3.0 FRONT-END ANALYSIS

3.1 ACCIDENT DELINEATION

3.1.1 Initiating Events and Success Criteria

The identification of generic and plant specific initiating events and the grouping of these events for the development of event trees is discussed in this section. As the internal flooding events required unique analysis of the plant specific conditions at North Anna the development of initiating events related to the wide range of flood scenarios is described in the internal flooding analysis section (Section 3.3.7). The minimum success criteria, in terms of the front line systems that can be used to prevent core damage, for each group of initiating events are described in this section. The description includes the basis for the criteria [previous studies, Updated Final Safety Analysis Report (Virginia Power, UFSAR, 1992), Westinghouse analyses, or realistic calculations using the MAAP code].

3.1.1.1 Development of Initiating Events

Methodology and Assumptions

An initiating event is either (1) an event resulting in a plant perturbation that potentially challenges the core thermal design criteria, requiring actuation of the reactor protection system and insertion of the control rods or (2) a perturbation resulting directly from insertion of the control rods, as a result of manual or automatic reactor trip. Such events will require the continued operation, or initiation, of the systems necessary to maintain decay heat removal from the fuel, Reactor Coolant System (RCS) and containment and thus prevent core damage. Events occurring at the plant during shutdown operations which might lead to loss of the capability to remove decay heat are specifically excluded from the IPE study. Also excluded at the present time are the majority of events initiated by so called external causes, for example earthquakes and fires. However, loss of offsite power and internal flooding resulting from pipe or tank ruptures within the unit confines are required to be included. The latter will be dealt with as a special case later in the study. Internal Initiating Events, that is, those which occur as the direct result of equipment failure or operator error are covered in this report. Examples are turbine trip, a break in the reactor coolant system leading to loss of coolant, the Loss of Offsite Power, errors during testing or maintenance leading to reactor trip or spurious signals causing a reactor trip.

In view of the recent detailed Individual Plant Examination (IPE) performed on Surry (Virginia Power, 1991), which is a Westinghouse PWR design similar to the North Anna Power Station (NAPS) Units, the approach is the same as used during the Surry IPE process. The Surry model was used as the starting base for North Anna. It was then modified as dictated by the design-specific features of North Anna.

The first stage of the analysis was to start with the list of initiators used in Surry IPE. As will become clear by the discussions that follow, the design differences between Surry and North Anna are such that only initiating events caused by failures of support systems comprise any differences in the overall list of initiating events. The various documents used in the identification of initiating events are listed in Table 3.1.1-1.

The second stage was to review the trips which have occurred at North Anna for the years 1980 - 1990, by examining the monthly operating reports. It should be noted that primarily the information for the years 1986-1990 was used for the quantification of the initiating event frequencies. The starting date for review of 1/1/86 was used since the resulting plant trips will be more representative of the recent operating characteristics of the plants than the earlier years, and five years of operation provide sufficient statistics for frequency quantification of anticipated plant transients. Plant trips that no longer apply because of design changes or equipment upgrades, would be excluded from the data analysis, and the initiating event grouping process. The LER review was extended to 1984-1990 for the T9-related precursors (loss of feeder power to 4160 V Buses 1H and 1J) due to the rarity of these events. One possible exception to this time interval is the consideration of the two incidents of Steam Generator Tube Ruptures (SGTR) that have occurred at North Anna. This issue is addressed in Appendix C.4 as part of the quantification of Initiating Event (IE) frequency.

The third stage consisted of identifying each of the front line and support systems at North Anna and performing a failure mode and effects analysis for those systems which interact with operating and standby systems. This included looking at spatial and environmental interactions. The list of subtle interactions identified by Sandia, and listed in NUREG/CR-4550 (Bertucio, 1990), were included in the development of the final list of initiating events.

On completion of the identification of the initiating events, those which have a similar impact on the availability or response of the systems required to prevent core damage are grouped together. This is done in order to avoid the unnecessary duplication of effort that would result from developing an event tree for each event. A group may consist of a unique event or many events. The final list of initiating event groups is shown in Table 3.1.1-2.

There are two units at North Anna site, and therefore there is the possibility that certain initiating events will affect both units. The two units share some systems and have the possibility to cross connect systems from one unit to another. The potential initiating event interactions are discussed in Section 3.1.1.1.2, which describes the transient initiating events.

The following assumptions were made in establishing the list of initiating events.

1. Failures which would not cause reactor shutdown directly or indirectly were eliminated.

2. Initiators which could possibly lead to shutdown through Technical Specification violations were not evaluated, unless they involved the unavailability of major plant system or system trains after plant trip.

3. Normal shutdown for refueling or administrative reasons were not included.

4. Those events which have occurred at power levels greater than 30% will be included when evaluating the frequency of each group. This power level criterion was chosen based on License Amendments 119 and 103 issued by the NRC on July 18, 1989, wherein the set point for direct reactor trip from a turbine trip was increased from 10 to 30 percent.

5. Events below 30% power will only be included if they represent conditions of special significance or relevance to power operation, judged on a case by case basis.

6. Events which involve over or under filling of the steam generators during start-up or manual shutdown are also excluded. Events during start-up are associated with low decay heat levels, and those during shutdown can be considered as manual shutdown events, which have been shown in past PRAs to be negligible contributors to core melt frequency. (In general the power level at which these events occur is less than 30%.)

3.1.1.1.1 Loss of Coolant Accidents

A loss of coolant accident (LOCA) is any random breach of the reactor coolant boundary that causes loss of inventory from the system at a rate higher than the capacity of the normal charging system and leads to conditions for the generation of a reactor trip and a safety injection (SI) signal on low pressurizer pressure. For North Anna this is a break size of approximately 3/8", and therefore this represents the lower bound for LOCAs. Any leakage lower than this is treated as a transient as the charging system

requirements and timing of events are such that no safety injection signal is generated, but manual shutdown would be required.

There are two basic categories of LOCAs: those which occur inside containment thus leading to the retention of all the water injected from the refueling water storage tank (RWST) within containment; and those which occur outside containment, in which case the water injected from the RWST is eventually transferred outside containment and is not available for recirculation. In order to evaluate the consequences of the various LOCAs it was necessary to consider the systems that can be used to provide cooling following a leak of a given size and a given location.

In addition to the direct occurrence of a leak as an initiating event, there is also the potential for consequential loss of coolant accidents. Potential sources inside containment for these events, which are the consequences of another initiator or system failure, are a failed open power operated relief valve following its opening during a transient, or failure of the reactor coolant pump seal following loss of seal injection and seal cooling (Westinghouse, 1988c). Similarly, there is the potential for a consequential LOCA by-passing containment following a steam generator tube rupture, if the affected steam generator is not isolated, and a steam generator atmospheric dump valve or safety valve lifts and fails to reseat. This will lead to a direct path from the RCS to outside containment. Each class of LOCA is described below and summarized in Table 3.1.1-3.

LOCA Inside Containment

Following a LOCA inside containment the water that is lost from the RCS through the break will eventually gather in the containment sump and is thus available for recirculation through the reactor vessel. Retention of the water within the containment also prevents the direct release of radioactivity to the environment. As discussed earlier, North Anna plant-specific analysis shows that a breach in the primary system boundary equivalent to a pipe size of 3/8 inch inside diameter (ID) is large enough to eventually lead to the generation of a safety injection (SI) signal. This ID can thus represent the lower bound on the break size for a small LOCA.

All components that could cause a LOCA inside containment were identified to determine the range of possible break sizes. The list of identified components and the ranges of the equivalent break sizes are given in Table 3.1.1-4.

Ultimately LOCAs are grouped into three break size categories. These category definitions are based on the requirements of mitigative functions (success criteria described in Section 3.1.1.2), and are consistent with other PWR PRAs. The lower bound for a small LOCA has been reduced to 3/8" compared with the value

of 1/2" given in NUREG/CR-4550, in order to meet the criterion for the definition of LOCA discussed above.

The very small LOCA category used in NUREG/CR-4550 was not included as a separate initiator group, but was modeled as a subgroup of S2. An event tree heading was added to distinguish between the very small and the small LOCAs. Although any break less than 3/8" is a transient, per the definition of this PRA, the very small break has substantial timing differences which affect the accident sequence delineation. The entire spectrum of LOCAs is covered by three groups: Large (A), Medium (S1) and Small (S2).

LOCA Outside Containment

There are two classes of potential LOCA outside containment. The first, the so called interfacing system LOCA, is the result of the failure of the closed valves representing the interface between high and low pressure systems. The second is the result of failure of high pressure piping outside containment.

These LOCAs are characterized by direct release of radioactivity to the environment, and the inability to recirculate the water that is lost out of the break. They are traditionally referred to as the "V sequence." Sources for the V sequence are the three listed in Table 3.1.1-3.

In North Anna, the RHR system is contained entirely inside the containment. Thus the only systems directly connected to the RCS that can cause a LOCA outside containment are the SI System (includes charging for high pressure injection - see Appendix A) and the Chemical and Volume Control System (CVCS). An additional potential source recently identified by Westinghouse is the failure of the RCP thermal barrier heat exchanger which would result in the release of coolant into the lower pressure Component Cooling Water (CC) system. Because of the small size of the equivalent break area, consistent with the criterion of excluding lines below 2 inch diameter adopted in NUREG/CR-5102 (Bozoks, 1989), this category is not examined further.

The specific containment penetrations for those systems that interface with the RCS are discussed below. They identify the candidates for the analysis and frequency quantification of the interfacing systems LOCA.

The study of the V-sequence for North Anna was performed by analyzing the piping and instrumentation diagrams (P&ID) for systems that interface with the Reactor Coolant System (RCS) with lines that eventually lead to the outside of containment. The following criteria were used to eliminate some lines from further consideration:

- Small Lines: Lines with diameters less than 2" were not considered (as was done in NUREG/CR-5102, 1989). In general, these breaks do not directly impact the safety systems and the resulting leakage is small.

- Breaks Inside the Containment: Breaks inside containment at interfaces between high- and low-pressure systems are included in the frequencies of LOCAs inside containment.

All lines penetrating containment and not eliminated using the above criteria were analyzed further. The following is a summary of information gathered after reviewing the P&IDs. Piping classifications are shown in Table 3.1.1-5 (Virginia Power, UFSAR, 1992). Each identified line is listed with its associated RCS loop number (if any).

High Head Safety Injection Pumps Discharge to RCS Cold Legs

Loop 1

The cold leg injection piping is class 1502 all the way to the charging pump discharge and penetrates the containment at penetrations #22 and #7. There are two check valves, 1-SI-83 and 1-SI-190, inside the missile barrier. Additionally, a check valve exists after each of the penetrations (1-SI-185 for #22 and 1-SI-79 for #7). There is also a normally closed motor operated valve, 1-SI-MOV-1836 for #22 and 1-SI-MOV-1867C for #7, in each line outside containment.

Loop 2

Similar to Loop 1, but with check valves 1-SI-86 and 1-SI-192 inside the missile barrier.

Loop 3

Similar to Loop 1 and 2 but with check valves 1-SI-89 and 1-SI-194 inside the missile barrier.

Low Head Safety Injection Pumps Discharge to RCS Cold Legs

Loop 1

The low head safety injection pipe to cold leg is class 1502 all the way to the containment penetration, #62. This line joins the line from high head safety injection before 1-SI-83 check valve which is inside the missile barrier. In addition, there is a check valve, 1-SI-195, just outside the missile barrier. There are no other valves between the penetration and the check valve, 1-SI-195. The are two normally open motor operated valves, 1-SI-MOV-1890C and D, which are in parallel, outside the penetration. The piping class changes from 1502 to 153A at these valves.

Loop 2
Similar to Loop 1 but with different valve mark numbers.

Loop 3
Similar to Loops 1 and 2 but with different valve mark numbers.

High Head Safety Injection to Hot Legs

Loop 1
The hot leg piping is made up of class 1502 all the way to the high head pump discharge. There is a check valve, 1-SI-99, inside the missile barrier and an additional check valve on each of the two lines upstream of the header, 1-SI-201 for penetration #114, and 1-SI-90 for #113, just inside containment. Two normally closed motor operated valves, one for each line, are on the outside of containment (1-SI-MOV-1869A for #114 and 1-SI-MOV-1869B for #113). The main lines which feed the high head pumps suction are class 1502.

Loop 2
Similar to Loop 1 but with 1-SI-95 check valve inside the missile barrier.

Loop 3
Similar to Loop 1 but with 1-SI-103 check valve inside the missile barrier.

Low Head Safety Injection to Hot Legs

Loop 1
The low head safety injection line joins the safety injection line from high head just before the check valve 1-SI-99. Again the line is entirely class 1502 inside the containment. This line starts from the LHSI pumps and penetrates the containment through two penetration points (#60 and #61). The lines then joins with the high head injection line through a check valve (1-SI-209) inside the missile barrier. There is a motor operated valve and check valve on each of the lines (1-SI-MOV-1890A and 1-SI-207 for #60 and 1-SI-MOV-1890B and 1-SI-206 for #61). The piping class changes to 153A at these two normally closed motor operated valves.

Loop 2
Similar to Loop 1.

Loop 3
Similar to Loop 1.

Normal Charging Line

Loop 2

The charging line has two check valves (1-CH-325 and 1-CH-496) inside the missile barrier. There is also a HCV-1310 upstream of the missile barrier which fails open. Inside the containment at penetration #15 is check valve 1-CH-322. There is also an 1-CH-MOV-1289A just outside the containment. Since the charging line is fed by the charging pumps, the piping class remains 1502 back to a header which is fed by each pump.

Loop Fill

Loop 1

The loop fill line is made up of class 1502 all the way to the charging pumps. There is a check valve, 1-CH-330, inside the containment but outside the missile barrier. The pipe size is 2" inside the containment and it changes to 4" upstream of valve 1-CH-FCV-1160. In addition to 1-CH-FCV-1160 there is HCV-1566A inside the containment both of which fail closed on loss of Instrument Air and electrical power.

Loop 2

Similar to Loop 1 but with different valve mark numbers.

Loop 3

Similar to Loop 1 and Loop 2 but with different valve mark numbers.

Auxiliary Spray Line

The auxiliary spray line connects to the charging line outside of the missile barrier. The piping class is 1502 all the way to the charging pumps. There is a check valve (1-CH-328) and 1-CH-HCV-1311 which fails closed, upstream of the check valve but before penetration 15. Isolation outside of containment is provided by 1-CH-MOV-1289A.

Letdown Line

Loop 1

The letdown line is class 1502 pipe until after the 1-CH-HCV-1200A/B/C which are after the regenerative heat exchanger. The reducing orifice acts to limit the flow through the line. The 1-CH-HCV-1200A/B/C valves fail closed. A section of the low pressure pipe (class 602) is shown to be inside containment. The letdown line penetrates containment at penetration number 28. Right before and after the

penetration there are trip valves (1-CH-TV-1204A/B) which fail closed.

Seal Water Injection Line

The seal water injection to RCPs, 1-RC-P-1A/1B/1C, takes place through three different penetrations (#36, #37, and #35) respectively, and discharge through penetration #19. Since the equivalent flow into these lines in case of seal failure is below that of a 2 inch equivalent break (Bertucio, NUREG/CR-4550, 1990), this line is not analyzed any further.

RCP Thermal Barriers

The lines in contact with the thermal barrier are 1-1/2 inch pipes, which are below the minimum pipe size for inclusion into the analysis.

Consequential Loss of Coolant Accidents

In addition to the occurrence of breaches in the RCS boundary during power operation, it is also necessary to consider the potential for RCS boundary failure following other initiating events, combinations of initiating events and system failures, or combinations of system failures. It is customarily assumed that with the exception of gross overpressure of the RCS pipe work, coincident failures will be unlikely during an accident outside the steam generators. Consequential LOCAs are therefore likely to arise from failure of pressurizer relief valves, pressurizer safety valves, RCS pump seals, and steam generator tube ruptures. Pipe work failures may occur following gross overpressure beyond the capacity of the safety and relief valves during the course of an ATWS event, and vessel failure has to be considered for pressurized thermal shock events. The possible contributions to a consequential LOCA are summarized in Table 3.1.1-3.

3.1.1.1.2 Transients

Transient initiating events are more complex than LOCAs, as in many cases there is significant interaction between the initiator and the systems which will be used to prevent core damage. The initiating events which have occurred at other PWRs and those which have occurred at North Anna are discussed in this section in relation to the plant design features. The grouping criteria used to aggregate the various transients are also discussed.

As in the case of the LOCAs, there is the potential for consequential failures. One important consequential failure is the

Anticipated Transient Without Scram (ATWS), which is the failure to scram following the initiating transient event. As the system requirements are very different following an ATWS, it is treated as a special initiating event, and separate event trees are developed for the various ATWS conditions. Another consequential event is pressurized thermal shock (PTS) which can occur if there is rapid cooldown of the RCS at the same time that pressure is maintained at or near normal operating conditions. This is discussed in Section 3.1.3.9.

A consequential event specific for North Anna is the failure of the Emergency Switchgear Room cooling following an initiating transient. This ESGR cooling consequential failure can be affected by a support system failure causing the transient (e.g., Loss of Offsite Power, T1, or Loss of Power from 4160 V Bus 1H, T9A). Separate event trees are developed to evaluate these consequential failures, identified by the 'Tr' code added to the original initiating event (e.g., the event tree T1Tr evaluates the Loss of Offsite Power initiator with the consequential loss of ESGR cooling).

Other consequential events of general interest in PRA's are the sticking open of a steam generator atmospheric dump valve or relief valve following an event such as an MSIV closure transient. Each of these is evaluated appropriately when performing the event tree analysis.

Grouping of Transient Initiators

As many of the initiating events require the same systems to operate under the same conditions in order to achieve the long term stable conditions defined in the event trees, it was possible to group them and so reduce the number of event trees which had to be developed. This grouping process is performed iteratively with the development of success criteria discussed in the Section 3.1.1.2. The process for grouping the transient initiators was simplified significantly because of the high degree of similarity between the NAPS and Surry designs.

Surry IPE (Virginia Power, 1991b) provided the starting point for the definition of transient initiating event groups. Departures from this list were implemented based upon plant specific analysis. The initiating event classifications in NUREG/CR-3862 and EPRI NP-2230 were also used for two purposes. First they helped to provide a more detailed categorization of initiating events than those provided in the Surry IPE and secondly this classification, which will be referred to as EPRI categories, is used to compare the plant specific initiating event frequencies with the generic data when evaluating the frequency for each initiator group in Table 3.1.1-2.

The NUREG/CR-4550 initiating event groups are shown in Table 3.1.1-6. Because of the importance of the availability of the Main Feedwater (MFW) system after reactor trip, it was decided to further subdivide the T2 group into two sub-groups as shown in Table 3.1.1-7. Table 3.1.1-8 identifies the relationship between grouping of events for the North Anna IPE and EPRI categories, based on the MFW availability criteria defined in Table 3.1.1-7 and plant operating experience.

In the course of performing the grouping analysis a number of initiating events were identified as having a direct impact on both units. Careful note has to be made of this to ensure that the use of systems on one unit to support the other unit is correctly handled when both units have tripped as the result of a common initiating event. This is discussed in detail in quantification of the core damage frequency section of this report.

The following events result in a two unit trip.

Loss of Offsite Power (T1)
Loss of Instrument Air (contributor to T2 and T4)
Loss of Service Water System (T6)

Analysis of Plant Operating Experience

The sources consulted for the compilation of North Anna Units 1 and 2 plant-specific operating experience are listed in Table 3.1.1-9. Data on all reactor trips was gathered to qualitatively evaluate all possible causes of trips that have occurred at North Anna, in the time period under review. Each initiating event was reviewed and characterized by listing important plant information such as date, power level, generator breaker position, generation of a SI signal, etc. In addition, for each recorded event, an event description and cause was added and the event was assigned to an EPRI category and a North Anna initiating event group. The data was also reviewed in the light of the current plant status, in order to exclude any potential initiating events that would no longer apply because of plant design or procedural changes. The other criteria for inclusion are as follows:

- Manual shutdowns (as opposed to manual scrams) are excluded.
- Scrams below 30% power are excluded, with the exceptions discussed before.
- Applicable data review period is 1980 - 1990 from which the data for the period 1986-1990 is included in Table 3.1.1-10. This is because these years provide a long enough time interval for quantification of the frequencies of most initiating event groups, and represent the most recent operating experience. The review was extended to 1984-1990 for the T9-related

precursors only; this longer interval was required by these less frequent events.

The analysis of the plant-specific trip data did not identify any initiators, which did not fall into the categories listed in Table 3.1.1-2.

3.1.1.1.3 Support System Failures

A list of systems at North Anna is shown in Table 3.1.1-11 and the dependency matrices for front line and support systems in Tables 3.1.1-13 and 14. The majority of these systems support power operation and perform non-safety related functions at the plant. However, a number of these systems provide the necessary heat removal functions following reactor trip, either directly or as a supporting system. Those supplying the direct function of reactivity control, core heat removal or containment heat removal are designated as front line systems, and those supporting the front line systems are referred to as support systems. Since a support system may serve more than one front line system; for example, the emergency 4160 V bus provides power to a number of pumps, there is concern that the failure of some of these systems may be significant contributors to core damage. Each of the support systems in the table was evaluated using the following criteria:

1. Does its failure have the potential to cause a reactor trip,

   and

2. Does it also result in failure of a train of one or more of the front line systems.

For each of the systems which satisfy the above criteria a Failure Modes and Effects Analysis (FMEA) was performed and is discussed in the following paragraphs.

The systems at North Anna for which an FMEA was performed are listed below and the results are summarized in Table 3.1.1-12.

1. Loss of 4160 V
2. Loss of 480 V
3. Loss of 120 VAC
4. Loss of 125 VDC
5. Loss of Service Water (SW)
6. Loss of Component Cooling (CC)
7. Loss of Emergency Switchgear Room cooling
8. Loss of Compressed Air (Instrument Air)
9. Loss of Bearing Cooling (BC)

The reasons for inclusion or exclusion of each system as an initiating event are also included in Table 3.1.1-12.

Loss of 4160 V

Loss of an individual non-emergency 4160 V bus will result in a reactor trip as the result of failures of Reactor Coolant Pump (RCP) and/or other balance-of-plant equipment, but will not impact any of the identified front line or support systems supplied from the diesel supported (emergency) 4160 V buses. Failures of non-emergency 4160 V buses are already counted in the transient categories described previously, and are not treated as unique initiators.

The loss of a single emergency 4160 V bus was considered to determine its potential as an initiating event. For front line and support systems directly using emergency 4160 V, each system is configured with two or more trains, with at least one of these trains assigned to the 1H 4160 V emergency bus and another train assigned to the 1J 4160 V bus. Loss of the standby ECCS or containment cooling 4160 V loads does not require a reactor trip and loss of an operating component emergency 4160 V load will not cause a reactor trip since the redundant train is still available. Loss of power to emergency 4160 V buses will also result in loss of power to the associated emergency 480 V buses, which is discussed in the support system failure section on loss of 480 V.

Loss of the 1H or 1J emergency 4160 V buses can cause a reactor trip due to loss of non-vital 480 V loads. Loss of the 1H 4160 V bus will result in loss of or power to the Unit 1 IRPIs, indicating that the shutdown and control banks have dropped into the core. Two North Anna Unit 2 LERs, 339-85005 and 339-90002 were evaluated which involved a momentary loss of the 2H 4160 V emergency bus due to switchyard or RSST (Reserve Station Service Transformer) faults. The earlier LER event resulted in the Control Room Operator initiating a manual reactor trip because the operator believed a reactor trip was occurring. The later LER event did not result in a reactor trip as the 2H 4160 V bus was powered by the 2H diesel. Loss of IRPIs does not require reactor trip if alternate rod indication (i.e., rod bottom lights, no change in NI power level, etc.) provides adequate assurance for continued operation. However, if either event had involved a total loss of the 4160 V 2H bus, and not just a momentary loss of the preferred switchyard power source with diesel startup and loading, then a reactor trip can be expected. Loss of the 1J 4160 V bus will de-energize the 1J1-1 480 V bus, which will isolate Component Cooling (CC) water to the Reactor Coolant Pumps (RCP). Loss of the 1J 4160 V bus has resulted in a manual reactor trip from high power, as reported in LER 338-85019.

The loss of 1H or 1J emergency 4160 V bus events are identified as support system special initiators T9A and T9B because of their impact upon a large number of systems. For T9A, the 1H bus loss affects the A and C ESGR chiller trains as well as the ESGR AHU 6. This leaves 1 chiller and 1 AHU for ESGR cooling. For T9B, the 1J bus loss affects the B ESGR chiller train and the ESGR AHU 7, and isolates CC cooling to the RCPs with the B Charging Pump unavailable for RCP seal injection. Because of the wide-scale impact of the loss of a 4160 V emergency bus, and the high probability of reactor trip on the total loss of a 4160 V emergency bus, these events are included as the support system initiators T9A for the 1H 4160 V bus and T9B for the 1J 4160 V bus.

Loss of 480 V

Loss of an individual non-emergency 480 V bus can lead to a reactor trip as the result of failures of RCP and/or other balance-of-plant equipment, but will not impact any of the identified front line or support systems supplied from the diesel supported (emergency) 4160/480 V buses. Failures of non-emergency 480 V buses are already accounted for in the transient categories described previously, and are not treated as unique initiators.

The loss of a single emergency 480 V bus was considered to determine its potential as an initiating event. For front line and support systems directly using emergency 480 V, each system is configured with two or more trains, with at least one of these trains assigned to the 1H or 1H1 480 V emergency buses and another train assigned to the 1J or 1J1 480 V buses. Loss of power to the emergency 480 V bus loads in the front line systems will not result in an automatic reactor trip. Failure of the emergency 480 V buses 1H1-4 and 1J1-1 will result in failure of 480 V power to the 120 VAC vital buses, and the 125 VDC chargers. The 120 VAC vital buses can also receive power from inverters supplied from 125 VDC buses, ultimately supplied from the station emergency batteries. Thus, failure of the 480 V emergency buses does not lead to failure of either the 120 VAC vital buses or the 125 VDC buses, and will not cause a reactor trip from these sources.

Based upon the above review, loss of the 480 V buses was not included as special initiators. However, loss of a 4160 V bus will result in loss of power to all lower level 480 V buses in that train. Hence, the T9A and T9B special initiators will include any 480 V bus dependencies modeled in this PRA.

Loss of 120 VAC Vital Bus

North Anna has five vital instrumentation buses powered from four invertor/transformer power supplies (the fifth bus is for North Anna Unit 2 Appendix R Distribution). These vital buses will

continue to receive power from the batteries through the inverters should power be lost to their parent 480 V or 4160 V buses. Reactor trips can be or have been caused by a loss of 120 VAC vital bus through three mechanisms:

1. reactor trip on loss of RC flow due to loss of power to a RCP sensor (which is now no longer susceptible to this fault by re-design),

2. reactor trip on loss of Circulating Water pumps (1-CW-P-1A/B/C/D) as a result of a loss of the condenser due to condenser vacuum breakers opening (at least 1 of 2 vacuum breakers on 2 or more water boxes) on loss of vital bus 1-I or 1-III power (which is now no longer susceptible to this fault by re-design), and

3. manual reactor trip on loss of RCP cooling (stator and bearing lube oil) necessitated by isolation of CC cooling water upon a loss of vital bus 1-I power or vital bus 1-III power.

None of these mechanisms is considered an initiator for the following reasons:

1. the first and second mechanisms have been corrected by adjustments to power sources,

2. loss of condenser is considered within the loss of feedwater initiator

3. the condenser dump valves are not required because steam generator safety and atmospheric dump valves provide adequate heat removal with several backups (only one of five safety valves is required for this initiator), and

4. the third mechanism, loss of RCP cooling, is similar to a loss of all RC forced flow event, as the loss of vital bus 1-I power does not affect any front line systems except the "A" steam generator atmospheric dump valve, 1-MS-PCV-101A.

In the same manner, the loss of vital bus 1-III power does not affect any front line systems except the "C" steam generator atmospheric dump valve, 1-MS-PCV-101C and the condenser dump valves. Hence, the loss of a 120 VAC Vital Bus is not considered an individual initiator because loss of bus 1-II and 1-IV do not result in a reactor trip, and loss of 1-I and 1-III are subsumed within the T3 initiator.

Loss of 125 VDC

Loss of 125 VDC bus 1-I or 1-III will lead to closure of all main feedwater control valves and bypass valves resulting in an immediate automatic reactor trip. These DC buses also provide control power to the 4160 and 480 V switchgear. Upon loss of DC bus 1-I the A train (H buses) switchgear will fail as is, so that the standby A train pumps will be inoperable. Similarly, loss of DC bus 1-III affects the B train (J buses) switchgear, so that the standby B train pumps will be inoperable. Although the loss of these DC buses has the appearance of the T2 loss of FW initiator (due to closure of the MFW control valves), these events are included as the separate support system initiators T5A and T5B because of their notable effect upon front line and support systems.

Loss of Service Water

The Service Water (SW) system at North Anna is common to both reactor units and is designed for the simultaneous operation of various subsystems and components of both units. The purpose of the SW system is to provide long term cooling for Design Basis Accidents (i.e., the ultimate heat sink) and to supply cooling water to safety-related components during normal plant operations. The sources of cooling water for the SW system are the Service Water Reservoir and the man-made Lake Anna.

The SW system provides cooling to the following front line and support system components:

1. Charging pump (1-CH-P-1A/B/C) seal and lube oil coolers,
2. Component Cooling system heat exchangers, 1-CC-E-1A/B and 2-CC-E-1A/B,
3. Emergency Switchgear Room chiller condensers, 1-HV-E-4A/B/C,
4. Instrument Air compressors, and
5. Recirculation Spray system heat exchangers, 1-RS-E-1A/B/C/D.

Note that the first four heat loads are in use during normal operation, as well as during accidents. CC, ESGR cooling for the emergency power switchgear and instrument air are support systems for front line systems.

Loss of SW will lead to a reactor trip through similar mechanisms to the loss of CC event, the loss of instrument air event, or the loss of ESGR cooling (T8). Because loss of SW affects several

front line and support systems, it meets the criteria for special initiators and will be included as T6.

Loss of Component Cooling Water (CC)

Loss of the CC system at North Anna Units 1 and 2 will not cause an automatic reactor trip, but will result in a manual reactor trip or shutdown if CC is not restored for RCP cooling (see LER 338-85019, where a manual reactor trip was initiated upon a loss of CC at a higher power level because manual shutdown could not be completed before RCP bearing temperatures exceeded procedural limits). Loss of CC will affect the following components:

1. RCP motor cooling (stator and oil systems),

2. RCP thermal barrier,

3. RHR heat exchangers.

The loss of CC does not fail RCP seal injection flow, main or auxiliary feedwater, or any other front line or support system required to maintain the plant in hot shutdown. Note that the RHR heat exchangers will not be available for plant cooldown on a loss of CC event, but RHR plant cooldown is not essential for any plant transients except the Steam Generator Tube Rupture (SGTR), which is itself a special initiator. Thus, this event is not considered by itself as an initiator, but is considered in conjunction with RCP seal injection flow failure in the T4 initiator.

Loss of Emergency Switchgear Room Cooling

The North Anna Emergency Switchgear Room contains most of the H and J bus 4160 V and 480 V switchgear, as well as the transformers, inverters and battery chargers. The ESGR is part of the Main Control Room envelope, and during accidents with radiological releases, the ESGR is isolated similar to the MCR to provide protection for the Control Room Operators. This isolation prohibits the use of significant outside ventilation for room cooling during accidents, and is the prime reason for modeling ESGR cooling loss during accidents.

During normal plant operation, the ESGR cooling requirements are more lenient, but a prolonged loss of the normal ESGR cooling system will eventually result in a manual reactor shutdown or an automatic reactor trip if certain vital equipment overheats (i.e., Solid State Protection system cabinets). ESGR cooling is dependent upon two component groups, two air handling units (AHU), 1-HV-AC-6/7, and three chillers, 1-HV-E-4A/B/C. Note that the chillers are cooled by SW and that both the AHUs and chillers are powered by emergency power sources. A reactor trip from loss of

ESGR cooling would involve the disabling of most front line and support system active components if ESGR cooling is not recovered. Because of the plant-wide impact of the loss of ESGR cooling initiator, this event is included as a support system initiator, T8.

Loss of Compressed Air (Instrument Air)

North Anna has several compressed air subsystems. For the purposes of the IPE, the outside containment subsystem (referred to as just "Instrument Air") and inside containment subsystem (referred to as "containment Instrument Air") are considered separate systems.

Loss of containment Instrument Air will not result in an automatic reactor trip, but a manual reactor trip may be required because RCP cooling water will be lost when the CC RCP supply and return trip valves close on loss of air. Affected equipment inside the containment include:

1. Pressurizer PORVs (nitrogen backup supply provided)

2. RCP stator and oil subsystems cooling

3. RCP thermal barrier cooling

4. RCS normal and excess letdown

Loss of containment Instrument Air is similar to loss of CC in that the event is not considered by itself as an initiator, but is considered in conjunction with RCP seal injection flow failure in the T4 special initiator.

Loss of outside Instrument Air can result in an immediate automatic reactor trip due to FW control valve closure or MS trip valve closure. Affected equipment inside and outside the containment include:

1. RCP stator and oil subsystems cooling (closure of outside isolation valves)

2. RCP thermal barrier cooling

3. RHR heat exchanger cooling

4. RHR pump seal cooling

5. Other CC loads within containment are isolated

6. FW control valves fail closed

7. MS condenser dump valves fail closed (MSTV may also close)

8. MS atmospheric dump valves (limited backup air supply provided)

9. MS trip valves fail closed

Loss of Instrument Air outside containment is functionally equivalent to a loss of feedwater transient, initiator T2, with the additional loss of condenser dump valves (which are not taken credit for in the T2 event tree) and RCP thermal barrier cooling. The loss of instrument air does not fail RCP seal injection flow, so for general transients, the loss of Instrument Air is subsumed within the T2 initiator. Similar to the loss of CC system, the loss of Instrument Air is considered in conjunction with RCP seal injection flow failure in the T4 initiator.

Loss of Bearing Cooling Water

The Bearing Cooling (BC) water system removes heat from several secondary plant components and transfers heat to the bearing cooling water towers. The only front line or support system components that are cooled by BC water are the main feedwater pumps and the condensate pumps. Loss of bearing cooling water is functionally equivalent to a loss of feedwater transient, initiator T2, so loss of this system is not considered an individual initiator.

3.1.1.1.4 Special Initiators

In addition to the events discussed in the previous sections, it was necessary to consider events identified in other PRA's and events identified as safety issues by the NRC. In addition, the IPE team performed a plant walkdown to identify special areas where specific events could have a major impact on the operation of front line systems. At this stage external events such as fires and flooding are excluded, since they will be addressed in future studies.

In the evaluation of LOCA initiators, one potential event was Reactor Vessel rupture. This can occur randomly, or it could occur as the result of the condition known as pressurized thermal shock (PTS). Reactor Vessel rupture as a random event is defined as a unique initiator, Rx. Rx is different from the A-LOCA in that the break location is defined to be below core height, thus precluding successful core cooling. All Rx sequences therefore result in core damage.

