diverse system for initiating the systems used to prevent and/or mitigate an ATWS. This independent system, the Redundant Reactivity Control System (RRCS), and associated modifications to existing systems are described below.

Redundant Reactivity Control System (RRCS) - The GESSAR-II PRA assumes the incorporation of an RRCS which includes the following added features as recommended by Alternate 3 of NUREG-0460 to reduce the impact of a failure to scram event.

a) A diverse initiation signal for the RRCS which is independent of the RPS initiation.

b) Alternate Rod Insertion (ARI) - this system is effective in reducing electrical common mode failure to scram. It is actuated by the Redundant Reactivity Control System (RRCS).

c) Scram Discharge Volume (SDV) - diverse and redundant water level sensors for the Scram Discharge Volume (SDV) were added to GESSAR-II (Grand Gulf also includes this feature). Additionally, changes to the instrument piping and installation of redundant SRV vent and drain valves are included in GESSAR-II to further reduce potential common mode failures.

d) Feedwater runback - the feedwater control system has been modified to provide for a feedwater runback function which stops feedwater flow to the reactor vessel during an ATWS situation. The feedwater runback reduces core inlet subcooling, thus helping to reduce the core power.

e) Standby Liquid Control (SLC) - the GESSAR-II PRA SLC system assumes model the Alternate 3 described in NUREG-0460 with two automatically initiated SLC pumps (43 gpm/pump). The system is actuated by the redundant reactivity control system (RRCS). Peach Bottom and Grand Gulf have two manually actuated SLC pumps.

f) Reactor Recirculation System (RRS) - The GESSAR-II PRA has been modified to include an ATWS recirculation pump trip designed to mitigate the consequences of an ATWS event by tripping the recirculation pumps. This trip is actuated by the RRCS.

Reactor Core Isolation Cooling System (RCICS) - the RCICS of GESSAR-II, Peach Bottom, and Grand Gulf are similar. The main differences are due to GESSAR-II modifications to the RCIC system that allow it to restart at the low reactor pressure vessel level (Level 2) following a trip due to high RPV water level (Level 8), and decrease the potential for a RCIC failure to start.

High Pressure Core Spray System (HPCSS) - the HPCSS of GESSAR-II and Grand Gulf are similar. The GESSAR-II design has an electric motor-driven pump which injects coolant at a variable flow rate (which is a function of the reactor vessel pressure) via a core spray ring. The makeup water is jetted as a spray over the top area of the fuel bundles. The equivalent system in Peach Bottom is the High Pressure Coolant Injection System which has a steam turbine driven main and booster pumps injecting coolant at a constant flow rate to the RPV via the main feed water line.

Automatic Depressurization System (ADS) - the ADS of GESSAR-II and Grand Gulf are similar. Both systems use 8 Safety relief valves (SRVs) for the automatic depressurization function and the ADS valves have two solenoid pilot valves. The ADS of Peach Bottom uses 5 SRVs for the automatic depressurization function and each ADS valve has one solenoid pilot valve.

The GESSAR-II PRA was performed under the assumption that the ADS control logic will automatically initiate the ADS function for events with low reactor pressure vessel (RPV) water level (level 1). So, in the GESSAR-II PRA the ADS signal is assumed to be initiated by low RPV water level and high drywell pressure (same as for Grand Gulf and Peach Bottom) OR low RPV level (Level 1) plus 10 minutes, i.e., low RPV level signal for 10 minutes. This modification allows the ADS to automatically perform its safety function for transient accident sequences without need for operator action. The modification, however, was not reflected in the logic diagrams of the SAR[4] at the time of this review.

Control Rod Drive System (CRDS) - there are no major differences in the GESSAR-II, Peach Bottom, and Grand Gulf CRD systems.

Low Pressure Coolant Injection System (LPCIS) - the LPCIS represents an operating mode of the Residual Heat Removal System. The GESSAR-II and Grand Gulf LPCIS have 3 pumps and 3 loops, whereas the Peach Bottom LPCIS has 2 loops with 2 pumps per loop. A major difference between the Peach Bottom LPCIS, and the GESSAR-II and Grand Gulf LPCIS is the location of injection. Peach Bottom LPCIS injects water into the recirculation loops using loop selection logic to ensure injection in the intact loop, whereas in GESSAR-II and Grand Gulf the LPCIS injects water directly into the core shroud above the top of the core through separate injection lines.

Low Pressure Core Spray System (LPCSS) - the GESSAR-II and Grand Gulf systems have a single pump, single loop system whereas the Peach Bottom equivalent system (Core Spray Injection System) is redundant, consisting of two independent loops and four pumps.

Residual Heat Removal System (RHRS) - the RHR systems for GESSAR-II and Grand Gulf are very similar. They consist of two separate loops for removing decay heat from the primary containment. Each loop includes an electric motor-driven pump, heat exchangers (two for GESSAR-II and one for Grand Gulf), and associated piping and valves. In GESSAR-II the heat exchangers receive their cooling water from the Essential Service Water System, which is an online system, and in Grand Gulf the cooling water for the heat exchangers is supplied by the Standby Service Water System. It is noteworthy that in the GESSAR-II PRA, credit was taken only for the suppression pool cooling mode of operation of the RHRS, whereas in the Grand Gulf-PRA credit was taken also for the reactor steam condensing mode of the RHRS.

In Peach Bottom, the system equivalent to the RHRS is the Low Pressure Recirculation System. This system is grouped in two loops, each consisting of two pumps in parallel, two heat exchangers in parallel, and associated piping and valves. The heat exchangers receive their cooling water from the High Pressure Service Water System.

Containment Sprays - GESSAR-II and Grand Gulf have an automatically actuated containment spray system which sprays the containment volume, whereas Peach Bottom has a manually actuated containment spray system which can spray either the drywell or the wetwell volumes.

[redacted]

main difference between GESSAR-II and Grand Gulf is that the SSWS is a standby system.

In Peach Bottom the equivalent system is actually composed of two systems: the High Pressure Service Water System (HPSWS) and the Emergency Service Water System (ESWS). The HPSWS comprises four redundant pumps, and its main function is to provide service water to the heat exchangers during long-term cooling. It is a standby system which must be manually initiated. The ESWS provides backup supply cooling water to the LPCRS and CSRS pump components and the CSRS pump lube oil coolers, and is the only source of cooling water to the diesel generators. Both the HPSWS and ESWS in Peach Bottom are shared by both units.

Plant Air Systems - the GESSAR-II does not have a description of all the plant air systems, and a comparison cannot be made with Peach Bottom and Grand Gulf. However, a description of the source of compressed air for the actuators of the ADS SRVs is given in GESSAR-II. In this system (the Pneumatic Supply System) GESSAR-II has more redundancy than Peach Bottom.

2.2 Initiating Events

This section discusses the initiating events which could lead to core damage. It is divided into three subsections. The first presents the approach used in the GESSAR-II PRA for initiating events and the BNL review of this classification. The other two compare the initiating event and groupings of the GESSAR-II, Peach Bottom, and Grand Gulf PRAs.

The GESSAR-II PRA considers three general classes of initiating events:

1. Loss-of-coolant Accidents (LOCAs),
2. Transients with successful scram, and
3. Anticipated transients without scram (ATWS).

The LOCA initiators have been differentiated in the GESSAR-II PRA, according to their location: inside the drywell (drywell LOCA), inside primary containment but outside the drywell (containment LOCA), and outside containment (ex-containment LOCA). These LOCA initiators were further subdivided into three groups according to the equivalent size of the break and the corresponding success criteria for the frontline systems (see Table 2.5). BNL has reviewed and accepted this classification.

The transient initiators with successful scram have been subdivided into five groups:

1. Reactor shutdown.
2. Turbine trip.
3. Isolation (includes MSIV closure and loss of feedwater).
4. Loss of offsite power.
5. Inadvertent open relief valve (IORV).

Each of the five groups represents a collection of transients that demand similar responses from plant systems. A summary of the GESSAR-II PRA grouping is given in Table 2.11. In the BNL review, these five major groups were retained and a sixth one was added. This additional group of initiating events encompasses all events that can lead to loss of dc power, but its frequency is dominated by the common mode loss of two dc buses (EDC125E and EDC125F). This group of initiating events requires a separate event tree to model the response of the plant (see Section 3.2).

Besides adding a sixth group, the BNL review used a list of generic transients different from that used in the GESSAR-II PRA and classified them differently in the five groups mentioned above. In particular, the BNL review used a list of 37 generic transient initiators for BWRs contained in an EPRI survey of the BWR operating experience[7] and reproduced here in Table 2.12. The major departure of the BNL grouping from the GESSAR-II PRA lies in the treatment of events noted by an asterisk in Table 2.13. Under the GESSAR-II PRA grouping scheme these events are classified as turbine trip transients. The GESSAR-II SAR (Chapter 15 - Accident Analysis) indicates, however, that these transient initiators cause a reactor vessel "Level 8" signal which in turn causes a trip of the main feedwater turbine in less than ten seconds. This feature of the GESSAR-II design requires the grouping of these ten transient initiators into the isolation group. It is BNL's speculation though that the number of actual turbine trip leading to a loss of feedwater would be less than what is assumed in this reassessment

If the Reactor Protection System fails to scram the reactor after a transient initiating event, then an ATWS results. Four groups of ATWS initiators were considered in the GESSAR-II PRA.

1. Turbine trip ATWS.

2. Isolation ATWS.

3. Loss of offsite power ATWS.

4. IORV ATWS.

In the GESSAR-II PRA, the specific transient events placed into these four groups correspond to those placed in the corresponding groups of transients with successful scram (see Table 2.11). In the BNL review the group "Turbine Trip" was eliminated for the following reason. A successful mitigation of turbine trip ATWS in GESSAR-II requires a recirculation pump trip (RPT). As a result of the RPT, a reactor vessel "Level 8" trip signal is generated resulting in the tripping of the feedwater turbine (see SAR Chapter 15); NEDE-24222 also shows that the reactor vessel Level 8 is reached in a few seconds. All the turbine trip ATWS events result, therefore, in a feedwater isolation, and hence the event trees that delineate the accident sequences for isolation ATWS are applicable.

2.2.1 Comparison with the Peach Bottom PRA

In the Peach Bottom PRA, all transient initiating events were grouped together and a single event tree was developed. The fifteen likely transient initiators used in the Peach Bottom PRA are given in Table 2.14. They are included in the BNL listing (Table 2.13). Since worst case assumptions were made about the required responses and availability of the frontline systems in the single transient event tree of the Peach Bottom PRA, the GESSAR-II approach of creating five groups of transient initiators is more realistic than the RSS approach. Furthermore, in the RSS, a failure to scram leads directly to core damage, whereas, in GESSAR-II, each failure to scram following a transient initiator is classified into one of the ATWS groups and a detailed plant response is considered. In this regard, GESSAR-II is also more realistic than the RSS.

For the LOCA initiators, both the RSS and GESSAR-II consider three groups according to the equivalent break size. GESSAR-II adds more detail by additionally separating each type of LOCA according to its location (drywell, containment, or ex-containment). GESSAR-II conservatively assumed, as in the RSS, that failure to scram following LOCA initiators leads to definite core damage.

The reactor vessel rupture initiator was handled the same way in both studies. That is, small and medium-size ruptures are considered to be among the small and medium LOCA initiators, respectively, and massive reactor vessel ruptures are excluded from the internally initiated accidents on the basis of their extremely low probability of occurrence. Massive reactor vessel ruptures are, however, considered in the "externally" initiated accident analysis and, in particular, the BNL seismic analysis.[8]

The RSS concluded that interfacing LOCAs were an insignificant contribution to the overall risk. The GESSAR-II PRA, however, includes an analysis of small LOCAs outside the containment which includes interfacing LOCA.

Thus, overall, the handling of the initiating events in the GESSAR-II PRA is more realistic and of finer detail than that in the RSS.

2.2.2 Comparison with RSSMAP Grand Gulf

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

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 GESSAR-II PRA treatment of initiating events is more detailed and realistic than that of the Grand Gulf study.

2.3 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 1978.
2. Hatch, S. W., "Reactor Safety Study Methodology Application Program: Grand Gulf No. 1 BWR Power Plant," NUREG/CR-1659/4, November 1981.
3. "Anticipated Transients Without Scram for Light Water Reactors," NUREG-0460, April 1978.
4. GESSAR-II BWR/6 Nuclear Island Design.

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

6. "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," GE Report NEDO-24708, August 1979.

7. "Anticipated Transients, A Reappraisal," EPRI-NP-2230.

8. "A Review of the GESSAR-II BWR/6 Standard Plant Seismic Probabilistic Risk Assessment," NUREG/CR-4315, Vol. 2, Draft, September 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 GESSAR-II

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)	Containment heat removal and/or containment spray*
7)	Scrub radioactivity from containment atmosphere

*Containment spray is required to prevent containment overpressurization for LOCAs which are located in the containment but outside the drywell.

Table 2.3 Frontline Systems for GESSAR-II

Safety Function	Frontline Systems
1) Reactor subcriticality	1) Reactor protection system
	2) Redundant reactivity control system including a) Alternate rod insertion b) Automatic standby liquid control c) Recirculation pump trip d) Feedwater runback
2) Reactor coolant system over-pressure protection	3) 19 Safety relief valves (SRV)
3) High pressure injection	4) High Pressure Core Spray (HPCS)
	5) Reactor Core Isolation Cooling (RCIC)
	6) Control Rod Drive (CRD*)
	7) Condensate and feedwater system with power conversion system
4) Depressurization	8) Automatic depressurization system (8 SRVs used for this function)
	9) Manual depressurization
5) Low pressure injection	10) Low Pressure Core Injection (LPCI)
	11) Low Pressure Core Spray (LPCS)
	12) Condensate pumps
6) Containment heat removal	13) Residual Heat Removal and Essential Service Water (RHR)
	14) Power Conversion System (PCS)
	15) Suppression pool
	16) Containment sprays**
7) Scrub radioactivity from containment atmosphere	17) Suppression pool
	18) Containment sprays

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

**For LOCAs in the containment (outside the drywell), spray is required to prevent overpressurization failure.

Table 2.4 Comparison of GESSAR-II, Peach Bottom, and Grand Gulf Safety Systems

	GESSAR-II	Peach Bottom	Grand Gulf -1
Power (MWT)	3579	3293	3833
Containment	MK-III(free standing steel)	MK-I(free standing steel)	MK-III
No. Relief Valves	19 SRVs	11 SRVs	20 SRVs
No. Safety Valves	----	2	----
RCIC	Steam turbine driven	Steam turbine driven	Steam turbine driven
HPCI	----	Steam turbine driven	----
HPCS	Electric motor driven	----	Electric motor driven
LPCI	3 pumps with 3 loops	4 pumps with 2 loops	3 pumps with 3 loops
LPCS	1 pump	2 loops; 2 pumps/loop	1 pump
ADS Valves	8 SRVs	5 relief valves	8 SRVs
RHRHX	4 cooled by essential service water (50% capacity each)	4 cooled by HPSW (100% capacity each)	4 cooled by the SSWS
EDG	3 with division 3 EDG for HPCS only	4 shared between two units	3 with division 3 EDG for HPCS only
RPS	Has ARI,RPT,feedwater runback	Has RPT	Has RPT
SLC	2 pumps, automatic actuation	2 pumps, manual actuation	2 pumps, manual actuation
HPSW	----	4 pumps, 100% each cross-connection with other unit considered; pumps normally not running	----
ESW	2 loops, 2 pumps/loop with 1 pump/loop normally operating*	1 100% pump per unit, pump normally not running	----
SSWS	----	----	3 loops, 1 pump/loop; loop dedicated to HPCS; standby system
HPCS ESW	This is a separate ESW system which supplies only HPCS system heat loads; standby system	----	----
FW and condensate	**	3 turbine-driven feed pumps and 3 electric-driven condensate pumps	3 motor-driven condensate pumps, 3 motor-driven condensate booster pumps and 2 (50% capacity) turbine-driven feedwater pumps
Containment Sprays	Automatically or manually actuated. Sprays only the containment area (not the drywell)	Manually actuated. Sprays either the drywell or wetwell	Automatically or manually actuated.

*This is the configuration assumed in the GESSAR-PRA system fault tree for ESW. The SAR provides no information as to the number of pumps for this system.

**Specific information on the FW and condensate system is not provided by the SAR.

Table 2.5 GESSAR-II LOCA* Success Criteria

[redacted]

Table 2.6 Peach Bottom LOCA Success Criteria

Large LOCA (A)		
Liquid Break: $A_b > 0.4$ ft^2	Injection:	4 of 4 CSIS or
Steam Break: $A_b > 0.4$ ft^2		3 of 4 LPCI and
		2 of 4 CSIS
	Recirculation:	1 of 4 CSIS or
		1 of 4 LPCI and
		1 LPRS
Intermediate LOCA (S_1)		
Liquid Break: 0.03 ft^2 < A_b < 0.4 ft^2		HPCI or 4 SRVs and
Steam Break: 0.12 ft^2 < A_b < 0.4 ft^2		1 of 4 LPCI or
		1 of 4 CSIS and 1 LPRS
Small LOCA (S_2)		
Liquid Break: 0.002 ft^2 < A_b < 0.03 ft^2		HPCI or RCIC or 4 SRVs and
Steam Break: 0.01 ft^2 < A_b < 0.12 ft^2		1 of 4 LPCI or
		1 of 4 CSIS and 1 LPRS

Table 2.7 Grand Gulf LOCA Success Criteria

	Core Cooling	Containment Heat Removal
Large LOCA:		
$A_b > 0.5\ ft^2$	HPCS or LPCS or 3 of 3 LPCI Loops	1 of 2 RHR Loops
Medium LOCA:		
$A_b < 1\ ft^2$	HPCS or RCIC or 4 SRVs and LPCS or 2 of 3 LPCI Loops	1 of 2 RHR Loops

Table 2.8 Transient Success Criteria

	GESSAR-II	Peach Bottom PRA	Grand Gulf PRA
Injection	[redacted]	HPCI or RCIC or FW or 4 SRVs and: 4 out of 4 LPCS pumps or 3 out of 4 CSIS pumps or 1 condensate pump	HPCS or RCIC or FW or 4 SRVs and: LPCS or 2 out of 3 LPCI loops
Containment Heat Removal		PCS or 1 RHR	PCS or 1 RHR

Table 2.9 ATWS Success Criteria for BWR/6 PRA Assuming ATWS Modifications

[redacted]

Table 2.11 GESSAR-II PRA Initiating Event Groupings

Reactor Shutdown
. Planned Shutdown
. Other Scrams - Inadvertent Scram - Flux or Pressure Scram - Single MSIV Closure
Turbine Trip
- Turbine Trip - Generator Load Rejection
Isolation
. Feedwater Failure - Recirculation Failure - Loss of All Feedwater - Feedwater Control Failure
. Immediate Isolation - Pressure Regulator Failure - MSIV Closure - Loss of Condenser Vacuum
Loss of Offsite Power
Inadvertent Opening of Safety/Relief Valves

Table 2.12 Summary of the Categories of BWR Transients Used to Classify Operating Experience Data on Anticipated Transients*

1. Electric Load Rejection
2. Electric Load Rejection with Turbine Bypass Valve Failure
3. Turbine Trip
4. Turbine Trip with Turbine Bypass Valve Failure
5. Main Steam Isolation Valve Closure
6. Inadvertent Closure of One MSIV (Rest Open)
7. Partial MSIV Closure
8. Loss of Normal Condenser Vacuum
9. Pressure Regulator Fails Open
10. Pressure Regulator Fails Closed
11. Inadvertent Opening of a Safety/Relief Valve (Stuck)
12. Turbine Bypass Fails Open
13. Turbine Bypass or Control Valves Cause Increase Pressure (Closed)
14. Recirculation Control Failure -- Increasing Flow
15. Recirculation Control Failure -- Decreasing Flow
16. Trip of One Recirculation Pump
17. Trip of All Recirculation Pumps
18. Abnormal Startup of Idle Recirculation Pump
19. Recirculation Pump Seizure
20. Feedwater -- Increasing Flow at Power
21. Loss of Feedwater Heater
22. Loss of All Feedwater Flow
23. Trip of One Feedwater Pump (or Condensate Pump)
24. Feedwater -- Low Flow
25. Low Feedwater Flow During Startup or Shutdown
26. High Feedwater Flow During Startup or Shutdown
27. Rod Withdraw at Power
28. High Flux Due to Rod Withdrawal at Startup
29. Inadvertent Insertion of Rod or Rods
30. Detected Fault in Reactor Protection System
31. Loss of Offsite Power
32. Loss of Auxiliary Power (Loss of Auxiliary Transformer)
33. Inadvertent Startup of HPCI/HPCS
34. Scram due to Plant Occurrences
35. Spurious Trip via Instrumentation, RPS Fault
36. Manual Scram -- No Out-of-Tolerance Condition
37. Cause Unknown

*EPRI-SAI Study (Reference 7).

Table 2.13 Grouping of Transient Initiators

(Number in parenthesis refers to transient numbers from Table 2.12)

Item	Transient Group**
1	Isolation Electric Load Rejection (1*, 2) Turbine Trip (3*, 4, 13*) Closure of All MSIVS (5) Loss of Condenser (8) Pressure Regulator Failures (9*, 10*) Recirculation Problems (15*, 17*, 19*) Distrubance of Feedwater (20*, 22) Loss of Auxiliary Power (32*)
2	Turbine Trip Partial Closure of MSIVs (6, 7) Bypass Fails Open (12) Recirculation Problems (14, 16, 18) Disturbance of Feedwater (21, 23, 24, 25, 26) Rod Withdrawal/Insertion (27, 28, 29) Fault in RPS (30) Inadvertent Startup of HPCS (33) Others (34, 35, 36, 37)
3	Loss of Offsite Power (31)
4	Inadvertent Open Relief Valve (11)

*These transients are normally grouped under turbine trip events; however, for the GESSAR-II, they are later characacterized as isolation events (see Section 2.2).

Table 2.14 WASH-1400 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

3. ACCIDENT SEQUENCE DEFINITION

The objective of this section is to provide a discussion and the major conclusions of the review on the following topics: 1) the GESSAR-II accident sequence definition and the qualitative description of the event trees, 2) the system fault trees that were used in the GESSAR-II PRA, and 3) the various aspects of human performance analysis that entered into the risk assessment.

3.1 Functional Event Tree

Subsection 3.1.1 provides an overview of the BNL comments on the functional event tree approach and the assumptions inherent in the GESSAR-II PRA. Subsection 3.1.2 presents a discussion of dependence analysis. Subsection 3.1.3 presents detailed discussions of the BNL review of the transient and manual shutdown events. Subsection 3.1.4 focuses on the ATWS functional event trees, and Subsection 3.1.5 on the LOCA functional event trees.

3.1.1 Overview

The GESSAR-II PRA employed the event tree/fault tree method in the evaluation of core damage sequences. A total of 13 sets of functional event trees were developed, one for every particular type of initiating event. These events include: turbine trip, isolation, loss of offsite power, inadvertent open of relief valve (four for successful scram and four for ATWS), manual shutdown, large, intermediate and small LOCAs inside the drywell containment, and LOCAs inside and outside containment. More detailed discussions of the initiators, their grouping and their frequencies are presented in Section 2.2 and Section 4.2.

Figures 3.1 through 3.3 depict typically the set of functional event trees used for the turbine trip initiator. The first tree, Figure 3.1, examines various ATWS prevention and mitigation functions and reactor pressure function. For those sequences where there is a successful insertion of control rods, an adequate pressure control, and feedwater injection, they are transferred to Figure 3.2, which continues to evaluate the availability of the high pressure and low pressure injection functions and the containment decay heat removal function. Given the occurrence of failure to scram and failure of the feedwater system to inject, the event is transferred to the isolation event tree. Failure to scram with feedwater available events are further developed in Figure 3.3. In this event tree, the various ATWS mitigating functions are evaluated. The overall event tree approach used in the GESSAR-II PRA is, in general, consistent with the state-of-the-art currently being adopted in all the PRA studies. Detailed discussions on the specifics of each of the event trees will be presented in Sections 3.1.3 and 3.1.4.

The end points of the transient event tree (for example, Figure 3.2) or the ATWS event tree (Figure 3.3) can be classified into four groups: 1) successful shutdown, 2) core damage, 3) loss of containment heat removal, and 4) transfers. Sequences which result in core damage are divided into plant damage states depending upon physical characteristics associated with core damage and containment response. The loss of containment heat removal events are further developed in containment event trees. Some of these sequences eventually lead to core damage, and others result in successful shutdown.

It is assumed in the GESSAR-II PRA that in a given accident sequence, if the peak clad temperature (PCT) is maintained at less than 2200°F and a viable capability to remove decay heat exists, then the sequence is considered a successful sequence. The 2200°F PCT criterion is based on the 10 CFR 50 Emergency Core Cooling Requirements which is a conservative license criterion.

In the development of functional event trees, the GESSAR-II PRA did not include a few of the functions which may be available to mitigate the event. These functions include the RHR steam condensing mode and the control rod drive pumps. Consideration of these functions may result in a reduction in the core damage frequency. However, a more detailed analysis of these systems, their support systems, and the dependence and interactions with other functions is necessary before one can conclude that their omissions in the event tree analysis lead to conservative results. BNL did not perform these analyses, and it was assumed for this review that these functions are not available.

3.1.2 Qualitative Dependence Analysis

The objective of this section is to provide a summary of the dependence modeling used in the GESSAR-II PRA and of the review comments and modifications by BNL. Detailed discussions on the quantification of these dependences are provided in Sections 3.1.3 and 3.2.

A comprehensive discussion on the various types of dependences is given in Reference 1. They can be classified as: 1) functional dependences, 2) physical dependences, and 3) humanly induced dependences. It should be noted that these three types of dependences are not necessarily mutually exclusive. A finer resolution of them 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.

System Functional Dependences - This type of dependence can be characterized by a functional relationship which exists between two or multiple systems. The GESSAR-II PRA is reported to have addressed, in general, this type of dependence in the functional event tree approach.

In the review of the GESSAR-II PRA, BNL evaluated specifically this type of dependence among frontline systems and also between frontline systems and support systems. The review identified a few areas of potential frontline system dependence that apparently were not adequately modeled in the GESSAR-II PRA. For instance, the PCS shares with the feedwater system the main condenser system which it serves as a common decay heat sink. Given that there is a failure of the feedwater system, the degradation in the power conversion system availability within the containment heat removal function is not observed in the GESSAR-II analysis. Moreover, as a result of the shared hardware between the low pressure coolant injection system and the RHR system, one would expect a compromise in the reliability of the RHR system given that the low pressure system is disabled. Another dependence that is identified in the BNL review is that of the RCIC system on the containment heat removal systems through the suppression pool. The RCIC system is driven by a steam turbine pump with the lube oil system cooled via diverted pump discharge flow. These items are addressed in greater detail in Section 3.1.3.

With regard to the dependence between frontline systems and support systems, BNL compiled a dependence matrix which illustrates how frontline systems depend on the various support systems, such as ac and dc power. This matrix is presented in Table 3.1. The left-hand column of the matrix contains the frontline systems. As one moves to the right, each successive column denotes a particular support system on which a frontline system may depend. For instance, the safety relief valve (SRV), depends only on the dc power for actuation and not on ac power. It is also dependent on the instrument air system or human action for actuation. Lastly, the suppression pool is required to condense steam release through the safety relief valves. The following frontline systems are included in the dependence matrix: SRV, ADS, HPCS, RCIC, LPCI, LPCS, RHR, and SLC.

A similar matrix delineating the dependence of one support system on another support system is not possible owing to the lack of information on these systems. Most of these systems, e.g., instrument air, service water, etc., are not part of the GESSAR Nuclear Island design,[2] and hence no specific information is available.

System Physical Dependences - Dependences of this type were treated in the GESSAR-II PRA to some limited extent. For instance, the effect of containment failure due to overpressure resulting from loss of containment heat removal was incorporated in the containment event trees. The GESSAR-II PRA assumed that only a fraction of the containment failures would lead to loss of coolant injection to the core. In the event of a station blackout, loss of room cooling was considered.

System Humanly Induced Dependences - These dependences were also addressed to a limited extent. These dependences include cognitive errors of the operator; an example included in the analysis is the failure to inhibit ADS in an ATWS event. Errors of commission were not included in the analysis.

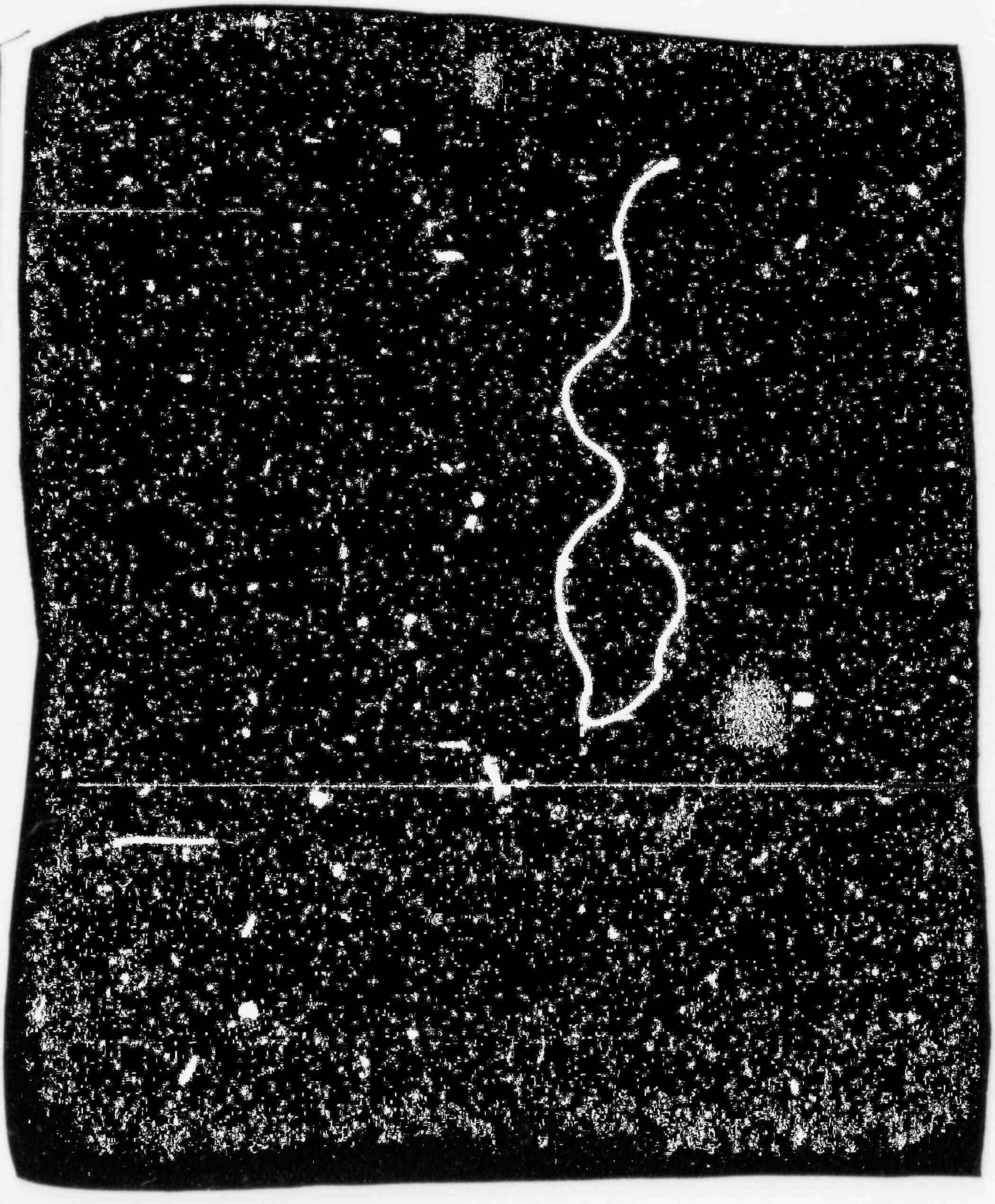
Component Functional Dependences - This type of dependence was partially addressed in the GESSAR-II PRA. Implicitly, the PRA assumed that the fault trees were developed up to a point that no functional dependence exists between the basic events (component failures).

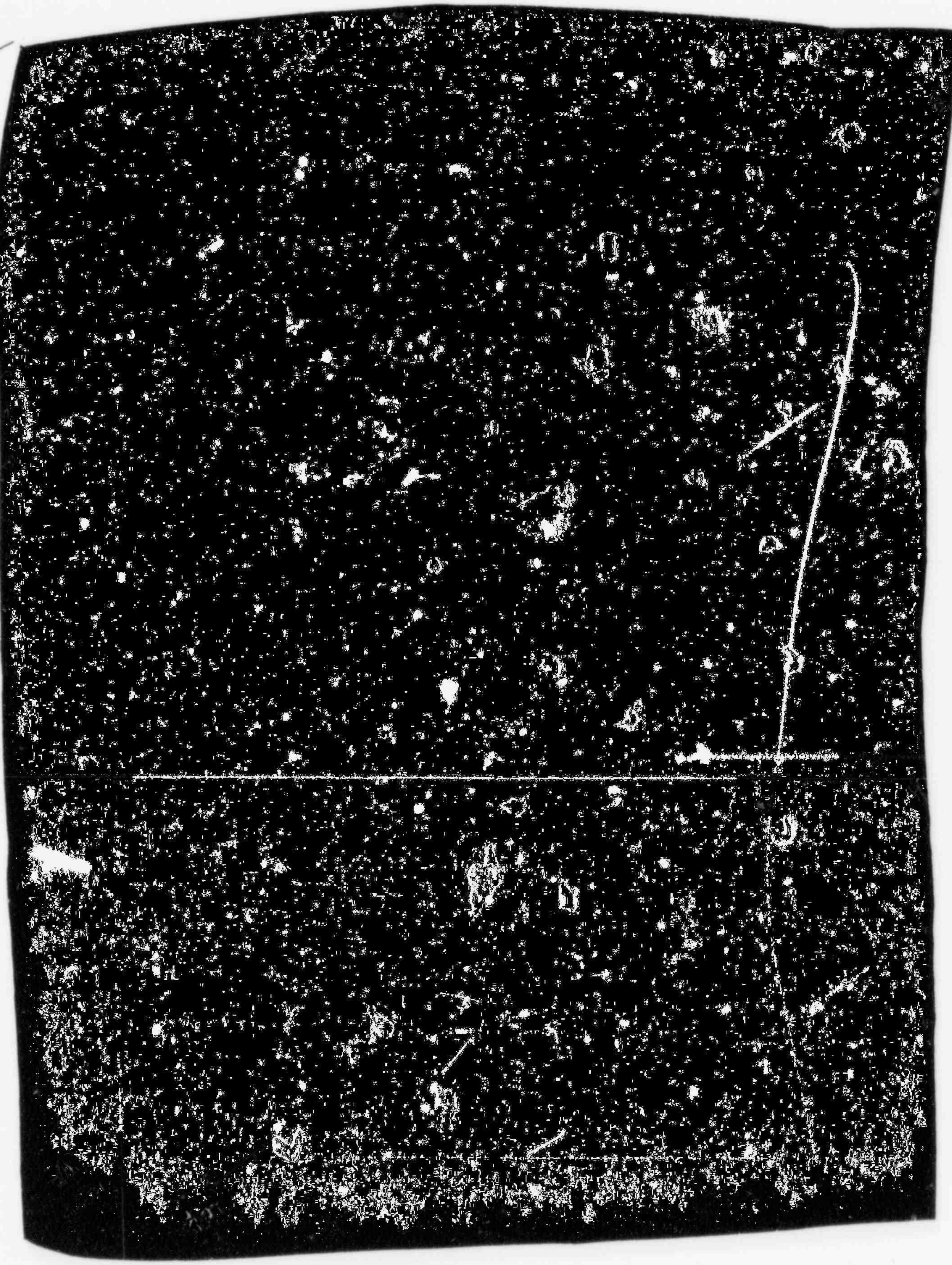
Component Physical Dependences - This type of dependence was not addressed in the GESSAR-II PRA.

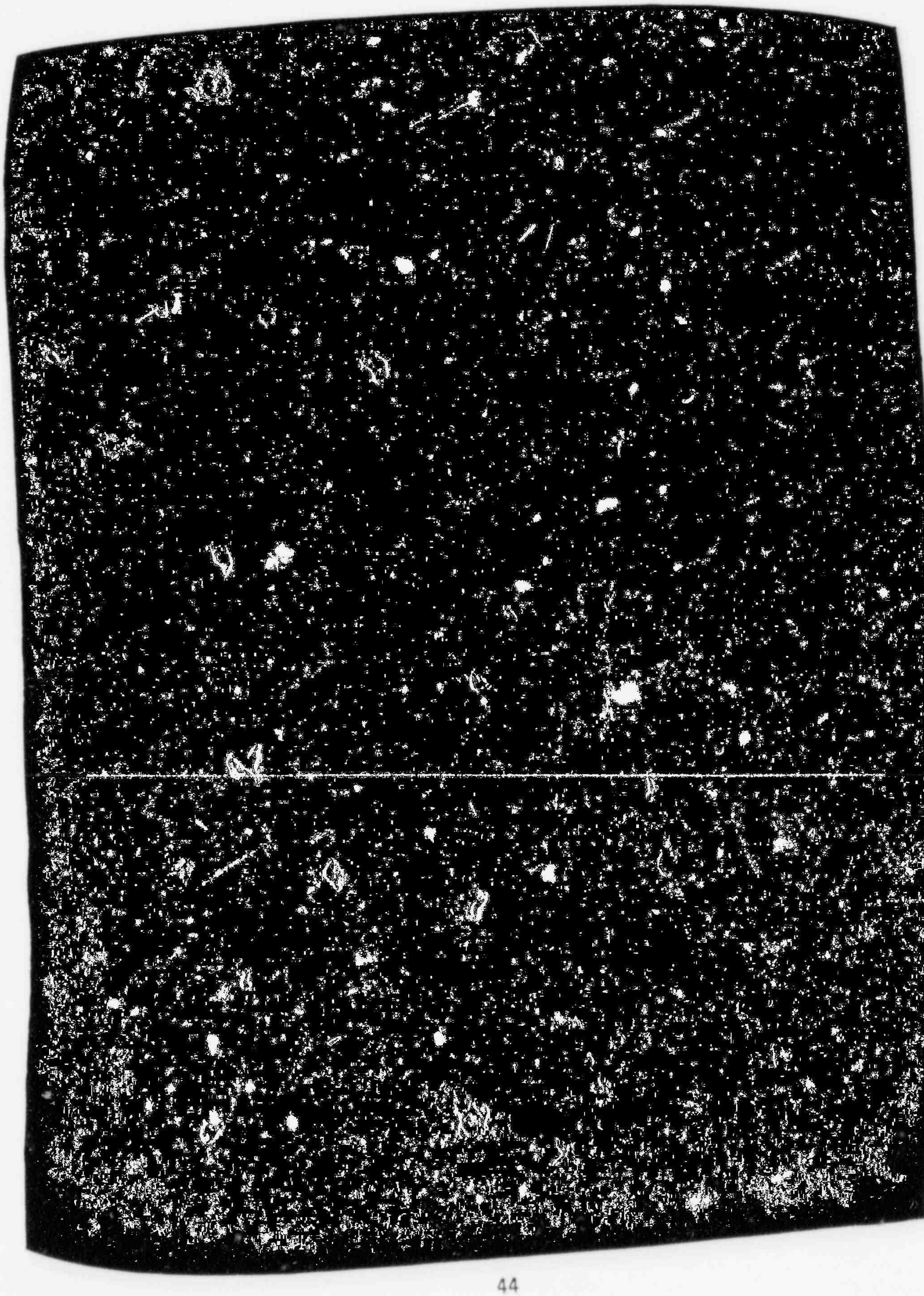
Component Human Interaction Dependences - This type of dependence was partially addressed by including in the analysis common mode failures of components because of operator errors during test and maintenance. Failures of multiple components owing to miscalibration were included in the system fault trees.

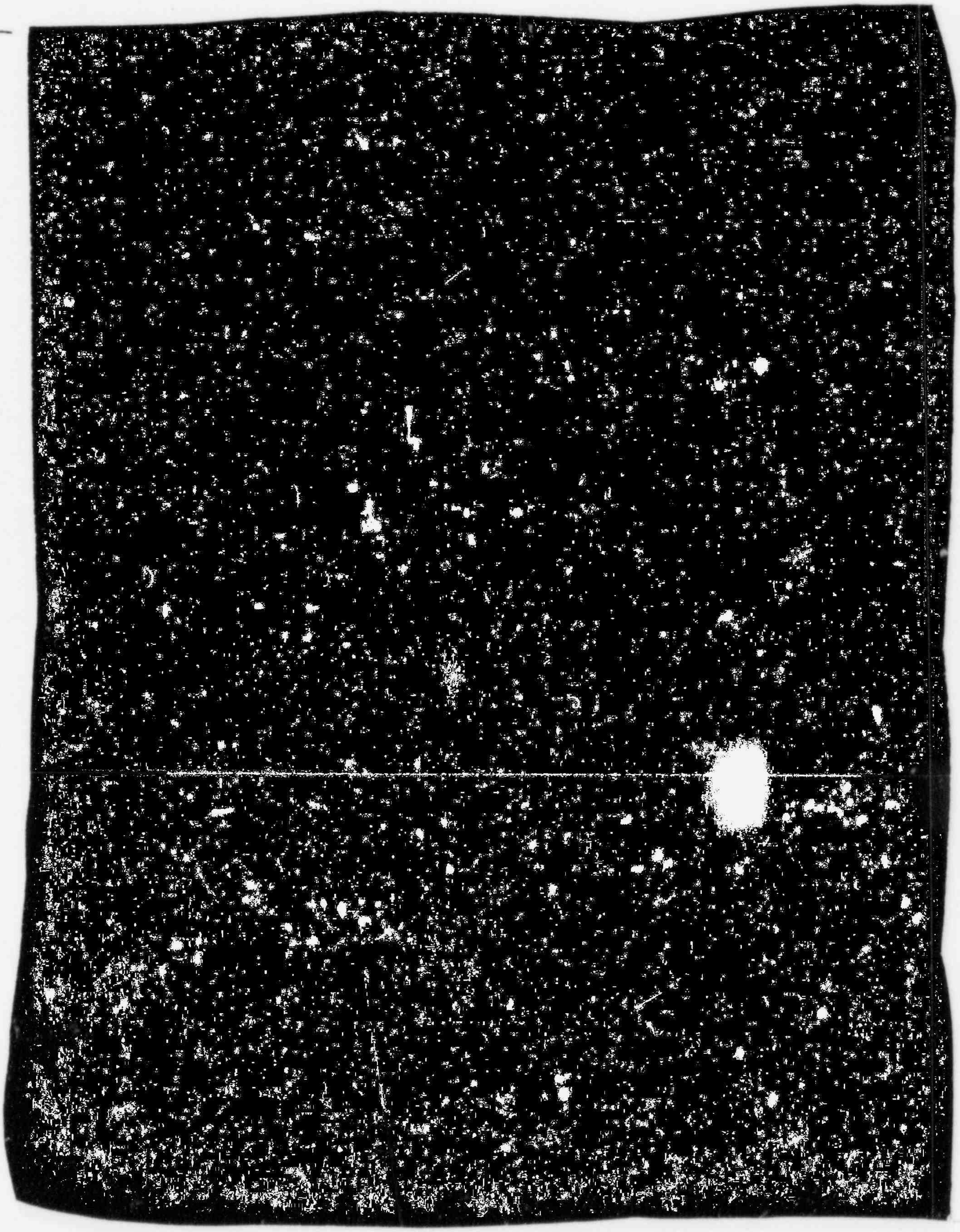
3.1.3 Transients and Manual Shutdown Events

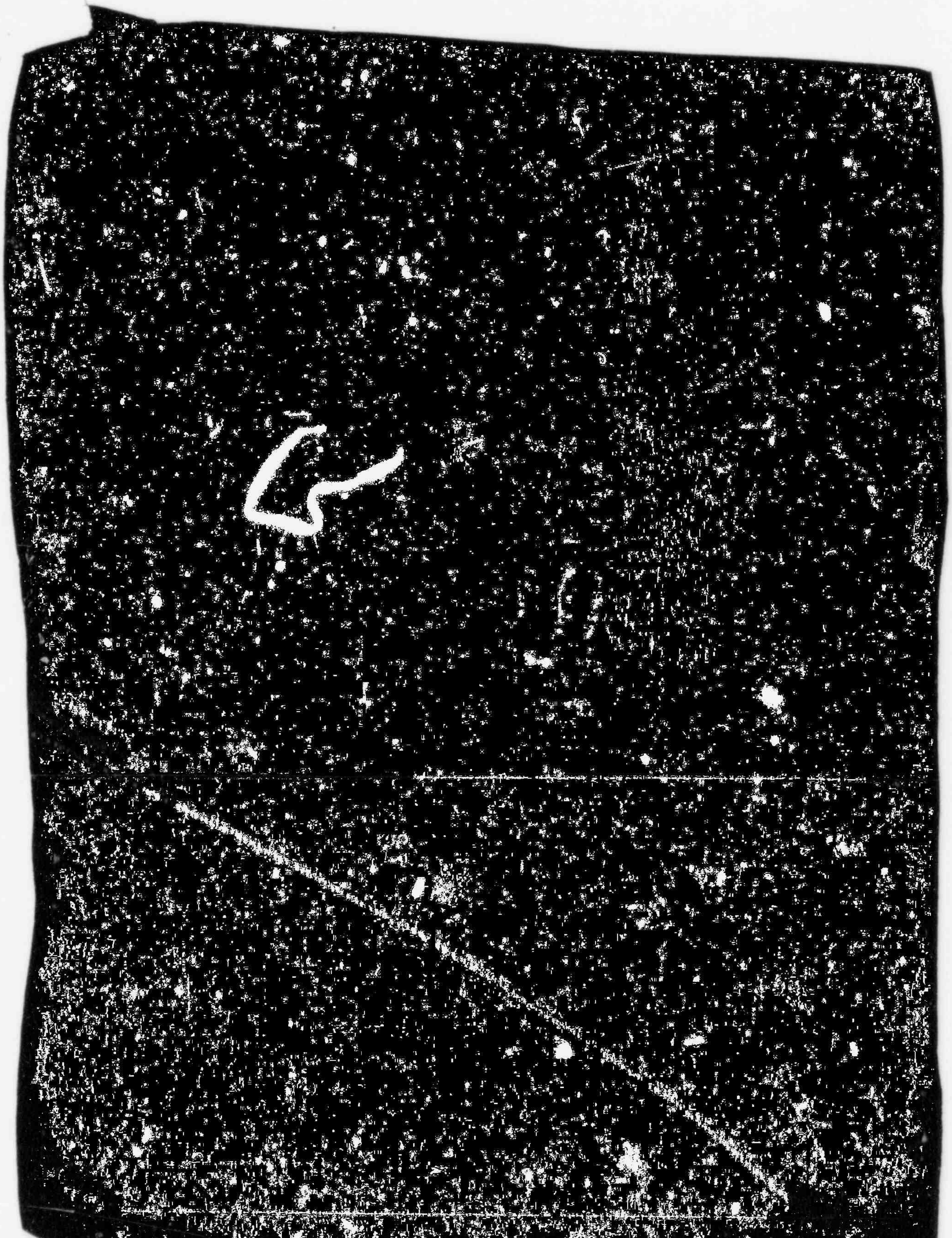
[redacted]

