2.0 EXAMINATION DESCRIPTION

2.1 INTRODUCTION

The North Anna PRA project is aimed at meeting two objectives, satisfying the NRC individual plant evaluation requirements and providing plant models for use in maintaining the safe reliable and efficient operation of the two units. In this section it is demonstrated that the project conforms with the NRC requirements. The general methodology and the information used in the course of the study are also described.

2.2 CONFORMANCE WITH GENERIC LETTER AND SUPPORTING MATERIAL

The general purposes of the examination specified in the generic letter are as follows:

1. To develop an appreciation of severe accident behavior.

2. To understand the most likely severe accident sequences that could occur at the plant.

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

4. To reduce, if necessary, the overall probabilities of core damage and fission product releases by modifying, when appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

Each of the issues in achieving this overall result and the reporting of this work are discussed in the following sections in the order presented in the generic letter.

2.2.1 Examination Process

In order to ensure that the Virginia Power personnel are fully conversant with the IPE and will be in a position to make maximum use of the subsequent living PRA the study has been performed at the Virginia Power offices by a joint Virginia Power/Halliburton NUS Corporation project team. Three Virginia Power engineers were dedicated full time to the PRA and have participated in performing all the tasks described in the general methodology section below. A fourth Virginia Power engineer assisted in the Containment Building evaluation analysis and performed the MAAP analyses in support of this task.

In order to ensure that the plant truly reflects the design and operating experience, review of the following work products was performed by North Anna plant staff.

1. The significant system models and the assumptions made concerning system design and operation were reviewed by individual station engineers.

2. The human reliability modeling and assumptions were reviewed by the simulator training staff and the Human Performance Evaluation System (HPES) coordinator.

3. The accident sequence delineation (i.e., event tree functions) were reviewed by a Shift Technical Advisor).

4. The potential improvements were reviewed by station management and the appropriate discipline area at the station, i.e., procedure development, training, maintenance, operations.

Additionally, an independent peer review of the study methodology and results was completed. This peer review team consisted of station personnel and independent PRA consultants. See Section 5 for peer review details.

2.2.2 External Events

The current study includes only core damage and fission product release assessment following internal events and internal flooding. Plant walkdowns have been conducted for the flooding analysis. The information gained from these walkdowns has been included in the appropriate analysis files and will be available when performing external event analyses.

2.2.3 Methods of Examination

The approach used for the North Anna IPE is Method 1, that is a modified Level 2 PRA using current methods and information including a Containment Building performance analysis which addresses the issues in Appendix 1 and Supplement 3 to letter 88-20. The PRA follows the method described in NUREG/CR-4550 amplified by more detailed procedures for each individual task in order to comply as closely as possible with the quality assurance requirements identified in 10CFR50 Appendix B. The PRA is based on a clearly defined plant status, which was current at the time of performing the systems analysis in 1991. As the QA for the development of the plant model has required full recording of all documentation used, it will be a straight-forward process to update the model if so desired in the future. The methodology is described in more detail in Section 2.3.

Details of the approach to the Containment Building analysis are given in Chapter 4 of this report.

2.2.4 Resolution of Unresolved Safety Issue USI-45

The PRA specifically addresses the performance of the shutdown decay heat removal systems following transients from high power or loss of coolant accidents. The overall contribution to core damage is assessed and the individual system vulnerabilities identified.

2.2.5 Severe Accident Sequence Selection

The results of the analysis which are reported in Sections 3.4 and 4.7 have been based on the screening analysis defined in Appendix 2 of the generic letter and amplified in the guidance document (NRC, 1989a). The identified vulnerabilities, unique safety features, and potential improvements are discussed in Chapter 6.

2.2.6 Documentation of Examination Results

As it is intended that the IPE shall be widely used within Virginia Power in the future, considerable care has been taken in developing the documentation for the study. The documentation of the completed study has three components. The first is this report which details the results and findings of the PRA given the current plant status. The second is the PC-based North Anna plant model created with the NUPRA/NUCAP+ software. This model can be updated when changes in design or procedures are made. The third is the set of quality assured analysis files which document all the assumptions, boundary conditions and documents used in developing the plant model on the PC. It is these documents which form the basis for the living PRA in that they provide the basis for comparison of new designs, data, operational procedures, training, testing, etc. with that used for the base scope study and, thus, identify what changes need to be made to the model.

One of the key items in this third tier of documentation is the document data base which relates all the documents (such as the Updated Final Safety Analysis Report (UFSAR), P&IDs, etc.) used and reports produced in the study to the appropriate area of work.

This report follows the format recommended in the guidance document.

2.3 GENERAL METHODOLOGY

The methodology used to perform the analysis for the Individual Plant Examination is designed to answer the following questions.