Analysis of main steam line break (MSLB) transients was screened out of NUREG/CR-4550 on the basis of low initiating event frequency. However, these transients were evaluated as part of the Surry IPE. At Surry the main steam lines pass through a building containing the AFW pumps so the potential for consequential failures exists. The results of the Surry evaluation showed that the MSLB was not significant even under these conditions. Accordingly, it has been omitted from the North Anna IPE.

The North Anna event trees include special trees that model transfers from consequential events. These are limited to loss of Emergency Switchgear Room cooling events that are partially caused by the normal initiating event (consequential events) or are loss of ESGR cooling events that occur during the 24 hour period following the initiator (coincidental events). An example is the T1Tr event tree, modeling the Loss of Offsite Power initiating event with the consequential loss of ESGR cooling. Implementation of these transfer event trees is discussed further in Section 3.3.6.1. These additional event trees are essentially variants of the T8 initiating event, with the precursor being a normal North Anna initiating event. Note that the transfers from coincidental events appear to be a subset of T8; however, human recovery during the T8 precursors does not have the burden of an ongoing accident, so that minor recoveries are reasonably justified. On the other hand, the coincidental events have an ongoing accident that can delay or inhibit recovery actions, which is reflected in the frequencies for T2Hv, T2AHv and T3Hv.

3.1.1.1.5 Summary of Initiating Events

The final groups of initiating events or single initiating events for which event trees have been developed are listed in Table 3.1.1-2. The frequency of each initiating events and the models used to derive these frequencies are described in Section 3.3.1 and 3.3.2.2.

3.1.1.2 Success Criteria

The success criteria for any initiating event are the minimal number of systems that are required to function to maintain adequate heat removal from the core and containment, ultimately establishing long term stable conditions and preventing core damage or containment failure. In accordance with industry practice and as assumed in NUREG/CR-4550, in general, sequences are terminated at 24 hours. This amount of time is considered appropriate, because of the low decay heat levels that allow extended recovery times for failed equipment or other corrective actions. Analysis of accidents leading to core melt after extended or stable reactor shutdown is beyond the scope of the IPE requirements. Stable conditions include hot shutdown or any other condition where heat

removal from the core and from the containment could continue for an extended period of time, with no requirement for additional systems to operate. If successful core heat removal is established but the decay heat is being transferred to the containment without containment heat removal, containment overpressurization will ultimately lead to its failure. This in turn may lead to failure of the operating injection system and loss of core heat removal. This is taken into account when developing the system success criteria.

In defining the sequence of events which will lead to core damage and the release of fission products, it is necessary to include operator interactions with the required systems based on the operating procedures. It is therefore necessary to know the time available to the operator to perform the necessary actions, to starting or recovering various systems. It is thus necessary to define acceptance criteria for fuel conditions, the containment failure pressure, and reactor coolant system pressure boundary. The following acceptance criteria based on information from the referenced sources were used in establishing the system success criteria and defining the accident sequences.

3.1.1.2.1 Acceptance Criteria

Fuel Boundary

In the IPE we are concerned with two levels of modeling, the identification of "core damage," and the risk posed by the release of fission products following core degradation and melting. For the core damage analysis, the conditions defined in 10 CFR 50.46, North Anna's UFSAR (Virginia Power, 1992) Section 15.3, have been used to define the onset of core damage.

1. The calculated peak fuel rod clad temperature is below the requirement of 2200°F.

2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of zircaloy in the reactor.

3. The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling, and the localized cladding oxidation limit of 17% is not exceeded during or after quenching.

4. The core remains amenable to cooling during and after a break in the RCS.

5. The core temperature is reduced and long term decay heat removal is established for an extended period of time.

In performing the core degradation analysis for the containment evaluation, some sequences will be identified in which core damage occurs but total core melt is arrested prior to vessel melt through, thus limiting the release of fission products. In these cases, the frequencies of fission product release of various magnitudes will be realistically assessed, and not based simply on the frequency of the onset of core damage.

Reactor Coolant Boundary

The acceptance criteria for maintaining integrity of the reactor coolant system boundary, as stated in NUREG/CR-4550, is only of real concern during an anticipated transient with failure to scram. In WCAP-11993 (Westinghouse, 1988c) 3200 psig is stated to be a conservative value applicable to all Westinghouse plants, corresponding to the ASME Boiler and Pressure Vessel Code Level C service limit stress criterion. In other transients there is the possibility of the pressurizer power operated relief valve opening resulting in the potential for sticking open and leading to a breach of the RCS (Westinghouse, 1981).

Containment Integrity

Containment evaluation for Surry performed in accordance with criteria and guidelines set forth in NUREG/CR-4551 showed the minimum theoretical yield capacity of the Surry containment is 128 psig. North Anna containment design was compared to that of Surry and was determined to be similar enough that its strength is assumed to be the same as that of Surry.

System Acceptance Criteria

Besides the above overall acceptance criteria, there are also specific acceptance criteria for some components and systems. Two of these which have an influence on the construction of the event and fault trees, and to a limited extent on the success criteria are:

1. NPSH for pumps connected to the containment sump,

2. maximum ambient temperature for all safeguard equipment.

The first of these is addressed in the derivation of the loss of coolant accident success criteria, and the second is addressed in the system modeling analysis described in Appendix B.

3.1.1.2.2 Front Line and Support System Safety Functions

The basic functions required for prevention of core damage and mitigation of the release of fission products can be divided into the six safety functions below:

1. Reactivity Control
2. Core Heat Removal
3. Secondary Heat Removal
4. RCS Integrity
5. Radioactivity Control
6. Containment Condition

For a given accident initiating event the systems that directly perform one or more of these safety functions are defined as front line systems. Support systems are those systems that affect the course of the accident sequence by supplying motive power, control power, cooling, etc., to the front line systems. The front line and support systems are identified in Table 3.1.1-11 and the dependencies of major front line system pumps and valves upon the support systems are shown in Table 3.1.1-13. The dependencies of some major support system components upon other support systems is shown in Table 3.1.1-14. The level and type of support function varies with the dependency. For example, cooling dependency are continuous, but DC power may only be required for initiation. Similarly, long term room cooling may not needed immediately, but may be required for a period of several hours after initiation of the system.

3.1.1.2.3 Initiating Event Success Criteria

The system success criteria have been developed for four groups of events: LOCAs, Transients, Steam Generator Tube Rupture (SGTR), and Anticipated Transients Without Scram, as the requirements for each of these groups are different. The sequence of events following a reactor trip is inherently dynamic and therefore the system requirements will change with time. This is handled in the tables by identifying early and late requirements. The dependency of the requirements for later systems on the success of the earlier systems is also indicated in these tables.

Success Criteria for Transients with Reactor Trip

By definition, a transient involves no substantial loss of coolant from the RCS; and, therefore, no immediate injection from the charging pumps or low head safety injection systems is required. As mentioned in Section 3.1.1.1.1, some transients may result in consequential small loss of coolant accidents as the result of the failure of a pressurizer PORV to reclose, or a steam generator tube rupture. Similarly, failure of all charging pumps and all CC

pumps will give conditions where there is possibility of reactor coolant pump seal failure (Westinghouse, 1988c). As the dominant failures of these two systems are not related to the majority of transient initiating events (the exception being Loss of Offsite Power leading to Station Blackout), the requirement for their operation has been excluded from the transient success criteria and will be addressed as a potential special initiating event equivalent to a small LOCA.

The success criteria for transients are shown in Table 3.1.1-15 and discussed in the following paragraphs.

Reactivity Control

In the case of transients that do not occur as the result of reactor trips or shutdown, automatic insertion of the control rods is required. The failure criterion is that two or more control rod assemblies fail to insert (Westinghouse, 1988a,b). Manual insertion of the control rods following failure of the reactor trip breakers is not considered success in this function, but is included in the success criteria for reactivity control following failure of automatic reactor trip in the ATWS success criteria.

Core Heat Removal

Heat transfer from the core to the steam generators (secondary heat removal) is performed by the RCS in either pumped or natural circulation modes, given the availability of secondary heat removal.

If secondary heat removal is not available, core heat removal is performed by bleed and feed cooling followed by high head recirculation. A Westinghouse investigation of total loss of feedwater events (Westinghouse, 1980a) showed that successful core cooling could be achieved by initiation of one charging pump and blocking open two PORVs within five minutes of the emptying of the steam generators. This effectively turns the transient into a LOCA. Depressurization is not rapid, as the flow capacities of the two PORVs is only just above the decay heat rate in the early stages of the event. Depending on the rate of RCS cooldown through the secondary system, the RCS may not be depressurized by the time the RWST is empty, and therefore high head recirculation is required to continue core heat removal.

A subsequent Virginia Power analysis using a RETRAN model of North Anna has determined that, for similar loss of feedwater events, only one PORV and one charging pump is required for successful feed and bleed cooling. This analysis found similar success for feed and bleed cooling after a T1 Loss of Offsite Power, again with only one PORV and one charging pump. These less stringent success

criteria are used for the North Anna IPE, since they were derived from plant specific models.

Secondary Heat Removal

Successful secondary heat removal is performed by use of any one of three Auxiliary Feedwater pumps, or one main feedwater pump and one condensate pump feeding one Steam Generator. Steam relief can be through any one of the MS atmospheric dump valves, safety valves, or condenser dump valves (Westinghouse, 1980a).

Note that at North Anna the flow from the turbine driven AFW pump (TDP) is limited to about the same flow rate as the motor driven pumps (MDP) by an orifice plate, and local manual action is required to restore the full TDP flow. Moreover, each Auxiliary Feedwater Pump (AFWP) delivers flow to only one steam generator, and again, local manual actions are required to reroute the flow from one pump to other steam generators.

Reactor Coolant System Integrity

During the course of some transients the RCS pressure boundary will be breached by the opening of a pressurizer PORV. If the PORV fails to reclose and the operator cannot close the associated block valve, then the RCS pressure boundary remains open, and the transient becomes a small LOCA with the similar success criteria requirements. The PORV sizing is equivalent to a small LOCA (Westinghouse, 1980b).

Containment Conditions

Following successful secondary heat removal there is no off-normal transfer of decay heat to the containment, and therefore no requirement for the spray systems.

In the event of bleed and feed, all decay heat is being transferred to the containment, and therefore containment heat removal will be required to prevent containment failure. The opening of a pressurizer PORV is equivalent to a break at the smaller end of the medium LOCA range (Westinghouse, 1980b), and is discussed in the LOCA success criteria for that break size. Analysis by Stone and Webster (SWEC, 1985) for Surry show that one of the four inside or outside recirculation spray trains is adequate to prevent containment failure. MAAP analysis for North Anna confirmed this to be true for North Anna.

For the purposes of the IPE, certain conservative assumptions were made about the need for Quench Spray and Casing Cooling to provide NPSH for the Recirculation Spray pumps. For large LOCAs or for

small LOCA's without successful SI injection, the Quench sprays were assumed to be necessary to provide NPSH for the IRS pumps and the Casing Cooling system was assumed to be necessary to provide NPSH for the ORS pumps.

Success Criteria for Loss of Coolant Accidents

The leak size can vary from negligible to a 31" equivalent diameter break. In the initiating event section, a LOCA is defined as a break of equivalent diameter greater than 3/8". As there are a number of different systems which can provide injection following the break, which may or may not be effective depending on the break size, LOCAs are divided into three break sizes (the appropriate references in each case are identified in the respective tables):

1. A break size of 6"- 31" equivalent diameter. The lower bound is the minimum break size that will cause rapid depressurization such that the accumulators and a low head SI pump will provide adequate core cooling. The upper bound of 31" represents the limit of two accumulators and one low head SI to provide adequate injection to meet the acceptance criteria. This is defined as a Large LOCA.

2. A break size less than 2" and greater than 3/8" equivalent diameter will require safety injection, but the break will not initially remove all the decay heat and therefore, there is an additional requirement for secondary heat removal. This is defined as a small LOCA.

3. A break size in the range of 2"- 6" equivalent diameter is defined as a medium LOCA.

Quench spray

The success criteria for each of these break sizes are discussed in the following sections.

Success Criteria for Large LOCA (A)

The success criteria for large LOCA are shown in Table 3.1.1-16.

<u>Reactivity Control</u>: No automatic reactor trip is required, as the rapid depressurization leads to significant void formation, and subsequent safety injection adds sufficient borated water to establish and maintain subcriticality.

<u>Core Heat Removal</u>: The success criteria are the same as for Surry (Virginia Power, 1991c). If the break is in one of the cold legs, then the associated accumulator will be ineffective, as the break

size will be considerably greater than the accumulator injection line. Similarly, the SI injection line into the cold leg in which the break occurs will be equally ineffective.

Following the depletion of the RWST, core heat removal is maintained by the use of one low head pump in its low pressure recirculation mode.

Secondary Heat Removal and RCS Integrity: Not applicable in the case of the Large LOCA.

Containment Condition: Following a large LOCA, the Inside and Outside Recirculation Spray systems will be initiated on Hi-Hi Containment pressure as indicated in the UFSAR (Virginia Power, 1992). The NPSH for these operations is marginally adequate as shown for Surry (Virginia Power, 1991b) until some of the RWST water has been emptied into the containment. Prior to the accumulation of adequate water inventory in the sump, sufficient NPSH for the IRS pumps would be assured by operation of the Quench Spray system and sufficient NPSH for the ORS pumps is assured by operation of the Casing Cooling system. In the long term, containment integrity is ensured by operation of one of the four recirculation spray trains, including its heat exchanger and associated service water (Donahue, 1985).

Success Criteria for Medium LOCA (S1)

The medium LOCA success criteria are shown in Table 3.1.1-17. Decay heat is removed through the break, but complete depressurization to the low head SI shutoff does not take place prior to violating the fuel acceptance criteria, in the absence of any other action.

Reactivity Control: As credit is taken in one of the success paths for cooldown using secondary heat removal, automatic shutdown by means of the RPS is required.

Core Heat Removal: Westinghouse analysis (Westinghouse, 1979) shows that one charging pump and two accumulators are the minimum requirements for the critical break size in this LOCA group (3 inch break), in the absence of secondary heat removal.

Alternatively, as shown for Surry (Beynon, 1988) one charging pump is adequate if secondary heat removal is available (success path two).

In both cases when the RWST is empty change over to high head recirculation is required for the lower bound of the medium LOCA size range.

If the charging pumps are unavailable, then core heat removal can be provided by accumulator injection and two low head SI pumps in conjunction with secondary heat removal as described in WCAP-9754 (Westinghouse, 1980b).

Secondary Heat Removal: For success path two, the analysis referenced above shows that one AFW pump to one steam generator will be sufficient for success.

For success path three, the analysis summarized in WCAP-9754 shows that steam dump to the condenser (to atmosphere in case of North Anna) and 50% of full auxiliary feedwater flow pumps are required to achieve the necessary depressurization. Because of the arrangement of one AFW pump aligned to one steam generator the success criterion is 2 of 3 AFW pumps and an operable pressure relief valve on each of the operable steam generators.

RCS Integrity: This is lost as a result of the initiating event.

Containment Conditions: The containment heat removal conditions are assumed to be the same as those for the large LOCA.

Success Criteria for Small LOCA (S2)

The small LOCA success criteria are shown in Table 3.1.1-18. Westinghouse has analyzed the various alternative methods to prevent core damage following small and medium LOCAs. The analysis results identify success paths with and without the operation of the high head charging pumps.

Reactivity Control: There is relatively little (if any) void formation in the early stages of the small LOCA. Therefore, the reactor protection system is required to operate for immediate shutdown. Shutdown is also achieved later by injection from the high head charging pump. If this is not available, depressurization through secondary heat removal cannot be achieved unless reactor trip is successful.

Core Heat Removal: In the first success path, this is achieved initially by one of the three charging pumps (Westinghouse, 1979). When the refueling water storage tank is empty, core heat removal

continues using the charging and low head SI pumps in high head recirculation.

In the second success path, core heat removal is performed using one of the three charging pumps and blocking open one PORV. For the upper bound break, it is clear that blocking open one PORV would increase the break size to the point where all decay heat would be removed. However, it is not clear generically (Westinghouse, 1980c) that blocking open one valve would be adequate for the entire range of break sizes in this category. However, plant specific MAAP calcs show that one PORV and one charging pump are adequate for the small end of the range. Later heat removal is performed by high head recirculation.

In the third success path core heat removal is performed by injection from two accumulators and both low head SI pumps (Westinghouse, 1980c). It is not clear that the accumulators are necessary for success or if success can be achieved with one low head pump, but the above criteria have to be used because success with lesser equipment is not demonstrated in the Westinghouse document.

<u>Secondary Heat Removal</u>: In the first success path, secondary heat removal is used to remove the decay heat in addition to the decay heat removal through the break. One of the three auxiliary feedwater pumps feeding one steam generator is adequate to remove all decay heat, regardless of the break size. The minimum auxiliary feedwater requirement was shown to be sufficient to maintain coolant temperatures even if all RCPs are operating (Virginia Power, 1991c).

In the second success path all secondary heat removal is assumed lost.

In the third success path analysis performed by Westinghouse (Westinghouse, 1980c) shows that in the absence of charging pump injection, the use of steam dump to the condenser (or alternatively, to the atmosphere through the steam generator atmospheric dump valves - SG ADV) and full auxiliary feedwater flow can achieve sufficient heat removal to cooldown and depressurize the RCS to the point where accumulators, and subsequently, the low head SI pumps can inject, and ensure core acceptance criteria are not violated. Surry specific MAAP analysis showed that success can also be achieved through 50% (i.e., 1/2 of maximum capacity) AFW flow utilizing one SG ADV or condenser dump valves. If the condenser steam dump option is not available due to the presence of an SI signal (which is assumed to be the case for all LOCAs), the SG ADVs are the only means for the rapid cooldown required in this path.

Due to dissimilarities between North Anna and Surry, the Surry criteria were not directly applicable to North Anna. Due to the AFW pump alignment at North Anna, 50% AFW flow can only be achieved by 2 pumps injecting into two generators. Thus, the success criteria adopted for North Anna were 2 pumps feeding two generators, each with an operable ADV.

RCS Integrity: This is lost as a result of the initiating event.

Containment Conditions: Surry MAAP analysis (Virginia Power, 1991c) shows that only one recirculation spray train is required in success path two. In the case of both success paths one and three, there are indications from previous analysis (Cybulskis, 1985) that the combination of secondary heat removal and normal containment heat losses may be adequate to remove all the heat from containment. This path was not pursued for the North Anna IPE, except for very small breaks where containment heat removal is not required.

Success Criteria for Anticipated Transient Without Scram (TH, TL)

As power generation continues following the initiating event in the event of an ATWS, the success criteria are very different from those following a normal transient. In particular, the resulting power generation/heat removal capability mismatch results in very rapid heat up of the RCS and subsequent pressure increase following the rapid expansion of the coolant, in the case of a loss of feedwater or turbine trip ATWS.

The most recent analysis which takes into account the full range of core loading schemes suggested in Westinghouse plants is WCAP-11993 (Westinghouse, 1988a), which clearly identifies the range of parameters which impact the sequence of events following an ATWS and the systems required to prevent core damage.

Earlier studies [NUREG/CR-4550, WCAP-8330 (Westinghouse, 1974)] concentrated on the effects of the moderator temperature coefficient (MTC) on limiting the peak pressure during the course of the transient and using cutoff values, such as -7 pcm, as a critical value for determining the percentage of time over-pressure transients would be restrained below 3200 psig. With extended life cores (18 - 24 months), this would give an overly conservative portion of time when the core would be vulnerable to ATWS events. In this study the approach recommended in WCAP-11993 has been adopted and therefore, the interrelated dependencies of core reactivity feedback, pressure relief capacity, and the time in core life when the transient occurs have been taken into account. Pressure relief capacity includes consideration of the availability of the PORVs as well as the safety valves.

The other events which impact the success criteria are the power level at which the ATWS occurs, the inclusion of the ATWS Mitigating Systems Actuation Circuitry (AMSAC) and the main feedwater availability. Therefore, the success criteria identified in Table 3.1.1-19 are drawn up for three sets of conditions which the Westinghouse analysis identifies as significant:

1. Reactor power less than 40%,

2. Reactor power greater than 40% and main feedwater available,

3. Reactor power greater than 40% and main feedwater not available.

The success criteria for each of these conditions, in terms of the five basic functions associated with core damage identified in Section 3.1.1.2.1, are shown in Table 3.1.1-19 and discussed in the following paragraphs.

Reactivity Control: Long term shutdown can be achieved either by injecting borated water using the charging pumps taking suction from the RWST, or the CVCS. In addition, if there is no mechanical failure associated with the control rods, they can be fully inserted by tripping the breaker at the rod drive motor generator sets or by manual reactor trip from the control room. The timing associated with these events in WCAP-11993 is 10 minutes.

Core Heat Removal: Core heat removal is performed by the reactor coolant system.

Secondary Heat Removal: It can be seen from the table that the secondary heat removal requirements are dependent upon the power level at the time the scram is required and the availability of main feedwater. If reactor power is below 40% one auxiliary feedwater pump must deliver flow to the steam generator. If reactor power is above 40% and total loss of feedwater is not the initiating event, one of the three main feedwater pumps must continue to operate. Following shutdown one auxiliary feedwater pump is adequate as for the normal transient.

For loss of feedwater initiated ATWS events there is a large imbalance in the heat source/sink relationship. This imbalance results in degradation in the heat transfer behavior between the primary and secondary systems. When the steam generator tubes are exposed secondary heat transfer is further reduced. As the result of this, reactor coolant temperature and pressure continue to rise and the pressurizer fills and releases water through the PORVs and safety valves. The peak pressure will depend on the moderator

temperature coefficient and availability of the PORVs and the safety valves to handle the volumetric in-surge of water into the pressurizer. The minimum requirement in this case is two auxiliary feedwater pumps. If all three auxiliary feedwater pumps are available the subsequent pressure relief requirements are lower, but this case was not incorporated into the IPE.

Reactor Coolant System Integrity: For limiting transient events for which peak RCS pressure could be of concern during an ATWS event, the occurrence of an over-pressure condition is a function of reactivity feedback (which will limit the peak power), pressure relief capacity, time in core life, and type of transient. The pressure relief includes the availability of the pressurizer power operated relief valves as well as the safety valves.

For transients from below 40% power or those in which feedwater is not lost there is no significant challenge to the RCS integrity. However, for loss of feedwater events above 40% power there is the necessity to ensure turbine trip (AMSAC), and the availability of adequate pressure relief. The required pressure relief is calculated for the various stages of core burnup during the cycle. The period of time when pressure relief is inadequate regardless of the number of available relief valves is calculated in accordance with WCAP-11993. The results are summarized in Appendix B.

Containment Conditions: In all cases, sequences which lead to core damage following failures of one of the above functions are not impacted by containment systems.

Success Criteria for Steam Generator Tube Rupture (T7)

The success criteria for the Steam Generator Tube Rupture (SGTR) initiating event are different from those for small LOCA and Transients in that an additional requirement exists for using the RHR system after failure to isolate the RCS from the outside environment. This happens in one of two ways. First, if the operator fails in early cooldown and depressurization, there is the potential for the SG safety relief valves or the ADV to fail open. Additionally, the operator could fail to isolate the ruptured steam generator. In both of these cases, core damage can only be prevented by a long term cooldown to atmospheric pressure, before RWST depletion, and use of RHR cooling thereafter. Success of HHSI is important only in so far as impacting the available time for operator actions, i.e., there is also a success path with failure of HHSI by cooldown and depressurization of the RCS with the secondary system to a pressure below that in the intact steam generators, which terminates the loss of RCS inventory.

Reactivity Control: As in small LOCA, reactor protection system is required to operate for immediate shutdown.

Core Heat Removal: This and all other event tree functions are a mixture of transient and small LOCA functions, because the amount of leakage from the rupture of one or two tubes is very small. The core will not be uncovered for a relatively long time without high pressure injection.

In the first success path, no injection is required as long as the RCS is rapidly cooled down and depressurized by the secondary system, and the affected steam generator is isolated, thus terminating the loss of primary system inventory early in the transient.

The second success path is used when the faulted steam generator is not isolated, and/or a secondary safety relief valve sticks open on the faulted steam generator. In this case, the stable end point can only be achieved by using high pressure injection initially, and the RHR system after the RCS is cooled down and depressurized.

Success paths three and four are similar to their counter parts in small LOCAs, with the use of feed and bleed, and core cooling recovery operations, respectively.

Secondary Heat Removal: In success path one, the ruptured steam generator is isolated and there are no stuck open safety relief valves (SRVs), and one of the two remaining steam generators is used with the associated SRVs using flow from one AFW pump. Success path two has the same requirements for secondary heat removal, with the difference that containment is bypassed with an intact core. Success path three has no secondary cooling required using path four represents core cooling recovery using enhanced AFW.

RCS Integrity: RCS integrity is achieved by cooldown and depressurization of the RCS and isolation of the affected steam generator in success path one and five. In other paths, RCS integrity is lost through induced LOCA (path 3), or an open path to outside containment with an intact core (paths 2 and 4).

Containment Conditions: Containment heat removal is not required, except for the case of feed and bleed operation in success path three.

3.1.1.2.4 Conclusions

The success criteria are essentially the same as those identified in the Surry IPE based on the same assumptions and acceptance criteria in that study. However, by using additional North Anna (and Westinghouse) analyses, slightly different success criteria for the small and medium LOCAs, and SGTR and ATWS are included in this study.

3.1.2 Event Trees

This section of the report describes the event trees that model the response of the North Anna systems and operations personnel to the initiating events identified in Section 3.1.1 and includes a brief discussion on the methodology used in their modeling. The event trees are constructed to reflect the success criteria discussed in Section 3.1.1. The event trees for transient with scram events are described in Section 3.1.2.1, those for LOCAs are in Section 3.1.2.2. The special initiators are discussed in Section 3.1.3.

The event tree model is the central analytical tool used in the determination of the frequency of core damage, the plant damage states and the various ways in which they can occur. As the principles of its development are well documented in the PRA Procedures Guide (NRC, 1983), and the Interim Reliability Evaluation Program Procedures Guide (NRC, 1982a), they are not described in detail in this report. Some discussion is, however, provided on those aspects of the development of the event trees that are specific to the present study.

The initiating event task identified the LOCA and transient initiating events, grouped according to the systems required to prevent core damage. The system success criteria task defines the specific requirements for each of the systems. The information from these two tasks combined with information from the dependency matrix for those initiating events involving failure of various support systems forms the basis for the construction of the event trees.

The most important aspect of developing the event trees from the above information is to reflect the inherent functional and physical dependencies between each phase of the sequence and, at the same time, the interaction between operators and systems as the sequence unfolds. Thus, the event tree is developed by first considering those functions (reactivity control, early core heat removal) that are required early and then those which are required in the long term. In this way, it is relatively straight forward to model dependencies between functions. For example, in the case of a large LOCA, injection is required immediately. If injection fails, it will not be necessary to consider recirculation.

Operator interactions, such as cooling down and depressurizing the RCS, when called for in the emergency procedures, are specifically identified, so that the relationship between the success or failure of various front line systems and the use of the various success criteria can be clearly identified.

Finally, it will be noted that, for initiating events that are not related to loss of a support system, the event tree addresses only systems which perform the functions identified in Section 3.1.1. The dependency of each of these systems on the various support systems is developed in the individual systems analysis. In the case of loss of a support system, such as Loss of Offsite Power, Loss of Service Water, or Loss of Emergency Switchgear Room Cooling, recovery of the support system is included in the event trees.

The results of this phase of the study are the identification of the individual plant damage state sequences and the detailed analysis requirements for determining the timing and progression of each accident sequence. The timing is required in order to evaluate the impact of the operator actions, and the time of occurrence of the automatic systems initiation signals.

3.1.2.1 Transients With Successful Scram

The classes of transient initiators are identified in Table 3.1.2-1. In order to accommodate the range of transients in the table, it is necessary to develop event trees for the following initiators:

T2:	Non Recoverable Loss of Feedwater
T2A:	Feedwater Isolation - Feedwater Recoverable
T3:	Main Feedwater Initially Available
T5A:	Loss of DC Bus 1-I (Non Recoverable)
T5B:	Loss of DC Bus 1-III (Non Recoverable)
T9A:	Loss of 4160 V Bus 1H (Non Recoverable)
T9B:	Loss of 4160 V Bus 1J (Non Recoverable)

The fundamental requirements for all transients in the above classes are the same. Differences in the availability of each function in Table 3.1.2-2 vary depending on the initiating event resulting in the necessity for the different event trees. In order to avoid repetition, the basic functions in each of the event trees are discussed under the heading of general transient. The event boundary conditions for quantification of each of these trees are described in Appendix B.

The remaining transient event trees are then described individually in Section 3.1.3.

It was decided to develop a specific event tree for the loss of RCP seal cooling, rather than include this failure as a consequential failure in all transient event trees as was done in NUREG/CR-4550 (Bertucio, 1990). With the exception of the Loss of Offsite Power leading to Station Blackout, the failure of both CC and Charging Pump cooling are unrelated to the transient initiating event, unless loss of both is the initiating event. Therefore, the loss of both systems is treated as an initiating event in a separate event tree in Section 3.1.3.

Table 3.1.2-2 gives a brief description of events used as event tree headings.

3.1.2.1.1 General Transient Event Tree

The functions appearing in the event trees T2, T2A, T3, T5A and T5B, T9A and T9B (see Figures 3.1 series) are discussed below:

K - Reactor Subcritical

Failure to achieve successful shutdown is identified as failure of the reactor trip breakers to open or automatic insertion of two or more full length control assemblies following the generation of a Reactor Protection System trip signal.

Hv - Emergency Switchgear Room Cooling Available

Analysis of the Emergency Switchgear Room (ESGR) shows that room cooling is required for continuous operation of the safeguards equipment. Failure of room cooling is assumed to result in failing open of all 4160 V breakers, resulting in a loss of emergency power. This function includes a dependency on Service Water. Successful room cooling is achieved with one ESGR AHU and one chiller for all transients. Failure of this function is transferred to the special event trees developed to model loss of Emergency Switchgear Room cooling.

Q - Reactor Coolant System Boundary Intact

Analysis (Westinghouse, 1981), shows that there are different demand probabilities for pressurizer PORV opening for different classes of initiating events. Reactor Coolant System integrity is maintained if, following the demand, the PORV recloses or the operator closes the appropriate block valve. The function Q is quantified by combining the probabilities of demand with the probability of the PORV failing open and the failure of the operator to close the block valve if the PORV fails open. The sizing of the PORV is such that a failed open PORV is equivalent to

a small LOCA; thus, failure of this event transfers to the small LOCA tree (S2).

L - Auxiliary Feedwater

Secondary heat removal is performed by a combination of Feedwater injection and steam relief. Successful operation of Auxiliary Feedwater requires one Auxiliary Feedwater Pump to supply rated flow to one Steam Generator. Successful operation implies long term success and, therefore, no additional front line systems are required. The required steam relief can be performed by either Condenser steam dump valve, atmospheric steam dump valve, or Main Steam safety valves.

M - Main Feedwater

If Auxiliary Feedwater is not available, one train of Main Feedwater - that is, one Feedwater and one condensate pump - can be used to maintain Steam Generator inventory. Since the Main Feedwater pumps at North Anna are motor driven for transients in which Feedwater is not lost, using the normally-running motor driven pumps is an option available to the operators. The steam relief requirements are the same as for Auxiliary Feedwater.

The emergency procedures instruct the operator to lower Steam Generator pressure and use the Condensate System to provide Feedwater to the Steam Generator in the event of failure of Main and Auxiliary Feedwater Pumps. However, he is also instructed to go to bleed and feed when any two wide range channels indicates less than 27% level. At this level in the SG, with failure of AFW and MFW at the time of scram, the pressure would not be low enough to use condensate alone; so, it has not been included in this function.

P - Feed and Bleed Initiation

In the event of total loss of secondary heat removal, decay heat removal is achieved directly by using the Charging Pumps and the pressurizer PORVs. Emergency procedures instruct the operator to open the two pressurizer PORVs and start one Charging Pump in the Safety Injection mode, taking suction from the Refueling Water Storage Tank (RWST) when any two wide range Steam Generator level channels indicates less than 27%. When the RWST is empty, it will be necessary to change over to High Head Recirculation. This function models failure to initiate feed and bleed by successfully opening at least one PORV.

D1 - Charging Pumps Available

Following successful opening of the PORVs, injection is required from one of three Charging Pumps in order to maintain RCS inventory.

H2 - High Head Recirculation

After successful bleed and feed operation, all the useable RWST inventory is eventually transferred to the sump. Continued decay heat removal requires the initiation of high pressure recirculation, using the Low Head Safety Injection System to take suction from the sump. The Low Head SI pumps provide the necessary NPSH for the Charging Pumps to continue injecting into the RCS.

Qs - Quench Spray

Quench sprays are asked at this point in the event tree for the purpose of plant damage state delineation. Quench Sprays are not needed to prevent core damage for feed and bleed sequences. The feed and bleed process will result in sufficient water in the Containment to provide adequate NPSH for the Recirculation Sprays.

However, some of the core damage sequences do not provide a source of RWST injection into the Containment. For these sequences, it is necessary to ask Quench Sprays in order to provide water inventory for Containment heat removal.

Rs - Recirculation Spray Operable

Containment heat removal is required to maintain sump water temperatures within acceptable limits and to reduce the pressure of the Containment atmosphere. This can be provided by any one train of the Inside or Outside Recirculation Spray Systems.

Ch - Containment Heat Removal

For the purpose of plant damage state delineation, Containment heat removal has been separated into two functions, the spray action and service water to the heat exchangers. For all sequences where the Rs function has been successful, Containment heat removal is asked. Success in this function is Service Water to at least one operable spray heat exchanger.

H1 - LHSI/LHSR Late

This function is asked for the purposes of plant damage state delineation of core damage sequences. The LHSI system can provide water to the Reactor Vessel and Reactor Cavity for debris bed cooling following core damage.

3.1.2.2 Loss of Coolant Accidents (A, S1, S2)

The potential ranges of loss of coolant accidents are covered by three event trees reflecting the various success criteria for the size and location of the LOCA. Each of these event trees is discussed in the following sections.

3.1.2.2.1 Large LOCA Event Tree (A)

The sequence of events following a large LOCA can be divided into four phases: blowdown, refill, reflood and long term cooling. Due to the rapid depressurization of the RCS and the rise in Containment pressure, a Safety Injection signal is generated within seconds giving a starting signal to the charging and Low Head SI pumps. As the pressure decreases below the pressure setpoint of the Accumulators, they will inject into the Reactor Vessel. As indicated in the UFSAR (Virginia Power, 1992), the steam blowdown rate raises the Containment pressure high enough to activate the Quench Spray, and Inside and Outside Recirculation Spray systems.

The RWST inventory decreases rapidly as injection continues with full flow from two charging pumps, two Low Head SI pumps and two Quench Spray pumps, all of which take suction from the RWST in the injection phase. When the RWST low level is reached automatic switchover will occur if the operator fails to manually swapover, the operator must ensure that the change over to recirculation occurs.

A large break in one of the cold legs can have an impact on the ability to cool the core in the long term as an extended period of boiling in the core may lead to the precipitation of sufficient boron to block flow through the core, thereby preventing a loss of cooling and ultimately core damage. To counteract boron precipitation, it is necessary to manually change over to hot leg recirculation, thereby producing reverse flow through the core.