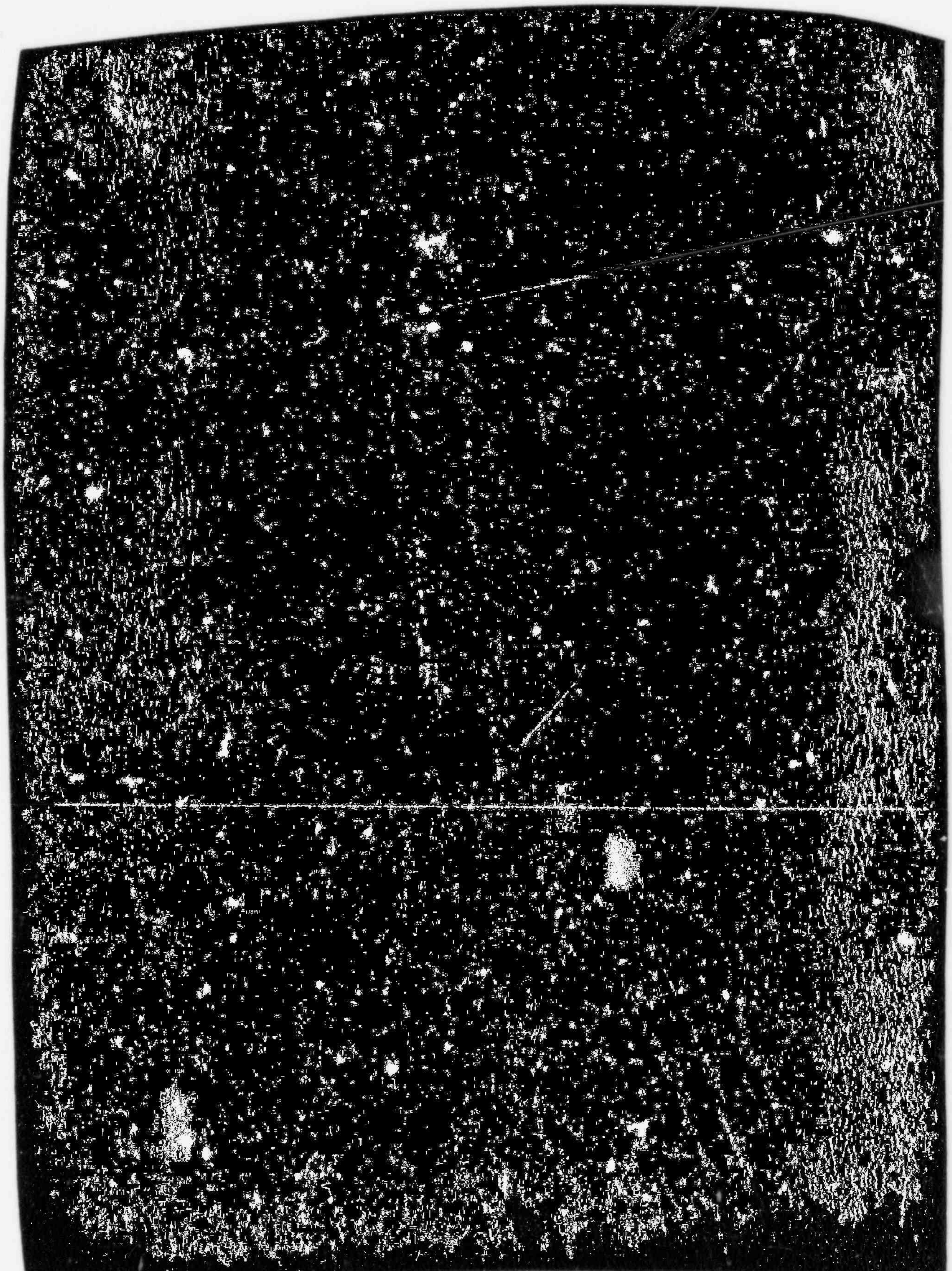


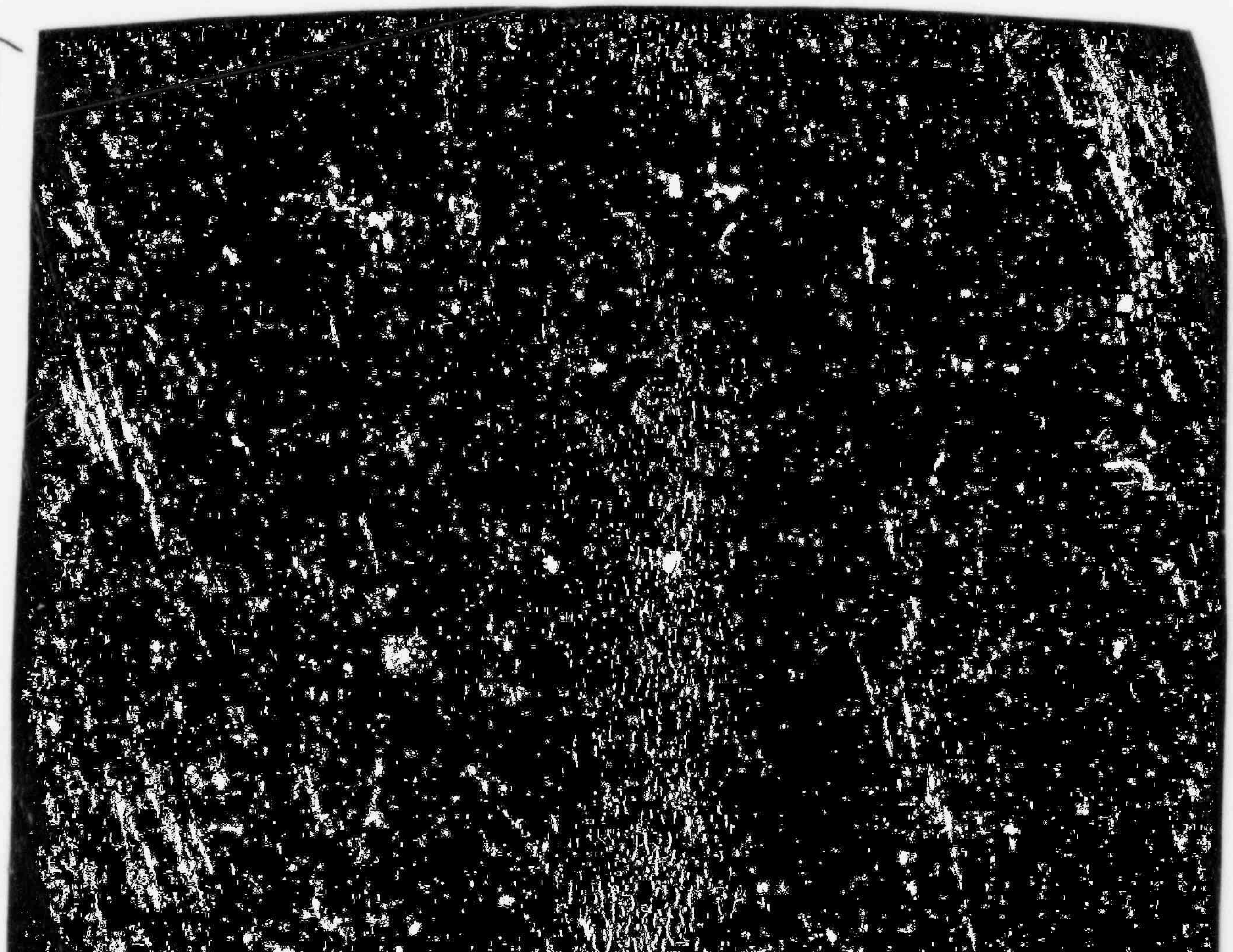


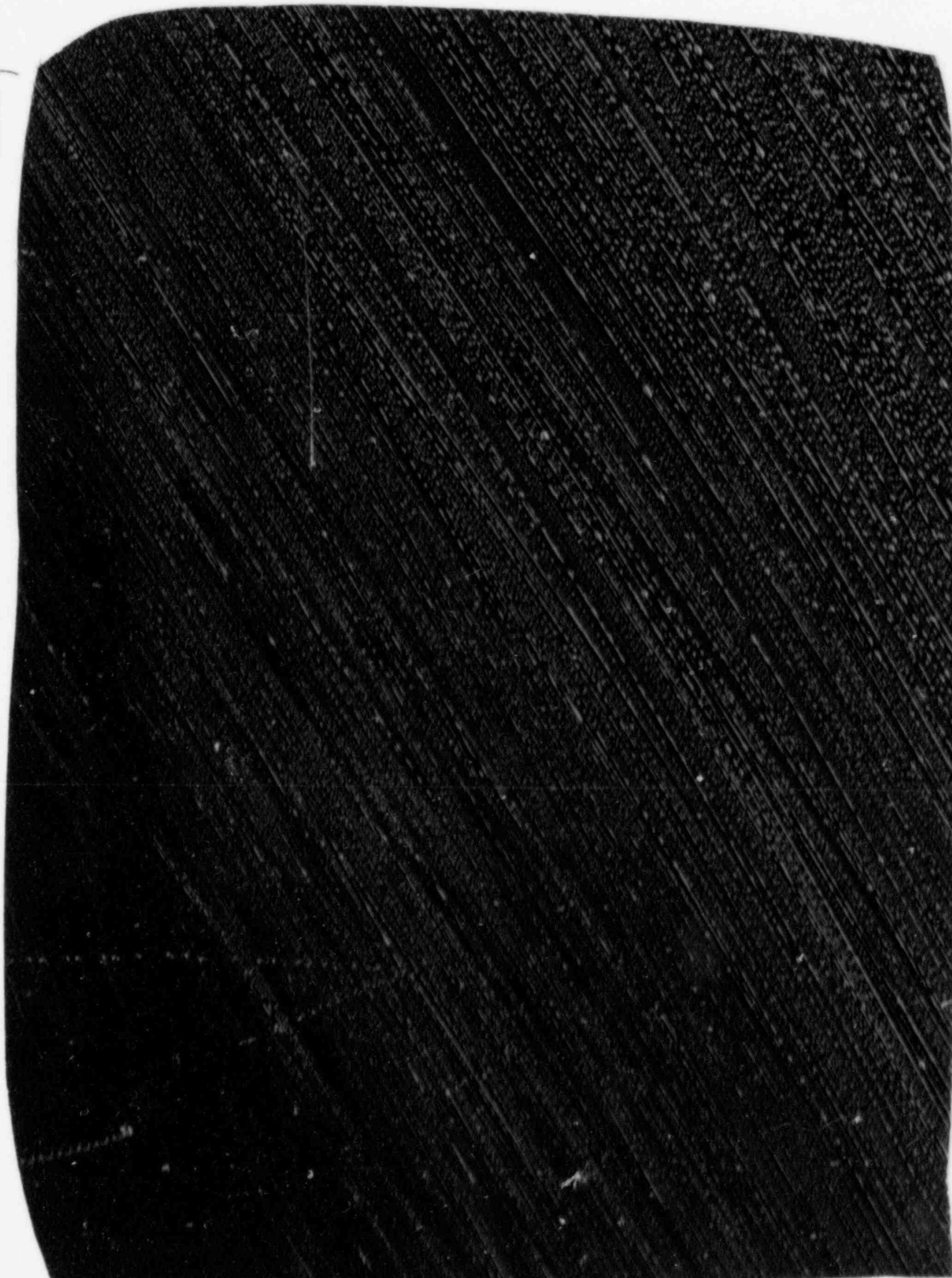


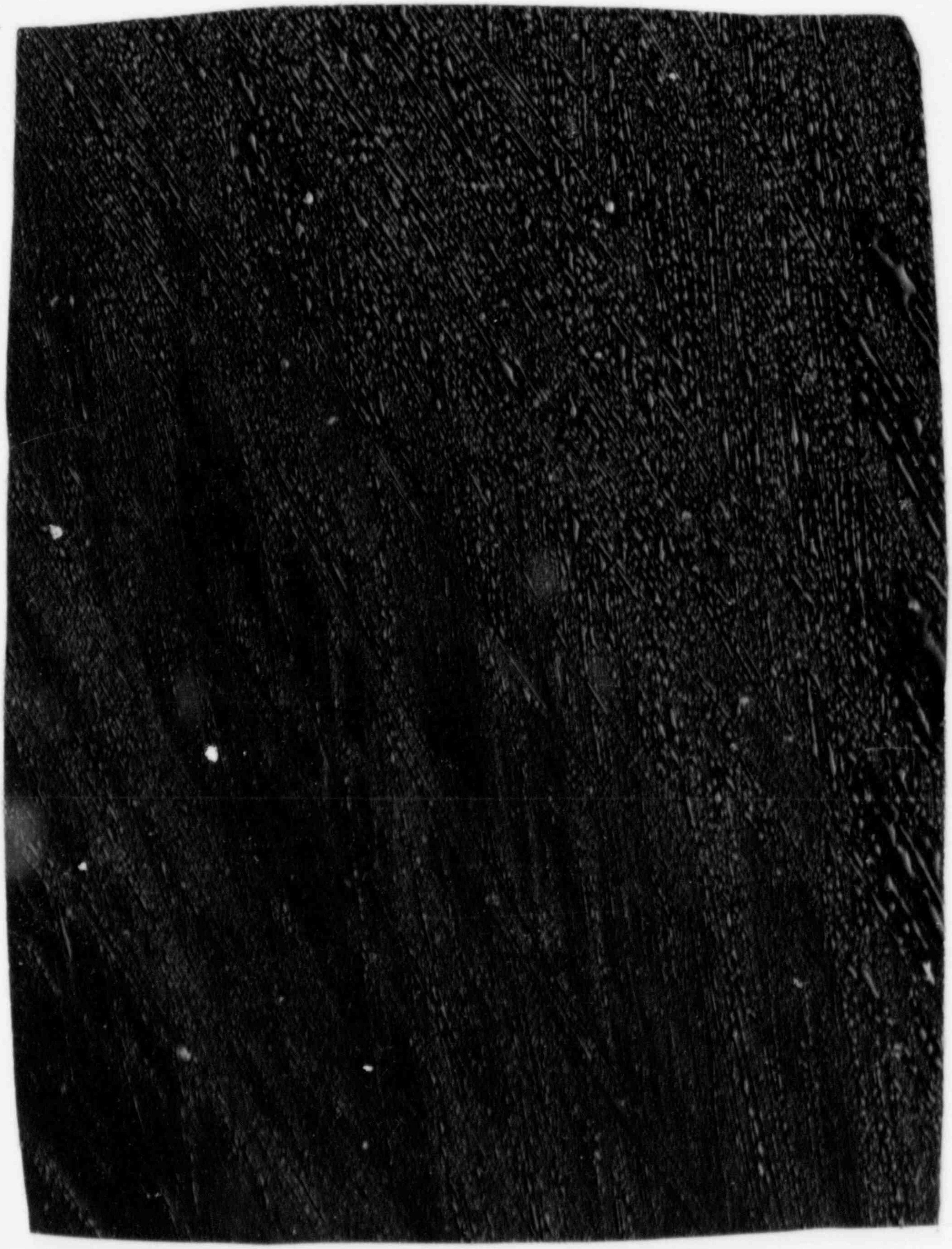












[redacted]

3.1.5 LOCAs

In the GESSAR-II PRA, three types of LOCA are considered to be possible inside the drywell: large, intermediate, and small. A separate event tree models the plant response for each of them (see Figures 3.18-3.20). In addition, three types of LOCAs inside containment--large, intermediate, and small--are discussed in the text of the GESSAR-PRA. The large and the intermediate LOCAs inside containment are assumed to contribute negligibly to the final core damage frequency, and details are not presented in the report. On the other hand, the small LOCA inside containment, which is believed to be a more credible event is treated in more detail, see Figure 3.21. Lastly, the small LOCA outside containment is also considered in the GESSAR study (Figure 3.22).

The GESSAR-II PRA presented a quite detailed study of the various types of LOCA events which could conceivably occur within the nuclear power plant. BNL concurs with the overall approach used in the development of these LOCA event trees, which were basically adopted in BNL's re-assessment of the LOCA core damage frequency. Minor improvements had been made to the event trees to give more credit for the availability of the feedwater system whenever it is possible. In the event that there is a loss of offsite power, the common mode failure of the diesel generators and the recovery of offsite power as well as the diesels are also included. The scram and ARI functions are modified to properly reflect the relative relationship of electrical and mechanical all line failures, and the ARI system.

3.2 System Fault Trees

The system fault trees are contained in Section D.2 of the GESSAR-II PRA. The following system fault trees are given in the PRA:

High pressure core spray (HPCS)

Reactor coolant isolation cooling (RCIC)

Automatic depressurization system (ADS)

Low pressure coolant injection (LPCI)

Low pressure core spray (LPCS)

Residual heat removal* (RHR)

Suppression pool makeup system (SPMS)

Essential service water (ESW)

Standby liquid control system (SLCS)

Electric power system (EPS)

Redundant reactivity control system (RRCS)

There are no system fault trees developed for the following systems:

a) Reactor protection system - the unavailability value given for the scram and ARI functions was $1x10^{-7}$ based upon an internal GE study. This value was not used in the BNL review (see Section 4.4.2).

b) Plant air systems (support system)

c) Feedwater and PCS

d) Condensate

The GESSAR-II PRA states that the main steam and feedwater systems (beyond the nuclear boundary), the main condenser and its supporting systems,

---

*This system has four trees associated with it. One tree applies for shutdown cooling, one for containment sprays, one for suppression pool cooling, and one for steam condensing. The steam condensing and the shutdown cooling functions of the RHR were not considered in the GESSAR-PRA accident sequence evaluation.

the condensate systems, portions of the essential service water system, the plant air supply systems, and the offsite electrical supply systems are not part of the GESSAR-II nuclear island. Thus, the PRA assumed values for the unavailability of these systems based upon judgement.

The impact of the omission of fault trees for the above systems has not been evaluated, but it may be important. If a full nuclear plant design, based on the GESSAR-II nuclear island, is to undergo risk evaluation, then reevaluation of these systems (including generation of system fault trees) should be considered to assess the impact on core damage frequency and risk.

The GESSAR-II PRA has a fault tree for the Redundant Reactivity Control System. However, since this is a system which was assumed to have been incorporated into the nuclear island design only for the purpose of the PRA, no drawings of the RRCS exist in the GESSAR-II SAR. Therefore, the fault tree could not be reviewed.

The BNL review of the fault trees resulted in some additions and revisions which are discussed next. The quantitative effects of these changes are discussed in Section 4.3. In general, the level of resolution in the fault trees (down to the component level, if data are available) is consistent with the state-of-the-art PRA practice. Two states are considered for each component in the fault trees: either the component operates as designed or it fails. The following items were not included in the analysis of the failure of a component (or system):

a) External events (including earthquakes, fires, and floods)

b) Sabotage

c) Operator errors of commission

The effect of external events is analyzed in a separate study, the review of which will be presented in a separate report.[1] Items (b) and (c) above are outside the scope of the PRA. The following items were also part of the BNL review:

a) Manual Actuation of Coolant Injection Systems: In the GESSAR-II PRA a grace period of 30 minutes, i.e., time availabe for initiation of coolant injection before core damage begins (see footnote on p. 3-9), for transients with successful scram, was assumed for manual initiation of coolant inventory makeup systems in the case of failure of auto-initiation. A description of the analysis of the BWR meltdown scenario given in Appendix F of the GESSAR-II PRA seems to support this grace period. It is noteworthy that for LOCA and ATWS events there is no grace period.

b) Dependences within a system and among systems: The GESSAR-II PRA states that those dependences were treated by using the same alphanumeric designator for components that appear several times in the same fault tree or in fault trees for more than one system. BNL review found that this policy was not followed consistently, and changes were made to correct this discrepancy.

c) The failure rates used in the GESSAR-II PRA system fault trees were point values and were meant to represent the average over the plant life. These data were based on References 5-9. The same failure rate were used in the BNL revised fault trees.

3.2.1 Summary of BNL Modifications to GESSAR-II System Fault Trees

A thorough review of each fault tree was performed using GESSAR-II SAR system descriptions, drawings and FMEAs. The failure rates used in the GESSAR-II PRA system fault trees were point values and represent average values over the plant life. These data were based on References 5-9. The same failure rates were used in the BNL revised system fault trees; the only exception is the common mode failure of the three diesel generators which BNL revised and used in the LOOP event tree (see Section 4.6.1).

The following is a list of changes that were made to the trees which changed some of the system unavailability values. A more detailed list of these changes is given in Appendix A.

High Pressure Core Spray

a) Included the failure of Division IV power (EV1D and EV3D) as a failure mode of channels D and H.

b) Accounted for failures of the HPCS pump suction supply in which automatic transfer from the CST to the suppression pool is precluded.

c) Accounted for mechanical failures of HPCS valves in which manual repositioning of the failed valves is precluded.

d) Changed some basic component names to account for commonalities with other systems.

e) Deleted the high drywell pressure signal to initiate the system for transient events.

Reactor Core Isolation Cooling

a) Added failures of bus dc E as a failure mode of RCIC.

b) Added failure of V1 and V3 power as a failure mode of RCIC initiation signal.

c) Changed names of basic components to properly account for commonalities with other systems.

d) Accounted for mechanical failures of RCIC valves in which manual repositioning of the failed valves is precluded.

e) Accounted for failures of the RCIC pump suction supply in which automatic transfer from the CST to the suppression pool is precluded.

f) Inserted the proper portion of the service water system fault tree to properly account for this support system.

Automatic Depressurization System (ADS)

a) The designator of the failure of the operator to open non-ADS SRVs and the failure of the operator to manually initiate ADS given auto-initiation failure was changed from two separate names (AHUSRV and AHUADS) to a single name (AHUDEP) to account for his common failure to depressurize.

b) Dc and ac support systems were changed from basic independent events to properly account for inter and intra system common mode failures. The corresponding gates in the EPS fault tree were used in the BNL modified fault tree for ADS.

c) The reactor low water level signal nomenclature for failures of the transmitter, ACU, isolator, and common mode miscalibration of the transmitter were changed to conform to that used in the low pressure ECCS system fault trees. This was done in order to properly account for commonalities between these systems.

d) LPCS and LPCI pump failures to run were changed from basic independent events to properly account for dependences. The corresponding gates in the LPCS and LPCI fault trees were used in the BNL modified fault tree for ADS.

e) Deleted the requirement to have a high drywell pressure signal for ADS auto-initiation, as assumed in the GESSAR-II PRA (p. 15.D.3-28).

Low Pressure Coolant Injection

a) Included failures of electric power as a failure mode of the system.

b) Accounted for failure of manual valves such that operator action could not change the position of the valve.

c) Included failures of service water as a mode of failure of the pumps.

d) Changed the names of many components in order to properly account for commonalities between systems.

Low Pressure Core Spray

a) Accounted for failure of the system if the minimum flow valve (normally closed according to FMEA in GESSAR-II SAR) fails to open upon system initiation while the pump is running but the injection valve is still closed waiting for a low reactor pressure permissive signal.

b) Changed some component names to properly account for commonalities between systems.

c) Included failures of the electric power system as a failure mode of the system.

d) Developed pumping system auxiliaries failures.

e) Included additional injection line valve failures.

RHR Suppression Pool Cooling Mode

a) Decoupled operator failures in LPCI from this system by changing the names of components.

b) Accounted for mechanical failures of the system's valves which would prevent the operator from taking corrective action.

c) Changed some component names to properly account for commonalities between systems.

d) Included failures of the electric power system as a failure mode of the system.

RHR Containment Spray

a) Accounted for mechanical failures of system valves which would prevent the operator from taking corrective action.

b) Changed some component names to properly account for commonalities between systems.

c) Included failure of the electric power system as a failure mode of the system.

d) Corrected some incorrect valve designators.

Suppression Pool Makeup

a) Accounted for system failure because of loss of electric power.

b) Accounted for common mode miscalibration of the suppression pool level instrumentation.

Essential Service Water

Changes to the ESW system were made, when appropriate, directly into the frontline system it serves, and, thus, are reflected in the frontline system fault trees.

RHR Steam Condensing Function

This tree was not evaluated because this function was conservatively not used in the GESSAR-II PRA.

Standby Liquid Control System (SLC)

a) The failure of a pipe cap downstream of valve F025 was removed since there is no cap on this line.

Electric Power

a) The failure of service water to the three emergency diesel generators was treated as a basic event in the PRA. These basic events were changed to transfer to the ESW system fault tree in order to properly account for commonalities.

b) Changed the common mode failure probability of the three diesel generators.

c) Included failures of several breakers left out in the GESSAR-II PRA.

d) Included the non-class IE battery chargers as a means of supplying rectified power to Division 1 through 4 125V ac buses. The PRA did not take credit for the non-class IE chargers.

e) Changed the loss of 6900V ac preferred power to bus G (Division 3) to account for the lack of an alternate offsite supply to the Division 3 bus.

In summary, it can be said that the GESSAR-II system fault trees present a more detailed and complete description of the faults leading to system failure than those used in the Peach Bottom and Grand Gulf PRAs.

3.3 Human Performance Analysis

The two types of human errors (cognitive and procedural) that can contribute to the unavailability of systems are addressed in the GESSAR-II PRA; the way they were treated is discussed below.

3.3.1 Cognitive Human Errors

In the GESSAR-II PRA, cognitive human errors are explicitly modeled either in the event trees or in the fault trees. Those human errors with a description of the required action and the time available (or assumed) for action are given in Table 3.2 for errors modeled in the event trees, and in Table 3.3 for those modeled in the fault trees.

The BNL review is in agreement with the qualitative modeling approach to most cognitive human errors. However, as will be discussed in Sections 4.2 and 4.3, BNL disagrees with the GESSAR-II PRA quantification for most of the human errors explicitly modeled in the event trees. More specifically, disagreements exist in the quantification of the following human errors:

a) Operator failure to manually scram the reactor given an IORV (C_I).

b) Operator failure to inhibit ADS or, if ADS is initiated, failure to prevent reactor vessel overflow during an ATWS event (P_A).

c) Recovery of FW and PCS (included in the heading Q and W in the event trees).

It is noteworthy that a human error that is important in the Peach Bottom[5] and Grand Gulf PRAs,[10] namely, the failure of the operator to timely depressurize the reactor vessel in the case of a transient and a failure of the high pressure injection function, is of less importance in GESSAR-II. This difference is due to the GESSAR-II PRA assumed modification in the ADS logic, i.e., ADS is automatically actuated in the presence of a low reactor pressure vessel (Level 1) plus 10 minutes.

3.3.2 Procedural Human Errors

Procedural human errors contribute to component and/or system unavailability through routine operations such as in test and maintenance acts. The GESSAR-II PRA followed, in most cases, the techniques recommended in NUREG/CR-1278[9] for quantification of the procedural human errors. The BNL review was mainly concentrated on including omitted human errors which lead to common mode failure; it did not take up requantification of procedural human errors.

3.4 References to Section 3

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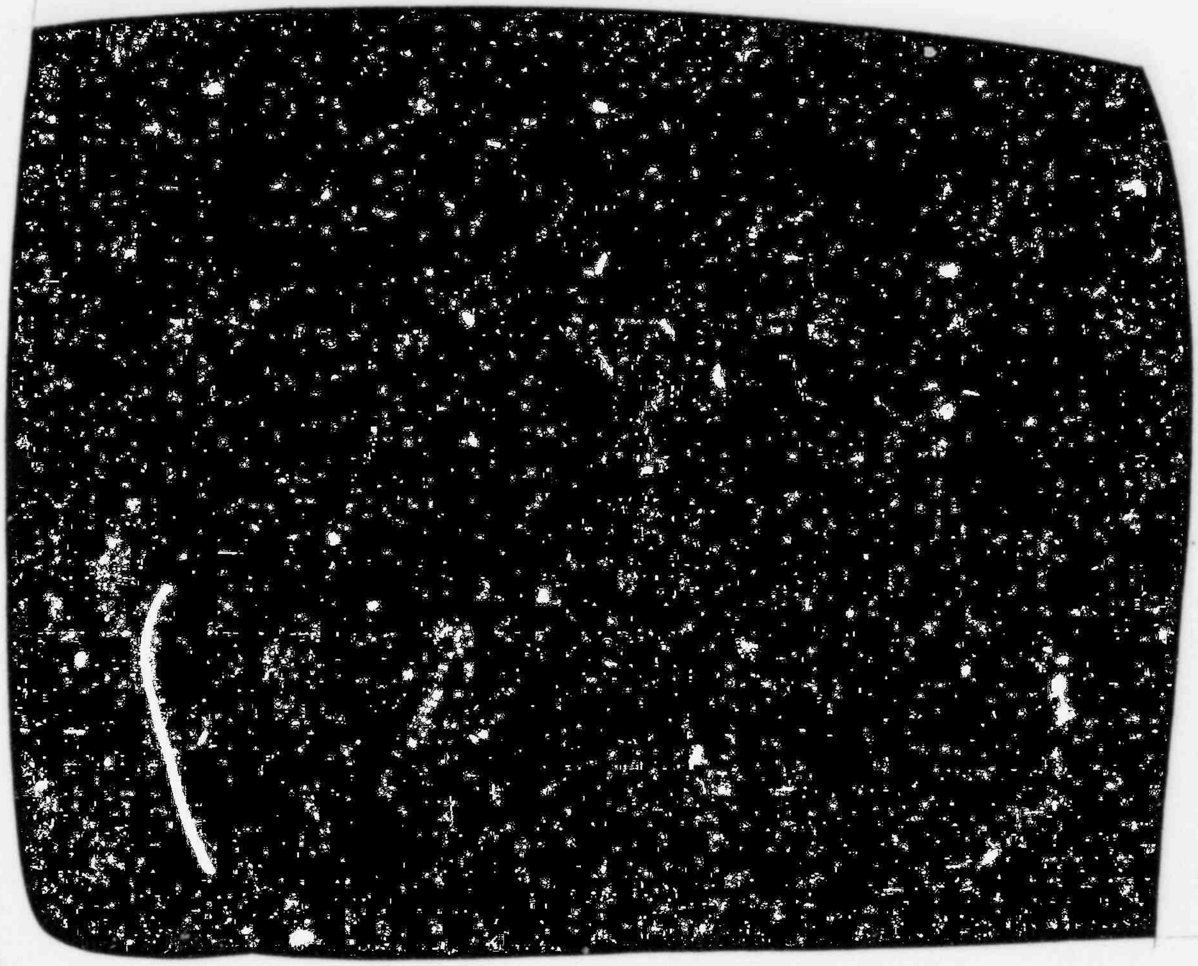

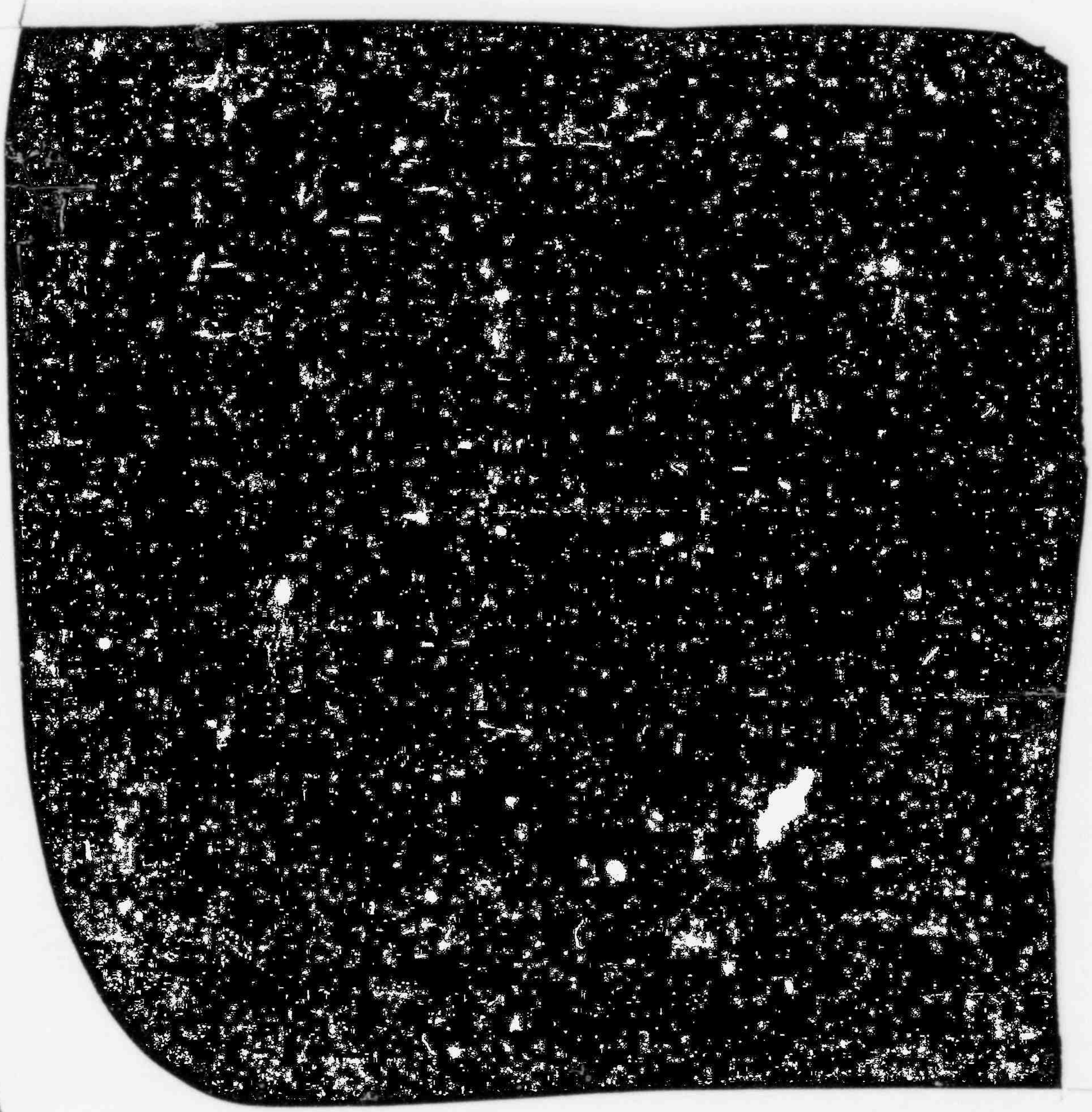

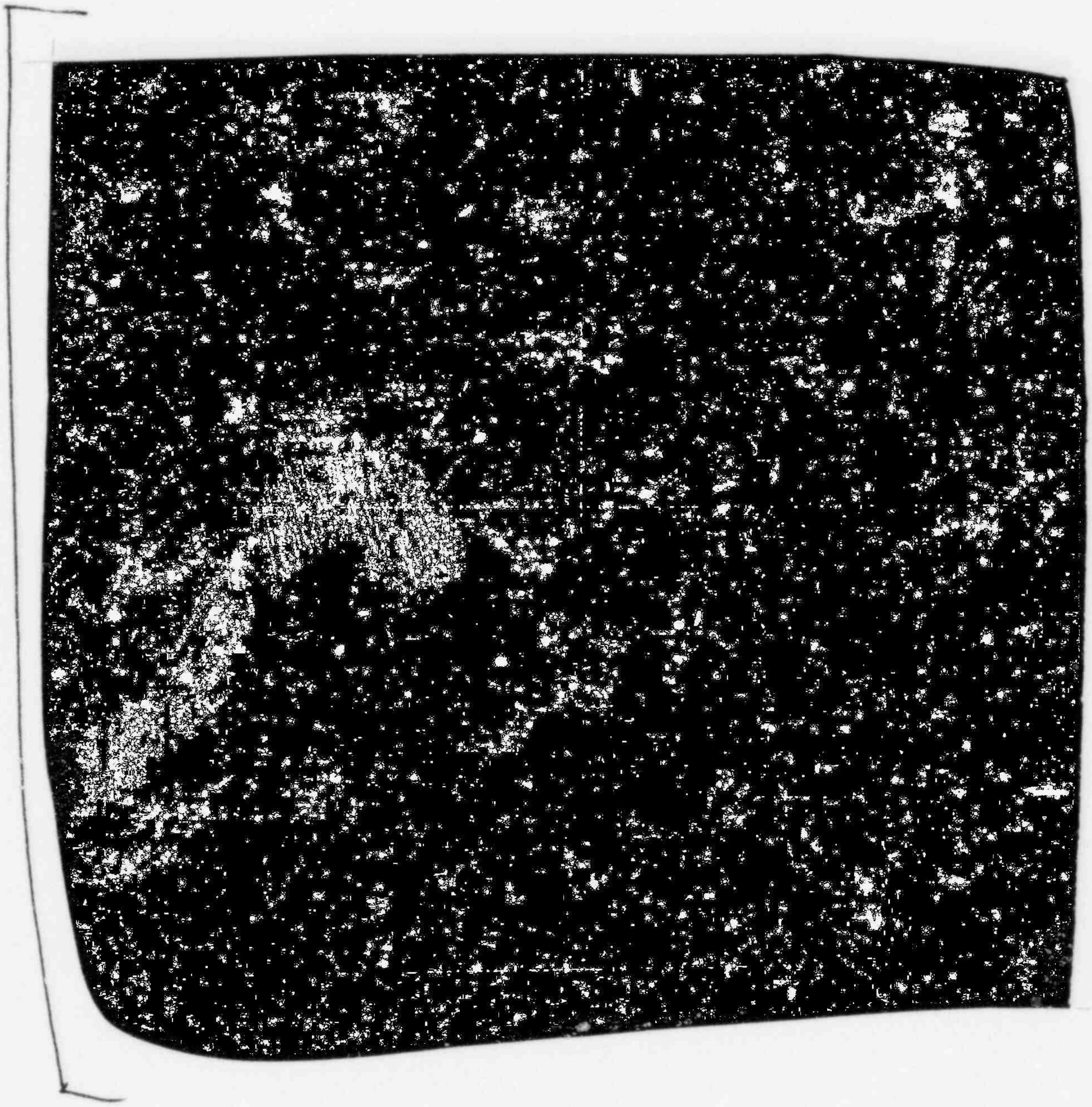

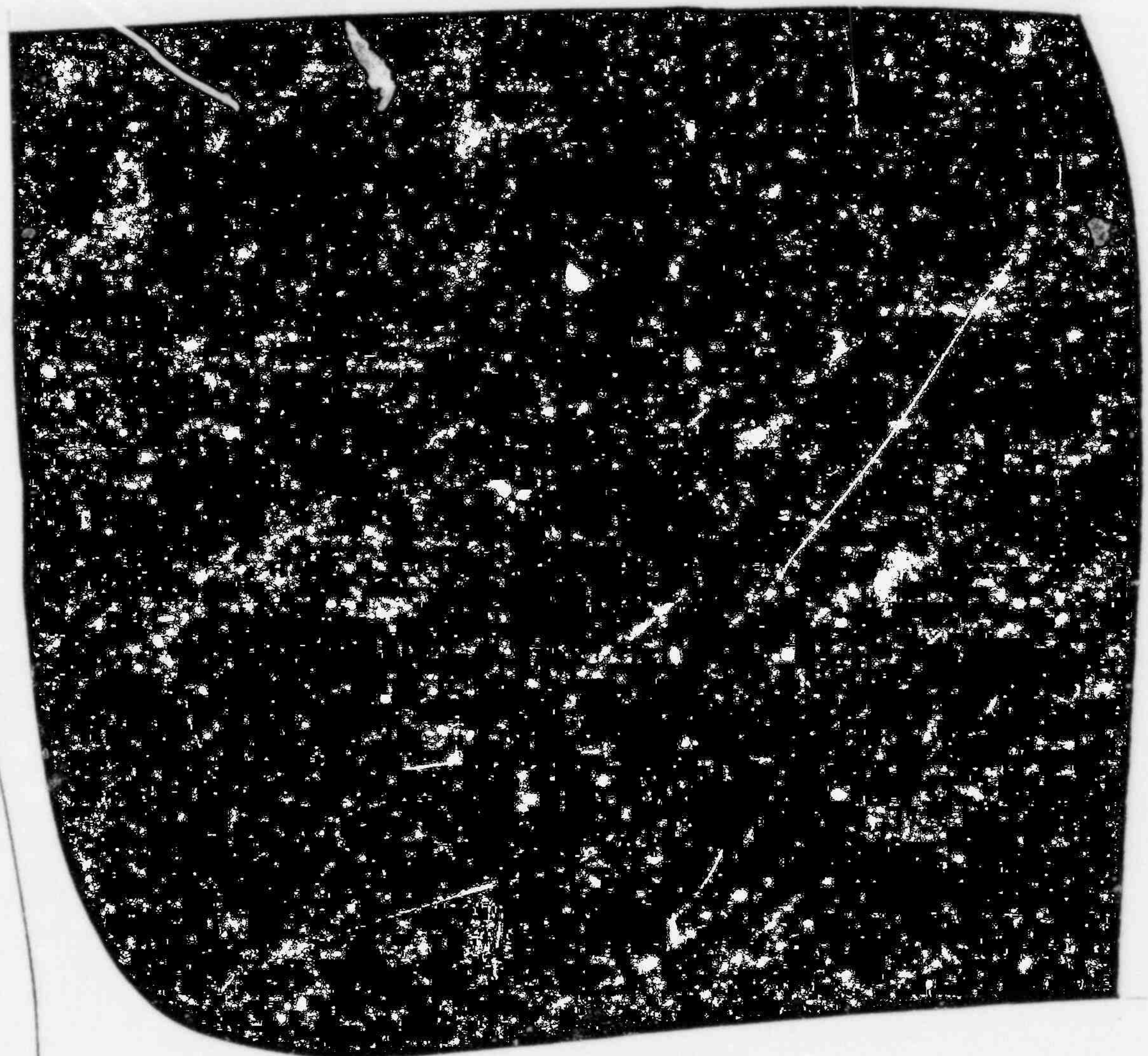

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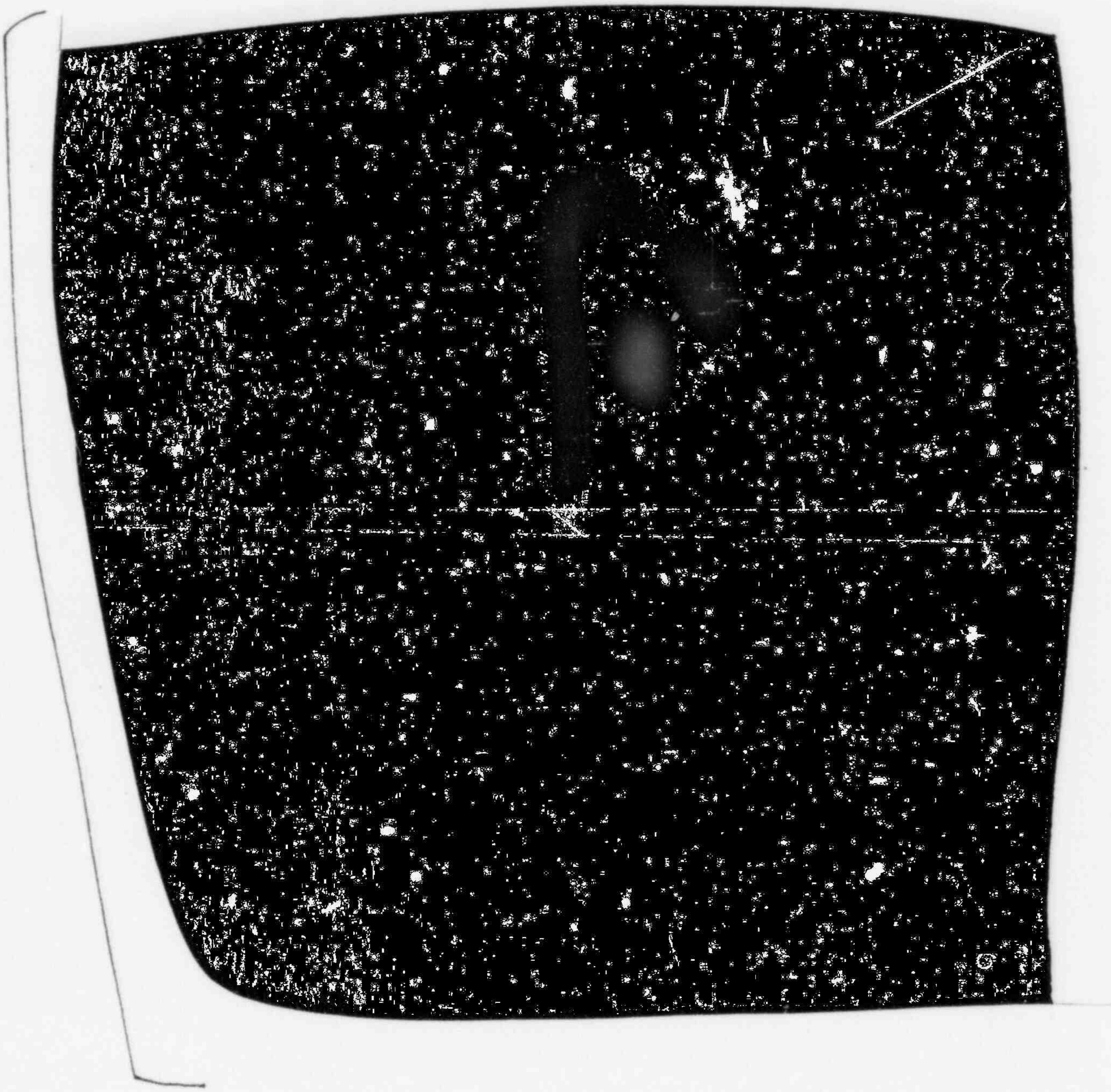

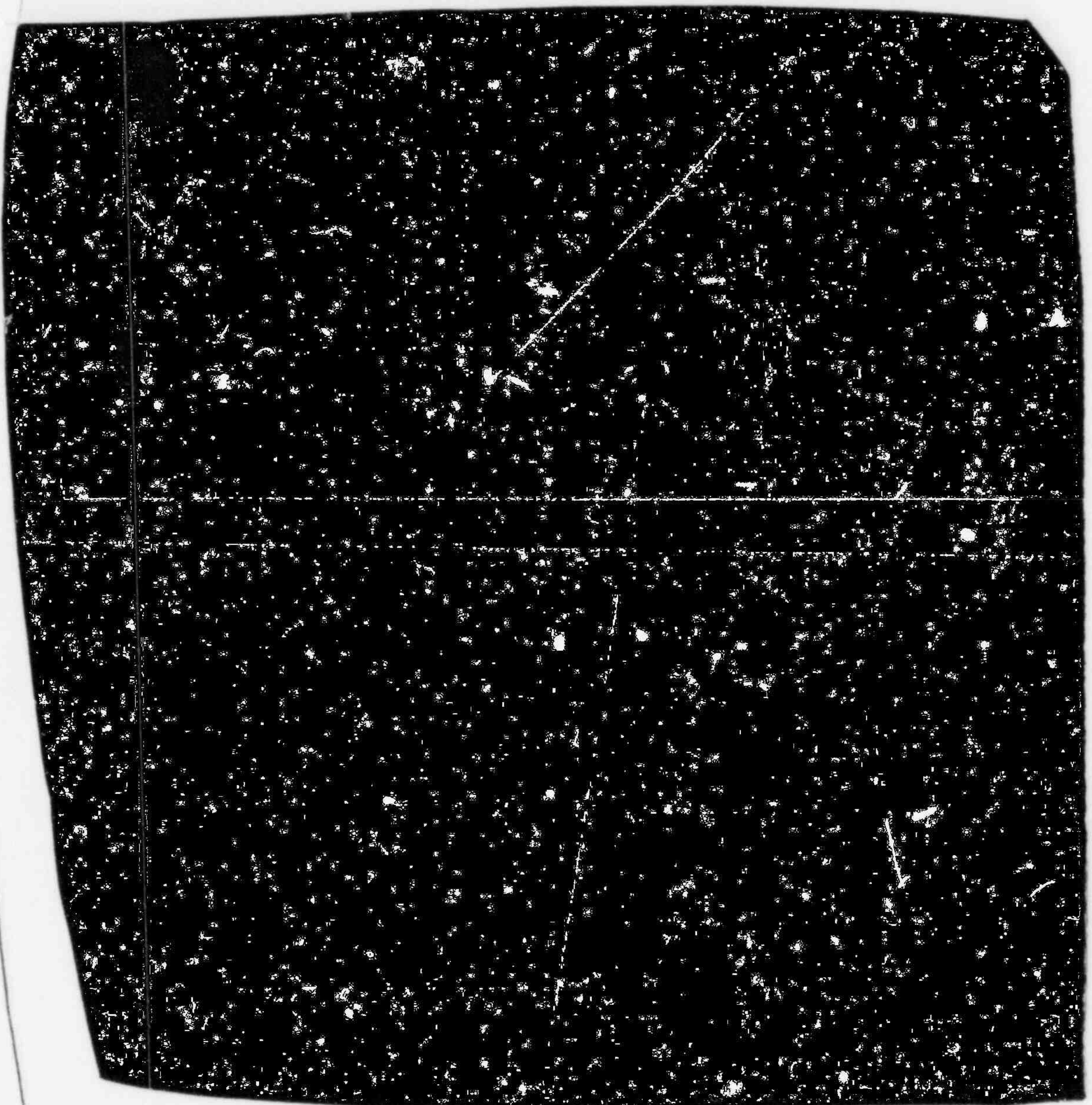

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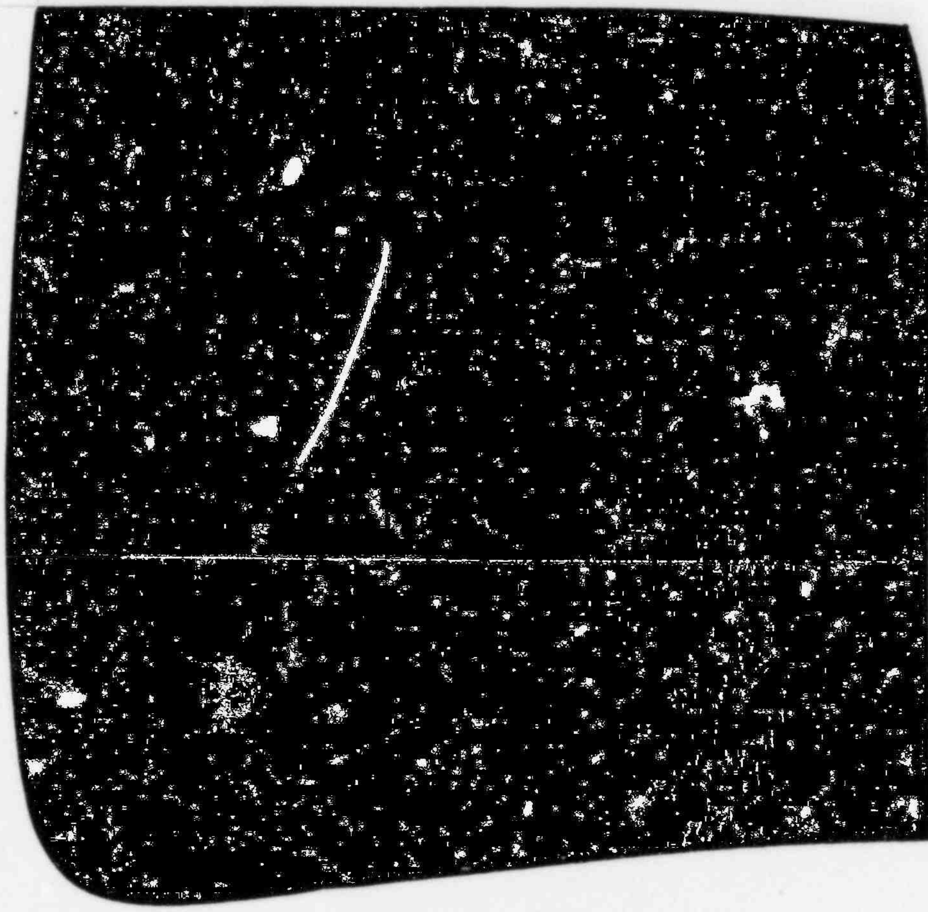