1. What events at the plant will result in a reactor trip or the necessity for the reactor to trip?

2. How can decay heat be removed following reactor trip?

3. How often (with what frequency) will core damage occur as the result of failure to remove decay heat?

4. How likely is containment failure to occur following core damage?

5. What fission product release is likely to occur and how often?

6. What are the significant contributors to the frequency of core damage, containment failure and fission product release?

The answers to the above questions are determined by carrying out the following basic steps shown in the flow diagram in Figures 2-1 and 2-2.

1. Identifying the causes and frequency of plant trips (Initiating Events).

2. Developing a model which represents the many different ways in which decay heat removal and containment systems can fail following the various initiating events (event trees and fault trees).

3. Assembling data for the component failures, maintenance unavailabilities, operator actions, etc. included in the event and fault tree models.

4. Quantifying the accident sequence and core damage frequency and performing importance analysis to identify the dominant contributors to the accident sequences (quantification, importance analysis).

5. Performing a Containment Building strength analysis to determine the likelihood of failure over a range of pressures and temperatures.

6. Determine the range of pressures and temperatures in the Containment Building for a range of possible accident sequence scenarios. Each scenario is identified by a specific set of conditions at the time of core damage known as the plant damage state.

7. Combine 5 and 6 to determine the likelihood of Containment Building failure and the magnitude of the fission product release.

The way in which each of these steps is performed is discussed in the following paragraphs. In order to ensure that the analysis was as complete as possible and that all the interfaces between the various sub-tasks of the PRA were clearly recognized, the first step in performing the analysis was to develop a set of comprehensive procedures to identify the way in which the analysis in each sub-task was to be performed. The procedures used are listed in Table 2-1.

2.3.1 Initiating Events and Success Criteria

Initiating event information was derived from the plant operating experience for the past 10 years, from past studies of North Anna and from NRC studies of other Westinghouse plants. Particular attention was paid to the Failure Mode and Effects analysis of support systems which resulted in initiating events associated with loss of Emergency Switchgear Room cooling, loss of Service Water, and loss of alternating current (AC) and direct current (DC) bus bars. The interaction between events occurring at Units 1 and 2 and events which would cause a plant trip at both units were also investigated.

The safety functions required in response to each of the initiating events were defined and the systems which could perform these functions were identified. These success criteria were determined by reviewing earlier analyses, the UFSAR, and Westinghouse analyses and by performing thermal hydraulic analyses using the MAAP code to confirm these requirements and to obtain estimates of timing.

Following the identification of the success criteria for each initiating event, it was possible to group together events which had similar system success criteria in order to reduce the number of event trees to be developed.

2.3.2 Accident Sequence Development

An event tree was developed for each initiating event or event group using the information developed for the system success criteria and including the operator actions identified in the various emergency operating procedures (EOPs). An individual path through each event tree (an accident sequence) then specifies the combination of system failures and successes and operator actions which will lead either to successful core cooling or to core damage through lack of cooling and failure of containment systems. The use of this information in the development of the plant damage states is described in Section 2.3.8.

In order to clearly define the boundary conditions in terms of timing for each sequence, it was necessary to perform a number of MAAP analyses. The results of these analyses enabled the timing of

actuation signals and the time available for operators to perform back up or recovery actions to be identified. The development of human error probabilities is briefly discussed in Section 2.3.6. Details of the human reliability analysis are presented in Section 3.3.3 and Appendix D.

2.3.3 Data Base

As both units at North Anna have been operating for more than ten years, it was decided to put a major effort into the acquisition of plant specific data. This data has been used to determine the maintenance unavailability for front line and important support systems, some initiating event frequencies, and key component failures. Generic data has been used for those items where plant specific data was not easily obtainable, either because they occur so infrequently or it was not possible to obtain estimates from the plant records.

2.3.4 Dependent Failures and Interaction

In addition to the failures directly associated with the components in the system performing the safety function, it is necessary to model the dependent failures. Direct functional dependencies (i.e., electric power, pump bearing cooling, initiating signals, etc.) are modeled by developing a dependency matrix for each component and including these dependencies in the system fault tree. The system performance is then evaluated by linking the fault tree for the supporting system into the appropriate point of the system fault tree. Appendix A provides a description of all the system models as well as the dependency matrices for all support systems.

Multiple failures of some components may occur as the result of influences which it is neither possible to identify or explicitly model in the fault tree for the system failure. In addition to the above there may also be interactions between systems as the result of location or the fluid being pumped by the system. An attempt has been made to model such failures by including common cause events in the fault trees for components which are assessed to be subject to such common failures, for example diesel generators, motor driven pumps, motor operated valves, and check valves. The quantification of these events has been based on a review of both plant design and industry data. The common cause failure basis events are included in the fault trees in Appendix A. The derivation of the common cause data is described in Section 3.3.4 and Appendix C.6.

