1.0 EXECUTIVE SUMMARY

1.1 BACKGROUND AND OBJECTIVES

This document reports the results of the Probabilistic Risk Assessment (PRA) performed for North Anna Nuclear Power Station. The purpose of the PRA was twofold: (1) to provide a living plant model for use by Virginia Electric and Power Company (Virginia Power) to respond to licensing and accident management issues, and (2) to meet the U. S. Nuclear Regulatory Commission (NRC) requirements for an Individual Plant Examination (IPE) as outlined in Generic Letter (GL) 88-20 (NRC, 1988) and Supplement.

The study was performed by a project team consisting of personnel from Virginia Power and Halliburton NUS Corporation at the Virginia Power offices.

The primary objective of the PRA was to develop a model of the plant at the time of performing the analysis that can be used to achieve the following:

1. Develop an appreciation of severe accident behavior.

2. Understand the most likely severe accident sequences that would occur at North Anna Units 1 and 2.

3. Gain a more quantitative understanding of the overall probabilities of core damage and fission product release.

4. Understand the underlying vulnerabilities associated with the dominant core damage sequences and most significant fission product releases.

5. Provide sufficient information for developing future risk and accident management strategies.

The secondary, but no less important, objectives of the study were:

1. To develop an in-house PRA capability within Virginia Power by involving utility staff in all aspects of the examination so that in the future the results of the study can be used by them as an integral part of the North Anna plant procedures evaluation and training programs.

2. To develop an in-house capability to modify the plant model such that PRA becomes a living document representing as closely as possible the current plant status at any time in the future.

3. To report the results of the study in accordance with the requirements of the IPE reporting guidelines (NUREG-1335, NRC, 1989a).

In the period 1985-1987 a PRA of Surry was performed under the auspices of Sandia National Laboratories (NUREG/CR-4550, Bertucio and Julius, 1990), and the final results were published in 1990. In the period 1989-1991 the IPE was performed for the Surry Power Station (Virginia Power, 1991a). The work performed for these two studies was reviewed in the preliminary phase of performing the IPE for North Anna. It was decided that the similarities between Surry and North Anna were such that the Surry IPE would make an excellent basis for the North Anna IPE for the following reasons:

1. The detailed documentation giving the basis for the results was very extensive and would meet the quality assurance (QA) requirements of a study intended for use in future licensing decisions.

2. The majority of ECCS systems are very similar at both plants. The containment designs are virtually the same.

3. A number of issues, e.g., room cooling requirements are similar for both plants.

4. A comprehensive plant-specific data analysis for Surry was available and could be augmented for North Anna.

In order to meet the objectives, the scope of this study is equivalent to a modified Level 2 PRA. Initiating events studied are those arising from internal events, Loss of Offsite Power (LOOP), and internal flooding. The fault and event trees developed to delineate the accident sequences and plant damage states (PDS) are drawn up with the knowledge that there is a requirement to perform external event analyses covering such initiators as fires and earthquakes. Use has been made of the Containment Building (Containment) capacity analysis performed for the NUREG-1150 study (NUREG-1150, NRC, 1989b) and new Modular Accident Analysis Program (MAAP) code investigations of core melt phenomenology. The Containment Building event trees (CETs) and source terms are based on this work and that performed for the Surry IPE as well as the work performed for the Level 2 analysis in the NUREG-1150 study. The frequencies of core damage and fission product releases are based on a combination of plant-specific and generic data as used in the Surry IPE. The uncertainties associated with the data and the phenomenologic assumptions are quantified and included in the results.

1.2 PLANT FAMILIARIZATION

The project team consisted of four Virginia Power engineers and eleven engineers from Halliburton NUS Environmental Corporation. One of the Virginia Power representatives had spent 6 years on site at North Anna, as Shift Technical Advisor (STA) and supervisor immediately prior to joining the project. The other Virginia Power engineers have made numerous visits to the station in connection with licensing and other issues. In the course of the study, the following visits were made to the plant in order for team members to familiarize themselves with particular aspects of plant design, operation, or maintenance:

- A one-day plant walkthrough by the HNUS project team members at the beginning of the project
- A number of visits to collect plant-specific data
- A number of visits to collect simulator data for the human reliability analysis (HRA)
- Plant walkdowns and use of the VIMS for internal flooding analysis

It should be emphasized that 95% of the work was done at the Virginia Power main offices with ready access to the latest controlled drawings and the appropriate design, licensing, and maintenance engineers. Work products were reviewed by personnel onsite as well as by the independent review team.

1.3 OVERALL METHODOLOGY

The methodology used to perform the IPE for North Anna Units 1 and 2 is based on the performance of a Level 2 PRA. The specific approach used is very similar to that used in the recent Sandia PRA of Surry for the NRC (NUREG/CR-4550, Bertucio, 1990) and the Surry IPE. Fundamentally, this is based on the use of event trees to develop the sequence of events following a plant transient or loss-of-coolant accident (LOCA), and fault trees to model the system failures and successes at each phase of the sequence. Each individual sequence is quantified by combining the main system and support system failures that lead to a given sequence of events. Each sequence defines a set of conditions leading to inadequate cooling of the core. The source term is determined by extending the analysis of events through the phases of core damage, Reactor Vessel failure, Containment Building Failure, and ultimate release of fission products. In order that this latter process may be represented in a logical manner, Containment Event Trees are developed to represent the range of possible events during the course of core damage and the events within the Containment Building. In order to investigate these events, the MAAP code was

used to model core damage, Reactor Vessel failure, and the corresponding variations in Containment pressure. The Containment Building strength analysis performed for NUREG-1150 (NUREG/CR-4551 Volume 3, Breeding et al., 1990) was used to estimate the probability of Containment Building failures at North Anna. This is considered acceptable as the design of both containments is the same.

Finally the source terms for the various Containment Building failure scenarios identified above were the same as those used in the Surry IPE which were based on the work done by Sandia National Laboratories (SNL) for the evaluation of severe accident risks (NUREG/CR-4551 Volumes 1 & 2, Gorham-Bergeron, 1988) and the use of new MAAP analyses for dominant sequences in each release category.

The methodology used to perform the analysis is described in more detail in Chapter 2. As the PRA is aimed at identifying specific strengths and weaknesses of the design and operation of North Anna Units 1 and 2, particular emphasis was placed on ensuring that the information used in key areas represented the current condition of the plant (December 1992 for design and operating practices, December 1991 for plant specific failure data).

In developing the sequences of events leading to core damage or fission product release, particular attention was paid to the relationship between system failures following a plant trip and the actions that the operators would be instructed to take in accordance with the Emergency Operating Procedures (EOPs). In this way, the EOPs were integrated into the analysis. The human reliability analysis was based on a careful evaluation of the EOP related to each scenario. A number of simulator observations and measurements were made in order to enhance the qualitative understanding of the way in which the operators used the procedures in the course of events and to assist in quantifying the failure probabilities of key actions. A recovery analysis was performed in order to ensure that proper credit was given for the operators' performance following various system failures.

Plant-specific data were collected and analyzed for system maintenance outages, initiating event frequencies, and a range of component failures.

One very important feature of the methodology is that the entire Level 1 and Level 2 PRAs were performed on the PRA workstations NUPRA (HNUS, 1992) and NUCAP+ (Fulford and Sherry, 1991), which allowed a completely integrated model from initiating event to source term release to be developed and maintained on a PC. This is essential for the establishment of a living PRA, i.e., one which can be easily maintained and modified as changes are made in design and operation of the plant. The workstation model is fully

supported by a comprehensive set of analysis files, which detail the assumptions and information sources used at each stage of model development.

A formal Quality Assurance Plan was developed for the project to ensure the appropriate level of review and documentation. The work products were reviewed at each stage, by project team members. North Anna station personnel reviewed key documents. In addition, an independent review was performed to ensure consistency within the overall methodology. All comments received have been addressed and retained within the appropriate analysis files.