In the long term, Containment heat removal is provided either by the Inside or Outside Recirculation Spray systems. Both of these systems are initiated by the high Containment pressure early in the accident. As there is only a small quantity of water in the sump, it is necessary for the Quench Spray to start at the same time to provide sufficient NPSH for the operation of the Inside

Recirculation Spray system. The headings for the large LOCA event tree shown in Figure 3.1-A are discussed below.

Hv - Emergency Switchgear Room Cooling Available

Success in this function is achieved by operation of one ESGR AHU and one chiller.

D2 - Accumulators Inject

The Accumulators are designed to inject when RCS pressure drops to 665 psia (Virginia Power, UFSAR, 1992). For a large LOCA, two effective Accumulators are required to inject. As this break is larger than the injection line, injection flow into the ruptured loop is not considered to be effective; therefore, the two remaining effective Accumulators are required to inject.

D3 - Low Pressure Safety Injection

For successful injection, one of the two Low Head SI pumps must start on receipt of the SI signal and inject water into the Reactor Vessel. Injection through the leg associated with the ruptured loop is also assumed to be lost, as in the case of Accumulator injection.

Qs - Quench Spray Available

The high Containment Building pressure will initiate the Quench Spray and the Inside and Outside Recirculation Spray Systems. Initially, the Quench Spray Systems will be taking suction from the RWST. Its operation is also necessary in providing sufficient NPSH for the operation of the Inside Recirculation Spray System. The minimum requirements are for successful operation of one train.

Rs - Recirculation Spray Available

If the Quench Spray system has operated, then success in this function is operation of either one of two Outside Recirculation Spray trains or one of two Inside Recirculation Spray trains.

If the Quench Sprays have not operated, then the IRS are assumed to have failed on inadequate NPSH. Thus, for these sequences, success in this function is operation of one of two Outside Recirculation Spray trains.

Ch - Containment Heat Removal

For the purposes of plant damage state delineation, Containment heat removal has been separated into two functions, the spray action and Service Water to the RS heat exchangers. For all sequences where the RS function has been successful, Containment heat removal is asked. Success in this function is Service Water to at least one operable Recirculation Spray heat exchanger.

H1 - Low head Recirculation

When the RWST level reaches its low setpoint, the Low Head SI System is automatically changed over to the recirculation mode. The sump suction valves open and the RWST suction valves close. Successful recirculation requires one train of the Low Head SI System to circulate water from the sump to the cold leg injection points.

Dh - Hot Leg Recirculation

To avoid channel blockage due to boric acid precipitation, the operator is required to transfer from cold leg to hot leg recirculation approximately 10 hours into the event. Failure in changeover is assumed to lead to core damage. Failure of the recirculation mode during the first 24 hours is modeled in event H1, so the only failures modeled in this function are the operator actions and the opening of the valves.

3.1.2.2.2 Medium LOCA (S1)

A medium LOCA is defined as a break in the RCS boundary in the range of 2" - 6". The loss of coolant will lead to a slower depressurization than in the case of the large LOCA but, as the break is large enough for all decay heat to be dissipated, secondary heat removal is not essential in order to prevent core damage. However, if secondary heat removal is available it can be used as an alternative to the Accumulators or the High Head Charging Pumps. The combination of systems which can be used to provide injection and cooling are discussed in the success criteria.

At the higher end of this break size, the Containment Building pressure will rise to the point where the Inside and Outside Recirculation Spray Systems will be initiated before the RWST is empty. A MAAP analysis was performed to confirm the existence of adequate NPSH. It was shown that there is adequate margin, as long as there is an operable injection source to the reactor vessel.

Failure of the Charging pumps to inject will lead to core damage, if the operators take no action. The operators are instructed to use all possible means (e.g., secondary heat removal) to cooldown and depressurize the RCS to allow the Low Head SI pumps to inject. However, the timing for this action is very short and; therefore, based on the human reliability analyses, the probability of achieving this condition is not high.

For any LOCA with a break diameter of 2" to 6" the injection flow should refill the reactor vessel and hot legs, hence the decay heat is transferred to the circulating RCS water. This limits boron precipitation and the necessity to change over to hot leg recirculation. The headings for the medium LOCA event tree shown in Figure 3.1-S1 are discussed below.

K - Reactor Subcritical

The system requirements for this function are the same as in the transient event trees.

Hv - Emergency Switchgear Room Cooling Available

This is the same as for large LOCA.

D1 - High Pressure Injection

One of three Charging Pumps is required to inject into the RCS. The Charging Pumps will receive an SI signal and will automatically align to the RWST. If Auxiliary Feedwater is available, immediate start is not required, and therefore, credit can be taken for the operators to start a pump. The time available for this and the resultant human error probability are evaluated separately from the other LOCAs.

D2 - Accumulator Injection

The requirements are the same as for the large LOCA; however, the Accumulator associated with the affected loop is not assumed failed.

L - Secondary Heat Removal

The Motor Driven Auxiliary Feedwater Pumps are automatically initiated by the SI signal and the MFW pumps tripped by the SI signal following the occurrence of the LOCA. If HHSI succeeds, the requirements are that one AFW pump inject into its corresponding Steam Generator. Credit was taken for manual starting of the

pumps. Following success of High Head injection, the operator will attempt to cooldown and depressurize with one AFW pump.

In the event of failure of HHSI and the subsequent need for core cooling recovery, the AFW requirements are 2 AFW pumps injecting into their corresponding generators. If both HHSI and the Accumulators fail, core damage is assured regardless of AFW availability.

O - Cooldown and Depressurize

Following successful HHSI injection, the operators are instructed to cooldown and depressurize. Successful completion of this operation prior to emptying the RWST will enable Low Head recirculation to be used instead of High Head recirculation.

Y - Core Cooling Recovery

For breaks less than four inches, core cooling recovery is required following failure of high pressure injection. This requires the operator to initiate Main Steam atmospheric dump to the atmosphere once he has indication of inadequate core cooling - that is core exit thermocouple readings of 1200°F. Success in this function will then reduce the RCS temperature and pressure to the point where the Low Head SI pumps can inject into the RCS.

D3 - Low Head Safety Injection

Following core cooling recovery actions, two of two Low Head SI pumps are required to inject to reflood the Reactor Vessel.

H2 - High Pressure Recirculation

If the operators have not depressurized, then at the lower end of the medium size LOCA, the RCS pressure will still be above that of the Low Head SI pumps. In this case, change over to one train of High Head recirculation will automatically take place on RWST low level.

H1 - Low Pressure Recirculation

In all cases when RCS inventory is being maintained by the Low Head SI pumps, the system will change over to low pressure recirculation on RWST low level.

Qs - Quench Spray

The requirements for this function are the same as for transients.

Rs - Recirculation Spray Operable

The requirements for this function are the same as for the large LOCA when the Quench Sprays have operated (i.e., either one of two Inside or one of two Outside Recirculation Spray trains).

Ch - Containment Heat Removal

The requirements for this function are the same as for the large LOCA.

3.1.2.2.3 Small LOCA (S2)

As a small break LOCA is not capable of removing all the decay heat following a reactor trip (on low pressurizer pressure and the generation of the SI signal), reactor pressure will remain high. Since all decay heat is not being removed through the break, Auxiliary Feedwater is required following the trip of the Main Feedwater by the SI signal. Both Low Head SI Pumps will be stopped as they will not be required in the short or medium term.

If secondary heat removal is not available, there will be a build up of pressure in the RCS until the pressurizer PORVs open leading to a loss of coolant through the PORV and the break. Loss of coolant will be greater than injection flow; therefore, RCS inventory is gradually reduced and the core is eventually uncovered. Under the conditions of loss of all secondary heat removal, the operator is instructed to go to bleed and feed operation when any two wide range Steam Generator level channels indicate less than 27%, thus maintaining core cooling and avoiding core damage.

With successful SI flow and secondary heat removal, the RCS pressure stabilizes at a pressure above the Accumulator injection pressure, and remains there for a considerable period of time. The operator has two courses of action depending on break size. For the upper end of the break size, operator initiated cooldown and depressurization, maintaining a cooldown rate of 100°F/hour will allow the Low Head recirculation point to be reached before the RWST is emptied. Thus, Low Head recirculation can be used rather than High Head recirculation. For the smaller end of the break spectrum, cooldown and depressurization will eliminate the need for recirculation altogether, allowing RHR closed cycle cooling to be used for long term decay heat removal. For both break sizes, if the operator does not cooldown, high pressure recirculation will

have to be used for long term inventory control and decay heat removal.

In the event of failure of all Charging Pumps, Westinghouse analysis has shown that secondary cooldown by Main Steam atmospheric dump valves can be used to cooldown and depressurize the RCS to allow the Accumulators and Low Head SI pumps to inject and restore decay heat removal. The cue for this action in the emergency procedures is the inadequate core cooling conditions of 700°F or 1200°F as indicated on the core outlet thermocouples.

The headings for the small LOCA event trees shown in Figure 3.1-S2 are discussed below:

K - Reactor Subcritical

The system requirements for this function are the same as in the transient event trees.

Hv - Emergency Switchgear Room Cooling

The system requirements for this function are the same as in the transient event trees.

D1 - High Pressure Injection

The system requirements for these functions are the same as for the medium LOCA.

L - Auxiliary Feedwater

As in the transient case, secondary heat removal is performed by the combination of Main Steam relief and Feedwater injection. Following successful initiation of high pressure injection success in this function requires one of three AFW Pumps supplying water to one of three Steam Generators.

In the event of failure of high pressure injection, two of three AFW Pumps are required to supply Feedwater to two Steam Generators in order to achieve core cooling recovery actions. The one-to-one AFW lineup at North Anna becomes the limiting factor in this sequence. Cooling to two generators is required for core cooling recovery to be effective. For core cooling recovery sequences, there is insufficient time for local realignment of the AFW system.

Fm - Small Break Size

This heading is asked to partition small LOCAs into small and very small LOCAs. The S2 break size includes breaks from 2" down to the point that they are no longer considered LOCA initiating events (i.e., do not cause an SI signal). The heading will separate those break sizes that do not cause a CDA signal from those breaks that do cause a CDA signal. Generation of a CDA signal occurs on a containment pressure Hi-Hi signal. This will activate Quench Sprays. If the QS pumps do not actuate, there is ample time for the operator to cooldown, depressurize, and never require ECCS recirculation.

This function is a split fraction representing those breaks that cause a CDA signal within four hours of the SI signal. Four hours was chosen as an appropriate time to expect operator action to cooldown and depressurize. As the RCS is depressurized, there is less and less potential for a CDA signal.

P - Bleed and Feed Operable

In the event of a small LOCA, if Auxiliary Feedwater fails, satisfactory core heat removal can be maintained by opening a pressurizer PORV to increase the effective LOCA size. At the upper limit of 2", opening one PORV would be sufficient. MAAP runs have shown that one PORV is also sufficient at the lower end of the break size.

O - Operator Cooldown

Operator cooldown and depressurization is the expected course of a small break. If the operator is successful in these actions, the need for high pressure recirculation is obviated for the larger break sizes (up on the Fm path) and the need for any recirculation at all is obviated for the smaller breaks (down on the Fm path).

Y - Core Cooling Recovery

The system requirements for these two functions are the same as for the medium LOCA. As the leak rate is much smaller, there will be more time available for the operator to perform the necessary actions.

D2 - Accumulators Injection
D3 - Low Head Safety Injection
H2 - High Head Recirculation
H1 - Low Head Recirculation

The system requirements for these functions are the same as for the medium LOCA.

Qs - Quench Spray

This heading is only asked for core damages sequences without an operable source of RWST injection into containment. The success criteria are the same as the large LOCA.

Rs - Recirculation Sprays Operable

In the case of the small LOCA, containment heat removal is required to prevent containment failure. The QS system is not required as the RS system will not be initiated until later on in the sequence when there will be adequate sump water to provide the required NPSH.

3.1.3 Special Event Trees

Separate event trees have been developed for a range of special initiating events identified in Section 3.1.1. The event trees developed in this section are as follows:

Loss of Offsite Power and Station Blackout (T1, T1A)
Loss of RCP Seal Injection and Cooling (T4)
Loss of Service Water (T6)
Steam Generator Tube Ruptures (T7)
Loss of Emergency Switchgear Room Cooling (T8)
Anticipated Transient Without Scram (TH, TL)
Interfacing System LOCA (Vx)

The system and function requirements to prevent core damage following each of the above initiators transients are discussed in the following paragraphs.

3.1.3.1 Loss of Offsite Power Event Trees (T1, T1A)

A complete Loss of all Offsite Power (LOOP) will lead to load rejection, Turbine trip, failure of all RCPs and loss of Feedwater, as well as reactor trip. The four diesel generators supplying the 4160 V Engineered Safety Features (ESF) buses receive a start signal and connect to their respective bus bars after acquiring the

necessary rpm. Emergency equipment is loaded on to the bus in the sequence determined by time delays associated with each pump.

North Anna has one diesel generator dedicated to each 4160 V ESF bus. If one of these diesel generators fails to start one train of equipment is lost in each front line and support system. The T1 event tree represents this possibility.

If diesel generator 1H and diesel generator 1J fail to start or are otherwise unavailable, a Station Blackout will occur at Unit 1. The event tree, T1A, is developed for this scenario and models the situation of station blackout at Unit 1. Operation of diesel generators at Unit 2 is handled in the fault trees for the shared systems as necessary.

If diesel generator 1H and diesel generator 1J connect to their buses, then Unit 1 has power and the event tree can be developed in a similar manner to the general transient tree.

The T1A tree represents station blackout at Unit 1. If the turbine driven Auxiliary Feedwater pump fails to start, the steam generators dry out, and the reactor coolant heats up and passes through the pressurizer PORVs, leading to core uncovery. If a PORV fails open early in the sequence of events, the time to uncover the core will be shorter than if it does not. The minimum time to core uncovery for a typical Westinghouse plant (NRC, 1985b) with a failed open PORV is approximately 80 minutes. If power is restored to the ESF buses in time to initiate safety injection, core damage is averted.

The time available to recover AC power is dependent upon a number of factors and plant conditions:

1. Whether a seal LOCA occurs as the result of failures of seal injection and thermal barrier cooling.

2. The length of time the battery will support the Steam Generator level instrumentation to allow operators to properly control flow from the turbine driven Auxiliary Feedwater Pump, and the possibility of operating the turbine driven AFW pump with no level indication.

3. Whether room cooling is required to prevent failures of the turbine driven AFW pump (not so during normal operation).

4. Whether or not there is a stuck open pressurizer PORV.

5. Depletion of the condensate supplies.

Each of these factors is evaluated for detailed sequence analysis in each of the event trees discussed below. The T1 event tree is

shown in Figure 3.1-T1. The T1A event tree is shown in Figure 3.1-T1A. A full discussion on the timing of the events is given in Appendix B.

3.1.3.1.1 Loss of Offsite Power (T1)

The T1 event tree is shown in Figure 3.1-T1. The events in this tree are described below.

K - Reactor Subcriticality

Same as for transients.

DG - EDG 1H or 1J Operable

Success of this function represents success of either of the 1H or 1J diesel or both of them. This function separates T1 from station blackout sequences. Success of this function is further delineated on this tree. Failure of this function transfers to the T1A tree (Station Blackout).

With the exception of RC pump seal cooling, all functions on this event tree have the very same success criteria as described for the T2 tree or the S2 event tree. When quantifying the functions, the boundary conditions are set for the T1 initiating event.

Slc - RC Pump Seal Cooling

This question is asked to determine if RC pump seal cooling is impaired by diesel generator failures. If the frequency of this sequence is significant, it will be further delineated via the T4 event tree, substituting proper functional assignments.

3.1.3.1.2 T1A - Station Blackout at Unit 1

The Station Blackout event tree is shown in Figure 3.1-T1A. In the SBO tree, the initiating event T1A is Loss of Offsite Power and failure of diesel generators 1H and 1J, sequence T1DG.

Q - Reactor Coolant System Boundary Intact

In the event of Station Blackout, power is unavailable to pressurizer PORV block valves and, thus, it is not possible to isolate a failed open PORV. This function is based on the probability of PORV demand, the probability a PORV fails open, and

the probability the block valve is closed at the time of the Loss of Offsite Power.

Lt - Turbine-Driven Auxiliary Feedwater Pump

Following Loss of Offsite Power and diesel generator failure, the only system available to provide Feedwater to support secondary heat removal is the turbine driven Auxiliary Feedwater pump. Success in this function is that the pump starts and runs until the operator stops it sometime after the loss of DC power and instrumentation. In the event of a failed open PORV, evaluations in the NRC study on Station Blackout (NRC, 1985b) indicate that the operation of the pump has little impact on the time to core uncovery, so it is omitted in sequences following a failed open valve.

Slc - RC Pump Seal Cooling Available

Following success in the previous function, it is necessary to determine if the cross connection from Unit 2 is used to provide RC pump seal cooling via CC to the thermal barrier. If CC flow is established then, seal failure will be prevented. Failure in this path will lead to the potential for seal failure. Cross connection of the charging system is conservatively ignored because of timing considerations.

In the event of failure of all seal cooling, there is the potential for seal failure. The time of occurrence of seal failure and the leakage rate under these conditions is extensively researched, but the results are inconclusive. In order to quantify the probability that core damage occurs as the result of a seal LOCA following Station Blackout, it is necessary to evaluate the probability that Offsite Power is not recovered by the time the core is uncovered following the occurrence of the seal LOCA. This probability is evaluated using a model for occurrence of seal LOCA and the probability of recovery of Offsite Power before core uncovery following seal failures based on the Westinghouse assessment of seal performance under these conditions (Westinghouse, 1988c). The details of the calculations are given in Appendix B.

B - Recovery of Offsite Power

The time by which offsite power has to be recovered in order to prevent core damage is determined by the performance of the turbine driven AFW pump, the occurrence of RC pump seal LOCA or the sticking open of the pressurizer PORV. The interaction between seal LOCA and offsite power recovery is integrated into the probability of non-recovery of AC power prior to core uncovery. The success of this function on the seal LOCA branch has two

components. Success can represent recovery of offsite power prior to seal failure, or it can represent recovery of offsite power with seal LOCA in progress, but prior to core uncovery. As the core uncovery time for a typical seal LOCA is about three hours, there is a significant window in which this latter sequence can occur.

Recovery of offsite power prior to seal failure requires no further functions for mitigation. Recovery of offsite power with seal LOCA in progress is treated similar to a S2 LOCA. The timing for recovery given failure of the Auxiliary Feedwater, failed open PORV or failure of the Auxiliary Feedwater in the long term is used to evaluate the probability of failure to recover offsite power.

As it is intended to take credit for the use of feed and bleed following offsite power recovery, the timing for recovery of offsite power is based on the time of Steam Generator dryout, which is the basis for initiating successful bleed and feed. A full discussion of the timing is given in Appendix B.

B1 - Recovery of AC Power Prior to Vessel Failure

This function is used to further partition plant damage states into those with power for recovery and those without.

B2 - Recovery of AC Power Prior to Containment Failure

This function is used to partition plant damage states into those with power for recovery and those without.

L - Auxiliary Feedwater Operable

Following the recovery of electrical power it will be necessary to start the motor driven Auxiliary Feedwater pumps for those sequences where the turbine driven AFW pump has failed. The success criteria are the same as for the loss of feedwater transient, as automatic initiating signals will occur and the operators are instructed to ensure the system is operating.

D1 - Safety Injection

For those sequences in which RC pump seal LOCA is in progress, safety injection will be required, similar to an S2 break. For sequences involving recovery of offsite power after a loss of feedwater, feed and bleed was assumed to be required. Although the motor driven feedwater pumps could be restored and supply feedwater to the Steam Generators, it is not clear from analysis available to the project that core cooling can be restored via Steam Generator

heat removal after the Steam Generators have dried out. Thus, feed and bleed was required.

P - Feed and Bleed

Same as T2 sequences.

QS - Quench Spray
Rs - Recirculation Spray
Ch - Containment Heat Removal
H1 - LHSI/LHSR Late

These functions are the same as for small LOCA.

3.1.3.2 Loss of RC Pump Seal Cooling (T4)

Complete loss of cooling to the RC pump seals can only occur as the result of failure of two independent systems, unless Loss of Offsite Power or Loss of Service Water is the initiating event. Failure of seal cooling due to loss of these systems is handled in the respective event trees for those initiators. This event tree handles loss of CC and loss of seal injection flow due to reasons other than Loss of Offsite Power and Loss of Service Water. Current analysis (Westinghouse, 1988c) indicates that seal failure is possible some time after loss of cooling, if no action is taken to cooldown and depressurize the RCS. For the purpose of the IPE it is assumed it will occur and the time of occurrence is in accordance with the model given in Appendix B.

At North Anna, loss of all CC (which is necessary for this initiator) will cause a two unit trip. Upon loss of all RCP seal cooling and loss of seal injection from the charging systems, the operator may cooldown and depressurize the RCS to minimize the possibility of seal failure; however, this action is not clearly indicated in the procedures. If seal failure occurs, HHSI is not available to mitigate the LOCA, as its failure is part of the initiator. Rapid cooldown is required to allow accumulators to inject and eventually enable intermittent use of the low head SI pumps to maintain RCS inventory. These pumps are not dependent on any form of cooling from the CC system.

The loss of RCP seal cooling event tree is shown in Figure 3.1-T4 and the function failures are discussed in the following paragraphs.

K - Reactor Subcritical

The system requirements for this function are the same as in the transient event tree. The reactor trip signal will result from a manual reactor trip signal or tripping of the RCP.

Hv - Emergency Switchgear Room Cooling Available

The system requirements for this function are the same as in the loss of feedwater transients event tree.

L - Auxiliary Feedwater Operable

Following this initiator, long term secondary heat removal is essential. One train of AFW is success in this function. Due to possible isolation of MFW during any potential seal LOCA, MFW was not credited for the long term. As the cooling failure causes charging pump failure, loss of secondary heat removal will result in core damage, as no charging flow is available for feed and bleed.

O - Operator Controlled Cooldown

The operator can cooldown and depressurize the RCS to prevent a seal LOCA. Inventory will be maintained by accumulator injection if the pressure is reduced low enough, otherwise the RCS will remain partially voided. If this is achieved in time, RC pump seal failure and core damage will be prevented. Success in this function is that the operator cools down in time to prevent a seal LOCA. Failure to do this implies that a seal LOCA occurs. Cooldown is achieved by use of auxiliary feedwater and Main Steam atmospheric dump valves.

Y - Core Cooling Recovery

If an RC pump seal LOCA occurs before the operator has achieved cooldown and depressurization, then the unavailability of charging pumps to maintain RCS inventory will lead to core uncovery and inadequate core cooling conditions (core exit temperature $\geq$ 1200°F) as pressure remains high. Success in this function is the same as that for the small LOCA, that is, the Main Steam atmospheric dump valves are used to perform a rapid cooldown allowing accumulator and low head SI pump injection.

D2 - Accumulator Injection
D3 - Low Head SI
H1 - Low Head Recirculation
QS - Quench Spray
Rs - Recirculation Spray
Ch - Containment Heat Removal
HI - LHSI/LHSR Late

The system success criteria for these functions are the same as those following small LOCA (Section 3.1.2.2.3). The loss of RC Pump seal cooling event tree is shown in Figure 3.1-T4.

3.1.3.3 Loss of Service Water (T6)

This initiator is defined as a total loss of Service Water and its event tree is shown in Figure 3.1-T6. This will result in a dual unit reactor trip, as Service Water is a shared system and as such will affect both units simultaneously. Loss of Service Water will cause a gradual heat up in the CC system. The RCPs will be tripped on high bearing temperature or high stator temperature within 30 minutes. Thus, the first effect of this initiator on plant systems will be loss of RCP seal injection flow or loss of RCP motor cooling, due to the loss of the charging system or loss of CC. The timing of the CC heat up is not known, but considered to be in the same time frame as loss of seal injection flow. Thus loss of Service Water is modeled to cause a Reactor trip in 30 minutes.

Loss of service water will gradually fail other systems. Recirculation sprays are unavailable should they be needed from the time of the initiator. Loss of Emergency Switchgear Room cooling is modeled to result in a loss of emergency power within 8 hours.

The event tree is shown in Figure 3.1-T6 and the event tree headings are discussed in the following paragraphs.

O - Operator Cooldown and Depressurize

In order to protect the RC Pump seals and allow time for the recovery of service water, it is necessary for the operator to cooldown and depressurize using the Auxiliary Feedwater System and rely on the accumulators for injection to the RCS. Without successful cooldown, core uncovery is assumed by 10 hours, in accordance with the RC Pump Seal LOCA model. Successful cooldown implies achievement of a stable state with secondary heat removal. Failure to prevent the seal LOCA will cause a faster path to core damage, as it is not possible to establish charging flow.

Lt - Auxiliary Feedwater After Loss of Emergency Power

Emergency Switchgear Room breakup is a slow process and does not occur in the early time frame. It is not until about 8 hours that loss of service water results in a loss of emergency power. After that time, only the turbine driven pump is available for AFW service. Considering the relative availabilities of the normal AFW system prior to loss of emergency power, and the turbine driven pump afterwards, only the turbine pump was asked. This question is principally asked to delineate the core damage sequences into plant damage states. If operator cooldown (O) succeeds but Auxiliary Feedwater (Lt) fails, core uncovery is assumed at about 10 hours (8 hours to loss of Emergency Power and 2 hours to core uncovery without AFW). If both operator cooldown (O) and AFW (Lt) are successful, core uncovery is estimated at about 22 hours (8 hours to loss of Emergency Power and 12 hours AFW run time and 2 hours to core uncovery without AFW), which is assumed to be 20 hours for analysis.

RC1 - Recovery of Service Water Prior to Core Uncovery and Vessel Failure at 10 Hours
RC2 - Recovery of Service Water Prior to Core Damage and Vessel Failure at 20 Hours
RC3 - Recovery of Service Water Prior to Containment Failure

These headings are asked to establish time windows for Service Water recovery and to further differentiate the plant damage states to allow for containment systems during core damage processes. The 10 hour and 20 hour windows are addressed in operator cooldown (O) and AFW (Lt) event discussions. Containment failure is assumed 18 hours after Reactor Vessel failure (Reactor Vessel failure is conservatively assumed at core damage). This yields the 30 hour and 40 hour windows based upon the 10 and 20 hour windows for core damage/Reactor Vessel failure.

D1 - HHSI/HHSR Late
QS - Quench Spray
Rs - Recirculation Spray
Ch - Containment Heat Removal
HI - LHSI/LHSR Late

These functions have the same requirements as in the loss of feedwater transients.

3.1.3.4 Steam Generator Tube Rupture (T7)

Although a Steam Generator Tube Rupture (SGTR) is similar to a small LOCA, a separate event tree (see Figure 3.1-T7) is developed because of the complex system and operator responses necessary for

successful prevention of core damage. One of the important considerations is the possibility of a LOCA bypassing the Containment if the faulted Steam Generator can not be isolated. On the other hand, the event can be limited to a transient if the primary to secondary leakage can be terminated by RCS cooldown and depressurization.

Based on experience at Ginna (NRC, 1982b) and North Anna, for a significant size of tube rupture the reactor will trip on low pressure and safety injection will be automatically initiated. At the same time, main feedwater pumps are tripped and auxiliary feedwater is initiated. If the leak rate is lower, the operator will diagnose the failure before a reactor trip occurs and start these systems manually.

The initial fall in pressurizer level will be restored and eventually the injection flow is stabilized to equal the break flow into the Steam Generator.

On the secondary side, automatic steam dump control is expected to be established and maintain the no load RCS temperature following reactor trip. However, the level, and possibly the pressure, in the affected Steam Generator will rise as break flow continues into the Steam Generator and eventually the SG PORV, and possibly SG safety valves, may be opened, particularly after isolation of the affected steam generator. The operator must isolate the affected SG to minimize the release of radiation. This includes the MS trip valves and SG blowdown valves, closure of the block valve if the SG PORV cannot be closed, and shutdown of the RCP in the affected loop. Once this is done, the RCS must be cooled down, maintaining subcooling (50°F). Cooldown and depressurization of the RCS continues until the RCS pressure is equal to that of the affected Steam Generator, thus terminating the leak. Finally, charging flow is terminated to prevent overfilling of the affected SG. Normal charging and letdown are established. The RCS is now in a quasi-stable state, and cooldown can continue until the RHR system is brought into service and cold shutdown conditions are established.

If termination of flow by depressurization and cooldown, from the RCS to the secondary side of the Steam Generator, is not established early on, then it is possible to overfill the Steam Generator and require the SG safety valves to pass water as well as steam, leading to the possibility of a valve sticking open. If this happens, it results in a LOCA bypassing containment and core damage can only be prevented if cooldown and depressurization to cold shut down condition (using RHR) is achieved prior to the emptying of the refueling water storage tank.

The headings for the SGTR event tree shown in Figure 3.1-T7 are discussed below:

K - Reactor Subcritical
Hv - Switchgear Room Cooling Available
D1 - High Pressure Injection
L - Auxiliary Feedwater
P - Bleed and Feed

The system requirements for these functions are the same as those following a small LOCA, as the break is equivalent to a small LOCA. The timing, however, will be different, and the number of SGs available for auxiliary feedwater injection is two not three.

SGI - Steam Generator Isolation

Outflow from the affected steam generator must be stopped to prevent a LOCA bypassing containment. Isolation is not achieved if:

1. Any safety or relief valve in the affected Steam Generator fails open.

2. The Main Steam trip valve on the affected Steam Generator fails to close, and Main Steam condenser dump valves are used for secondary heat removal.

3. Feedwater (AFW or MFW) is not isolated to the affected Steam Generator.

4. SG Blowdown trip valves fail to close.

5. Turbine driven AFW pump steam supply line from the affected Steam Generator is not isolated.

6. Common steam headers are not isolated.

If isolation is not achieved, then the only recourse to the operator is cooldown as discussed in the next function.

O - Operator Cooldown and Depressurize

This function and the previous one, isolation of the Steam Generator, are interrelated. If this action is not performed early, overfill of the Steam Generator will occur and it is assumed that this will also result in failure of the ability to achieve satisfactory isolation. The immediate impact of a Steam Generator Tube Rupture is similar to that of a small LOCA. Therefore, the operator is instructed to cooldown and depressurize. In this case the requirements are more specific. It is necessary to cooldown the RCS to a temperature of 50°F below the saturation temperature corresponding to the pressure until it is equal to that in the Steam Generator. These actions serve to terminate the loss of

coolant flow to the affected Steam Generator. If the pressure in the affected Steam Generator does not stabilize, then failure of isolation has occurred.

The systems which can be used for successful cooldown are Main Steam condenser dump valves or the intact Steam Generator's PORVs. The normal spray, auxiliary spray or pressurizer PORV can be used to depressurize.

O2 - Late Cooldown

Operator failure to achieve early cooldown leading to loss of isolation because of a failed open Steam Generator PORV/SRV, or operator failure to isolate the affected Steam Generator results in a continuous loss of RCS inventory through the affected Steam Generator. The operator can achieve cooldown using secondary heat removal and RHR in a similar manner to core cooling recovery in the case of a small LOCA. He will have to reach 212°F and atmospheric pressure before all injection is lost. This may require refill of the RWST, depending on how rapidly the operator recognizes the requirements and the achieved rate of cooldown.

D2 - Accumulators Injection
D3 - Low Head Safety Injection

In the event of failure of high pressure injection and failure of the operator to cooldown and depressurize the RCS, there is the possibility that the Steam Generator PORV/SRV will fail open. In this case, it will be necessary, as in the case of the small LOCA, to perform a rapid cooldown and depressurization through Main Steam condenser or atmospheric dump valves in order to allow the accumulators to inject, followed by low head safety injection and recirculation. This cooldown can continue until RHR entry conditions are reached. As the water loss is through the Steam Generator and not to containment, the environmental conditions in containment will not prevent operation of this RHR system.

W - RHR Cooling

In the event of failure to isolate the affected steam generator, all the water injected into the RCS from the RWST will be discharged into the atmosphere and will not be available in the sump for recirculation cooling. In this case, long term heat removal must be provided by one train of the RHR system.

H2 - High Head Recirculation
QS - Quench Spray
Rs - Recirculation Sprays Operable
Ch - Containment Heat Removal

The system requirements for each of these functions are the same as those in small LOCA event trees.

3.1.3.5 Loss of Emergency Switchgear Room Cooling (T8)

Emergency Switchgear Room cooling is required at North Anna under normal and accident conditions in order to prevent room heat up and eventual failure of the 4160 V and DC power supplies. Therefore the cooling of this room is modeled as an initiating event as well as in the event tree following each initiator. The loss of ESGR cooling event tree shown in Figure 3.1-T8 has been developed for this initiator. If Emergency Switchgear Room cooling is lost there is no immediate impact on the unit. Loss of ESGR cooling will cause a gradual heatup of the ESGR. At some temperature above 120°F, three detrimental effects have been identified. 1) The thermal overload design margin in the bus feed breakers will disappear, causing the breakers to spuriously open. 2) The inverters will sustain thermal damage. 3) Control Relays may develop spurious signals. For the purpose of this analysis, the event assumes the thermal overload margin disappears at 120°F. The feed breakers open causing loss of emergency power. In two more hours, battery depletion occurs.

Heat up to 120°F is modeled to occur in 8 hours. Prior to this time, all systems are nominally available. Reactor shutdown by manual trip is assumed to occur prior to reaching 120°F in the ESGR. AFW or MFW is required from the moment of shutdown. At the time loss of emergency power occurs, the turbine driven AFW pump must provide feedwater. If room cooling is restored, all sequences can be successfully mitigated.

Lt - Turbine Driven Pump Available

Feedwater is assumed to be available prior to loss of emergency power. All three AFW pumps and the main feedwater systems are nominally available at the time of loss of ESGR cooling. Their probability of failure in the ensuing 8 hours, compared to the unavailability of the turbine pump after loss of emergency power, is negligible. Thus only the turbine driven pump is asked. If the turbine driven pump is successful (Lt) then the time available for recovery of room cooling and restoration of ESF systems will be longer. If Lt fails, core uncovery is assumed at about 10 hours (8 hours to loss of Emergency Power and 2 hours to core uncovery without AFW). If Lt is successful, core uncovery is assumed at

about 20 hours, conditional upon operator cooldown and depressurize (O) discussed below.

O - Operator Cooldown and Depressurize

In order to protect the RC pump seals and allow time for recovery of ESGR cooling, it is necessary for the operator to cooldown and depressurize. Failure to prevent RC pump seal LOCA requires the additional inventory makeup from HHSI in the event ESGR cooling is restored before core uncovery. If both AFW (Lt) succeeds and operator cooldown (O) succeeds, core uncovery is estimated at about 22 hours (8 hours to loss of Emergency Power and 12 hours AFW run time and 2 hours to core uncovery without AFW), which is assumed to be 20 hours for analysis. If AFW (Lt) succeeds but operator cooldown (O) fails, core uncovery is estimated at about 18 hours (8 hours to loss of Emergency Power and core uncovery in another 10 hours, in accordance with the RC pump seal LOCA model), which is assumed to be 20 hours for analysis.