The dependency of systems on environmental conditions has been included in the analysis. For each system the running time and room temperature have been evaluated over the range of accident

scenarios to determine if room cooling is required. The temperature of the water to be recirculated in the Containment Building has been determined for all the LOCA events with and without Quench Spray (QS) in order to establish the operability of the Low Head Injection and Recirculation Systems in terms of net positive suction head (NPSH) or absolute water temperature. This dependency is discussed in the individual systems descriptions in Appendix A.

The dependencies between initiating events and systems required to prevent core damage or the release of fission products are modeled in the event and fault trees. Similarly, the dependency of a function in an event tree on the success or failure of a previous function has been modeled. This is accounted for in the development of the event trees described in Sections 3.1.2 and 3.1.3 and in the quantification of each function described in Section 3.3.5 and Appendix B.

2.3.5 System Modeling

As insufficient data exist for determining the probability of failure of each system directly, it is necessary to model the system in a logical way, breaking it down to individual components for which failure data are available or can be estimated. The basic model used in this study, as is in other PRAs, is the fault tree.

The systems for which fault tree models are required are defined in the development of the success criteria and event trees. In addition to the front line systems, the list includes the support systems such as electric power, cooling water, control air, and instrumentation and control. For each system a fault tree has been developed either to the component level or, if it is not important relative to core damage frequency, to a lesser level of detail but always including the dependency on supporting systems.

Such methods allow the analyst to identify the way in which system failure may occur, assign failure probabilities to individual plant components (e.g., pumps, valve, relays, bus bars), and human actions related to the system's operation. Unavailability due to test or maintenance is also included. The failure probabilities of individual components of the model are combined to give the overall system failure probability.

2.3.6 Human Reliability Analysis

System and safety function failures may result from the operators failure to perform certain actions or the performance of incorrect actions. Therefore operator actions are included in the event and fault trees. Two types of actions are identified.

1. Errors occurring prior to the plant trip which result in equipment (e.g., valves) left in the wrong position at the time of the demand on the system or result in miscalibration of instruments such that a system is not initiated at the proper time.

2. Post trip errors which result in the operator failing to take the correct action in response to the plant state.

The second group can in fact be broken down into the following three subgroups: failing to take the correct action in accordance with procedure, taking the incorrect action and failing to take appropriate recovery actions following system failures.

A systematic approach was used to perform the human reliability analysis. There are three essential steps in this method. The first is identification by the event and fault tree analyst of the operator action that can occur which will impact the operation of the system either before or after the plant trip.

The second stage is to evaluate each action in terms of how the action would be performed for various combinations of prior system failures, how long the operator has to perform the action, what cues he has to perform the action, and how it relates to the procedures. At this stage it may be necessary to modify the event or fault tree in order to more accurately model the operator action. It is also necessary to examine the combinations of operator errors that occur in a given sequence of events in order to determine the relationship between success and failure of one action to the success and/or failure of the second and/or third action in the combination.

The third and final stage is the quantification of the human error probability basic events included in the event and fault trees. The quantification of pre accident operator actions followed the guidelines of NUREG/CR-1278. The quantification of post accident operator actions included a time dependent probability that the operator could complete the task in sufficient time to prevent core damage, and a time independent probability determined from NUREG/CR-1278. This is achieved by a detailed review of test, maintenance, and operating procedures. A summary of the HRA results may be found in Section 3.3.3 and the detailed analysis in Appendix D.

2.3.7 Accident Sequence Quantification

In this step the information gathered in the preceding steps is used to quantify the individual accident sequences. In the NUPRA code quantified, linked fault trees are developed for all the events in the event tree. Care is taken to ensure that for each sequence the system fault tree used reflects the boundary

conditions in terms of initiating event and actuation conditions appropriate to that sequence. Each sequence is then quantified by combining the results from the linked fault trees taking care to ensure that both success and failure paths are accounted for in each sequence. The final result is entered into a data base which stores the frequency of each sequence, the various functional failures and successes, and a list of cut sets of the failure contributors to the sequence.

2.3.8 Plant Damage State Analysis

The interface between Level 1 and Level 2 is the set of plant damage states which reflect the operability of plant systems and Containment Building systems, as well as the Reactor Coolant System state at the onset of core damage. In order to group the individual core damage sequences into the various plant damage states, it is necessary to define the grouping criteria. The criteria have been developed based on those aspects of plant and Containment Building systems, or reactor state, at the onset of core damage which have a significant impact on the accident progression and ultimate fission product release. This information is derived from previous level 2 studies performed for PWRs.