1.4 SUMMARY OF MAJOR FINDINGS

1.4.1 Results of Core Damage Frequency for Internal Events

Core damage is defined as failure of decay heat removal such that the maximum fuel temperature will exceed the licensing basis temperature of 2200°F or the core exit thermocouples will reach 1200°F and long-term cooling cannot be established. Although these criteria are slightly conservative, the increase in the time to the onset of significant core damage following failure of decay heat removal compared with the time to 2200°F is not significant in terms of system recovery or actions by the operators. In a number of sequences, the time it takes to achieve this temperature limit is based on actions taken by the operators when the core exit thermocouples indicate 1200°F. Each event tree was extended to include the containment systems and where appropriate the recovery of cooling injection after core damage or vessel failure in order to accurately define the plant damage states which were the basis for the containment accident progression and source term analysis.

The internal events portion of the PRA identified 61 core damage sequences with an annual frequency of greater than 1.0E-7, which contributed 96% of the overall core damage frequency. An additional 161 sequences with a point estimate frequency of greater than 1.0E-9/year contributed the remaining 4% of the overall core damage frequency. The accident grouping by initiating event class is shown in Table 1-1 and Figure 1-1.

The internal events core damage model gave a point estimate frequency of 6.8E-5 per reactor-year. The combined frequency of the 161 sequences below the 1.0E-7 cutoff is less than 2.9E-6. An uncertainty analysis was performed to evaluate the uncertainty on core damage frequency resulting from the uncertainties on the parameter values of the core damage model. The cumulative distribution function for the core damage frequency is shown in Figure 1-2.

Some significant parameters of the core damage frequency distribution function are as follows:

Mean	1.66E-4
Standard Deviation	1.03E-3
95th Percentile	3.41E-4
Median	7.41E-5
5th Percentile	2.74E-5

The difference between the mean value, obtained from the uncertainty analysis, and the point estimate, results from the correlation of the samples of those basic event probabilities that are based on the same parameter value distribution. This is the so-called state of knowledge correlation (Apostolakis and Kaplan, 1981). Several of the cut sets that are affected have point estimate frequencies in the 1.0E-8 range. The parameter values that contribute to these cut sets are generally based on generic estimates. The reason they contribute significantly to the difference is that the representation of the uncertainty on the parameters results in a large variance on the parameter value. This is in many respects somewhat arbitrary; for example, the choice of the lognormal distribution was based on accepted industry practice; the use of large error factors is a way of increasing the mean value with respect to a given median value [e.g., air-operated valves (AOVs)], but it also increases the variance. Thus, the difference between the point estimate and mean value is potentially exaggerated by the way in which the uncertainty characterization of parameter estimates was established.

On review of the cut sets, it did not appear that the overall characterization of the safety of the plant, in terms of the contributors and their relative importance, would be significantly altered by using the uncertainty analysis for the estimation of core damage frequency. Therefore, the point estimate results were used in the remainder of the analysis. In further support of this approach, it should be noted that the point estimate values chosen for the parameters were either realistic (when sufficient data were available) or conservative.

An event importance analysis was performed on the overall core damage model. In this analysis the relative importance of each basic event was calculated with respect to three different measures: Fussell-Vesely, risk reduction worth, and risk achievement worth. The results are shown in Table 1-2.

The Fussell-Vesely importance is a measure of the contribution of the given component to the overall core damage frequency by comparing the sum of cut sets in which that basic event occurs with the total sum of all cut sets. The risk reduction worth shows the reduction in the core damage frequency that would be achieved if the component were perfect or its failure probability were zero.

Three of the top four highest ranking events for risk reduction are the Loss of Offsite Power initiating event (IE-T1), the small LOCA initiating event (IE-S2), and the steam generator tube rupture event (IE-T7). (Note the complement events indicated by "C-xxx" and the 1EE-BAT-i-2HR Battery failure in 2 hours after SBO are not true events and should not be considered in the interpretation of results.) This is consistent with the core damage profile where T1 accounts for 29.2% of CDF (this includes the station blackout contribution), S2 accounts for 14.8% of CDF, and T7 accounts for 10.3% of CDF. In Table 1-2, the Fussell-Vesely importance values for these initiators are precisely these percentages. Having an initiating event group as the top risk reduction item indicates the risk from these initiators is spread over many components and involves several aspects of accident mitigation. Alternatively, it can be said that there are no single component improvements or changes that would have a dominant impact on accident mitigation for all these initiating events. The frequencies for the T1, S2, and T7 initiators are generic industry values as opposed to plant specific data. The S1 LOCA and T8 loss of Emergency Switchgear Room cooling initiating events are the fifth and sixth most important risk reduction events having F-V values of .098 and .097, respectively.

The most important component for risk reduction is the 1H Emergency Diesel Generator. This component is the most important single component. The seventh, eighth and eleventh events (or numbers 9, 13 and 17 in the listing) represent different fault modes of EDG 1H. As such, they can be combined to yield one F-V measure of unavailability for EDG 1H which is .23 (the sum of the three F-V values). This is due to 1) the relatively high fault probabilities for the EDG 1H compared to other components and 2) the higher Loss of Offsite Power (T1, T1A and T1Tr) and partial loss of switchyard feeder power (T9A and T9ATr) contribution to the total CDF (35% for all 5 events, T1, T1A, T1Tr, T9A and T9ATr).

The second most important component for risk reduction is the turbine driven Auxiliary Feedwater pump. The ninth, 16th, 24th and 46th events (or numbers 15, 23, 32 and 57 in the listing) represent different fault modes of the turbine driven Auxiliary Feedwater pump. As such they can be combined to yield one F-V measure for unavailability of the turbine driven pump. If the four values are added, the resultant F-V for the turbine driven AFW pump is .18. This is due to 1) the relatively high fault probabilities for the turbine driven pump compared to other components (high fault probabilities for turbine driven pump is typical) and 2) the increased reliance on the turbine driven Auxiliary Feedwater pump for initiators such as T9A, T9B, T5A, T5B, and T7, where one motor driven pump is unavailable due to the initiator, or in the case of T7, is aligned to the affected generator. Having the turbine driven pump as a significant component for risk reduction indicates the risk profile is dominated by loss of steam generator heat removal following the initiating event.

The third most important event and the most important operator action (number five in the listing) is failure of operator action to initiate High Head Safety Injection. This human action appears in T1 and T1A sequences involving loss of AFW and in several Hv transfer sequences (e.g., event trees T1Tr, T2Tr, T2ATr, etc.) involving restoration of Emergency Power before core damage, but where HHSI is required to prevent a RC Pump Seal LOCA. Although the human action to manually initiate HHSI is important, the split between Loss of Offsite Power and other transient initiators indicates that two human action models would be more appropriate, yielding the same combined importance but with an apportionment between the two transient types.

The next most important operator action is the 10th event (number 16 in the listing), recovery actions for loss of Unit 1 ESGR cooling using Unit 2 ESGR chilled air. Initiating events for transients with MFW available and large LOCA are listed next. The 20th listed event is failure of operator action to rapidly depressurize the Steam Generators during a medium break LOCA.

The event listed 22 represents unavailability of Emergency Diesel Generator 1J. It can be combined with events 25 and 41, which represent other failure modes of EDG 1J. Adding these three events together yields an overall F-V importance value of .13 for EDG 1J. This places it fifth in true ranking, behind the S2 initiator. The asymmetrical dependence between the 1J and 1H diesel is due to the greater dependence of ESGR cooling components upon the 1H bus (2 chillers) than on the 1J bus (1 chiller).

The events ranked in order of risk achievement worth are shown in Table 1-3. Risk achievement worth must be viewed with an understanding of how it is calculated. The risk achievement worth for an event represents the increase in core damage frequency if that event's probability is 1.0. This can be interpreted as guaranteeing that the failure will occur. The two top events for risk achievement are modeled to lead straight to core damage. These are Reactor Vessel rupture and Interfacing System LOCA initiating events. Also, they have very low probabilities in the base case CDF profile. Thus, if their probabilities are increased to 1.0, the resultant increase in CDF is very high.

