7.0 SUMMARY AND CONCLUSIONS

The results of the North Anna IPE and its comparison with the results for the Surry IPE (Virginia Power, 1991a) are discussed in the following paragraphs. As there are many similarities between the two plants, it was possible to use much of the basic analysis already performed for the Surry IPE, for example, the containment strength analysis and source term evaluation. However, in order to identify initiating events unique to North Anna as part of the level one analysis, a full failure modes and effects analysis was performed for all the supporting systems. New system models were developed in order to ensure that the final model was an accurate representation of North Anna rather than a surrogate model.

The core damage frequency is approximately 7.2E-5 per year which is made up of 6.8E-5 per year from internal events and 3.6E-6 per year from internal flooding events. The containment bypass frequency is 9.1E-6 per year and the containment failure frequency 8.6E-6 per year. The contribution by initiating event is shown in Table 7-1 and Figure 7-1. The core damage frequency distribution resulting from the uncertainty analysis is shown in Figure 7-2.

The results of this study are compared with the results of the Surry IPE and the Surry PRA reported in NUREG/CR-4550 (Bertucio, 1990) in Table 7-1.

7.1 CORE DAMAGE FREQUENCY

The core damage frequency from internal events and from internal flooding are discussed in Section 7.1.1 and 7.1.2 respectively.

7.1.1 Core Damage Frequency from Internal Events

All comparisons of Probabilistic Risk Analysis (PRA) studies must be made in an informed manner to avoid drawing inappropriate conclusions. Differences in PRA study results can be due to differences in plant design, modeling techniques and guidelines, study scope, failure data, and success criteria. A comparison of results must identify the study differences and understand the reasons for these differences. The comparison with the Surry IPE should highlight design differences or plant specific data differences as the generic data and success criteria assumptions are the same for both studies.

The total CDF from internal events for the North Anna IPE is 6.8E-5 per reactor-year and 7.4E-5 for the Surry IPE; the NUREG/CR-4550 CDF is 3.4E-5 per reactor-year. Further breakdown reveals that the expanded scope in the IPE to investigate areas that were screened away in NUREG/CR-4550 increased the CDF by 2.4E-5 per year and differences in success criteria for small breaks and Steam

Generator tube rupture (SGTR) by 1.4E-5 per year. Reasons for these changes are discussed below. Thus, with comparable scope and comparable success criteria, the CDF for the North Anna IPE is 3.0E-5 and for the Surry IPE is 3.6E-5, and that for NUREG/CR-4550 is 3.4E-5. Within this residual CDF, certain groups of sequences were changed up and down slightly due to modeling differences and data differences. A summary of this comparison is shown in Table 7-1.

7.1.1.1 Comparison between North Anna and Surry IPE

The comparison of the results for North Anna and Surry show that there is little difference in the overall core damage frequency, as would be expected from plants with very similar designs. However within this overall result there are individual differences reflecting differences in details of the design in some areas. It should be borne in mind that plant specific data has been used for pumps and maintenance activities in the ECCS system so this has given rise to minor differences in areas where no difference would be expected such as the large and medium loss of coolant accidents. Differences are discussed for each of the accident groups.

Loss of Coolant Accidents

The only difference between the plants for this group of initiating events is that in the case of the small LOCA cross connect of the RWST is less available at North Anna. It can be seen that the impact of this on the core damage frequency from small LOCAs is negligible. Data differences lead to minor variations in the core damage frequency for other LOCAs.

Loss of Offsite Power

A plant specific analysis of the Loss of Offsite Power frequency was available for Surry from the NUREG-1150 work. Since no such analysis was available for North Anna, the NUREG/CR-5032 Common Incidence Rate Model LOOP frequency was utilized. The mean frequency for Surry was found to be 7.69E-2/yr, while the North Anna IPE was quantified using a T1 frequency of 1.14E-1/yr. As a sensitivity calculation, the North Anna contribution to core damage frequency can be adjusted to reflect the T1 initiating event frequency used for Surry resulting in

$$T1_{sen} = \frac{.0769}{.114} * 1.2E\text{-}5 = 8.1E\text{-}6$$

This sensitivity shows that the difference in contribution to core damage frequency from loss of offsite power is due almost entirely to the initiating event frequency. Since the generator connections

to the offsite power lines are similar for both plants, it is expected that the North Anna result is conservative.

The station blackout contribution to core damage frequency is about the same for both stations, even with the different initiating event frequency for loss of offsite power. This result shows that the extra diesel generator at North Anna is quite significant.