RC1 - ESGR Cooling Recovered Prior to Core Uncovery and Vessel Failure at 10 Hours
RC2 - ESGR Cooling Recovered Prior to Core Damage and Vessel Failure at 20 Hours
RC3 - ESGR Cooling Recovered Prior to Containment Failure

Failure to restore ESGR cooling leads to a loss of emergency power and core damage results with or without an RC seal LOCA. The partitioning of these sequences is further discussed in Section 4.3 when developing the plant damage states. The 10 hour and 20 hour windows are addressed in operator cooldown (O) and AFW (Lt) event discussions. Containment failure is assumed 18 hours after Reactor Vessel failure (Reactor Vessel failure is conservatively assumed at core damage). This yields the 30 hour and 40 hour windows based upon the 10 and 20 hour windows for core damage/Reactor Vessel failure.

D1 - HHSI/HHSR Late
QS - Quench Spray
Rs - Recirculation Spray
Ch - Containment Heat Removal
H1 - LHSI/LHSR Late

These headings are the same as for the S2 event tree.

3.1.3.6 Transients with Failure to Scram (ATWS)

Transients without reactor trip (scram) represent a unique type of event for core damage analysis. Very specific system and operator

responses are required in the short term (<30 minutes) following an ATWS. Conversely, an ATWS has no long term mitigation requirements. If subcriticality can not be restored in a relatively short period of time, ATWS events are assumed to result in core damage.

PWRs exhibit an inherent shutdown characteristic during a heatup, in that reactivity (hence reactor power) is reduced for a sustained coolant heatup. The magnitude of the negative reactivity inserted during heatup is a function of fuel enrichment, burnup, loading arrangement, burnable absorber design and exposure and chemical shim (boron) concentration. The limiting ATWS event is one which results in RCS pressure beyond the ASME Code Level C limit (that is, events which are the result of an adverse combination of the above, combined with degradation of the heat transfer capability between primary and secondary systems). The magnitude of the heatup, and therefore the pressure transient, is determined by the ability to maintain primary to secondary heat transfer, the primary pressure relief capacity, and the inherent shutdown characteristics of the core at the time of the transient. An ATWS event is the result of a transient generating the conditions for a reactor trip followed by failure to insert the rod cluster control assemblies. If this event occurs at low power, or if Main Feedwater is available, a less severe power mismatch between primary and secondary occurs and the peak pressure will be lower (Westinghouse, 1988a and 1988b). However, with the loss of Main Feedwater, a large imbalance in the heat source and heat sink will occur as a result of the rapidly falling level in the Steam Generator. The imbalance leads to a heat buildup in the primary system indicated by rising coolant temperature and pressurizer level. The rapid inflow of water into the pressurizer as the result of the expansion due to the heatup leads to a rising system pressure. As the Steam Generators empty, the primary to secondary heat transfer rapidly declines so the primary system temperature and pressure continues to rise. The pressurizer PORVs and safety valves lift to release the reactor coolant volumetric insurge.

Due to reactivity feedback conditions, the increase in RCS temperature causes core power to be reduced. This rate of reduction combined with available relief capacity will determine whether the peak RCS pressure exceeds the ASME Code Level C conditions. If this peak pressure is not exceeded during the early phase of the transient, then a quasi-stable condition can be established with the power, equivalent to the Auxiliary Feedwater/Main Feedwater flow to the Steam Generator, at an elevated RCS temperature.

There are several mechanisms which can then be used to achieve plant shutdown. The operator can deenergize the control rod drive mechanisms or attempt to drive the control rods in. Alternatively, boration of the RCS can be performed either through the CVCS or using RWST and charging pumps in the Safety Injection mode.

Two event trees are developed based on the system success criteria, the first for events occurring below 40 percent power and the second for events at or above 40 percent. These event trees are shown in Figures 3.1-TL and 3.1-TH for the low power and high power events respectively.

3.1.3.6.1 ATWS Event Tree for Initiators Below 40 Percent Power (TL)

Each of the functions appearing in the event tree in Figure 3.1-TL is discussed below:

K - Reactor Subcritical

Success in this event requires automatic control rod insertion following the generation of a Reactor Protection System reactor trip signal, or manual reactor trip by the operator early enough to limit the peak RCS pressure. If failure to trip is not the result of failure of the rods to mechanically insert or mechanical faults in the reactor trip breakers, then manual reactor trip from the control room within one minute is included in the modeling of function K. Success in this function leads to the normal transient condition modeled in the earlier event trees. The timing of one minute is based on achieving scram before peak pressure is reached in the RCS (WCAP-11993).

L - Auxiliary Feedwater

The Westinghouse (1988a) analysis shows that for transients where the power is less than 40%, minimum Auxiliary Feedwater flow is adequate to prevent RCS over-pressure (3200 psig): that is, one Auxiliary Feedwater pump, either motor driven or turbine driven, must deliver flow to one Steam Generator. For those transients which Main Feedwater remains available, credit will be taken for Main Feedwater flow.

Q - Reactor Coolant System Boundary Intact

It is considered that for low power ATWS, the pressurizer PORVs open at the same demand rate as for T2. There is the possibility of a breach of the RCS boundary if one fails to reclose. The specific impact of a failed open PORV is that the charging pump will be required to maintain RCS inventory. This will achieve the dual function of assuring subcriticality (by the addition of borated water from the BIT) and maintaining inventory. The other small LOCA success criteria are not applicable, as they require the cooldown of the RCS and would therefore add reactivity, increasing the primary to secondary power mismatch.

Since the frequency of the sequence representing failure to scram, followed by a failed open PORV and successful charging pump injection, is very much lower than its equivalent small LOCA sequence, the subsequent requirement for recirculation and containment heat removal have been omitted from the event tree.

MS1 - Manual Scram Late

This heading specifically models the operator action to remove power from the control rod drive mechanisms. This can be done by tripping the output or input breakers to the motor generator sets. This is an out of control room action and would reasonably take 1-2 minutes. Thus, its success would obviate the need for emergency boration, but not for Auxiliary Feedwater initiation, turbine trip, or primary pressure relief.

D4 - Emergency Boration

Long term shutdown can be achieved by the injection of borated water, either by using the charging pumps to inject through the normal charging header or the boron injection tank (BIT).

In the event of failure of a PORV to reclose, the requirement is determined by the consequential small LOCA, that is one charging pump is required in the SI mode. In this case, boration will occur naturally as part of the SI process.

The remaining functions, Qs, Rs, Ch and H1, are the same as for the transients.

3.1.3.6.2 ATWS for Initiators Above 40 Percent Power (TH)

Each of the functions appearing in the event tree in Figure 3.1-TH is discussed below.

M - Main Feedwater

For ATWS events where Main Feedwater is available, irrespective of power level, the peak RCS pressure will not exceed 3200 psig.

Tt - Turbine Trip

If Main Feedwater is lost and the reactor trip has failed, it is necessary for the turbine to trip in order to avoid the very rapid loss of primary to secondary heat transfer and hence over pressure in the RCS. Success in this function is tripping of the turbine within 1 minute. Failure to trip will occur if the reactor trip

breakers fail to open or other turbine trip signals do not occur. The AMSAC modifications required by the ATWS Rule (10CFR50.62) for tripping the turbine and starting the Auxiliary Feedwater pumps are installed and operational at North Anna Units 1 and 2.

L - Auxiliary Feedwater

For transients greater than 40% power the Westinghouse analysis shows that the minimum feedwater requirement is one turbine driven Auxiliary Feedwater pump or two motor driven Auxiliary Feedwater pumps to all three Steam Generators. However, because of the AFW alignment in North Anna (i.e., each pump dedicated to a specific generator), this analysis will assume that success is defined as two of three AFW pumps to two of three Steam Generators.

Pr - Pressure Relief

Pressure relief is required to prevent the RCS pressure exceeding 3200 psig in the event of a loss of Feedwater ATWS at greater than 40% power. The pressure relief requirements depend on the following factors in addition to the Auxiliary Feedwater flow:

- time in cycle life that transient occurs,
- reactivity feedback as a function of cycle length,
- pressurizer pressure relief and safety valve capacity,
- manual insertion of RCCA bank.

The combination of these factors is used to determine the proportion of time during the cycle life when pressure relief capacity will be insufficient to prevent pressure exceeding 3200 psig. This is defined as the Unfavorable Exposure Time (UET). It is necessary to perform a number of calculations in order to establish the value for event PR. The method for performing these calculations is described in WCAP-11993 (Westinghouse, 1988b).

Success in this function requires that sufficient pressure relief is available to prevent RCS pressure exceeding 3200 psig. For conditions resulting in favorable reactivity parameters (i.e., not UET), success criteria are three pressurizer SRVs or two SRVs and two pressurizer PORVs open.

Q - Reactor Coolant System Boundary Intact

In all cases Pressurizer PORVs will be challenged. Success in this function requires that all PORVs reclose or the block valve is closed for any PORV which fails open.

MS1 - Manual Scram Late

This heading specifically models the operator action to remove power from the control rod drive mechanisms. This can be done by tripping the output or input breakers to the motor generator sets. This is an out of control room action and would reasonably take 1-2 minutes. Thus, its success would obviate the need for emergency boration, but not for auxiliary, turbine trip, or primary pressure relief.

D4 - Emergency Boration

The requirement for this function is the same as in Section 3.1.3.6.1.

3.1.3.7 Interfacing System LOCA Event Tree (VX)

The maximum size of the Interfacing System LOCA is limited by the piping connected to the RCS, and the flow restrictors fitted to the piping. From the analysis of initiating events in Section 3.1.1, this was determined to be approximately the equivalent of six inch piping. A break in the piping outside Containment, and therefore no possibility will exist for recirculation when the RWST is empty. The RHR system in Containment will be unaffected by the LOCA.

In NUREG-1150, the frequency of an Interfacing System LOCA was determined to be 1.6E-6 per year based on expert elicitation. This will also be used for the North Anna analysis, based upon the similarities between North Anna and Surry for related piping and components.

The frequency of the Interfacing System LOCA is the frequency of core damage. An event tree was drawn for possible future development. The event tree is shown in Figure 3.1-VX.

3.1.3.8 Reactor Vessel Rupture (RX)

Any break in the RCS beyond the capacity of the low pressure injection/Accumulator systems is defined as Reactor Vessel rupture. There is no mitigation for a break beyond the design basis, so Reactor Vessel rupture is expected to lead directly to core damage. Therefore, this event tree consists of a single event as shown in Figure 3.1-RX.

3.1.3.9 Reactor Vessel Rupture and Pressurized Thermal Shock

An early PTS risk evaluation (Westinghouse, 1982) has identified that LOCAs, secondary side breaks and SGTR may lead to cooldown

transients at high RCS pressure, hence these initiators are the ones that must be addressed in terms of pressurized thermal shock (PTS). The NRC has also addressed this issue and identified a range of sequences which have the potential to result in PTS (NRC, 1982c). In addition to the initiating events, the key conditions which influence the likelihood of its occurrence are:

- Weld material composition, particularly copper,
- Accumulated fluence at each weld,
- Frequency and severity of overcooling transients.

The first two of these functions influence the reference temperature for nil ductility transition (RTNDT). As indicated in NUREG/CR-4550, the value established at Surry Unit 1 was 269°F. Those factors were considered in the evaluation of the potential for pressurized thermal shock leading to vessel rupture in the NUREG/CR-4550 study for Surry. It was concluded in that study that core damage due to PTS at Surry was minimal compared to core damage from other causes, therefore the same conclusions have been adopted in this study.

Surry is limited by the copper content in a specific weld. These North Anna vessels were constructed by a different manufacturer using low copper content weld flux. Therefore, North Anna has significantly more margin (i.e., a lower RTNDT) than Surry so core damage due to PTS is considered negligible for North Anna.

3.1.4 Sequence Grouping and Back End Interface

The interface between the Level 1 Systems Analysis and the Level 2 Containment Analysis consists of a set of plant damage state (PDS) groups. The plant damage states define a set of functional characteristics for system operation which are important to accident progression, Containment failure and source term definition. Each PDS contains Level 1 sequences with sufficient similarity in system functional characteristics that the Containment accident progression for all sequences in the group can be considered to be essentially the same. Each PDS defines a unique set of conditions regarding the state of the plant and Containment Systems and the physical state of the core, Primary Coolant System and the Containment boundary at the time of core damage/vessel failure. The important functional characteristics for each PDS were determined by defining the critical parameters (system functions) which impact these key results. The sequence characteristics which are important were defined by the requirements of the Containment accident progression analysis. They include the type of accident initiator, the operability/non-operability of important systems, the value of important state variables (e.g., Primary System pressure) which are defined by system operation, and timing of key events. The binning criteria

grouping parameters, plant damage state characteristics and binning results are fully described in Section 4.3.

The event trees developed in Section 3.1.2 are based on the success criteria to prevent the onset of core damage as defined in Section 3.1.1. They include all the functions associated with the operation of Containment systems (e.g., heat removal or isolation). Also for those sequences where core damage occurs as the result of failure of injection early in the event tree the performance of the Quench Spray and Recirculation Spray systems has been considered.

Since the Containment is operated at subatmospheric pressure the probability of Containment bypass as a result of failure to isolate is very low for all sequences. Hence this function has been excluded from individual event trees. The frequency of bypass used in evaluating Containment bypass for all sequences is that used for the Surry analysis in NUREG/CR-4551.

3.2 SYSTEM ANALYSIS

The North Anna IPE was performed using the "small event tree - large fault tree" approach, as discussed in NUREG/CR-2300 (NRC, 1983). In using this type of method, the fault tree analysis of the plant systems becomes the major underlying task of the Level I analysis.

In using this method, the event tree analysis identifies safety functions necessary for the mitigation and containment of initiators and accidents. The systems that perform these functions are known as "front-line" systems. Through a process of accident analysis with thermal hydraulic calculational support, success criteria are developed for the front line systems. Success criteria specify minimal combinations of equipment that must operate for the various safety functions to be supplied. These success criteria form the starting point for the fault tree analysis (also called systems analysis). The success criteria determine the top event for the front line system fault trees. The fault trees for the front line systems in turn identify requirements for support services, such as actuation, cooling, or AC power. These interfaces then determine the top events for the support systems.

The systems analysis for the North Anna IPE was conducted in accordance with Task Plan SM of this project. The task plan provides specific guidance for all facets of fault tree development, including specifying component boundaries, level of detail for modeling, modeling guidelines for restoration errors, and test and maintenance unavailability. The task procedure was developed and reviewed to ensure that it would meet all applicable requirements of NUREG-1335 (NRC, 1989) and NUREG/CR-2300 (NRC,

1983) for fault tree analysis. The task plan will not be discussed in this section.

This section provides a brief description of each system, the functions the system provides and the success criteria for the system. Detailed reporting of the systems analysis task is found in Appendix A of this report and will not generally be repeated here. Specifically, due to the size and volume of the dependency matrices and the fault tree assumptions, they will not be listed here.

3.2.1 Accumulator Model

The Accumulators provide an initial influx of borated water to reflood the reactor core following a large loss of coolant accident (LOCA) or a medium LOCA at the upper end of this LOCA size. The Accumulators are a front line safety system designed to provide core heat removal.

3.2.1.1 Accumulator Description

The Accumulator System consists of three tanks filled with borated water and pressurized with nitrogen. Each of the Accumulators is connected to one of the Reactor Coolant System (RCS) cold legs by a line containing a normally open motor operated valve and two check valves in series. The check valves serve as isolation valves during normal reactor operation and open to empty the contents of the accumulator when the RCS pressure falls below 650 psig.

The Accumulators are dependent on the Nitrogen System to maintain a head on the Accumulators. The nitrogen is supplied by a nitrogen supply header from bottles located outside of Containment. The Accumulators are fully instrumented to indicate an abnormal pressure condition. Due to the small fault exposure time of four hours, this dependency was not further developed.

The Accumulators are initially filled with borated water from the Refueling Water Storage Tank (RWST). The Accumulators are filled and the valves are closed. Instrumentation verifies that the level remains above a minimum value. Therefore, no dependencies were modeled between the Accumulators and the RWST.

3.2.1.2 Accumulator Logic Model

The success criteria for the Accumulators vary depending on the application in the event tree analysis. The success criterion for the Accumulators following a large LOCA, conservatively assuming a cold leg break, is injection of the contents of the two Accumulators associated with the intact cold legs into the RCS.

The success criterion for the Accumulators following a medium LOCA is injection of the contents of two or more Accumulators into the RCS. These success criteria are translated into the following top events associated with the large and medium LOCA size breaks, respectively:

- Failure of two of two Accumulators to inject their contents into the RCS.

- Failure of two of three Accumulators to inject their contents into the RCS.

- Failure of three of three Accumulators to inject into the cold legs.

3.2.2 Auxiliary Feedwater System Model

The Auxiliary Feedwater System (AFW) provides Feedwater to the Steam Generators (SG) to remove core heat from the Primary System after reactor trip. The AFW System is a front line safety system.

3.2.2.1 AFW System Description

The North Anna AFW System is a three train system, with two electric motor driven pumps and one steam-driven pump. The electric motor driven AFW Pumps have a capacity of 350 gpm, and the turbine-driven AFW Pump has a capacity of 700 gpm. Each pump draws a suction through an independent line from the 110,000 gallon emergency condensate storage tank (ECST). Additionally, a 300,000 gallon CST, the Service Water System and the firemain can be used as water supplies for the AFW Pumps. The normal AFW valve line-up has each AFW Pump discharging to one and only one SG. The valve alignment can be reconfigured to allow any pump to provide Auxiliary Feedwater flow to any or all of the three Steam Generators. However, this requires local manual action which is modeled in the IPE. Flow from each AFW pump to its generator is through a series of diverse valves. The TDP discharges through a normally open MOV. One motor driven pump discharges through an AOV and an HCV, while the other motor driven pump discharges through an AOV and an MOV. All AFW lines discharge through a check valve to a line which joins the Main Feedwater line to a Steam Generator.

All pumps automatically start on receipt of a Safety Injection actuation signal, trip of Main Feedwater pumps, low Steam Generator level in any Steam Generator, or loss of offsite power.

The AFW System is dependent on the AC power buses for motive power to the AFW motor driven pumps, and motive and control power to the AFW MOVs. The AFW System is also dependent on the DC power buses for control power to the electric motor driven AFW Pumps, and the

Solid State Protection System (SSPS, SI actuation system) for actuation of the AFW Pumps. The turbine-driven pump turbine inlet valves require Instrument Air and DC power for control, however, on loss of either Instrument Air or DC power the valves fail open allowing steam flow to the pump turbine. Hence, no dependencies were modeled in these cases to represent system success.

3.2.2.2 AFW System Logic Model

The success criteria for the North Anna AFW System vary depending on the application in the event tree analysis. The majority of initiating events require AFW flow from any one AFW Pump to its Steam Generator for successful secondary heat removal. The success criterion for AFW during an ATWS event is flow from any two AFW pumps to two steam generators. Success criterion for AFW flow during core cooling recovery is the same as for ATWS. The success criterion for AFW following a SGTR is flow from the TDP or the B motor driven pump to an associated steam generator. The C generator is assumed to be the affected generator.

House events were used to model conditional criteria other than the success criteria. There were thirteen AFW functions in all. The principal success criteria described above translate into the following top events for the AFW fault trees:

- Insufficient flow to at least one of three Steam Generators from at least one AFW pump

- Insufficient flow to two Steam Generators from 2 AFW Pumps

- Insufficient flow to at least one of the two intact Steam Generators from at least one AFW Pump

3.2.3 Chemical Volume and Control System (CVCS)

The CVCS at North Anna provides many functions. The functions of interest to the PRA are Reactor Coolant Pump seal injection flow, emergency boration, and auxiliary pressurizer spray. The Charging System was modeled as a front line system in the PRA.

3.2.3.1 CVCS Description

Under normal operating conditions, one of three Charging pumps provides normal RCS makeup and seal injection flow to the Reactor Coolant Pump seals. The Charging System operates in a closed cycle, taking suction from the Volume Control Tank (VCT), which in turn is replenished from the letdown flow. The normal charging flow is split between the charging flow and the seal injection

flow. Part of the seal injection flow goes through the seals and into the RCS, while part is recirculated through the seal return line.

The auxiliary spray function from the CVCS is intended for use when the normal pressurizer spray from the RCP's is unavailable. Auxiliary spray is activated by opening an air operated valve to allow cold charging flow to enter the pressurizer spray line.

In the emergency boration mode, the Charging pumps function just as in the charging mode, with the exception that the boric acid transfer pumps deliver concentrated boric acid to the Charging Pump suction from the boric acid transfer tanks. To perform this operation, the operator must switch the normally operating BAT pump to fast speed and open an MOV allowing the boric acid to flow to the Charging Pump suction.

3.2.3.2 CVCS Logic Model

The CVCS was modeled as a front line system. It supplies functions for emergency boration when the pressurizer PORV's reclose and when one or more pressurizer PORV's stick open. It provides a contributing function for RCS depressurization and a separate function for RCP seal cooling. The functions are described below:

- Failure to provide emergency boration via 1/2 Charging Pumps taking suction from the VCT with the BAT pump taking suction from the BAT tank.
- Failure to provide emergency boration via SI flow from the RWST for ATWS sequences with failure of pressurizer relief valves to reclose.
- Auxiliary sprays can be provided from 1/2 Charging Pumps injecting through the auxiliary spray valve, in the event that normal spray is unavailable.

3.2.4 Component Cooling Water System Model

The Component Cooling Water (CC) System is a closed cycle cooling system which provides cooling to many systems including the Residual Heat Removal (RHR) and the Reactor Coolant System (RCS). The CC System, as defined for this analysis, includes only that portion of the CC System required to provide cooling water to the Reactor Coolant Pump (RCP) thermal barriers and the Residual Heat Removal System. RCP seal cooling is a front line function, while RHR heat exchanger and RHR pump cooling is a support function.

3.2.4.1 CC System Description

The CC System at the North Anna station is a shared system which provides CC to both units. The system consists of four CC pumps in parallel and four CC heat exchangers. The system is normally configured into two trains, each supplying a unit. The CC System is a closed cycle system. The CC pumps take suction from the return line header, which includes heat loads from the RCS pump thermal barriers, RHR pumps, and RHR heat exchangers. The pumps, arranged in parallel, discharge to a header that feeds the two CC heat exchangers also arranged in parallel. The discharge of the heat exchangers is delivered to two supply headers, which in turn feed the RCS pump thermal barriers, RHR pumps, and RHR heat exchangers. After cooling these loads, the flow is returned to the CC pump suction headers. Makeup to the CC System is provided from a surge tank in the system.

During reactor operation, one CC pump and one heat exchanger per unit are normally in operation. In the event of failure of the heat exchanger, the parallel component is manually placed in service. The containment isolation valve on the thermal barrier cooling water outlet closes on loss of Instrument Air or receipt of a Containment Depressurization Actuation (CDA) signal, resulting in loss of flow to the thermal barriers.

The CC System is dependent on the AC power buses for motive power for the CC pumps and the DC power buses for control power to the CW pumps and the thermal barrier throttle valves. Also, the CC System is dependent on the Instrument Air System for motive power to the thermal barrier throttle valves.

3.2.4.2 CC System Logic Model

The general success criterion for the CC System is that continued CC flow is provided to the RCS pump thermal barriers, Residual Heat Removal pumps, and Residual Heat Removal heat exchangers following reactor shutdown.

This success criterion translates into the following top events in the CC System fault trees:

- Failure to provide flow from 1/3 charging pumps or 1/2 CC pumps to all RCS pump thermal barrier coolers.

- Failure to provide CC flow to Unit 1 RCS pump thermal barrier heat exchangers during station blackout using CC System from Unit 2.

CC is also used as a support system for the Residual Heat Removal heat exchangers.

3.2.5 Containment Depressurization Actuation Signal

The Containment Depressurization Actuation (CDA) automatically actuates the Containment Safeguards Systems following receipt of indicated high (28 psia) Containment pressure. The CDA is a support system for the Quench Spray, Inside Recirculation Spray and Outside Recirculation Spray.

3.2.5.1 CDA Description

The North Anna CDA is composed of four Containment pressure sensors, each feeding a signal comparator. The output of each signal comparator is input into two separate two out of four logic trains. These logic trains automatically actuate the Containment Safeguards System components.

The CDA is dependent on the vital instrumentation buses and the DC buses for operation of the primary sensors and the relay logic network.

3.2.5.2 CDA Logic Model

CDA is a support system, and as such it does not appear directly on the event trees. There are two major output signals in the CDA model (train A and train B). Each output signal activates several other relays, which in turn activate the appropriate front line systems. Several top events for the CDA were modeled, each one included failure of the output signal and the output relay. These top events were linked directly into the front line system fault trees.

3.2.6 Containment Isolation

A Containment Isolation function was identified for the Level II analysis. Containment Isolation was not included in any of the plant damage event trees as it was determined not to have an effect on any of the accident sequences. It was also determined that traditional fault tree analysis was not appropriate for this function. A reliability analysis of Containment Isolation was performed and the results were used directly in the Level II analysis.

3.2.7 Emergency Diesel Generator

Each North Anna unit has two diesel generators which are dedicated to that unit. Each diesel feeds one of the 4160 V emergency busses. Cross tie of the diesel generators to another bus or

another unit is not possible. Cross tie of certain busses can be accomplished under some conditions.

3.2.7.1 Diesel Description

Each diesel generator is a self-contained 3000 kW continuous rating generating unit. The diesels are self-cooled (water cooled with water-air radiators), are provided with self contained starting air system, batteries, take suction directly on outside air, and are each provided with a separate day tank. The EDG batteries provide control power and generator field flashing. There are two underground fuel tanks, each with four fuel oil transfer pumps, one dedicated to each diesel. The only dependency with other plant systems is that DC power required to close the diesel output breaker is supplied from the station vital batteries. However, in the event that this battery is unavailable, the diesel can be loaded onto the emergency buses manually with spring loaded closing mechanisms.

3.2.7.2 Diesel Modeling

Each Emergency Diesel Generator was modeled as a separate system to provide alternate AC power to its 4160 volt emergency bus. The diesel generators were used as a front line system to define station blackout.

3.2.8 Electrical Power Distribution System

The Electrical Power Distribution System is divided into two parts, the Emergency Electrical (EE) Distribution System and the Normal Electrical Power (EP) Distribution System. The Emergency Electrical Power Distribution System provides AC and DC power to safety related components. The normal electrical power system provides AC power to non safety related components. The Normal Electrical Power System receives its electricity from off site transmission lines or from the Unit 1 or 2 main electrical power generators. The Emergency Electrical Power System receives its electricity from the Normal Electrical Power System or from emergency diesel generators.

3.2.8.1 Emergency Electrical Power System

The Emergency Electrical Power at Unit 1 consists of two 4160 V buses, four 480 V buses, eleven 480 V motor control centers, four 120 VAC vital instrumentation buses, four 125 VDC buses, two dedicated diesel generators, and their associated breakers, transformers, uninterruptible power supplies, and batteries. The EE System at Unit 2 is symmetric to Unit 1.

The following description applies to the EE System at Unit 1. Since the EE System is symmetrical to Unit 2, the description is equally applicable with the appropriate change of designator (2H for 1H, 2J for 1J). Each 4160 V bus is normally powered from the switchyard. Upon loss of offsite power the supply breakers open, the diesel generators start and their associated EDG output breakers close to load the diesels on the emergency buses. North Anna Power Station has four emergency diesel generators, one dedicated to each 4160 V emergency bus. Each diesel is a self contained, self cooled unit with its own battery for starting power. The diesel battery is independent of the station batteries. The 4160 V buses provide power to the large pumps such as the HHSI, LHSI, and AFW pumps. The 4160 V stub buses each power one CC and Residual Heat Removal pump. The stub bus is shed on undervoltage on the main bus and must be reloaded by the operator.

The 1H 4160 V bus feeds two 480 V buses (1H and 1H1) through separate transformers. The 1H1 480 V bus is primarily used to power small pumps and an MCC. The 1H 480 V bus feeds five motor control centers (MCCs). MCC 1H1 provides power to three uninterruptible power supplies used to charge the DC batteries, and power to the 1-I and 1-II 120 VAC vital instrumentation buses. The 1J buses have a similar arrangement.

The 125 VDC bus provides control power to the switchgear for the pumps powered from the 4160 V buses. The 125 VDC buses are powered from a 480 V bus, as noted above, and in the event of loss of the AC power source, is powered from DC battery.

3.2.8.2 EE System Logic Model

The EE System is a support system that interfaces with almost all front line safety systems and support systems. The events identified for the EE System represent the modeled interfaces of the EE System with the system requiring electrical power. These interfaces were modeled to the Motor Control Center level. The following buses and MCCs are included in the EE fault trees:

4160 V Bus 1H
4160 V Bus 1J
4160 V Stub Bus 1H
4160 V Stub Bus 1J
480 V Bus 1H
480 V Bus 1H1
480 V MCC 1H1-1
480 V MCC 1H1-2N
480 V MCC 1H1-2S
480 V MCC 1H1-3
480 V MCC 1H1-3A
480 V MCC 1H1-4
480 V Bus 1J

480 V Bus 1J1
480 V MCC 1J1-1
480 V MCC 1J1-2N
480 V MCC 1J1-2S
480 V MCC 1J1-3
120 VAC Vital Instrumentation Bus 1-I
120 VAC Vital Instrumentation Bus 1-II
120 VAC Vital Instrumentation Bus 1-III
120 VAC Vital Instrumentation Bus 1-IV
125 VDC Bus 1-I
125 VDC Bus 1-II
125 VDC Bus 1-III
125 VDC Bus 1-IV
120 VAC Semi-Vital Bus 1A
120 VAC Semi-Vital Bus 1B
120 VAC Semi-Vital Bus 1C

3.2.8.3 Normal Electrical Power System

The Normal Electrical Power System consists of switchyard buses, transfer buses, station service buses and intake structure buses. The switchyard buses (500 and 34.5 kV) receive power from the offsite transmission lines and the Unit 1 and 2 main electrical power generators. Three transfer buses connect the switchyard to the emergency buses and to the station service buses. Each unit has three station service 4160 V buses which supply electrical distribution to 480 V and 120 VAC buses. The station service buses receive power from the switchyard through the transfer buses or through the line connecting the main generator to the switchyard. The CW intake structure has two 4160 V buses which supply power to 480 V then to 120 VAC buses.

3.2.8.4 Normal Electric Power System Logic Model

The key components which receive electrical power from the Normal Electrical Power System are large 4160 V pumps such as the Reactor Coolant (RC), Main Feedwater (MFW), Condensate (CN) and Circulating Water (CW) pumps. The major 4160 V and 480 V buses were included in the fault tree models. The unavailability of buses not included in the model can be approximated by selecting a bus which corresponds to the desired bus.

The Normal Electrical Power Supply System was included in the North Anna PRA to show the support system dependencies it provides to the Emergency Electrical Power Supply. The EP system model also allows studying the reliability of non Technical Specification required equipment.

The following buses are included in the Normal Electric Power System fault trees:

500 kV Switchyard Bus #1
500 kV Switchyard Bus #2
34.5 kV Switchyard Bus #3
34.5 kV Switchyard Bus #4
4160 V Transfer Bus 1D
4160 V Transfer Bus 1E
4160 V Transfer Bus 1F
4160 V Intake Bus 1G
4160 V Intake Bus 2G
4160 V Station Service Bus 1A
4160 V Station Service Bus 1B
4160 V Station Service Bus 1C
4160 V Station Service Bus 2A
4160 V Station Service Bus 2B
4160 V Station Service Bus 2C
480 V Station Service Bus 1A1
480 V Station Service Bus 1A2
480 V Station Service Bus 1A3
480 V Station Service Bus 1B1
480 V Station Service Bus 1B2
480 V Station Service Bus 1B3
480 V Station Service Bus 1C1
480 V Station Service Bus 1C2
480 V Station Service Bus 2A1
480 V Station Service Bus 2A2
480 V Station Service Bus 2B1
480 V Station Service Bus 2B2
480 V Station Service Bus 2C1
480 V Station Service Bus 2C2
480 V Intake MCC 1G1-1
480 V Intake Bus 1G2
480 V Intake Bus 1G3
480 V Intake MCC 2G1-1
480 V Intake Bus 2G2

3.2.9 Emergency Switchgear Room Cooling

Emergency Switchgear Room cooling is necessary in order to maintain the temperatures in the Emergency Switchgear Room below acceptable limits. Acceptable room temperature limits have been defined for this project as 120°F. At temperatures above these equipment qualification limits, electrical equipment aging is enhanced. Three other temperature related affects are that relays may spuriously transfer, the thermal overload margin in the load breakers may be lost causing tripping of the load breakers, and the solid state components in the inverters may become inoperable. The temperatures at which each of these effects will occur was not specifically determined. Heat up times for combinations of heat loads and cooling equipment availability made for licensing purposes were available for use in the IPE. Heat up times were compared with the 24 hour mission times used in the IPE and success

criteria for the ventilation function were developed based on the above equipment qualification temperature limit.

3.2.9.1 ESGR Cooling System Description

ESGR cooling is provided at each unit by a dedicated system of air handling units and chillers. The chillers are a closed cycle loop, consisting of three pumps and three chillers per unit. Each chiller Condenser is cooled by an open system using Service Water from the SW supply headers. Each chiller has a dedicated pump which takes suction from either SW supply header. The chilled water is circulated around a closed cycle system which contains four ESGR air handling units, two for the ESGR and two for the control room. In the event that the Chilled Water System at one unit fails the ESGR from the other unit can be utilized to provide cooling by opening the doors between ESGRs and placing portable fans in strategic locations.

3.2.9.2 ESGR Logic Model

The ESGR cooling function was modeled on the event trees as a front line safety function. The effect of loss of ESGR cooling is universal as it will eventually lead to loss of all safety systems. There is one success criterion for the HV function:

- Failure to provide ESGR cooling using 1/3 chiller units and 1/2 air handling units

Loss of ESGR cooling was also modeled as an initiating event. Loss of ESGR cooling can lead to reactor trip because the charging pumps fail, which leads to a seal LOCA.

3.2.10 High Head Safety Injection (HHSI) System

At North Anna, the Charging System provides normal coolant makeup to the Reactor Coolant System (RCS) and cooling flow to the Reactor Coolant Pump (RCP) seals under normal operating conditions. The Safety Injection System uses the same Charging Pumps to provide primary coolant injection and recirculation following an accident, as well as maintaining flow to the RCP seals. The Charging System also functions to deliver boric acid to the RCS from the Boric Acid Transfer System if emergency boration is required.

3.2.10.1 HHSI System Description

Under normal operating conditions, one of the three Charging Pumps provides normal RCS makeup and cooling to the RCP seals by taking

suction from the Volume Control Tank (VCT) through two motor operated valves (MOVs) in series.

Upon indication of a loss of RCS coolant or steam line break (i.e., low pressurizer level, high Containment pressure, high pressure differential between Main Steam header and any steam line, or high steam flow with low average coolant temperature or low steam line pressure), the Safety Injection Actuation System, Solid State Protection System (SSPS), initiates emergency coolant injection. Emergency coolant injection differs from normal coolant makeup in three ways. First, the suction source is the Refueling Water Storage Tank (RWST) rather than the Volume Control Tank (VCT). Second, the pump discharge is directed to the cold legs instead of the Loop 2 hot leg. Finally, the emergency injection flow is from two pumps. The SSPS signals the charging line isolation valves to close, the standby Charging Pumps to start, the valves from the VCT to close, the boron injection tank isolation valves to open, and the suction valves from the RWST to open. An additional path to the RCS cold legs through a manually operated, normally closed MOV is also available. The line to the RCP seals remains open throughout the event.