The third most important event in risk achievement worth is mechanical binding of the control rods. This has a high risk achievement worth because, it leads directly to core damage when combined with any initiator and it has a very low probability in the base case.

The next event (#4) involves common cause failure of the Service Water Reservoir screens, which fails both Unit 1 ESGR cooling, and its recovery, Unit 2 ESGR cooling. It has a high risk achievement worth because it affects all of the Hv Transfer event trees. The next two events, 1QSMV--PG-1Q38, and 1SICKV-CC-838689, cause common

mode failure of the HHSI and LHSI systems. The QS term is plugging of the manual isolation valve on the discharge of the RWST and the SI term is common cause failure of check valves 83, 86, and 89 which are located in the SI injection lines into the cold legs.

The next several events involve faults of a 4160 V or 480 V bus. Both 4160 V buses, the 480 V buses, and several MCC's are represented. These events appear in virtually all the sequences at lower frequencies. Note that the 1H buses characteristically have a higher risk achievement worth than comparable 1J buses, again due to the greater dependence of ESGR cooling components upon the 1H buses.

1.4.2 Core Damage Frequency from Internal Flooding

The core damage frequency from internal flooding is 3.6E-6/year which is approximately 5% of the overall core damage frequency. The dominant contribution is from service water floods in the Auxiliary Building.

It can be seen that the base case results show that core damage from internal flooding is not a vulnerability at North Anna. This is the result of identifying a number of minor modifications during the course of the study, as potential flooding vulnerabilities were identified. The required plant modifications included in the IPE model are as follows:

1. Back flow prevention devices are fitted in the charging pump cubicles' floor drains in order to prevent floods in the Auxiliary Building and Quench Spray Pump House spreading to the charging cubicles.

2. A flood barrier is erected in the pipe tunnel between the Quench Spray Pump House and the Auxiliary Building to prevent the spreading of floods from one to the other.

3. The Chiller Room doors are modified to prevent flooding of the Instrument Rack Room and Emergency Switchgear Room following a flood in the Chiller Room.

1.4.3 Containment Building Performance

The North Anna Containment Building structures and systems are robust with respect to the challenges presented by severe accidents. Because of the high assessed strength of the Containment structure, both early as well as late over-pressure failure of the Containment is very unlikely. The North Anna Containment Building is operated in a subatmospheric mode; consequently, the probability of loss of isolation is extremely remote since any significant preexisting leakage would be easily

detected. The major threat of early, large radionuclide leakage at North Anna results from core damage Containment bypass sequences, particularly SGTRs. Figure 1-3 shows a breakdown of the predicted North Anna Containment Building performance for severe accidents. Table 1-4 compares the North Anna IPE, the Surry IPE and NUREG-1150 results. In general, the results from the three studies are quite similar. The differences that exist stem mostly from the difference in contributions from the different initiators. Section 7.2 discusses this in more detail.

1.4.4 Comparison of Results

The major purpose of this study, was to ensure that the PRA model was developed and understood by the Virginia Power staff and represented the as-built-and-as-operated condition of North Anna Units 1 and 2 at the time of the performance of the PRA. The guidance for performing the IPE indicated that heavy reliance could be placed on the results of the previous studies performed for similar plants. Therefore, the work performed for the Surry IPE was used as the starting point for the North Anna Analysis.

All the systems at North Anna were analyzed and new fault trees developed for each one. There are differences in the support system (electric power, cooling water) design which resulted in the identification of different initiating events. The results from the North Anna IPE are compared with the Surry IPE and the NRC PRA of Surry reported in NUREG/CR-4550.

1.4.4.1 Comparison of Core Damage Frequencies

The comparison of core damage frequencies is shown in Table 1-5. It can be seen that the results for Surry (Virginia Power, 1991a) and North Anna IPEs are very similar and somewhat higher than those from the NRC study of Surry. However, investigations of the design requirement for room cooling, the capability of removing heat from the Containment Building, and the requirements for RHR following an SGTR resulted in the introduction of new sequences associated with loss of Emergency Switchgear Room cooling, consequential loss of ESGR cooling after other initiators, loss of Containment heat removal (Surry only), and a revised frequency for core damage sequences following an SGTR.

Whereas in the NUREG/CR-4550 study the LOOP leading to station blackout was the dominant contributor to core damage, it can now be seen from Figure 1-2 that, although loss of offsite power is still a high contributor, LOCA, SGTR, and transients are all significant contributors. It should be noted that the increase in the loss of feedwater initiating event contribution to core damage frequency is entirely due to the dependency on Emergency Switchgear Room cooling and not on poor performance of the front-line decay heat removal systems.

The relative reduction in the station blackout core damage frequency from NUREG/CR-4550 is the result of three changes. First, the IPEs credited successful Turbine-Driven Auxiliary Feedwater Pump (AFWP) operation after battery depletion. Thus, although the battery depletion time was similar to that in NUREG/CR-4550, AFW was potentially available until Emergency Condensate Storage Tank (ECST) depletion. An extension of the time to core uncovery is probable for the case in which the AFW pumps are running at the time of battery depletion. (Operators have indicated that they are not instructed by procedure to trip the pumps at that time and, thus, that they would not do so.) Second, the RC Pump seal LOCA model used for the IPE predicted an average core uncovery time due to seal failure of about 9 hours, rather than the 3.5 hours used in NUREG/CR-4550. The IPE seal LOCA model is based on Westinghouse seal performance analysis. Third, the common-cause failure probabilities for diesel generators was lower than that used in the NUREG. A rigorous analysis of industry data was performed to generate as realistic a value as possible for the potential for common-cause failures of the diesel generators.

Finally, the ATWS sequence frequencies are somewhat lower as the result of more accurate analysis of the pressure relief requirements at the various stages of core burnup. Although the results for North Anna and Surry are approximately the same there are differences in the design which individually would have been expected to give different results for the two stations. The joint Westinghouse Owners Group/Westinghouse program for the ATWS rule administration described in WCAP-11992 (Westinghouse, 1988) identified a more rigorous method for determining the probabilities of core damage based on evaluating the pressure relief requirement during core burnup, following an anticipated trip without scram. It also discussed the impact of fitting the AMSAC modification. The AMSAC modification has been installed at North Anna but was not installed at Surry at the time of the IPE. The calculated unfavorable exposure time (UET) for North Anna, Unit 1 was 27.7% compared with zero for Surry. The most likely reason for the higher UET is a combination of the higher nominal inlet temperatures at hot full power at North Anna and larger power defects from the higher power North Anna cores. Thus the reduction in core damage frequency from the fitting of the AMSAC modifications is offset by an increase due to the unfavorable exposure time, when the pressure relief is inadequate.

1.4.4.2 Fission Product Release

There are several factors that would tend to produce small releases at North Anna: the Containment Building is strong; there is a high degree of redundancy in the sprays; as the plant is subatmospheric, there is a very low probability of its being in a non-isolated state; and the piping arrangement in the Safeguards Building is such that most interfacing LOCAs (V) will vent releases under

water. The cavity is not connected to the sump directly at floor level but rather through a somewhat elevated vent path. This means that it is difficult to get water into the cavity other than by operation of the Quench/Recirculation Sprays or in vessel injection (following reactor vessel failure). This has advantages and disadvantages; a wet cavity means debris cooling, but it also can impose a large heat load on the Containment.

The sprays play several roles, all of which are important with regard to source terms: they can "wash out" airborne radionuclides in the Containment, they provide the major pathway for the introduction of water into the cavity and onto the debris, and they are the vehicle for Containment heat removal.

The MAAP-derived release fractions (calculated for 11 of the 24 source term categories) confirm what is already known from other work (NUREG-1150, for example) the Containment Building bypass sequences (interfacing LOCA [V] and SGTR) have the greatest release potential. This is because of the relative scarcity of mitigating features in the release pathways. Following a SGTR, the Steam Generator with the broken tube is likely to be dry when core damage and fission product release occurs. The SGTR sequence is also significant on a frequency basis (see Section 4.7.4).