Transients

Loss of feedwater transients are a little over two times more likely to lead to core damage at North Anna than at Surry. Although these sequences are not dominant, it is interesting to note that the difference is primarily due to the AFW cross-connect capability at Surry. The unavailability of AFW at North Anna is about a factor of two higher than at Surry.

At North Anna the contribution to core damage frequency from the loss of Emergency Switchgear Room cooling initiating event is over a factor of two lower than at Surry. There are system differences and modeling considerations which explain this result. North Anna has three ESGR room chillers per unit, while Surry has three ESGR chillers for the plant. However, Surry has two independent, backup chillers that can be used for recovery. Recovery at North Anna is limited to using equipment from the opposite unit, but recovery of 1 of 6 similar chillers is more likely and this has improved the North Anna recovery for loss of ESGR cooling.

Loss of DC Bus transients leading to core damage are negligible contributors to core damage frequency. However, the North Anna initiators contribute about a factor of six more to core damage frequency than those at Surry. The reason for this result is the same as for the loss of feedwater transients. That is, the unavailability of AFW is higher at North Anna, about a factor of ten, for these initiating events.

The loss of power from the 4160 V buses has been included, as these faults have historically resulted in reactor trip at North Anna. The difference between the 1H and 1J bus failure contribution to core damage frequency is due primarily to the fact that the 1H bus supplies power for two of the ESGR chillers, so the Hv function is much more likely to fail for the 1H bus than the 1J bus.

Steam Generator Tube Rupture

The initiating event frequency used at North Anna is 1.0E-2 compared with 1.6E-2 at Surry. The value of 1.0E-2 is in line with industry data at the time of this report (1992). Use of this data would bring the Surry assessment in line with North Anna.

Anticipated Transient Without Scram

The AMSAC modification has been fitted at North Anna. The joint Westinghouse Owners Group/Westinghouse Program ATWS Rule administrative process (Westinghouse, 1988) was used to determine the pressure relief requirement following an anticipated trip with failure to scram. The calculated unfavorable exposure time (UET), that is the period of time that pressure relief would not be adequate was 27.7%, compared with zero at Surry. The most likely reason for the higher UET is a combination of the higher nominal inlet temperature at HFP and larger power defects at North Anna. The net effect of this is that the ATWS contribution to core damage frequency at North Anna is slightly higher than at Surry.

7.1.1.2 Comparison of the IPE with the NRC Study of Surry

Both the North Anna and Surry IPEs identified a functional failure scenario contributing significantly to CDF that had been screened away in NUREG/CR-4550 as being a minimal contributor. The failure of Emergency Switchgear Room cooling had not been identified as important in NUREG/CR-4550, but it was calculated to have a contribution to CDF of 2.3E-5 at North Anna and 2.1E-5 at Surry. North Anna functional failures are discussed further in Section 3.4.1.2.

Small Break Size Partitioning and Frequency

The IPEs have three break sizes for LOCAs. The smallest size (S2) includes breaks between 3/8 inch and 2 inches in diameter. The frequency of S2 is 2.1E-2 per year. Timing considerations and success criteria for the small break event tree analysis were based on the upper end of the spectrum (a break size of 2 inches). This is standard practice for PRA analysis.

NUREG/CR-4550 used four break sizes, the smallest being less than 0.5 inch. The S2 and S3 break frequencies in NUREG/CR-4550 sum to 1.4E-2 per year, which is comparable to the IPE frequency for S2, but the 2 inch break size has a frequency of 1E-3 per year. The event tree analysis for the S3 break is based on timing considerations for a 0.5 inch break. The effect of this is to allow much more time for mitigation of the accident before it goes to recirculation and more time for recovery from additional failures. Specifically, Emergency Core Cooling System (ECCS) recirculation is not required for S3 breaks if AFW is operable, and the operator takes action to cool down and depressurize within 6 hours of the initiator. Also, S3 breaks with failure of High-Head Safety Injection (HHSI) have approximately 1.5 hours before core uncovery for recovery actions to be implemented. The S2 break, however, only has about 40 minutes. Thus, in NUREG/CR-4550, the smaller break size, which accounts for 93 % of small break

frequency, is allotted much more potential for mitigation and recovery.

In the North Anna IPE, the frequency of 2.1E-2 was split between the two break sizes (Fm) and allowance made for successful cooldown and depressurization in the core of the very small LOCA.