2.3.9 Risk Contribution Evaluation

The results of both the accident sequence quantification and the source term releases have been analyzed to identify the dominant contributors to each in terms of initiating events, system and component failures, operator actions, containment capacity, and phenomenological sensitivity. This has been achieved by performing three sets of analyses: sensitivity, importance, and uncertainty. The use of importance analysis leads to the identification of the dominant contributors to the frequency of an accident sequence. The identification of dominant or significant contributors such as an initiating event, component failure, operator action, maintenance item, or Containment Building failure mode, leads to the identification of the areas in which improvements would have the most impact. The important contributions to decay heat removal failure and the evaluation of these potential vulnerabilities is addressed in the risk contribution evaluation task.

2.3.10 Dual Unit Interactions

The North Anna Power Station is a two unit station with several shared systems. "Shared system" is defined here as a single system which provides the same function/service to both units. In addition, several systems that are not shared have cross-tie capability to the other unit. This type of configuration gives

rise to dual unit interactions which must be addressed in the IPE. For the purpose of the IPE, five types of dual unit interactions have been identified. They are:

1. A single event that can cause reactor trip at both units.

2. A single event or series of events that can impair safety system operability at both units.

3. An event at one unit that can cause reactor trip at the other unit.

4. Event(s) at one unit that can impair system operability at the other unit.

5. Loss of a system or function at one unit that can be recovered by supplying that function from a similar system at the other unit.

In general, types 1, 2, and 5 have been systematically searched for and included in the event trees, fault trees, and recovery models as appropriate. Types 3 and 4 have been searched for by reviewing events that have occurred at the two units as well as from FMEAs and are included if considered to be important to the core damage profile.

In general, Unit 1 event trees and fault trees do not have any events that are occurring at, are confined to, or generated from systems and components solely associated with Unit 2. Events in shared systems that have the same effect on both units are considered for inclusion in the event trees and fault trees as appropriate. Modeling of cross-ties will consider failure of equipment at Unit 2.

Specifically, the following guidelines were followed when modeling dual unit interactions:

1. Events that cause both reactors to trip were looked for and specifically identified in the Initiating Event (IE) Task. For these initiators, the success criteria for the station as a whole was considered. Modeling of system cross ties during these initiators must consider that both units may require safety systems as a result of the initiator.

2. Operator errors on one unit causing problems on the other unit were not considered as none have been specifically identified from IEs that have occurred.

3. Dual core damage frequency was not calculated. Source terms were evaluated for a single unit.

4. The impact of any accident condition at one unit on the other unit was not considered. The impacts of normal operational requirements at one unit on success criteria for the other unit were considered. An example of the type of interaction that was not considered would be the impact on Unit 1 of Auxiliary Building habitability problems caused by an accident at Unit 2. Similarly, the impact on Unit 1 equipment from high energy breaks on Unit 2 was not considered.

5. Cross-tie of systems was modeled where appropriate. In order to be appropriate, the following criteria must be met:

   a) Cross-tie must be directed by a procedure applicable to the accident being modeled.

   b) Success criteria must consider the demand, if any, at the other unit.

   c) Mode status of the other unit and its impact on equipment operability must be considered (for example, cross-tie of other unit's HHSI Pumps is only possible when the other unit is shutdown.

   d) Availability of standby equipment on the other unit must be ensured by a Technical Specification or, if no Technical Specification is applicable, realistic estimates of unavailability must be used.

   e) When modeling cross-ties from the other unit to recover common cause failure of equipment, the potential for common cause failure of similar equipment across both units must be considered.

6. Cross-ties from the other unit were modeled with fault trees, showing explicit equipment unavailability and operator actions.

2.4 INFORMATION ASSEMBLY

The first step in the performance of the IPE was the assembly and review of the documentation necessary to perform the study. This consisted of the UFSAR, the IPE performed for Surry, an earlier fault tree analysis of the Service Water system, the piping and instrumentation diagrams, normal and emergency operating procedures, control room logs, and selected thermal hydraulic analyses performed by Westinghouse and Stone & Webster. During the course of the study a number of earlier analyses performed for specific areas (such as room heat up calculations, flooding rates and NPSH calculations) were used. Each of these documents is referenced in the specific area of work to which it is applicable.

The members of the project team from Virginia Power are familiar with the North Anna plant. The remaining members of the team made a number of site visits early in the project for system walkdowns, data collection, and observations of operator training and performance in the simulator in order to ensure familiarity with the areas of work for which they were responsible. Further visits were made during the project to review specific areas of modeling and provide further information for the internal flooding, containment evaluation, and human interactions tasks. For each of these visits, the personnel present, the purpose of this visit, and the information acquired during the visit is documented in the appropriate analysis file.