In the recirculation mode of operation, the HHSI is used to provide core heat removal late in an accident sequence. The Charging Pumps draw suction from the discharge of the Low Head Safety Injection pumps in the pump recirculation mode. After the RWST level decreases below 29%, the Control Room Operator realigns SI to the recirculation mode by opening the LHSI supply valves from the Containment sump, the Charging Pump supply valves from LHSI, then closes the RWST supply valves to the LHSI and Charging Pumps.

The HHSI System interfaces with the Quench Spray System and the Low Pressure Injection System at the common RWST. In the recirculation mode, the system interfaces with the Low Head Safety Injection System at the recirculation suction valves for the Charging Pumps. The HHSI is dependent on the RWST for fluid inventory, and the Service Water System for Charging Pump seal cooling and lube oil cooling. The HHSI System is also dependent on the AC power buses for motive power to the Charging Pumps and motive and control power to the MOVs, the DC power buses for control power to the pumps, and the SIAS for actuation. In the recirculation mode, the system is dependent on the Low Head Safety Injection System for fluid inventory, and the SIAS for actuation of the switchover from injection.

Technical Specifications require two Charging Pumps to be operable at all times.

The HHSI System is limited to the simultaneous operation of two of the three Charging Pumps. Further, the two operable pumps must be powered from different 4160 V buses. The third Charging Pump is placed in the "auto-after-stop" position. In this position, the

pump remains aligned to an AC bus. If an SI signal occurs, the pump will automatically start.

3.2.10.2 HHSI System Logic Model

The success criteria for the High Head pumps generally require flow from any one of three Charging Pumps to the RCS cold legs in response to a LOCA (automatic actuation), or in the "feed and bleed" mode with manual actuation.

These success criteria translate into the following top events for these functions:

- Failure to provide high pressure flow to the cold legs from at least one Charging Pump, taking suction on the RWST.

- Failure to provide flow to the RCS cold legs from at least one Charging Pump in the recirculation mode, taking suction from either LHSI pump discharge.

3.2.11 Instrument Air and Compressed Air

Compressed air is required at North Anna to operate the pressurizer PORV's, the Steam Generator PORV's, the Turbine-Driven AFW Pump Steam Admission Valves, Condenser Steam Dump Valves and the main and auxiliary pressurizer spray valves. Some of these valves have important functions, while some of them have very limited importance. The more important valves have a limited need for Instrument Air. For example, the pressurizer PORV's have backup nitrogen supplies, while the turbine-driven supply valve fails open on loss of air.

3.2.11.1 Instrument Air and Compressed Air Description

The Containment Instrument Air subsystem can operate independently of the other Instrument Air subsystems. In practice, the Instrument Air compressors are used to supply both the containment instrument air and plant instrument air requirements.

3.2.11.2 Instrument Air and Compressed Air Logic Model

Fault tree analysis was not performed on the Instrument Air System. Fault tree analysis of the front line systems identified the importance of Instrument Air for each load, based on backup supplies or alternate methods of valve operation. The results of the fault tree analysis on all other systems indicated that

Instrument Air could be modeled as a basic event for those conditions where it was important.

The containment instrument air dependencies in the PRA model include primarily the pressurizer PORVs and the RHR flow control valves. The PORVs can be operated with the backup nitrogen bottles if air supply fails. The RHR flow control valves fail safe. Therefore, the components supplied through the containment instrument air subsystem are less vulnerable to failure than those components supplied through the instrument air subsystem. As a result, separate basic events were used for the system, even though they are no longer operated independently of each other.

3.2.12 Low Head Safety Injection

The Low Head Safety Injection System provides emergency coolant injection and recirculation following a loss of coolant accident when the Reactor Coolant System (RCS) depressurizes below 175 psia. In addition to the direct recirculation of coolant during the recirculation phase, once the RCS is depressurized, the LHSI discharge provides the suction source for the HHSI following drainage of the Refueling Water Storage Tank (RWST).

3.2.12.1 LHSI Description

The North Anna LHSI System is composed of two 100% capacity pump trains. In the injection mode, the pump trains share a common suction header from the RWST. Each pump draws suction from the header through a normally open motor operated valve (MOV), and locked open manual valve in series. Each pump discharges through a check valve and normally open MOV in series to a common injection header. The injection header contains two normally open MOVs in parallel and branches to three separate lines, one to each cold leg. Each of the lines to the cold legs contain two check valves in series to provide isolation from the high pressure RCS. The LHSI System is not cooled by any other system.

In the recirculation mode, the pump trains draw suction from the Containment sump through a parallel arrangement of suction lines to a common header. Flow from the suction header is drawn through a normally closed MOV. Discharge of the pumps is directed to either the cold legs through the same lines used for injection or to a parallel set of headers which feed the Charging Pumps, depending on the RCS pressure.

In the hot leg injection mode, system operation is identical to cold leg injection with the exception that the normally open cold leg injection valve must be closed and one or more normally closed hot leg injection valves must be opened.

Upon indication of a loss of RCS coolant or a main steam line break (i.e., low pressurizer level, high Containment pressure, high pressure differential between Main Steam header and any steam line or high steam flow with low average coolant temperature or low steam line pressure), the Solid Station Protection System (SSPS, SI actuation system) initiates LHSI operation. The SSPS signals the low pressure pumps to start. All valves are normally aligned to their injection position. If the Reactor Coolant System pressure remains above the pump shutoff head, the pumps will discharge to the RWST through two normally open minimum flow recirculation lines until the RCS pressure is sufficiently reduced to allow inflow.

Upon receipt of a low RWST level signal, the SIAS signals the low pressure pump suction valves from the RWST and the valves in the minimum flow recirculation lines to the RWST to close, and the suction valves from the Containment sump to open.

At approximately 10 hours following the initiation of a large LOCA, the emergency procedures call for switchover from cold leg recirculation to hot leg recirculation. The operator must open valves 1-SI-MOV-1890A and 1890B, and close 1-SI-MOV-1864A and 1864B.

The LHSI System interfaces with the Quench Spray System and the High Pressure Injection System at the common RWST. The LHSI System is dependent on the RWST for fluid inventory and the SIAS for actuation of components. The LHSI System interfaces with the high pressure Recirculation System at the recirculation suction valves for the HHSI. The LHSI System depends on the AC power buses for motive power to the pumps, and motive and control power to the MOVs, and the DC power buses for control power to the pumps.

3.2.12.2 LHSI Logic Model

The success criteria for the LHSI vary depending on the application in the event tree analysis. The general success criterion for the LHSI is flow from one or more low pressure pumps to the RCS cold legs in response to a loss of primary coolant inventory.

- Insufficient flow from at least one low pressure pump to the cold legs from the RWST.

The success criteria for the recirculation modes of operation are continued flow from either of the two low head pumps to the cold legs and switchover to hot leg recirculation at 10 hours or sufficient flow from either of the two low pressure pumps to the Charging Pump suction header.

These success criteria translate into the following top events in the LHSI fault trees:

- Failure to switch to hot leg recirculation at 10 hours following a large LOCA.
- Insufficient flow from at least one low head pump to the cold legs from the Containment sump.
- Insufficient flow from at least one low head pump to the Charging pump suction header from the Containment sump.

3.2.13 Main Feedwater System

3.2.13.1 Description

The Main Feedwater System consists of the Main Feedwater pumps, the condensate pumps and the hotwell inventory. Because North Anna has electric driven MFW pumps, it is possible to supply Feedwater using the MFW System, without having the turbine bypass and Steam Condensing Systems available. The inventory of the hotwell (with the condensate storage tank as a backup supply) was calculated to be sufficient for all mission times of interest. The Feedwater regulating valves will close after a reactor scram, due to plant control logic. However, the Feedwater pumps remain on, unless an SI signal occurs, and the miniflow valves open. Feedwater can then be provided to the SGs, through the Feedwater regulating bypass valve.

The MFW System is dependent on DC power and Instrument Air.

3.2.13.2 MFW Logic Model

The success criteria for the MFW System was:

- Failure of at least one Main Feedwater pump to provide flow to at least one Steam Generator.

3.2.14 Main Steam System

The Main Steam System was modeled for the capability to depressurize and cooldown in certain sequences, as well as to provide Steam Generator isolation.

3.2.14.1 Main Steam System Description

The portion of the Main Steam System that was analyzed for the IPE consisted of two steam flowpaths: steam dumped to the main Condensers and steam dumped to the atmosphere. During RCS cooldown below 543°F, one or two of the condenser steam dump valves can be

used depending on the demand signal. Above 543°F, all eight condenser steam dump valves can be used for RCS cooldown. The atmospheric cooldown path uses the SG atmospheric steam dump, if it is available.

The SG Relief System is composed of the code safety relief valves and one power operated relief valve (PORV) for each Steam Generator. The SG relief valves were modeled for core cooling recovery, station blackout and Steam Generator tube rupture transients. The PORVs provide SG pressure relief at a set point below the SRVs. Each PORV is provided with a manually operated block valve which is normally open unless a PORV is leaking. The PORVs automatically open on high SG pressure or are manually opened at the direction of the operator. All of the relief valves are upstream of the Main Steam trip valves and all discharge directly to atmosphere outside of the Containment.

The following paths provide a potential loss of SG integrity should they fail to isolate: a) steam can flow from the Steam Generator through a manual isolation valve and check valve to the steam supply for the AFW Turbine-Driven Pump, b) flow from the SG blowdown line through the blowdown coolers to the Blowdown Treatment System, and c) failure to close the Main Steam Trip Valves (MSTV).

3.2.14.2 Main Steam Logic Model

The Main Steam fault trees were used directly in or to support four basic frontline functions. A total of seventeen functional equations were generated to account for conditional equipment availabilities for certain sequences. The four basic success functions are:

- Failure to cooldown after a small LOCA or SGTR, by providing heat removal from one Steam Generator to the atmosphere or to the Condenser.

- Failure to cooldown to atmospheric conditions in 10 hours after a SGTR given a failure to cooldown and depressurize.

- Failure to isolate the ruptured Steam Generator after a tube rupture initiator.

- Failure to support core cooling recovery by heat removal from 2/2 Steam Generators to the atmosphere through 2/2 Steam Generators PORV's.

3.2.15 Quench Spray System Model

The Quench Spray System (QS) provides Containment pressure reduction following an accident by spraying cool water from the Refueling Water Storage Tank (RWST) to condense steam in the Containment. The QS is a front line system designed to protect the Containment. In addition, the QS performs a support function for the Inside Recirculation Spray System.

3.2.15.1 QS Description

The North Anna QS is composed of two 100% capacity spray injection trains. The QS has no recirculation or sump cooling capability. Each spray train draws water from the Refueling Water Storage Tank through independent suction lines. Each QS pump takes suction through a normally open motor operated valve (MOV) and discharges through a normally closed MOV to its associated Quench Spray header.

The QS automatically starts on receipt of a CDA (28 psia) signal. The CDA signals open the pump outlet valves and start the QS pumps.

The QS interfaces with the Safety Injection System at the common Refueling Water Storage Tank. The QS is dependent on the RWST for fluid inventory. The QS System also depends on the AC power buses for motive power to the QS pumps and motive and control power to the MOVs, the DC power buses for control power to the QS pumps, and the CDA for actuation of the QS components.

3.2.15.2 QS Logic Model

The success criterion for the North Anna QS is the same for each application in the event tree analysis. The success criterion is one of the two QS trains provide flow to its Quench Spray header. This translates into the following top event in the QS fault tree:

- Insufficient flow from 1 of 2 QS pumps to 1 of 2 spray headers.

3.2.16 Reactor Coolant System

The important functions of the Reactor Coolant System modeled in the IPE involve the Primary Pressure Relief System. This system also supports core heat removal during "feed and bleed" cooling.

3.2.16.1 RCS Pressure Relief System Description

RCS Pressure Relief at North Anna is composed of three code safety relief valves (SRV) and two power operated relief valves (PORVs). The code safety valves are important only in the anticipated transient without scram (ATWS) analysis. The PORVs provide RCS pressure relief at a set point below the SRVs. The PORVs discharge to the pressurizer relief tank. Each PORV is provided with a motor operated block valve.

The PORVs automatically open on high RCS pressure or are manually opened at the discretion of the operator. The block valves are normally open unless a PORV is leaking.

The PPRS is dependent on the AC power buses for motive and control power to the PORV block valves, DC power for control power to the PORVs, and the Containment Air System for motive power to the PORVs. However, the PORVs are provided with air bottles sized to provide approximately 80 openings of each valve. Therefore, no dependencies on the Containment Air System were included in the system models.

3.2.16.2 RCS Pressure Relief Logic Model

The success criteria for the PPRS vary depending on the application in the event tree analysis. The success criterion for the PPRS following a transient event demanding PORV opening is that the PORVs successfully reclose.

These success criteria translate into the following top events:

- One or more PORVs fail to reclose following a transient.
- Failure to provide adequate pressure relief during an ATWS event.
- Failure of 1 of 2 PORVs to open to support feed and bleed.

3.2.17 Reactor Protection System

The Reactor Protection System (RPS) is designed to automatically scram the reactor following receipt of indications of abnormal conditions.

3.2.17.1 Reactor Protection System Description

The RPS is an actuation system that receives signals from several different types of sensors. The sensor signals are combined in

various logic matrices which function to trip the control rod drive mechanisms' supply circuit breakers, (the scram breakers). In addition to the sensor signals automatically tripping the scram breakers, a circuit is installed that allows the scram breakers to be manually tripped from the Control Room. The RPS System is dependent on the vital AC instrumentation and DC buses for power to the sensors and logic network.

3.2.17.2 Reactor Protection Logic Model

The success criterion for the RPS requires insertion of sufficient number of control rods to make the reactor subcritical.

- Failure of the RPS to shutdown the reactor by inserting control rods.

3.2.18 Recirculation Spray System

The Recirculation Spray (RS) System provides long term Containment pressure reduction and Containment heat removal following an accident by drawing water from the Containment sump and spraying the water into the Containment atmosphere. Heat is removed from the sump water through Service Water cooled heat exchangers. The RS System is a front line system designed to protect the Containment.

3.2.18.1 Recirculation Spray System Description

The RS System is composed of four independent Recirculation Spray trains. Two trains are located entirely inside Containment (IRS) while two trains have the pumps located outside the Containment (ORS). The spray trains draw water from the Containment sump. The ORS and IRS Systems draw from the same sump, although the sump is compartmentalized. Each ORS train has its own separate compartment. Each ORS System pump has an individual suction line from the header with a normally open motor operated valve (MOV). Each pump discharges through a normally open MOV, check valve, and a Service Water heat exchanger. The cooled water is then directed to an independent spray header. The IRS trains have no MOV's. In order to support adequate net positive suction head for the ORS System pumps during the early phase of a loss of coolant accident (LOCA), the Casing Cooling subsystem is provided. This consists of a 100,000 gallon tank of borated water and two pumps each of which discharges to the suction of one ORS pump. Casing Cooling is initiated on the same signals as the ORS pumps and injects until the Casing Cooling tank is empty. A similar function is provided for the IRS pumps by a take-off line from the Quench Spray System.

The RS System automatically starts on receipt of a hi-hi (28 psia) Containment pressure signal from the Containment Depressurization Actuation System (CDA). The CDA signals start the RS System pumps and ensure that the pump inlet and discharge valves are open. An Agastat timer in the pump start circuit delays pump start for 210 seconds for the ORS and 195 seconds for IRS to ensure adequate sump inventory and the correct diesel generator loading sequence in the event of loss of offsite power.

The RS System depends on the AC power buses for motive power to the ORS System pumps and motive and control power to the RS System MOVs, the DC power buses for control power to the RS System pumps, and the CDA for actuation of the RS System pumps.

3.2.18.2 RS System Logic Model

There is one success criterion for the North Anna RS System that is the same for each application in the event tree analysis. This success criterion is that at least one of the four ORS or IRS trains provide flow and cooling to its spray header. A second criterion is that at least one of the two ORS trains provide flow and cooling to its spray header. The second criterion is necessary because the IPE assumes that the IRS system fails if the Quench Spray System fails for the large LOCA. This translates into the following top events in the ORS System fault tree:

- Insufficient flow and cooling from at least one RS System train.
- Insufficient flow and cooling from at least one outside RS system train for large LOCA.

3.2.19 Residual Heat Removal System Model

The Residual Heat Removal (RHR) System provides shutdown cooling when the Reactor Coolant System (RCS) depressurizes below 450 psig and cools below 350°F. Residual Heat Removal is a front line system designed to provide long term decay heat removal.

3.2.19.1 RHR System Description

The North Anna RHR System is composed of two pumps and two RHR heat exchangers in parallel. The RHR pumps take suction from the RCS loop 1 hot leg through two normally shut motor operated valves (MOVs). The discharge of the pumps is headered together and feeds two heat exchangers arranged in parallel. The RHR pumps and heat exchangers are cooled by Component Cooling Water (CC). An air operated valve (AOV) controls bypass flow around the heat exchangers, another controls flow through the heat exchangers. The

two AOVs work together to control the cooldown rate of the RCS. The discharge of the flow control valves feeds into the SI/Accumulator piping and is delivered to the RCS loop 2 and loop 3 cold legs. Each path has a normally shut MOV isolating the RHR from the high pressure RCS during normal plant operations. Makeup to the RHR System is provided by the RCS.

The RHR is manually initiated. An interlock prevents opening the Hot Leg RHR isolation MOVs until RCS pressure is below 450 psig. Only one RHR pump and heat exchanger are required for plant cooldown although both pumps and heat exchangers are normally used immediately following a reactor shutdown, to provide a faster cooldown. Following a loss of offsite power, the stub buses powering the RHR pumps are shed from the emergency buses and must be manually reconnected to restore power to the RHR pumps.

The RHR System is dependent on AC power for motive power for the pumps, and the DC buses for control power to the RHR pumps and the heat exchanger throttle valves. Additionally, the RHR System requires the Instrument Air system for motive power to the heat exchanger throttle valves. The RHR System is dependent on the RCS water level to avoid air binding of the pumps.

Prior to placing the RHR System in service, RCS pressure must be below 450 psig and RCS temperature must be below 350°F. Following a loss of offsite power, the stub buses which power the RHR pumps are automatically shed and must be normally reloaded as the main bus by the operator to restore power to the pumps.

3.2.19.2 RHR System Logic Model

The success criterion for the Surry RHR System requires RHR flow to be provided from one of two pumps through one of two heat exchangers to the RCS following reactor shutdown and cooldown to 450 psig, 350°F. This criterion translates into the following top event in the RHR System fault tree:

- Failure to provide cooled RHR flow to the RCS.

3.2.20 Safety Injection Actuation System Model

The Solid Station Protection System (SSPS, SI actuation system) automatically initiates the Safety Injection Systems, following an indication of the need for primary coolant makeup, and automatically initiates the switchover of the suction of the low pressure injection pumps from the Refueling Water Storage Tank (RWST) to the Containment sump and the switchover of the suction of the high pressure injection pumps from the RWST to the low pressure injection pump discharge upon low RWST level.

3.2.20.1 SSPS Description

The North Anna SSPS is composed of two independent trains used to automatically actuate the low and high pressure injection systems and the motor driven AFW Pumps.

The portion of the SSPS which supports recirculation is composed of four independent RWST level sensors, each feeding two separate two out of four relay matrices. These two relay matrices automatically actuate the components required to perform the switchover to the recirculation mode of the low and high pressure systems. The SSPS is dependent on the AC vital instrumentation buses and the DC buses for operation of the relay logic network.

3.2.20.2 SSPS Logic Model

The SSPS was modeled as a support system to be linked into the components which are activated by the SI signals.

3.2.21 Service Water System

The Service Water System is common to both reactor units and is designed for the simultaneous operation of various subsystems and components of both units. SW System provides long term cooling after a loss of coolant accident (LOCA) and supplies cooling water to the following safety-related components during normal plant operations:

1. Component Cooling (CC) heat exchangers;
2. Recirculation Spray (RS) heat exchangers;
3. Control Room/ESGR air conditioning chiller condensers;
4. charging pump seal coolers, gear reducers, lube oil coolers; and
5. Instrument Air compressors.

The SW System also serves as a backup source of water to the Auxiliary Feedwater System.

The sources of cooling water for the SW System are the SW reservoir and Lake Anna. These two, independent sources of water form the ultimate heat sink for the North Anna Power Station.

3.2.21.1 SW Description

The SW System has two modes of operation: reservoir-to-reservoir, and lake-to-lake. It is normally operated in the reservoir-to-reservoir mode, which uses the SW reservoir as the ultimate heat sink. The SW reservoir is a large pond with a sufficient supply of treated water to provide cooling for both operating units with one of the units suffering from a loss of coolant accident (LOCA). There are two spray headers in the reservoir that spray returning SW into the air to assist in dissipating the heat acquired while cooling the various plant components. Each spray header consists of two pairs (four total) of individual controllable spray arrays. The spray arrays can be bypassed by two spray bypass lines (one per header) leading directly to the reservoir.

There are four SW pumps, of which one pump per unit is normally in operation. The pumps draw SW from the reservoir through a set of traveling screens that filter out debris. The SW pumps provide the motive force for the flow of the SW through the various components cooled by the SW System.

The SW System supplies cooling water through the plant with two supply headers. Two return headers collect the SW from the cooled components and return the water to the reservoir. At the reservoir, the return headers divide the returning SW among the two spray headers or spray bypass lines.

In the lake-to-lake mode of operation, two auxiliary SW pumps draw water directly from Lake Anna through the Circulating Water (CW) System traveling screens. The lake-to-lake mode is used as a backup and during SW System maintenance. The auxiliary SW pumps discharge the SW to the same supply headers as the SW pumps used in the reservoir-to-reservoir mode. The return headers have an auxiliary return header that directs the return SW to the lake. The auxiliary return header is also monitored for radioactivity by a radiation monitor. Two of the CW screen wash pumps serve as makeup pumps for the SW System and can add lake water to the SW reservoir. The auxiliary SW pumps can also be used to provide makeup water to the SW reservoir.

There are two SW supply headers, 1 and 2, that provide SW to all the plant components and systems in both units. Each header is capable of providing 100 percent of the necessary SW flow.

The SW supply headers distribute SW to the following loads of interest to the PRA:

1. four Component Cooling heat exchangers (two per unit);
2. eight Recirculation Spray heat exchangers (four per unit);

3. six Control Room/ESGR air conditioning chiller condensers (three per unit);

4. six Charging Pumps (three per unit):

   - seal coolers,
   - gear reducers, and
   - lube oil coolers; and

5. two Instrument Air compressors.

3.2.21.2 SW System Logic Model

The SW System is a support system and, as such, was linked into the front line systems where necessary. A separate top event was developed for each of the supply headers.

Loss of SW is also treated as an initiating event. An event tree was developed to evaluate the impact of core damage from this initiator. A separate fault tree was developed to find the annual frequency of the T6 initiator.

3.3 SEQUENCE QUANTIFICATION

3.3.1 List of Generic Data

This section discusses the approach used to assemble the set of generic parameter values that was used in the quantification of the PRA logic model, with the exception of those parameters estimated in the human reliability and common cause failure analysis which are discussed in Sections 3.3.3 and 3.3.4, respectively. This set of generic parameter values is henceforth called the generic data base. The parameters include initiating event frequencies, failure probabilities and standby and operating failure rates of components, and unavailabilities of components or trains of components due to test, and due to maintenance. The logic model also contains certain undeveloped events, which because they have a PRA model specific definition are discussed in Section 3.3.2.

Section 3.3.1.1 summarizes the main characteristics of the initiating event and basic event probability models and defines the parameters. The sources of estimates of these parameters and the method used to select the most appropriate are discussed in Section 3.3.1.2. The generic data base is presented in Section 3.3.1.3.

3.3.1.1 Simple Probability Models and Their Parameters

Table 3.3.1-1 presents in summary form, the main characteristics of the probability models used for initiating events, and the component related basic events.

The table presents the assumptions behind the models, identifies the parameters of the models, and the data needed to estimate those parameters. Typically the generic data sources discussed in Section 3.3.1.2, present estimates of initiating event frequencies, standby or operating failure rates, probabilities of failure on demand, or unavailabilities as appropriate.

3.3.1.2 Generic Data Sources

The sources used to provide parameter estimates are largely documents published by the USNRC, EPRI, or other industry groups, or other PRA reports. They are listed in Table 3.3.1-2. Care has to be exercised in using these references as the parameters may be defined differently from reference to reference. For example, different PRAs may have varying definitions of initiating event groups, or of the component boundaries, or indeed even the component failure modes. Thus, in constructing the generic data base, considerable effort was spent in reviewing the basis for the estimates in each reference to identify those that most closely match the requirements of the North Anna PRA model. The selection criteria adopted are discussed in Appendix C. In general, only one source was used as opposed to combining several different sources.

3.3.1.3 The Generic Parameter Data Base

Each of the parameters used is characterized in terms of a probability distribution which characterizes the uncertainty in the parameter value. The mean value of this distribution is used as the point estimate. The generic estimates of initiating event frequencies are given in Table 3.3.1-3. For those initiating events not appearing on Table 3.3.1-3, plant specific data were used (see Section 3.3.2). The generic data for equipment failures and unavailabilities are given in Table 3.3.1-4.

3.3.2 Plant Specific Data and Analysis

The aim of the PRA is to model the North Anna plant as accurately as possible. This includes using the plant operating experience data to provide a basis for estimating some of the model parameters on a plant specific basis. This is discussed in Section 3.3.2.1. There are other events of the model, for which experience data are not available, but whose occurrence rates are expected to be plant specific. These include those initiating events, which are caused

by failures of support systems, and whose frequencies are a function of the plant specific design and operation. In addition, the plant logic model contains some undeveloped events which have a plant and model specific definition (i.e., they are defined specifically for this PRA). For these two types of events, the frequencies in the first case, and probabilities in the second, are estimated by constructing models which represent the plant logic and operating practices, or by expert opinion using input from plant personnel. This plant specific analysis is discussed briefly in Section 3.3.2.2. Finally the plant specific North Anna PRA data base is presented in Section 3.3.2.3.

3.3.2.1 Collection and Reduction of Plant Specific Data

There are many potential sources of operating experience data. These include: Licensee Event Reports (LERs), Work Planning and Tracking System (WPTS), Nuclear Plant Reliability Data System (NPRDS) reports, the Technical Specification Action Statement Log, North Anna Monthly Reports, and maintenance and test procedures. Each of these sources is discussed in Appendix C.

The types of data needed to provide parameter estimates are summarized in Table 3.3.1-1. With the volume of plant records and the large number of components in the plant model, it is clearly impractical to analyze this data for all components. Hence the data reduction was performed for only a small number of the components, which were chosen because they are generally key contributors to PRA results. Thus it was decided to try to obtain plant specific data to enable estimates to be made of:

- initiating event frequencies for the more common transients,

- fail to start probabilities of Engineered Safeguards standby or alternating pumps, and diesel generators,

- unavailabilities due to maintenance for the trains of important standby systems, and

- other faults for components unique to the North Anna configuration such as Emergency Switchgear Room chillers and Service Water System components.

The details of the data gathering, reduction and analysis are presented in Appendix C. The results of the data reduction are summarized in Tables 3.3.2-1, 3.3.2-2, and 3.3.2-3. In general, only the five years of data (1986-1990) were analyzed for the component reliability parameters as this was felt to be adequate to represent the current plant status.

Generally, data for North Anna 1 and North Anna 2 was pooled to increase the sample size (this, of course is not done for common systems and components). Note that 10 years of reactor trip information was reviewed for the Initiating Events T2, T2A, T3, TH and TL, but that only the latest 5 years of data was used to better represent current plant design and operation. The details are provided in Appendix C.

3.3.2.2 Estimation of Probabilities of Undeveloped and Special Initiating Events

The frequencies of the special common cause initiating events, loss of RC pump seal cooling (T4), loss of service water (T6), loss of Emergency Switchgear Room cooling (T8), loss of power from 4160 V Bus 1H (T9A) and loss of power from 4160 V Bus 1J (T9B) are estimated on the basis of fault tree models. Note that T9A and T9B event trees do not use the estimated frequencies for the T9A and T9B initiating events, but rather use equations generated by the fault tree solution. This allows the incorporation of T9A or T9B dependencies within the remainder of the event tree functions. For example, a 1H diesel generator fault for T9A could also affect the solution of the T9A function QS05 for Quench Spray pumps. This consideration is discussed in detail in Appendix C.

The probabilities of undeveloped events were estimated either on the basis of a direct assessment based on input from plant personnel (e.g., the probability that a leaking Main Steam Power Operated Relief Valve is isolated), or by constructing a simple model that captures the key factors of the mechanisms leading to the event (e.g., the Main FW pump lube oil system is simplified to the dominant subsystem fault, a PAT-FR event). These are discussed on a case by case basis in Appendix C.

3.3.2.3 The North Anna Specific PRA Data Base

The parameter data base incorporating plant-specific data used for the quantification of the North Anna PRA model is presented in Tables 3.3.2-4 and 3.3.2-5 in a summarized form. Each parameter is characterized by a probability distribution which is given in Appendix C.

The plant specific data was in general used to update the generic data base using the Bayesian approach, with some exceptions as follows. For the initiating events T2, T2A, T3, TH and TL, a non-informative prior rather than a generic prior was used. For the unavailabilities due to maintenance, the plant specific estimates were used directly without Bayesian updating of the generic estimates.

3.3.3 Human Reliability Analysis

This section summarizes how the human reliability analysis (HRA) task was organized, how the data was derived and used to support the accident sequence quantification, and how the human reliability insights were generated. Detailed descriptions of the methods used and the individual assessments to derive human error probabilities (HEPs) are documented in Appendix D.

3.3.3.1 Scope of the HRA Task

The human reliability analysis (HRA) task of the North Anna IPE project was performed over a period of about 18 months. The task was organized into five steps:

1. identification of preliminary human error basic events by the system analysts,

2. collection of operator response times from the simulator,

3. determination of engineering time windows using the MAAP code,

4. detailed evaluation of each operator error included in the NAPS IPE, and

5. review of the final quantification results to confirm proper modeling of operator errors.

Several factors influence the practical implementation of an HRA task. First, the content of the analysis procedures and instructions together with the available resources (time, budget, personnel) determine how the task is organized and integrated with other PRA study tasks. An equally important factor is the analytical style/experience of the event tree and fault tree modeler; the final decision whether to include/exclude certain HEPs is made in the event tree and fault tree tasks, whereas the human reliability expertise primarily provides input with respect to derivation of HEPs and interpretation of results.

The HRA task has addressed two major groups of human interactions (HIs): pre-accident operator actions, Type A HEPs (testing, calibration and maintenance), and post-accident operator actions, Type C HEPs (Control Room crew responses to transients by following the applicable emergency procedures).

The basic HRA methodology adopted for the North Anna IPE is quite similar to the methodology used in other, current U.S. PRA projects. Expert judgement has been an essential aspect of the estimation of human interaction basic event probabilities, referred to as HEPs; but, in order to provide a North Anna specific

assessment, the HRA has been supported by making observations at the simulator.

A series of simulator experiments was conducted during the winter of 1992. The emphasis of these experiments was to generate qualitative insights on a range of operator actions associated with emergency response following SGTR initiators. In addition, response time data were extracted for the operator actions for later use when quantifying non-response probability.

Much of the recent human reliability model development has focused on the post-accident operator actions, and specifically the time-dependent actions taken by crew members in response to an initiating event. This time dependency stems from the constraints imposed on the operators by the plant hardware and thermal-hydraulic characteristics. A common representation of the post-accident operator actions is the event tree style of Figure 3.3.3-1 with the p_1, p_2 and p_3 end points. The p_1-parameter represents an error of cognition which is unrecovered, p_2 represents the probability of non-response given the operators are on the correct "cognitive path," and p_3 represents failure to correctly implement a step in an emergency procedure.

Where data was available from simulator exercises, the operator response times were used as a basis for the quantification of the parameter P_2. The approach adopted to the estimation of the remaining parameters is described in Appendix D.

In addition to analyzing the potential operator non-response, a set of specific non-proceduralized recovery actions were analyzed. These actions were included in the event tree/fault tree models as part of the final sequence quantification to acknowledge the possibility of restoring failed equipment or functions given sufficient time is available to the operators. The specific operator actions for North Anna IPE are summarized in Tables 3.3.3-1 and 3.3.3-2.

3.3.3.2 Organization of the HRA Task

The HRA task was performed in parallel with the accident delineation and system analysis tasks, and there were numerous iterations between the system analysts and the human reliability analysts. The HRA task was organized into five steps as follows:

1. Identification of preliminary human error basic events by the system analysts. Using the experienced gained from the Surry IPE, the system analysts were able to include basic events representing Type A and Type C HEPs in each system fault trees. Point estimate values were assigned based on the Surry values for the initial quantification.

2. Collection of operator response times from the North Anna simulator. Seven simulator sessions were observed during the winter of 1992. These sessions presented various SGTR events during which operator response time data was obtained for the procedures 1-E-0, "Reactor Trip or Safety Injection" and 1-E-3, "Steam Generator Tube Rupture."

3. Determination of engineering time windows using the MAAP code. Each Type C HEP basic event was evaluated to create one or more accident transients which were run with MAAP to determine how quickly the operators must respond to a plant condition to prevent core damage.

4. Detailed evaluation of each operator error included in the NAPS IPE. This task included reviewing all fault trees and event trees to identify every function which includes operator actions. Then an evaluation was performed which identified station procedures related to the operator action. A detailed analysis was performed to calculate a probability based on the procedures, simulator data and MAAP runs. See Appendix D for details of each evaluation.

5. Review of the final quantification results to confirm proper modeling of operator errors. The most significant basic events were reviewed as well as the sequence equations to ensure the operator errors occurred where expected. Procedure enhancements could also be evaluated using the final quantification results.

During the winter of 1992 members of the North Anna IPE team collected information on shift crew responses to simulated SGTR accidents at the North Anna Training Simulator. The training exercises were part of the normal Licensed Operator Requalification Program (LORP) but modified somewhat to meet the data needs of the IPE. Information was collected by observing the crews in action, and video tapes with sound tracks allowed the investigators to prepare detailed event chronologies and to identify when in time a diagnosis or decision was made. Time data on operator responses to a total of seven scenarios were collected and analyzed; see Appendix D.

3.3.3.3 Overview of HEPs

Human error probability basic events are a significant part of the overall North Anna PRA model. To understand the HEPs several studies were performed.

The importance of human error probabilities (HEPs) basic events in the North Anna IPE can be seen in Table 3.3.3-3. This table was

made using an overall equation (NAPS.EQN) which combined cut sets above 1E-11 from all core damage sequences. The ranking number is the overall importance ranking for the basic events, there were 710 in the overall equation. Non HEPs basic events were removed from the table so that the importance of HEPs too each other could more easily be understood. A brief discussion of the most important operator actions included in the North Anna IPE follows.

HEP-1FRH:1-11, point estimate 5E-2, represents the operators failing to establishing safety injection flow when feed and bleed cooling is necessary. This HEP is used in the FB400 fault tree and appears in 30 sequences for the failure of function D102 or D105.