The calculated release fractions generally agree in magnitude with values reported for NUREG-0956 and NUREG-1150. A comparison of the IPE values and those reported in NUREG-1150 is shown in Figures 1-4 and 1-5. Sensitivity studies demonstrated that the sprays are important in minimizing releases and that different modeling assumptions regarding tellurium release from the fuel can affect its release fraction significantly. While no direct analyses of uncertainty were performed, the extensive NUREG-1150 work has indicated that in most cases two orders of magnitude is not unreasonable uncertainty for many of the release fractions for any given source term category (STC).

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TABLE 1-1
ACCIDENT GROUPING BY INITIATING EVENT CLASS

Initiating Event Type	Point Estimate Frequency (per year)	Percentage of Total
Internal Events:		
LOCA (A, S1, S2, RX)	2.1E-5	31
Loss of Offsite Power (T1, T1A, T1TR)	2.0E-5	29
Transient (T2, T2A, T3, T4, T5A T5B, T6, T8, T9A, T9B, T2TR, T2ATR, T3TR, T9ATR, T9BTR)	1.8E-5	27
Steam Generator Tube Rupture (T7)	7.0E-6	10
Interfacing System LOCA (VX)	1.6E-6	2
ATWS (TH, TL)	4.2E-7	1
Total Internal Events	6.8E-5	100
Internal Flooding:		
Auxiliary Building	2.6E-6	72
Air Conditioning Chiller Room	9.7E-7	27
Turbine Building	0	0
Total Internal Flooding	3.6E-6	100
Combined CDF:		
Total Internal Events	6.8E-5	95
Total Internal Flooding	3.6E-6	5
	7.1E-5	100

TABLE 1-2
BASIC EVENTS RANKED BY THE FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
1	IE-T1	1.139E-1	2.923E-1	3.27	1.413
2	IE-S2	2.100E-2	1.479E-1	7.90	1.174
3	1EE-BAT-I-2HR	1.000E+0	1.442E-1	1.00	1.169
4	C-LT01	9.068E-1	1.436E-1	1.01	1.168
5	HEP-1FRH:1-11	4.824E-2	1.163E-1	3.29	1.132
6	IE-T7	1.000E-2	1.033E-1	11.23	1.115
7	IE-S1	1.000E-3	9.785E-2	98.77	1.108
8	IE-T8	6.579E-3	9.665E-2	15.59	1.107
9	1EGEDG-FS-1H	1.434E-2	8.702E-2	6.98	1.095
10	C-Y02	9.800E-1	8.556E-2	1.00	1.094
11	C-RC303	8.750E-1	8.554E-2	1.01	1.094
12	1EE-BAT-II-2HR	1.000E+0	8.499E-2	1.00	1.093
13	1EGEDG-FR-1H	1.330E-2	8.029E-2	6.96	1.087
14	C-SGI01	9.890E-1	7.934E-2	1.00	1.086
15	1FWTRB-FR-12HP2	5.742E-2	7.282E-2	2.20	1.079
16	HEP-0AP55-10HR	4.949E-3	7.078E-2	15.23	1.076
17	1EGEDG-UM-1H	1.781E-2	6.081E-2	4.35	1.065
18	IE-T3	1.350E+0	6.078E-2	0.98	1.065
19	IE-A	4.999E-4	6.027E-2	121.49	1.064
20	HEP-1FRC:1-11-S1	1.000E+0	5.962E-2	1.00	1.063
21	C-P02	9.870E-1	5.411E-2	1.00	1.057
22	1EGEDG-FS-1J	1.434E-2	4.804E-2	4.30	1.050
23	1FWTRB-FS-1FWP2	1.854E-2	4.678E-2	3.48	1.049
24	1HVCHU-UM-1HVE4B	9.440E-2	4.579E-2	1.44	1.048
25	1EGEDG-FR-1J	1.330E-2	4.455E-2	4.30	1.047
26	NON-REC-B103	6.799E-1	4.431E-2	1.02	1.046
27	C-QS05	9.460E-1	4.416E-2	1.00	1.046
28	REC-SCREEN-TURNS	1.000E-1	4.348E-2	1.39	1.045
29	1SWSCN-CC-SWRES	6.392E-5	4.318E-2	676.43	1.045
30	1IAIAS-LF-OUTIA	2.520E-4	4.257E-2	169.90	1.044
31	REC-1AP28	1.017E-1	4.257E-2	1.38	1.044
32	1FWTRB-FR-24HP2	1.115E-1	4.127E-2	1.33	1.043
33	NON-REC-B02	3.400E-1	4.072E-2	1.08	1.042
34	1CHCKV-FO-1CH254	1.147E-3	3.956E-2	35.44	1.041
35	HEP-NO-PROCEDURE	1.000E+0	3.910E-2	1.00	1.041
36	HEP-1E3-13	2.180E-2	3.881E-2	2.74	1.040
37	C-FM01	4.800E-2	3.866E-2	1.77	1.040
38	HEP-1ES1:2-S1	1.000E+0	3.860E-2	1.00	1.040
39	HEP-0AP55-20HR	2.600E-4	3.677E-2	142.42	1.038
40	1EE-BAT-III-2HR	1.000E+0	3.614E-2	1.00	1.037
41	1EGEDG-UM-1J	1.781E-2	3.391E-2	2.87	1.035
42	HEP-1AP22:5	1.750E-4	3.367E-2	193.37	1.035

TABLE 1-2 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
43	1HVCHU-FS-1HVE4B	4.545E-2	3.016E-2	1.63	1.031
44	NON-REC-B16	7.499E-3	3.005E-2	4.98	1.031
45	C-D102	9.400E-1	2.601E-2	1.00	1.027
46	IE-T2A	5.500E-1	2.510E-2	1.02	1.026
47	HEP-1OP49:1	1.326E-1	2.497E-2	1.16	1.026
48	1RCRV--FC-1456	9.988E-3	2.474E-2	3.45	1.025
49	REC-B12AVE	1.056E-1	2.441E-2	1.21	1.025
50	PROB-FM01	9.522E-1	2.383E-2	1.00	1.024
51	IE-VX	1.600E-6	2.358E-2	14737.41	1.024
52	1QSMV--PG-1QS38	6.749E-5	2.352E-2	349.45	1.024
53	T9A-FREQ-500KV-1	1.786E-1	2.330E-2	1.11	1.024
54	1MSRV--FC-101C	9.988E-3	2.310E-2	3.29	1.024
55	1MSMV--LK-1MS97	3.999E-2	2.305E-2	1.55	1.024
56	REC-1OP14:1	1.043E-1	2.288E-2	1.20	1.023
57	1FWTRB-UM-1FWP2	1.366E-2	2.267E-2	2.64	1.023
58	T9A-FREQ-4160-1H	5.999E-3	2.248E-2	4.73	1.023
59	1SICKV-CC-838689	6.339E-5	2.208E-2	349.26	1.023
60	1SICKV-FC-1SI47	6.339E-4	2.155E-2	34.97	1.022
61	1SIMOV-FO-1862B	1.090E-2	2.153E-2	2.95	1.022
62	1SIMOV-FC-1860B	1.090E-2	2.153E-2	2.95	1.022
63	C-B103	3.200E-1	2.081E-2	1.04	1.021
64	NON-REC-B117	6.799E-1	2.045E-2	1.01	1.021
65	1SIPSB-CC-FS1A1B	4.934E-4	2.025E-2	42.03	1.021
66	1RCPORV-DMDSBO	2.000E-1	1.928E-2	1.08	1.020
67	NON-REC-B01	4.799E-1	1.928E-2	1.02	1.020
68	1SIMOV-FC-1860A	1.090E-2	1.771E-2	2.61	1.018
69	1SIMOV-FO-1862A	1.090E-2	1.771E-2	2.61	1.018
70	1CHPAT-CC-FS1ABC	4.968E-4	1.692E-2	35.04	1.017
71	HEP-0AP55-40HR	1.250E-1	1.656E-2	1.12	1.017
72	1SIMOV-CC-1860AB	3.903E-4	1.598E-2	41.92	1.016
73	1SWTCV-FC-SW102B	1.812E-2	1.589E-2	1.86	1.016
74	1SWPSB-UM-1SWP-4	8.290E-2	1.560E-2	1.17	1.016
75	1RCRV--FO-1456	2.500E-2	1.548E-2	1.60	1.016
76	IE-T2	5.000E-2	1.515E-2	1.29	1.015
77	1RCRV--FO-1455C	2.500E-2	1.514E-2	1.59	1.015
78	1EGEDG-CC-1H-1J	2.663E-4	1.411E-2	53.96	1.014
79	C-P01	1.000E+0	1.347E-2	1.00	1.014
80	HEP-1FRC:1-11-S2	1.062E-2	1.332E-2	2.24	1.013
81	1SIMOV-CC-867836	3.903E-4	1.327E-2	34.99	1.013
82	1SIMOV-CC-1115CE	3.903E-4	1.323E-2	34.88	1.013
83	1SIMOV-CC-1115BD	3.903E-4	1.323E-2	34.88	1.013