One of the major concerns when developing a living IPE/PRA model is that it represents the as-built, as-operated plant. In order to ensure that the North Anna model, described in this report, meets this requirement the following specific actions were taken:

a) The study was performed at Virginia Power's Innsbrook Technical Center in Richmond, Virginia where the Virginia Power PRA engineers are based. Design documentation used was acquired from Records Management which holds the latest quality assured documents.

b) Analysis files and a PRA document data base were set up for each phase of the model development to ensure that the documents used and decisions made on the basis of a given document were recorded. This ensures that comparison between the model and subsequent design change packages (DCP) can be made in a controlled manner.

c) Mechanical systems at North Anna have been walked down by the responsible system engineer. These system engineers reviewed many of the system models for correctness and assumptions concerning system operations.

d) The current set of operating procedures were used in performing the human reliability analysis. Certain of the actions were observed in the simulator and discussed with the training staff.

e) Data was acquired directly from control room logs, Licensee Event Reports and Deviation Reports.

f) Visits were made to the plant to identify the flooding sources and flood propagation pathways between the various flood areas.

g) The Containment Building strength evaluation was based on a Surry analysis performed by the architect/engineer (A/E). The Surry analysis was reviewed and found to be applicable to North Anna. Site visits, drawings and the

Videodisc Information Management System (VIMS) were used to confirm the layout and arrangement of Containment Building systems.

h) In addition to reviews of parts of the IPE model made by the appropriate system engineers, intermediate reviews of the work products and a review of the draft report were performed by station personnel from the operations, training and engineering departments as part of the independent review.

2.4.1 Plant Layout and Containment Building Information

The North Anna Power Station is a two-unit site. Each unit, designed by the Westinghouse Corporation, is a three loop pressurized water reactor rated at 2893 MWt and is housed in a subatmospheric Containment Building designed by Stone & Webster Engineering Corporation. The balance of plant systems were engineered and built by Stone & Webster Engineering Corporation. Unit 1 started commercial operation in June 1978. Some important system design features of the North Anna plant are described in Table 2-2.

The applicability of the unique features associated with the subatmospheric Containment are discussed in Section 4.1 of the report. Another feature of the design, which impacts the Interfacing system LOCA, is the low pressure pipe work in the Safeguards Building. This is a relatively small structure separate from the Auxiliary Building which houses the low pressure safety injection pumps, the Outside Recirculation Spray (ORS) Pumps, and the associated piping and valves.

2.4.2 Review of PRA Studies Associated with North Anna

The North Anna plant is very similar to Surry; and, therefore, PRAs performed for Surry have been reviewed. The Surry Nuclear Power Plant has been subject to two previous PRAs, WASH-1400 (NRC, 1975) and as part of the NUREG/CR-4550 (Bertucio et al, 1990) and NUREG/CR-4551 (Breeding et al, 1990). Other studies reviewed for information pertaining to either internal events, internal flooding or Containment Building evaluation analysis were those performed for the Westinghouse three loop plants. The Surry IPE was used as the basis for the performance of the North Anna IPE.

For the Level 1 analysis, other PRA studies were reviewed to identify the dominant contributors to core damage and any plant specific vulnerabilities associated with equipment or systems in these plants which are similar to equipment or systems at North Anna. In Table 1-5 the results of this study are compared with the results of the NUREG/CR-4550. Important insights from some of

these PRAs are listed in Table 2-3 and the core damage frequency for a range of plants in Table 2-4 and Table 2-5.

For the Level 2 analysis, prior studies were reviewed to identify 1) plant systems and components important to accident progression and mitigation, 2) phenomenological events and processes which are important to Containment failure and source term definition, and 3) potential Containment loading and failure mechanisms and modes. These studies were reviewed to determine the criteria and parameters used in the definition of plant damage states and the physical processes, systems operation, operator actions, etc., which were considered of sufficient importance to Containment accident progression to be included as a Containment event tree heading in these analyses. In Sections 4.3.4 and 4.7.4, the results from the Containment analyses are compared with the results from the Surry IPE and NUREG-1150 analyses.

2.5 PLANT FAMILIARIZATION

In order to ensure that all members of the project team were familiar with the plant, a plant familiarization task was performed. This consisted of reviewing the system descriptions and the recently completed IPE study of the Surry plant. A plant visit was made at the beginning of the study so that each analyst could familiarize himself with the configuration of the systems for which he was responsible.

2.5.1 Human Reliability Analyses

A number of sequences containing significant operator action were identified. Performance of these actions in the simulator was observed and timing measurements taken by the project staff during the course of training. This work is fully reported in Appendix D.