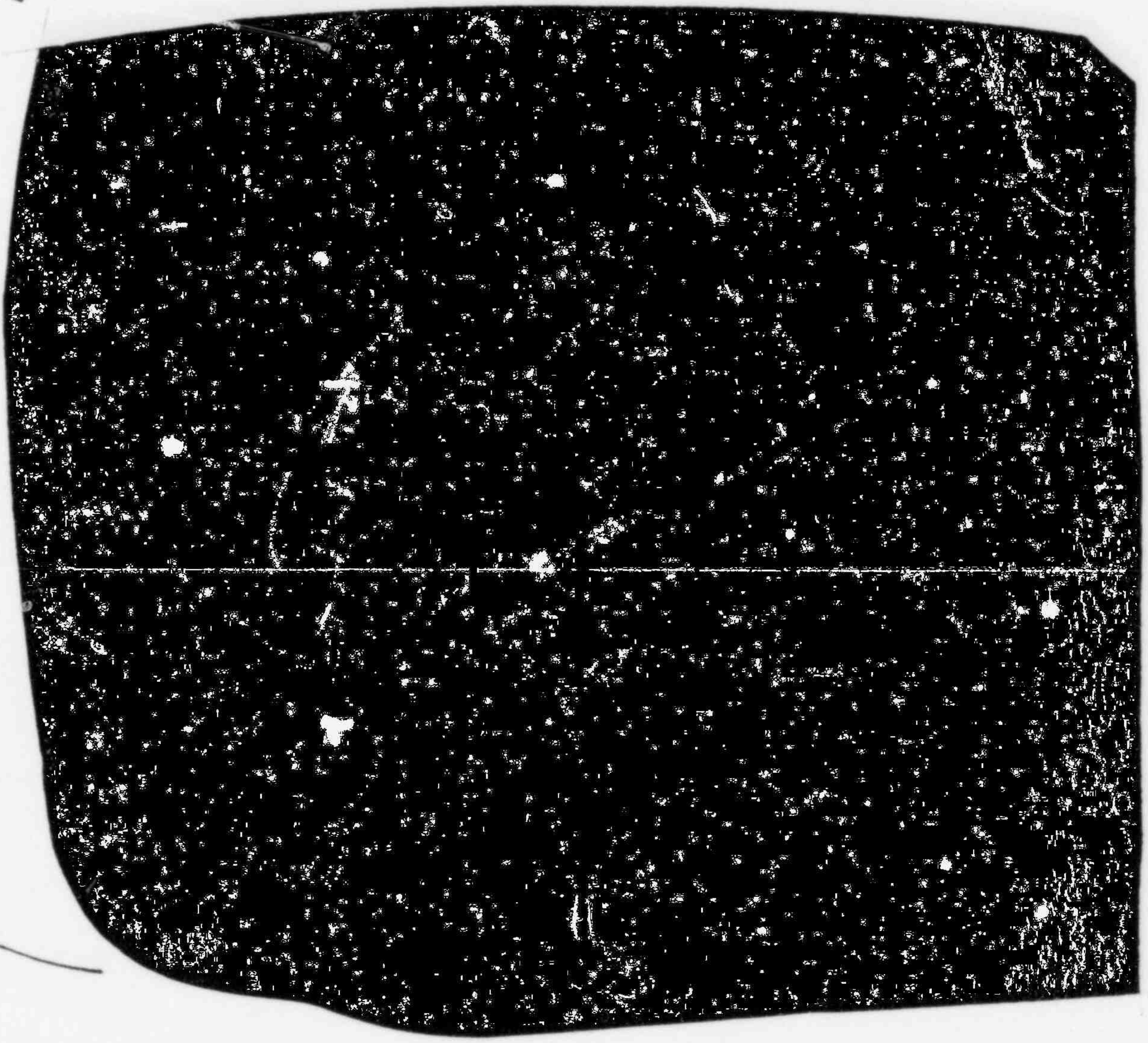

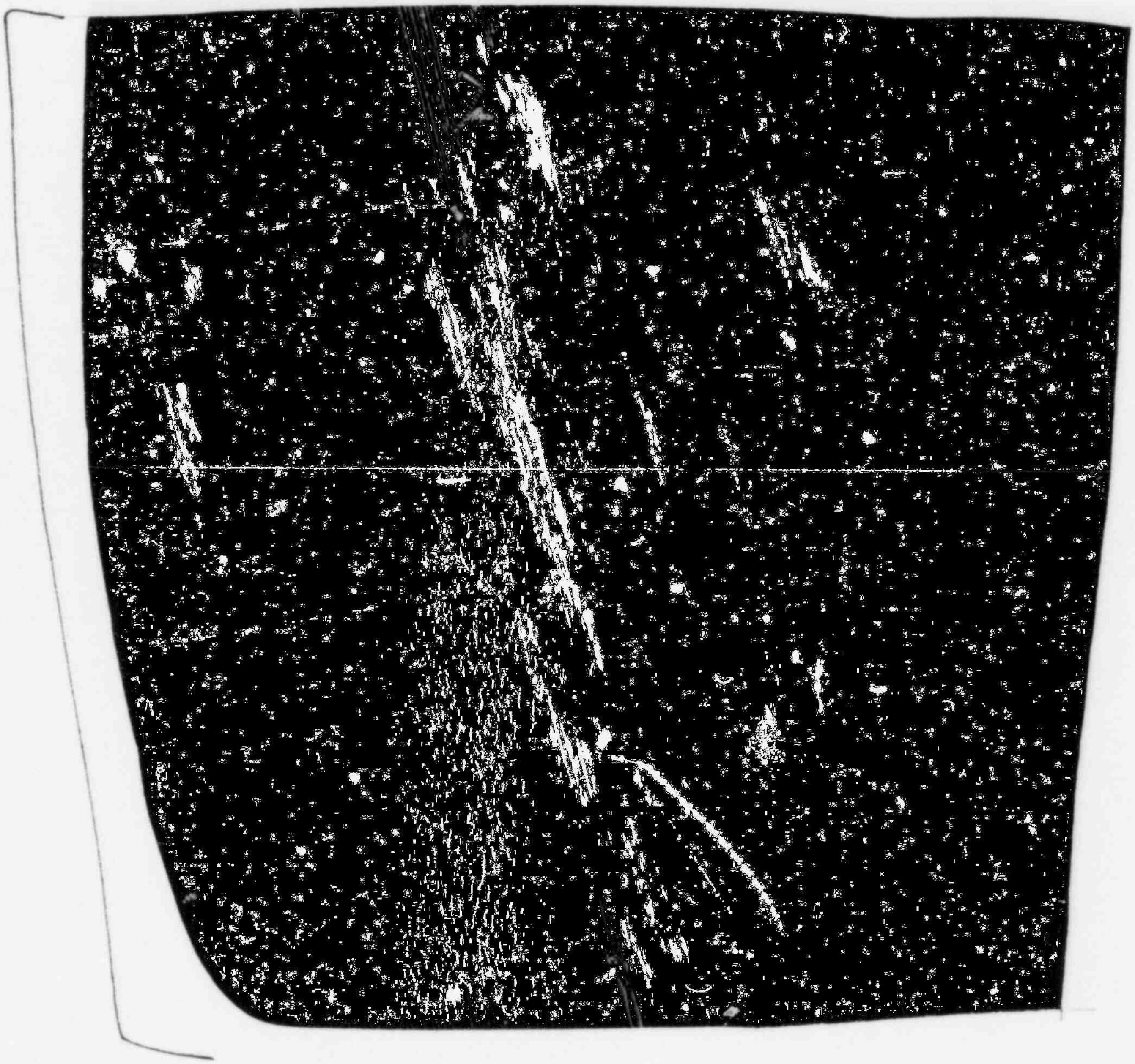

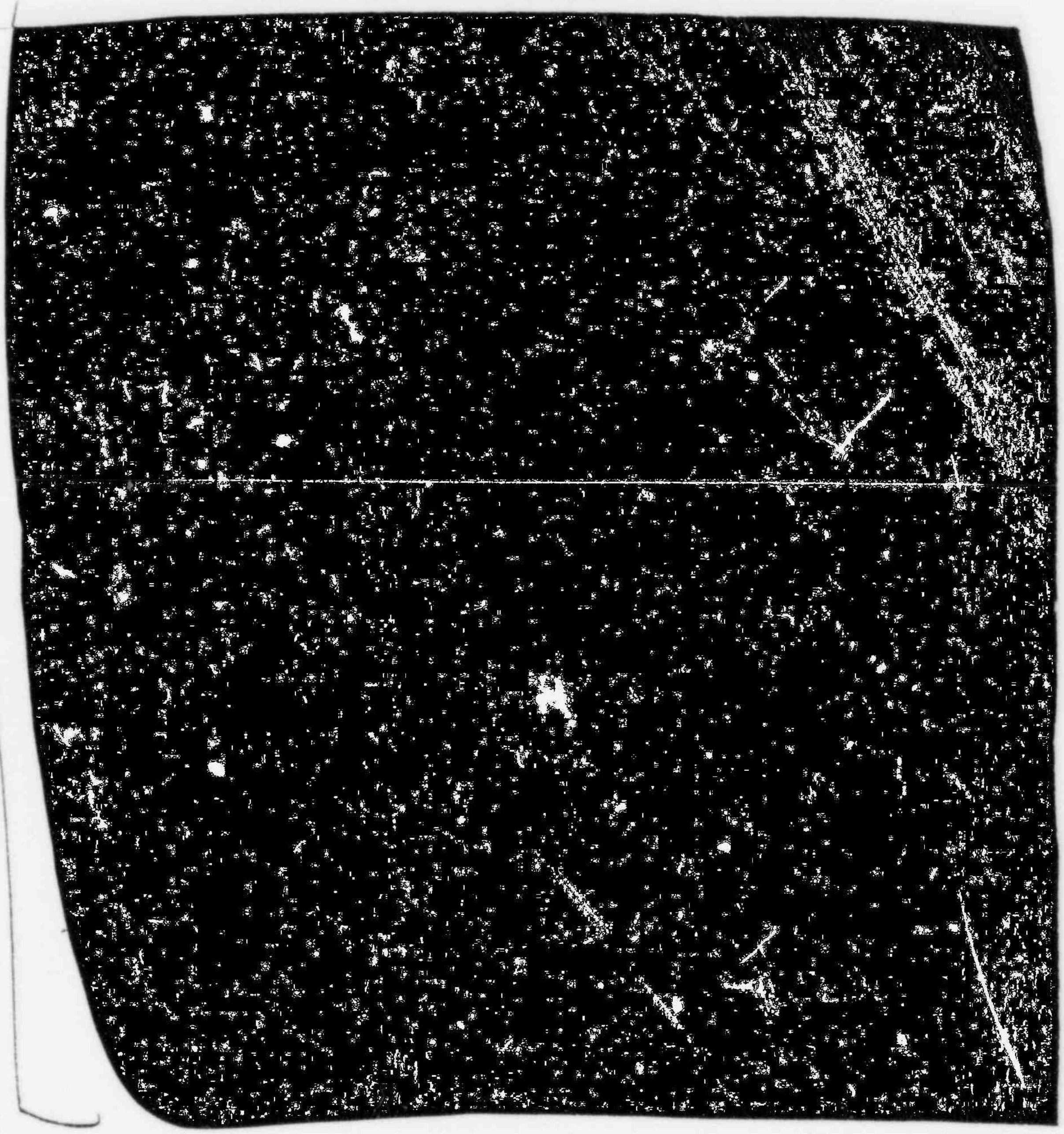

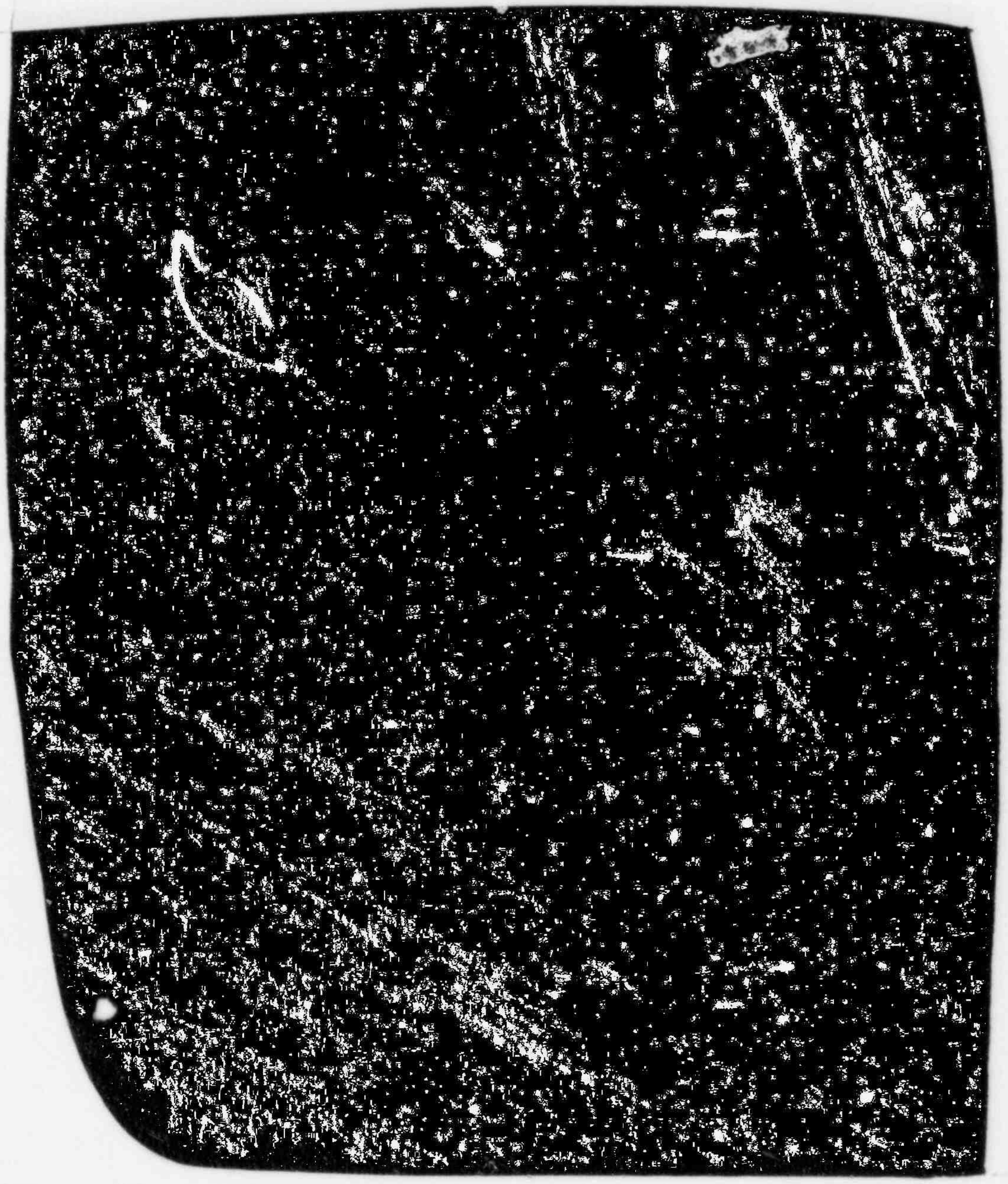

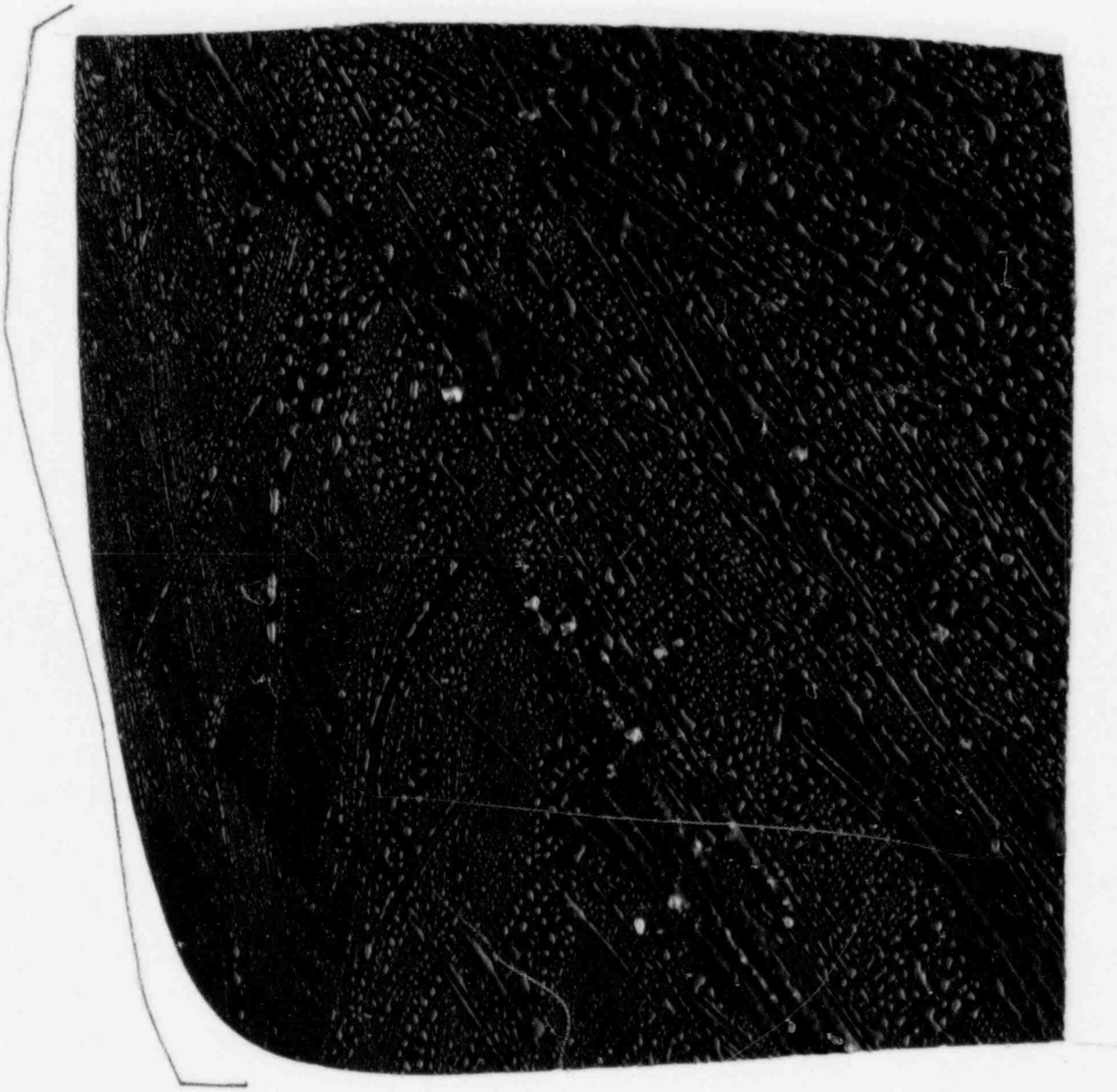

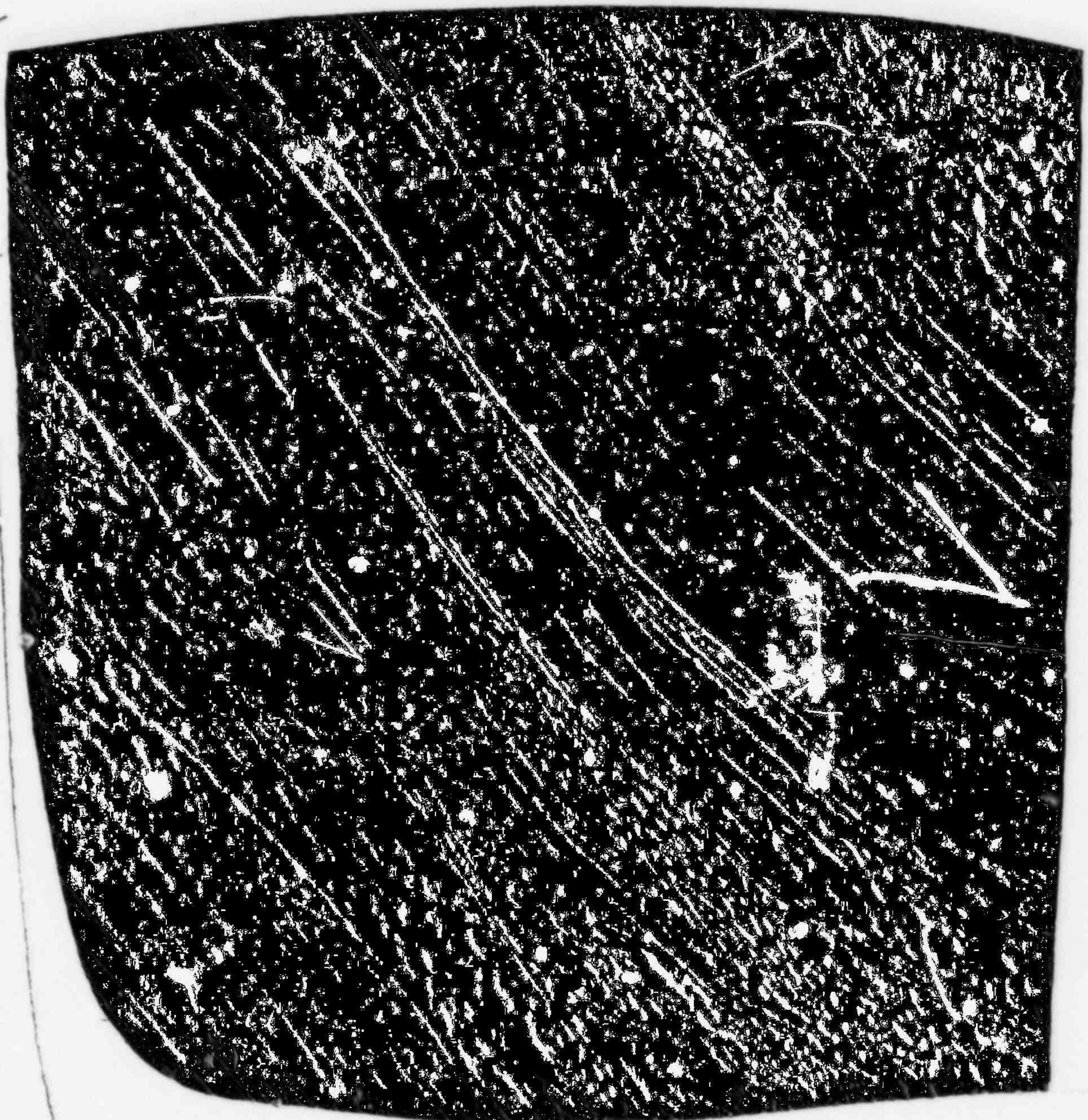

Table 3.1 Frontline/Support System Dependence Matrix

System	AC Power	DC Power	Cooling	Compressed Air	Suction	Human	Other
SRV	---	For manual or relief mode actuation.	---	Instrument air system for manual or relief mode actuation, require plant air. Each SRV has an accumulator which is available upon the loss of plant air.	---	For manual actuation of SRVs.	Suppression pool required for SRV discharge steam condensation.
ADS	---	For initiation and actuation.	---	Pneumatic Supply System. Same as above.	---	For manual ADS actuation.	Same as above.
HPCS	Class IE (Dedicated EDG No.3)	For initiation and control.	Independent ESW system which only supplies HPCS.	---	CST with automatic switchover to the suppression pool on low CST level, or high SP level.	For manual initiation.	---
RCIC	Class IE (for room cooling)	For initiation and control.	ESW system for room cooling.	---	CST with automatic switchover to the suppression pool on low CST level, or high SP level.	For manual initiation.	Suppression pool, lube oil.
LPCI	Class IE (EDG No.1 or No.2).	For initiation and breaker control.	ESW for room cooler, bearing cooler and motor cooling.	---	Suppression pool.	To control flow rate to RPV.	---
LPCS	Class IE (EDG No.1)	Same as above.	Same as above.	---	Suppression pool.	Manual actuation.	---
RHR	Class IE (EDG No.1 or No.2).	For breaker control.	ESW for heat exchanger, room cooler, bearing cooler, and motor cooling.	---	Suppression pool or RPV.	To control flow rates and place system into operation.	---
SLC	Class IE (EDG No.1 or No.2).	For initiation	---	---	Single poison water tank.	For manual initiation.	---
Containment Spray	Class IE (EDG No.1 or No. 2).	For initiation and breaker control.	ESW for heat exchanger, room cooler, ing cooler, and motor cooling.	---	Suppression pool.	For manual initiation.	---
EDG	---	For initiation, breaker control, and field flash.	ESW	EDG air compression and reservoirs.	---	---	---

Table 3.2 Human Errors Modeled in Event Trees

Symbol	Description of Required Action	Time Assumable for Action
L_H	Limit high reactor vessel level during ATWS.	Minutes
V_C	Failure of this system includes the failure of the operator to initiate makeup water to the condenser hotwell.	Hours
C_I	This value includes the probability of operator failure to manually scram the reactor given an IORV.	15 minutes
P_A	This value includes the probability of operator failure to inhibit ADS or if ADS is initiated to prevent reactor vessel overfill during an ATWS.	Minutes*
W_X	Failure to actuate an adequate external water supply given a small LOCA outside containment.	Many hours
W,Q	Recovery of FW and PCS (included in the heading "W" and "Q" in the event trees)	Minutes for FW, and hours for PCS.

*The time available for operator action is sequence dependent.

Table 3.3 Human Errors Modeled in System Fault Trees

Description of Required Action	Time Assumable for Action
HPCS	
1. Manual actuation of HPCS upon failure of auto-start signal	1/2 hour
2. Open CST supply line valve to HPCS pump given that it was shut	1/2 hour
3. Transfer HPCS suction from CST to SP upon failure of auto-transfer action	1/2 hour
4. Open HPCS discharge valve given that it failed to automatically open	1/2 hour
5. Failure to shut CST test line valve given that it was open and failed to automatically shut	1/2 hour
6. Failure to shut valve F023 given that it failed open and diverts HPCS flow	1/2 hour
7. Manually start HPCS pump given that it failed to automatically start	1/2 hour
RCIC	
1. Manually initiate RCIC given that it has failed to automatically initiate	1/2 hour
2. Manually open valve F010 given that it was shut and failed to open automatically	1/2 hour
3. Manually open F031 given that it failed to open automatically	1/2 hour
4. Manually open valve F13 given that it failed to open automatically	1/2 hour
5. Manually open MOV022 and 059 given that they failed to open automatically	1/2 hour
6. Manually close MOV019 to stop diversion of RCIC flow to the suppression pool given that it failed to automatically shut	1/2 hour
ADS	
1. Manually depressurize plant given that automatic depressurization has failed	Minutes
LPCS	
1. Failure to manually start the LPCS pump given that it failed to start automatically	1/2 hour
2. Failure to manually open LPCS pump discharge valve given that it failed to open automatically	1/2 hour

Table 3.3 (Continued)

Description of Required Action	Time Assumable for Action
LPCI	
1. Failure to follow procedures resulting in failure to line up an alternate LPCI pump suction path given that the normal path is unavailable	1/2 hour
2. Manually open alternate path crossover valves given that an alternate suction path to the LPCI pump is required	1/2 hour
3. Manually initiate the LPCI pumps given that they failed to auto-start	1/2 hour
4. Manually open LPCI pump discharge valves given that they failed to auto-open	1/2 hour
5. Manually initiate LPCI given that it failed to auto-initiate	1/2 hour
RHR (Containment Spray Mode)	
1. Manually align RHR system for containment spray mode given that auto-alignment of containment spray has failed	hours
2. Manually initiate containment spray given that it failed to automatically initiate	hours
RHR (Suppression Pool Cooling Mode)	
1. Start suppression pool cooling when required	hours*
2. Correct valve misalignments during line up of system	hours*
3. Manually open pump suction valves in alternate suction lines given that normal suction line valves have failed	hours*
4. Initiate RHR pump room cooling	hours*
Suppression Pool Makeup	
1. Manually initiate the system given that it failed to auto-initiate	1/2 hour
SLC	
1. Manually initiate SLC given that it failed to auto-start	< 5 minutes
2. Recognize leak in SLC poison water storage tank in daily inspection	---

*The time available for operator action is sequence dependent.

4. ACCIDENT SEQUENCE QUANTIFICATION

This Section discusses the quantification of the accident sequences in the GESSAR-II PRA. Section 4.1 presents an overview of the accident sequence quantification approach used in the GESSAR-PRA, and Section 4.2 a discussion of the quantification of accident initiators. Section 4.3 presents the results of system fault tree evaluations. Section 4.4 discusses the BNL revisions to the various functional event trees. Section 4.5 gives the results of the quantification of these revised trees and the dominant accident sequences. Sections 4.6 and 4.7 provide discussions on the uncertainty analysis and the importance analysis, respectively, that were done by BNL on the basis of the revised functional event trees.

4.1 Overview

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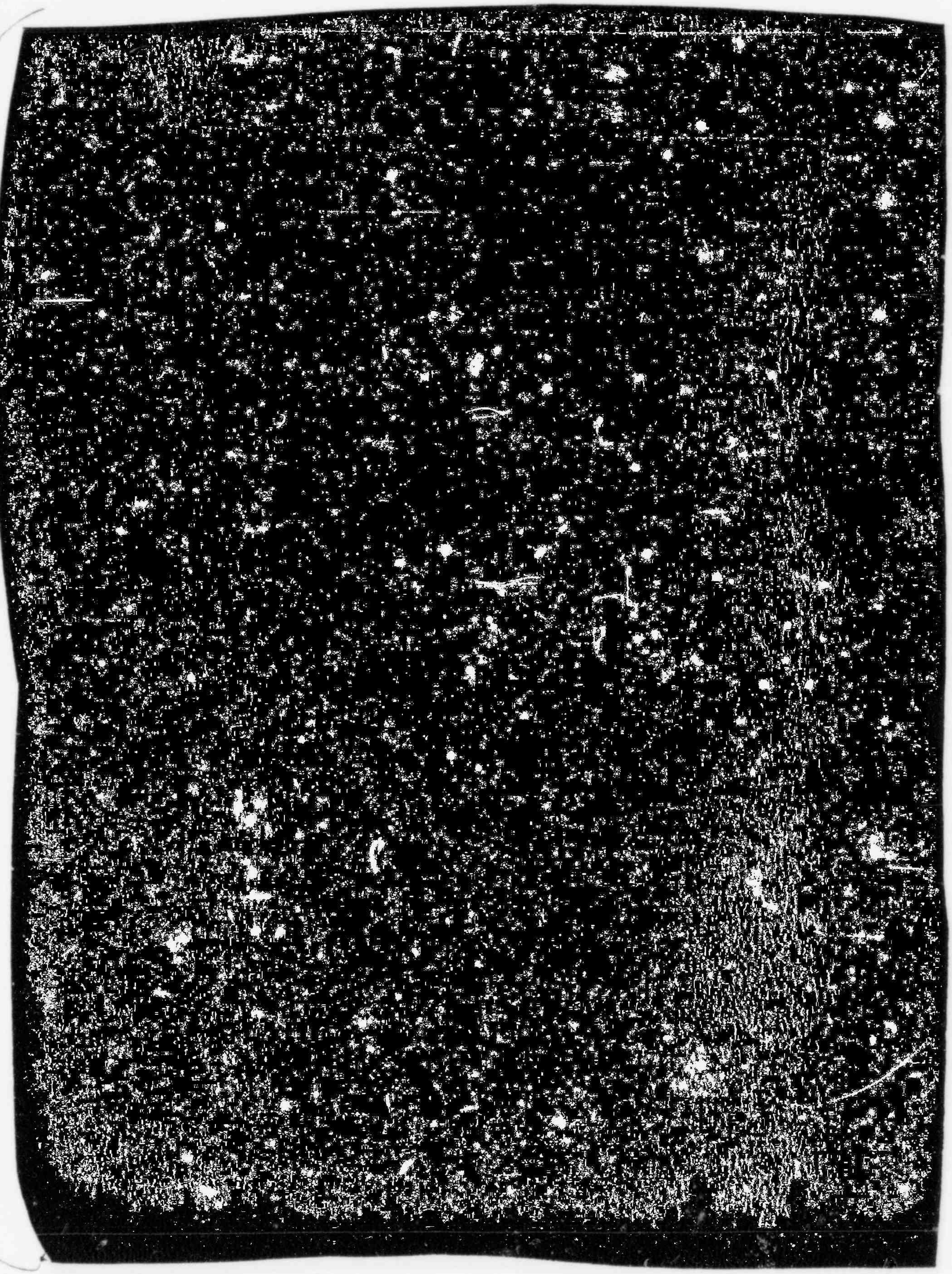