HEP-0AP55-10HR, point estimate = 5E-3, represents the operators failing to restore ESGR cooling within ten hours of equipment failure. This HEP is used in the FFT00 fault tree and appears in 22 sequences for the failure of function RC102 and RC103.

HEP-1FRC:1-11-S1, point estimate = 1.0, represents the operators failing to depressurize the Steam Generators within 3.5 minutes during a medium break LOCA and loss of HHSI, success of the Accumulators and Auxiliary Feedwater. The time until core exit temperatures reach 1200°F is too short for the operators to respond. This HEP is used in the FFT00 fault tree and appears in 4 S1 sequences for the failure of function Y01.

HEP-NO-PROCEDURE, point estimate = 1.0, represents the operators failing to identify that SI flow diversion is occurring through 1-CH-P-1A when its discharge check valve remains open. Flow diversion through a discharge check valve is possible when 1-CH-P-1A was the running pump, approximately 1/3 of the time, and both 1-CH-P-1B and 1-CH-P-1C are in auto-after-stop. The SI logic will stop the 1A pump and start 1B and 1C. If the discharge check for the 1A pump remains open, flow from the other pumps will recirculate through the 1A pump preventing full SI flow to the RCS. Currently there are no administrative procedures to prevent occurrence of this event and the operators have no procedure guidance for correcting this condition. This HEP appears in 3 fault trees, FB400, HH100 and HR100, which are used for 14 functions, D1, D4 and H2 series, resulting in 68 sequences with this basic event.

HEP-1E3-13, point estimate = 2.18E-2, represents the operators failure to initiate a RCS cooldown so that the RCS pressure is less than the Steam Generator pressure within 68 minutes. This HEP appears in the FFT00 and OD200 fault trees, the O06 function and sequences T7P03 and T7P04.

HEP-1ES1:2-S1, point estimate = 1, represents the operators failure to initiate as RCS cooldown and depressurization within 23 minutes after a medium break LOCA. This HEP appears in the FFT00 fault tree, is used in the 001 function which appears in the sequences S1P08 to S1P12 and VXP03.

HEP-0AP55-20HR, point estimate = 2.6E-4, represents the operators failing to restore ESGR cooling within twenty hours of equipment failure. This HEP is used in the FFT00 fault tree and appears in 24 sequences for the failure of function RC202.

HEP-1AP22:5, point estimate = 1.75E-4, represent the operators failing to begin refilling the ECST when its level drops below 40% before it becomes empty, failing the AFW pumps.

The top HEPs in the importance analysis shows which operator actions could most benefit from future HRA analysis or station attention. This potential future work could include improved operator training, procedure enhancements and improved management oversight during accidents.

3.3.3.4 HEP Sensitivity Studies

The North Anna IPE results are sensitive to the point estimates selected for the human error probability basic events. A sensitivity of HEPs on core damage frequency (CDF) was performed by changing the value of each HEP to one. The results are shown in Table 3.3.3-4. The table was organized to place the HEPs with the greatest effect on CDF at the top, followed by the HEPs with less effect on CDF. Recall the North Anna IPE overall core damage frequency is 6.8E-5/year.

Table 3.3.3-5 shows the importance listing of HEPs when all are set equal to one with an overall core damage equation truncated to 1E-8. These are the ranking of operator actions which the most credit was taken during the North Anna IPE.

As a comparison to setting all HEPs equal to one a study was made setting all HEPs equal to 1E-15. This assumes all operator required actions are performed perfectly. The results show that overall core damage frequency decreases from 6.8E-5 to 3.5E-5/year. More can be learned from studying the effects of increasing the point estimates of HEPs than by decreasing HEPs. This is due to the distribution of all core damage sequences resulting in no outlier cut sets and truncation of low value cut sets. Future sensitivity studies of HEPs will produce the most significant results when HEP values are increased rather than decreased from the initial point estimates.

3.3.3.5 HRA Comparison To Surry

The HRA method used for the North Anna IPE was similar to the Surry IPE. The Type C, operator actions after accident initiation, HEPs for North Anna and Surry are the similar except where the time windows used to calculate the p_2, time dependent probability, term was significantly different. Several HEPs associated with the SGTR are different between North Anna and Surry IPEs. The North Anna SGTR time windows are based on operator response times from simulator data and detailed MAAP analysis. The Surry SGTR time windows are engineering estimates. Four non SGTR Type C HEPs which are significantly different between North Anna and Surry IPEs discussed in Appendix D. Future Surry PRA work may refine the Surry point estimates to be more similar to the North Anna HEPs.

The North Anna and Surry PRA models can be studied to determine if the differences in the results are due to human error probabilities or due to equipment modeling differences. Details may be found in Appendix D. Generalizations concerning the two PRAs may be made. However, detailed analysis must be used to confirm these observations. The large, medium, small and intersystem LOCAs, and the Reactor Vessel failure accident initiators have similar HEPs and equipment models for both the North Anna and Surry PRAs.

T5A and T5B are examples of equipment differences between North Anna and Surry PRAs where the HEPs are similar. The T5s are different because of the differences in the DC bus design differences. There are eight DC buses are North Anna and five at Surry (two per unit and one non emergency power bus). Other examples are the T2, T2A and T3 accident initiators which have approximately the same HEPs but the Auxiliary Feedwater systems are different between the stations. Surry has AFW cross connects between units. North Anna can not deliver AFW flow from one unit to another.

The T7, SGTR, CDF differences in North Anna and Surry appear to be a result of the HEP differences and not equipment differences. The North Anna more accurately defined the engineering allowable time windows by using MAAP analysis and improved the operator response times by utilizing simulator data.

3.3.3.6 Sensitivity Results For Procedure Improvements

The point estimates for the human error probability basic events were developed assuming that several procedure changes would be made to minimize operator errors for actions significant to the IPE. This section will summarize a sensitivity study on these assumed procedure changes included in this PRA.

AFW Full Flow Recirculation Valves

Revising all procedures which open the Auxiliary Feedwater full flow recirculation manual valves to include independent verification that the valves are closed upon completion of the procedure was identified as a potential procedure improvement. When the recirculation valves are independently verified to be closed, the core damage frequency remains at 6.8E-5. If the MOVs are not independently verified to be closed, then the CDF increases to 7.2E-5 (+6%). The procedure improvement is important to minimizing core damage frequency.

QS and RS Piping

Revising all procedures which realign the Quench Spray or Recirculation Spray headers for testing to include independent verification that the headers have been restored to fully operable was identified as a potential procedure improvement. When system restoration specifically identifies all test devices, including exact location of each device, to be removed (and how to restore the piping), and the QS and RS system piping is independently verified to be operable by operator performing a walkdown at the end of each refueling outage, then the CDF remains at 6.8E-5/year. This independent verification assumes the use of a procedure which includes a list of all valves (and the operable position), all normally installed spool pieces, flanges, and elbows and the location of all potential test devices (flanges, elbows, etc.) which should have been removed. If independent verification is not added to the procedures, the CDF increases to 7.0E-5/year (+3%).

Alternate SI Header, 1-SI-MOV-1836

Revising 1-E-0, Reactor Trip or Safety Injection, to add opening 1-SI-MOV-1836 to the RNO of step 14, verify SI flow, or to the RNO of step 16, check Charging Pump alignment was identified as a potential procedure improvement. This MOV allows flow to the alternate SI flow path to the cold leg.

When procedures are revised for the operator to open 1-SI-MOV-1836 after failures in the normal SI flow path, the CDF remains at 6.8E-5/year. If 1-E-0 is not revised then the CDF increases to 7.1E-5/year (+5%).

3.3.4 Common Cause Failure Data

The common cause failure analysis was performed following the general guidelines of the procedure documented in NUREG/CR-4780. As a result of a qualitative screening following the guidance for stage 2 of the procedure, and supplemented with insights drawn from

experience in performing data analyses, the following common cause failure component groups were identified for inclusion in the plant logic models.

Standby pumps - fail to start and run

Standby and alternating pumps - fail to start and run after Loss of Offsite Power (LOOP)

Operated valves - fail to open/close

Check valves - fail to open/close

Diesel generators - fail to start and run

Batteries - fail to supply power

Chillers for ESGR cooling - fail to start and run

Instrument Channels - loss of function

SW Reservoir Screenwells - plugging

The rationale for the choice of groups is presented in Appendix C.

A quantitative screening was performed using a Beta factor model with a beta factor of 0.1. This indicated that a detailed analysis should be performed for Auxiliary FW motor driven pumps, LHSI motor driven pumps, diesel generators, and MOVs using the approach described in NUREG/CR-4780. Following this approach the event data in Nuclear Power Experience was reviewed and a pseudo plant specific data base developed. This was then used to evaluate the CCF basic event probabilities using the basic parameter model.

The probabilities of the CCF basic events are included in Table 3.3.4-1. For all of the events not analyzed in detail, the screening values were retained in the final analysis.

3.3.5 Quantification of Unavailabilities of Systems and Functions

The North Anna IPE was performed using a linked fault tree approach as described in NUREG/CR-2300 using the NUPRA PRA workstation. In performing a PRA in this manner, the functions defined by the event tree headings are the main building blocks of the quantification process. The quantified functions can be used to determine the overall contribution of a given function failure, such as decay heat removal, to core damage. Each function is representative of failure of one or more front line system and/or human actions. Support systems are linked into the front line systems they support. A given function or combination of systems may be quantified several times to reflect different boundary conditions

as the result of different initiating events. The results of the quantification of each function are summarized in Table 3.3.5-1. The quantification process and the results are presented in detail in Appendix B of this report.

3.3.5.1 Summary of Function Quantification Process

The accident sequence analysis task identifies safety functions that must be provided in order to prevent core damage following an initiating event. The function is defined by the system success criteria, the mission time, important operator actions, and any sequence specific equipment unavailability, environmental, or phenomenological conditions that exist following a specific initiating event. The combination of the above used to develop the function fault tree for a given initiator are first combined and linked with the appropriate support systems. In order to avoid developing a separate fault tree for each condition in which a system is required to respond, a "switch" known as a "house event" is used to switch in or out various sections of the system fault tree for each condition. A "house event Basic Event Data (BED) file" is constructed for each initiating event. This defines the status (on or off) of each house event for that initiating event.

The functional quantification process is very straight forward. The appropriate fault tree(s) for the function of interest are updated against the house event file appropriate for the initiator of interest. This updating will turn on and off gates as appropriate. The fault tree is then solved and quantified using appropriate truncation values. The result is a Boolean equation which is used in the sequence quantification as well as giving a quantified value for that function. Some functions may be single events and would not require this process.

3.3.5.2 Summary of Function Unavailabilities

The functional unavailabilities are shown in Table 3.3.5-1. This table identifies the function, and describes the success criteria. Each function is named after the function identifier in the event tree. For example, the function D1 represents failure of high head injection. In the event trees it is found that high head injection is required for nine different sets of conditions. Thus the functions are identified as D101, D102, D103, ..., D109. The exact function used in each sequence of each event tree is printed on the event tree.

3.3.6 Quantification of Sequence Frequencies

This section will describe how the quantification of sequence frequencies was accomplished. The methodology is discussed and an

example of how one sequence was quantified is given. A brief explanation of the computer code modules used to perform the quantification is given at the appropriate stage. The full quantification documentation is given in Appendix B.

Sequence frequency quantification is accomplished by combining the combinations of failures for each function resulting from the quantification of the function unavailabilities (Section 3.3.5), as determined by the event tree structure (Sections 3.1.2 and 3.1.3). The event trees define the 667 plant damage sequences, 42 transfer sequences and 75 non-core damage sequences. NUPRA was used to perform the task of sequence quantification. The non-core damage sequences, OK status, are shown on the event trees for convenience, but are not quantified since their results have no application. These non-core damage sequences will not be discussed further.

3.3.6.1 Initial Quantification of Sequence Frequencies

Each core damage sequence is quantified in a number of stages. In the first stage all the combinations of failures and the initiating event are combined to give a set of combinations of failures (known as minimal cut sets) which will lead to core damage. However in any given sequence a number of functions may have been successful and therefore some of the combinations may not be applicable as the particular piece of equipment may have been successful. If for example a specific electrical supply appears in both a successful and failed function it cannot be both failed and successful. In the second stage the combinations of failures are compared with the successful functions and all the non-allowed failures removed.

In the third stage the remaining combinations are compared with disallowed combinations of failures, for example Trains A - B in planned maintenance and these combinations of failures removed.

The final list of combinations (cut sets) represents the contributors to core damage and the sum of the frequency of each combination, the sequence core damage frequency. The sum of all the core damage frequencies gives a point estimate of the overall core damage frequency.

The PRA code NUPRA was used to perform the quantification in the following manner.

The NUPRA Sequence Function Equation Assignment Module was used to automatically generate quantification control (merge) files for each sequence in each of the 26 event trees based on the event tree construction and function assignment. This file is then used to automatically perform the quantification of all the sequences in the event tree. The initial cutoff selected for each step of the sequence quantification was a truncation value of 1E-9 for intermediate steps and 1E-11 for final sequence step. The results

from function quantification used in the quantification of each sequence are in the form of equations containing the minimal cut sets above a cutoff value. The cutoff value varies over a wide range, but was always three orders of magnitude smaller than the function unavailability. Then each sequence is quantified automatically by combining the cut sets for each function.

The event tree merge control files perform two processes for each core damage and transfer sequence. The first process is to combine the function equation failures with the initiating event function equation using Boolean binary conjunction, AND logic. The second process is to delete the functional successes using Boolean binary subtraction of each cut set. The deleted cut sets are only for those successful functions for each sequence as defined by the event tree structure. The event tree merge control file names were named using the corresponding event tree name with .OCL as the filename extension. For example for large LOCA (A) this would be A.OCL.

The output from the quantification consists of two files. One output file is the sequence equation file which contains all of the cut sets which are greater than the truncation value. It should be emphasized that this file contains ONLY cut sets and no data or cut set values. Another output file is the sequence printout file which contains the top 100 cut sets ranked in order of magnitude using the data in the data base at the time of quantification. Appendix B contains the first page from the printout files for every sequence resulting in a core damage frequency greater than 1.0E-7. The sequence file names are the same as the event tree names combined with the corresponding sequence number. The equation files have a filename extension of .EQN and the printout file has a filename extension of .MGP.

The sequence equation and printout files were then modified to take account of Technical Specification limitations, to delete successful function cut sets for the transfer sequences or to correct the value for success branches of event trees when the failure branch is greater than 0.05. These three operations are described in more detail in the following paragraphs. The sequence files were manipulated using NUPRA Cut Set Equation Merging Module. Two cut set merge control files were created to perform the intended operations. The output sequence files were named the same as the input files, and a truncation value of 1.0E-12 was utilized to maintain the same level of cut set information.

Technical Specification Limiting Conditions of Operations prohibit certain multiple trains of equipment from being inoperable during power operation. A fault tree, DAM00 (see Appendix B), was created to provide cut sets which represent Technical Specification disallowed maintenance combinations. The cut set function equation for disallowed maintenance, DAM.EQN, was used to perform Boolean subtraction from every sequence using a cut set merge control file

for each event tree. These files were called by the event tree name with the three letters DAM added (e.g., A event tree was called ADAM).

During the event tree development, 42 core damage sequences were identified as potential transfers from one event tree to another. The results from 35 of these sequences indicate that a transfer is not actually necessary because either the transfer sequence frequency is well below the initiating event frequency of the tree to which it would be transferred or the sequence could be recovered. See Appendix B tables for a summary of these transfer frequencies.

Transfer sequence T1P63 or T1Hv was used as the initiator for the T1Tr event tree, modeling a loss of Emergency Switchgear Room (ESGR) cooling caused by a Loss of Offsite Power (T1) and ESGR component faults. The T1Tr event tree is similar to T8 in that the same events are used. However, since T1 results in an immediate reactor trip, the T6 event tree structure is used, as it more closely represents the event timing anticipated for a T1Hv sequence.

Transfer sequence T1P64 was actually used as the initiating event frequency for Station Blackout at Unit 1 (event tree T1A). The T1P64 sequence file may be used directly as input to T1A.

Transfer T2P20 or T2Hv was used as the initiator for the T2Tr event tree, modeling a loss of ESGR cooling caused by a T2 loss of Main Feedwater transient and ESGR component faults. Similarly, transfers T2AP21 or T2AHv and T3P21 or T3Hv were used as the initiators for T2ATr and T3Tr event trees, modeling a loss of ESGR cooling caused by T2A or T3 transients and ESGR component fault. All three event trees, T2Tr, T2ATr and T3Tr, have similar structure and events to the T8 event tree. As the T2, T2A and T3 initiators do not directly fail support systems, the ESGR cooling faults are more greatly influenced by mission time than by demand faults as when T1, T9A or T9B occur.

Transfer sequences T9AP23 or T9AHv and T9BP23 or T9BHv were used as initiators for the T9ATr and T9BTr event trees, modeling a loss of ESGR cooling caused by T9A or T9B transients and ESGR component faults. Both T9A and T9B fail support system required by ESGR cooling, so the event tree structure for T6 is used to more closely represent the event timing anticipated for T9AHv and T9BHv sequences. Both T9ATr and T9BTr event trees have the same events used in T8.

When the failure probability of a function is greater than 0.05 then the success branch approximation of one can be overly conservative. NUPRA automatically uses a success approximation value of one, unless the option for complements is exercised. NUPRA allows complement functions to be entered for any specified

success path, which reduces the frequencies of affected sequences for a more realistic result. Since each complement function requires manpower to calculate the complement probability, modify the event tree, complement fault tree, BED and PRM files, etc., only a limited number of complements were generated, primarily for those sequences which have higher CDF values. Generally, complements were not generated for functions with a probability less than 0.05. After this final processing, the sequence equation files and the sequence printout files were then ready to determine total core damage frequency or as input for Back-End Analysis.

3.3.6.2 Example of How Sequence Frequencies Were Calculated

The sequence T7SGIW will be used as an example of how sequence frequencies were calculated (Figure 3.1-T7). This sequence results in one of the higher core damage frequencies. In the event tree for Steam Generator Tube Rupture, (T7), sequence number six is defined as the initiating event T7 combined with the failure of functions SGI01 and W01, and the success of functions, K01, HV01, D101, L07 and O202.

An event tree merge control file, T7.OCL, was automatically created by NUPRA using the T7 event tree structure and function assignments. This file is contains the logic for the Boolean binary conjunction of the initiating event and the function failures and the Boolean subtraction of the function success. The logic may be represented as follows:

```
SGI01     *    W01   =    INT1
INT1      *    T7    =    INT3
INT3      \    K01   =    INT4
INT4      \    HV01  =    INT5
INT5      \    D101  =    INT6
INT6      \    L07   =    INT7
INT7      \    O202  =    T7P06
```

Where * represents a Boolean AND operation and \ represents a delete operation

3.3.7 Internal Flood Analysis

Internal plant flooding encompasses the effects from the accumulation, spraying or dripping of fluids arising from the rupture, cracking or incorrect operation of components within the plant. In practice major internal floods have occurred in nuclear power plants, for example, from the rupture of pipes, valves and expansion joints as well as from operator errors during plant maintenance activities. All potential internal flood sources and causes are considered in this analysis, with the exception of those that result from the loss of primary or secondary reactor coolant

outside the Containment (i.e., interfacing LOCAs or steam line breaks with failure to isolate). Such events are considered in the internal initiating events analysis.

Internal flooding, like other so called "external events" merits consideration as a potentially significant risk contributor because of its potential for common cause equipment failures and/or human actions which may result in an accident initiating event (e.g., loss of Main Feedwater) and loss of one or more accident mitigating systems. The detailed analysis of such events is very plant specific, since their likelihood of progression and subsequent impact on plant systems is highly dependent on factors such as, layout, piping arrangements, drainage as well as the prevailing flood protection features and programs. Furthermore, in evaluating the frequency of flood induced accident sequences, the probability of coincident random equipment failures and operator errors must also be taken into account.

In theory, at least, the risk from all flood sources anywhere in the plant could be assessed in a detailed realistic manner. However, this is impractical due to the large number of potential flood sources, and unnecessary since floods in many areas can be shown to be insignificant contributors by simple bounding arguments and analyses. For the sake of efficiency the analysis is usually performed in several levels of detail, each one more refined than the previous. At the conclusion of each level of analysis some plant areas (or particular flood scenarios) are determined not to be significant while others are singled out for further analysis at the next level of detail. Plant specific information required to evaluate the impact of flooding at North Anna was mainly obtained from the Appendix R safe shutdown submittal (Virginia Power, 1992c), UFSAR, (Virginia Power, 1992), general arrangement and piping drawings. Previously performed deterministic evaluations of flood levels in Turbine Building and Auxiliary Building at North Anna (SWEC, 1974, 1977) were also reviewed. In addition, information obtained during plant walk-downs and in discussions with North Anna plant staff proved invaluable. The methodology involved in performing the flood analysis and details of the analysis itself are summarized in Appendix E. The results of the screening analysis are given in Section 3.3.7.2 and the detailed analysis in Section 3.3.7.3.

3.3.7.1 Screening of Flood Areas

The initial screening analysis resulted in the identification of twenty nine areas which could be considered independent with respect to internal flooding events. The potential flood damage in each of these areas and the consequent initiating event (cause of a reactor trip) are summarized in Table 3.3.7-1. The areas of significance identified as the result of the screening analysis were:

Turbine Building
Auxiliary Building
Unit 1 Quench Spray Pump House
Unit 2 Quench Spray Pump House
Unit 1 Air Conditioning Chiller Room
Unit 2 Air Conditioning Chiller Room

Each of these areas, the flood sources and flood propagation pathways are described in detail in Appendix E.

As the result of performing the screening analysis several flood propagations pathways were identified. In order to reduce the frequency of floods in the Auxiliary Building and ESGR, the following hardware modifications were identified to the flood propogation pathways:

1. Installation of backflow prevention devices in the common drain lines to prevent the propagation of the floods from the Auxiliary Building basement floor to the charging pump cubicles via the floor drain system. This modification has increased the critical flood height in the Auxiliary Building from 24" to 44" which in turn increases the critical flood volume and the time available for isolation of a flood source. Without the backflow devices, or maintaining the charging pump cubicles sealed to 44", the total flooding core damage frequency increases to 9.0E-5/year.

2. Reinforcement of the present fire barrier in the pipe penetration between the Auxiliary Building and the QSPH such that it limits the flooding flow rate reaching the Auxiliary Building to less than approximately 300 gpm. This modification will keep a majority of the flood water originating in the Safeguards Building or QSPH from damaging equipment in the Auxiliary Building. Without improving this pipe penetration, the total flood core damage frequency increases to 7.0E-5/year.

3. Modifications to the Chiller Room/Fan Room door and the Chiller Room/Turbine Building door. The modification to the Chiller Room/Fan Room door is to add a 3'3" high dike. This will cause a flood originating in the Chiller Room to fill the room to 3' deep and overflow the existing 3' dike protecting the Chiller Room/Turbine Building door. The Chiller Room/Turbine Building door must be modified to allow a sufficient gap at the bottom to allow the worst case flood to leak under the door, around the missile shield protecting the door and into the Turbine Building. Due to the large area of the Turbine Building and relatively small flood flow rate, there will be no significant hazard to the Turbine Building. These door modifications will allow the redirection of the flood water away from the ESGR. This flood will cause a loss of all ESGR cooling for one unit. This is less severe than

allowing the flood to propagate to the ESGR where it will cause a loss of AC and DC emergency electrical power and all instrumentation to both units. Without the modifications which direct flooding from the Chiller Room to the Turbine Building and away from the ESGR, the total flooding core damage frequency increases to 6.4E-5/year.

The final quantification of the internal flooding core damage frequency of 3.6E-6/year takes account of these modifications.

3.3.7.2 Auxiliary Building Floods

The frequency of floods in the Auxiliary Building from all sources is shown in Tables 3.3.7-2 through 3.3.7-7. The combinations of the various flood sources and flood rates resulted in the identification of two flood damage states in the Auxiliary Building: 1AB2 and 1AB4. The contribution of each of these to the overall core damage frequency is discussed in the following paragraphs.

Flood damage state 1AB2 arises from service water floods in the Auxiliary Building. Flooding results in damage to the charging pumps, the CC pumps and the loss of the Service Water header (cause of the flood). This results in a loss of seal cooling initiating event and the core damage frequency is evaluated using the loss of seal cooling event trees (T4). In addition containment heat removal is not available because of the loss of SW.

Due to the presence of accident mitigating systems (such as the Auxiliary Feedwater System, the Accumulators, and Low Head Safety System), the CDF is calculated to be 2.6E-6 per year.

A review of SW supply headers to the CC heat exchanger indicate that SW induced flooding events in the AB can be divided into two types, Manual Isolable Floods and Remotely Isolable Floods, depending on the location of the rupture. The LTI and STI flooding events are defined as follows:

Manually Isolable Floods are those which occur due to breaks in the isolation valves (1-SW-MOV-108A/B and 2-SW-MOV-208A/B) or in the piping between the isolation valves and the SW supply header to the CC heat exchangers. Tripping of the SW pumps is not effective in terminating this flooding event due to the siphoning effects in the service water pipework. Floods can only be isolated by closing the SW pump's discharge valves which are located in the SW pump house. This room is not manned and, in this analysis, it is assumed that it would take at least 30 minutes for an operator to be dispatched and valve closure to be completed.

Remotely Isolable Floods are those which occur due to breaks in the SW piping upstream of the isolation valves (108s and 208s) on the

SW supply header to the CC heat exchangers. Tripping of the SW pumps in combination with closure of the isolation MOVs will terminate flow from the SW reservoir. However, the water in the SW return header would siphon into the Auxiliary Building. In summary:

Flood Damage State:	1AB2
Frequency of Occurrence:	1.0E-4
Core Damage Frequency:	2.6E-6

Flood damage state 1AB4 occurs as the result of the fire protection system failing in the Auxiliary Building. This flood would lead to failure of the charging pumps as well as CC pumps leading to a loss of seal cooling. The core damage frequency is therefore calculated using the T4 event tree. In summary:

Flood Damage State:	1AB4
Frequency of Occurrence:	6.2E-7
Core Damage Frequency:	1.6E-8

3.3.7.3 Quench Spray Pump House Floods

Because of plant modifications to eliminate the potential flood flow path to and from the Auxiliary Building, a flood in this area will be confined to this building. This flood is not expected to result in a plant trip, and core damage frequency from this scenario is insignificant.

3.3.7.4 Chiller Room Floods

The frequency of floods in the Chiller Room from all sources is shown in Table 3.3.7-8. The contribution of service water flooding to the frequency of flood damage state 1AC1 is presented in Table 3.3.7-9. The combination of the various flood rates and sources resulted in the identification of flood damage state 1AC1. The contribution from this FDS to the overall core damage frequency is discussed below.

Flood Damage State:	1AC1
Frequency of Occurrence:	5.6E-4
Core Damage Frequency:	9.7E-7

This flood damage state arises from service water system floods in the air conditioning and chiller room which result in the loss of the control and relay room HVAC. This results in loss of cooling to the Emergency Switchgear Room, and is equivalent to the T8 initiating event. Loss of switchgear room cooling eventually leads to loss of all AC power and hence a station blackout. The sequence of events following loss of cooling is developed by using a modified internal event tree for T8.

The critical flood volume is 11,000 gallons at which point the level reaches 28" and results in the loss of the chiller.

3.3.7.5 Summary of Base Core Analysis Results

A summary of the flood damage state frequency calculations is presented in Table 3.3.7-10. As discussed, above modified internal event trees were solved to obtain the contribution to core damage frequency for each flood damage state. The core damage frequency resulting from events initiated by internal flooding is 3.63E-6/yr which is approximately 5% of the total core damage frequency of 7.15E-5/yr. The dominant contributor is from Service Water floods in the Auxiliary Building.

3.3.7.6 Summary and Conclusions

A number of propagation pathways which would lead to loss of essential equipment were identified during the performance of the screening analysis. The plant modifications discussed in Section 3.3.7.2 were identified and the core damage frequency from flooding evaluated following the completion of these modifications. The current core damage frequency from internal flooding is 3.63E-6 which is approximately 5% of the overall core damage frequency. The dominant contribution is from service water floods in the Auxiliary Building.

3.4 RESULTS AND SCREENING PROCESS

Core damage for the North Anna Units 1 and 2 IPE is defined either as the condition when the core exit temperature exceeds 1200°F and there is no probability of recovery of core cooling or the licensing basis of maximum fuel temperature of 2200°F is reached during the course of the transient. Although both these criteria are slightly conservative, the analysis shows that the additional time available to recover cooling systems before serious core degradation occurs is small and would not significantly lower the core damage frequency for the dominant sequences. However, in evaluating the frequency of release of fission products, consideration has been given to recovery of systems and reestablishment of decay heat removal capability prior to vessel failure or prior to Containment failure. Thus, the slight conservatism in the arbitrary definition of the frequency of core damage has not been carried over into the frequency of fission product release. The frequency of each accident sequence contributing to the various plant damage states takes this into account.

In this section the results and screening process used to identify the core damage frequency are discussed. Results associated with

the Level 2 analysis, that is the contribution of sequences to the Containment failure frequency and radionuclide source term category frequencies are summarized in Sections 4.7 and 7.0.

3.4.1 Application of Generic Letter Screening Data

The majority of the event trees were constructed using individual systems for the event headings. The remaining headings were represented by functions combining one or more systems and operator actions. The following screening criteria have been used for inclusion of the results in this section of the report. These have been based on the criteria for systemic sequences rather than functional sequences.

1. All sequences quantified above the cutoff of 1.0E-7 have been included in the summary of core damage frequency by initiating event (Table 3.4.1-1).

2. Any sequence contributing 1E-7 or more per reactor year to core damage, grouped by initiator (Table 3.4.1-2).

3. All sequences that are within the upper 95% of the total core damage frequency (Table 3.4.1-3).

4. System sequences that contribute to a Containment bypass frequency in excess of 1E-8 per reactor year (Table 3.4.1-4).

5. Any system sequence that is determined from previous applicable PRAs or by engineering judgment to be an important contribution to core damage frequency or poor Containment performance which is not already included in Tables 3.4.1-1 through 3.4.1-4.

6. Identification of sequences that, but for low human error rates in post initiator operator actions, would have been above the applicable core damage frequency screening criteria (Table 3.4.1-13).

7. All systemic sequences within the upper 95% of the total Containment failure frequency (Table 3.4.1-10).

In order to meet the above screening criteria the accident sequence frequency calculations were performed using a cut set cutoff of 1.0E-10. Recovery actions were included after the initial quantification by modifying the applicable cut sets in a given sequence. No cut sets were cut off after adding the recovery actions therefore information was retained on all sequences that meet the original screening criteria of 1E-7.

The submittal guidance requires that all sequences that, but for low human error rates in post initiator operator actions, would have been above the applicable core damage frequency screening criteria be identified. By using an initial cutoff two to three orders of magnitude below the basic screening criteria and not eliminating any sequences after applying recovery action, all such sequences are retained. These sequences are listed in Table 3.4.1-13.

3.4.1.1 Core Damage Frequency

The internal events portion of the PRA identified 61 core damage sequences with an annual frequency of greater than 1.0E-7 contributing 96% of the overall core damage frequency. An additional 161 sequences with a point estimate frequency of greater than 1.0E-9 per year contributing the remaining 4% of the overall core damage frequency. Each initiating event's contribution to the overall CDF is shown in Table 3.4.1-1. The CDF results given, except where stated otherwise are point estimates.

The point estimate frequency of core damage is 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 per reactor year. 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 3.4.1-1. 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, 1991). 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 is made on the basis of industry accepted practice; the use of large error factors is a way of increasing the mean value with respect to a given median value

(e.g., AOVs), but it also increases the variance. Thus, we believe that the difference between the point estimate and mean value is potentially exaggerated by the way in which the uncertainty characterization of parameter estimates has been established.

On review of the cut sets, it does 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 are used in the remainder of the analysis. In further support of this approach, it should be noted that the point estimate values for the parameters have been chosen to be either realistic (when sufficient data are available) or conservative.

The dominant initiating event type is a small LOCA contributing 14.8% to the overall core damage frequency, followed by Station Blackout at 11.7%, Loss of Offsite Power with a consequential loss of Emergency Switchgear Room cooling at 10.7%, and Steam Generator Tube Rupture at 10.3%. As a complete class, LOCAs contribute 31 per cent, transients 27 per cent and loss of offsite power (including station blackout) 29 per cent. ATWS is a very small contributor. These results are summarized in Table 3.4.1-5 and Figure 3.4.1-2.

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. The three measures are Fussell-Vesely, risk reduction worth and risk achievement worth.

The dominant basic events ranked in order by Fussell-Vesely and risk reduction measures are shown in Table 3.4.1-6. 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 all cut sets in which that basic event occurs compared with the sum of all cut sets. The risk reduction worth shows the ratio of the original core damage frequency to the reduced core damage frequency if the component was perfect or its failure probability is zero.

It should be noted that the ranking of events by the Fussell-Vesely measure and the risk reduction measure are identical so the highest ranked items for these two measures are discussed in the following paragraphs.

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 3.4.1-6, 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 for risk reduction 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 a need for manually initiated feed and bleed, 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 for 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 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 listed 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 3.4.1-7. 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.

3.4.1.2 Functional Failures Leading to Core Damage

In order to evaluate the relative contribution of the failure of various systems or functions, other than the initiating events, to the overall core damage frequency it is possible to group the core damage sequences by functional failure. The percentage contribution for the following functional failures are shown in Table 3.4.1-8.

- Failure of Emergency Switchgear Room cooling (T8, Hv)
- High Head/Low Head Recirculation (H1, H2)
- Recovery of offsite power (B)
- Auxiliary Feedwater (L, Lt)
- RC Pump Seal LOCA (T4, Slc)
- Operator cooldown and depressurization (O, Y)
- Failure of Safety Injection (D1, D2, D3)
- Failure of Bleed and Feed (P)

The sum of these events is greater than 100% as a number of the sequences contribute to more than one category of functional failure. For example some sequences consist of failure of Auxiliary Feedwater and failure of feed and bleed.

Failure of Safety Injection (HHSI-D1, Accumulators-D2 or LHSI-D3) contributes 42% to the core damage frequency and is dominated by D1. These sequences fall into three major groups: 1) failure of required injection during a LOCA (e.g., S2D1D3, S1D1Y or AD2), 2) failure during transient after AFW (L) fault (e.g., T1LD1 or T2LD1), and 3) failure in Hv Transfer event (e.g. T1Tr) following failure of operator cooldown (O) but following recovery of ESGR cooling (e.g., T1TrOD1, T3TrOD1 and T2ATrOD1).

Failure of operator cooldown and depressurization contributes 36% and involves three basic groups. For medium and small LOCAs and SGTR, when HHSI (D1) is available, O represents normal operator cooldown. If HHSI is not available, Y represents operator cooldown without HHSI. In these cases, failure to cooldown will prevent the use of Low Head Safety Injection pumps to maintain Reactor Coolant System inventory. Finally, for events with imminent loss of emergency power (T6, T8, and the initiators with consequential loss of ESGR cooling sequences, T1Hv, T2Hv, etc.), operator cooldown O is needed to avoid RC pump seal LOCA since RC pump seal cooling will also be lost with loss of emergency power.