RHR Cooling and Steam Generator Tube Rupture

Both the IPEs and NUREG/CR-4550 included a SGTR initiating event. In these studies, a key assumption in the event tree analysis for this initiator was that action is required to cope with a ruptured, faulted Steam Generator. It was also assumed that further depressurization and continued HHSI are required. However, different interpretations of "successful accident mitigation" lead to significant differences in CDF between the IPE and NUREG/CR-4550 results.

According to NUREG/CR-4550, successful mitigation is achieved with depressurization to minimum Steam Generator operating conditions, continued AFW, and HHSI, with a possible cross-tie of the refueling water storage tank (RWST) from the opposite unit. The plant status at the end of the 24 hour mission time could be characterized as follows: plant condition stable, continued HHSI operation necessary for an indefinite period, and RWST capacity available for at least 48 hours. The IPEs modeled a failure of the operator to cool down and depressurize early, leading to Steam Generator filling, a failed-open valve (i.e., a Steam Generator pressure-operated relief or safety relief valve), and a bypass core damage sequence.

The IPEs required cooldown to atmospheric conditions and the operation of RHR cooling for all sequences in which Containment Building bypass has occurred. The plant status of the IPE sequence could be characterized as follows: plant condition stable and RHR operable for an indefinite period of time if a safety/relief valve fails open. The IPE requirement for RHR cooling, which was not in NUREG/CR-4550, accounts for 3.1E-6 of CDF at North Anna. Although this is a small percentage of CDF, these sequences have a significant impact on the Level 2 results. The difference in CDF between the IPE and NUREG/CR-4550 is attributed to different interpretations of requirements for successful accident mitigation.

Station Blackout

In the case of the NRC study of Surry the contribution to station blackout from sequences involving battery depletion, reactor coolant pump seal LOCA and early auxiliary feedwater failure were all higher than at North Anna or Surry resulting in the higher contribution to core damage frequency.

7.1.2 Core Damage Frequency for Internal Flooding

The core damage frequency from flooding at North Anna is only 3.6E-6 per year compared with a frequency of 5.1E-5 at Surry (Virginia Power, 1991b). The major contribution at North Anna is from service water system floods in the auxiliary building leading to loss of seal cooling to the reactor coolant pumps and the requirement to cool down and depressurize to prevent seal LOCA. At Surry, because of the completely different arrangement of circulating and service water systems in which a canal is used to provide gravity feed to the systems, the major contribution comes from floods in the Turbine Buildings and mechanical equipment room No. 3. Thus the resultant core damage frequency at North Anna is lower.

During the course of the analysis at North Anna it was found that the impact of floods in the Auxiliary Building as well as the air conditioning and chiller room could be reduced by making a few plant design changes. The changes identified following the initial analysis have been used as the basis for the final analysis. Individual changes are discussed in Section 6.2.1.

7.2 SUMMARY OF CONTAINMENT FINDINGS

The North Anna Containment Building and systems are robust with respect to the challenge presented by severe accidents. Because of the high assessed strength of the Containment Building, overpressure failure of Containment either early or late in time is unlikely. The North Anna Containment Building is operated in a subatmospheric mode. Consequently, the probability of a 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 4.7.4-4 shows a breakdown of the predicted North Anna Containment Building performance for severe accidents. The Containment results of the North Anna IPE are compared to those for the Surry IPE and to NUREG-1150 in Table 7-2. A discussion of this comparison is included in the following subsections.

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 nonisolated state; and the piping arrangement in the Safeguards Building is such that most interfacing LOCAs (Event 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 unless the Quench or Recirculation Sprays are operating.

The sprays play several roles, all of which are important with regard to source terms. They can "wash out" airborne radionuclides in the Containment Building, 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 heat removal function not only prevents long-term over-pressure failure but also minimizes outflow from the Containment Building in those circumstances when its integrity has already been breached (e.g., loss-of-isolation sequences).

The MAAP-derived release fractions (calculated for 12 of the 24 source term categories) confirm what is already known from other work (NUREG-1150, for example) that the Containment Building bypass sequences (interfacing LOCAs and SGTR) have the greatest release potential. This is because of the relative scarcity of mitigating features in the release pathways. In particular, the Steam Generator with the ruptured tube is likely to be boiled dry when core damage and fission product release begins.

The calculated release fractions generally agree 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 7-3 and 7-4. The IPE analysis 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.

7.3 COMPARISON OF THE NORTH ANNA IPE AND SURRY IPE CONTAINMENT RESPONSE ANALYSES

Table 7-2 compares the North Anna IPE results to those in the Surry IPE. The biggest difference is that the relative frequency of early Containment failures has increased at North Anna while the frequency of late failures has decreased. In addition, because of the decrease in late failures, the frequency of the Containment not failing (with the RV ruptured) has increased for the North Anna study.