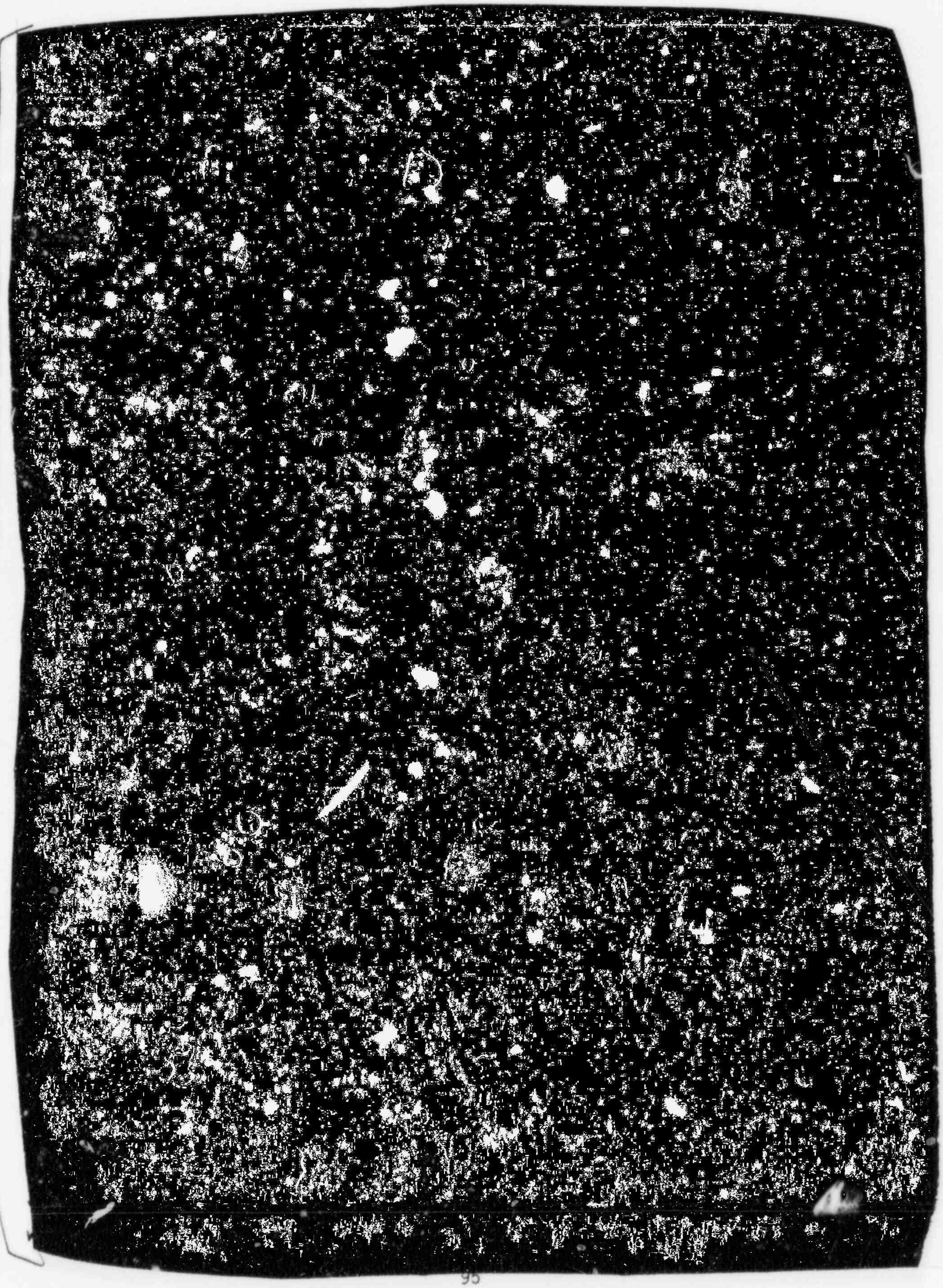

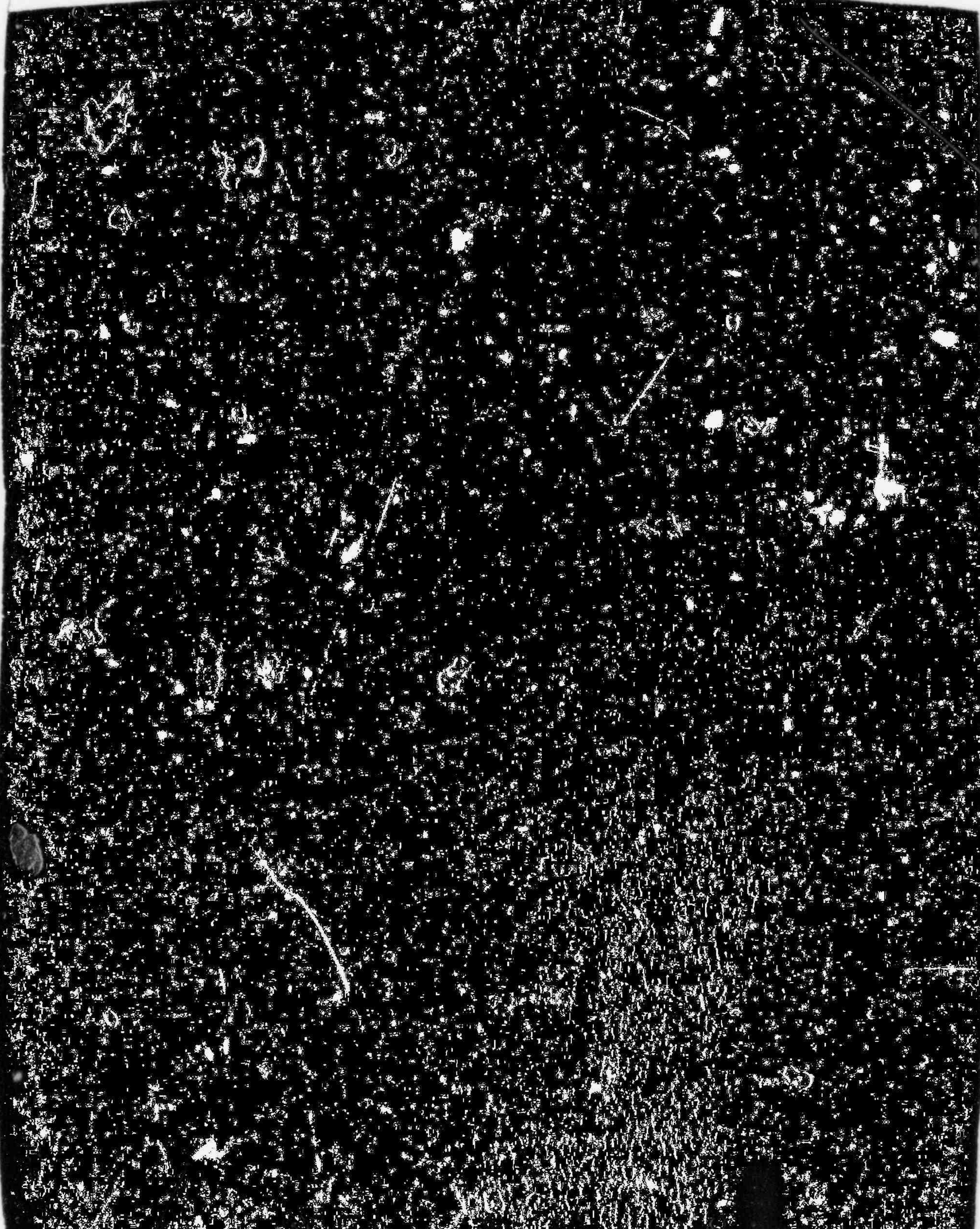

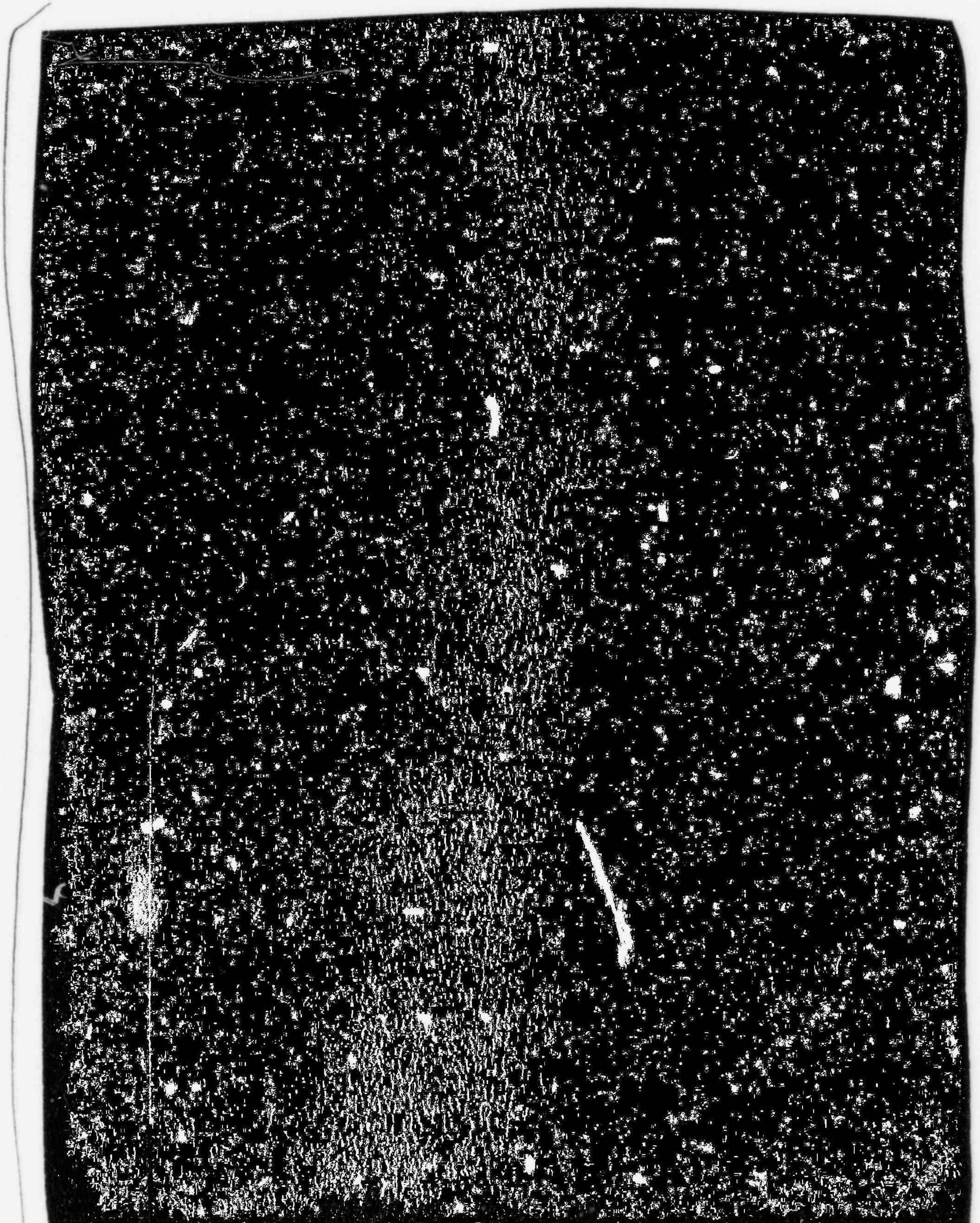

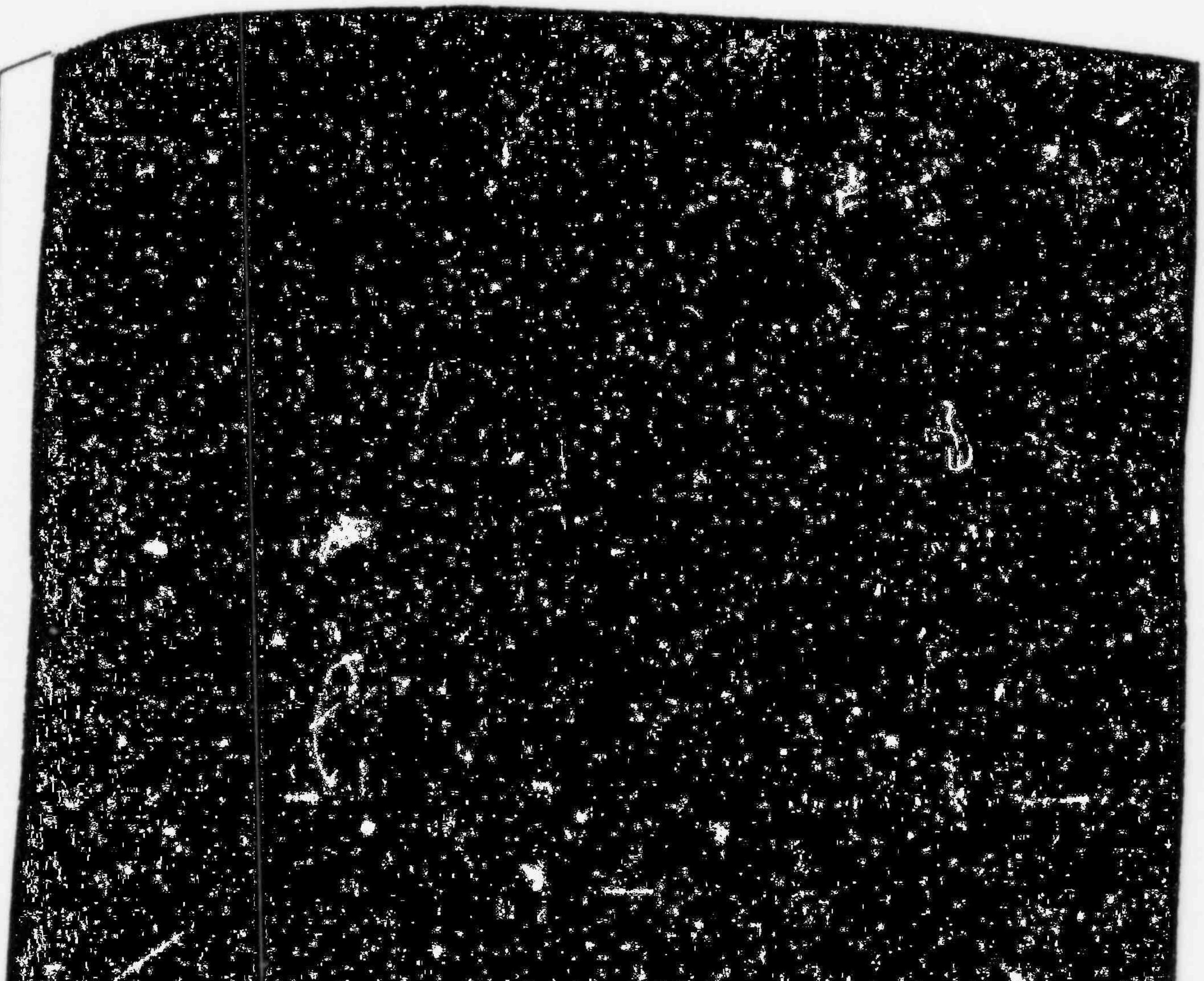

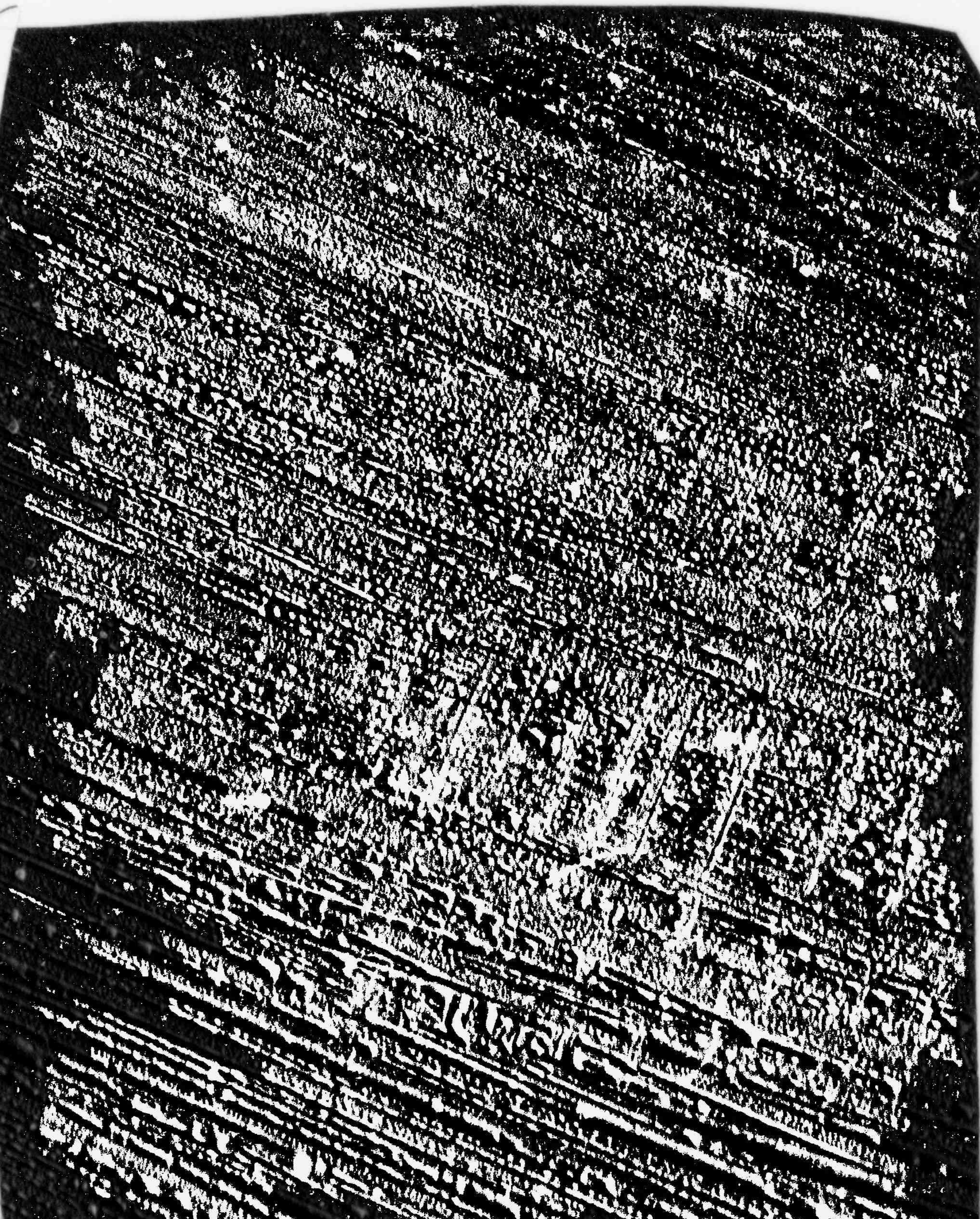

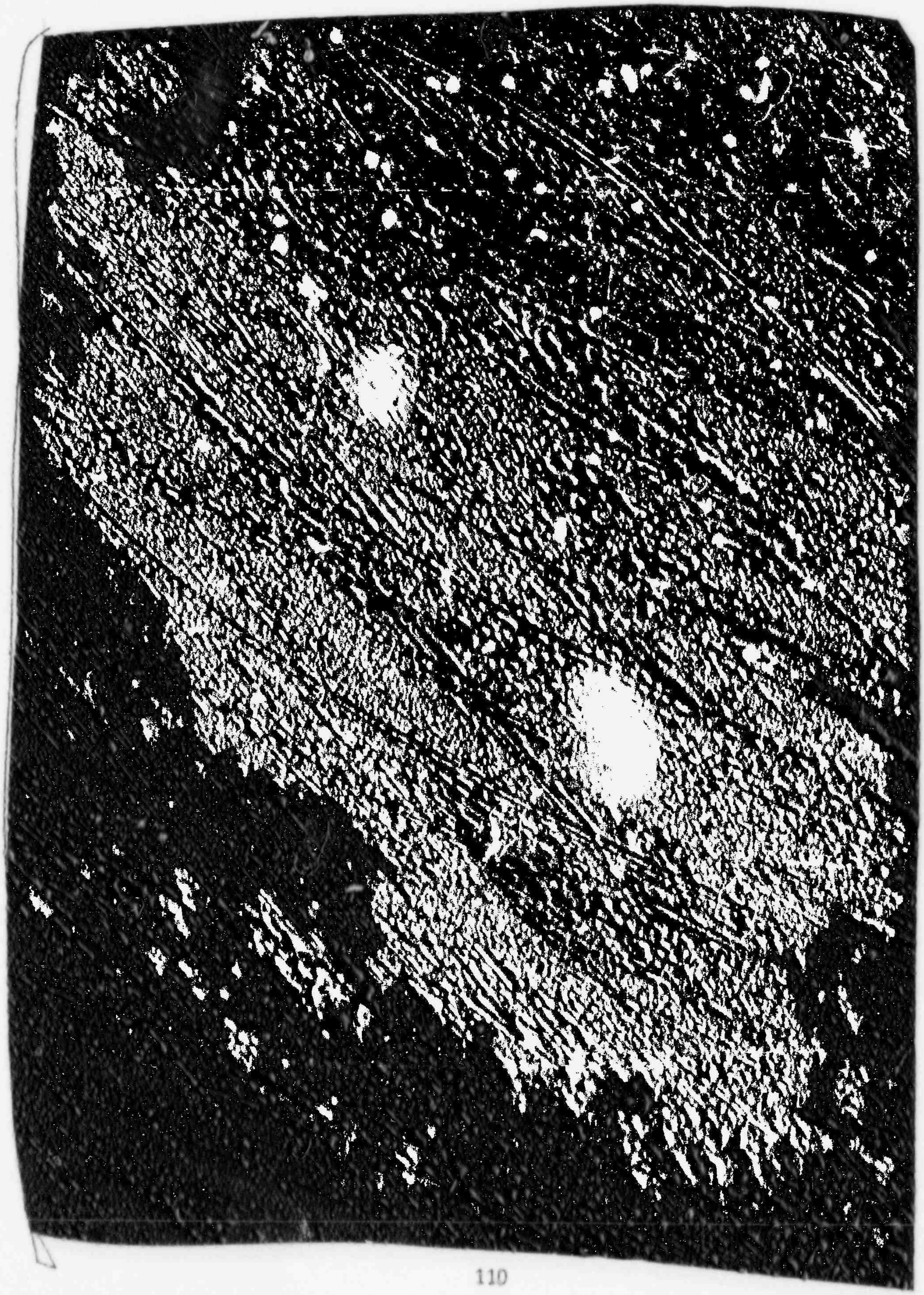

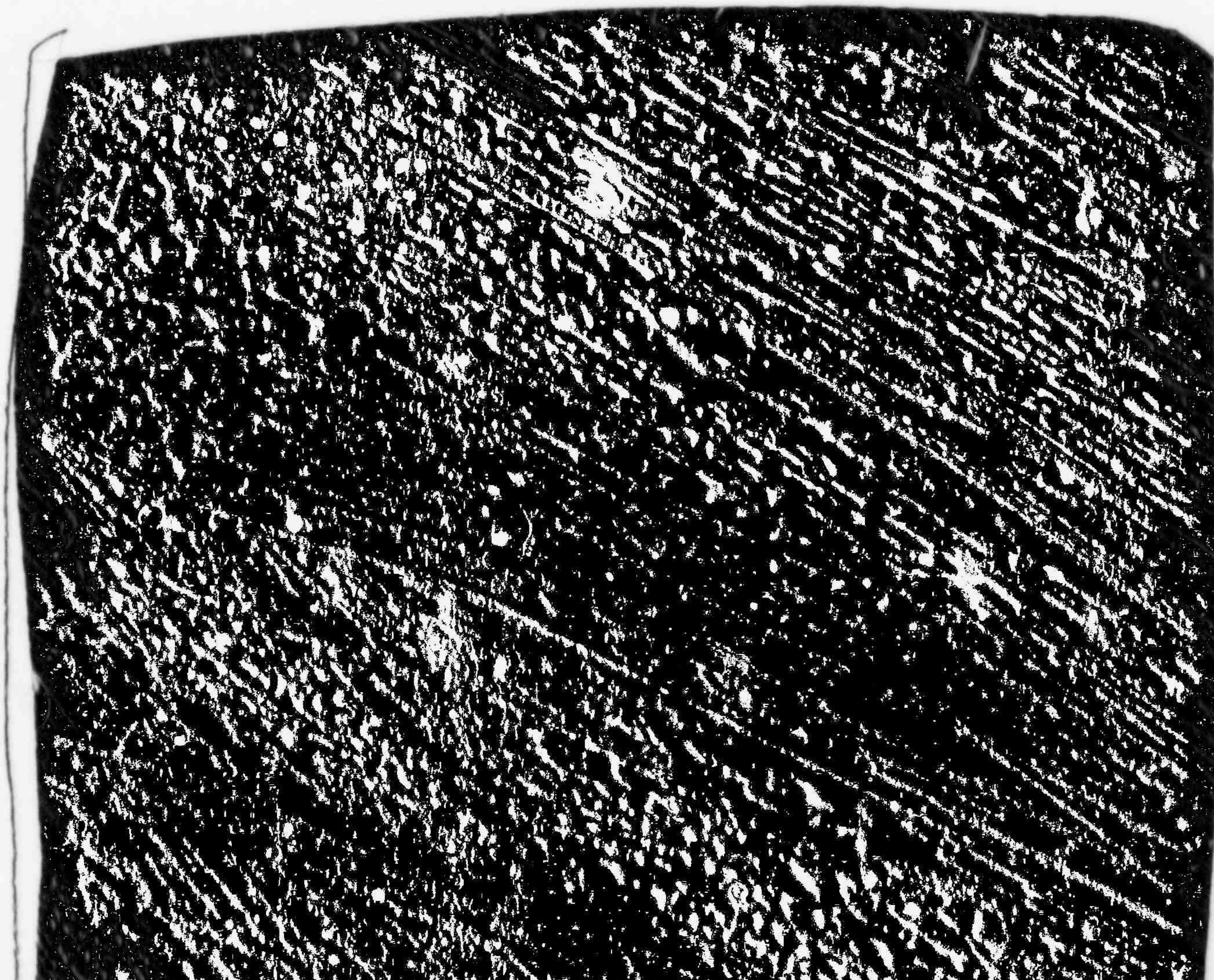

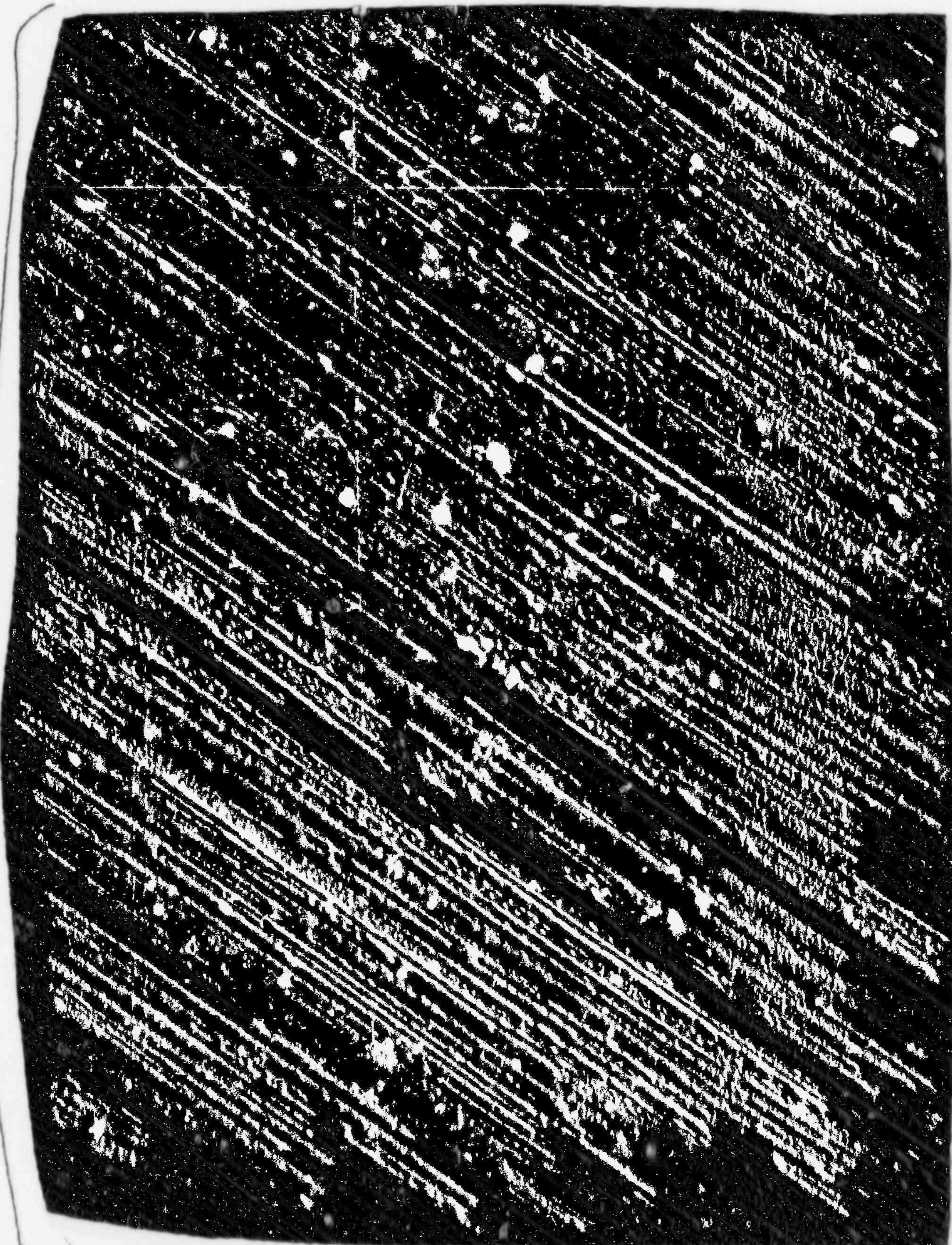

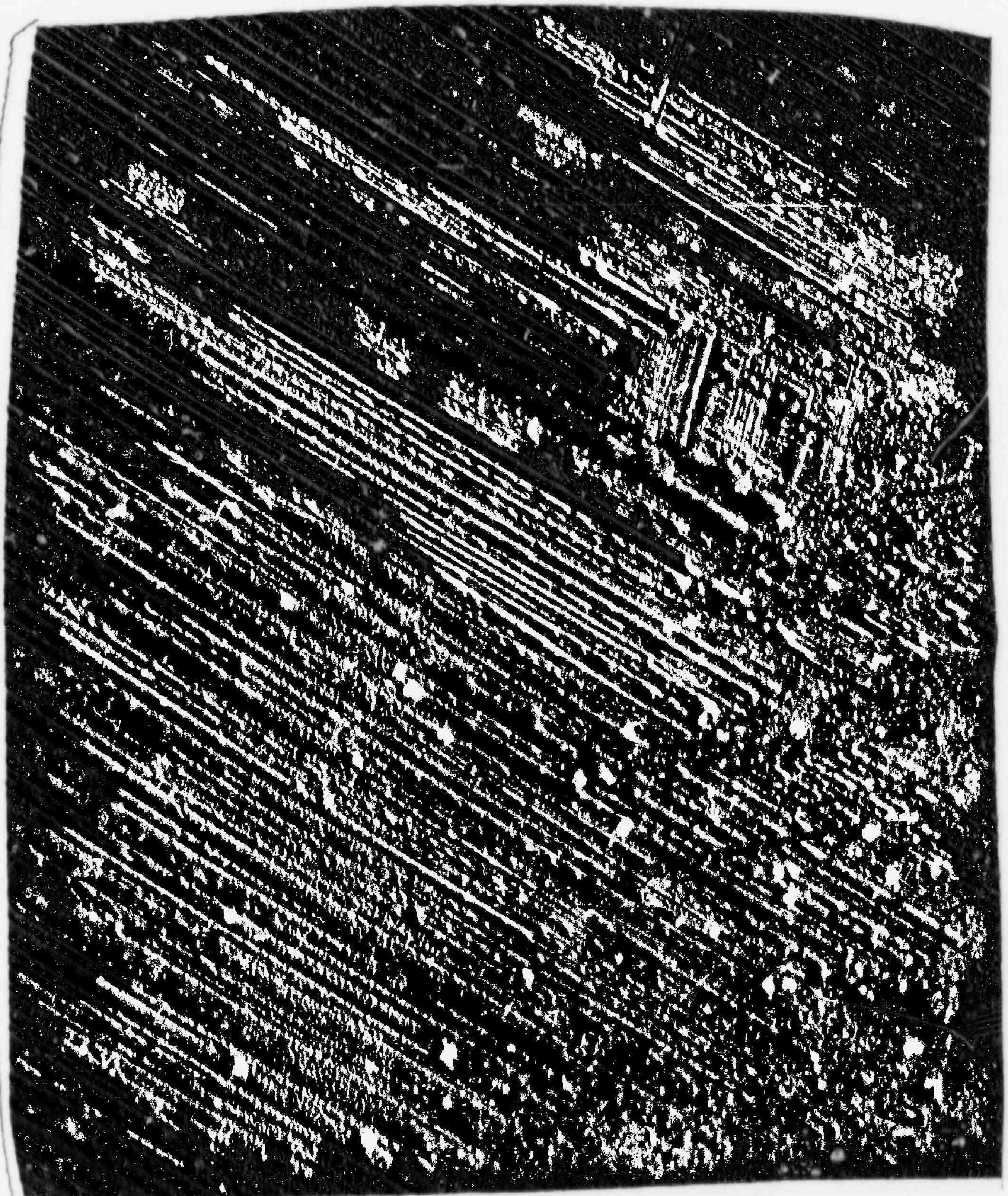

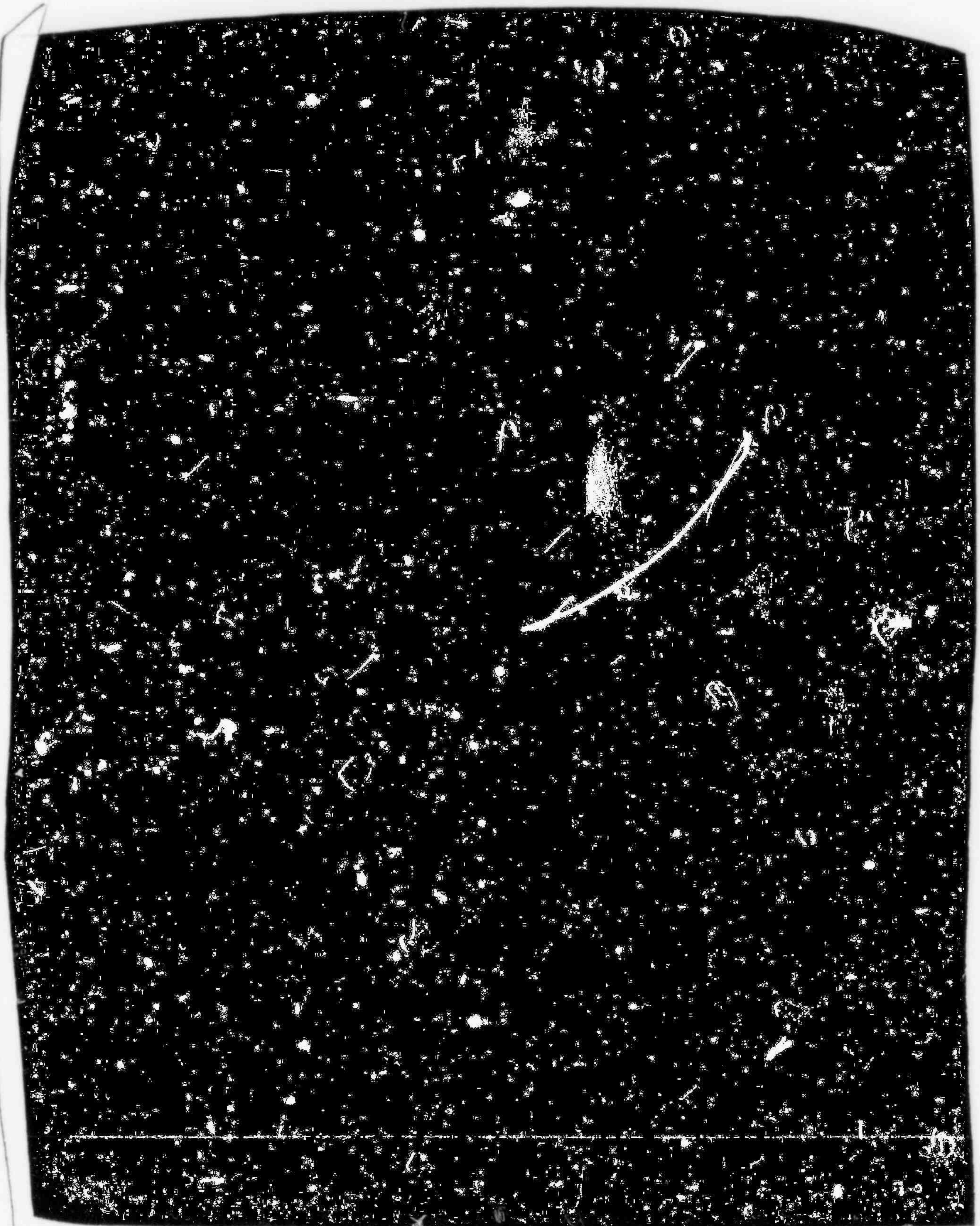

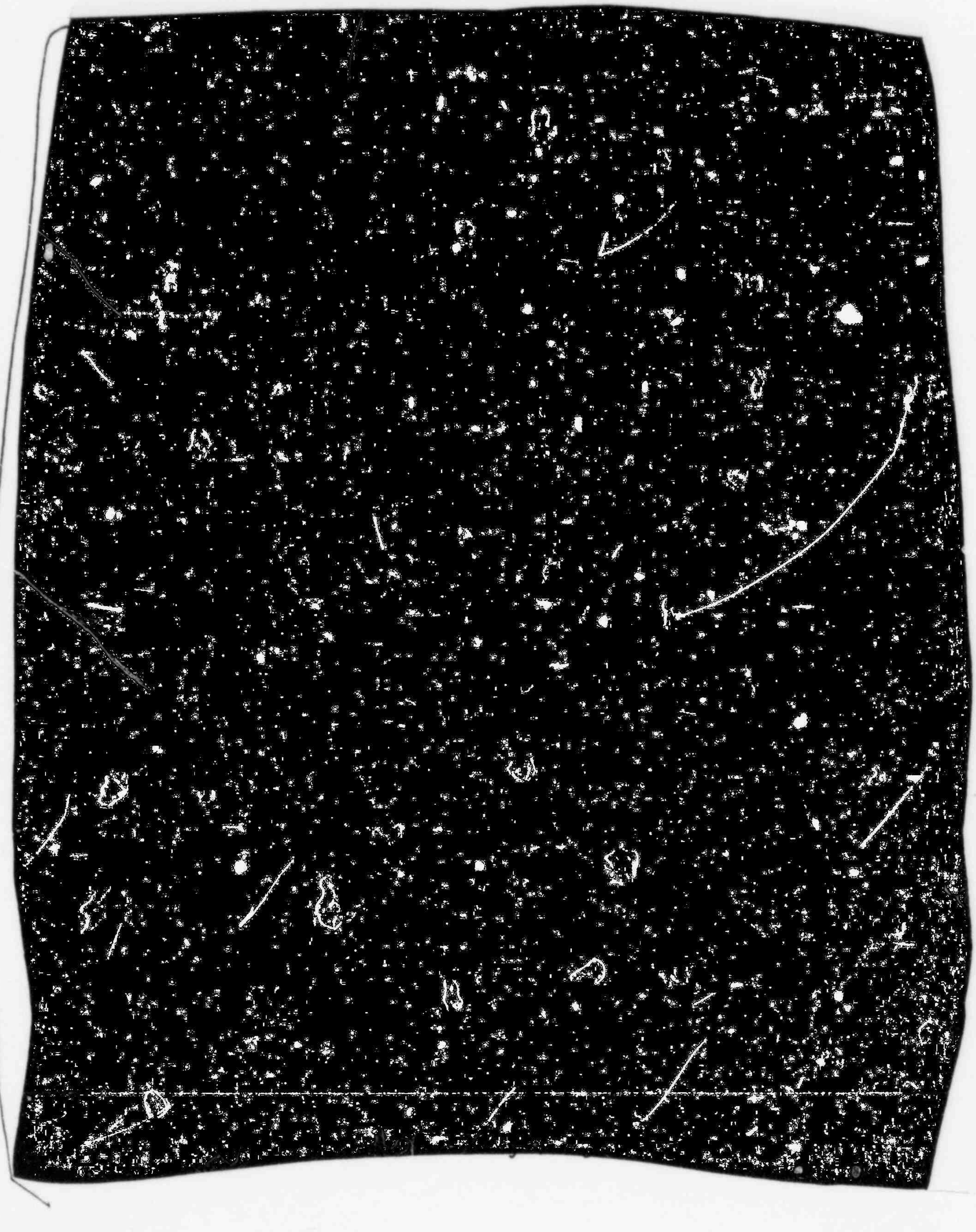

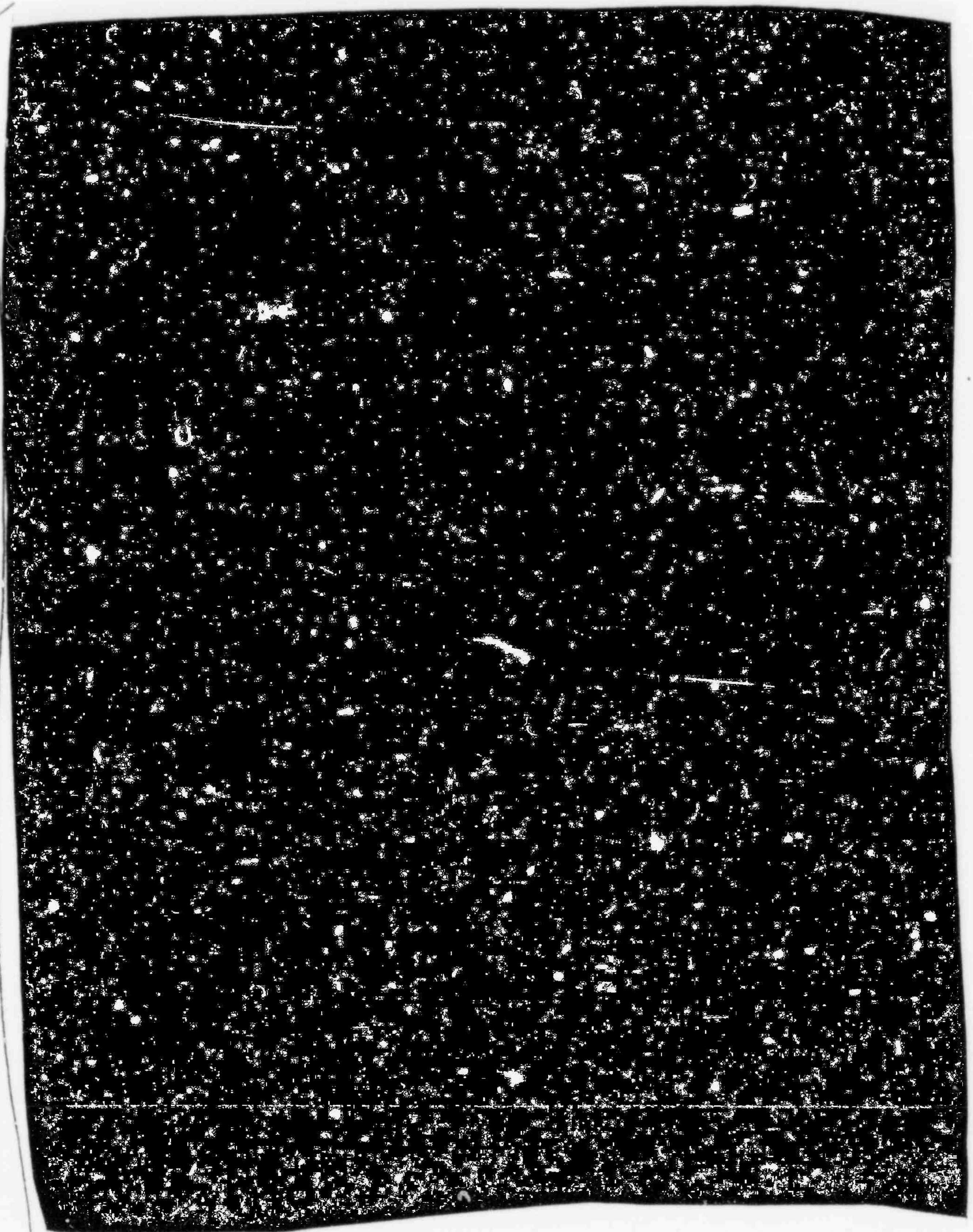

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4.6 Uncertainty Assessment

This section presents an assessment of the uncertainties about the frequency of core damage. These uncertainties should be interpreted as being introduced by uncertainties in the values of the various input parameters, given the modeling assumptions of the GESSAR-II PRA as revised by BNL and described in the previous sections. They are not meant to include uncertainties introduced by failure modes and other modeling assumptions considered outside the scope of the GESSAR-II PRA.

An uncertainty analysis was submitted by GE at the time this report was being written. A limited review of this analysis was performed, and the following comments can be made:

1. The methodology used by GE is similar to that used in this review.
2. The results are presented in terms of release categories while in the BNL review the results are presented in terms of core damage classes.
3. The analysis for release categories 1, 2, and 3 (in the GE analysis) does not include several sequences. The reason for this omission is the probability cut off used in obtaining the cutsets for reactor water injection (Table 3.1 and 3.2 of the GE uncertainty analysis.)

4. The equations used for the uncertainty analysis for release categories 1, 2, and 3 (in the GE analysis) do not correspond to the cutsets for reactor water injection (Tables 3.1 and 3.2 of the GE analysis). These equations include the failure to recover offsite power in 2 hours (given failure to recover offsite power in 1/2 hour), which is not correct, because the common mode failure of the 3 diesel generators and failure to recover the DGs in 1/2 hour, together with failure to recover offsite power in 1/2 hour and failure of the RCIC system, leads to core damage whether the offsite power is recovered in 2 hours or not.

Because of limitations in the level of effort of the BNL review of the GESSAR-II PRA, a rigorous propagation of the uncertainties was not performed. Instead, a rather conservative approach was used for the assessment of the uncertainties in the frequency of core damage. The approach consisted of the following general steps:

1. The uncertainties in the initiating events and in the frontline and support systems were quantified by fitting lognormal distributions to evaluate uncertainty measures (mean and variance).
2. The uncertainties in each accident class were quantified, using the most important accident sequences in each class (contributing 95% of the class frequency) and the distributions assessed in step 1.
3. The uncertainties in the frequency of the total core damage were evaluated, using the most important accident sequences (contributing 95% of the core damage frequecy) and the distributions assessed in step 1.

The main approximation of this approach lies in the evaluation of the uncertainties for the various systems. The procedure is further described in the remainder of this section.

Initiating Events: Frequency distributions were derived for each of the 37 transient accident initiators using the methodology described in Reference 4. Then, the several initiators that form the turbine trip and the isolation groups were added by simply adding the means and the variances. Finally, the results for each of the four transient groups and manual shutdown were fitted to lognormal distributions. Table 4.38 gives the mean values and probability intervals for the five initiators. For LOCA initiators, the mean values given in the GESSAR-II PRA were used as the mean of lognormal distributions, and an error factor of 5 was assumed.

Frontline Systems: The analysis of the minimum cutsets for the frontline systems indicates that hardware failures are responsible for most of the unavailability of those systems. Therefore, lognormal distributions with the mean equal to the calculated unavailability and an error factor of 3 were assumed for all automatically initiated frontline systems. For the LPCCS, an error factor of 5.0 was used because human error (miscalibration) is the dominant cutset. For the ADS, the operator failure to depressurize the reactor given a failure of auto-initiation was treated with an error factor equal to 10.0.

Quantification of Uncertainties: For the quantification of uncertainties, all dominant accident sequences that contribute up to 95% of the core damage frequency in each class were determined. The accident sequences for each class were then input to the SAMPLE[2] code, which provided the uncertainties for each class frequency; details of the SAMPLE input are described in Appendix D. No uncertainty was assumed in the offsite power and diesel recovery probabilities used in the event trees. Table 4.39 shows means, medians, and 5% and 95% probability intervals for the frequencies of all classes. Note that in Table 4.39, the distribution for each class uses the "National Average" initiation frequency for LOOP.

The total core damage frequency was obtained by combining the uncertainties in the accident sequences that contribute 95% to total the core damage frequency using the SAMPLE[2] code. The results for the final distribution using the "National Average" and the Mid-Atlantic Area Council are also presented in Table 4.39.

4.7 Importance Analysis

In the GESSAR-II PRA, an importance analysis (sometimes referred to as the sensitivity analysis) was not included as part of the risk evaluation study. Therefore, this presentation, which includes a description of the methodology (Section 4.7.1) and results (Section 4.72.) of an importance analysis, reflects the work of a BNL evaluation rather than a review or a reassessment.

4.7.1 Methodology

Various types of importance measures have been defined to serve as indicators of relative importance with respect to different risk indices, for instance core damage frequencies. This study will utilize two of these measures; namely, the Birnbaum measure and the Fussel-Vesely measure. The Birnbaum measure of importance is defined as

$$I_B = \frac{\partial f}{\partial \alpha}, \tag{1}$$

where α is the unavailability of the system (or component) of interest and f is the total core damage frequency, i.e., the summation of all the core damage sequences. This measure expresses the slope of change in core damage frequency due to the change in the unavailability of the system. The Fussel-Vesely measure is defined as

$$I_{FV} = \frac{\alpha}{f} \frac{\partial f}{\partial \alpha} . \tag{2}$$

This measure is equivalent to the logarithmic derivative of both f and α. It is also the Birnbaum measure weighted by the ratio of the unavailability of the system to all the core damage sequences. An alternative interpretation of this measure is that it gives the percentage of change in the risk indices per unit change in the system unavailability.

In order to evaluate these two importance measures for the various systems with respect to core damage frequency, BNL examined the cutsets as generated from the core damage fault trees and identified all the cutset elements which contributed to dependence between systems. The cutsets were then reduced and grouped according to systems. Dominant accident sequences contributing up to 95% of the core damage frequency) were then defined in terms of these systems. Those cutset elements which are major contributors to syste dependence were retained in the sequence definition and were not reduced. The computer code Set Evaluation Program (SEP)[17] was used to calculate the partial derivatives.

An alternative method of depicting the importance of a system is by examining its risk achievement worth or the risk reduction worth as proposed by Vesely.[18] The approach considers two different states: 1) when the system is perfectly reliable $\alpha(0)$, and 2) when the system is in a failed state, $\alpha(1)$. The risk reduction worth of the system α is defined as

$$R_R = \frac{f}{f(\alpha(0))} \tag{3}$$

where $f(\alpha(0))$ is the sum of core damage sequences with α being perfectly reliable. Similarly, the risk achievement worth of system α is defined as

$$R_A = \frac{f(\alpha(1))}{f} \tag{4}$$

where $f(\alpha(1))$ denotes the core damage sequences with α in a failed state.

The inverse of R_R can be interpreted as a reduction factor which reduces the core damage frequency if the system α is rendered perfect. Conversely, R_A is a factor which increases the core damage frequency if α is in a failed state.

Both R_A and R_R can be derived by the method described in Reference 18. However, they can also be expressed in terms of the Birnbaum and the Fussel-Vesely importance measures. Let the core damage sequences be written as

$$f = \sum_{ijk} a_i\alpha + b_j\overline{\alpha} + c_k \tag{5}$$

where $\overline{\alpha}$ is the NOT event of system α, i.e., its success, the coefficients a_i, b_j denote all other cutset elements; and c_k represents cutsets which do not contain either system α or its NOT event. The core damage expression with a perfect system α can then be written as:

$$f(\alpha(0)) = f\,[1 - I_{FV}] \quad (6)$$

Furthermore,

$$f(\alpha(1)) = I_B + f\,[1 - I_{FV}]. \quad (7)$$

It follows that the risk reduction worth becomes

$$R_R = [1 - I_{FV}]^{-1} \quad (8)$$

and the risk achievement worth becomes

$$R_A = \frac{1}{f} I_B + [1 - I_{FV}] \quad (9)$$

Both Eq. (8) and Eq. (9) exhibit one very important feature: given the partial derivative of the core damage sequence function f and the unavailability value of α, both the risk reduction and achievement worth can be readily calculated.

4.7.2 Importance Analysis Results

The two importance measures were calculated for each of the ten accident sequence classes based on the "National Average" LOOP initiator (Case 4 of Table 4.25), and the results are presented in Tables 4.40 to 4.49. An explanation of the acronyms used in these tables can be found under nomenclature. For instance, in Table 4.40 the first column enumerates the cutset elements of the accident sequences. The second and the third columns present the Birnbaum measure and the Fussel-Vesely measure, respectively. The cutset elements are given in the ascending order of the Fussel-Vesely measure.

The two importance measures of the cutset elements with respect to the total core damage frequency for the "National Average" LOOP initiator were also calculated, and the results are presented in Table 4.50. It is evident that perturbations of the electric power related failures, such as the LOOP initiator or the common mode three-diesel failure, will result in substantial changes to the total core damage frequency. For instance, a 50% reduction in the three-diesel common mode failure probability results in a 20.5% reduction in core damage frequency. The HPCS and the RCIC systems rank third and fifth respectively in the list. A 100% change of the mechanical failure to scram function, C_M, will lead to an 8% change in the core damage frequency. Among those that rank low on the list are the low pressure systems, V4, the

ARI, the electrical failure to scram, C_E, manual shutdown transient initiator, T_M, and the multiple IORV function, I_M.

The core damage achievement worth and reduction worth for various systems are presented in Table 4.50 and Figure 4.32. It can be seen that the common mode failures of two or three diesels have quite marked core damage achievement worth. The same is true for the failure of reactor scram due to either mechanical or electrical failures. The depressurization function X similarly exhibits a high core damage achievement worth. In summary, it appears that there are a number of systems or functions in the GESSAR design which warrant efforts to maintain their reliability.

On the other hand, the more significant core damage reduction worth functions are the failure of three or two diesels due to common mode and the high pressure injection systems. Results indicate that improvements made to these systems would have the most notable impact in the reduction of total core damage frequency.

One of the findings of this analysis appears to point to the overall importance of the various electrical subsystems, be it ac or dc, offsite or onsite power.

4.8 References to Section 4

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14. "Assessment of BWR Mitigation of ATWS," GE Report NEDE-24222, Vols. 1 and 2, 1979.

15. "ATWS in Light Water Reactors," NUREG-0460, April 1978.

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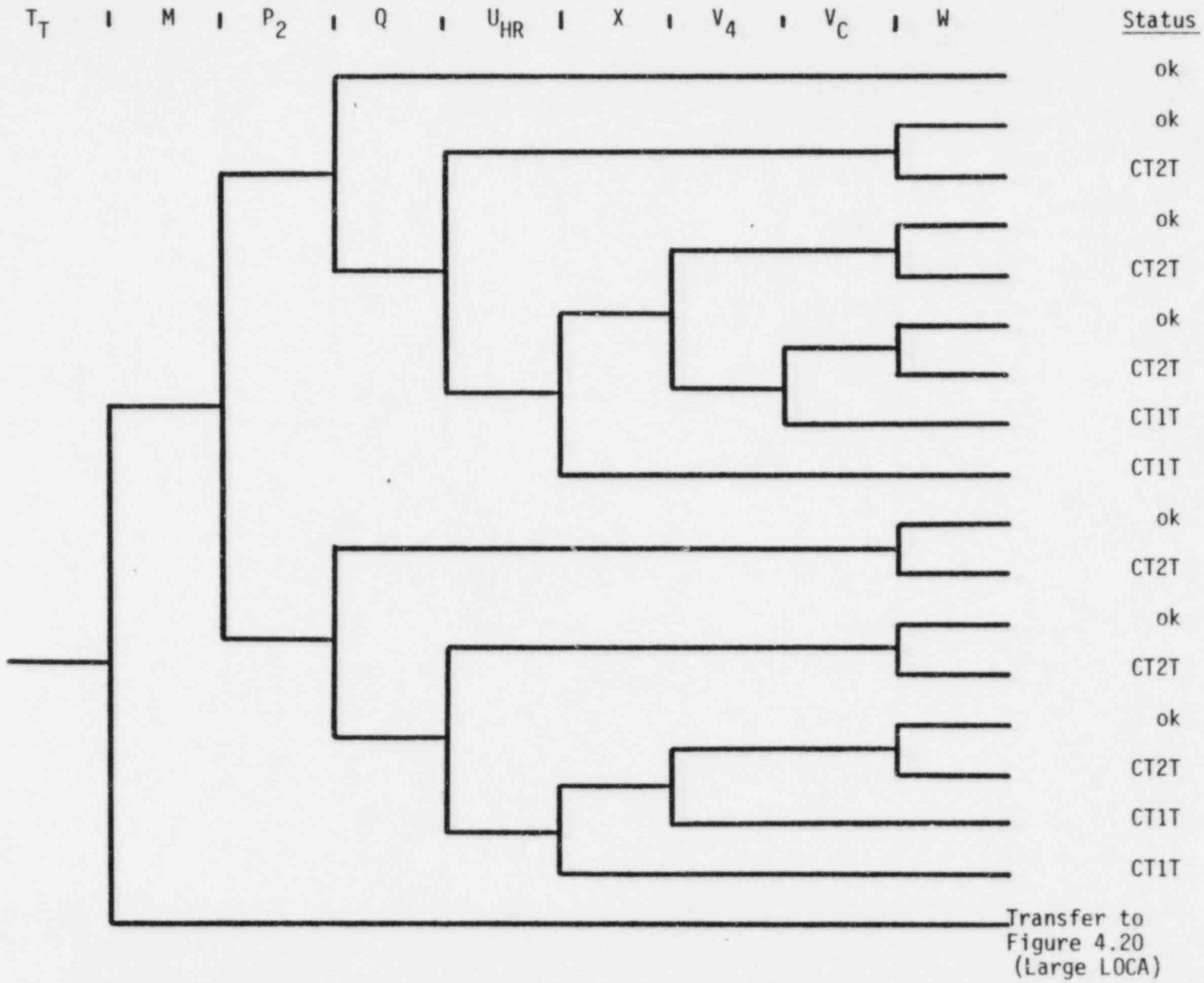


Figure 4.1 BNL revised functional event for turbine trip.

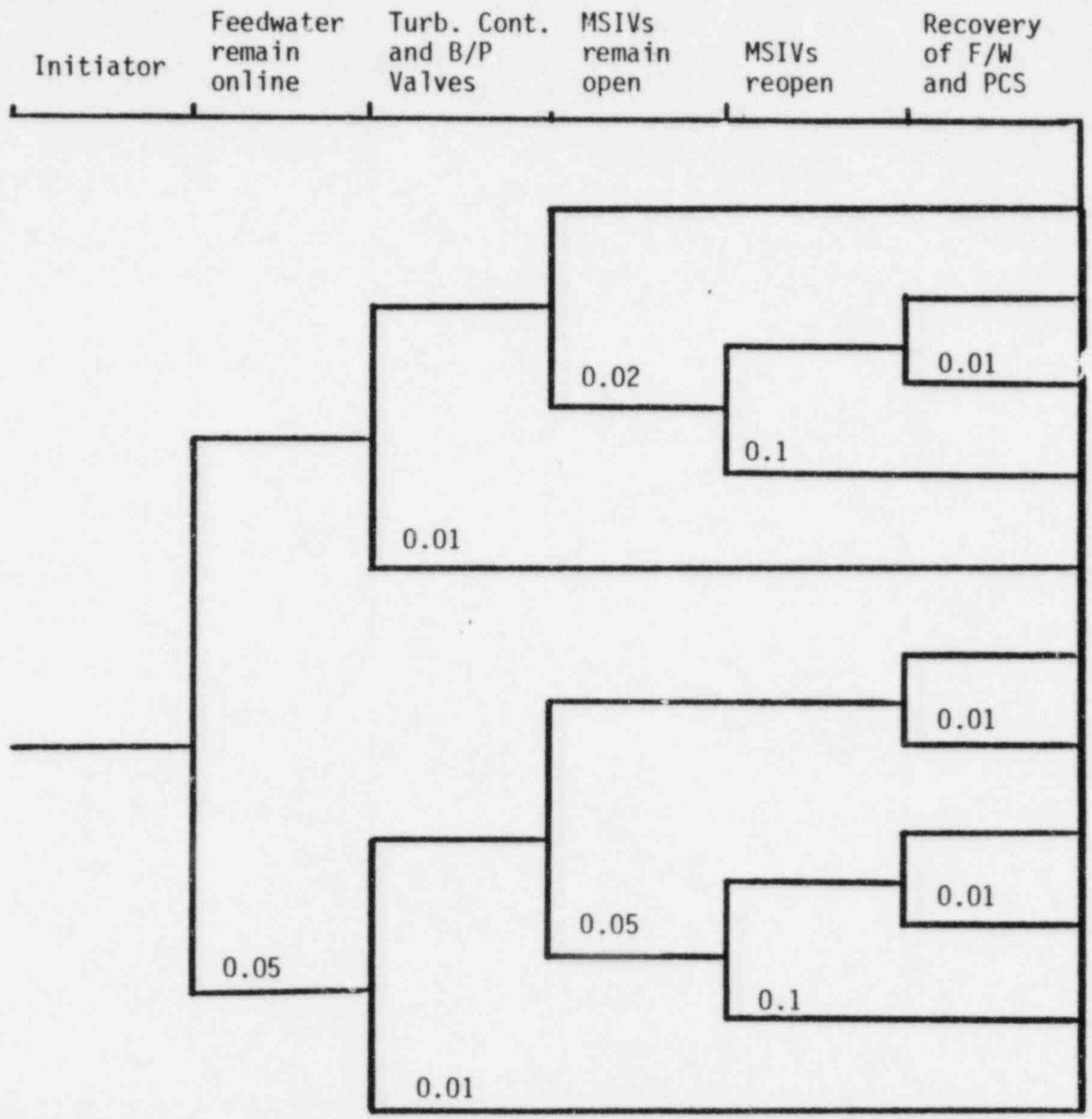


Figure 4.2 Functional level event tree of the feedwater system with no SORV for turbine trip.

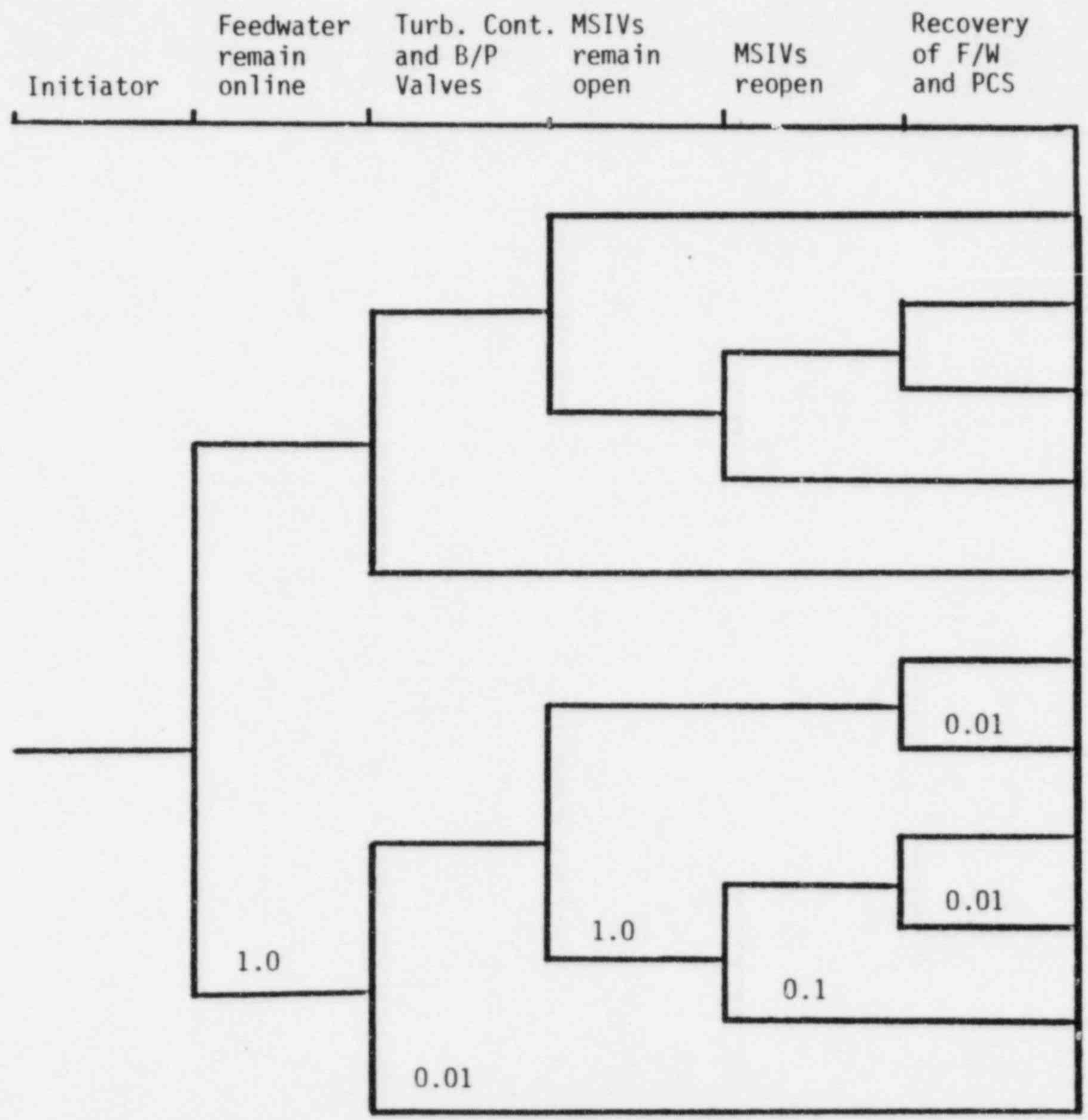


Figure 4.3 Functional level event tree of the feedwater system with 2 or more SORV for turbine trip.

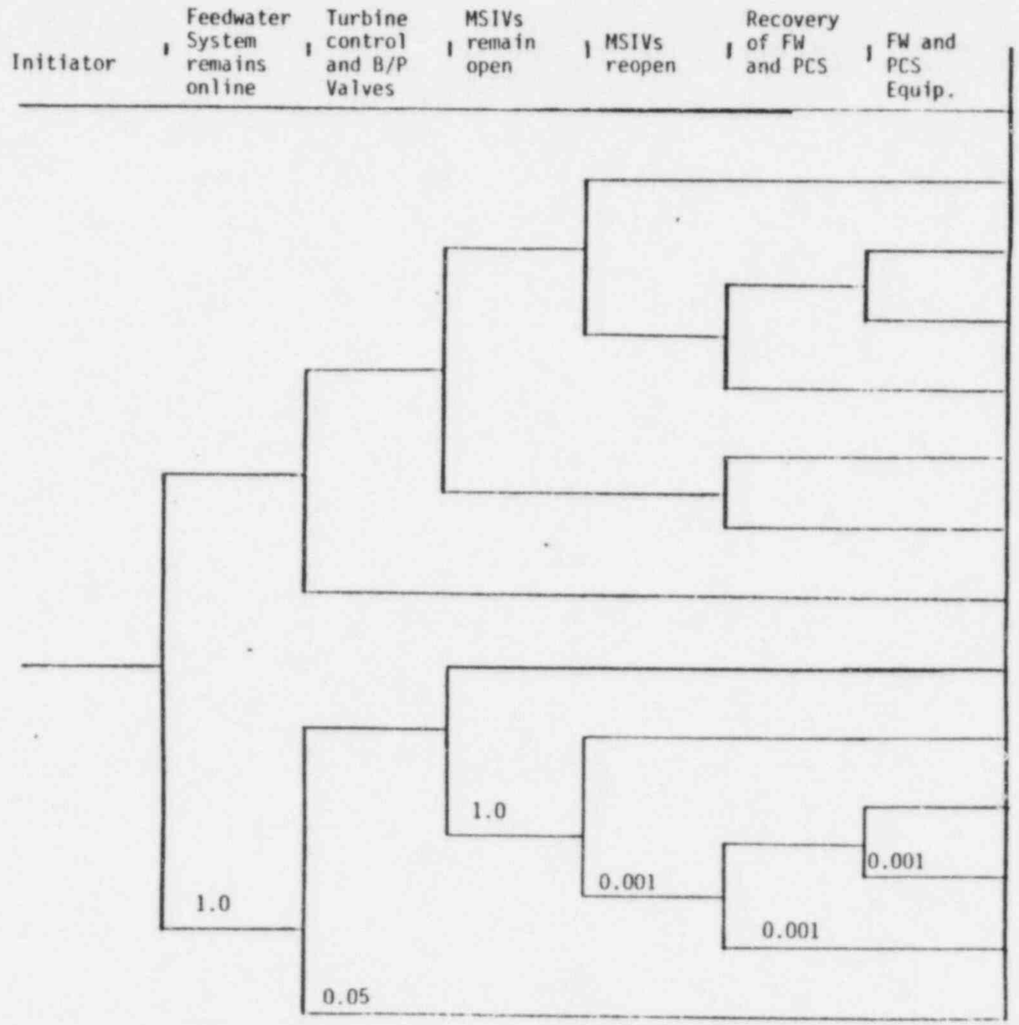


Figure 4.4 Functional level event tree of the long-term PCS with turbine control valves failed and no SORV for turbine trip.

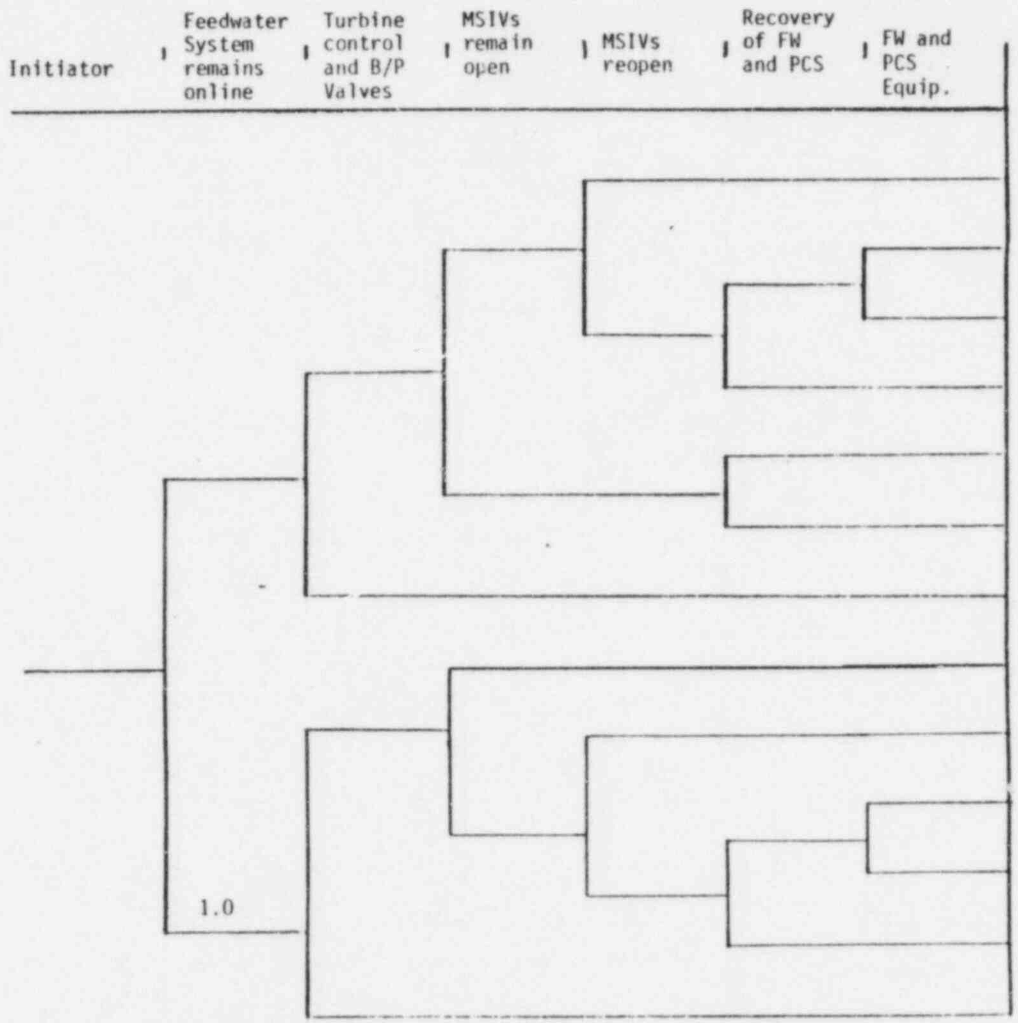


Figure 4.5 Functional level event tree of the long-term PCS with turbine control valve operational and no SORV for turbine trip

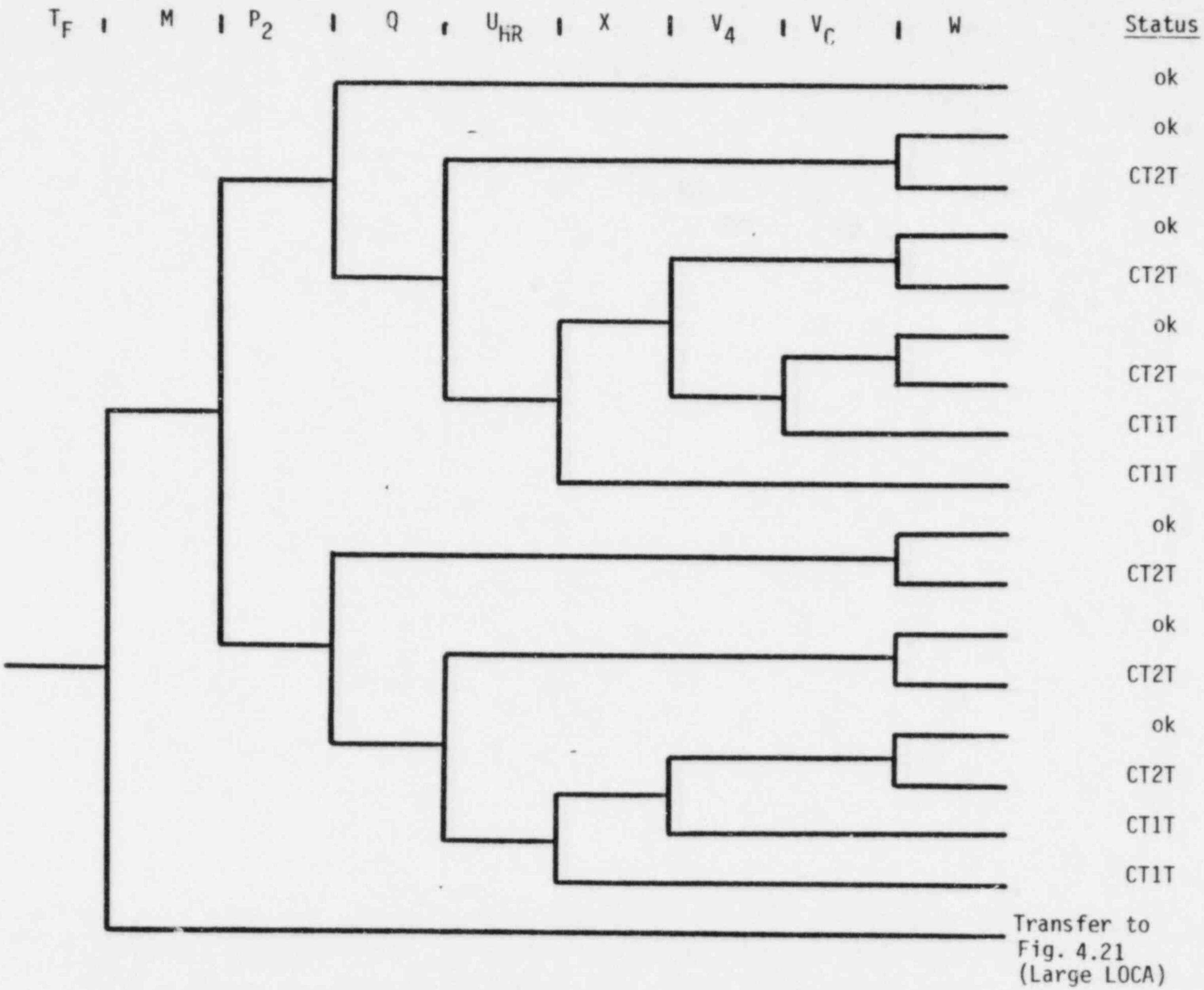


Figure 4.6 BNL revised functional event tree for isolation.