TABLE 1-3
BASIC EVENTS RANKED BY THE
RISK ACHIEVEMENT WORTH IMPORTANCE MEASURE

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
1	IE-RX	2.664E-7	3.946E-3	4814.19	1.004
2	IE-VX	1.600E-6	2.358E-2	4737.41	1.024
3	1RPROD-LF-CRODS	1.800E-6	5.718E-3	3178.03	1.006
4	1SWSCN-CC-SWRES	6.392E-5	4.318E-2	676.43	1.045
5	1QSMV--PG-1QS38	6.749E-5	2.352E-2	349.45	1.024
6	1SICKV-CC-838689	6.339E-5	2.208E-2	349.26	1.023
7	1EEBKR-SO-15H8	3.356E-5	9.561E-3	285.88	1.010
8	1EEBKR-SO-14H1	3.356E-5	9.332E-3	279.06	1.009
9	1EEBUS-LU-1H1	1.215E-5	3.334E-3	275.48	1.003
10	1EEBUS-LU-1H-480	1.215E-5	3.334E-3	275.48	1.003
11	1EEBUS-LU-1H	1.215E-5	3.330E-3	275.14	1.003
12	1EETFM-LP-1H	1.899E-5	5.134E-3	271.32	1.005
13	IE-T4	6.001E-7	1.573E-4	263.09	1.000
14	1EEBUS-UM-1H	1.000E-5	2.556E-3	256.64	1.003
15	1EEBKR-SO-14H2	3.356E-5	8.525E-3	255.01	1.009
16	1EEBUS-UM-1H-480	1.000E-5	2.498E-3	250.86	1.003
17	1EEBUS-LU-1H1-4	1.215E-5	2.993E-3	247.47	1.003
18	1EEBUS-UM-1H1-4	1.000E-5	2.280E-3	229.03	1.002
19	HEP-1AP22:5	1.750E-4	3.367E-2	193.37	1.035
20	1FWCKV-CC-ALLAFW	6.339E-5	1.210E-2	191.93	1.012
21	1FWCKV-LEAKAGE	1.000E-5	1.802E-3	181.25	1.002
22	1IAIAS-LF-OUTIA	2.520E-4	4.257E-2	169.90	1.044
23	HEP-0AP55-20HR	2.600E-4	3.677E-2	142.42	1.038
24	IE-A	4.999E-4	6.027E-2	121.49	1.064
25	1EEBKR-SO-14J1	3.356E-5	3.646E-3	109.63	1.004
26	1EEBKR-SO-15J8	3.356E-5	3.646E-3	109.63	1.004
27	1EETFM-LP-1J	1.899E-5	1.974E-3	104.94	1.002
28	1EEBUS-LU-1J-480	1.215E-5	1.258E-3	104.55	1.001
29	1EEBUS-LU-1J1	1.215E-5	1.258E-3	104.55	1.001
30	1EEBUS-LU-1J	1.215E-5	1.248E-3	103.72	1.001
31	IE-S1	1.000E-3	9.785E-2	98.77	1.108
32	1EEBKR-SO-14J4	3.356E-5	2.807E-3	84.64	1.003
33	1EEBUS-LU-1J1-1	1.215E-5	9.851E-4	82.11	1.001
34	1EEBUS-UM-1J-480	1.000E-5	8.015E-4	81.16	1.001
35	1EEBUS-UM-1J	1.000E-5	7.929E-4	80.30	1.001
36	1EEBUS-UM-1J1-1	1.000E-5	5.544E-4	56.44	1.001
37	1EGEDG-CC-1H1J2J	9.576E-5	5.252E-3	55.84	1.005
38	1EGEDG-CC-ALL	6.090E-5	3.334E-3	55.74	1.003
39	1EGEDG-CC-1H-1J	2.663E-4	1.411E-2	53.96	1.014
40	1EGEDG-CC-1H1J2H	9.576E-5	5.043E-3	53.66	1.005
41	1EEBAT-CC-I-III	1.050E-6	5.426E-5	52.68	1.000
42	1SIPSB-CC-FS1A1B	4.934E-4	2.025E-2	42.03	1.021
43	1SIMOV-CC-1860AB	3.903E-4	1.598E-2	41.92	1.016

TABLE 1-3 (Continued)
BASIC EVENTS RANKED BY THE
RISK ACHIEVEMENT WORTH IMPORTANCE MEASURE

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
44	1SICKV-CC-FC926	6.339E-5	2.574E-3	41.60	1.003
45	1SICKV-CC-FC116	6.339E-5	2.571E-3	41.55	1.003
46	1RPBKR-CC-RTARTB	1.300E-5	4.696E-4	37.13	1.000
47	1EEBUS-UM-DC-III	2.000E-4	7.079E-3	36.39	1.007
48	1CHCKV-FO-1CH254	1.147E-3	3.956E-2	35.44	1.041
49	1CHPAT-CC-FS1ABC	4.968E-4	1.692E-2	35.04	1.017
50	1SIMOV-CC-867836	3.903E-4	1.327E-2	34.99	1.013
51	1SICKV-FC-1SI47	6.339E-4	2.155E-2	34.97	1.022
52	1SIMOV-CC-1115CE	3.903E-4	1.323E-2	34.88	1.013
53	1SIMOV-CC-1115BD	3.903E-4	1.323E-2	34.88	1.013
54	1SICKV-CC-79185	6.339E-5	2.121E-3	34.46	1.002
55	1CESTR-CC-SUMPPG	5.000E-5	1.670E-3	34.39	1.002
56	1SIMV--PG-1SI46	4.499E-5	1.497E-3	34.26	1.001
57	1EEBUS-UM-DC-I	2.000E-4	5.508E-3	28.54	1.006
58	1EEBUS-LU-DC-I	1.215E-5	3.301E-4	28.17	1.000
59	1FWPSB-CC-MDP3AB	1.418E-4	3.839E-3	28.07	1.004
60	1EEBUS-LU-DC-III	1.215E-5	3.285E-4	28.05	1.000
61	1FWPCV-CC-159AB	1.369E-5	3.099E-4	23.64	1.000
62	2EEBKR-SO-24H4	3.356E-5	6.664E-4	20.86	1.001
63	2EEBKR-SO-24H1	3.356E-5	6.664E-4	20.86	1.001
64	2EEBKR-SO-25H8	3.356E-5	6.664E-4	20.86	1.001
65	2EETFM-LP-2H	1.899E-5	3.484E-4	19.35	1.000
66	2EEBUS-LU-2H1-1	1.215E-5	2.225E-4	19.32	1.000
67	2EEBUS-LU-2H1	1.215E-5	2.225E-4	19.32	1.000
68	2EEBUS-LU-2H	1.215E-5	2.225E-4	19.32	1.000
69	2EEBUS-LU-2H-480	1.215E-5	2.225E-4	19.32	1.000
70	1SICKV-CC-FC1229	6.339E-5	1.081E-3	18.06	1.001
71	2IAIAS-LF-OUTIA	2.520E-4	3.980E-3	16.79	1.004
72	2EEBKR-SO-24H2	3.356E-5	5.242E-4	16.62	1.001
73	IE-T8	6.579E-3	9.665E-2	15.59	1.107
74	2EEBUS-LU-2H1-4	1.215E-5	1.757E-4	15.46	1.000
75	HEP-0AP55-10HR	4.949E-3	7.078E-2	15.23	1.076
76	1EP-LOOP-24	3.120E-4	3.850E-3	13.34	1.004
77	2EEBUS-UM-2H1-1	2.000E-4	2.323E-3	12.62	1.002
78	2EEBUS-UM-2H-480	2.000E-4	2.323E-3	12.62	1.002
79	2EEBUS-UM-2H	2.000E-4	2.323E-3	12.62	1.002
80	IE-T6	6.270E-6	6.611E-5	11.54	1.000
81	IE-T7	1.000E-2	1.033E-1	11.23	1.115
82	1RCRV--CC-RCPORV	9.988E-4	9.040E-3	10.04	1.009
83	1SICKV-FC-1SI161	6.339E-4	5.540E-3	9.73	1.006
84	1SICKV-FC-1SI125	6.339E-4	5.534E-3	9.72	1.006
85	1SICKV-FC-1SI127	6.339E-4	5.534E-3	9.72	1.006