2.5.2 Data Analysis

Plant specific operational data were acquired from the Station Records identified during visits to the plant. The information recorded in the Control Room log was discussed as necessary with the Control Room staff. Discussions were also held with maintenance staff and system engineers to ensure that all possible sources of data were identified.

2.5.3 System Modeling

The lead system analyst from Virginia Power visited the plant to discuss key system models with the responsible system engineers. As complete system walkdowns have been conducted by system

engineers, they were able to advise on the accuracy of each system model. Any queries arising out of this meeting were resolved by examining the system in the plant.

2.5.4 Dependencies

The potential for system interactions was reviewed during the initial plant walkdown and by the use of the VIMS System at the headquarters. This system which consists of a very large number of color photographs on a video disk allows areas inside and outside Containment to be examined in detail.

2.5.5 Internal Flooding

Plant visits were made to review piping arrangements and potential flood progression paths from one area to another. The VIMS System was also used to evaluate details of piping and valve layout in areas not accessible during the plant visits.

2.6 REFERENCES

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Virginia Power (Virginia Electric and Power Company), Probabilistic Risk Assessment Surry Nuclear Power Plant Units 1 and 2 for the Individual Plant Examination, Richmond, Virginia, August 1991.

Virginia Power (Virginia Electric and Power Company), Updated Final Safety Analysis Report (UFSAR), Revision 21, Richmond, Virginia, July 1992.

TABLE 2-1
LIST OF PRA ANALYSIS PROCEDURES

Procedure	Number
Project Plan	272M
Quality Assurance Plan	0581
Initiating Events and Success Criteria	322M.N
Accident Sequence Development	321M.N
Data Base Analysis	319M.N
Dependant Failure Analysis	323M.N
System Modeling	320M.N
Human Interactions Analysis	324M.N
Containment Evaluation	326M.N
Quantification of Core Damage Frequency	325M.N
Internal Flooding	327M.N
Risk Contribution Evaluation	328M.N
Final Report	330M.N

TABLE 2-2
SUMMARY OF DESIGN FEATURES: NORTH ANNA UNIT 1

1.	Coolant Injection System	a.	High-Pressure Safety Injection and Recirculation System with 2 trains and 3 pumps. System provides normal makeup flow with crosstie to Unit 2.
		b.	Low-Pressure Injection and Recirculation System with 2 trains and 2 pumps.
2.	Heat Removal Systems	a.	Power Conversion System.
		b.	Auxiliary Feedwater System (AFW) with 3 trains and 3 pumps (2 MDP, 1 TDP)*.
		c.	RHR System with 2 pumps and 2 trains inside Containment.
		d.	2 pressurizer power operated relief valves.
3.	Reactivity Control Systems	a.	Control rods.
		b.	Chemical and Volume Control (CH) System.
4.	Key Support Systems	a.	DC power provided by 2 trains of batteries.
		b.	Emergency AC power provided by 2 dedicated diesel generators (both self-cooled).
		c.	Component Cooling Water provides normal cooling to RCP thermal barriers.

TABLE 2-2 (Continued)
SUMMARY OF DESIGN FEATURES: NORTH ANNA UNIT 1

		d.	Service Water is normally fed from a reservoir. Lake Anna serves as an alternate supply of Service Water. The SW system provides heat removal from Containment following an accident.
5.	Containment Structure	a.	Subatmospheric (10 psia).
		b.	1.82 million cubic feet.
		c.	45 psig design pressure.
		d.	Reinforced concrete.
6.	Containment Systems	a.	Quench Spray injection initiated at 28 psia with 2 trains and 2 pumps.
		b.	Inside Recirculation Spray (IRS) initiated at 28 psia with time delay with 2 trains and 2 pumps (both pumps inside Containment).
		c.	Outside Recirculation Spray initiated at 28 psia with time delay with 2 trains and 2 pumps (both pumps outside Containment).
		d.	The Inside and Outside Spray Recirculation Systems provide the only form of Containment heat removal after a LOCA.

*MDP - motor driven pumps
TDP - turbine driven pumps

TABLE 2-3
SIGNIFICANT INSIGHTS

NUREG/CR-4550 Surry PRA (1990)

1. Station Blackout accidents are the dominant contributor to core melt (68%).

   Significant contributors: Battery depletion
   Seal LOCA
   Diesel Generator failures

2. Loss of Coolant accidents contribute 15%, Interfacing LOCA, 4%.

Oconee Unit 3 (EPRI, 1984)