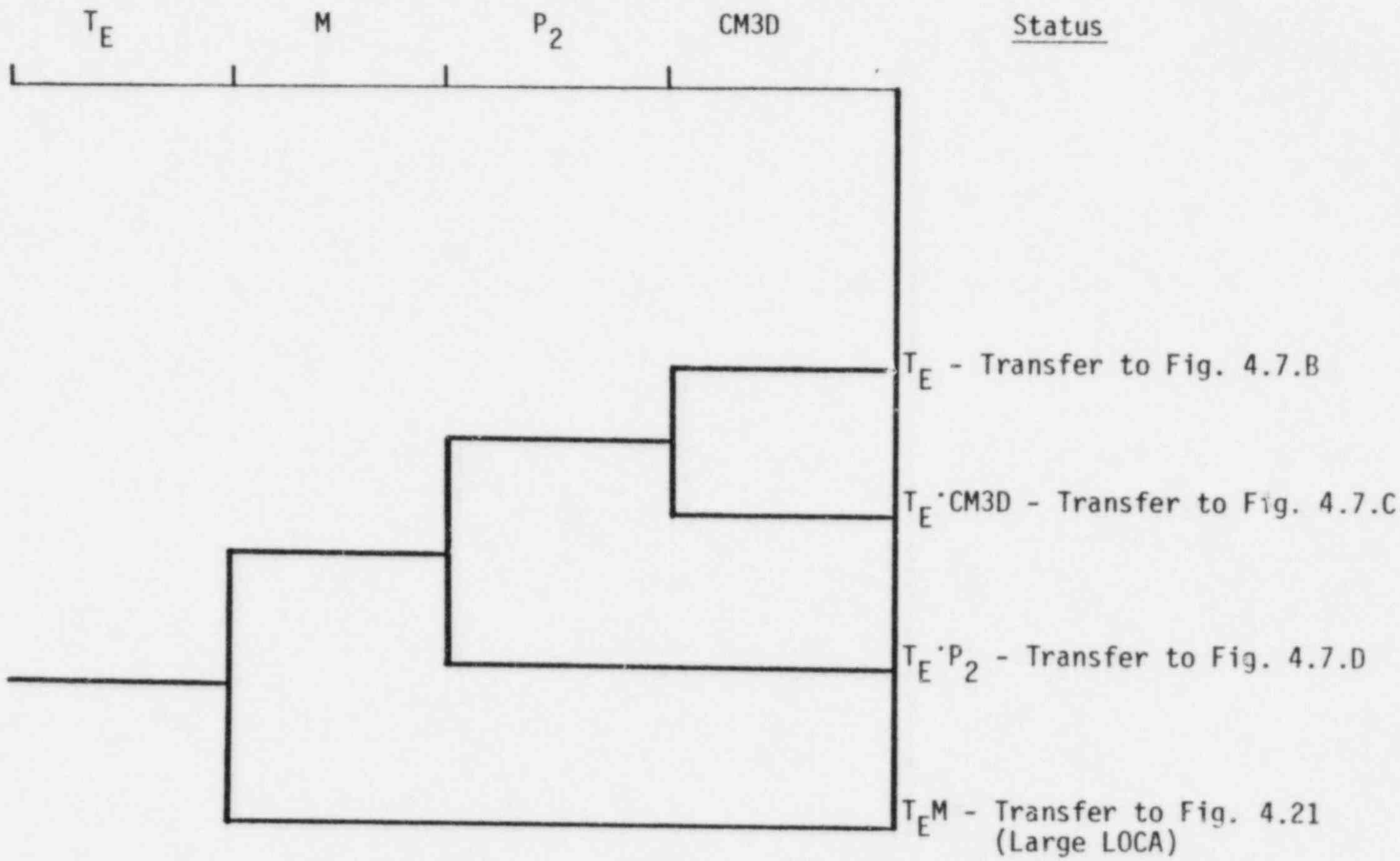


Figure 4.7.A BNL revised functional event tree for LOOP.

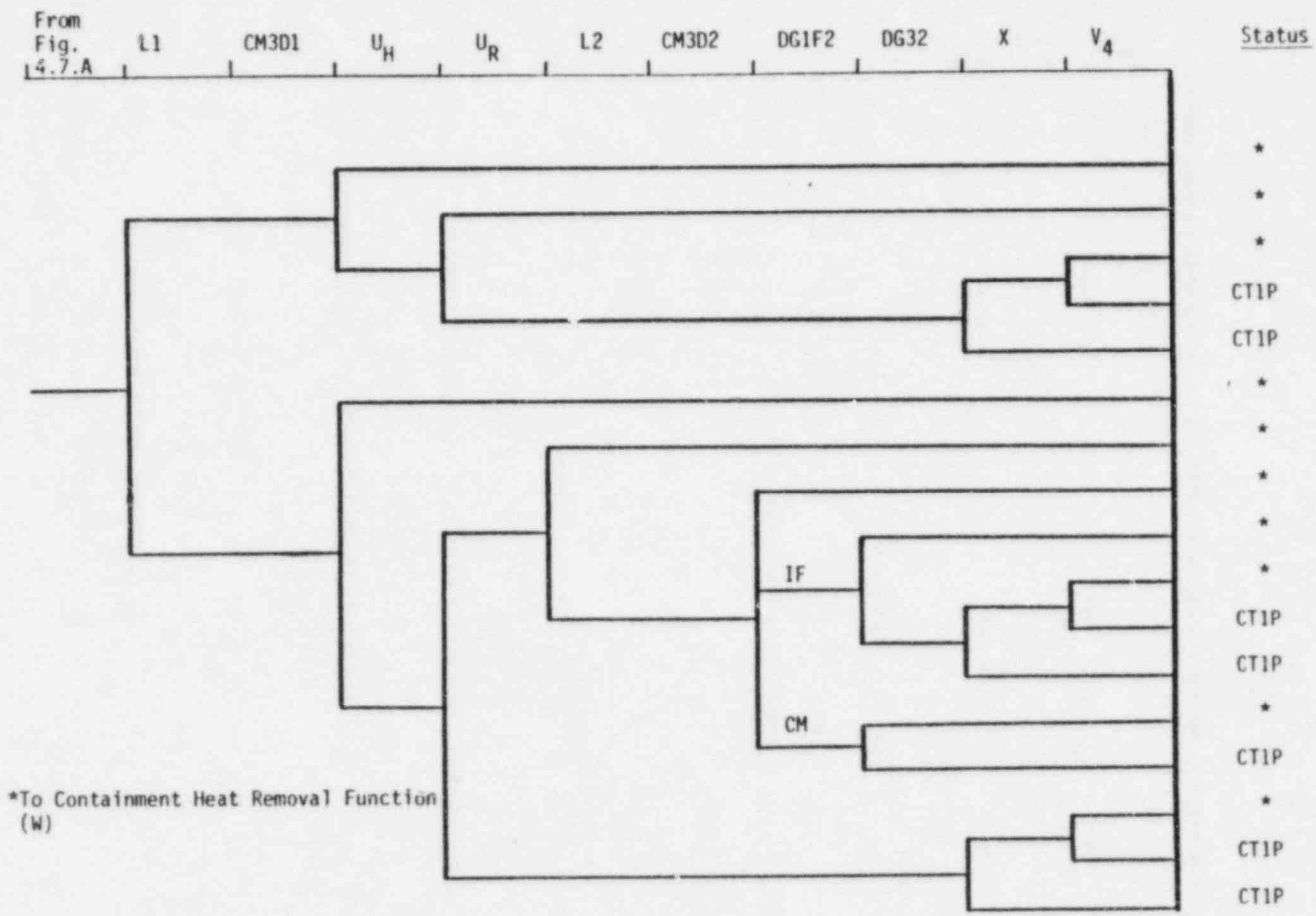


Figure 4.7.B BNL revised functional event tree for LOOP (continued).

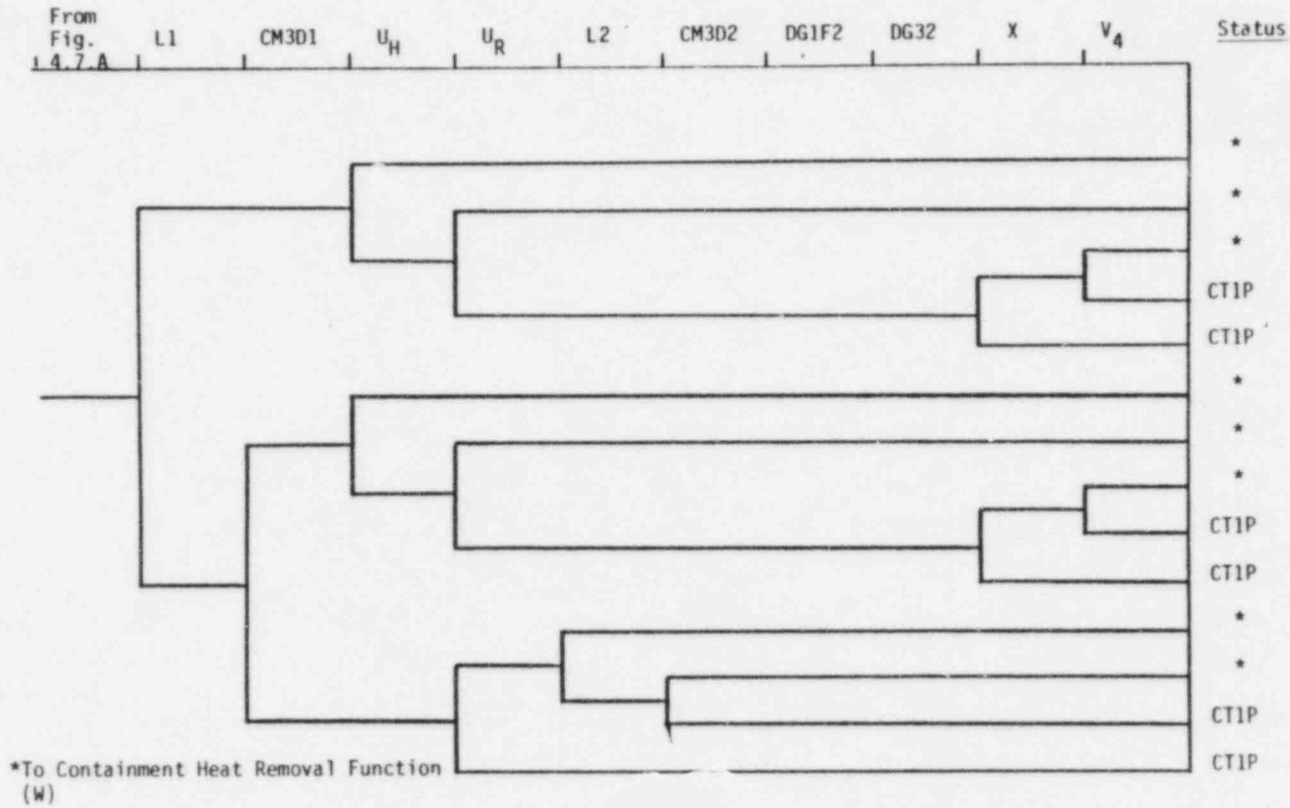


Figure 4.7.C BNL revised functional event tree for LOOP (continued).

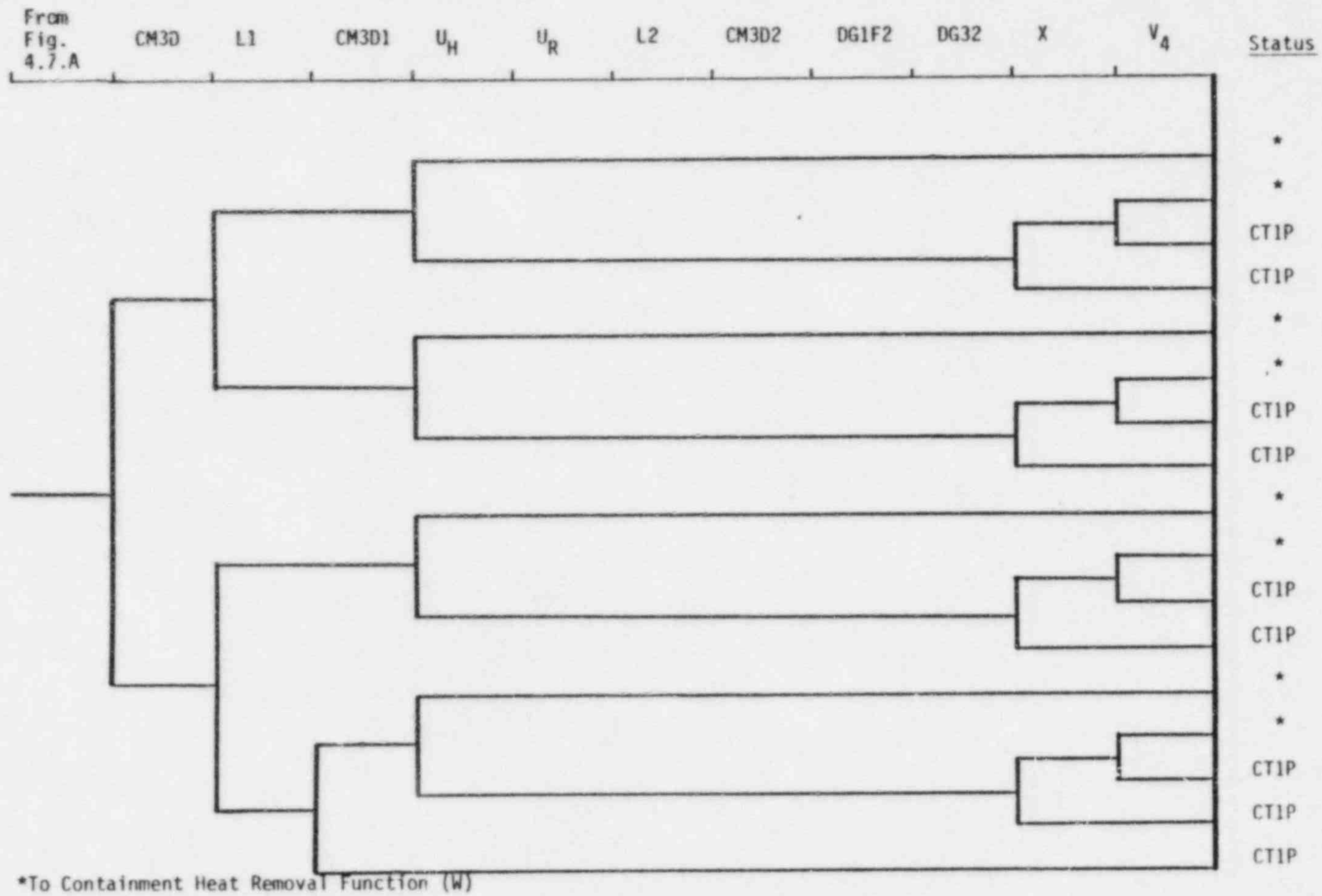


Figure 4.7.D BNL revised functional event tree for LOOP (continued).

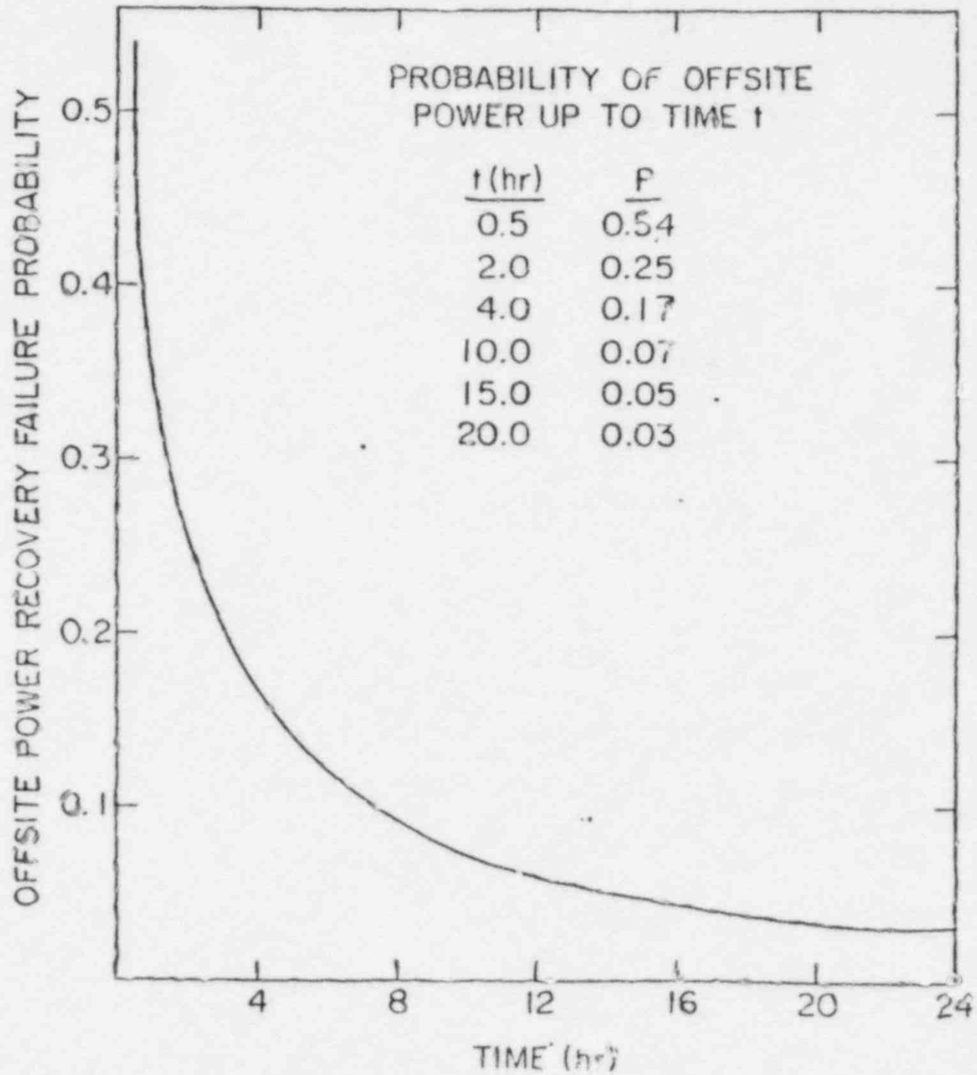


Figure 4.8 Offsite power recovery failure probability.

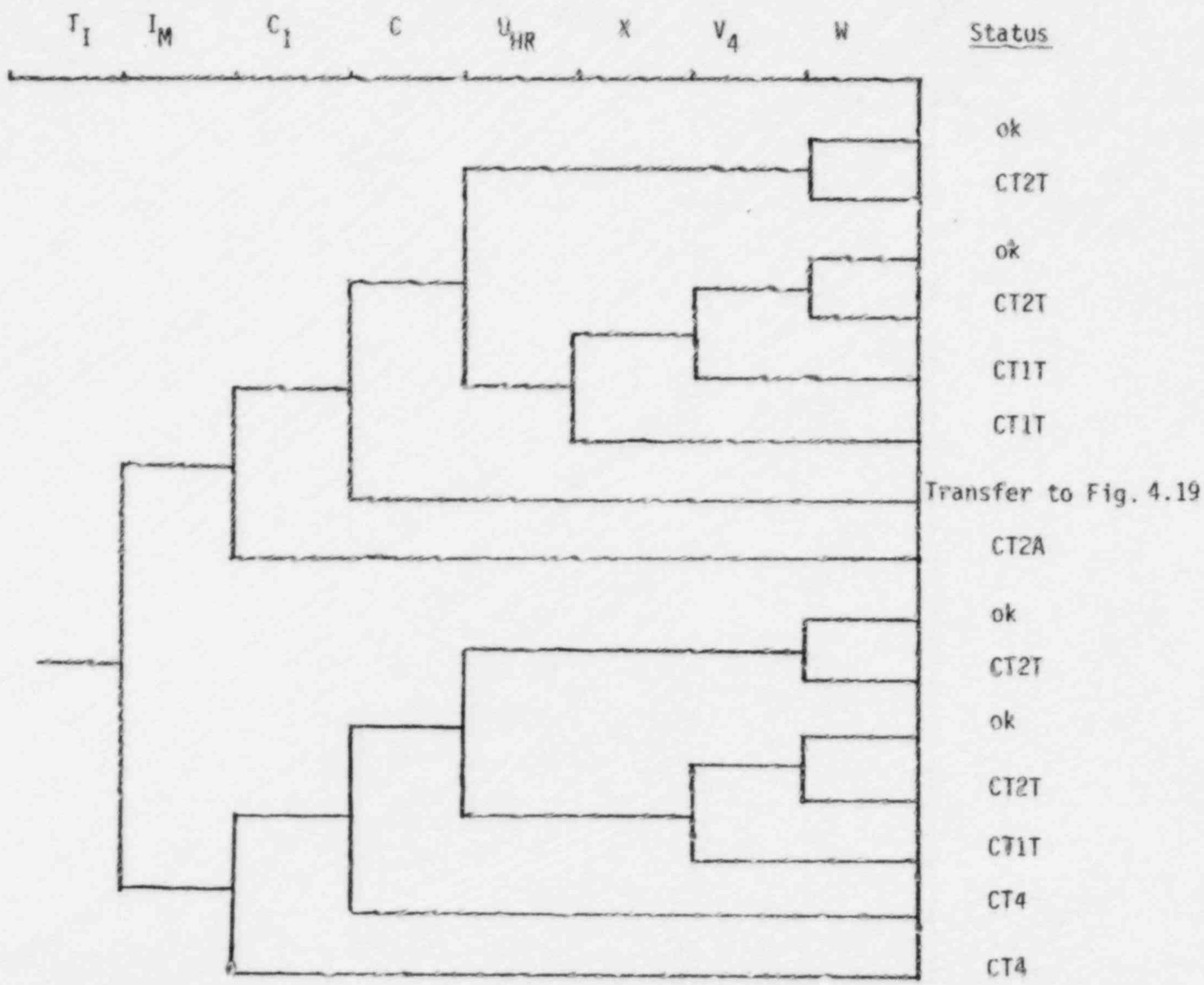


Figure 4.9 BNL revised IORV functional event tree.

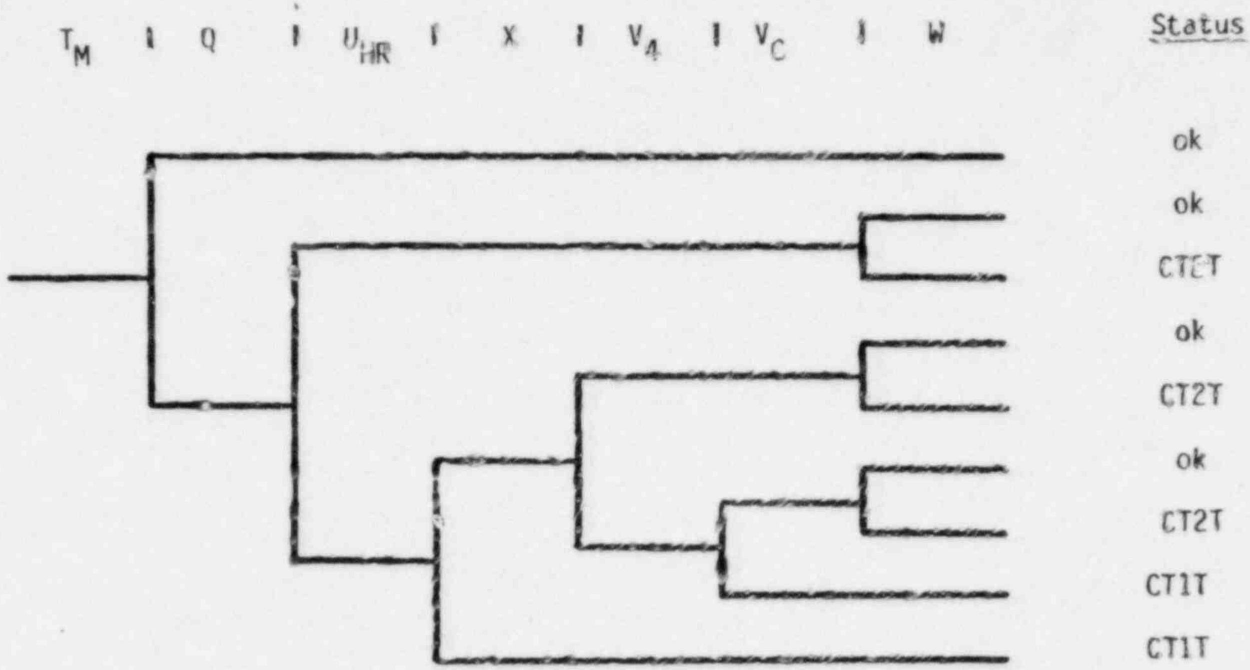


Figure 4.10 BNL revised manual shutdown functional event tree.

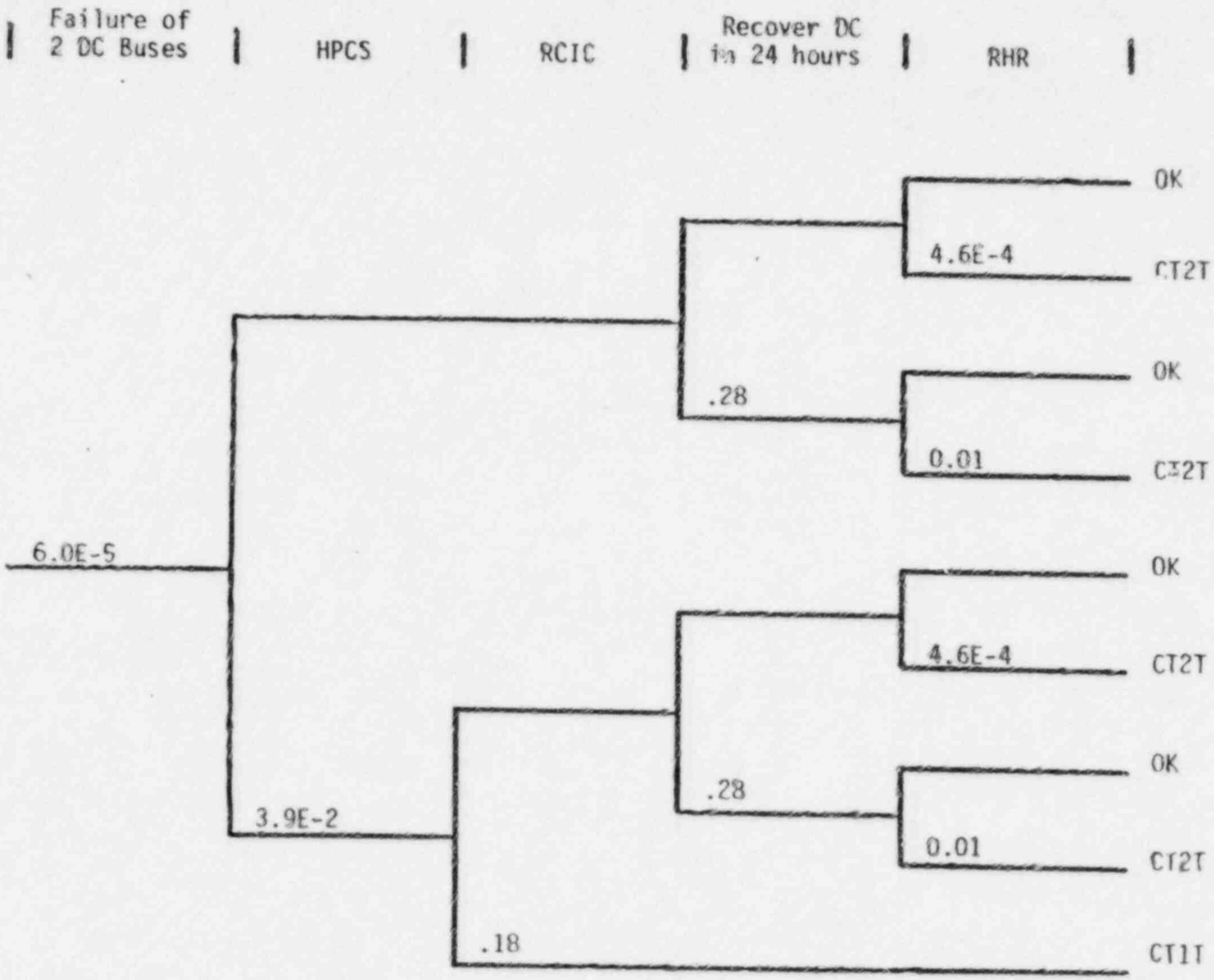
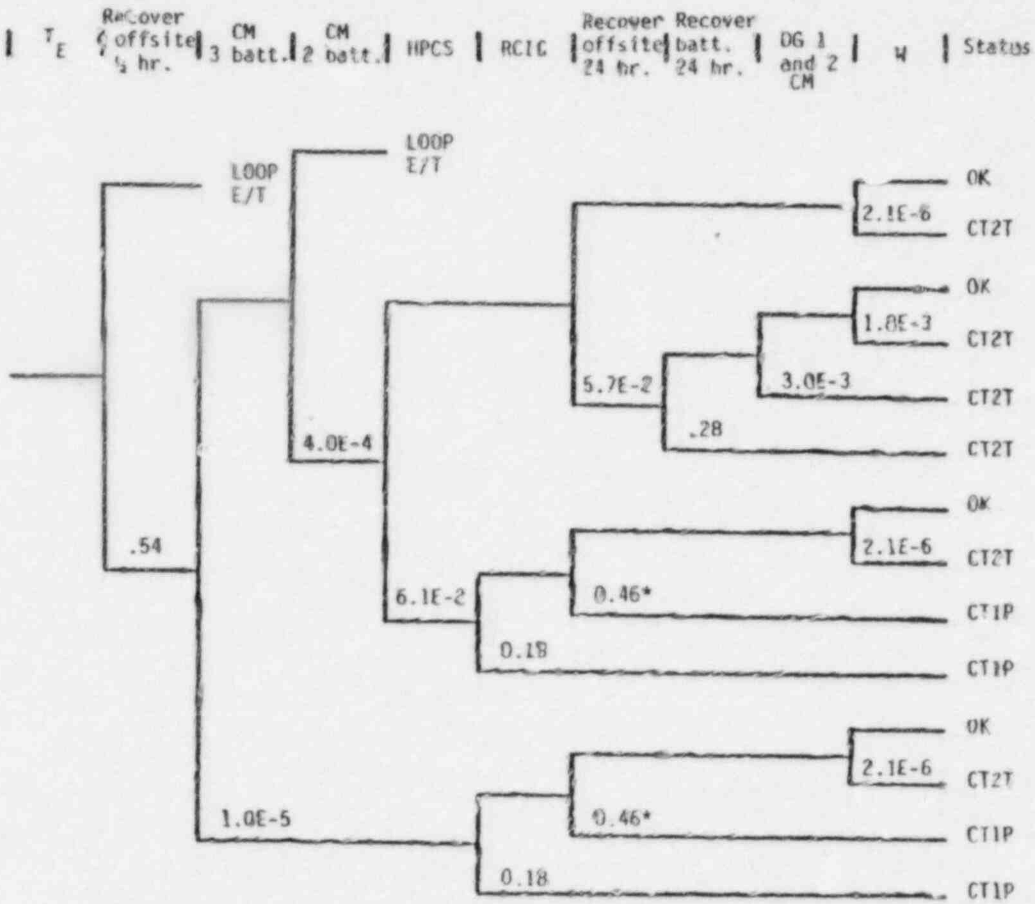
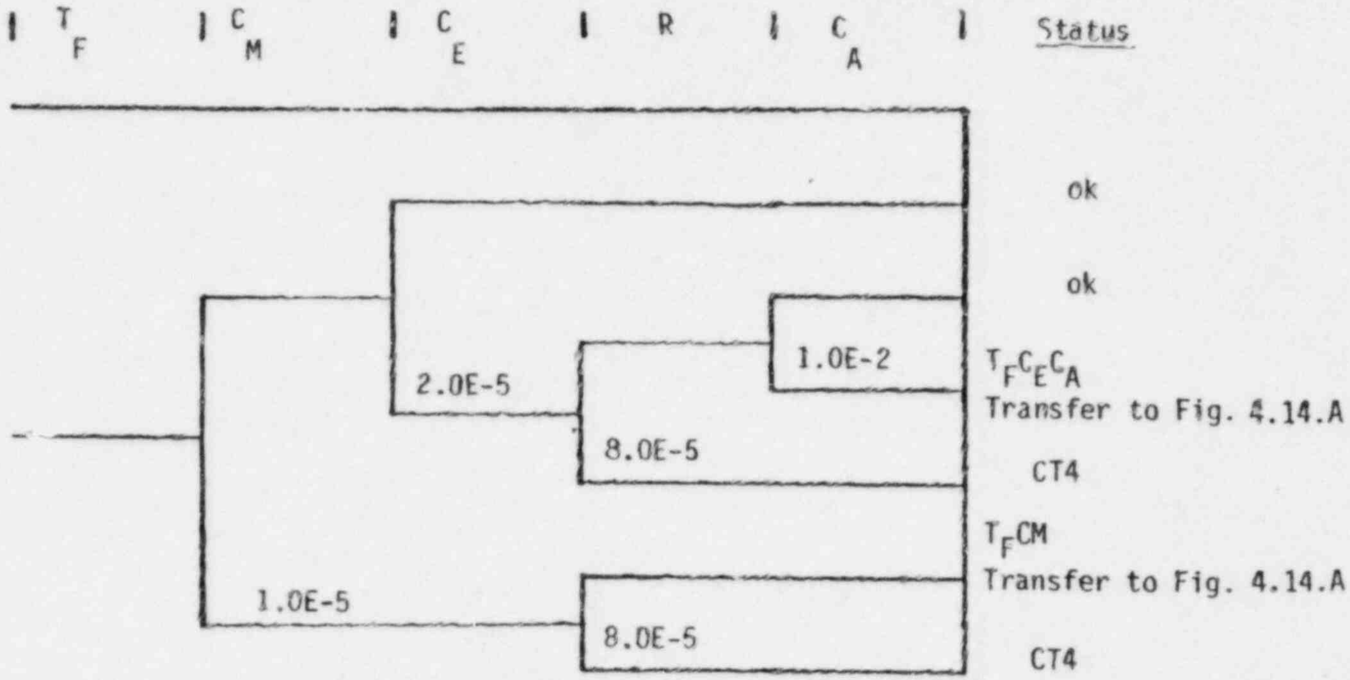


Figure 4.11 Loss of 2 dc buses event tree.



* value for failure to recover offsite power in 2 hours given failure at ½ hr.

Figure 4.12 Loss of dc power event tree.



Transfer = 9.59E-5

Figure 4.13 BNL revised isolation ATWS prevention event tree.

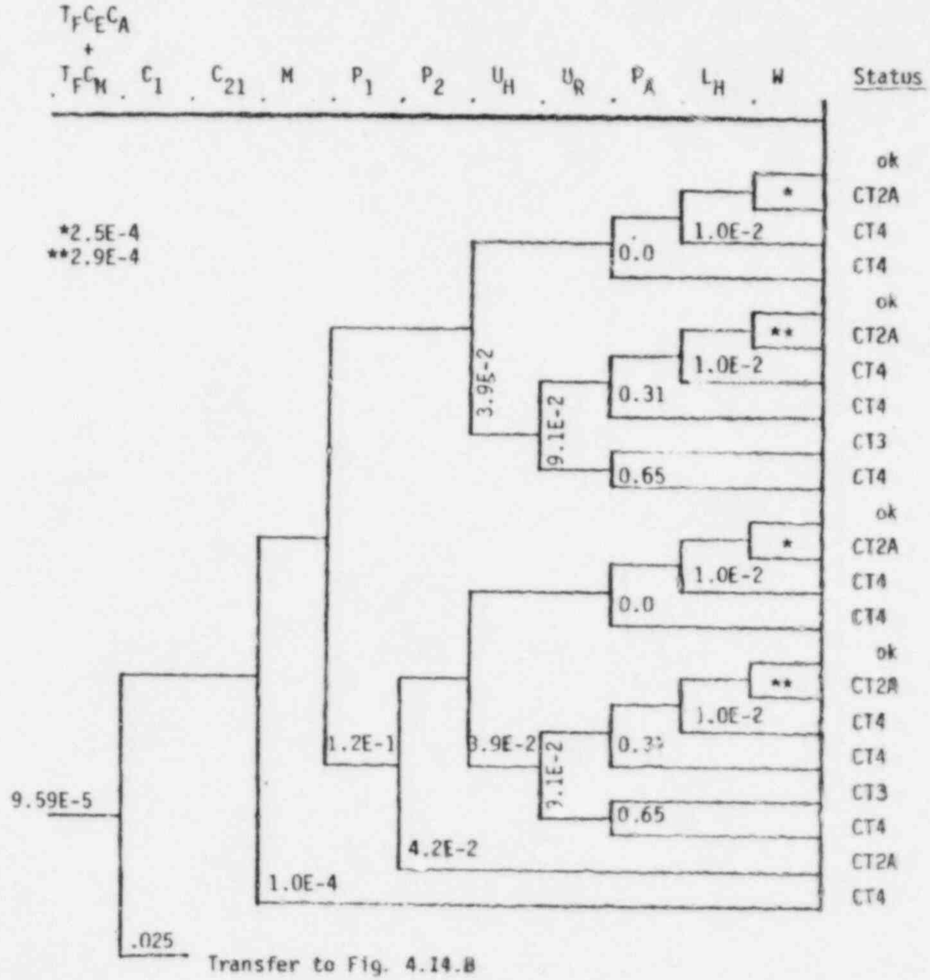


Figure 4.14.A BNL revised isolation ATWS mitigation event tree.

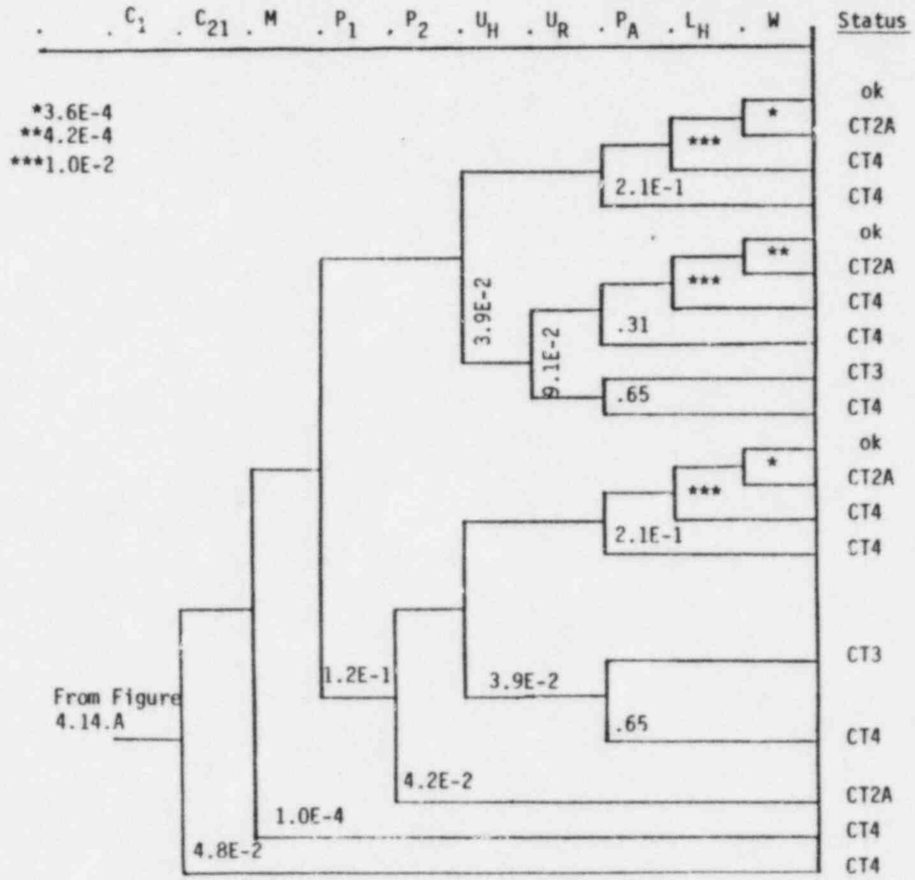


Figure 4.14.B BNL revised isolation ATWS mitigation event tree (continued).

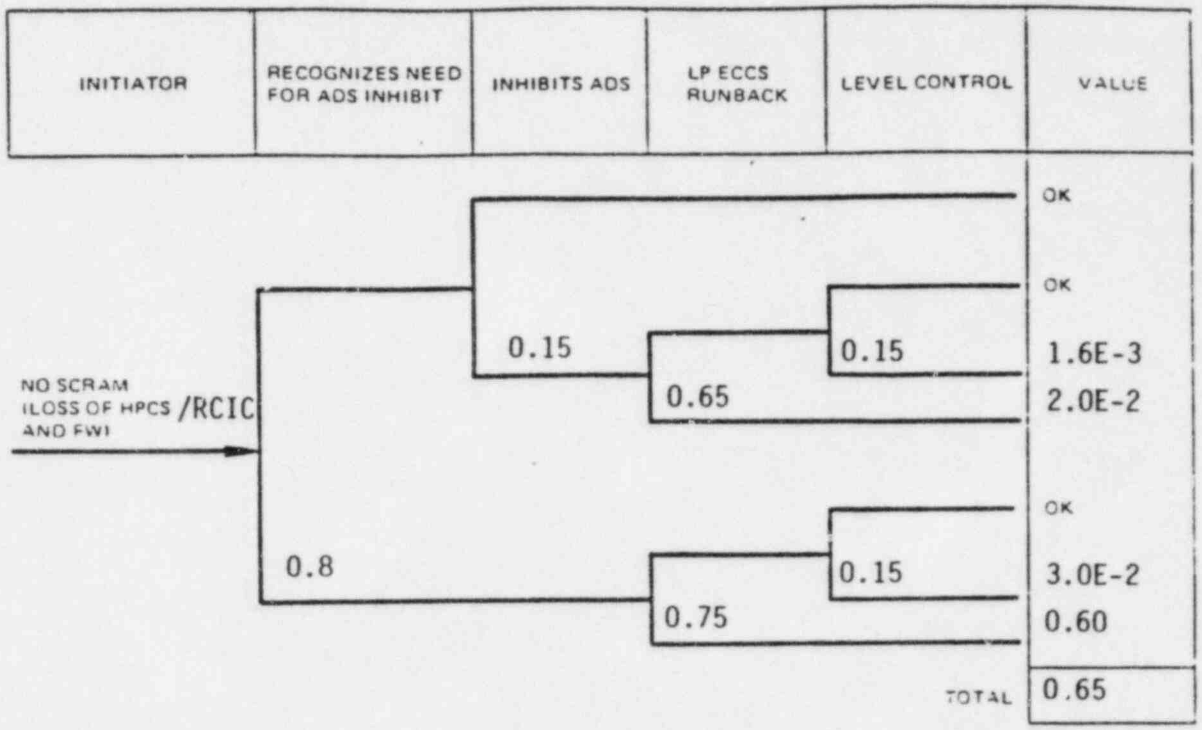


Figure 4.15 BNL revised ADS inhibit failure with loss of HPCS/RCIC and FW.

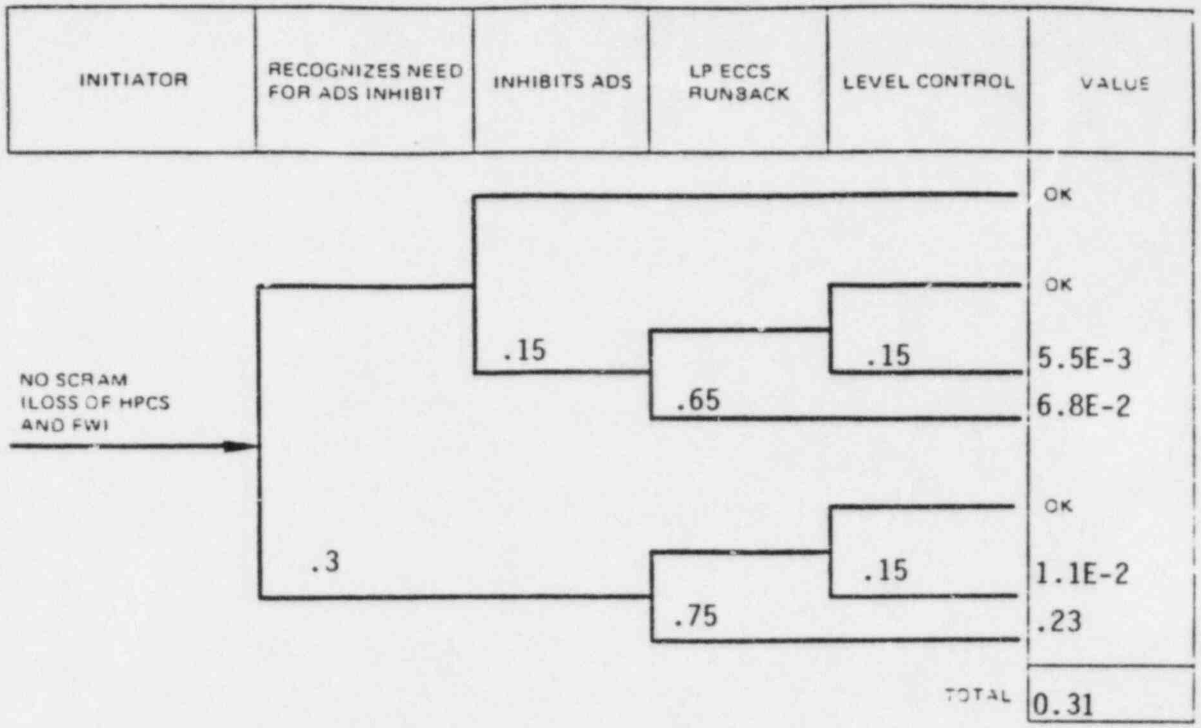


Figure 4.16 BNL revised ADS inhibit failure with loss of HPCS and FW.

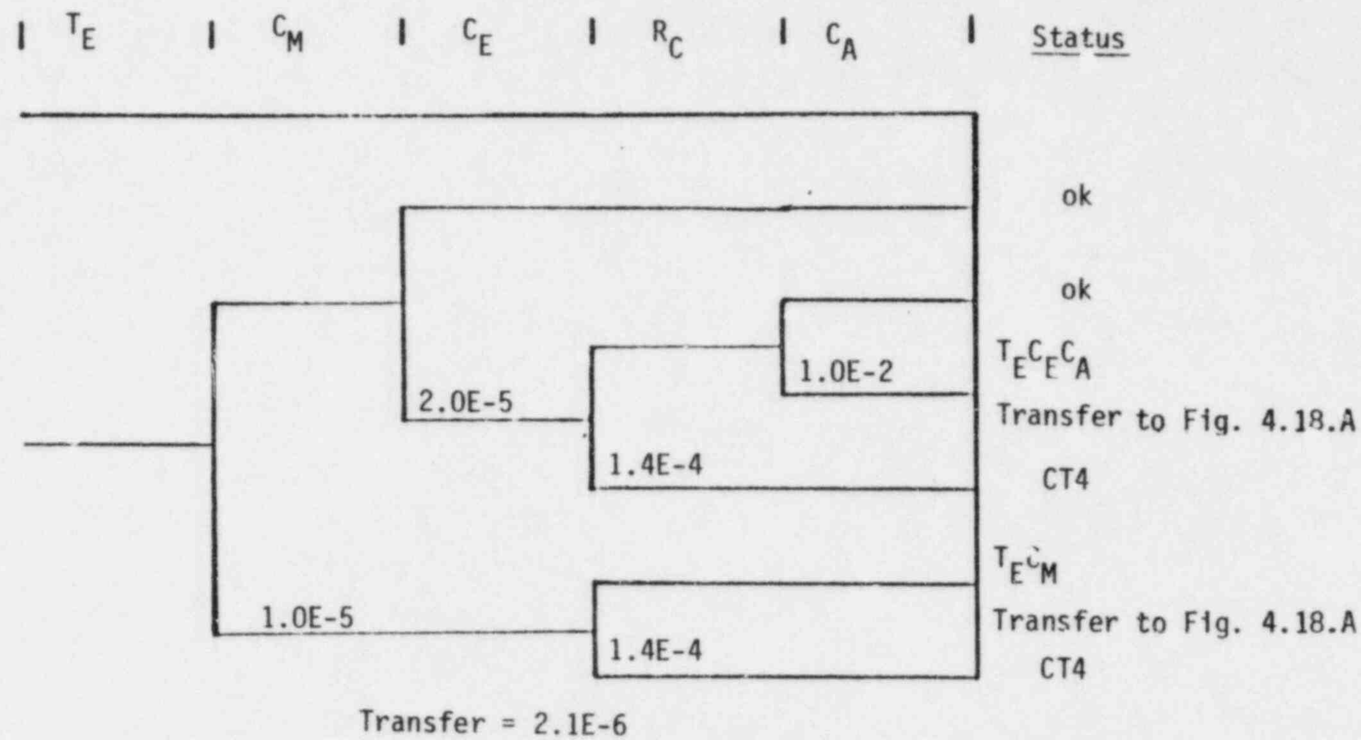


Figure 4.17 BNL revised LOOP ATWS prevention event tree.