Loss of Emergency Switchgear Room cooling contributes to 34% of the core damage frequency, through the T8 initiator and through the consequential loss and coincidental loss of ESGR cooling for several initiators. These latter events are the initial event in the T1Tr, T2Tr, T2ATr, T3Tr, T9ATr and T9BTr event trees. Since loss of ESGR cooling results in a loss of emergency power, core damage will occur through an RC pump seal LOCA if there is no cooldown, or through loss of core heat removal capability when the turbine driven AFW pump eventually fails (including SG overfill).

Sequences involving loss of Auxiliary Feedwater contribute 24% to the overall core damage frequency. One of the reasons for this is that six of the top seven initiating events require the operation of Auxiliary Feedwater following the initiator.

Failure of recirculation contributes 13% and failure of bleed and feed following loss of Auxiliary Feedwater contributes 1%.

As station blackout is only a 10% contributor to the overall core damage frequency, failure to recover offsite power only contributes 10%. The contribution from seal LOCAs is less than 1%.

3.4.1.3 Dominant Accident Sequences

The top 22 dominant accident sequences (core damage frequency greater than 1.0E-6/yr) are discussed in detail in this section. A complete list of the sequences and a list of the dominant cut sets for those sequences with frequency greater than 1.0E-7/yr are given in Appendix B. The sequences discussed in this section contribute approximately 75% of the core damage frequency and the sequences in the Appendix with frequency greater than 1.0E-7/yr contribute 96% of the core damage frequency.

Sequence S2D1D3

Frequency: 5.15E-6 Contribution: 7.6%

This sequence is initiated by a small break LOCA. The high head safety injection system fails to provide coolant make-up to the reactor. As blowdown through the break continues, the subcooling decreases and the core starts to heatup and eventually uncover. When the core outlet thermocouples reach 1200F, the operators are directed into functional restoration procedure 1-FR-C.1, which will direct the operators to perform core cooling recovery. In this sequence, the LHSI pumps fail to provide adequate flow to re-establish core cooling. All containment systems function during the core damage process, resulting in plant damage state #21. Dominant contributors to this sequence involve plugging of the RWST discharge isolation valve (1QSMV--PG-1QS38) and common cause failure of the check valves on the cold legs SI injection lines

(1SICKV-CC-838689). Each of these events result in failure of both HHSI and LHSI. Other typical cut sets involve the diversion of flow from the normal flow path through the failed open check valve at the discharge of a failed charging pump. Failure of the check valve (1SICKV-FC-15147) downstream of the RWST manual isolation valve appears in several cut sets among the top ten.

Sequence S2D1Y

Frequency: 1.19E-6 Contribution: 1.8%

This sequence is initiated by a small break LOCA. The high head safety injection system fails to provide coolant make-up to the reactor. As blowdown through the break continues, the subcooling decreases and the core starts to heatup and eventually uncover. When the core outlet thermocouples reach 1200°F, the operators are directed into functional restoration procedure 1-FR-C.1, which will direct the operators to perform core cooling recovery. In this sequence however, the operator fails to perform the actions necessary for core cooling recovery. The allowable time for operator action in this sequence is relatively small. The operator is not directed into 1-FR-C.1 until outlet thermocouples reach 1200°F, while the criteria for core damage is 2200°F cladding temperature. This presents a small window for successful operator action. This sequence is dominated by the failure of operator action. The failure causes for HHSI are the same as for sequence S2D1D3 described above.

Sequence T7OW

Frequency: 1.98E-6 Contribution: 2.9%

This is a steam generator tube rupture sequence where there is failure of early cooldown (but successful late cooldown) and failure of the RHR system to provide long term heat removal. The dominant events are the failure of the operator to initiate early cooldown (HEP-1E3-13); the failure of local recovery of the failed RHR valves (REC-1OP14-1); the failures of the A and B RHR heat exchangers (1RHHEX-CF-1RHE1A/B) and the fail closed during the mission of air operated valve controlling component cooling water flow to the B heat exchanger and seal cooling for both A and B RHR pumps.

Sequence S1D1Y

Frequency: 4.04E-6 Contribution: 5.9%

This is a medium LOCA sequence with the failure of high pressure injection. Injection from the accumulators and Auxiliary Feedwater

are available, but core cooling recovery fails because the initiator will result in containment insolation thereby preventing the opening of the steam dumps or SG PORVs. The contributors to the sequence are failure of the high pressure injection system check valve (SI-47) to open, common cause failure of MOVs 1115C and E to close, common cause failure of MOVs 1115 B and D to open, or common cause failure of the discharge MOVs to the cold legs MOV 1867C, 1867D and 1836.

Sequence S1OH2

Frequency: 2.45E-6 Contribution: 3.6%

This sequence is initiated by a medium LOCA with all the high pressure injection systems available. The operator does not cooldown and depressurize so that high head recirculation is required when the RWST is empty. Subsequent failure of high pressure recirculation leads to core damage. The dominant contributors to this sequence in order of importance are the operator's failure to cooldown due to lack of time to do so; the common cause failure of the low head pumps (1S1PSB-CC-FS1A1B); and failures associated with the SI valves SI-1860A and B, SI-1862A and B and SI-1863A and B.

Sequence S2H1

Frequency: 2.45E-6 Contribution: 3.6%

This is a small LOCA initiated sequence with success of the high pressure injection and Auxiliary Feedwater as well as cooldown and depressurization by the operator. Core damage occurs as the result of failure of low head recirculation. The dominant contributors to the failure of recirculation are common cause failure of the low head pumps (1SIPSB-CC-FS1A1B) and failures associated with the SI valves SI-1860A and B and SI-1862A and B.

Sequence AD2

Frequency: 2.12E-6 Contribution: 3.1%

This is a large LOCA sequence with failure of two of the accumulators to inject. Low head injection is successful as are the Quench Spray and recirculation systems. Cold leg and hot leg recirculation are both successful. Although the thermal hydraulic analysis indicates that some core damage will occur the contribution to containment failure and offsite release is negligible. The contributions to this sequence are the failure to open of any of the check valves or the plugging of either MOV in the two effective accumulator lines.

Sequence T1ALtBB1

Frequency: 2.99E-6 Contribution: 4.4%

In this sequence and in sequence T1ALtB, core damage occurs as the result of station blackout with failure of the Turbine Driven AFW pump and failure to recover offsite power leading to a failure of all core cooling functions. In addition, for this sequence there is failure to recover offsite power before core damage and before vessel failure; but power is recovered before containment failure. The dominant contributions following the loss of offsite power are failure to recover power within .6 hours and failure of the diesel generators to start and run. Failure of the turbine driven pump to start and the turbine driven pump in maintenance are also significant contributors. Although the turbine is expected to run for 24 hours (mission time) it is very likely that offsite power would be recovered by this time so the failure to run probability was taken as 12 hours and combined with a time averaged probability of failure to recover offsite power.

Failure to recover offsite power before vessel failure (B1) results in core damage and vessel failure. Subsequent recovery will allow containment heat removal to be established.

Sequence T7O02

Frequency: 2.98E-6 Contribution: 4.4%

This Steam Generator tube rupture sequence results in core damage as a result of failure of the operator to achieve early cooldown and depressurization followed by failure to achieve late cooldown. The early cooldown failure leads to overfilling of the steam generator which is assumed to lead to failure in the open position of a relief valve on the steam generator. Failure of late cooldown is failure to cooldown and depressurize the RCS to enable RHR to be used before the RWST is empty. The dominant contributors to this sequence are the failure of the operator to perform the appropriate steps in the procedures (HEP-1ECA3:1-16 and HEP-1E3-13); common cause failure of both pressurizer PORVs to open (1RCRV-CC-RCPORV) and loss of electric power due to spurious opening of 4160 V and 480 V breakers.

Sequence VxFm

Frequency: 1.52E-6 Contribution: 2.2%

This sequence consists of the interfacing system LOCA leading to discharge of the RCS outside containment in such a way that even if injection to the RCS is achieved in the short term no long term

make up or recirculation is possible. Approximately 95% of interfacing LOCAs fall into this category.

Sequence T7SGIW

Frequency: 1.10E-6 Contribution: 1.6%

This is a Steam Generator tube rupture sequence with failure to isolate the faulted Steam Generator followed by failure of the RHR system to provide long term heat removal. The significant events in this sequence in order of importance are the operator's failure to recover the failed valves in the RHR system by local action (REC-10P14:1); failure open of the decay heat release check valves from the intact steam generators (1MSCKV-FO-1MS58/19) and failure of the RHR heat exchangers (1RHHEX-LF-1RHE1A/B).

Sequence T1TrOD1

Frequency: 4.00E-6 Contribution: 5.9%

This sequence is initiated by a loss of offsite power. At least one diesel starts, but ESGR cooling fails due to chiller faults. Operator cooldown and depressurization fails due to the fact that the PORV block valve is closed to prevent leakage. ESGR cooling is restored prior to vessel failure, but safety injection is not successful after power restoration due to operator error.

Sequence T8LtRC1

Frequency: 3.17E-6 Contribution: 4.7%

This sequence is one of two in the top twenty two initiated by a loss of ESGR room cooling. The turbine driven AFW pump fails and the operators fail to recover ESGR cooling in 10 hours in the top three cut sets. Failure of the turbine driven AFW pump and loss of Unit 2 instrument air also leads to core damage. ESGR cooling is recovered before Containment failure and all Containment systems are success.

Sequence T1LD1

Frequency: 2.71E-6 Contribution: 4.0%

The emergency diesel generator(s) start after the initial loss of offsite power transient. The dominant cut set is failure of AFW due to depletion of the CST combined with the failure of feed and bleed due to operator failure to establish Safety Injection flow.

Subsequent cut sets consist of failure of the AFW pumps and failure to establish Safety Injection flow.

Sequence T8RC2

Frequency: 2.52E-6 Contribution: 3.7%

The sequence is initiated by a loss of switchgear room cooling followed by successful operation of the turbine driven AFW pump. The operator successfully performs the cooldown and depressurization procedure. The dominant contribution is the failure to restore ESGR cooling before core damage in 20 hours. Subsequent cut sets include failures of Unit 2 components needed for successful recovery of the Unit 1 ESGR using the standby equipment in the Unit 2 ESGR.

Sequence: T1TrOD1Qs

Frequency: 2.22E-6 Contribution: 3.3%

This sequence is similar to T1TrOD1, which is discussed above. However, in this sequence the Quench Spray System also fails. A review of the two sequences indicates that the cut sets are quite different. In this sequence a cut set has the additional requirement for quench spray failure. Since quench spray failure requires both trains to fail, faults to both electrical buses dominate the top cut sets for this sequence. These cut sets are various combinations of breaker failures and EDG maintenance terms.

Sequence: T3TrOD1

Frequency: 1.67E-6 Contribution: 2.5%

Following a trip with main feedwater available ESGR cooling is lost leading to loss of the motor driven AFW pumps. The turbine driven pump operates successfully, but the operators are unable to cooldown and depressurize. ESGR room cooling is restored within twenty hours, but the operators are unsuccessful in their attempts to initiate safety injection. The failure of instrument air to the pressurizer sprays is the dominant contributor to failure of the cooldown function. Safety injection valve failures are responsible for the failure of safety injection in the significant cut sets other than the top cut set.

Sequence: T3TrRC2Ch

Frequency: 1.57E-6 Contribution: 2.3%

This loss of main feedwater sequence has two functional failures. Failure to recover room cooling within twenty hours and failure of the containment heat removal function. The cut sets contain common cause failure of the screens at the intake to the service water pumps in the service water reservoir combined with failure of the auxiliary service water pumps.

Sequence: T9ATrLtRC1

Frequency: 1.53E-6 Contribution: 2.2%

Following reactor trip due to loss of the 4160 V H Bus, ESGR cooling is lost due to chiller maintenance. The operators successfully cooldown and depressurize, but the turbine driven AFW pump fails. ESGR room cooling is restored after core damage, but before containment failure. The dominant contributor to failure of ESGR cooling recovery prior to core damage is the operator error (HEP-OAP55-10hr).

Sequence: T1ALtB

Frequency: 1.41E-6 Contribution: 2.1%

This sequence is analogous to sequence T1ALtBB1 with regard to core damage. Also, all of the Containment systems are successful in both sequences. As a result, the cut sets are the same as those in sequence T1ALtB discussed above, except for the difference in the B1-function.

Sequence: T1ABB1

Frequency: 1.38E-6 Contribution: 2.0%

This sequence is a station blackout in which all mitigation actions are success. The turbine driven AFW pump operates and there is no seal LOCA. However, AC power is not recovered in time to avoid core damage due to loss of RCS inventory. Therefore, the dominant cut sets are various EDG failures combined with the non-recovery of offsite power failure probability.

Sequence: T1TrOH1

Frequency: 1.01E-6 Contribution: 1.5%

The initiating event for this sequence is a loss of offsite power followed by startup of at least one EDG. ESGR cooling is lost due primarily to chiller maintenance. Operator cooldown and depressurization is unsuccessful, but ESGR cooling is restored prior to core damage. High pressure injection is initiated, but fails during the recirculation phase, leading to core damage. The dominant HV fault is chiller maintenance. Failure to initiate RCS cooldown (HEP-1ES1:3) is one contributor to failure of the O-function. Other contributors include failure of the SG PORVs and the pressurizer PORVs.

3.4.1.4 Contribution to Containment Bypass and Failure

The development of the Containment response following core damage for the various plant damage states is fully described in Chapter 4. This includes a complete analysis of which sequences and plant damage states contribute to the various Containment failure modes, plant damage states and source terms. The significant sequences most likely to contribute to Containment failure or bypass are summarized in this section. The frequency of Containment failure or bypass is 1.77E-5 per year. As shown in Figure 3.4.1-3 sequences resulting in Containment bypass represent 12.6% of the core damage frequency, and those resulting in Containment failure 13.4% of the sequences. The sequences contributing 95% of the total to each of these failures and the plant damage states in which these sequences occur are discussed below. The plant damage state grouping criteria are shown in Figure 3.4.1-4.

Containment Bypass

The Containment bypass frequency is 9.1E-6. From Table 3.4.1-9, it can be seen that the plant damage states contributing 95% of this frequency are PDS 25 (77%) and PDS 24 (18%). Plant damage state 24 contains the V sequence, that is the Interfacing System LOCA and it contributes 100% of this plant damage state. Plant damage state 25 contains all Steam Generator tube rupture sequences resulting in core damage in which failure of Steam Generator isolation occurs.

Containment Failure

The Containment failure frequency is 8.6E-6. The plant damage states which contribute to the Containment failure frequency are listed in Table 3.4.1-9. Of these, the top nine plant damage states contribute to over 96% of the total. The sequences which contribute to each of these 9 plant damage states are listed in

Table 3.4.1-10. It will be noted that a single plant damage state does not always result in Containment failure. For example the frequency of plant damage state 14 in Figure 3.4.1-4 is 1.27E-5. The proportion of this plant damage state which leads to Containment failure shown in Table 3.4.1-9 is 9.1E-7. The way in which the assessment of whether or not Containment failure occurs for any given plant damage state is described in Section 4.

The dominant contribution to Containment failure (26%) is from sequence T1TrP21, which is initiated by a loss of offsite power followed by a loss of ESGR cooling. A seal LOCA occurs because operator cooldown is not successful. Although room cooling is restored before core damage, a random failure of the HHSI system causes core damage. In addition, the failure of the Quench Spray system leads to no Containment heat removal resulting in early Containment failure.

Sequence T3TrP03 contributes 18% to the Containment failure frequency and is the second largest contributor. This is a transient with subsequent loss of ESGR cooling leading to a station blackout. The turbine driven Auxiliary Feedwater pump is initially running, but fails at approximately 20 hours leading to core damage. Room cooling is recovered after vessel failure, but before Containment failure, so that Quench and Recirculation Sprays are available. However, long term heat removal is failed because the SW system is not available in this sequence. Containment failure occurs mostly in the late time frame.

The next significant contribution is from the T2ATrP03 sequence (7.3%). Except for the initiator, this sequence is identical to the sequence described above (T3TrP03).

The fourth most significant sequence, contributing approximately 7%, is T8P06, which is initiated by a loss of switchgear room cooling followed by failure to restore room cooling in time to restore decay heat removal or containment heat removal and therefore results in late failure of containment.

Each of the remaining sequences listed in Table 3.4.1-10 individually contribute less than 3% to the containment failure frequency.

3.4.1.5 Comparison of Results With and Without Recovery Actions

In developing the event trees and initial core damage sequences each function was developed based on the automatic initiation of systems combined with the procedure directed actions that the operator would be instructed to take in the applicable emergency procedures. For example following a small LOCA the operator is instructed to confirm that Injection Systems operate satisfactorily and that changeover to recirculation takes place when the RWST low

level occurs. Therefore in the sequence S2H1 which models a small LOCA followed by successful injection and cooldown and failure of low head recirculation only failures associated with the Low Head System were modeled in the initial event sequence quantification. All possible operator actions and events can not be modeled in the event trees and fault trees. After initial quantification, the dominant sequences were examined and further potential operator actions for these sequences identified. This is the process of recovery analysis. Each of the sequences were reviewed and where the failures in a given cut set would result in the operator going to another back up procedure and there was sufficient time to make the recovery action, a new basic event was added to the cut set and the sequence requantified. See Appendix B, Section 3 for details of the recovery analysis.

Thus, operator actions applicable to a sequence following a plant trip are represented by two sets of events in the final list of cut sets. Those actions modeled in the event tree are designated HEP, and those identified in later analysis are designated as a recovery action (REC). The distinction between the two sets of operator actions is determined by the size and complexity of the event tree and fault tree models, and does not represent a distinction between actual operator actions during an event.

The recovery actions with descriptions and quantified values are listed in Table 3.4.1-11. The sequences impacted by these recovery actions are listed in Table 3.4.1-12. The result of including these recovery actions lowered the core damage frequency from 2.2E-4 to 6.8E-5 a factor of 3.2 (or 69% reduction).

The dominant recovery actions (that is those which have the greatest impact on reducing the core damage frequency) are:

REC-1OP31:1 - Recovery of main feedwater.

REC-0AP10-24HR - Loss of electrical power recovered in 24 hours.

REC-1OP14:1 - Local opening of RHR valves to recover the RHR following a steam generator tube rupture.

REC-1MOP6:70 - Recovery of 1H emergency from maintenance in 12 hours.

Certain sequences are below 1E-7 because of inclusion of operator actions (HEP) following initiating events and equipment failure. In order to ensure that no sequences are artificially eliminated due to low HEPs, these sequences were examined for the impact of HEP values or core damage frequency. The sequences which fall into this category and the significant HEPs in each sequence are listed in Table 3.3.3-3.

All of the operator actions included in HEPs are based on procedures and therefore would be performed in the normal course of events following the failure of specific components. In fact, for the North Anna IPE, all operator actions are identified by the procedure number and procedure step which specify the appropriate action. The revised sequence frequency is based on the extreme case of assuming that the operator fails to take any action (HEP = 1.0) and therefore represents the upper bound for this sequence frequency for these sequences. It can be seen that there are two sequences which would be raised above 1.0E-5 and thirteen sequences which would be raised to between 1.0E-6 and 1.0E-5. The HEPs associated with these sequences are shown in Table 3.3.3-3.

Table 3.3.3-2 lists all sequences and shows their CDF with the HEPs set equal to 1.0 and to the values of Table 3.3.3-1. A sensitivity analysis indicates that if all of the HEPs are set equal to 1.0 the CDF will change from 6.8E-5 to 2.2E-2/year. The Type A HEPs above would change the CDF to 2.3E-3 and the Type C HEPs alone would change the CDF to 1.7E-2. If all HEPs were set equal to 1E-15 the CDF would change from 6.8E-5 to 3.5E-5/year.

3.4.1.6 Sensitivity Analysis

Seven initiating events namely small LOCA, steam generator tube rupture, loss of offsite power, station blackout, intermediate LOCA and loss of switchgear room cooling contribute 74% of the core damage frequency. The core damage sequences for each of these initiators were reviewed to establish if there were any areas where the success criteria were based on assumptions which did not have a solid basis on past experience (data) or sound thermal hydraulic analysis. Similarly, the sensitivity of the results to the modeling of human errors and the use of potential cross ties was also examined. The results of the sensitivity analyses performed are summarized in the following paragraphs.

The station blackout sequences were quantified based on the following assumptions in three key areas:

1. The modeling of the seal LOCA assumes that a seal LOCA will occur between 1 and 10 hours after the loss of seal injection and cooling. This is based on the Westinghouse analysis of seal LOCA and is fully described in Appendix B.

2. It is assumed that the auxiliary feedwater pump which does not require dc power to continue to run, will run until the condensate storage tank is empty.

3. Station blackout will occur following loss of offsite power if the diesel generators fail to start or fail while running before recovery of offsite power. The mission time for all sequences is nominally 24 hours from the time of the

initiating event, after which the analysis transfers to the analysis of shutdown risk. As the diesel generators fail to start and fail to run appear together as failures and power recovery requirements are based on failure of the diesel generators to start at time zero this gives a conservative result. As in NUREG/CR-4550 for Surry the diesel run time is set to 6 hours in order to give a realistic value for the failure to run and failure to recover offsite power.

The following sensitivity analyses were performed to assess the impact of alternative assumptions in each of these areas:

Assumption	SBO Contribution to CDF	Delta	CDF	% Change
1. Base Case	8.0E-6		6.8E-5	
2. Seal LOCA occurs at 1 hour	6.8E-5	5.4E-5	1.3E-4	91%
3. No Seal LOCA occurs	7.8E-6	-2.0E-7	6.8E-5	---
4. AFW fails at the time dc power fails (4 hrs.)	1.7E-5	9.0E-6	7.1E-5	4%
5. Diesel generators modeled as failure to run for 24 hrs.	3.3E-5*	2.1E-5	9.8E-5	44%

*Impacts station blackout and loss of offsite power

The core damage frequency from station blackout ranges from 7.6E-6 when there is no seal LOCA to 7.0E-5 when seal LOCA is assumed to occur one hour after station blackout. Similarly if auxiliary feedwater is assumed to fail on loss of dc power the core damage frequency station blackout is increased by 9.0E-6 to 1.8E-5. Assumpting that the diesel generators fail to run for 24 hours is lumped together and treated as failure at time zero for the restoration of offsite power, results in an increase in loss of offsite power frequency to 2.1E-5, a factor of four higher. It is therefore important to accurately model the occurrence of the seal LOCA and the failure to run of the diesel generators.

subsequent failure of Low Head Safety Injection (LHSI) after successful core cooling recovery. The success criterion for LHSI is 2/2 trains successful. If this success criterion were changed to 1/2 LHSI pumps, the sequence frequency would be reduced by about a factor of two.

The independent review team identified a sequence with potentially conservative success criteria. The sequence is AD2, a large LOCA followed by failure of 2/3 accumulators to inject. A MAAP run was made to look at the results of a large LOCA with only one accumulator successful and with minimum safeguards otherwise. The results show very little difference between the case with two accumulators injecting and one accumulator injecting.

Finally the sensitivity of the overall results to the values used for human reliability data were examined. The inclusion of recovery actions is discussed in Section 3.4.1.5, where it is shown that the core damage frequency is reduced from 2.2E-4 to 6.8E-5 by their inclusion.

The current values for all human reliability basic events have been based on a systematic approach to the quantification of the human actions. If it is assumed that there is a bias in this analysis such that all the events were assessed to be too low then the core damage frequency would be higher. To assess overall sensitivity to the HEP all the operator errors were increased to 1.0. The resultant core damage frequency increased from 6.8E-5 to 3.1E-4, a factor of 5 increase. This indicates that operator actions do play a significant part in determining the core damage frequency and emphasizes the importance of the instructions and guidance in the emergency procedures. However if the operator were perfect (i.e., probability of failure is zero) the core damage frequency would only be reduced to 4.0E-5/year indicating that the current performance as evaluated in the PRA is very good.

3.4.2 Vulnerability Screening

A concise definition of a vulnerability is not given in the documentation associated with the performance and reporting of the IPE. In the response to questions in Appendix C to the Submittal Guidance Document (NRC, 1989), mention is made of expanding sequences that are above the screening criteria in order to determine if a weakness exists. Such a weakness may be defined as a vulnerability and a fix proposed. In another response it is suggested that a vulnerability is in fact an outlier. In this study Importance and Sensitivity measures have been used to determine the most significant contributors to the core damage frequency, Containment Systems performance, fission product source term frequencies, and decay heat removal functions.

The importance measure identifies the contribution of a given failure to the core damage frequency, etc., with the current failure probabilities and also the impact on core damage frequency of making that component perfect. To be a vulnerability, such a failure (component fault or human error) must be significantly greater than any others, i.e., contribute more than ten percent to overall core damage frequency or be a factor of three greater than the next highest similar event.

3.4.2.1 Internal Event Core Damage Vulnerabilities

The contribution to core damage from functional failures in Table 3.4.1-8 shows that there are no outliers. Loss of the safety injection function contributes 42% compared with failure to cooldown and depressurize which contributes 36%, and failure of Emergency Switchgear Room cooling at 34%. The highest contribution comes from failure associated with safety injection and therefore it is appropriate to determine which failures in this system are contributing most to the core damage frequency. The importance analysis in Table 3.4.1-6 lists all components in order of their contribution to core damage at the current core damage frequency and also shows the impact on core damage frequency if this component was perfect. This could be considered the same as considerably improving the components reliability or having an alternative path. The highest ranked events associated with components in the plant are the failure to start and failure to run of the 1H Emergency Diesel Generator, and the failure to run of the turbine driven Auxiliary Feedwater pump. The EDG 1H fail to start and fail to run events can be combined with the EDG 1H unavailability due to maintenance event to yield a combined contribution to core damage frequency of 23%. The relative importance of the EDG 1H is a product of the total Loss of Offsite Power event (T1) and the partial Loss of Offsite Power or loss of a switchyard bus or Reserve Station transformer events, T9A. Note that EDG 1J has a similar importance of lower contribution to CDF, due to differences in powering support systems (e.g., 1H supplies 2 ESGR chiller trains while 1J supplies only 1 ESGR chiller train). The turbine driven AFW pump contributes 16% to CDF through two fail to run events (12 hour and 24 hour missions) and one fail to start event. The turbine driven AFW pump is the only source of AFW for Station Blackout (T1A) and for the loss of Emergency Power from T6, T8 or the Hv transfer events T1Tr, T2Tr, T2ATr, T3Tr, T9ATr and T9BTr. In view of the relatively low contribution of these individual failures (<10%) they are not considered a vulnerability.

Failure of injection is the highest ranked functional failure, and contributes 42% to the core damage frequency. A review of the importance analysis in Table 3.4.1-6 shows that there are no individual events associated with component failure above 4% with the highest contribution coming from 1-CH-254 failing open at 3.96%. All the other failures are in fact below 3 per cent. Thus

there are no individual vulnerabilities associated with these systems. However, the human action of highest worth, HEP-1FRH:1-11, involves manual initiation of High Head Safety Injection. This HEP appears in T1 and T1A sequences involving loss of AFW and the need for manually initiated feed and bleed, and in several Hv transfer sequences involving restoration of Emergency Power before core damage, but where HHSI is required for a RC Pump Seal LOCA. Although the importance of this HEP is high at 12% CDF, the existence of other HEPs with 7% and 6% CDF importance confirm that the event is not an outlier, and it is not considered a vulnerability. Also, although this human action to manually initiate HHSI is felt to be important, the split between LOOP and other transient initiators indicates that two human action models would be appropriate, yielding the same combined importance but with apportionment between the two transient types.

The room cooling requirement for the Emergency Switchgear Room is an important aspect of design as failure of room cooling will lead to failure of all emergency power. This particular event has been analyzed in some detail in order to establish the range of cooling requirements for normal and off normal plant operation as well as to examine and evaluate component failure and maintenance history. Individual components do not contribute over 5% CDF and human action contributes 8% CDF or less. Note that ESGR cooling system support systems (1H and 1J Emergency Power, Service Water) have some noteworthy importance above 3% CDF, but it is difficult to separate the proportion of this importance attributable to ESGR chiller support. Although the group importance is high (upwards of 30% from Table 3.4.1-8), no individual component or human action contributes above 10%, so they are not considered a vulnerability.

3.4.2.2 Flooding Core Damage Vulnerabilities

The total core damage frequency from internal flooding is approximately 3.6E-6/year or 5% of the total core damage frequency. As such there are no plant vulnerabilities associated with internal flooding. The dominant contribution comes from service water floods in the Auxiliary Building. A small portion of these floods are the result of failure of pipes which require manual isolation to prevent the syphoning of water from the pond.

3.4.2.3 Containment Vulnerabilities

The North Anna Containment Structure and mitigation systems are robust with respect to their capability to withstand the challenges resulting from severe accidents. The Containment capacity is such that the threat of early Containment failure is very low. Because of the redundancy and diversity of the Containment and Recirculation Spray Systems, failure of the sprays and of CHR is

very unlikely (with the exception of station blackout sequences). Containment vulnerabilities are discussed in Section 4.4.

Since North Anna is a subatmospheric Containment the probability of having a large unisolated leak pathway prior to a core damage initiator is virtually impossible. No Containment System design flaws, operational errors, or phenomenological processes were identified which lead to a significant probability of early Containment failure or leakages or a large atmospheric source term. The dominant accident sequences from the standpoint of risk are associated with Containment bypass (particularly SGTR). These sequences are largely not impacted by Containment structure strengths or weaknesses.

3.4.2.4 Post Quantification Enhancements

As mentioned in Section 2.0, the IPE project is subject to a QA plan which requires internal review of all analysis files. When preparing to close out the accident sequence analysis file, it was discovered that the T6 and T8 restoration functions could be modeled better. We reviewed the impact of the changes and found that it was minimal on the T8 tree and the transfer trees (T1Tr, T2ATr, etc.). The impact of the SW restoration functions on the core damage frequency was negligible. However, the contribution of loss of SW initiator to core damage increased substantially. Fortunately, the contribution is on the order of 5E-9/year in the T6 tree presented in the report. With a better modeling of the SW restoration functions, the contribution increases to about 5E-7/year.

It should be noted that none of the top sequences are impacted. In fact, the list of sequences with frequency greater than 1E-7/year is not impacted except by one sequence at the bottom of the range. It is also very important to note that the model presented in the IPE report provides sufficient accuracy to determine the procedure modifications that were required. These modifications are discussed in Chapter 6.

3.4.3 Decay Heat Removal Evaluation

The objectives of Task Action Plan A-45 are to evaluate the safety adequacy of Decay Heat Removal (DHR) Systems in existing light water reactor nuclear power plants and to assess the value and impact (benefit-cost) of alternative measures for improving the overall reliability of the DHR function if required. A program was developed by the NRC and case studies performed to investigate this issue. One of the studies was that performed by Sandia National Laboratories for a typical Westinghouse 3 loop PWR (Sanders et al., 1986).

In order to understand the adequacy of the results of the IPE performed for North Anna Units 1 and 2 for resolution of this issue it is appropriate to compare the approach used by Sanders with that used in the IPE.

The initial steps in assessing the adequacy of DHR in the A-45 program were to 1) characterize the units in terms of their physical parameters, i.e., number and location of safety pumps, number of redundant emergency power trains etc.; and 2) develop a set of qualitative screening criteria against which the plant characteristics could be compared. This set of qualitative screening questions was based upon a thorough review of existing guidance such as the Standard Review Plan, the various Regulatory Guides, previous Probabilistic Risk Assessments, and special topical studies such as the Auxiliary Feedwater studies. The intent was to establish a set of questions which would reveal potential deficiencies in DHR capabilities for both Design Basis Events and for beyond Design Basis situations. This screening process resulted in the identification of plants for the DHR study.

The analysis procedure used to perform the study was a shortened Probabilistic Risk Assessment approach. The approach that has been used to perform the North Anna IPE has been a full PRA, including the analysis of Containment Systems and Containment performance analysis. Thus, the performance of the North Anna IPE meets the performance requirements for A-45 and exceeds that of earlier studies.

In performing the A-45 studies SNL limited themselves to the number of initiating events and systems which were considered. In the North Anna IPE a much greater number of initiating events and all front line and support systems, which can affect the progression of events following these initiators, have been considered. A comparison of initiating events is shown in Table 3.4.3-1 and of front line and support systems in Table 3.4.3-2.

The delineation of the accident sequences, system analysis, and quantification are fully described in the proceeding Sections 3.1 through 3.4.1. The identification and ranking of the plant vulnerabilities are described in Section 3.4.2. However, the concern in issue A-45 is to identify the specific vulnerabilities associated with sequences identified as potentially leading to core damage if feed and bleed, secondary blowdown, recirculation, and Residual Heat Removal are failed.

The relative contribution to core damage from failure of the functions which support decay heat removal are shown in Table 3.4.3-3. Failure of secondary heat removal includes Main and Auxiliary Feedwater and steam relief and Residual Heat Removal; failure of feed and bleed including Charging Pumps and the PORVs; failure to cooldown includes Steam Generator depressurization for RCS cooldown; failure of recirculation includes failure of high and

low head recirculation and failure of injection includes failure of high pressure and low pressure injection.

The overall contribution of individual basic events to the core damage frequency is shown in Table 3.4.1-6. The highest ranked component associated with this decay heat removal function is failure to start of the turbine driven feedwater pump (1FWTRB-FR-12HP2) with a Fussell-Vesely importance of .07 and a risk reduction worth of 1.08. The second highest contributor is failure to run of the turbine driven pump (1FWTRB-FS-1FWP2) with an importance of .05 and a risk reduction worth of 1.05. The remaining contributors from individual components are each less than five per cent.

It is considered that these results show that there are no particular vulnerabilities of the North Anna Unit 1 and 2 systems that are used to perform decay heat removal as any failures contribute 10% or less.

There are a number of reasons why this is the case. Firstly, North Anna has three motor driven Main Feedwater pumps. Secondly, procedures are in place for the use of feed and bleed in the event of failures of all secondary heat removal capability. Procedures are also in place for the use of emergency cooldown in the event of failure of charging flow following a small LOCA. Finally the ability to be able to cross connect charging flow can be used to considerably extend the time available to establish recirculation for small LOCA and Steam Generator tube rupture events.

3.4.4 USI and GSI Screening

There are two issues in addition to A-45 which have been considered during the performance of the IPE and which are currently unresolved at the North Anna plant. These are:

A-17	System Interactions
GI-23	RCP Seal Failure.

The resolution of each of these issues is based on the following deliberations:

1. The ability of the methodology to identify vulnerabilities associated with the issue being addressed.

2. The contribution of the issue to core damage frequency or unusually poor Containment performance, including sources of uncertainty.

3. The technical basis for resolving the issue.

3.4.4.1 Systems Interactions Due to Internal Flooding (USI A-17)

Generic letter 89-19 informs licensees of the final resolution of USI A-17 "Systems Interactions in Nuclear Power Plants." The staff has identified actions to be taken by the NRC to resolve USI A-17 and has made the judgment that these actions, together with other ongoing estimates should reduce the risk from adverse system interactions.