TABLE 1-4
COMPARISON OF NORTH ANNA IPE, SURRY IPE AND NUREG-1150
CONTAINMENT RESPONSE ANALYSES
(Excluding Flooding Sequences)

	Total Frequency (Fraction)		
	North Anna IPE	**Surry IPE**	**NUREG-1150**
No Containment Failure (No RV Failure)	7.1E-6 (.10)	6.5E-6 (.09)	1.9E-5 (.48)
No Containment Failure (RV Failure)	4.3E-5 (.64)	3.4E-5 (.45)	1.4E-5 (.35)
Basemat Melt-Through	7.4E-7 (.011)	2.9E-6 (.04)	2.4E-6 (.06)
Late Over-pressure Failure	6.9E-6 (.10)	1.9E-5 (.25)	
Alpha Mode	1.5E-7 (.002)	2.2E-7 (.003)	1 E-7 (.003)
Early Over-pressure Failure	7.6E-7 (.011)	2.3E-7 (.003)	2 E-7 (.005)
Containment Bypass	9.1E-6 (.14)	1.3E-5 (.17)	4.9E-6 (.12)
TOTAL	6.8E-5	7.5E-5	4.0E-5

TABLE 1-5
OVERALL COMPARISON OF RESULTS OF THE NORTH ANNA IPE WITH THE SURRY IPE AND NUREG/CR-4550 (SURRY) RESULTS

	Core Damage Frequency		
Initiating Event[2]	North Anna IPE	Surry IPE	Surry[1] NUREG/CR -4550
Loss of Coolant Accident			
Small LOCA	1.0E-5	1.1E-5	1.1E-6
Medium LOCA	6.6E-6	5.3E-6	3.1E-6
Large LOCA	4.1E-6	4.6E-6	2.0E-6
Interfacing System LOCA	1.6E-6	1.6E-6	1.2E-6
Loss of Offsite Power			
Loss of Offsite Power	1.2E-5	7.1E-6	<1.5E-7
Station Blackout	8.0E-6	8.1E-6	2.1E-5
Transients			
Loss of ESGR Cooling	6.6E-6	1.8E-5	N/A
Other Transients	6.1E-6	4.8E-6	N/A
Loss of 4160 V Bus 1H	3.7E-6	-	N/A
Loss of Feedwater	1.0E-6	4.7E-7	1.7E-6
Loss of 4160 V Bus 1J	6.5E-7	-	N/A
Loss of DC Bus 1-I	1.1E-7	6.8E-7	1.4E-7
Loss of DC Bus 1-III	1.1E-7	6.8E-7	1.4E-7
Steam Generator Tube Rupture	7.0E-6	1.0E-5	1.9E-6
ATWS	4.2E-7	3.2E-7	1.4E-6
Total of Internal Events	6.8E-5	7.4E-5	3.4E-5
Internal Flooding	3.6E-6	5.1E-5	-

NOTE 1: From NUREG/CR-4550 Vol. 3 Rev. 1 Table 4.10-5.

NOTE 2: For North Anna, Hv transfer event tree (namely, consequential loss and coincidential loss of ESGR cooling) contributions to core damage frequency have been summed with those of the parent tree for comparison to Surry.

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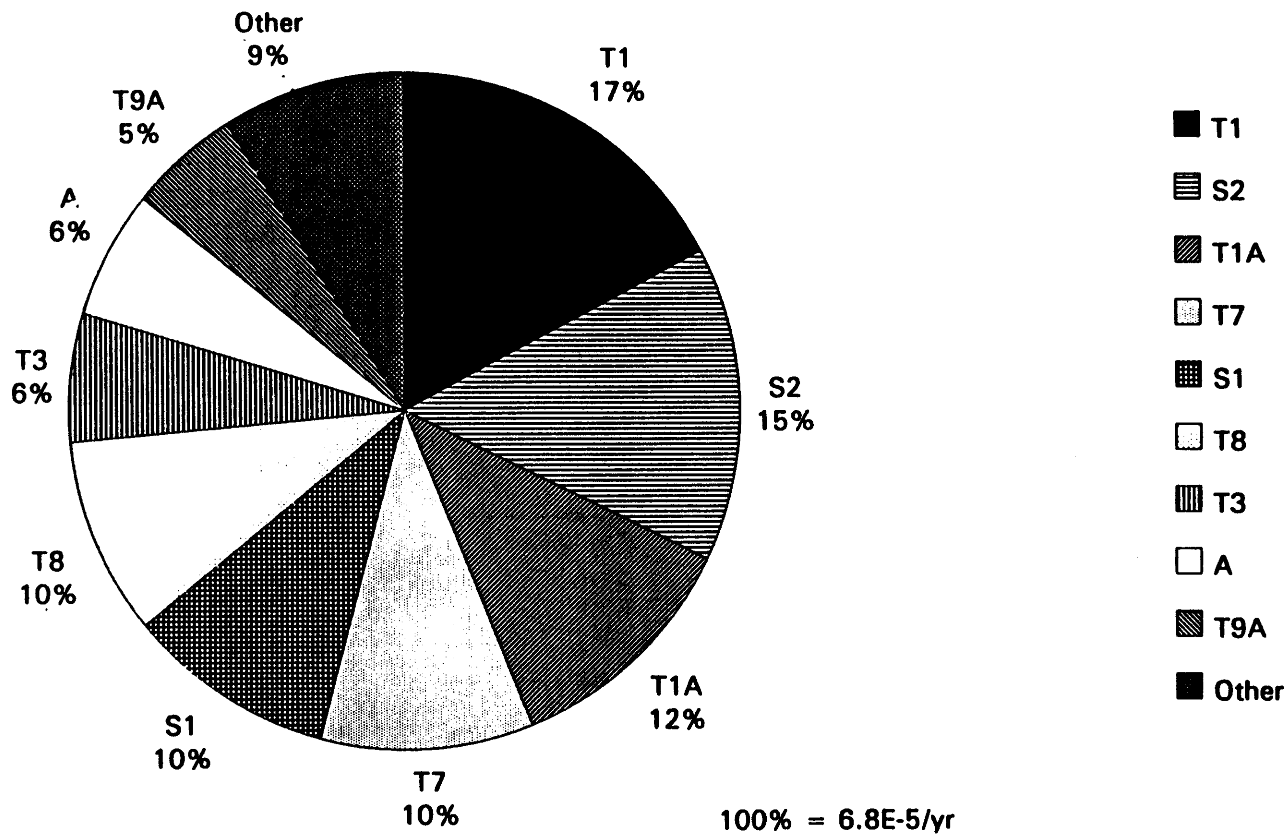