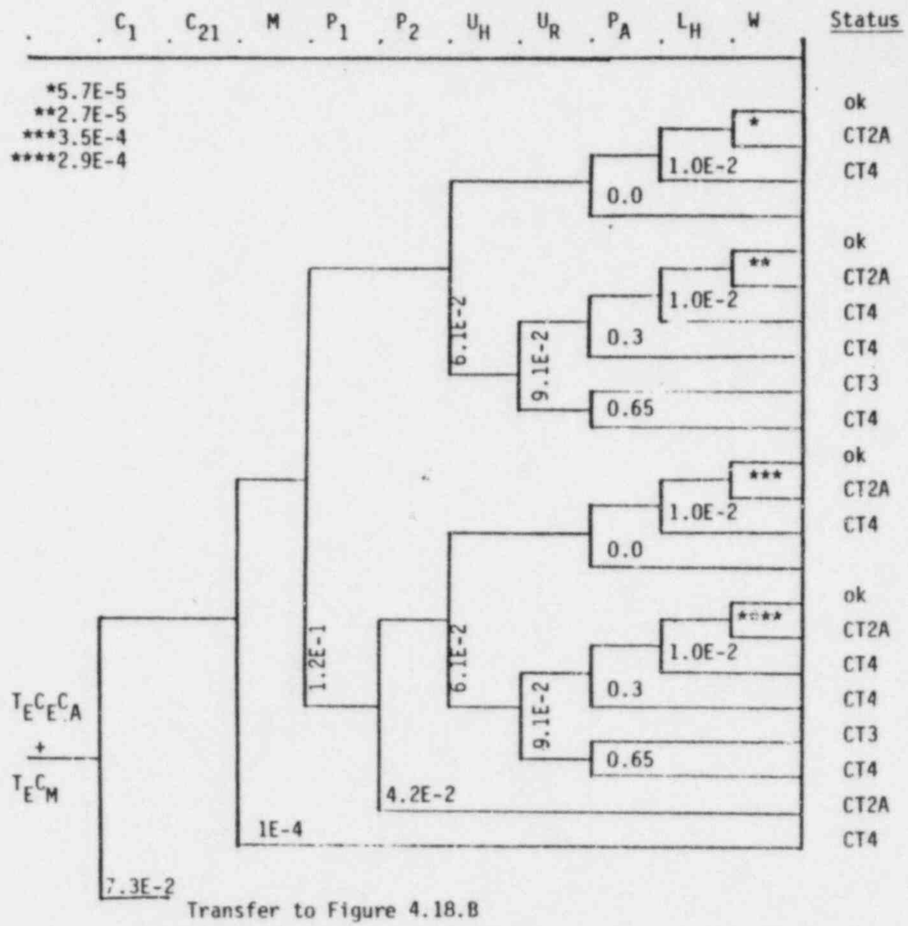


Figure 4.18.A BNL revised LOOP ATWS mitigation event tree.

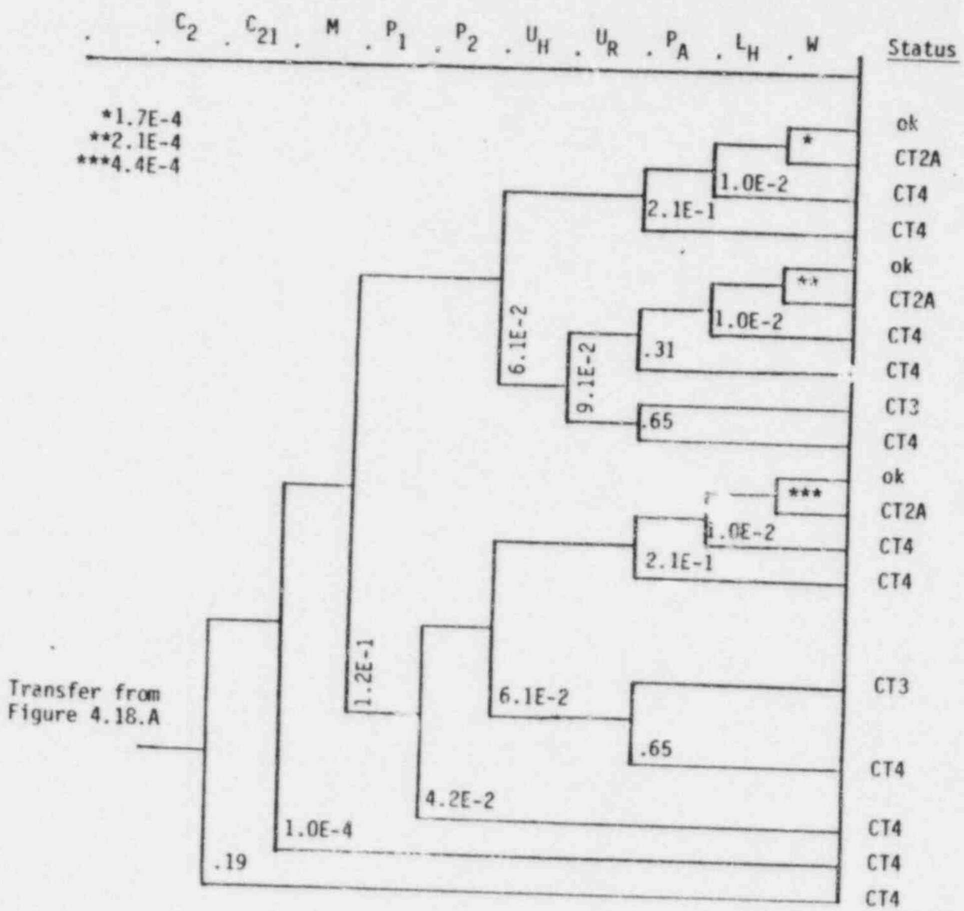


Figure 4.18.B BNL revised LOOP ATWS mitigation event tree (continued).

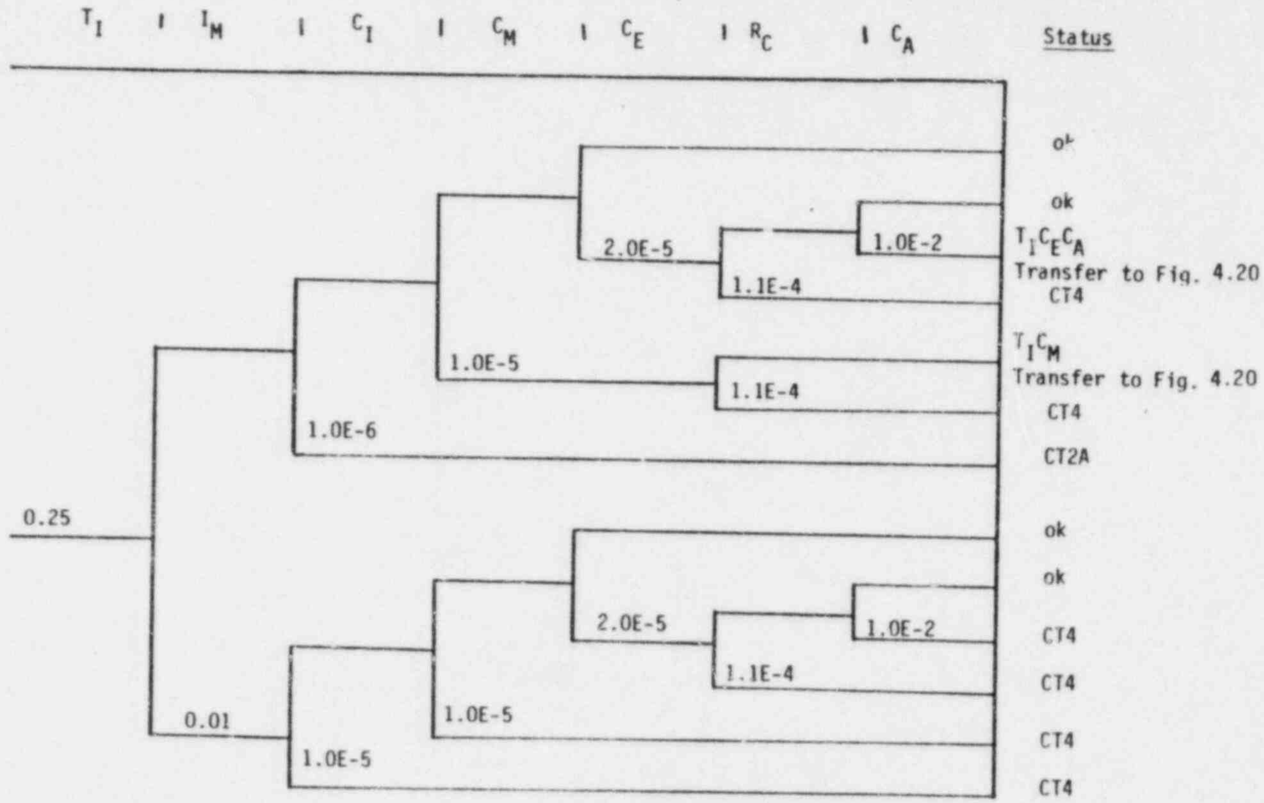


Figure 4.19 BNL revised IORV ATWS prevention event tree.

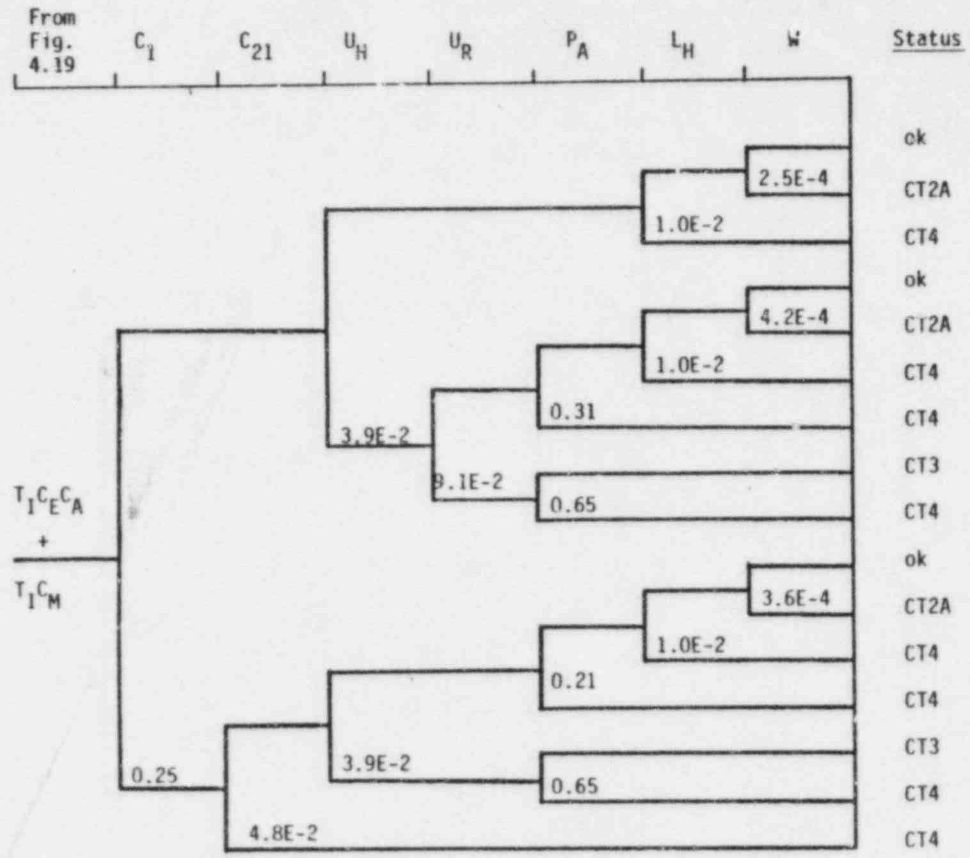


Figure 4.20 BNL revised IORV ATWS mitigation event tree.

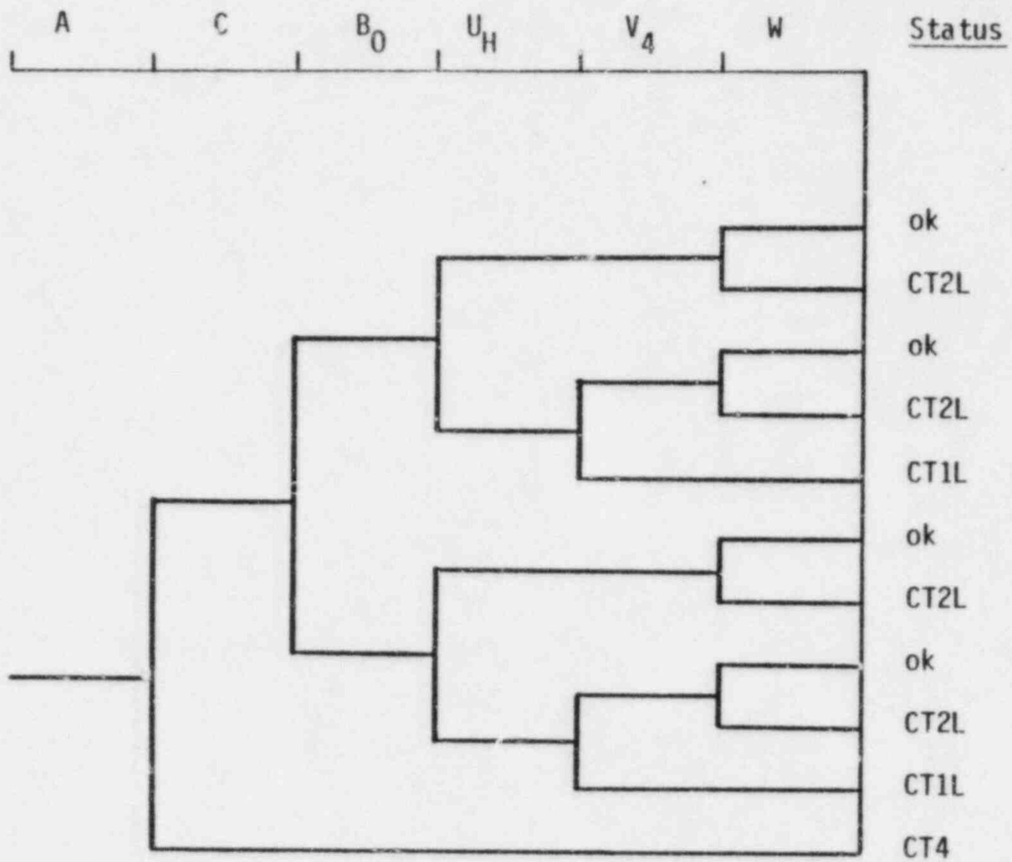


Figure 4.21 BNL revised functional event tree for large LOCA in drywell.

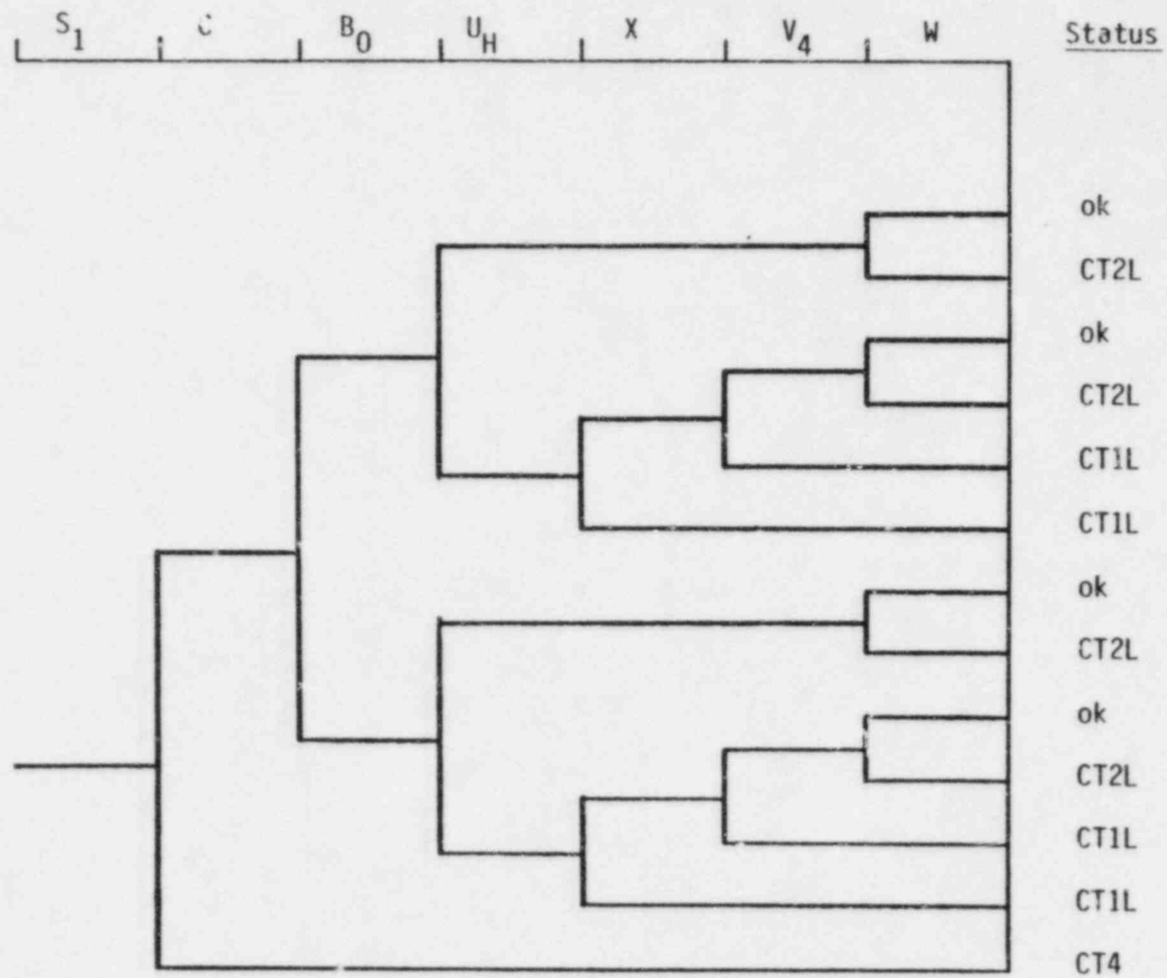


Figure 4.22 BNL revised functional event tree for intermediate LOCA in drywell.

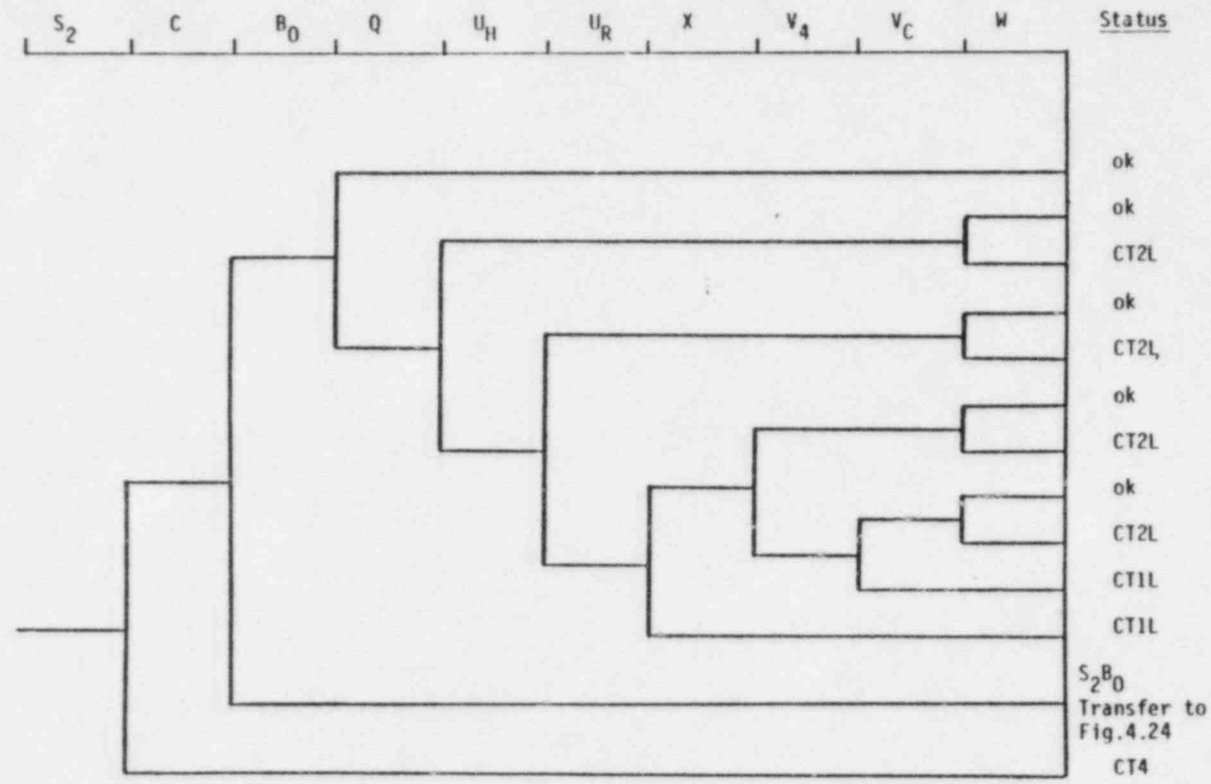


Figure 4.23 Small LOCA in drywell.

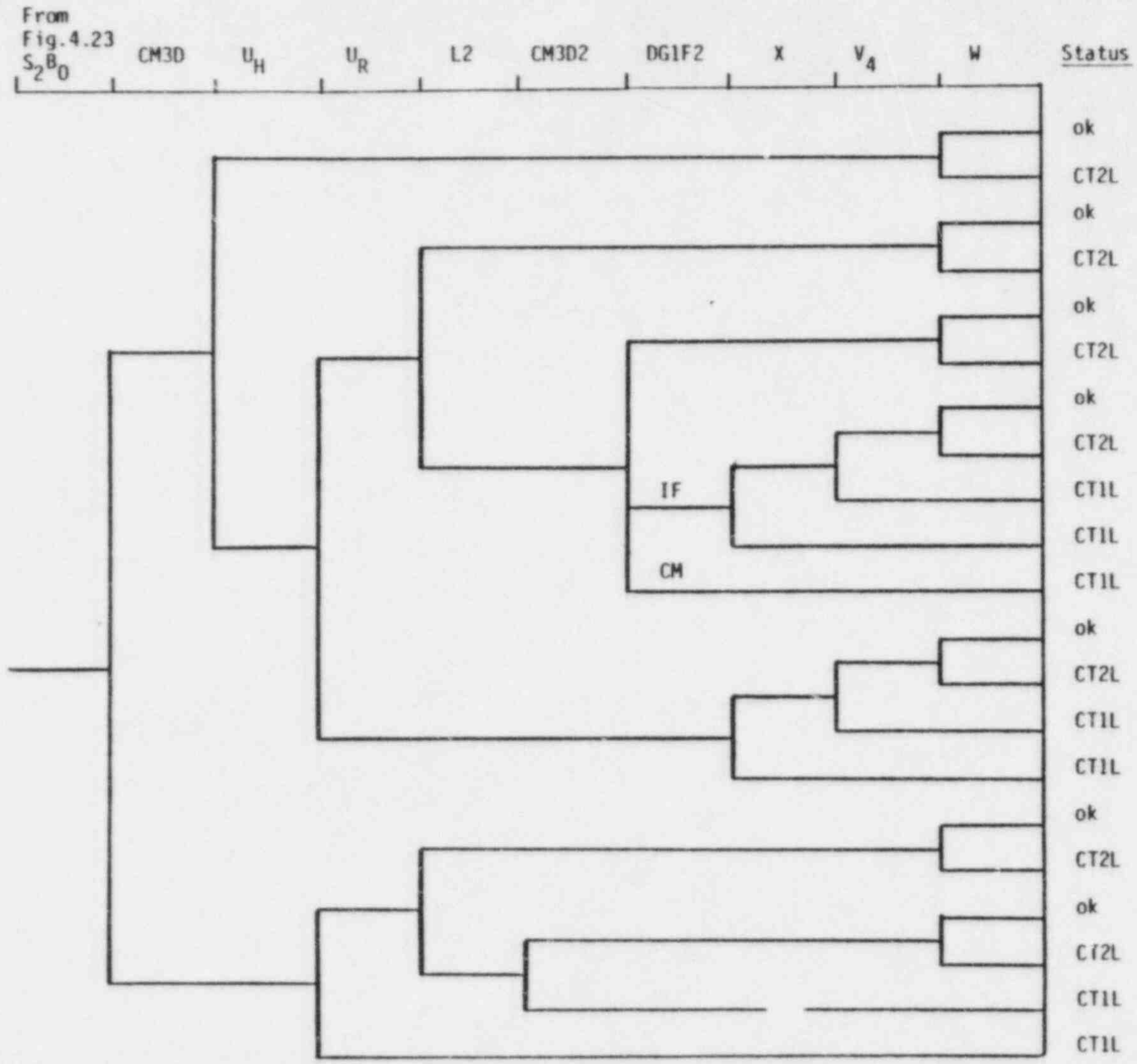


Figure 4.24 BNL revised functional event tree for small LOCA in drywell.

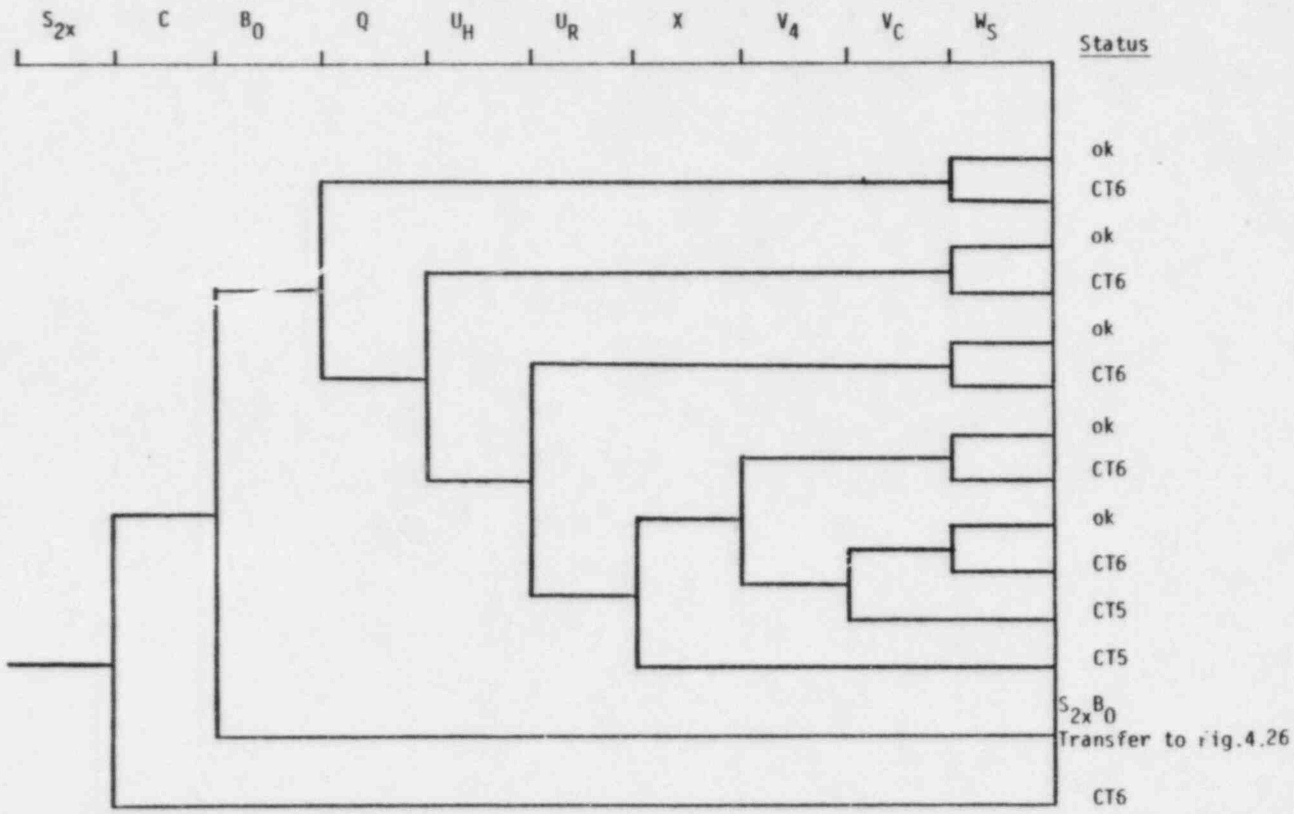


Figure 4.25 BNL revised functional event tree for small LOCA in containment.

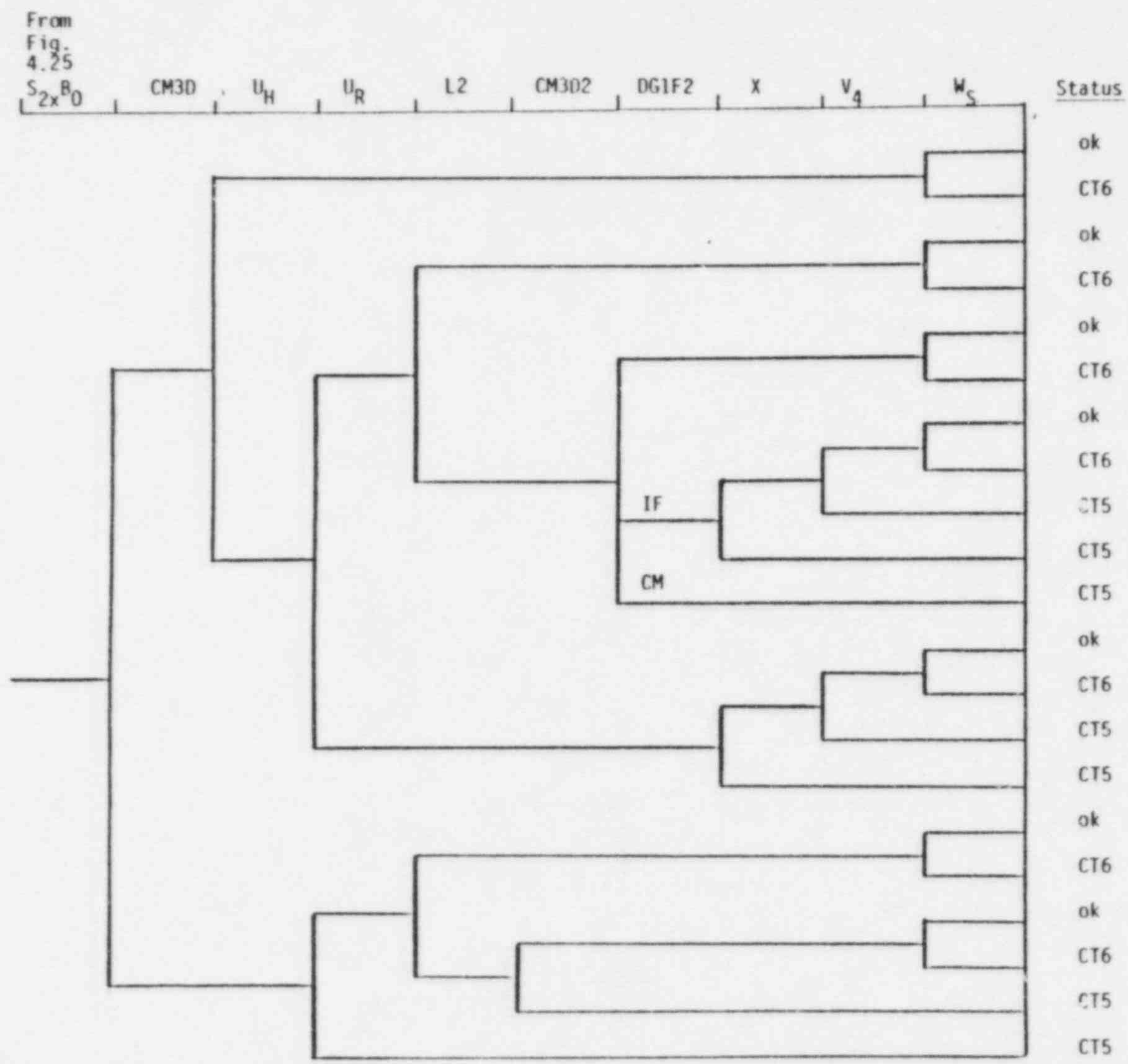


Figure 4.26 BNL revised functional event tree for small LOCA in containment with loss of offsite power.

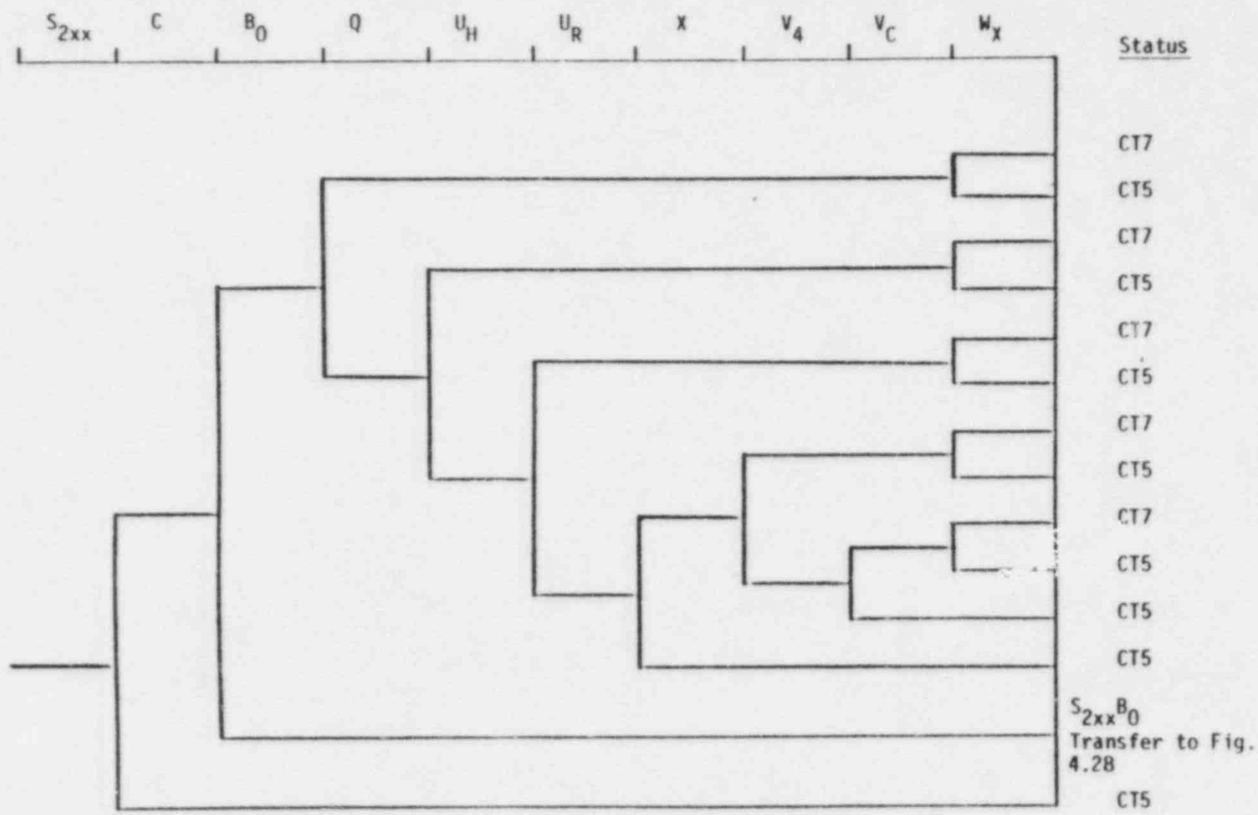


Figure 4.27 Small LOCA outside containment.

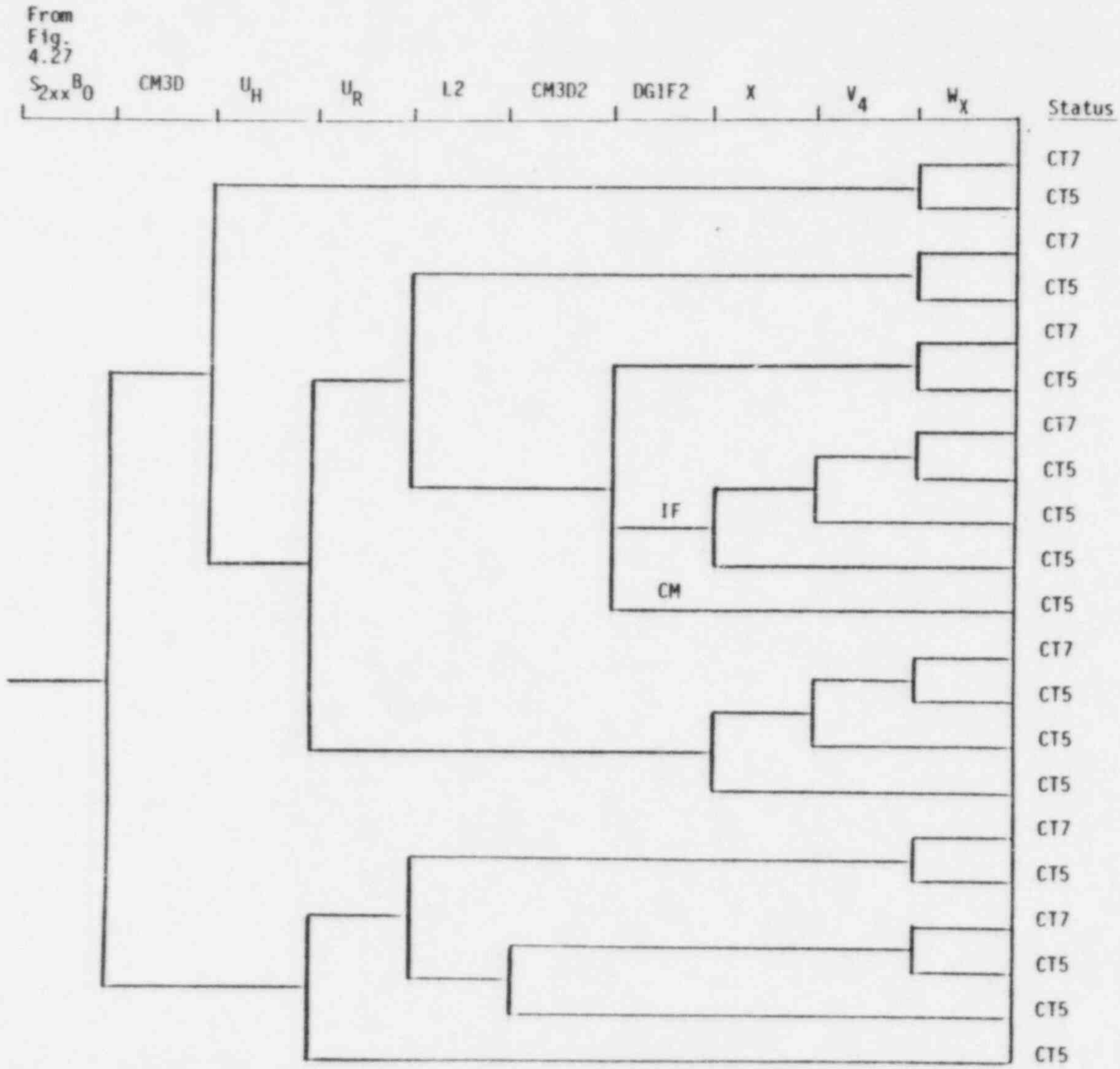


Figure 4.28 BNL revised functional event tree for small LOCA outside containment.

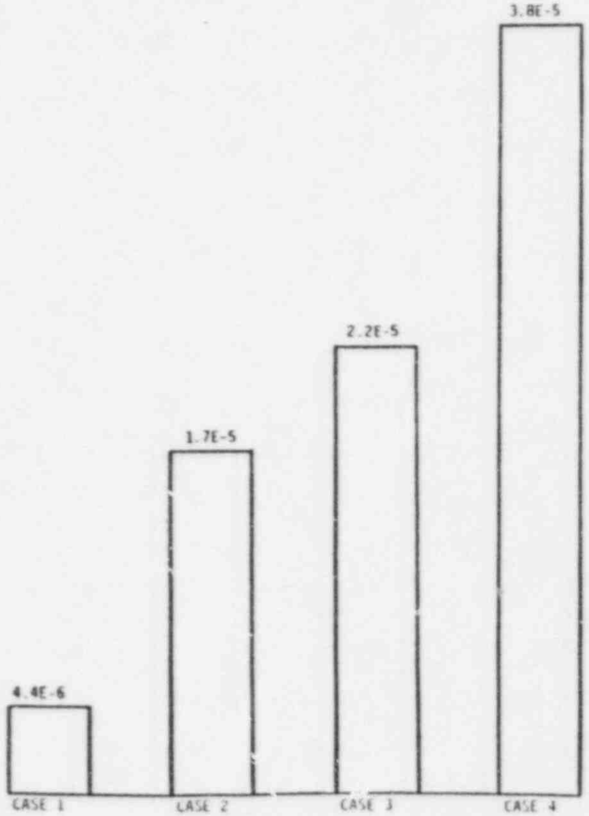


Figure 4.29 Total core damage frequency for the four cases.

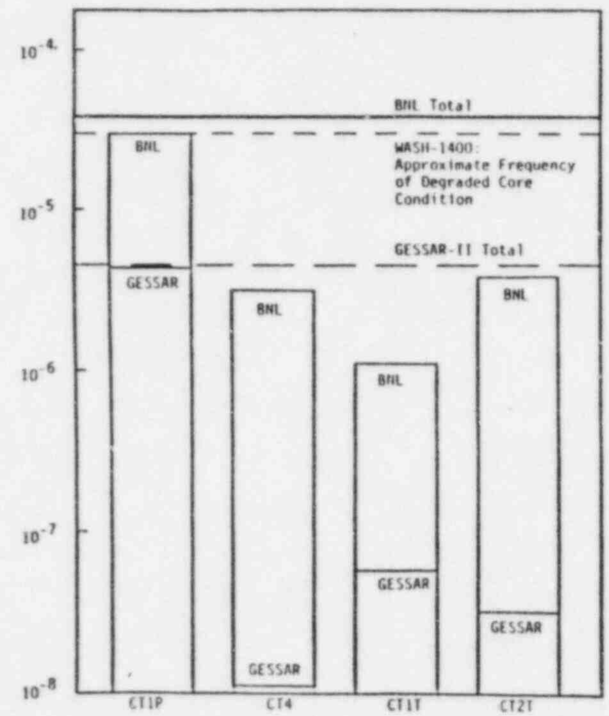


Figure 4.30 Summary of the accident sequence frequencies leading to degraded core conditions summed over all accident sequences within a class (other classes are not shown).

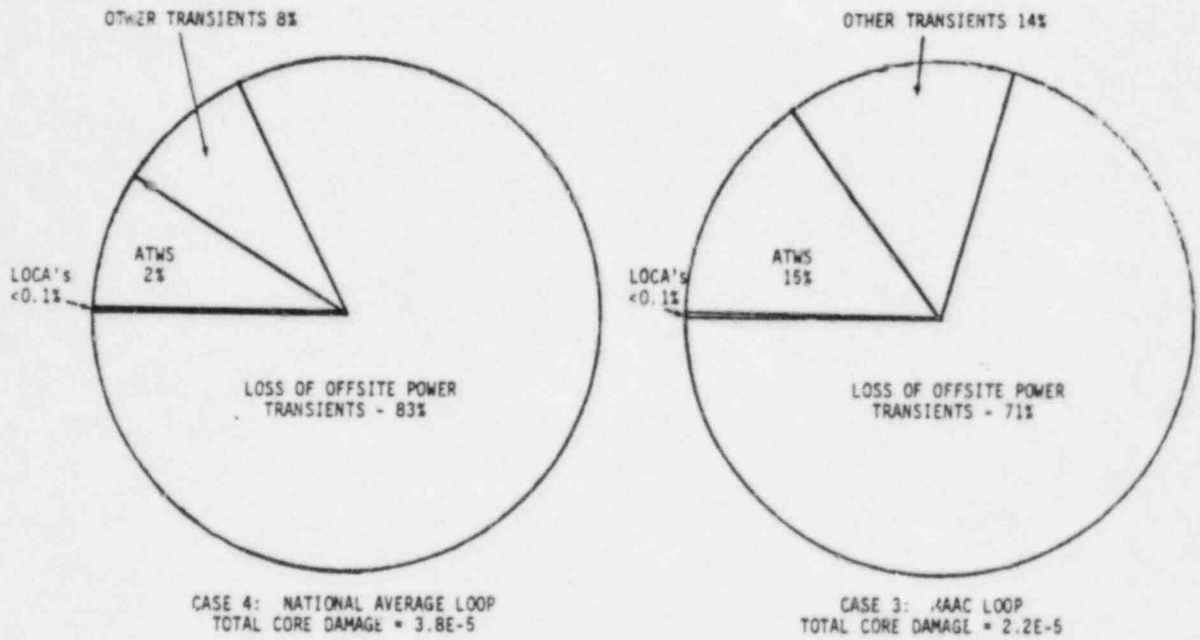


Figure 4.31 Contribution of accident initiators to the frequency of core damage (per reactor year).

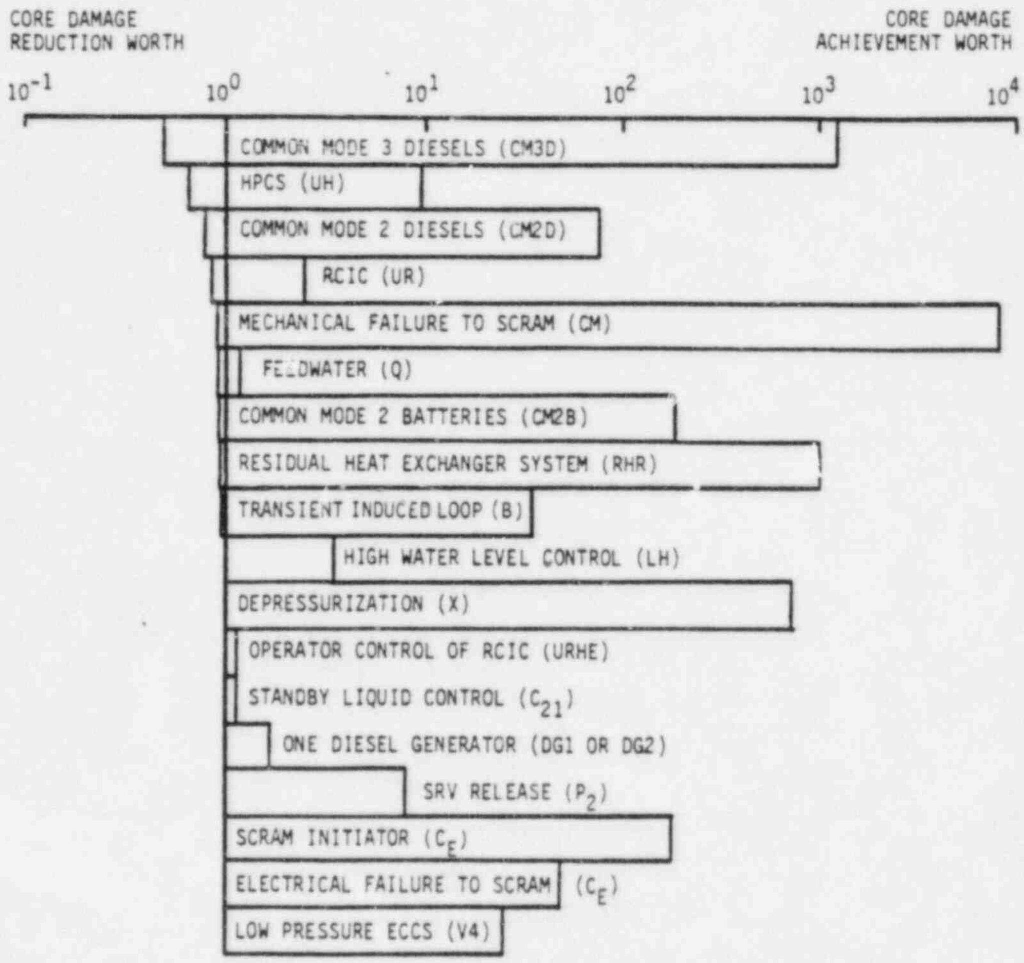


Figure 4.32 Core damage achievement and reduction worth for various systems and functions.

Table 4.1 Transient Initiator Frequencies for GESSAR-II

	GESSAR-II PRA	BNL
Planned Reactor Shutdown (T_M)	[redacted]	3.20
Turbine Trip (T_T)		4.93
Isolation (T_F)		4.28
Loss of Offsite Power (T_E)		0.22*
		0.11**
Inadvertent Open Relief Valves (T_I)		0.25
Turbine Trip ATWS		0.0
Isolation ATWS		9.21
Loss of Offsite Power ATWS		0.22*
		0.11**
Inadvertent Open Relief Valve ATWS		0.25
Loss of Two DC Buses		6.0E-5

*National Average
**Middle Atlantic Area Council (MAAC)
***Includes recovery in the first half hour.

[redacted]

Table 4.3 Comparison of Systems Fault Trees Results

	GESSAR-II PRA	BNL
HPCS	[redacted]	3.9E-2
RCIC	[redacted]	9.1E-2
ADS	[redacted]	2.5E-5
LPCCS*	[redacted]	3.0E-5
RHR (Suppression Room Cooling)	[redacted]	5.3E-4
RHR (Containment Spray)	[redacted]	5.7E-4
SLC	[redacted]	1.2E-3

*LPCCS includes the LPCI and LPCS systems.

Table 4.4 Conditional Probabilities of Core Damage Given an Accident Initiator

	CLASS CTIT/CTIP		CLASS CT2T*	
	BNL	GE	BNL	GE
Turbine Trip	6.5E-9	[redacted]	1.4E-7 / 2.0E-8	[redacted]
Isolation	1.3E-7	[redacted]	3.2E-6 / 4.7E-7	[redacted]
IORV	5.4E-7	[redacted]	8.4E-7 / 1.2E-7	[redacted]
Reactor Shutdown	3.1E-9	[redacted]	2.7E-8 / 3.9E-9	[redacted]
Loss of Offsite Power	1.3E-4	[redacted]	1.5E-5 / 6.8E-6	[redacted]
Loss of DC Power	7.2E-3	[redacted]	3.1E-3 / 4.9E-4	[redacted]

*Only the bottom value in each row represents core damage frequency (see Section 4.1).

Table 4.5 Core Damage Frequency for Transient Initiators

	CLASS CTIT/CTIP		CLASS CT2T*	
	BNL	GE	BNL	GE
Turbine Trip	3.2E-8	[redacted]	6.7E-7 9.8E-8	[redacted]
Isolation	5.5E-7	[redacted]	1.4E-6 2.0E-6	[redacted]
IORV	1.3E-7	[redacted]	2.1E-7 3.1E-8	[redacted]
Reactor Shutdown	9.8E-9	[redacted]	8.6E-8 1.2E-8	[redacted]
Loss of Offsite Power	3.0E-5**	[redacted]	3.5E-6** 1.6E-6	[redacted]
	1.5E-5***	[redacted]	1.7E-6*** 7.8E-7	[redacted]
Loss of DC Power	7.2E-9	[redacted]	1.9E-7 3.0E-8	[redacted]
Total with:				
National Average LOOP	3.1E-5**	[redacted]	1.8E-5** 3.8E-6	[redacted]
MAAC LOOP	1.6E-5***	[redacted]	1.6E-5*** 2.9E-6	[redacted]

*Only the bottom value in each row represents core damage frequency (see Section 4.1.