The increase in the frequency of early Containment failures at North Anna is due to the fact that the ratio of the power rating to Containment volume is 17% larger at North Anna than for Surry. Therefore, calculated post-accident Containment pressures at North Anna are higher, leading to a higher probability of Containment failure (the probability of failure for a given pressure is approximately equal for both Surry and North Anna). The frequency of late Containment failure at North Anna has decreased because credit was taken for electric power recovery between the time of

vessel rupture and Containment failure. This recovery leads to the possibility of recovering Recirculation Sprays and Containment Heat removal in SBO sequences, thus reducing the probability of late failures.

7.4 REFERENCES

Bertucio, R. C., and Julius, J. A., Analysis of Core Damage Frequency, Surry Unit 1: Internal Events, NUREG/CR-4550, Volume 3, Revision 1, Part 1, Sandia National Laboratories, Albuquerque, New Mexico, April 1990.

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Virginia Power (Virginia Electric and Power Company), Internal Flooding Analysis Supplemental Report, Surry Nuclear Power Plant Units 1 and 2 for Individual Plant Examination, Richmond, Virginia, November 1991b.

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TABLE 7-1
OVERALL COMPARISON OF RESULTS OF THE NORTH ANNA IPE WITH THE SURRY IPE AND NUREG/CR-4550 (SURRY) RESULTS

Initiating Event[2]	North Anna IPE	Surry IPE	Surry[1] NUREG/CR-4550
	Core Damage Frequency		
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 coincidental loss of ESGR cooling) contributions to core damage frequency have been summed with those of the parent tree for comparison to Surry.