1. Turbine Building Floods are dominant sequences (40%).

2. Transients contribute 12% and LOCAs about 5%. Other external events contribute the remainder.

TABLE 2-4
COMPARISON OF PRA RESULTS FOR INTERNAL EVENTS

Power Station	Core Damage Frequency (per year)
Three Mile Island	4.4E-4
Midland	2.9E-4
Beaver Valley	1.9E-4
Surry (IPE)	7.4E-5
Diablo Canyon	8.5E-5
North Anna	6.8E-5
Seabrook	6.0E-5
Millstone 3	5.5E-5
McGuire	4.0E-5
Surry (NUREG/CR-4550)	3.4E-5
Turkey Point	4.0E-5
Monticello	2.0E-5
Oconee	1.8E-5
Yankee Rowe	1.7E-5
Millstone 1	1.1E-5
James A. Fitzpatrick	1.9E-6
Susquehanna	1.1E-7

TABLE 2-5
COMPARISON OF PRA RESULTS FOR INTERNAL FLOODING

Power Station	Core Damage Frequency (per year)
Surry	5.1E-5
Beaver Valley 2	7.3E-6
Monticello	6.8E-6
Oconee	5.5E-6
North Anna	3.6E-6
Diablo Canyon	3.2E-6
Seabrook	2.8E-6
McGuire	9.0E-7
Millstone 3	8.5E-7
Turkey Point 3 & 4	5.0E-7
Susquehanna	4.0E-9