**National Average LOOP frequency.

***MAAC LOOP frequency.

Table 4.6 Conditional Probabilities of Core Damage for ATWS Given an Initiator

	CLASS CT3		CLASS CT4		CLASS CT2A*	
	BNL	GE	BNL	GE	BNL	GE
Turbine Trip	--	[redacted]	--	[redacted]	--	[redacted]
Isolation	1.3E-8	[redacted]	3.0E-7	[redacted]	5.3E-8 / 8.6E-9	[redacted]
IORV	1.6E-8	[redacted]	5.2E-7	[redacted]	1.0E-6 / 1.6E-7	[redacted]
Loss of Offsite Power	2.1E-8	[redacted]	5.7E-7	[redacted]	5.3E-8 / 8.6E-9	[redacted]

*Only the bottom value in each row represents core damage frequency (see Section 4.1).

Table 4.7 Core Damage Frequency for ATWS

	CLASS CT3		CLASS CT4		CLASS CT2A*	
	BNL	GE	BNL	GE	BNL	GE
Turbine Trip	--	[redacted]	--	[redacted]	--	[redacted]
Isolation	1.2E-7	[redacted]	2.8E-6	[redacted]	4.9E-7 / 8.0E-8	[redacted]
IORV	3.9E-9	[redacted]	1.3E-7	[redacted]	2.5E-7 / 4.1E-8	[redacted]
Loss of Offsite Power	4.8E-9	[redacted]	1.3E-7	[redacted]	1.2E-8 / 1.9E-9	[redacted]
	1.3E-7	[redacted]	3.1E-6	[redacted]	7.5E-7 / 1.2E-7	[redacted]

*Only the bottom value in each row represents core damage frequency (see Section 4.1).

Table 4.8 Core Damage Frequency for LOCA in Drywell Initiators

	CLASS CT1L		CLASS CT2L*		CLASS CT4	
	BNL	GESSAR	BNL	GESSAR	BNL	GESSAR
Large LOCA	5.4E-10	[redacted]	1.0E-7 / 7.7E-9	[redacted]	2.2E-9	[redacted]
Intermediate LOCA	1.9E-9	[redacted]	4.0E-8 / 5.8E-9	[redacted]	6.7E-9	[redacted]
Small LOCA	6.1E-10	[redacted]	9.2E-10 / 1.3E-10	[redacted]	1.2E-8	[redacted]
	3.0E-9	[redacted]	1.4[illegible]-7 / 1.4E-8	[redacted]	2.1E-8	[redacted]

*Only the bottom value in each row represents core damage frequency (see Section 4.1).

Table 4.9 Core Damage Frequency for LOCA in the Containment Initiators

	CLASS CT5		CLASS CT6*	
	BNL	GESSAR	BNL	GESSAR
Large LOCA	--	[redacted]	--	[redacted]
Intermediate LOCA	1.7E-14	[redacted]	4.8E-9 / 7.8E-10	[redacted]
Small LOCA	3.0E-12	[redacted]	2.6E-9 / 4.2E-10	[redacted]
	3.0E-12	[redacted]	7.4E-9 / 1.2E-9	[redacted]

*Only the bottom value in each row represents core damage frequency (see Section 4.1).

Table 4.10 Core Damage Frequency for LOCA Outside Containment Initiators

	CLASS CT5		CLASS CT7*	
	BNL	GESSAR	BNL	GESSAR
Large LOCA	7.4E-13	[redacted]	--	[redacted]
Intermediate LOCA	5.9E-14		--	
Small LOCA	1.9E-11		1.7E-7	
	2.0E-11		1.7E-7	

*Values in Class CT7 do not contribute to core damage (see Section 4.1).

Table 4.11 Turbine Trip: Accident Class Core Damage Frequencies

Case	CT1T	CT2T**	CT3	CT4	CT2A**	CD**
1. GESSAR PRA	[redacted]					
2. BNL Modifications in F/T-E/T	1.4E-8	3.0E-7 / 4.4E-8	--	--	--	3.1E-7 / 5.8E-8
3. BNL Initiators	3.2E-8	6.7E-7 / 9.8E-8	--	--	--	7.0E-7 / 1.3E-7

Table 4.12 Isolation: Accident Class Core Damage Frequencies

Case	CT1T	CT2T**	CT3	CT4	CT2A**	CD**
1. GESSAR-II PRA	[redacted]					
2. BNL Modifications in F/T-E/T	2.6E-7	3.0E-7 / 9.2E-7	5.3E-8	1.2E-6	2.2E-7 / 3.6E-8	8.0E-6 / 2.5E-6
3. BNL Initiators	5.5E-7	1.4E-5 / 2.0E-6	1.2E-7	2.8E-6	4.9E-7 / 8.0E-8	1.8E-5 / 5.5E-6

** Only the bottom value represents core damage frequency (see Section 4.1).

Table 4.13 Loss of DC Power: Accident Class Core Damage Frequencies

Case	Class					
	CT1T	CT2T*				CD
1. GESSAR-II PRA	--	--	--	--	--	--
2. BNL Modifications in F/T-E/T	--	--	--	--	--	--
3. BNL Initiators	4.3E-7	1.9E-7 / 3.0E-8	--	--	--	6.2E-7 / 4.6E-7

Table 4.14 LOOP: Accident Class Core Damage Frequencies

Case	Class					
	CT1T	CT2T*	CT3	CT4	CT2A*	CD*
1. GESSAR PRA	[redacted]					
2. BNL Modifications in F/T-E/T	1.3E-5	1.5E-6 / 6.9E-7	2.0E-9	5.5E-8	5.1E-9 / 8.3E-10	1.5E-5 / 1.4E-5
3. BNL Initiators -MAAC Loop	1.5E-5	1.7E-6 / 7.8E-7	2.3E-9	6.3E-8	5.8E-9 / 9.4E-10	1.7E-5 / 1.6E-5
4. BNL Initiators -Nat. Avg. Loop	3.0E-5	3.5E-6 / 1.6E-6	4.8E-9	1.3E-7	1.2E-8 / 1.9E-9	3.4E-5 / 3.2E-5

Table 4.15 IORV: Accident Class Core Damage Frequencies

Case	Class					
	CT1T	CT2T*	CT3	CT4	CT2A*	CD*
1. GESSAR-II PRA	[redacted]					
2. BNL Modifications in F/T-E/T	2.7E-8	4.2E-8 / 6.1E-9	8.0E-10	2.6E-8	5.0E-8 / 8.1E-9	1.5E-7 / 6.8E-8
3. BNL Initiators	1.3E-7	2.1E-7 / 3.1E-8	3.9E-9	1.3E-7	2.5E-7 / 4.1E-8	7.2E-7 / 3.4E-7

* Only the bottom value represents core damage frequency (see Section 4.1).

Table 4.16 Manual Shutdown: Accident Class Core Damage Frequencies

Case	Class					
	CT1T	CT2T*				CD*
1. GESSAR PRA	[redacted]					
2. BNL Modifications in F/T-E/T	6.1E-9	5.3E-8 / 7.7E-9	--	--	--	5.9E-8 / 1.4E-8
3. BNL Initiators	9.8E-9	8.6E-8 / 1.2E-8	--	--	--	9.6E-8 / 2.2E-8

Table 4.17 Large LOCA in Drywell: Accident Class Core Damage Frequencies

Case	Class					
	CT1T	CT2T*	CT3			CD*
1. GESSAR PRA	[redacted]					
2. BNL Modifications in F/T-E/T	5.4E-10	1.0E-7 / 7.7E-9	2.1E-9	--	--	1.0E-7 / 8.2E-9
3. BNL Initiators	5.4E-10	1.0E-7 / 7.7E-9	2.2E-9	--	--	1.0E-7 / 8.2E-9

Table 4.18 Medium LOCA in Drywell: Accident Class Core Damage Frequencies

Case	Class					
	CT1T	CT2T*				CD*
1. GESSAR PRA	[redacted]					
2. BNL Modifications in F/T-E/T	1.9E-9	4.0E-8 / 5.8E-9	6.7E-9	--	--	4.9E-8 / 1.4E-8
3. BNL Initiators	1.9E-9	4.0E-8 / 5.8E-9	6.7E-9	--	--	4.9E-8 / 1.4E-8

* Only the bottom value represents core damage frequency (see Section 4.1).

Table 4.19 Small LOCA in Drywell: Accident Class Core Damage Frequencies

Case	Class					
	CT1T	CT2T*	CT4			CD*
1. GESSAR-II PRA	[redacted]					
2. BNL Modifications in F/T-E/T	6.1E-10	9.2E-10 / 1.3E-10	1.2E-8	--	--	1.3E-8 / 1.3E-8
3. BNL Initiators	6.1E-10	9.2E-10 / 1.3E-10	1.2E-8	--	--	1.3E-8 / 1.3E-8

Table 4.20 Medium LOCA in the Containment: Accident Class Core Damage Frequencies

Case	Class					
	CT5	CT6*				CD*
1. GESSAR-II PRA	[redacted]					
2. BNL Modifications in F/T-E/T	1.7E-14	4.8E-9 / 7.8E-10	--	--	--	4.8E-9 / 7.8E-10
3. BNL Initiators	1.7E-14	4.8E-9 / 7.8E-10	--	--	--	4.8E-9 / 7.8E-10

Table 4.21 Small LOCA in the Containment: Accident Class Core Damage Frequencies

Case	Class					
	CT5	CT6*				CD*
1. GESSAR-II PRA	[redacted]					
2. BNL Modifications in F/T-E/T	3.0E-12	2.6E-9 / 4.2E-10	--	--	--	2.6E-9 / 4.2E-10
3. BNL Initiators	3.0E-12	2.6E-9 / 4.2E-10	--	--	--	2.6E-9 / 4.2E-10

* Only the bottom value represents core damage frequency (see Section 4.1).

Table 4.22 Large LOCA Outside Containment: Accident Class Core Damage Frequencies

Case	Class: CT5					CD
1. GESSAR PRA	[redacted]					
2. BNL Modifications in F/T-E/T	7.4E-13	--	--	--	--	7.4E-13
3. BNL Initiators	7.4E-13	--	--	--	--	7.4E-13

Table 4.23 Medium LOCA Outside Containment: Accident Class Core Damage Frequencies

Case	Class: CT5					CD
1. GESSAR-II PRA	[redacted]					
2. BNL Modifications in F/T-E/T	5.9E-14					5.9E-14
3. BNL Initiators	5.9E-14	--	--	--	--	5.9E-14

Table 4.24 Small LOCA Outside Containment: Accident Class Core Damage Frequencies

Case	Class: CT5	CT7				CD*
1. GESSAR PRA	[redacted]					
2. BNL Modifications in F/T-E/T	1.9E-11	$\frac{1.7E-7}{0.0}$	--	--	--	$\frac{1.7E-7}{1.9E-11}$
3. BNL Initiators	1.9E-11	$\frac{1.7E-7}{0.0}$	--	--	--	$\frac{1.7E-7}{1.9E-11}$

* Only the bottom value represents core damage frequency (see Section 4.1).

Table 4.25 Total Core Damage Frequency of Various Accident Classes

Class	Case 1 GESSAR-II PRA	Case 2 BNL Modifications in E/T F/T	Case 3 BNL Initiators -MAAC Loop	Case 4 BNL Initiators -Nat. Avg Loop
CT1T	5.8E-8	7.3E-7	1.1E-6	1.1E-6
CT1P	4.3E-6	1.3E-5	1.5E-5	3.0E-5
CT2T[a]	1.0E-6 / 3.3E-8	8.4E-6 / 1.7E-6	1.6E-5 / 2.9E-6	1.8E-5 / 3.8E-6
CT3	7.5E-10	5.6E-8	1.3E-7	1.3E-7
CT4	1.1E-8	1.3E-6	3.0E-6	3.1E-6
CT2A[a]	5.2E-9 / 2.4E-10	2.7E-7 / 4.5E-8	7.4E-7 / 1.2E-7	7.4E-7 / 1.2E-7
CT1L	3.1E-9	3.0E-9	3.0E-9	3.0E-9
CT2L[a]	1.1E-7 / 3.4E-9	1.4E-7 / 1.4E-8	1.4E-7 / 1.4E-8	1.4E-7 / 1.4E-8
CT5	1.9E-11	2.3E-11	2.3E-11	2.3E-11
CT6[a]	7.3E-9 / 3.4E-10	7.4E-9 / 1.2E-9	7.4E-9 / 1.2E-9	7.4E-9 / 1.2E-9
CT7[a]	1.7E-7 / 0.0	1.7E-7 / 0.0	1.7E-7 / 0.0	1.7E-7 / 0.0
Total Core Damage	4.4E-6	1.7E-5	2.2E-5	3.8E-5
Total Accident Sequences	5.5E-6	2.4E-5	3.6E-5	5.3E-5

[a]In these rows, the first value is for accident sequence frequency, and the second value is for core damage frequency.

Table 4.26 Ranking of BNL and GESSAR-II PRA Sequences by Core Damage Frequency

BNL (3.8E-5) Dominant Sequences		
1) T_EUV	3.0E-5	79.0%
2) T_FQW	1.9E-6	5.0%
3) T_EW	1.6E-6	4.2%
4) $T_FC_MU_HP_A$	8.7E-7	2.3%
5) $T_FC_ML_H$	7.6E-7	2.0%
6) T_FQUX	5.3E-7	1.4%
7) $T_{DC}UV$	4.3E-7	1.1%
8) $T_FC_MC_1P_A$	3.9E-7	1.0%
9) $T_FC_MUP_A$	1.8E-7	0.5%
10) $T_FC_MP_1U_HP_A$	1.1E-7	0.3%
11) $T_FC_MC_1C_{21}$	1.1E-7	0.3%
12) T_FC_MU	1.0E-7	0.3%
13) T_IUX	1.0E-7	0.3%

GESSAR-II (4.4E-6) Dominant Sequences		
1) T_EUV	4.0E-6	90.0%
2) T_EUX	1.3E-7	2.1%
3) T_EP_1UV	1.2E-7	2.6%
4) T_FP_1UV	1.6E-8	0.4%
5) T_FQUX	1.3E-8	0.3%
6) T_IUV	1.2E-8	0.3%

Table 4.27 Class CT1T Dominant Sequences

1)	$T_F QUX$	5.3E-7
2)	$T_D CUV$	4.3E-7
3)	$T_I UX$	1.0E-7
4)	$T_T QUX$	2.9E-8
5)	$T_I UV$	2.7E-8

Table 4.28 Class CT1P Dominant Sequences

1)	$T_E UV$	2.7E-5
2)	T_E(CM2BATT)UV	1.7E-6
3)	T_E(CM3BATT)UV	6.9E-7

Table 4.29 Class CT2T Dominant Sequences*

1)	$T_F QW$	$\frac{1.3E-5}{1.9E-6}$
2)	$T_E W$	$\frac{2.6E-6}{1.1E-6}$
3)	T_E(CM2BATT)W	$\frac{7.3E-7}{4.7E-7}$
4)	$T_T P_2 W$	$\frac{4.2E-7}{6.1E-8}$
5)	$T_F P_2 W$	$\frac{2.7E-7}{3.9E-8}$
6)	$T_T QW$	$\frac{2.0E-7}{3.0E-8}$
7)	$T_{DC} W$	$\frac{1.8E-7}{3.0E-8}$

* Only the bottom value represents core damage frequency (see Section 4.1)

Table 4.30 Class CT3 Dominant Sequences

1)	$T_F C_M U$	1.0E-7
2)	$T_F C_M P_1 U$	1.3E-8
3)	$T_E C_M U$	3.6E-9
4)	$T_I C_M U$	3.0E-9
5)	$T_F C_M C_1 U$	2.4E-9

Table 4.31 Class CT4 Dominant Sequences

1)	$T_F C_M U_H P_A$	8.7E-7
2)	$T_F C_M L_H$	7.6E-7
3)	$T_F C_M C_1 P_A$	3.9E-7
4)	$T_F C_M U P_A$	1.8E-7
5)	$T_F C_M P_1 U_H P_A$	1.1E-7
6)	$T_F C_M C_1 C_{21}$	1.1E-7
7)	$T_F C_M P_1 L_H$	9.9E-8
8)	$T_F C_M C_1 P_1 P_A$	5.1E-8
9)	$T_E C_M U_H P_A$	3.2E-8
10)	$T_E C_M C_1 C_{21}$	3.1E-8
11)	$T_I C_M U_H P_A$	2.7E-8
12)	$T_I I_M C_I$	2.5E-8
13)	$T_I I_M C_M$	2.5E-8
14)	$T_F C_M P_1 U P_A$	2.4E-8
15)	$T_E C_M C_1 P_A$	2.4E-8
16)	$T_I C_M L_H$	2.3E-8
17)	$T_F C_M C_1 P_A$	2.1E-8
18)	$T_F C_M U_H L_H$	1.9E-8

Table 4.32 Class CT2A Dominant Sequences*

1)	$T_F C_M P_1 P_2$	4.5E-7	7.3E-8
2)	$T_I C_I$	2.5E-7	4.1E-8
3)	$T_F C_M W$	1.9E-8	3.1E-9

Table 4.33 Class CT1L Dominant Sequences

1)	$S_1 U_H V$	7.9E-10
2)	$S_1 U_H X$	6.7E-10
3)	$S_1 UV(B_0)$	4.4E-10
4)	$AU_H V$	2.6E-10
5)	$AU_H V(B_0)$	2.8E-10
6)	$S_2 UV(B_0)$	7.6F-11

* Only the bottom value represents core damage frequency (see Section 4.1)

Table 4.34 Class CT2L Dominant Sequences*

1)	AW	9.8E-8	7.6E-9
2)	$S_1 W$	3.8E-8	5.6E-9
3)	$AU_H W$	4.0E-9	3.1E-10

Table 4.35 Class CT5 Dominant Sequences

1)	$S_{2xx} QW_x$	1.3E-11
2)	$S_{2xx} W_x$	3.2E-12
3)	$S_{2x} QUX$	2.1E-12
4)	$S_{2xx} C_M$	1.7E-12
5)	$S_{2x} UV(B_0)$	7.6E-13
6)	$S_{2xx} QU_H W_x$	4.9E-13

Table 4.36 Class CT6 Dominant Sequences*

1)	S_{1x}	$\frac{4.8E-9}{7.8E-10}$
2)	$S_{2xx}QW_S$	$\frac{1.9E-9}{3.1E-10}$
3)	$S_{2x}W_S$	$\frac{4.7E-10}{7.6E-11}$

* Only the bottom value represents core damage frequency (see Section 4.1)

Table 4.37 Class CT7 Dominant Sequences

1)	$S_{2xx}Q\overline{W}_x$	1.3E-7
2)	$S_{2xx}\overline{W}_x$	3.2E-8

Table 4.38 Uncertainty Measures for Transient Initiators and Manual Shutdown

Initiator	5%	Median	Mean	95%
Turbine Trip	3.10	4.77	4.93	7.30
MSIV	2.09	3.97	4.28	7.55
IORV	0.05	0.19	0.25	0.65
LOOP (National Average)	0.06	0.17	0.22	0.53
LOOP (MAAC)	0.009	0.06	0.11	0.28
Manual Shutdown	0.85	2.56	3.20	7.68

Table 4.39 Core Damage Frequency Distribution

Class	X_{05}	X_{50}	Mean	X_{95}
CT1P	3.4E-6	1.7E-5	2.9E-5	9.1E-5
CT1T	5.7E-8	4.3E-7	1.1E-6	3.7E-6
CT3	2.9E-9	4.5E-8	1.2E-7	4.6E-7
CT4	2.6E-7	1.5E-6	2.8E-6	9.4E-6
CT1L	2.7E-10	1.4E-9	2.9E-9	8.9E-9
CT2T[a]	1.1E-6 / 4.1E-7	7.0E-6 / 1.9E-6	1.7E-5 / 3.7E-6	6.3E-5 / 1.2E-5
CT2A[a]	8.0E-8 / 4.2E-9	3.9E-7 / 4.1E-8	7.0E-7 / 1.2E-7	2.4E-6 / 4.4E-7
CT2L[a]	8.0E-9 / 3.0E-10	6.3E-8 / 4.1E-9	1.4E-7 / 1.3E-8	5.1E-7 / 5.2E-8
CT5	1.7E-12	1.1E-11	2.2E-11	7.4E-11
CT6[a]	1.4E-9 / 6.8E-11	5.3E-9 / 5.3E-10	7.3E-9 / 1.2E-9	2.0E-8 / 4.3E-9
CT7[a]	2.0E-8 / 0.0	1.0E-7 / 0.0	1.7E-7 / 0.0	5.3E-7 / 0.0
Total Core Damage with				
National Average LOOP[b]	6.8E-6	2.5E-5	3.6E-5	1.1E-4
MAAC LOOP[c]	3.7E-6	1.3E-5	2.6E-5	6.0E-5

[a]In these rows, the first value is for accident sequences and the second value is for core damage (see Section 4.1).

[b]With the "National Average" for the frequency of LOOP.

[c]With the Middle Atlantic Area Council average frequency of LOOP.

Table 4.40 Importance Measures of CT1T

Cutset Element	I_B	I_{FV}
U_H	2.87E-5	1.00
U_R	9.80E-6	0.79
X	2.64E-3	0.60
Q	1.95E-6	0.50
T_F	1.23E-7	0.47
T_{DC}	7.15E-3	0.38
URHE	2.36E-6	0.19
T_I	5.28E-7	0.12
T_T	5.95E-9	0.03
V_4	7.15E-3	0.02
T_M	2.98E-9	0.01

CT1T Core Damage Frequency: 1.1E-6/yr

Table 4.41 Importance Measures of CT1P

Cutset Element	I_B	I_{FV}
T_E	1.27E-4	0.96
CM3D	4.45E-2	0.65
U_H	2.59E-4	0.32
CM2D	2.47E-3	0.26
U_R	5.07E-5	0.23
CM3B	5.35E-3	0.07
B_0	1.20E-3	0.04
T_T	1.27E-7	0.02
T_F	1.27E-7	0.02
URHE	3.78E-6	0.01
T_I	1.27E-7	ϵ

CT1P Core Damage Frequency: 3.0E-5/yr

Table 4.42 Importance Measures of CT2T

Cutset Element	I_B	I_{FV}
E	3.97E-6	0.82
I	1.14E-5	0.63
RHR	3.82E-2	0.63
Q	7.44E-6	0.59
T_F	2.02E-6	0.55
T_E	6.91E-6	0.41
CM2D	2.94E-4	0.24
CM2B	1.18E-3	0.13
T_T	1.13E-7	0.03
P_2	5.36E-5	0.02
B_0	6.53E-5	0.02
T_I	6.91E-9	ϵ
CT2T Core Damage Frequency: 3.8E-6/yr		

Table 4.43 Importance Measures of CT3

Cutset Element	I_B	I_{FV}
C_M	1.21E-2	1.0
U_H	3.07E-6	1.0
U_R	1.33E-6	1.0
T_{FA}	1.24E-8	0.95
P_2	8.13E-6	0.11
T_E	1.58E-8	0.03
T_I	1.22E-8	0.03
C_{21}	1.27E-8	0.02
CT3 Core Damage Frequency: 1.3E-7/yr		

Table 4.44 Importance Measures of CT4

Cutset Element	I_B	I_{FV}
C_M	2.82E-1	0.98
T_{FA}	2.93E-7	0.94
P_A	2.65E-6	0.60
U_H	3.36E-5	0.44
L_H	3.22E-5	0.33
C_{21}	3.22E-6	0.21
P_2	1.82E-4	0.10
U_R	1.08E-6	0.03
T_I	4.02E-7	0.03
T_E	3.79E-7	0.03
C_E	1.72E-3	0.01
C_I	2.50E-3	0.01
I_M	5.00E-6	0.01
B_0	3.58E-6	0.01
ARI	3.44E-6	0.01

CT4 Core Damage Frequency: 3.1E-6/yr

Table 4.45 Importance Measures of CT2A

Cutset Element	I_B	I_{FV}
I	5.83E-7	1.0
E	1.08E-7	0.69
C_M	7.64E-3	0.66
T_{FA}	8.30E-9	0.66
P_2	4.63E-5	0.63
C_I	4.02E-3	0.34
T_I	1.61E-7	0.34
RHR	5.12E-5	0.03

CT2A Accident Sequence Frequency: 1.2E-7/yr

Table 4.46 Importance Measures of CT1L

Cutset Element	I_B	I_{FV}
U_H	5.02E-8	0.79
S_1	2.83E-6	0.76
V_4	3.67E-5	0.44
B_0	7.90E-7	0.31
X	2.63E-5	0.27
A	2.45E-6	0.21
CM3D	1.28E-6	0.21
CM2D	6.70E-8	0.08
S_2	6.30E-8	0.03

CT1L Core Damage Frequency: 3.0E-9/yr

Table 4.47 Importance Measures of CT2L

Cutset Element	I_B	I_{FV}
RHR	2.25E-4	1.00
I	6.74E-8	1.00
E	1.41E-8	0.79
A	3.58E-5	0.58
S_1	8.39E-6	0.42

CT2L Accident Sequence Frequency: 1.4E-8/yr

Table 4.48 Importance Measures of CT5

Cutset Element	I_B	I_{FV}
S_{2xx}	1.10E-4	0.87
W_X	1.70E-7	0.79
Q	5.44E-11	0.60
S_{2x}	4.74E-7	0.13
X	8.00E-8	0.10
U_H	5.47E-11	0.10
U_R	2.49E-11	0.10
C_M	1.70E-7	0.08
CM3D	1.42E-9	0.03
B_0	7.40E-10	0.03
CM2D	5.40E-11	0.01
CT5 Core Damage Frequency: 2.3E-11/yr		

Table 4.49 Importance Measures of CT6

Cutset Element	I_B	I_{FV}
I	5.8E-9	1.0
E	1.08E-9	0.69
S_{1x}	1.62E-1	0.67
S_{2x}	6.60E-5	0.33
W_S	9.27E-7	0.33
Q	1.08E-9	0.27
CT6 Accident Sequence Frequency: 1.2E-9/yr		

Table 4.50 Importance Measures of the Total Core Damage Frequency

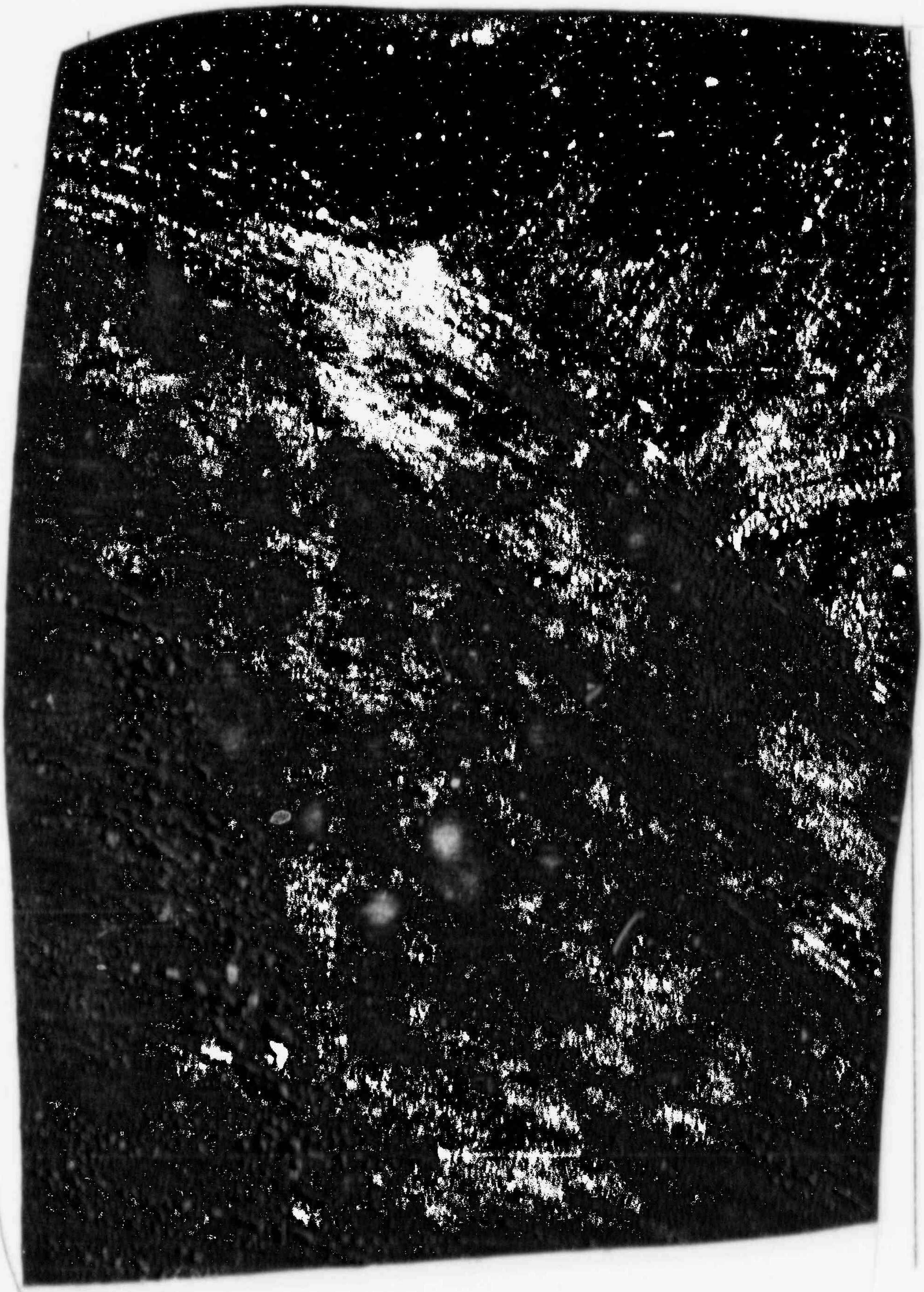
Cutset Element	I_B	I_{FV}	R_A	R_R
T_E	1.34E-4	0.80	3.8	4.90
CM3D	4.45E-2	0.51	1.2E+3	2.03
U_H	3.21E-4	0.34	9.4	1.52
CM2D	2.76E-3	0.23	7.6E+1	1.29
U_R	6.29E-5	0.16	2.6	1.18
T_F	1.04E-6	0.12	0.91	1.14
E	4.09E-6	0.08	1.0	1.09
C_M	3.01E-1	0.08	8.2E+3	1.10
Q	9.39E-6	0.07	1.2	1.08
CM2B	6.54E-3	0.07	1.8E+2	1.08
I	1.21E-5	0.07	1.3	1.07
RHR	3.85E-2	0.06	1.0E+3	1.07
T_T	4.69E-7	0.06	0.95	1.07
P_A	2.31E-6	0.04	1.0	1.04
B_0	1.27E-3	0.03	3.5E+1	1.04
L_H	9.39E-5	0.03	3.5	1.03
X	2.67E-2	0.02	7.2E+2	1.02
URHE	5.92E-6	0.02	1.1	1.02
C_{21}	3.22E-6	0.02	1.1	1.02
DG1	2.72E-5	0.02	1.7	1.02
DG2	2.72E-5	0.02	1.7	1.02
T_{DC}	7.15E-3	0.01	1.9E+2	1.01
P_2	2.61E-4	0.01	8.1	1.01
T_I	1.24E-6	0.01	1.0	1.01
C_I	6.52E-3	ϵ	1.8E+2	1.0
C_E	1.72E-3	ϵ	4.8E+1	1.0
V_4	8.85E-4	ϵ	2.5E+1	1.0
I_M	4.83E-6	ϵ	1.1	1.0
ARI	3.44E-6	ϵ	1.1	1.0
T_M	2.98E-9	ϵ	1.0	1.0

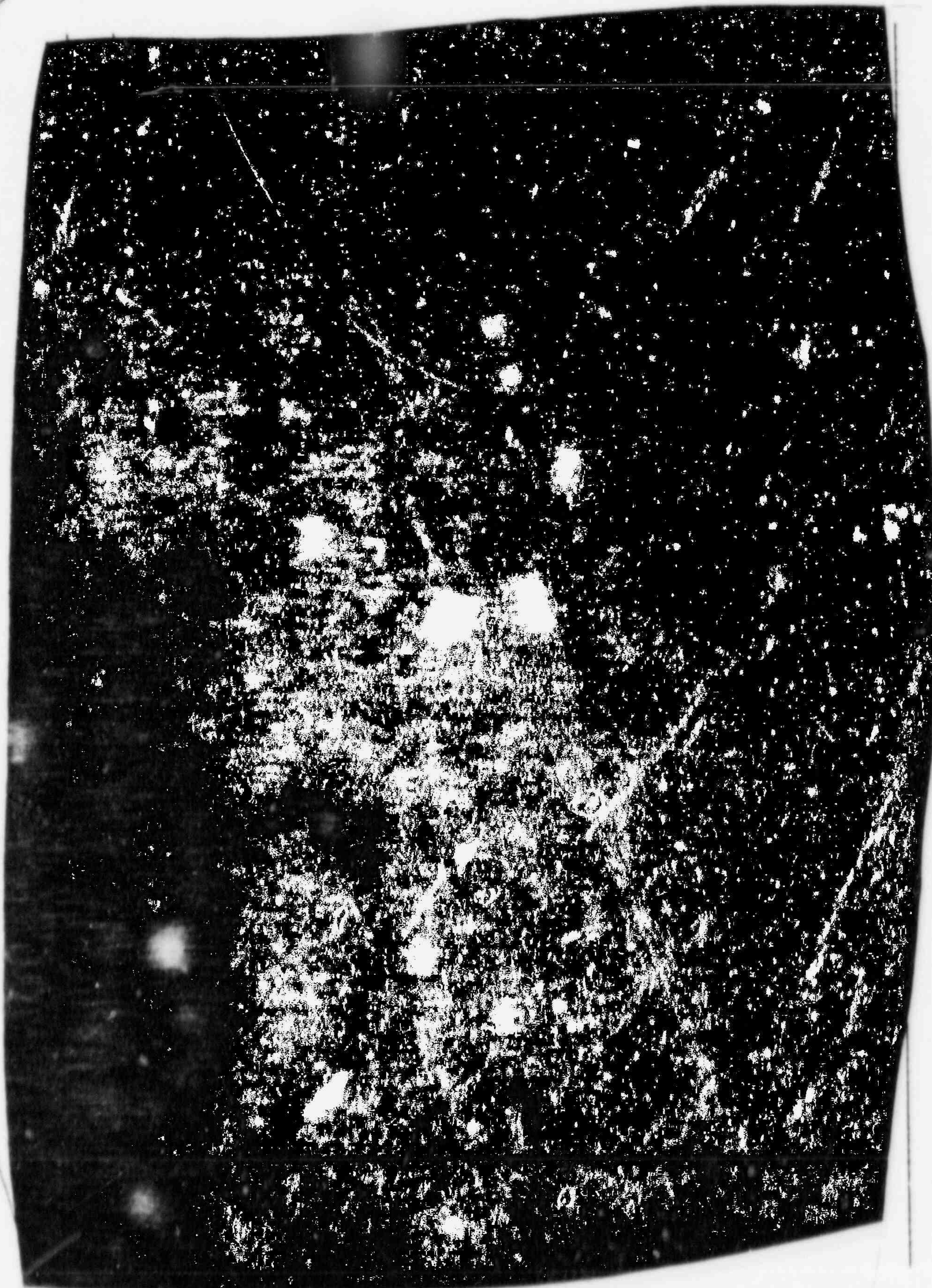
APPENDIX A

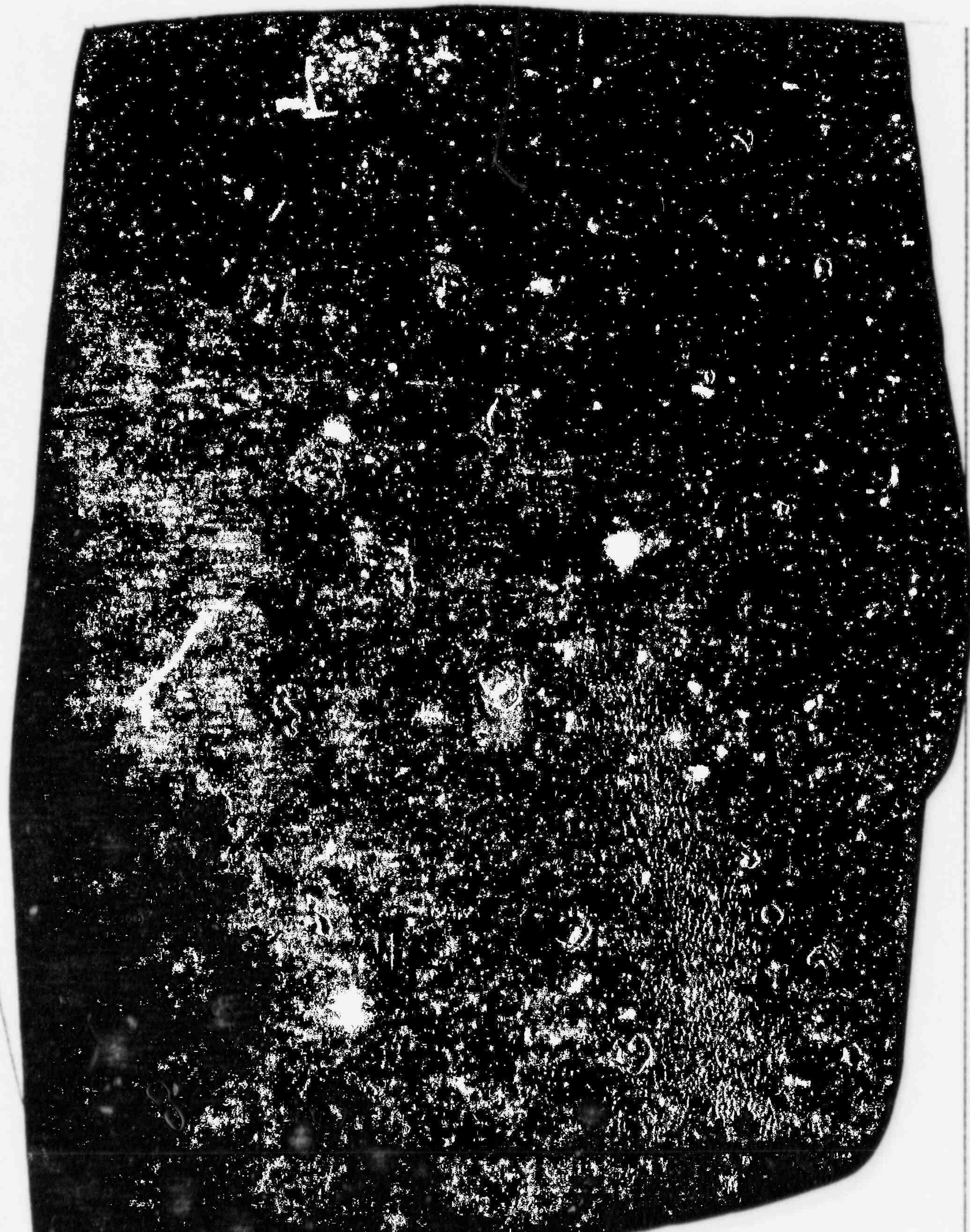
BNL MODIFICATIONS TO GESSAR-II PRA FAULT TREES

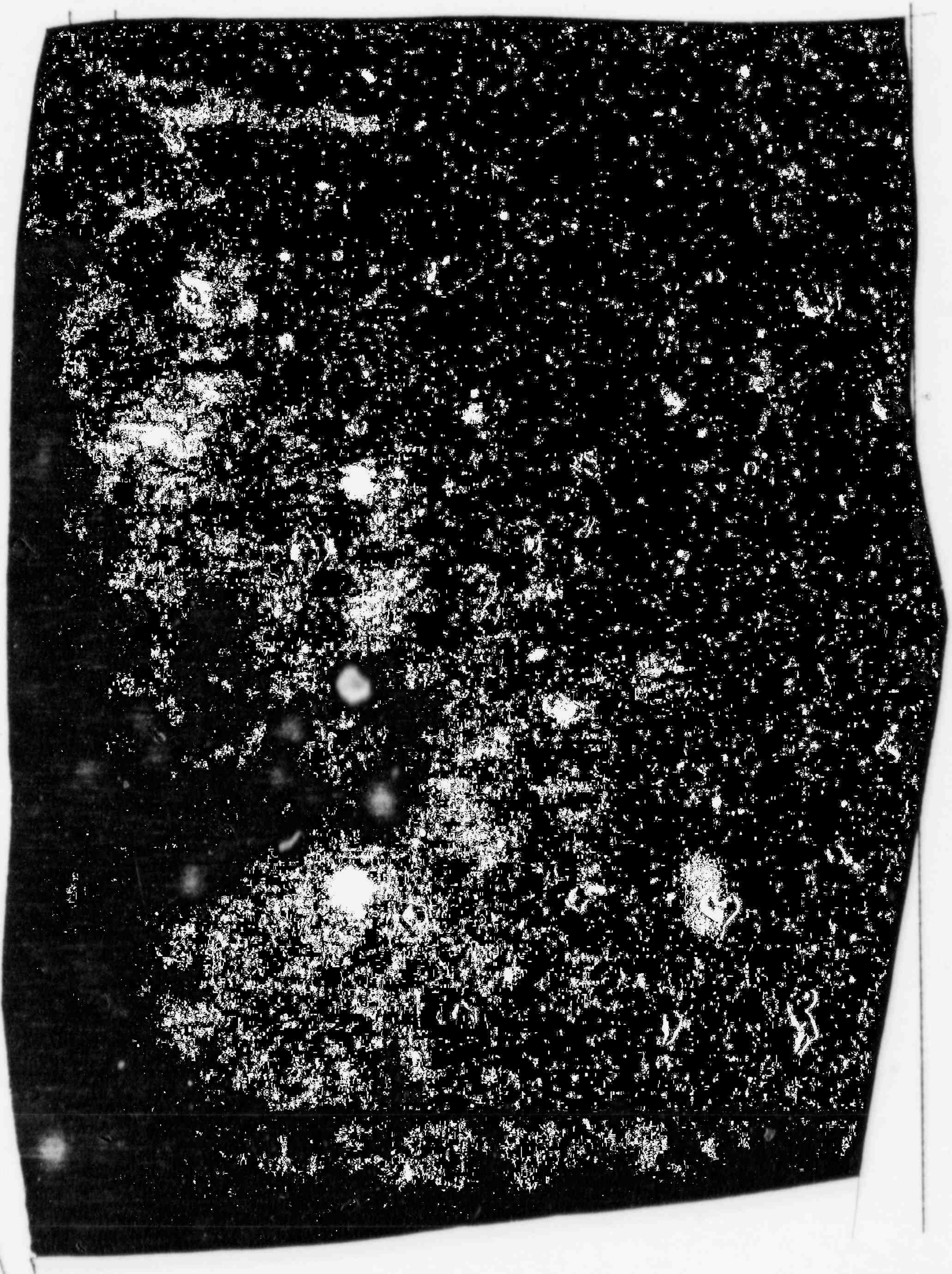
This appendix presents the details of the modifications made by BNL to the GESSAR-II PRA fault trees for the following systems:

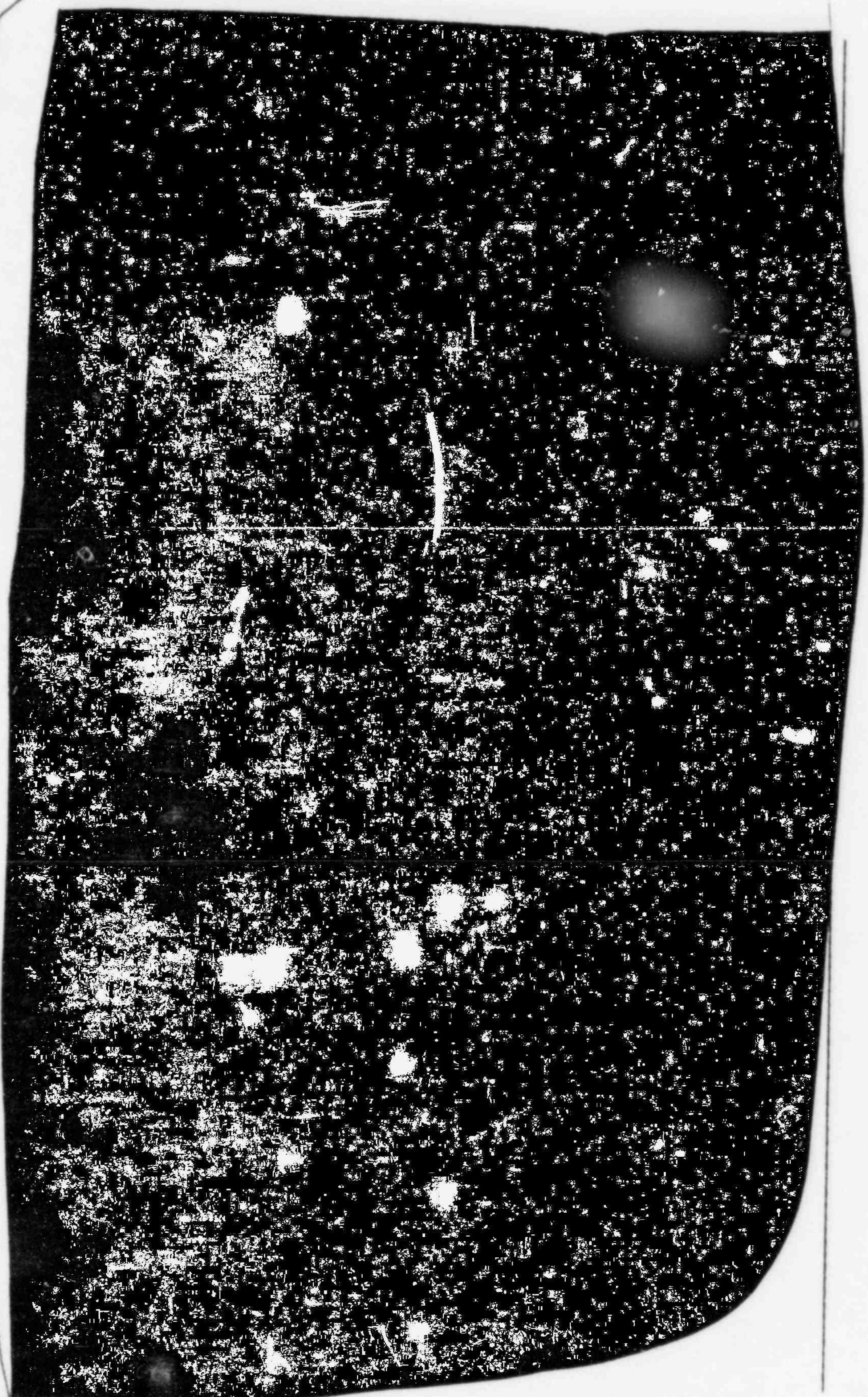
- High Pressure Core Spray System
- Reactor Core Isolation Cooling System
- Automatic Depressurization System
- Low Pressure Core Injection System
- Low Pressure Core Spray System
- Residual Heat Removal System
- Suppression Pool Makeup System
- Electric Power System
- Standby Liquid Control System.