TABLE 7-2
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

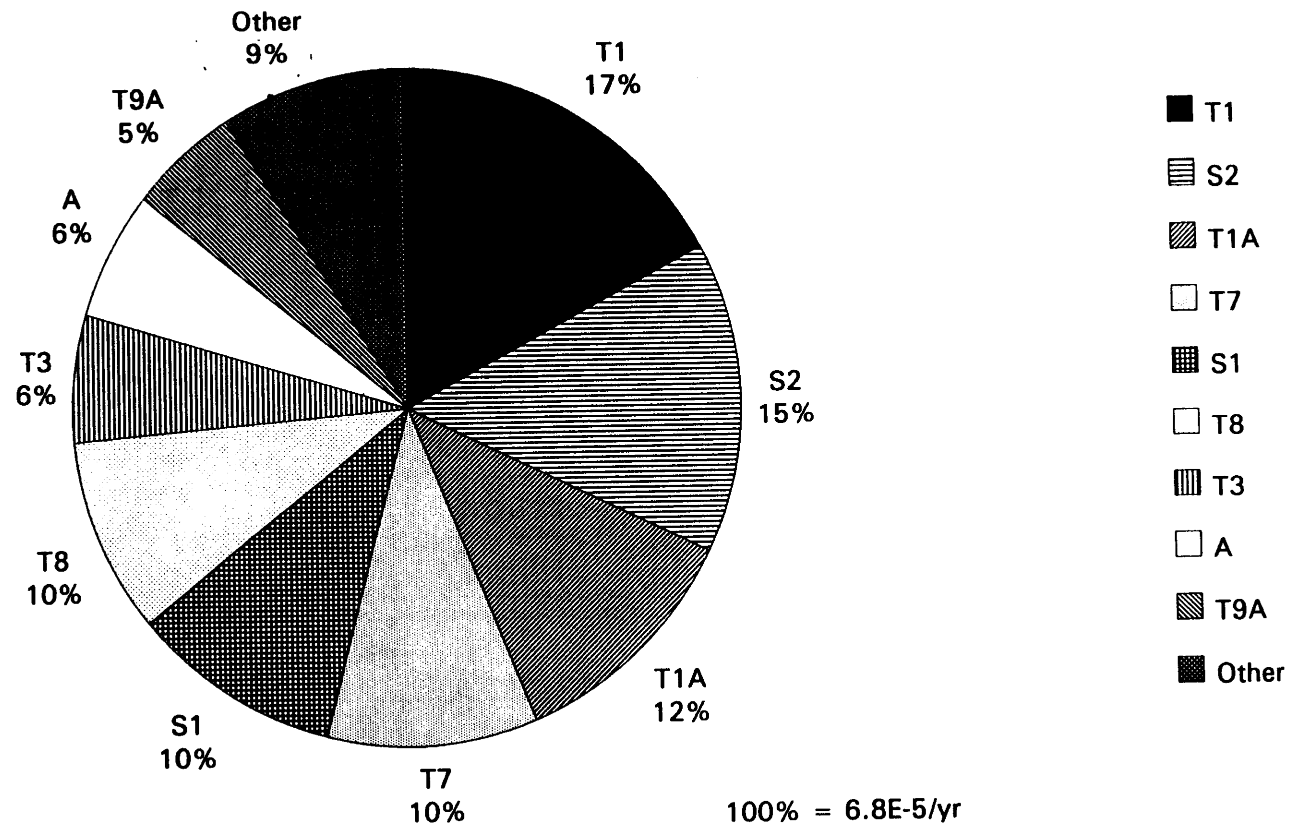

Figure 7-1
Contribution of Initiators to Core Damage Frequency

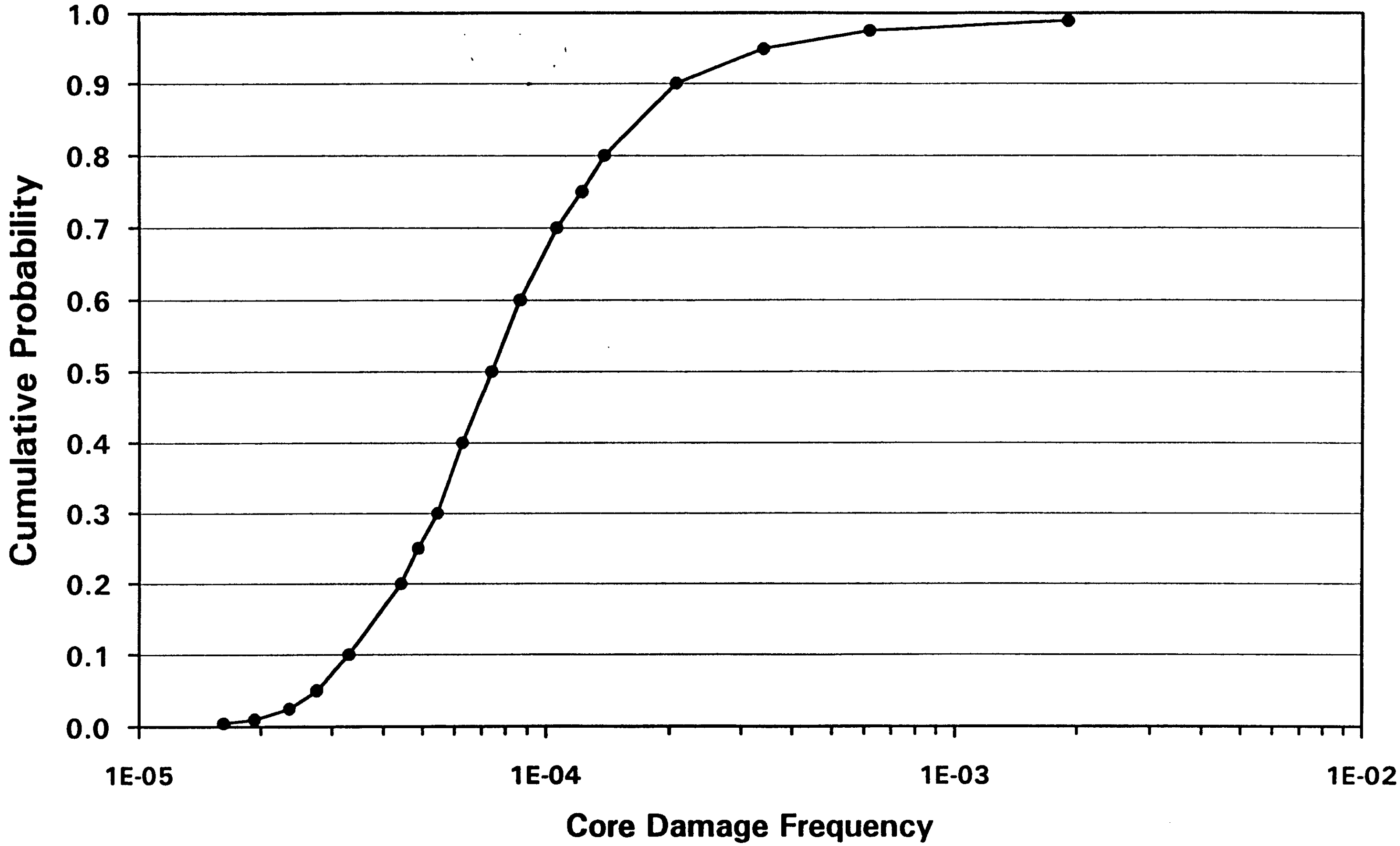

Figure 7-2