*Yankee Rowe, James A. Fitzpatrick, Millstone 1 - screened out/no numbers

FIGURE 2-1 OVERVIEW OF LEVEL 1 PRA METHODOLOGY

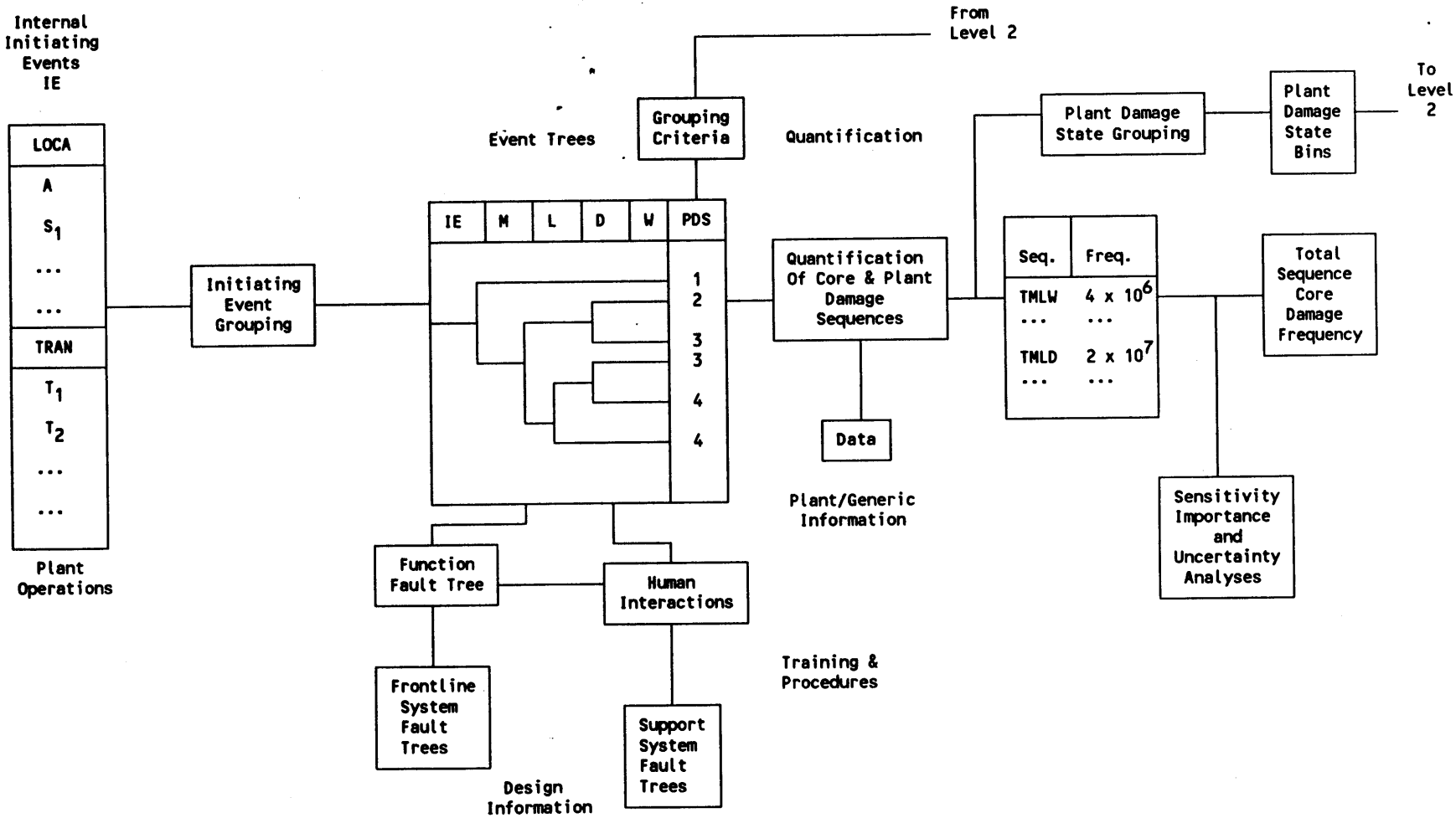
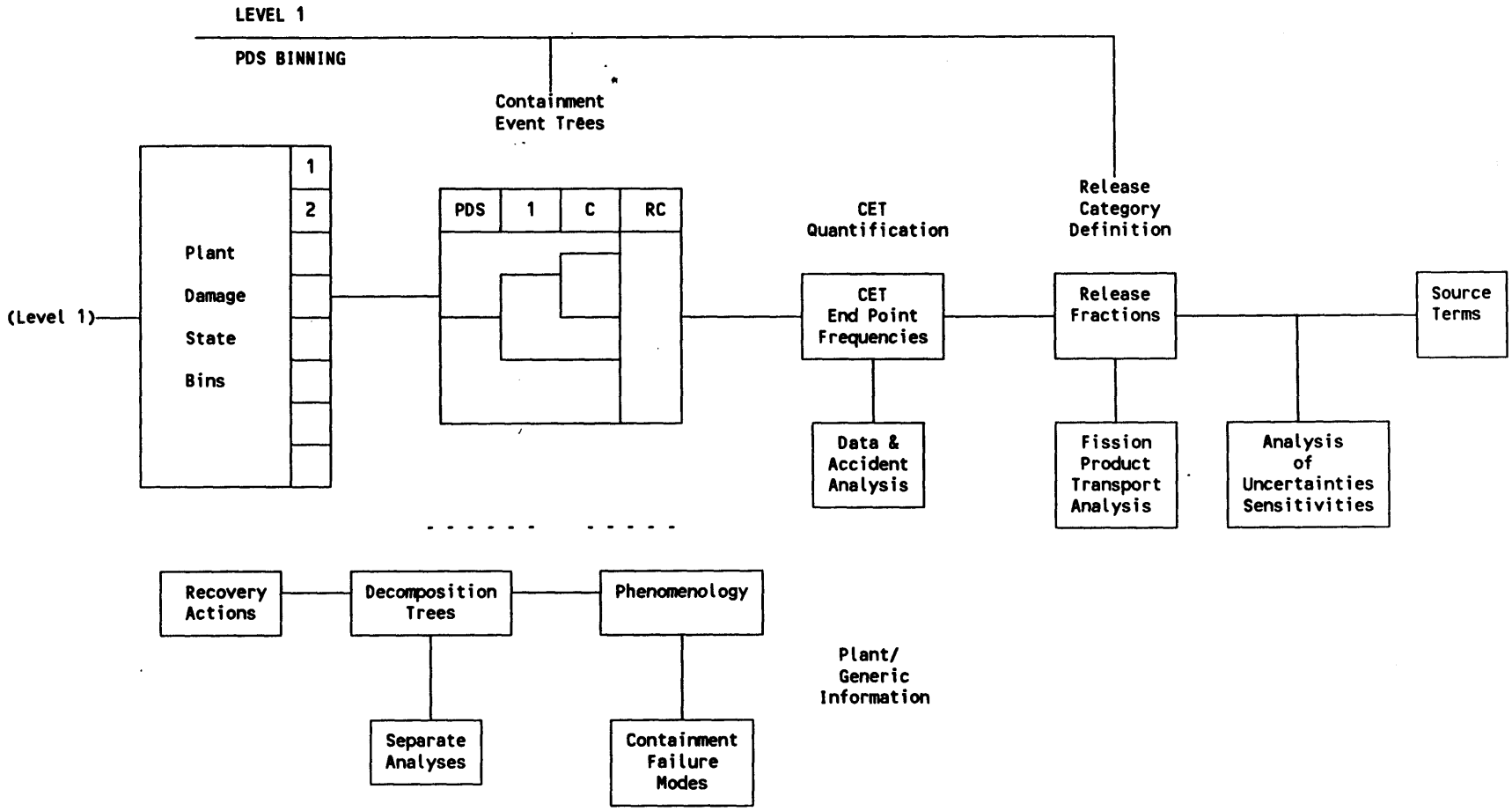
FIGURE 2-2 OVERVIEW OF LEVEL 2 PRA METHODOLOGY