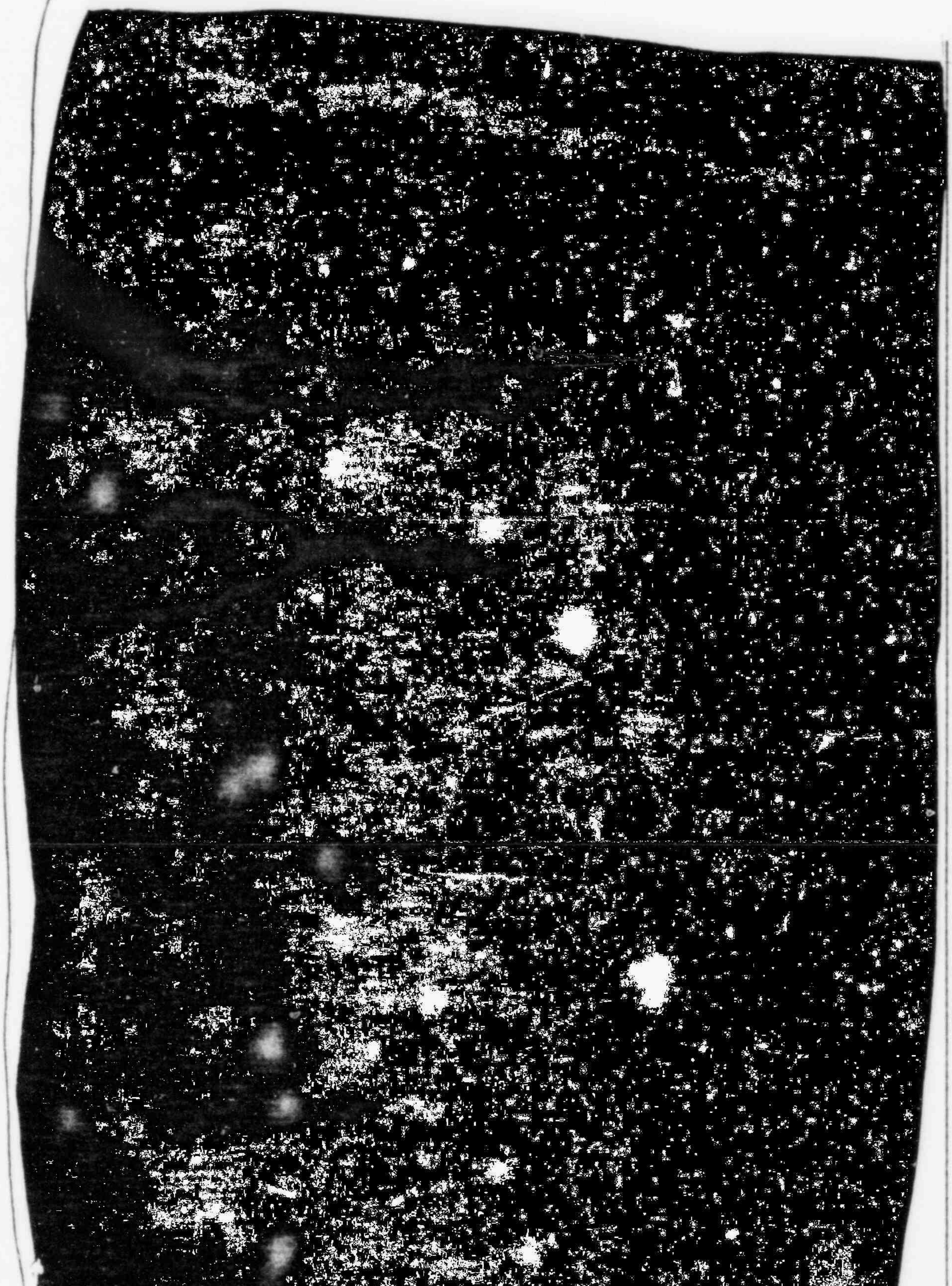












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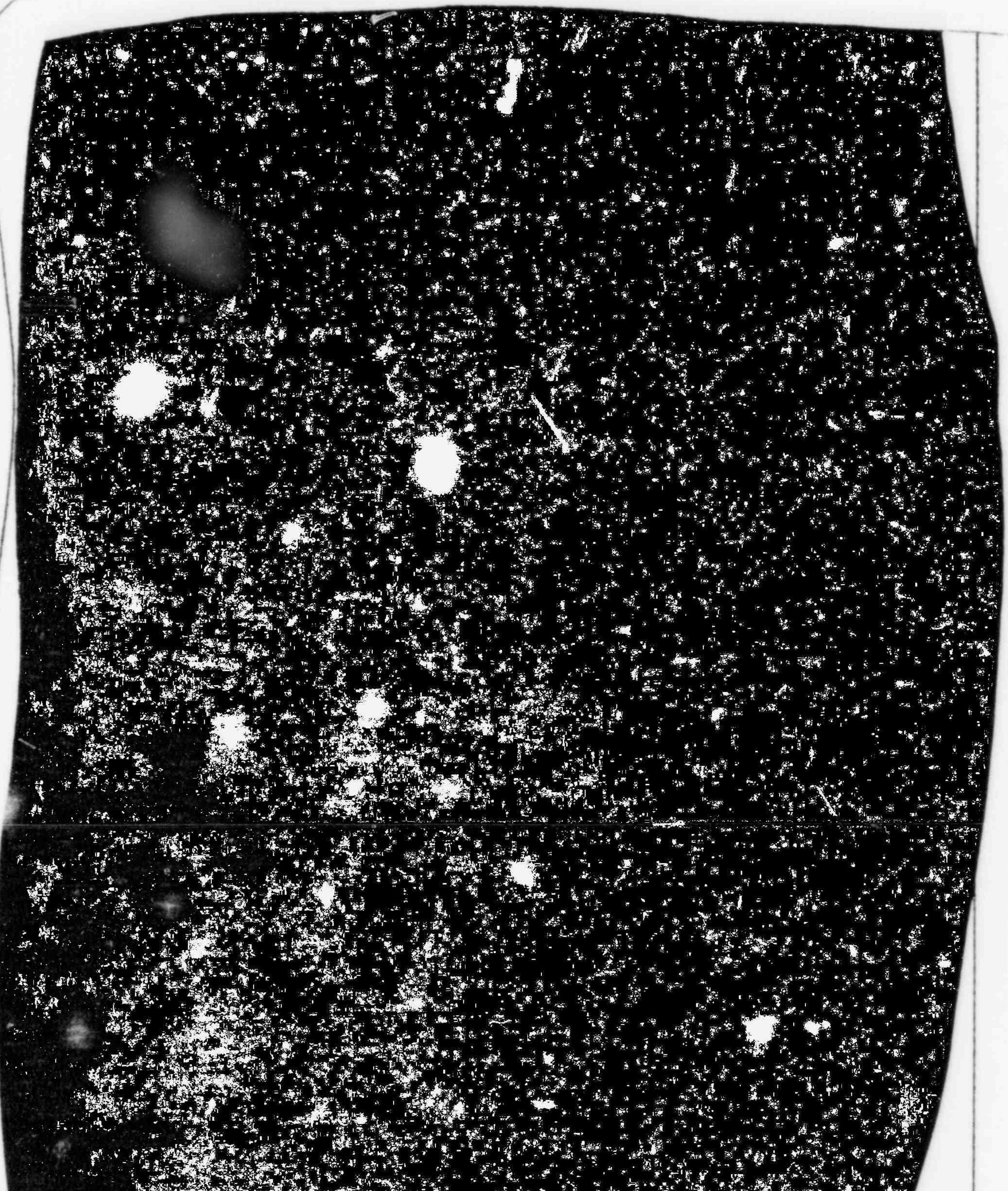

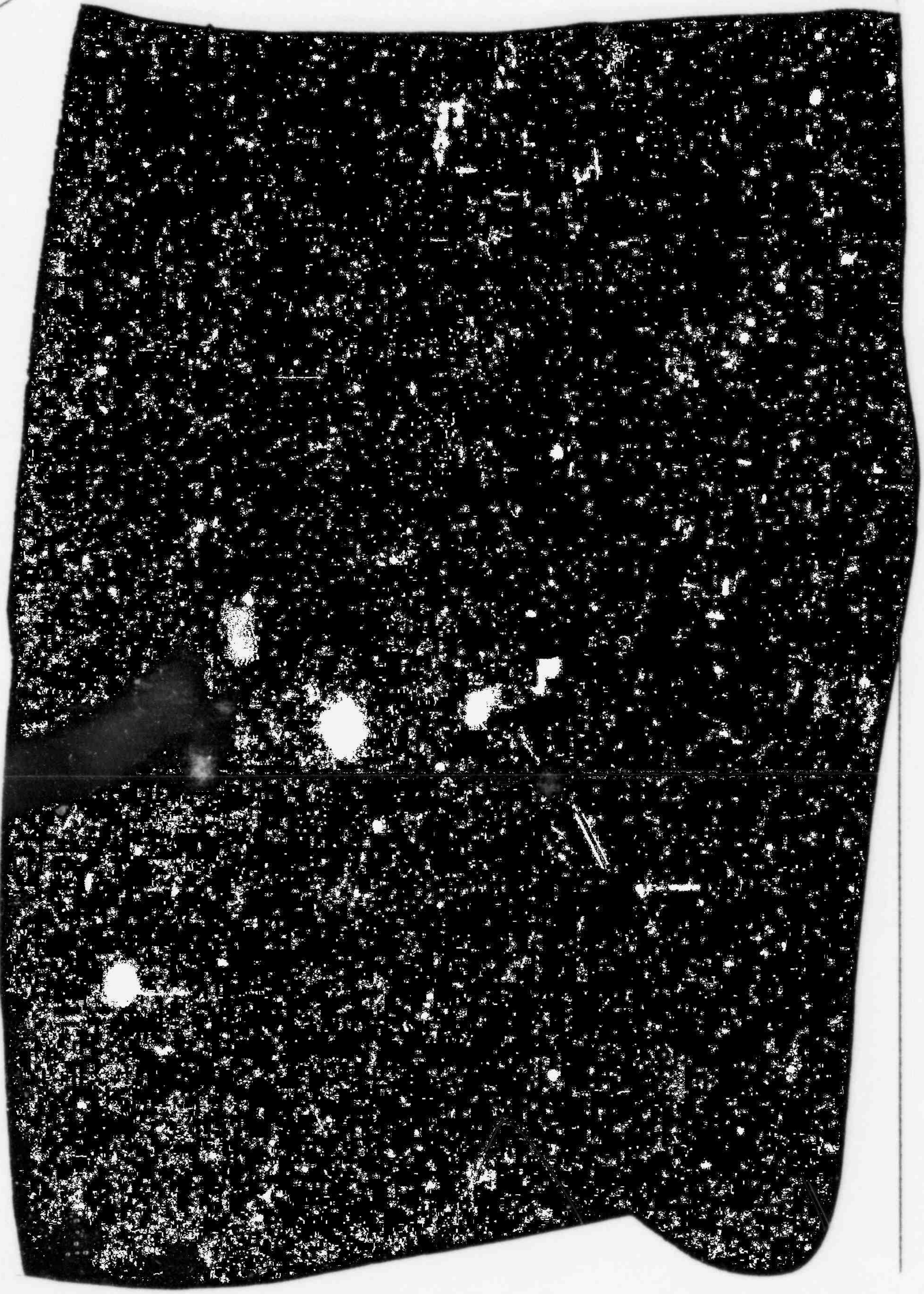

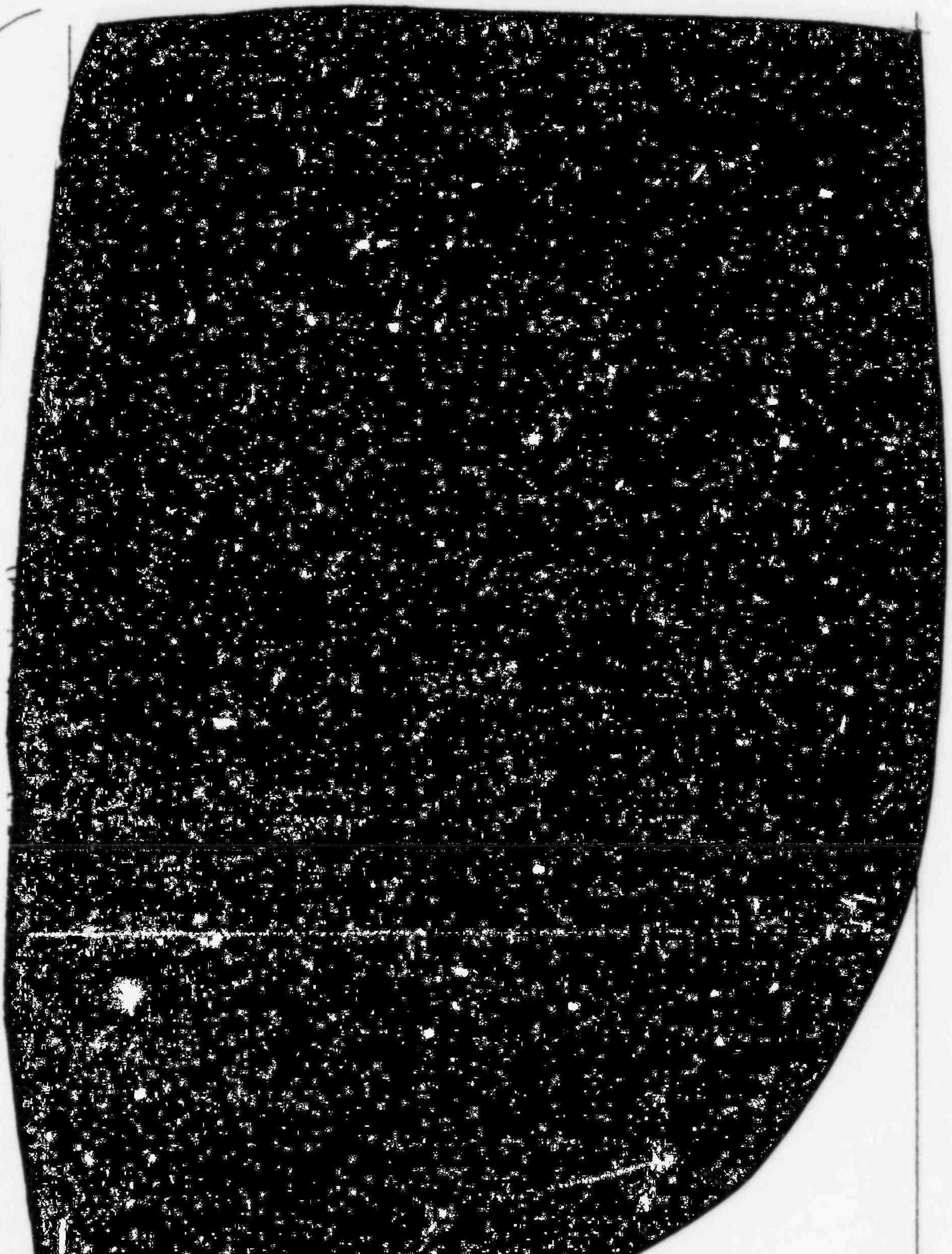

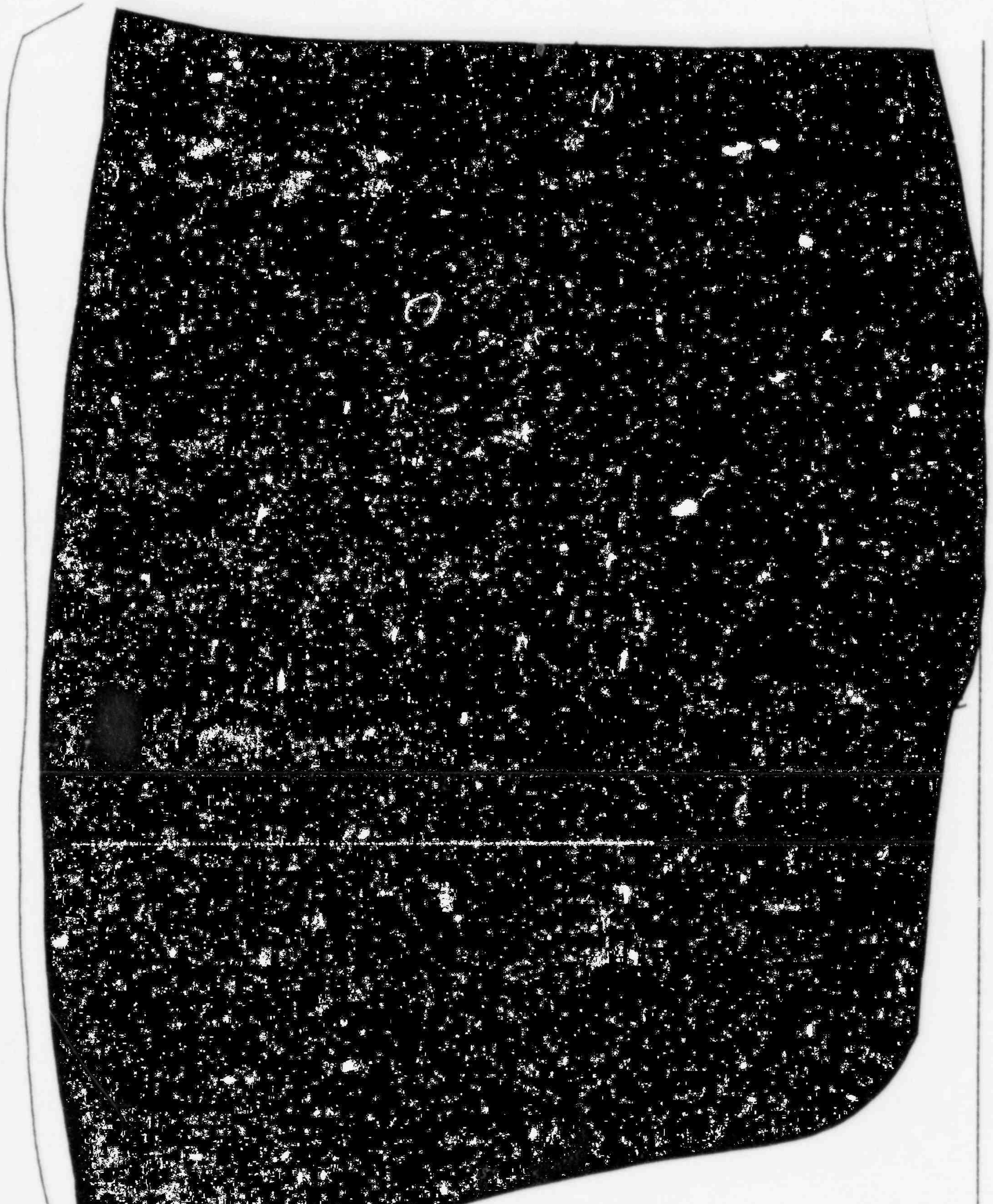

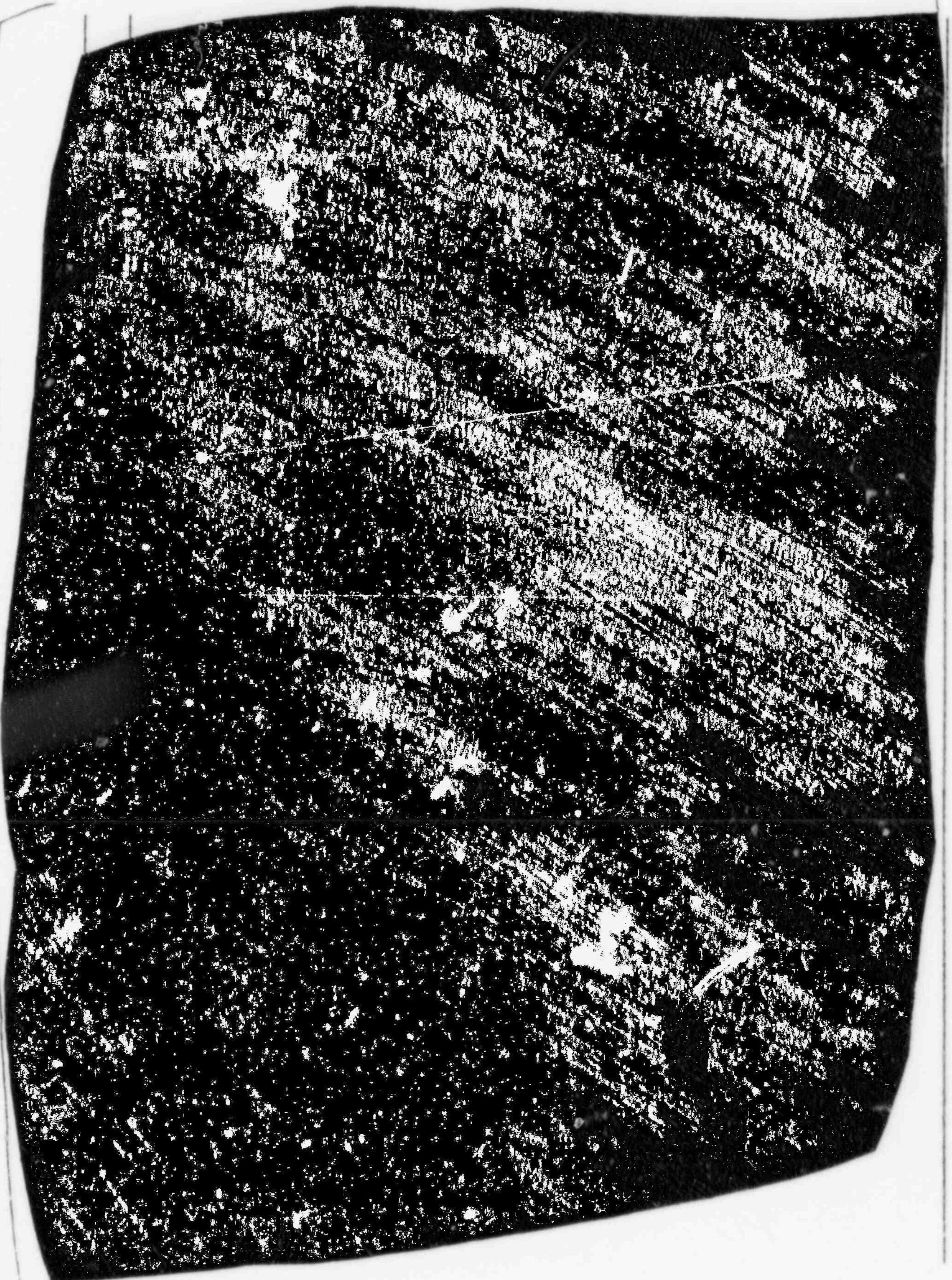

APPENDIX B

EVALUATION OF BNL MODIFIED SYSTEM FAULT TREES

This appendix presents the results obtained from the quantification of the BNL modified system fault trees; the SETS code was used for this quantification. The BNL modifications to the GESSAR-II PRA fault trees were presented in Section 3.2 and Appendix A.

Tables B.1 through B.5 present the most important cutsets generated for the HPCS, RCIC, LPCCS (LPCI and LPCS), ADS, and RHR systems, respectively. In these tables, the cutsets are given together with their description and contribution to the system unavailability. It is important to note that in some cutsets (mainly the cutsets with system maintenance events) the non-occurrence of some events was not shown for clarity of presentation.

Table B.1 Cutsets for BNL Modified HPCS System Fault Tree

	Probability	% of System Unavailability	Cutset	Description
1.	9.0E-03	23.1	HPCSRSTRT	Restarts
2.	7.2E-03	18.5	IMOOSWDWI	SWS Pump
3.	6.5E-03	16.7	HPM001DWI	HPCS Pump
4.	5.0E-03	12.8	HOOUNVHLI	Maintenance
5.	2.0E-03	5.1	IHERCFHWI	Cooler (Room Cooling)
6.	1.7E-03	4.4	HSSOOOHWI	Spargers
7.	1.6E-03	4.1	HMV004HWI	Pump Discharge Valve
8.	1.4E-03	3.6	HMD001DRI	HPCS Motor
9.	1.0E-03	2.6	EST12NSG	Static Switch Div.3
10.	7.1E-04	1.8	HCA000HBI	Cable
11.	5.0E-04	1.3	HTKCSTHFI*HOOERROR2	CST Suction Line Plugged and Failure to Transfer to SP Suction.

Table B.2 Cutsets for BNL Modified RCIC System Fault Tree

	Probability	% of System Unavailability	Cutset	Description
1.	2.1E-02	23.1	RLS068HWI	RCIC Switch
2.	2.0E-02	22.0	RCICRSTRT	Restart and Quick Trip
3.	1.1E-02	12.1	RTU001DHI	Turbine Failure
4.	1.1E-02	12.1	ROOUNVHLI	Maintenance
5.	6.5E-03	7.1	RPM001DWI	RCIC Pump
6.	2.0E-03	2.2	RHERFCHWI	Cooler (Room Cooling)
7.	1.6E-03	1.8	RMV045HPI	RCIC Steam Supply Valve
8.	1.6E-03	1.8	RMV013HPI	Pump Discharge Valve
9.	1.0E-03	1.1	RLU001DWI	Turbine Lubrication
10.	1.0E-03	1.1	RPR83BHWI	Steam Line Differential Pressure Transmitter
11.	1.0E-03	1.1	RPR84BHWI	Same as above
12.	1.0E-03	1.1	RPR83AHWI	Same as above
13.	1.0E-03	1.1	RPR83BHWI	Same as above
14.	1.0E-03	1.1	RPRPFBHWI	Steam Supply Pressure Transmitter
15.	1.0E-03	1.1	RPRPF1HWI	Same as above
16.	1.0E-03	1.1	RPRPFSHWI	Pump Suction Pressure Transmitter
17.	1.0E-03	1.1	RPR56AHWI	Turbine Exhaust Pressure Transmitter
18.	1.0E-03	1.1	RPR56BHWI	Same as above

Table B.3 Cutsets for BNL Modified LPCC System Fault Tree*

	Probability	% of System Unavailability	Cutset	Description
1.	2.0E-05	66.7	DLHU005	Miscalibration of Pressure Transmitters on Miniflow Valves
2.	1.0E-06	3.3	ZSP000DWI	Suppression Pool Water Unavailable Due to High Temperature
3.	1.0E-06	3.3	EST12NSE*EST12NSF	Static Switches (Div. 1 and 2 EPS)
4.	9.8E-07	3.3	DTMABC*LPM001DRI	All LPCI LOOPS in Maintenance and LPCS Pump Failures
5.	3.0E-07	1.0	DPPSUPHFI	Suppression Pool Water Unavailable Due to Rupture

*It includes failure of LPCI and LPCS systems.

Table B.4 Cutsets for BNL Modified ADS Fault Tree

	Probability	% of System Unavailability	Cutset and Description*
1.	2.0E-06	8.0	AHUDEP*AHU003
2.	2.0E-06	8.0	AHUDEP*DLHU001
3.	9.0E-07	3.6	AHUDEP*DTL91FHWI*LTL91EHWI
4.	9.0E-07	3.6	AHUDEP*APX95BHWI*LTL91EHWI
5.	9.0E-07	3.6	AHUDEP*DTL91BHWI*LTL91EHWI
6.	9.0E-07	3.6	AHUDEP*DTL91FHWI*APX95AHWI
7.	9.0E-07	3.6	AHUDPE*APX95BHWI*APX95AHWI
8.	9.0E-07	3.6	AHUDEP*DTL91BHWI*APX95AHWI
9.	9.0E-07	3.6	AHUDPE*DTL91FHWI*LTL91AHWI
10.	9.0E-07	3.6	AHUDEP*APX95BHWI*LTL91AHWI
11.	9.0E-07	3.6	AHUDEP*DTL91BHWI*LTL91AHWI
12.	6.0E-07	2.4	AHUDEP*DPP23HHFI*LTL91EHWI
13.	6.0E-07	2.4	AHUDEP*DPP23HHFI*APX95AHWI
14.	6.0E-07	2.4	AHUDEP*DPP23HHFI*LTL91AHWI
15.	6.0E-07	2.4	AHUDEP*DTL91FHWI*LPP13HHFI
16.	6.0E-07	2.4	AHUDEP*APX95BHWI*LPP13HHFI
17.	6.0E-07	2.4	AHUDEP*DTL91BHWI*LPP13HHFI
18.	4.0E-07	1.6	AHUDEP*DPP23HHFI*LPP13HHFI
19.	3.0E-07	1.2	AHUDEP*EST12NSF*LTL91EHWI
20.	3.0E-07	1.2	AHUDEP*EST12NSF*APX95AHWI
21.	3.0E-07	1.2	AHUDEP*EST12NSF*LTL91AHWI
22.	3.0E-07	1.2	AHUDEP*DTL91FHWI*EST12NSE
23.	3.0E-07	1.2	AHUDEP*APX95BHWI*EST12NSE
24.	3.0E-07	1.2	AHUDEP*DTL91BHWI*EST12NSE

*AHUDEP = Operator failure to initiate ADS and non-ADS valves, given failure of ADS auto initiation.

AHU003 = Miscalibration of LPCCS pump discharge pressure transmitters.

DLHU001 = Miscalibration of transmitters (reactor level).

DTL---HWI
LTL---HWI } = Transmitters
APR---HWI

DPP---HFI
} = Instrument lines
LPP---HFI

EST12NSE
} = Static switches
EST12NSF

Table B.5 Cutsets for BNL Modified RHR System Fault Tree*

	Probability	% of System Unavailability	Cutset	Description
1.	1.4E-04	26.4	DTMAB	Both RHR loops in maintenance
2.	4.6E-05	8.7	DPMO2ADRI*DPMO2BDRI	Both RHR pumps
3.	4.0E-05	7.6	SPDHUU002	Operator fails to follow procedures
4.	2.0E-05	3.8	SPDHU001	Operator initiation
5.	1.4E-05	2.6	DTMA*DPMO2BDRI	Loop A in maintenance and pump loop B
6.	1.4E-05	2.6	DTMB*DPMO2ADRI	Loop B in maintenance and pump loop A
7.	1.4E-05	2.6	DCSO2AHWI*DPMO2BDRI	Coolers on loop A and pump loop B
8.	1.4E-05	2.6	DCSO2BHWI*DPMO2ADRI	Coolers in loop B and pump loop A
9.	6.8E-06	1.3	DLUO2AHWS*DPMO2BDRS	Lubrication pump A and pump loop B
10.	6.8E-06	1.3	DLUO2BHWI*DPMO2ADRI	Lubrication pump B and pump loop A

*RHR system operating in the suppression pool cooling mode. Note that recovery of the RHR system is not included in this fault tree, even though it is addressed in quantification of the accident sequences.

APPENDIX C

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Table C.1 Results of BNL Review of the First Stage for Accident Classes CT2T, CT2A, CT2L and CT6*

	Classes			
	CT2T	CT2A	CT2L	CT6
Turbine Trip (T_T)	6.7E-7	---		
Isolation (T_F)	1.4E-5	4.9E-7		
IORV (T_I)	2.1E-7	2.5E-7		
Reactor Shutdown (T_M)	8.6E-8	---		
Loss of Offsite Power (T_E)	3.5E-6**	1.2E-8**		
	1.7E-6***	5.8E-9***		
Loss of DC Power (T_{DC})	1.9E-7	---		
Large LOCA (A)			1.0E-7	---
Intermediate LOCA (S_1)			4.0E-8	4.8E-9
Small LOCA (S_2)			9.2E-10	2.6E-9

*Results for all classes are presented in Tables 4.5 through 4.9 in this report.
**National Average LOOP Frequency
***MAAC LOOP Frequency

Table C.2 Basis for Assessed Probabilities for the Containment Event Tree (Figure C.1)

Function	Initiator	Probability	Branch in ET	Description/Comment
RHR Recovery	T_T, T_F, T_I, T_M • Class CT2T	0.85	B	a) The probability of failing to restore RHR prior to loss of containment integrity given that it has not been restored in the first stage. Probability of repair from 24 hrs to 27 hrs[4] with a MTTR equal to 19 hrs.
	ATWS (T_F, T_I, T_E) • Class CT2A	1.0	B	b) The dominant accident sequences for ATWS (see Table 4.32 of this report) do not have failure of RHR. BNL review uses this value because no deterministic calculations were performed to assess the pressure/temperature history in the containment. This value is similar to the one used in the GESSAR-II PRA.
	T_E • Class CT2T	1.0	B	c) A sequence accounting for about 34% of the Class CT2T for a LOOP initiator (the 6th sequence in Figure 4.7.8) includes in its first stage the failure to recover offsite power and the failure to recover the DG1 and DG2 (common mode failure) in 24 hrs; in this sequence core cooling is via the HPCS injection (DG3). So, no credit was given for failure to recover power (onsite or offsite) from 24 hrs to 27 hrs.

Table C.2 (Continued)

Function	Initiator	Probability	Branch in ET	Description/Comment
RHR Recovery	T_E • Class CT2T	1.0	B	d) A sequence accounting for about 21% of the Class CT2T for a LOOP initiator (the 6th sequence in Figure 4.11) is very similar to the one in item c) above; the only difference is that there is a common mode failure of the batteries instead of two DGs.
	T_E • Class CT2T	0.85	B	e) For all the other LOOP sequences the same comment as in item a) applies.
	T_{DC} • Class CT2T	1.0	B	f) Since for 85% of the Class CT2T for a T_{DC} the recovery of DC power (Divisions 1 and 2) and initiation of RHR without DC power is included in the first stage (for 24 hrs) no credit was given for the time interval between 24 and 27 hrs.
	A • Class CT2L	0.24	B	g) Same as item a). However, since no credit for RHR recovery was given in the first stage, the probability of recovery in 27 hrs is given by: $\exp(-27/19) = 0.24$
	S_1 and S_2 • Class CT2L	0.85	B	h) Same as item a).

Table C.2 (Continued)

Function	Initiator	Probability	Branch in ET	Description/Comment
Injection from Suppression Pool	All	0.5	C	i) The probability that erroneous RPV level signal due to adverse environment will trip the high pressure ECCS; for all dominant sequences the high pressure ECCS is the core cooling medium. This value is used for the cases where the containment integrity is not lost.
	All	0.75	D	j) Same as above, but with loss of containment integrity.
Maintain Injection from CST	T_T, T_F, T_I, T_M A, S_1, S_2	0.1 0.05 0.2	E F G	k) The probability that core cooling can be maintained with CRD or condensate booster pumps: • 0.1 used for the case where containment integrity is maintained, and injection from suppression pool fails. • 0.2 used for the case where containment integrity is lost, and injection from suppression pool fails. • 0.05 used for the case where containment integrity is lost, but injection from suppression pool does not fail.

Table C.2 (Continued)

Function	Initiator	Probability	Branch in ET	Description/Comment
Maintain Injection from CST	T_E	1.0	F,G	l) For cases c) and d) above, where offsite power is not recovered; the CRD and condensate booster pumps require offsite power.
	T_E	0.1 0.05 0.2	E F G	m) Same as item k) above.
Drywell Leakage	All	1.0E-4	H	n) The probability of excessive drywell leakage.

Table C.3 Core Damage Frequencies for Accident Classes CT2T, CT2L, CT2A and CT6

	Classes			
	CT2T	CT2A	CT2L	CT6
Turbine Trip	9.8E-8	---	---	
Isolation	2.0E-6	8.0E-8	---	
IORV	3.1E-8	4.1E-8	---	
Reactor Shutdown	1.2E-8	---	---	
Loss of Offsite Power	1.6E-6*	1.9E-9	---	
	7.8E-7**		---	
Loss of DC Power	3.0E-8	---	---	
Large LOCA	---		7.7E-9	---
Intermediate LOCA	---		5.8E-9	7.8E-10
Small LOCA	---		1.3E-10	4.2E-10
CLASS TOTAL	3.8E-6* 2.9E-6**	1.2E-7	1.4E-8	1.2E-9

*National Average LOOP Frequency
**MAAC LOOP Frequency

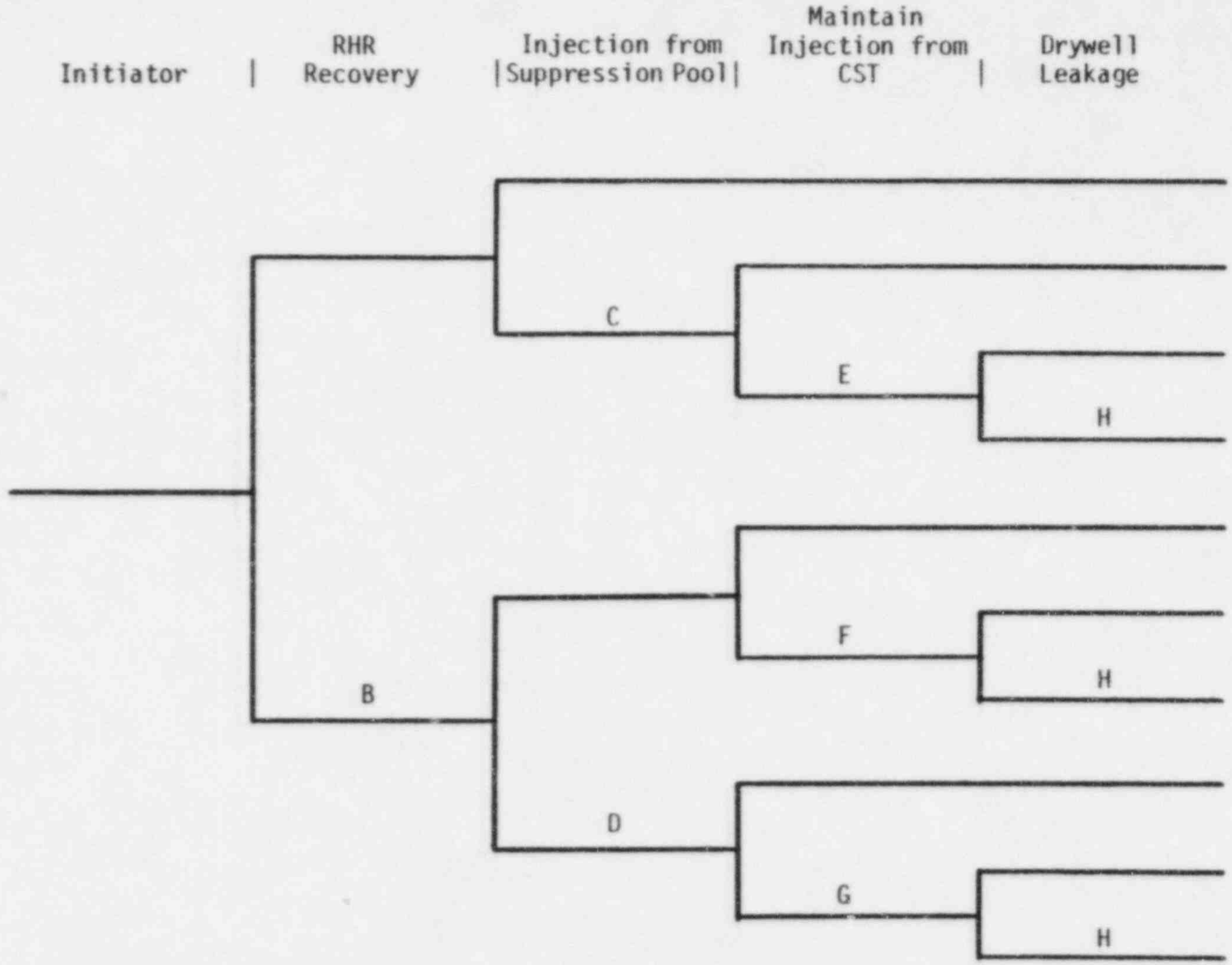

Figure C.1 BNL revised containment event tree for class 2 type accident sequences.

References

1. GESSAR-II BWR/6 Nuclear Island Design, Section 15.D.

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

3. Limerick PRA, Philadelphia Electric Co., 1982.

4. Personal communication with W. T. Pratt (BNL), 1984.

APPENDIX D

UNCERTAINTY ANALYSIS

This appendix presents the details of the functions used in the uncertainty analysis performed in this revision of the GESSAR-II PRA. As discussed in Section 4.6, the uncertainties in each accident class were quantified, with the SAMPLE code, using the most important accident sequences in each class (contributing 95% of the class frequency). The most important accident sequences in each class, as given in Tables 4.27 through 4.37 in the main text, were transformed to SAMPLE functions as required by the SAMPLE code. These functions and the numerical data for all the variables are given in Table 4.B.1 and 4.B.2, respectively.

The following notes are required for better understanding of the SAMPLE functions as compared to the accident sequences:

1. To account for the unavailability of some systems being accident sequence dependent, it was assumed that the ratio of the various unavailabilities will remain constant and equal to the ratio of the base case. For example, the unavailability of the Q function depends on the accident initiator. For the uncertainty calculations, it was assumed that the unavailability of the Q function given an MSIV closure initiator will be the "independent" variable (that is characterized by a distribution) and that the unavailabilities of the Q function given other initiators will be a multiple of the former. For example,

$$Q_i = r_i \cdot Q_F,$$

where r_i is the ratio of the point estimates Q_i to Q_F, that is,

$$r_i = \frac{Q_i}{Q_F} .$$

The numerical values of the various r-coefficients are directly inserted in the expressions given in Table D.1.

2. Since, as explained in Section 4.6, no uncertainty was assumed in the offsite power and diesel recovery probabilities, the coefficients used in the SAMPLE functions given in Table D.1 also include the numerical values of these probabilities.

3. To properly account for the uncertainties in sequences where dependences between systems are important, the SAMPLE functions represent the sequences by using the minimal cutsets at the component level. This representation is used for sequences with the failures of the RCIC and ADS (sequences of TUX type).

Table D.1 SAMPLE Functions

Class CT1P

```
SAMPLE= ( X(5)+X(6)*(X(1)+X(2)+X(3)))*(1.0-X(12))*
        ( 1.650E-1* (1.0-0.98791*X(8))* X(29) +
        2.2571E-1*(1.0-X(29))*X(30)*X(7)*(1.0-0.98791*X(8)) +
        5.0680E-1* X(29) * X(8) +
        7.8665E-1* (1.0-X(29))*X(30)*X(7)*X(8) +
        (1.5522*X(7)*(1.0-X(54))*X(53)*
        (0.25*(1.0-X(8))*(1.0-X(55)) +
        (X(8)*(1.0-X(55))+X(55))) +
        1.4897E-01*(1.0-X(29))*(1.0-X(30))*(1.0-9.8791E-01*X(8))*
        X(7)*X(44)*X(45) ))
```

Class CT1T

```
SAMPLE=((X(1)+4.8276E-2*X(2)+2.4138E-2*X(4))*X(11)*(1.0-X(12)) +
       (1.0-X(13))*X(3)) * X(28) * X(7)*
       (5.5832E-1*X(8)*X(10) +
       X(14) + X(15)* (X(16)+X(17)+X(18)+X(19)*X(20)) +
       X(16)* (X(21)+X(22)*X(20)+X(23)*X(20))+
       X(24)* ((X(25)+X(19)+X(26))*X(20)+X(18)) +
       X(22)* (X(18)+ (X(25)+X(19)+X(26)+X(17))*X(20)) +
       X(25)* (X(27)+X(23))* X(20) +
       X(19) * X(20)* (X(27)+X(21)+X(23)) +
       X(27)* (X(18)+X(26)*X(20)) +
       X(21) * X(18)) +
       (1.0-X(13))*X(3)*X(7)*X(8)*X(9) +
       X(56)*X(7)*(X(8)*(1.0-X(55)) + X(55))
```

Class CT3

```
SAMPLE=X(38)*X(32)*X(7)*X(8)*(1.0-5.5556E-1*X(35))*(1.0-X(34))*
       (1.0-X(33))* (8.5797E-1 +7.0497E-1*X(12)+1.1023E-1*X(36)) +
    1.2662*(X(5)+X(6)*(X(38)+X(3)))*X(32)*X(7)*X(8)*(1.0-X(35))*
         (1.0-X(34))*(1.0-X(33)) +
       0.975*X(3)*X(32)*X(7)*X(8)*(1.0-7.6389E-1*X(35))*
          (1.0-X(34))
```

Table D.1 (Continued)

Class CT4

```
SAMPLE= X(39)*(1.0-5.5556E-1*X(35))*(X(32)*(0.075*(1.0-X(33))*
        (8.7997E-01* ( 4.7619E-1*X(7)*X(34)*(1.0-X(8)) +
          X(39)*(1.0-X(7)) + X(7)*X(8)*X(34) +
          X(7)*X(39)*(1.0-X(8))*(1.0-4.7619E-1*X(34))) +
        72.509*X(12)*( 4.7619E-1*X(7)*X(34)*(1.0-X(8)) +
          X(39)*(1.0-X(7))+ X(7)*X(8)*X(34))) +
        1.2526E-1*X(36)*(1.0-X(33))*( 2.8656E-1*X(34)*(1.0-X(7))+
          23.613*X(12)*X(34)*(1.0-X(7)) + 4.1903E-1*X(34)*X(7)*
          (1.0-X(8)))+ 6.3158E-03 * X(36)) +
        8.5797E-01*X(37)*X(51)*(1.0-X(33))*(1.0-X(32))*
          (4.7619E-1*X(7)*X(34)*(1.0-X(8))+X(39)*(1.0-X(7)))) +
        (X(5)+X(6)*(X(39)+X(3)))*X(32)*(1.0-X(35))*
          (0.073*X(36) + 0.927*(1.0-X(33))*( 6.5041E-01*X(7)*
          X(34)*(1.0-X(8)) + 8.7997E-01*X(39)*(1.0-1.5522*X(7)))+
          8.9182E-2*X(36)*X(34)*(1.0-X(33))*(1.0-1.5522*X(7)))+
        X(3)*( X(13)* (X(40)+ (1.0-X(40))*X(32)) +
          0.975*X(32)*(1.0-0.1*X(40))*(1.0-X(13)) *
          (1.0-7.6389E-1*X(35))*(4.7619E-1*X(34)*X(7)*
          (1.0-X(8)) + (1.0-X(7))*X(39)))
```

Class CT1L

```
SAMPLE= X(42)*((1.0-X(6))*(X(7)*X(9) + X(7)*X(10)*X(28)) +
          X(6)*(X(29) + 1.5522*X(7)*(1.0-X(29))*
          (X(30) + 26.667*X(9)))) +
          X(43)*((1.0-X(6))*X(52)*X(28)*X(7) *
         (5.5832E-1*X(8)*X(10) +
         X(14) + X(15)* (X(16)+X(17)+X(18)+X(19)*X(20)) +
         X(16)* (X(21)+X(22)*X(20)+X(23)*X(20))+
         X(24)* ((X(25)+X(19)+X(26))*X(20)+X(18)) +
         X(22)* (X(14)+ (X(25)+X(19)+X(26)+X(17))*X(20)) +
         X(25)* (X(27)+X(23))* X(20) +
         X(19) * X(20)* (X(27)+X(21)+X(23)) +
         X(27)* (X(18)+X(26)*X(20)) +
         X(21) * X(18)) +
          1.650E-01*X(6)*X(29)*(1.0-X(8))) +
          X(41)*((1.0-1.955*X(6))*X(7)*X(9) +
          1.955*X(6)*(X(29)+1.5522*(1.0-X(29))*X(7)*
          (X(30)+26.667*X(9))))
```

Table D.1 (Continued)

Class CT2T

```
SAMPLE=(X(1)*(1.8017E-1*X(11)*X(31)*(1.0-X(7)*X(8))*(1.0-X(12)) +
              X(12)*(1.0-1.1724*X(11))*X(31)) +
        3.871E-01* (X(5)+X(6)*(X(1)+X(2)+X(3)))* X(11)*X(31)*
              (1.0-X(12))* (1.0-X(29))* (1.0-1.5522*X(7)) +
        X(2)* (2.3948E-03*X(11)*X(31)*(1.0-X(7)*X(8))*(1.0-X(12)) +
              X(12)*X(31)*(1.0-3.7931E-01*X(11)))) *
        (0.854* (X(60)*X(61)+ (1.0-X(60))*0.25*X(61)) +
        4.867E-2* X(60)*X(61)) +
        1.8E-3*(X(5)+X(6)*(X(1)+X(2)+X(3)))*X(30)*X(60)*
        (1.0-X(29))*(1.0-X(12))*(1.0-1.522*X(7)) +
        7.245E-3*(X(5)+X(6)*(X(1)+X(2)+X(3)))*(1.0-1.522*X(7))*
        (1.0-X(54))*X(53)*X(60)
```

Class CT2A

```
SAMPLE=(X(38)*X(32)*(1.0-5.5556E-01*X(35))*(1.0-X(33))*
        (3.0922*X(12) + 3.6006 *(1.0-X(7))*(1.0-X(39))*X(31)) +
         0.1 * X(3)*(1.0-X(13))*X(40) ) *
         (X(60)*X(61) + (1.0-X(60))*0.25*X(61))
```

Class CT2L

```
SAMPLE=7.7579E+00*X(41)*X(31)*(1.0-1.955*X(6))*
        ((1.0-X(7))+X(7)*(1.0-X(9)))*
        (0.242*(X(60)*X(61)+(1.0-X(60))*0.25*X(61)) +
        2.527E-01*X(60)*X(61)) +
        X(42)*(1.0-X(6))*(1.0-X(7))*X(31)*
        (0.854*(X(60)*X(61)+(1.0-X(60))*0.25*X(61))+
        4.867E-02*X(60)*X(61))
```

Class CT5

```
SAMPLE=X(48)*(X(32)+(1.0-X(6))*X(50)*( X(52)*
        ((1.0-X(7))+X(7)*(1.0-X(8))) +
         (1.0-X(52)))) +
         X(47)*((1.0-X(6))*X(52)*X(28)*X(7) *
        (5.5832E-1*X(8)*X(10) +
        X(14) + X(15)* (X(16)+X(17)+X(18)+X(19)*X(20)) +
        X(16)* (X(21)+X(22)*X(20)+X(23)*X(20))+
        X(24)* ((X(25)+X(19)+X(26))*X(20)+X(18)) +
        X(22)* (X(18)+ (X(25)+X(19)+X(26)+X(17))*X(20)) +
        X(25)* (X(27)+X(23))* X(20) +
        X(19) * X(20)* (X(27)+X(21)+X(23)) +
        X(27)* (X(18)+X(26)*X(20)) +
        X(21) * X(18)) +
        X(6)*(1.650E-1*X(29)*(1.0-X(8))+X(29)*X(8)+
        2.5611E-1*(1.0-X(29))*X(7)*(1.0-X(8))*X(30)))
```

Table D.1 (Continued)

Class CT6

```
SAMPLE=( X(46)+X(47)*(1.0-X(6))*X(49)*
        (X(52)*(1.0-X(7)) + (1.0-X(52))) )*
        (X(60)*X(61)+ (1.0-X(60))*0.25*X(61))
```

Class CT7

```
SAMPLE=X(48)*(1.0-X(6))*(1.0-X(50))*(X(52)*(1.0-X(7))+
        (1.0-X(52)))
```

Table D.2   Sample Code Input Variables

Sample Code Variable	Event Designator	Event	Median	Error Factor
X(1)	T_F	MSIV Closure	3.97E+0	1.9
X(2)	T_T	Turbine Trip	4.77E+0	1.5
X(3)	T_I	IORV	1.87E-1	3.5
X(4)	T_M	Manual Scram	2.56E+0	3.0
X(5)	T_E	Loss of Offsite Power	1.74E-1	3.0
X(6)	B	LOP Due to Grid Instability	6.20E-4	5.0
X(7)	U_H	HPCS	3.14E-2	3.0
X(8)	U_R	RCIC	7.28E-2	3.0
X(9)	V_4	LPCCS (LPCI and LPCS)	1.86E-5	5.0
X(10)	X	ADS	2.04E-4	3.0
X(11)	Q	FW Recovery (Transients)	1.80E-1	5.0
X(12)	P_2	S/RV Reclosure	9.83E-4	5.0
X(13)	I_M	Multiple IORV	8.00E-3	3.0
X(14)	DLHU001	Miscalibration of Level Transmitters	7.51E-6	10.0
X(15)	DTL91BHWI	Transmitter B21-N091B	2.40E-3	3.0
X(16)	LTL91AHWI	Transmitter B21-N091A	2.40E-3	3.0
X(17)	LACTU091A	Analog Comparator Unit B21-N091A	1.04E-3	3.0
X(18)	EST12NSE	Static Switch (EPS Division 1)	8.00E-4	3.0
X(19)	LPP13HHFI	Instrument Line Rupture or Leakage	1.60E-3	3.0
X(20)	ROERROR1	Operator Failure to Initiate RCIC	7.38E-1	1.3
X(21)	DACTU091B	Analog Comparator Unit B21-N691B	1.04E-3	3.0
X(22)	DPP23HHFI	Instrument Line Rupture or Leakage	1.60E-3	3.0

Table D.2 (Continued)

Sample Code Variable	Event Designator	Event	Median	Error Factor
X(23)	EST12NSF	Static Switch (EPS Division II)	8.00E-4	3.0
X(24)	DTL91FHWI	Transmittor B21-N091F	2.40E-3	3.0
X(25)	LTL91EHWI	Transmittor B21-N091E	2.40E-3	3.0
X(26)	LACTU091E	Analog Comparator Unit B21-N691E	1.04E-3	3.0
X(27)	DACTU091F	Analog Comparator Unit B21-N691F	1.04E-3	3.0
X(28)	AHUDEP	Manual Initiation of ADS and non-ADS values given Auto-Initiation Failure	3.75E-2	10.0
X(29)	CM3D	Three DGs Common Mode	2.60E-4	5.0
X(30)	CM2D	Two DGs Common Mode	1.86E-3	5.0
X(31)	RHR	RHRS	3.70E-5	5.0
X(32)	C_M	RPS Mechanical	6.20E-6	5.0
X(33)	M	Limit Pressure (ATWS)	6.20E-5	5.0
X(34)	P_A	Inhibit ADS (ATWS)	6.37E-1	1.4
X(35)	R	RPT (ATWS)	8.92E-5	5.0
X(36)	C_{21}	SLC 2nd LOOP	1.52E-1	3.0
X(37)	C_E	RPS Electrical	1.24E-5	5.0
X(38)	T_FA	Isolation ATWS	8.96E+0	1.5
X(39)	L_H	Limit High Level (ATWS)	6.20E-3	5.0
X(40)	C_I	Scram Initiation (IORV)	6.20E-6	5.0
X(41)	A	Large LOCA (Inside Drywell)	1.36E-4	5.0
X(42)	S_1	Intermediate LOCA (Inside Drywell)	4.15E-4	5.0
X(43)	S_2	Small LOCA (Inside Drywell)	7.44E-4	5.0

Table D.2 (Continued)

Sample Code Variable	Event Designator	Event	Median	Error Factor
X(44)	DG1	DG No.1 Independent Failure	1.81E-2	3.0
X(45)	DG2	DG No.2 Independent Failure	1.81E-2	3.0
X(46)	S_{1x}	Intermediate LOCA in the Containment	2.97E-4	5.0
X(47)	S_{2x}	Small LOCA in the Containment	3.66E-6	5.0
X(48)	S_{2xx}	Small LOCA Outside the Containment	1.05E-7	5.0
X(49)	WS	Containment Spray	3.36E-4	3.0
X(50)	WX	External Water Supply	6.20E-5	5.0
X(51)	CA	ARI (ATWS)	8.00E-3	3.0
X(52)	QL	FW Recovery (Small LOCA)	8.05E-1	1.2
X(53)	CM2B	Two Batteries (Common Cause)	2.48E-4	5.0
X(54)	CM3B	Three Batteries Common Cause)	6.20E-6	5.0
X(55)	RCICHE	Operator Failure to Actuate and Control RCIC w/o dc Power	3.75E-2	10.0
X(56)	T_{DC}	Two dc Buses Divisions 1 and 2 EPS)	3.72E-5	5.0
X(57)	Not Used			
X(58)	Not Used			
X(59)	Not Used			
X(60)	E	Instrument Failure	7.45E-1	1.2
X(61)	I	Maintain Injection	1.24E-1	5.0

Note: Events CT2T, CT2A, and CT2L were not used in the final uncertainty analyses review.

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A review of the Probabilistic Risk Assessment (PRA)of the BWR/6 Standard Plant (GESSAR-II) was conducted with the broad objective of evaluating the contribution of the internally generated accidents to the frequency of core damage. The review included a technical assessment of the assumptions and methods used in the GESSAR-II PRA study. The BNL staff reevaluated the main results of the study within the scope and general methodological framework, including both qualitative and quantitative analyses of accident initiators, and accident sequences which result in core damage. The review assessed the relative importance of various accident sequences as well as systems with regard to their contribution to the core damage frequency. The effect of uncertainties was considered throughout the review process,and the uncertainty bands for the core damage frequency were quantified.

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