Figure 1-1
Contribution of Initiators to Core Damage Frequency

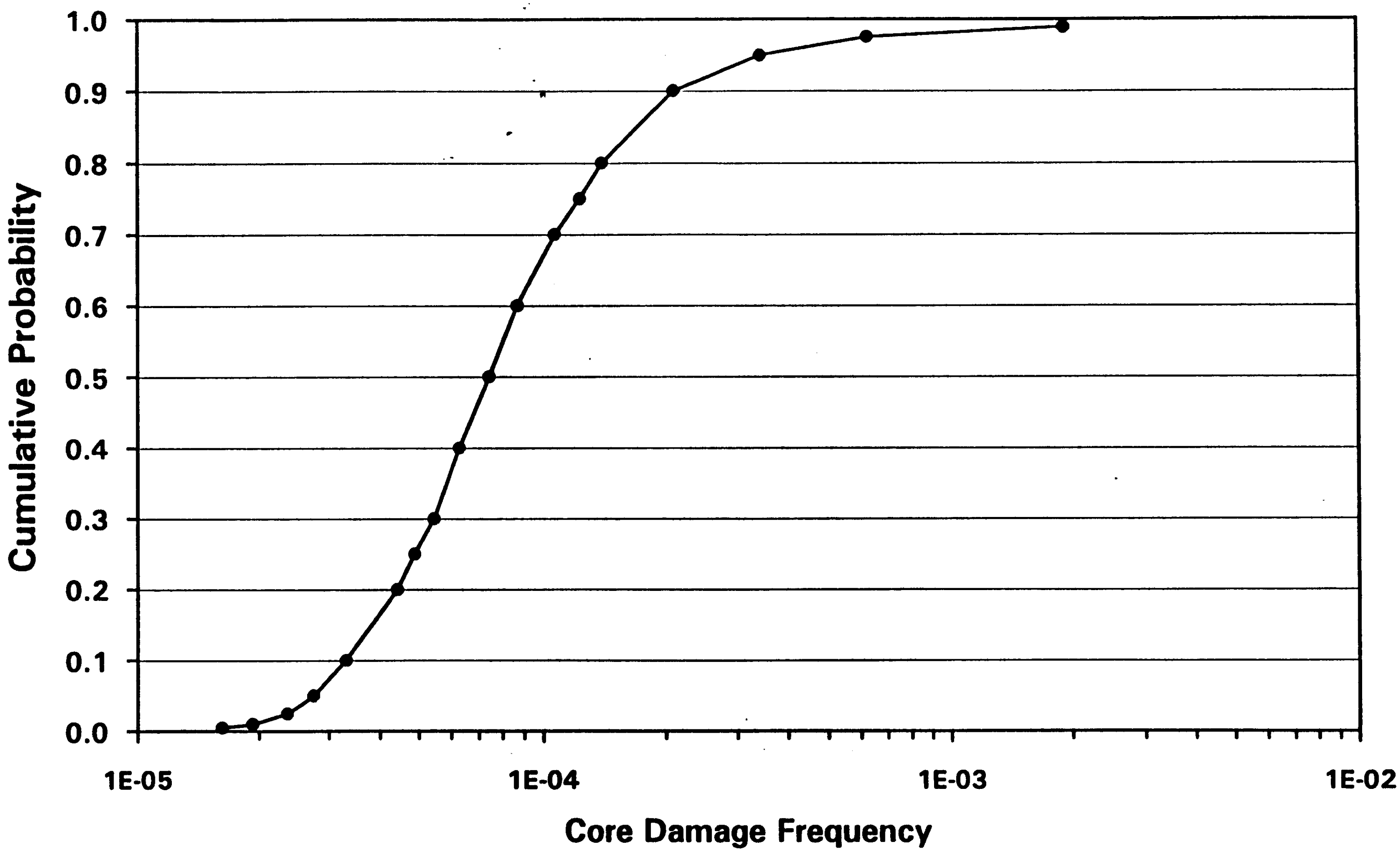

Figure 1-2
North Anna Core Damage Frequency Distribution

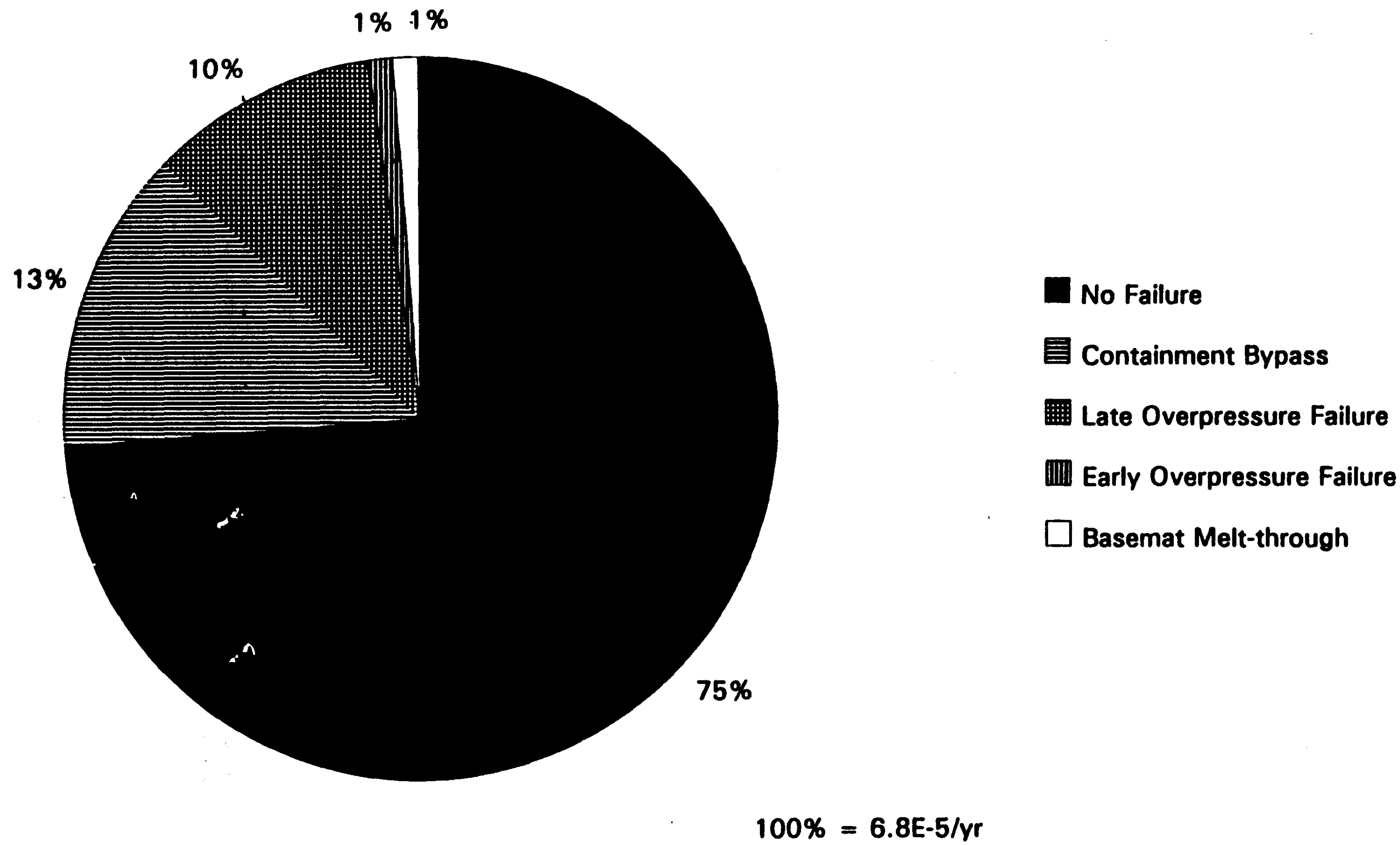

Figure 1-3
Breakdown of Containment Failure by Type

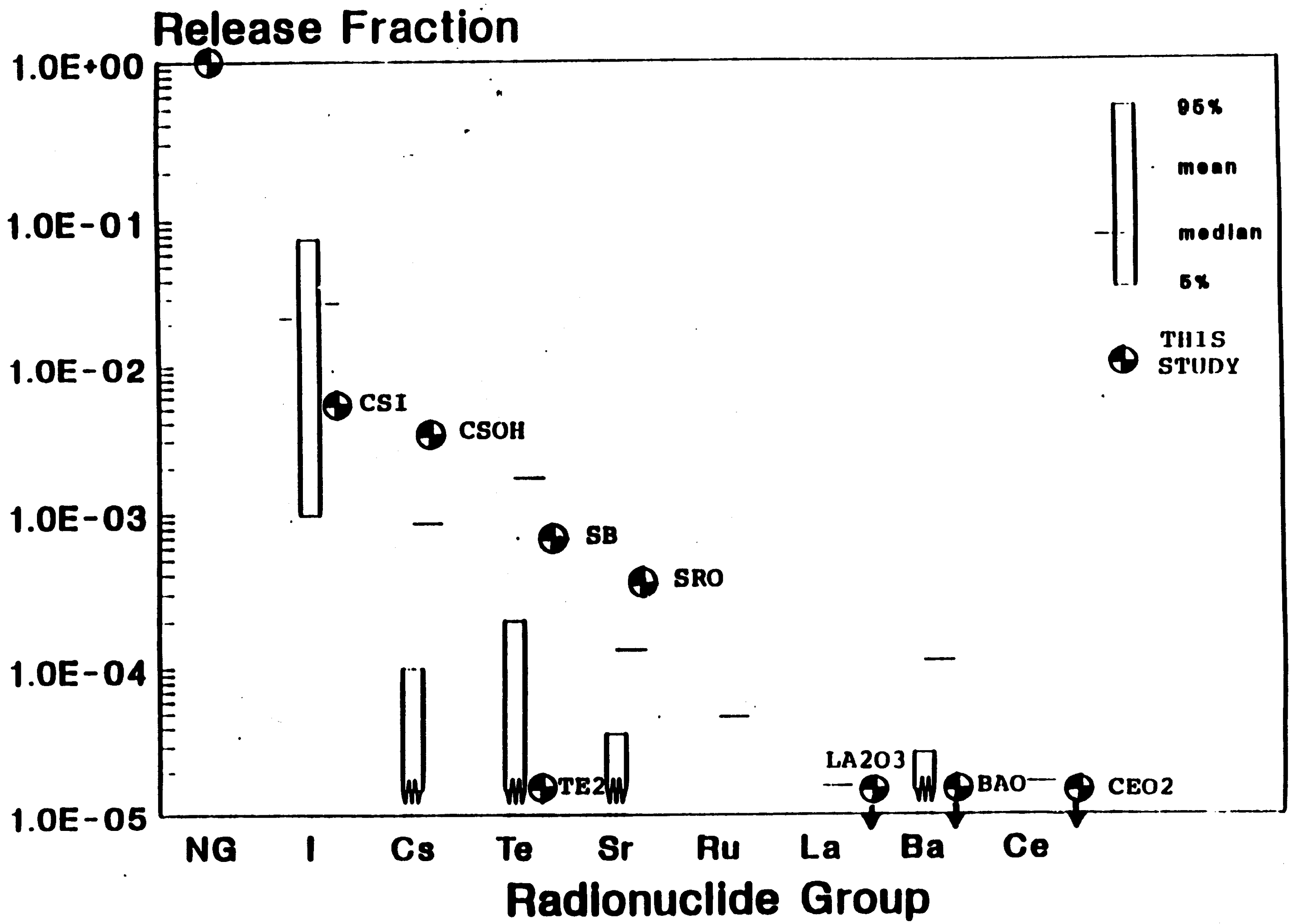