North Anna Core Damage Frequency Distribution

Figure 7-3 Source Term Distributions for Containment Bypass at North Anna

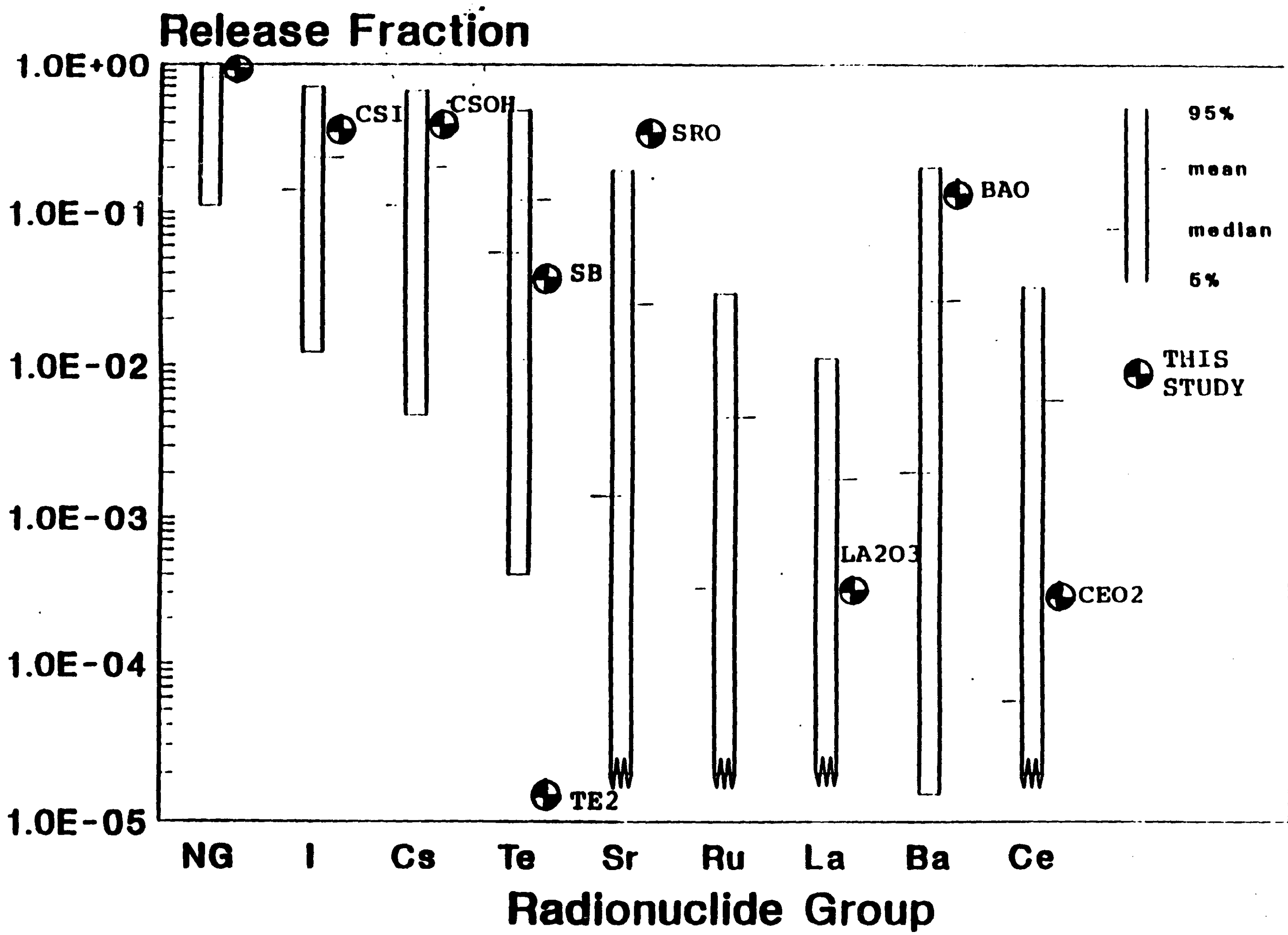

Figure 7-4 Source Term Distributions for Late Failure at North Anna