FIGURE 1-4
SOURCE TERM DISTRIBUTIONS FOR LATE FAILURE AT NORTH ANNA (COMPARISON WITH NUREG-1150)

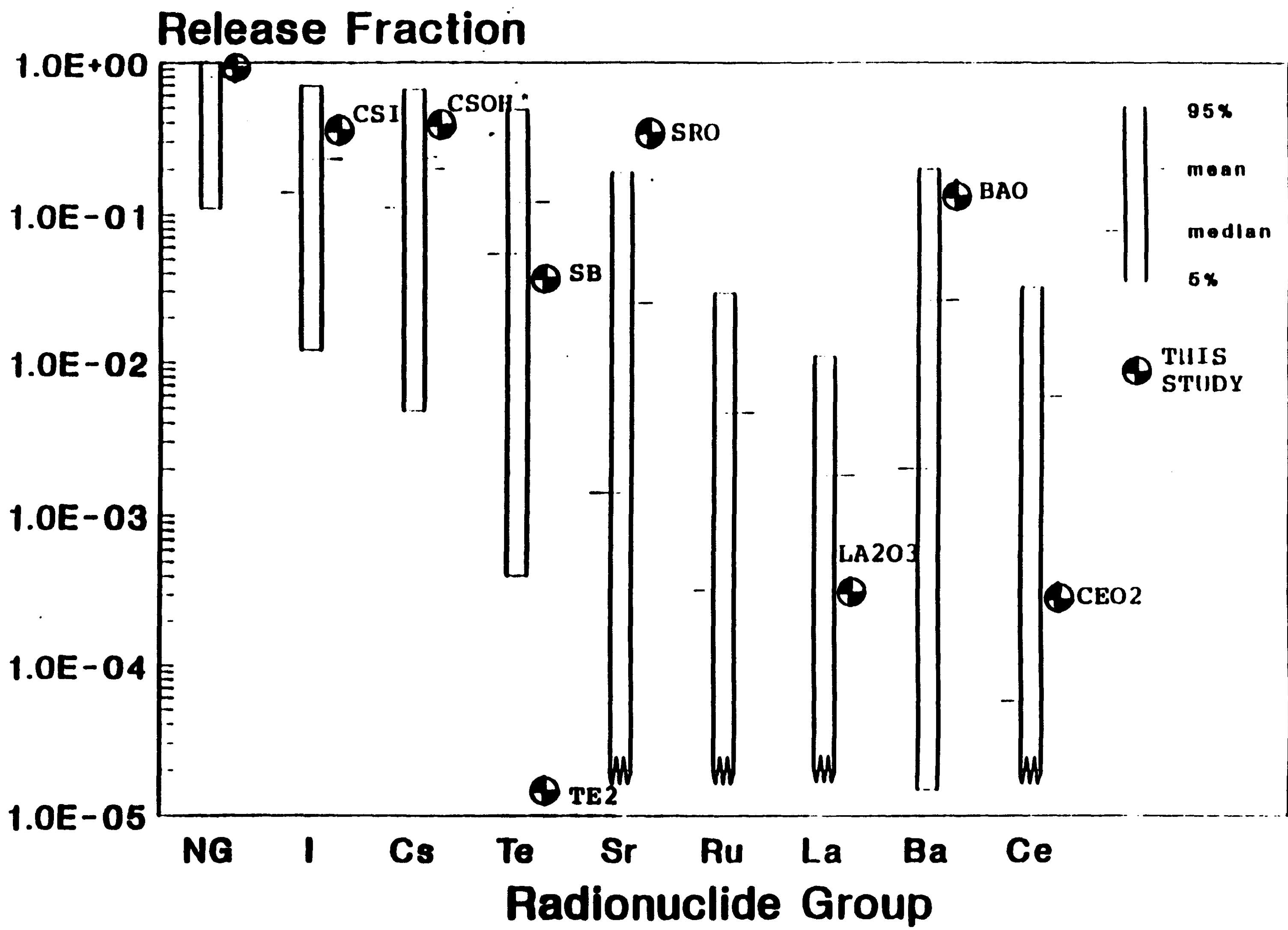

FIGURE 1-5
SOURCE TERM DISTRIBUTIONS FOR CONTAINMENT BYPASS AT NORTH ANNA (COMPARISON WITH NUREG-1150)

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