As part of the resolution the staff has identified that water intrusion and flooding of equipment from internal plant sources may result in a risk significant adverse system interaction. The appendices to NUREG-1174 (NRC, 1988) provide insights regarding plant vulnerabilities to flooding and water intrusion from internal plant sources. The licensee is expected to take action to review their plant in the light of this NUREG and to continue to review information on events at operating nuclear power plants.

The Virginia Power response to these requirements has been to perform an extensive probabilistic internal flooding analysis as part of this IPE submittal. The results and findings of this analysis are reported in Section 3.3.7 and Appendix E.

The internal plant flooding analysis encompasses the effects from the accumulation, spraying or dripping of fluids arising from the rupture cracking or incorrect operation of components within the plant. All potential flood sources, including such effects as the inadvertent initiation of fire fighting systems have been included in the analysis. The potential for these floods resulting in a plant trip and at the same time resulting in the common cause failure of one or more systems used to prevent core damage or mitigate the fission product release has been considered. Flood progression from one area to another has been examined and the evaluation of the plant damage state frequencies includes the random failures of equipment not affected by the flood. The response of the operator to each flood and his ability to terminate the flood or take other actions necessary to prevent core damage has been included in the study.

The above analysis resulted in the conclusion that the core damage frequency from internal flooding events is low (~5% of the total CDF) and therefore does not represent a vulnerability at North Anna. The performance of this study and the extensive review and discussions of the results have resulted in an awareness of internal flooding and its potential effects both at the plant and in the general office engineering staff. Based on the studies performed and the actions taken during the course of the study, it is considered that the internal flooding issue identified in USI A-17 is resolved for the North Anna plant.

3.4.4.2 RCP Seal Failure (GI-23)

This issue has recently been the subject of significant activity with the issue of Generic Letter 91-07, "GI-23 Reactor Coolant Pump Seal Failure" and its possible effect on station blackout, (NRC, 1991a), the Draft Regulatory Guide DG-1008 (NRC, 1991b), and the report Regulatory Analysis for Generic Issue 23: Reactor Coolant Pump Seal Failure (NRC, 1991c).

Reactor Coolant Pump (RCP) seal failure results in loss of integrity of the Primary Coolant System pressure boundary. If the failure is large enough, it would result in leak rates in excess of the normal capacity of the Reactor Coolant make up system and is potentially a small loss of coolant accident (LOCA). In order to address this issue, the potential for RCP seal LOCAs leading to small LOCAs has been considered in this study. Three sets of conditions have been considered:

1. Random failure of one RCP seal due to either material faults or local failure of seal injection or thermal barrier cooling.

2. Failure of multiple seals due to loss of the thermal barrier cooling and Seal Injection Systems (non-loss of electric power).

3. Failures due to loss of all cooling and injection during station blackout.

The first two events can occur during normal operation and the third event occurs during loss of offsite power. Therefore, the first two events were analyzed by the inclusion of random failures in the small LOCA event tree and by the development of a special event tree for the failure of seal injection and thermal barrier cooling. The third event is considered in detail in the development of the loss of offsite power event trees.

As discussed earlier, there are a number of specific design features at North Anna which ensure that the contribution to core damage frequency from RCP seal induced LOCAs is small. As identified in this study, these plant features significantly reduce the impact of the random occurrence of a seal LOCA.

Firstly, the design allows cross connection of both seal injection and thermal barrier cooling. Therefore, the contribution to core damage from random failure of these systems is 7.7E-7/yr which is approximately 1% the overall core damage frequency from internal events.

In the case of random failures leading to small LOCAs, the ability to cross tie the High Pressure Safety Injection Systems, ensures that the contribution to core damage from all small LOCAs is only

20%. The initiating frequency used for small LOCAs in this study is 2.1E-2/yr and that in Appendix A in NUREG-1401 is 1.3E-2/yr. If the latter frequency is used for the proportion of small LOCAs which are seal failure at North Anna, the contribution to core damage frequencies would be 12% and the frequency 7.9E-6/yr compared with the estimated values in NUREG-1401 of 2.79E-5/yr for the base case, where recommendations 1 and 2 of the NUREG are not performed, and 1.06E-5/yr if the recommendations are performed.

The current design at North Anna ensures that the core damage frequency from random failures during normal operation is well below the postulated value in NUREG-1401 of 1.12E-5. In addition, action has been taken as the result of performance of the IPE to ensure that adequate instructions are available to provide backup seal injection and thermal barrier cooling by adding steps to ECA-0.0, which refer the operators to the seal cooling abnormal procedure (AP-33.2 NAPS).

It is considered that based on the current design at North Anna and the actions taken following the IPE, GI-23 is resolved for normal operation and no further actions are required.

In the event of station blackout, seal injection and RCP thermal barrier cooling is lost. Thus, there are two competing conditions which will lead to core damage. Either the RCP seals will fail leading to a LOCA and core damage before AC power is restored, or the Turbine-Driven Auxiliary Feed Pump will fail due to the loss of the DC battery or depletion of condensate, leading to loss of heat sink and core damage, if AC power is not restored. The contribution of each of these to core damage is dependent upon the time at which the seal fails (given no restoration of AC power). The seal failure model used in this study is very similar to that described in NUREG-1401, based on the work done by AECL and the Westinghouse analysis of seal failure (this is discussed in Appendix B). This resulted in an assessed core damage frequency for station blackout sequence of 6.2E-6/yr. A sensitivity analysis (Section 3.4.1.6) was performed to determine the impact of seal LOCA assumptions on the core damage frequency. This showed that if the seals were considered perfect the core damage frequency was reduced from 6.2E-6 to 6.0E-6, a reduction of 2.0E-6/yr. This contribution to core damage from seal failure is very small because of the ability to cross connect seal cooling from the other unit when the diesel generators are running. On the other hand, if a seal LOCA is postulated to occur one hour after station blackout, the core damage frequency would increase to 4.0E-5. In other words, the contribution would increase by 3.4E-5, almost a factor of seven.

In view of the uncertainty of seal performance under these conditions, Virginia Power has decided to install equipment which will ensure the provision of seal injection under these off normal

conditions, as suggested in the draft regulatory guide. This action satisfactorily resolves the off-normal requirements of GI-23 (Virginia Power, 1991).

3.5 REFERENCES

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TABLE 3.1.1-1
SOURCES OF INITIATING EVENT INFORMATION

1. LERs for North Anna Unit 1 and 2 for 1986 - 1990.
2. NUREG/CR-3862, "Development of Transient Initiating Event Frequencies for use in PRA, May 1988".
3. NUREG/CR-4550, Volume 3.
4. Review of Support System Drawings for North Anna Units 1 and 2.
5. North Anna monthly operating reports 1980 - 1990.
6. Review of past PRAs on Westinghouse PWRs.

TABLE 3.1.1-2
LIST OF INITIATING EVENTS

Abbreviations	Descriptions
T1	Loss of Offsite Power
T2	Transients with non-recoverable Loss of Main Feedwater
T2A	Transients with recoverable loss of Main Feedwater following FW Isolation
T3	Transients with Main Feedwater initially available
T4	Loss of RC Pump Seal Injection and Thermal Barrier Cooling
T5A	Non-recoverable Loss of DC Bus 1-I
T5B	Non-recoverable Loss of DC Bus 1-III
T6	Loss of Service Water System
T7	Steam Generator Tube Rupture
T8	Loss of Emergency Switchgear Room Cooling
T9A	Loss of 4160 V Bus 1H
T9B	Loss of 4160 V Bus 1J
A	Large LOCA 6" - 31"
S1	Medium LOCA 2" - 6"
S2	Small LOCA 3/8" - 2"
VX	Interfacing System LOCA
RX	Reactor Vessel Rupture

TABLE 3.1.1-3
CLASSIFICATION OF LOCA CATEGORIES

LOCA Classification		Definition
1. Inside Contain-ment	a. Large	> 6"
	b. Medium	2" ≤ ID ≤ 6"
	c. Small	3/8" ≤ ID ≤ 2"
	d. Very Small	RCP Seal
	e. Consequential	1. Failed Open RCS PORV or SRV 2. RCP seal failure following loss of seal injection and thermal barrier cooling (small) 3. SGTR following steam leak inside containment (small) 4. Reactor Vessel/ Pipe failure following ATWS overpressure event (greater than large) 5. Reactor Vessel/ Pipe failure following pressurized thermal shock (greater than large)
2. Outside Contain-ment V Sequence	a. Valve Failures	Interfacing System LOCA
	b. High Pressure Pipe Ruptures	
	c. Consequential	Failed Open SG ADV or SRV After SG Tube Rupture

TABLE 3.1.1-4
LOCA COMPONENTS AND RANGE OF EQUIVALENT BREAK SIZES

COMPONENT	SIZE RANGE (INCHES)
RCS & Connected Piping	3/8 - 31
SG Tubes*	0.875
SG Inlet & Outlet Nozzles	31
SG Manways (RCS Side)	16
Pressurizer (PZR) Sprays	4
PZR Relief & Safety Valves **	4

* Cause LOCA outside containment - considered as a separate initiating event group.

** Not necessarily equal to the valve throat area.

TABLE 3.1.1-5
PRESSURE RATING OF PIPES

Piping Class	Maximum Operating Conditions Temperature (°F)/Pressure (PSI)
152	350°F/195 psi, 300°F/210 psi, 250°F/225 psi, 200°F/240 psi, 100°F/275 psi
153	350°F/195 psi, 300°F/210 psi, 250°F/225 psi, 200°F/240 psi, 100°F/275 psi
1502	200°F/2900 psi, 300°F/2830 psi, 400°F/2780 psi, 500°F/2740 psi, 600°F/2730 psi, 650°F/2720 psi
1503	200°F/3200 psi, 400°F/2500 psi, 650°F/2100 psi

TABLE 3.1.1-6
NUREG/CR-4550 INITIATING EVENT GROUPS

Abbreviation	Description
T1	Loss of Offsite Power
T2	Transients with Loss of MFW
T3	Transients with MFW Initially Available
T5A	Non-Recoverable Loss of DC Bus A
T5B	Non-Recoverable Loss of DC Bus B
T7	Steam Generator Tube Rupture
A	Large LOCA, 6" - 31"
S1	Medium LOCA, 2" - 6"
S2	Small LOCA, 1/2" - 2"
S3	Very small LOCA, less than 1/2"
V	Interfacing System LOCA

TABLE 3.1.1-7
TRANSIENT INITIATING EVENT T2 SUB-GROUP

Initiating Event Group	Representative Initiators	Comments
T2 Non-recoverable Loss of Main FW	Failure of Main Feedwater. Loss of Instrument Air (IA) system. Main Feed-water Line Break.	Includes MFW failures (i.e., disabled pumps), failure in hotwell FW flow path, and insufficient condensate inventory, loss of IA.
T2A Recoverable Loss of MFW	Steam Generator Hi Hi Level. Inadvertent SI. Main Steamline Break.	FW recovered by start of 1 MFW pump and flow through 1 FRV or bypass valve.
	Lo Tavg coincident with Reactor Trip.	FW recovered by flow through 1 FW bypass valve and 1 MFW pump maintained on recirculation.

TABLE 3.1.1-8
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T2A, T3	1	**Loss of RCS Flow (1 loop)** - An inadvertent hardware or human error interrupts the flow in one loop of the reactor coolant system. **SI***
T3	2	**Uncontrolled Rod Withdrawal** - One or more control rods are withdrawn inadvertently.
T3*	3	**CRDM Problems and/or Rod Drop** - Failures in the control rod drive mechanism (CRDM) occur that lead to out-of-tolerance conditions in the primary system. The transient may include dropping of one or more control rods into the core as part of the CRDM failure. [Assumes no turbine runback-use category 33 with turbine runback].
T3*	4	**Leakage from Control Rods** - Primary system leakage around the control rod drive mechanism is excessive and reactor shutdown is required.
T3*	5	**Leakage in Primary System** - Primary system leakage through various piping components is excessive and reactor shutdown is required. This transient does not include: No. 4 - Leakage from control rods No. 7 - Pressurizer leakage No. 26 - Steam generator leakage
T3	6	**Low Pressurizer Pressure** - Pressurizer pressure falls below the lower operating limit.
T3*	7	**Pressurizer Leakage** - Pressurizer components allow excessive primary system leakage and reactor shutdown is required.

TABLE 3.1.1-8 (Continued)
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T3	8	**High Pressurizer Pressure** - Pressurizer pressure climbs above the upper operating limit.
T2A	9	**Inadvertent Safety Injection Signal** - Hardware or operator error initiates a safety injection.
T3	10	**Containment Pressure Problems** - Hardware or operator error results in containment pressure exceeding limits.
T3	11	**CVCS Malfunction - Boron Dilution** - Hardware or operator error results in a CVCS malfunctions such that reactor power is affected.
T3	12	**Pressure/Temperature/Power Imbalance - Rod Position Error** - Poor control rod positioning from mechanical or operator error causes a scram based on a pressure, temperature, or power imbalance.
Not Applicable*	13	**Startup of Inactive Coolant Pump** - An inactive coolant pump is started at an improper power and flow condition. [Unit operation with inactive coolant loop is precluded by Technical specifications.]
T3	14	**Total Loss of RCS Flow** - A hardware or operator error causes a loss of reactor coolant system flow.
T3*	15	**Loss or Reduction in Feedwater Flow (1 loop)** - One feedwater pump trips or another occurrence results in an overall decrease in feedwater flow.

TABLE 3.1.1-8 (Continued)
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T2, T2A	16	**Total Loss of Feedwater Flow (all loops)** - A simultaneous loss of all main feedwater occurs, excluding that due to loss of all offsite power (Category 35).
T2A	17	**Full or Partial Closure of MSIV (1 loop)** -One main steam isolation valve (MSIV) closes, the rest remaining open, or partial closure of one or more MSIV occurs. [Can result in Steam Generator Lo-Lo Level reactor trip.]
T3	18	**Closure of all MSIV** - One of various steam line or nuclear system malfunctions requires termination o f steam flow from the vessel. The closure of one MSIV may cause an immediate closure of all other MSIVs; this occurrence is also included in this transient definition. However, any closure that is the result of another initiator is not included. [Can result in Steam Generator Lo-Lo Level reactor trip.]
T3*	19	**Increase in Feedwater Flow (1 loop)** - An increase in feedwater flow occurs in one loop. [Steam Generator Hi-Hi Level Turbine Trip/Reactor Trip causes Feedwater Isolation.]
T3*	20	**Increase in Feedwater Flow (All Loops)** - An increase in feedwater flow occurs in one loop.[Steam Generator Hi-Hi Level Turbine Trip/Reactor Trip causes Feedwater Isolation.]

TABLE 3.1.1-8 (Continued)
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T2A*, T3*,	21	**Feedwater Flow Instability - Operator Error** - Feedwater is being controlled manually, usually during startup or shutdown, and excessive or insufficient feedwater flow occurs.
T2A, T3*	22	**Feedwater Flow Instability - Miscellaneous Mechanical Causes** - Excessive or insufficient feedwater flow results form hardware failures in the feedwater system.
T2, T3	23	**Loss of Condensate Pumps (1 loop)** - One condensate pump fails, reducing feedwater flow. [Can result in Feedwater pump trip on low suction pressure]
T2	24	**Loss of Condensate Pumps (all loops)** - All condensate pumps fail, causing a loss of feedwater flow.
T3	25	**Loss of Condenser Vacuum** - Either a complete loss or decrease in condenser vacuum results from hardware or human error.Can use atmospheric steam dump without condenser, Feedwater pumps will not trip as long as hotwell inventory lasts.
T3*, T7	26	**Steam Generator Leakage** - Excessive primary system to secondary leakage occurs in the steam generator.
T3	27	**Condenser Leakage** - Excessive secondary system leakage occurs in the condenser.[a] [Feedwater heater level Turbine Trip].

TABLE 3.1.1-8 (Continued)
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T3	28	**Miscellaneous Leakage in Secondary System** - Excessive leakage occurs in the secondary system other than in the condenser (see Category 27).
T2A, T3*	29	**Sudden Opening of Steam Relief Valves** - A secondary system steam relief valve opens inadvertently, causing an unacceptably low pressure in the secondary system. [Can result in Feedwater Isolation from SI or Steam Generator Hi-Hi Level Turbine Trip/Reactor Trip.]
T2A*, T3*	30	**Loss of Circulating Water** - Circulating water is not available to the plant. [Can result in loss of condenser vacuum - see Category 25.]
T3	31	**Loss of Component Cooling** - Excessive temperature of critical components is a result of a loss or decrease in component cooling water flow.
T3	32	**Loss of Service Water System** - The service water system fails to perform its function.
T2A*, T3*	33	**Turbine Trip, Throttle Valve Closure, EHC Problems** - A turbine trip occurs; or turbine problems occur which in effect decrease steam flow to the turbine, causing a rapid change in the amount of energy removed from the primary system. [Turbine runback can result in Steam Generator Hi-Hi Level or Steam Generator Lo-Lo Level, causing Feedwater Isolation.]

TABLE 3.1.1-8 (Continued)
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T3*	34	**Generator Trip or Generator Caused Faults** - The generator is tripped due to electrical grid disturbances or generator faults.
T1	35	**Loss of All Offsite Power** - All power to the plant from external sources (the grid or a dedicated transmission line to another plant) is lost.
T3	36	**Pressurizer Spray Failure** - The pressurizer spray system spuriously actuates or fails upon demand.
T3	37	**Loss of Power to Necessary Plant Systems** -Power is lost to a component or group of components such that plant shutdown is necessary. It does not include loss of power to those components whose failure causes another defined transient to occur.
T3	38	**Spurious Trips - Cause Unknown** - A scram occurs and no out-of-tolerance condition can be detected; the cause of the scram cannot be determined. [Use Category 9 if scram by SI reactor trip (and SI is spurious).]
T3*	39	**Automatic Trip - No Transient Condition** - An auto scram is initiated by a hardware failure in instrumentation or logic circuits and no out-of-tolerance condition exists.

TABLE 3.1.1-8 (Continued)
CORRESPONDENCE BETWEEN NORTH ANNA IE GROUP AND EPRI CATEGORIES

North Anna IE Group(s)	EPRI Category	Title and Definition
T3	40	**Manual Trip - No Transient Condition -** The operator initiates a scram for any reason when no our-of-tolerance condition exists.
T3	41	**Fire Within Plant** - A plant shutdown is necessitated by a fire in some part of the plant.

* Evidenced in North Anna data
\+ Manual reactor trip only
[ ] North Anna specific

TABLE 3.1.1-9
SOURCES OF DATA FOR PLANT-SPECIFIC INITIATORS

1. North Anna Licensee Event Reports (LERs) for the period 1986 - 1990[1],

2. North Anna Power Station "Monthly Operating Report" for the period 1986 - 1990,

3. NUREG/CR-3862 for reactor trips within the interval 1978 through 1981, and for the power level of some reactor trip events over the interval 1982 through 1983. Note that North Anna "Monthly Operating Reports" were scanned to identify any unusual initiating events for the interval 1980 through 1990.

[1] North Anna LERs were reviewed for the period 1984-1990 for the T9-related precursors involving loss of feeder power to the 4160 V buses 1H and 1J.

TABLE 3.1.1-10
LIST OF NORTH ANNA REACTOR TRIP EVENTS, 1986-1990

VaP Unit	EPRI Cat	Yr	Date	ID	IE Group	Pwr Lev	Bkr Cls	Description	Cause	SI	References
N1	15	90	01/23/90	1	T3	100	Y	RT ON STEAM/FEEDWATER FLOW MISMATCH DUE TO A FAILED DRIVER CARD ON A FRV.	NA 1 EXPERIENCED AN AUTO Rx TRIP FROM 100% POWER DUE TO LOW LEVEL IN THE C SG WITH STEAM FLOW/FW FLOW MISMATCH. THE MISMATCH RESULTED FROM THE CLOSURE OF THE C MF REG. VLV DUE TO A FAILED PCB DRIVER CARD IN THE VALVE CONTROLLER	N	LER 90-001-00
N1	21	89	12/05/89	2	T2A	90	Y	AUTO REACTOR TRIP RESULTING FROM EHC SYSTEM TRANSIENT. REACTOR WAS INITIALLY AT 90% POWER AND RAMPED DOWN UNTIL TRIP.	UNIT 1 EXPERIENCED AN AUTO REACTOR TRIP FROM 7% POWER DUE TO A LO LO LEVEL IN THE B SG CAUSED BY FW ISOLATION. PRIOR TO THE REACTOR TRIP, THE POWER WAS BEING REDUCED DUE TO EHC SYSTEM PRESSURE TRANSIENTS WHICH WAS CAUSED BY LEAKING TURBINE OPC VLVS.	N	LER 89-017-00
N1	33	89	07/19/89	3	T3	90	Y	REACTOR TRIP DUE TO A LOSS OF EHC SYSTEM PRESSURE.	UNIT 1 EXPERIENCED AN AUTO Rx TRIP FROM 90% POWER DUE TO A LOSS OF EHC SYSTEM PRESSURE WHICH WAS CAUSED BY A FAILED O-RING ON THE TURBINE TRIP SOV 20-ET, RESULTING IN THE CLOSURE OF THE TURBINE STOP VALVES GENERATING THE TURBINE TRIP SIGNAL.	N	LER 89-014-00
N1	15	89	02/25/89	4	T3	76	Y	REACTOR TRIP DUE TO A MAIN FEEDWATER REGULATING VALVE CLOSURE AND SUBSEQUENT SG TUBE LEAK.	UNIT 1 EXPERIENCED AN AUTO Rx TRIP FROM 76% POWER DUE TO 'C' SG STEAM FLOW/FW FLOW MISMATCH COINCIDENT WITH A LOW SG LEVEL. THE MISMATCH WAS CAUSED BY THE CLOSURE OF THE C MF REG. VALVE, ON THE LOSS OF AIR.	N	LER 89-005-00

TABLE 3.1.1-10 (Continued)
LIST OF NORTH ANNA REACTOR TRIP EVENTS, 1986-1990

VaP Unit	EPRI Cat	Yr	Date	ID	IE Group	Pwr Lev	Bkr Cls	Description	Cause	SI	References
N1	15	88	08/06/88	5	T9	100	Y	REACTOR TRIP ON STEAM FLOW/FEED FLOW MISMATCH COINCIDENT WITH A LOW LEVEL DUE TO MFRV CLOSURE.	AUTO Rx TRIP FROM 100% POWER DUE TO THE MISMATCH OF SG FEED FLOW/SG COINCIDING WITH A LOW LEVEL. THE MISMATCH RESULTED FROM A CLOSURE OF THE 'B' MF REG VLV WHICH WAS CAUSED BY A DEGRADED VOLTAGE CONDITION ON THE 1J EMERGENCY BUS, CAUSED BY AN RSST (RESERVE STATION SERVICE TRANSFORMER) FAULT.	N	LER 88-020-00
N1	33	88	03/19/88	6	T3	004	N	TURBINE TRIP/REACTOR TRIP-EHC SYSTEM MALFUNCTION. NOT INCLUDED BECAUSE OF LOWER POWER LEVEL.	UNIT 1 EXPERIENCED AN AUTO Rx TRIP FROM 3.5% POWER DUE TO SPIKE IN THE TURBINE IMPULSE PRESSURE WHICH CAUSED A TURBINE TRIP & ENABLED THE LOGIC FOR A REACTOR TRIP WHEN A TURBINE TRIP CONDITION EXISTED.	N	LER 88-013-00
N1	33	88	01/13/88	7	T2A	015	Y	AUTOMATIC REACTOR TRIP DUE TO HI-HI STEAM GENERATOR LEVEL.	AUTO TURBINE TRIP/Rx TRIP FROM 15% POWER DUE TO A TURBINE SOLENOID TRIP WHICH RESULTED WHEN A HI-HI LEVEL (>75%) WAS DETECTED ON 2/3 LEVEL CHANNELS IN THE B SG. THE HI-HI LEVEL CAUSED FW ISOLATION AND WAS THE RESULT OF SG LEVEL OSCILLATIONS.	N	LER 88-005-00
N1	30	88	01/08/88	8	T2A	100	Y	MANUAL REACTOR TRIP IN ANTICIPATION OF LOSS OF THE MAIN CONDENSER.	Rx WAS MANUALLY TRIPPED FROM 100% POWER IN ANTICIPATION OF LOSS OF THE MAIN CONDENSER AFTER THE THREE RUNNING CW PUMPS TRIPPED SIMULTANEOUSLY & CONDENSER VACUUM WAS OBSERVED TO BE DECREASING RAPIDLY. CAUSE OF PUMPS FAILURE COULD NOT BE FOUND.	N	LER 88-002-00

TABLE 3.1.1-10 (Continued)
LIST OF NORTH ANNA REACTOR TRIP EVENTS, 1986-1990

VaP Unit	EPRI Cat	Yr	Date	ID	IE Group	Pwr Lev	Bkr Cls	Description	Cause	SI	References
N1	22	87	11/23/87	9	T3	100	Y	REACTOR TRIP GENERATED FROM 5A FEEDWATER HI-HI LEVEL SIGNAL.	REACTOR TRIPPED FROM 100% POWER DUE TO A TURBINE SOLENOID TRIP WHICH RESULTED FROM A 5A FEEDWATER HEATER HI-HI LEVEL SIGNAL WHICH WAS GENERATED WHEN A LEVEL SWITCH FAILED.	N	LER 87-020-00
N1	26	87	07/15/87	10	T2A	100	Y	MANUAL REACTOR TRIP DUE TO INDICATIONS OF EXCESSIVE RCS LEAKAGE THROUGH STEAM GENERATOR TUBE.	REACTOR WAS MANUALLY TRIPPED FROM 100% POWER DUE TO INDICATIONS OF A SG TUBE LEAKAGE IN THE C SG. -20 MIN. LATER SAFETY INJECTION SYSTEM WAS AUTOMATICALLY INITIATED. THE ROOT CAUSE HAS BEEN LABELED A SG TUBE RUPTURE; HOWEVER, CONSIDERING SG REPLACEMENT, THIS EVENT WAS CATEGORIZED T3 AS A SG TUBE LEAK REQUIRING MANUAL REACTOR TRIP.	Y	LER 87-017-01
N1	33	87	06/29/87	11	T3	018	Y	REACTOR TRIP DUE TO 5A FEEDWATER HEATER HI-HI LEVEL.	Rx TRIPPED FROM 18% POWER DUE TO A TURBINE SOLENOID TRIP WHICH RESULTED FROM A 5A FW HEATER HI-HI LEVEL SIGNAL. THE HI-HI LEVEL IN THE 5A FW HEATER WAS CAUSED BY AN IMPROPER VLV LINE-UP FOLLOWING A REFUELING OUTAGE.	N	LER 87-015-01
N1	3	87	04/19/87	12	T3	067	Y	REACTOR TRIP CAUSED BY DROPPED CONTROL ROD.	REACTOR TRIPPED FROM 67% POWER DURING A CONTROLLED RAMPDOWN INTO A REFUELING OUTAGE DUE TO NUCLEAR INSTRUMENTATION SYSTEM POWER RANGE HIGH NEGATIVE FLUX RATE CAUSED BY A SINGLE DROPPED ROD.	N	LER 87-004-00

TABLE 3.1.1-10 (Continued)
LIST OF NORTH ANNA REACTOR TRIP EVENTS, 1986-1990

VaP Unit	EPRI Cat	Yr	Date	ID	IE Group	Pwr Lev	Bkr Cls	Description	Cause	SI	References
N1	33	86	08/27/86	13	T3	100	Y	MANUAL TURBINE/REACTOR TRIP DUE TO HIGH TURBINE/GENERATOR VIBRATION.	TURBINE/REACTOR WERE MANUALLY TRIPPED WHEN NA1 WAS AT 100% POWER DUE TO HIGH VIBRATION OF TURBINE/GENERATOR BEARING VIBRATION. VIBRATION CAUSE WAS BREAKAGE OF A 13 INCH PIECE OF TURBINE BLADE FROM THE LAST STAGE OF THE 'A' LOW PRESSURE TURBINE.	N	LER 86-015-00
N1	16	86	05/20/86	14	T3	100	Y	REACTOR TRIP FROM STEAM FLOW/FEED MISMATCH COINCIDENT WITH LOW STEAM GENERATOR LEVEL.	Rx TRIP OCCURRED FROM 100% POWER DUE TO A TRIP SIGNAL GENERATED FROM A STEAM FLOW/FEED FLOW MISMATCH (ALL 3 FW REG VLVS CLOSED BY SPURIOUS FW ISOLATION SIGNAL TO FRVS ONLY) CONCURRENT WITH A LOW LEVEL (2/3 LESS THAN/EQUAL TO 25% N.R. LEVEL) IN THE SG.	N	LER 86-008-00
N1	17	86	03/26/86	15	T2A	100	Y	REACTOR TRIP DUE TO A SAFETY INJECTION TRIP SIGNAL.	Rx TRIPPED FROM 100% POWER DUE TO A SI CAUSED BY THE CLOSURE OF THE B MAIN STEAM LINE TRIP VALVE. THIS RESULTED IN REACTOR AND TURBINE TRIP. THE SI WAS INITIATED DUE TO HIGH STEAM FLOW COINCIDENT WITH LOW STEAM LINE PRESSURE IN 'A' & 'C' SGs.	Y	LER 86-006-00
N1	39	86	05/31/86	16	T3	100	Y	REACTOR TRIP DUE TO LOSS OF A POWER TO 120 VAC VITAL BUS.	Rx TRIPPED FROM 100% POWER DUE TO FAILURE OF VITAL BUS WHICH POWERS THE RELAY THAT SENSES THE BREAKER POSITION OF 'A' RCP. DE-ENERGIZED RELAY, LEAD TO Rx TRIP SIGNAL BECAUSE THE RPS SENSED THE 'A' RCP BREAKER OPEN COINCIDENT WITH REACTOR POWER >30%.	N	LER 86-009-00

TABLE 3.1.1-10 (Continued)
LIST OF NORTH ANNA REACTOR TRIP EVENTS, 1986-1990

VaP Unit	EPRI Cat	Yr	Date	ID	IE Group	Pwr Lev	Bkr Cls	Description	Cause	SI	References
N1	33	86	02/23/86	17	T3	100	Y	REACTOR/TURBINE TRIP - TURBINE CONTROL SYSTEM MALFUNCTION.	Rx TRIP/TURBINE TRIP OCCURRED FROM 100% POWER. THE REACTOR TRIP SIGNAL WAS GENERATED BY A LO-LO LEVEL IN 'B' SG, DUE TO CLOSURE OF THE TURBINE GOVERNOR VALVES, CAUSING SHRINKAGE IN ALL SG WITH 'B' SG REACHING THE Rx TRIP SETPOINT FIRST.	N	LER 86-002-00
N1	33	86	01/19/86	18	T3	004	Y	REACTOR/TURBINE TRIP DUE TO A TURBINE FIRST-STAGE IMPULSE PRESSURE SPIKE. NOT INCLUDED BECAUSE OF LOW POWER LEVEL.	TURBINE TRIP/REACTOR TRIP OCCURRED FROM 4% POWER DUE TO A TURBINE FIRST-STAGE IMPULSE PRESSURE SPIKE AS PLANT PERSONNEL WERE SETTING UP FOR A TURBINE-GENERATOR OVERSPEED TRIP TEST.	N	LER 86-001-00
N2	21	90	11/02/90	19	T3	15	Y	REACTOR TRIP FROM 9% POWER DUE TO LOSS OF NORMAL FEEDWATER. REACTOR WAS INITIALLY AT 15 % POWER.	AUTO REACTOR TRIP OCCURRED FROM 9% POWER DUE TO A LO-LO LEVEL IN 'A' SG WHILE RETURNING TO POWER OPER. THE REACTOR TRIP OCCURRED ~8 MIN. FOLLOWING AN AUTO TURBINE TRIP FROM ~15% POWER. THE CAUSE OF EVENT WAS PERSONNEL ERROR TO RESET FW BYPASS VALVE.	N	LER 90-010-00
N2	15	86	06/29/86	20	T9	100	Y	REACTOR TRIP DUE TO LOW STEAM GENERATOR LEVEL COINCIDENT WITH A STEAM FLOW/FEED FLOW MISMATCH.	Rx TRIP OCCURRED FROM 100% POWER DUE TO LOW SG LEVEL COINCIDENT WITH A STEAM FLOW/FEED FLOW MISMATCH DURING EMERGENCY RAMPDOWN, DUE TO LOSS OF 2/3 MFW PUMPS CAUSED BY A LOSS OF POWER TO 1 OF 2 500kV SWITCHYARD BUSSES.	N	LER 86-009-00

TABLE 3.1.1-10 (Continued)
LIST OF NORTH ANNA REACTOR TRIP EVENTS, 1986-1990

VaP Unit	EPRI Cat	Yr	Date	ID	IE Group	Pwr Lev	Bkr Cls	Description	Cause	SI	References
N2	34	86	04/11/86	21	T3	071	Y	UNIT 2 REACTOR TRIP DUE TO A TURBINE TRIP CAUSED BY A MAIN ELECTRICAL GENERATOR TRIP.	REACTOR TRIP OCCURRED FROM 71% POWER DUE TO A TURBINE TRIP CAUSED BY A MAIN ELECTRICAL GENERATOR TRIP, DUE TO ACTUATION OF A GENERATOR DIFFERENTIAL LOCKOUT RELAY UPON LOSS OF EXCITATION FIELD SIGNAL CAUSED BY FAILURE OF THE PERMANENT MAGNET GENERATOR.	N	LER 86-008-00
N2	33	86	04/16/86	22	T3	004	Y	REACTOR TRIP CAUSED BY TURBINE FIRST STAGE PRESSURE SPIKE. NOT INCLUDED BECAUSE OF LOW POWER LEVEL.	REACTOR TRIPPED FROM 4% POWER DUE TO TURBINE 1ST STAGE PRESS. SPIKE, CAUSED BY PERFORMING A THROTTLE VALVE/GOVERNOR VALVE TRANSFER WITH TURBINE IN AUTO CONTROL. THE PRESS. SPIKE CLEARED THE P-7 Rx TRIP BLOCKS CAUSING Rx TRIP DUE TO TURBINE TRIP.	N	LER 86-007-00
N2	3	86	05/29/86	23	T3	100	Y	UNIT 2 REACTOR TRIP OCCURRED FROM A NEGATIVE FLUX RATE TRIP.	Rx TRIP OCCURRED FROM 100% POWER DUE TO A NEGATIVE FLUX RATE CAUSED BY THE OPENING OF THE STATIONARY COIL POWER SUPPLY DISCONNECTED TO ROD CONTROL POWER DISTRIBUTION CABINET 1AC, CAUSING 12 RODS TO DROP INTO THE CORE. PERSONNEL ERROR CAUSED THE EVENT.	N	LER 86-005-00

TABLE 3.1.1-11
SUMMARY OF NORTH ANNA SYSTEM REVIEW FOR INITIATING EVENTS

System	System Symbol	Front line or Support	Detailed Analysis
Ambient Air Monitoring	AM	Neither	No
ATWS Mitigation System Actuation & Control (AMSAC)		Front line	No
Auxiliary Boilers	AB	Neither	No
Auxiliary Feedwater	AFW	Front Line	Yes
Auxiliary Steam	AS	Neither	No
Batteries, 125VDC	BY	Support	Yes
Bearing Cooling	BC	Support	Yes
Bearing Lube	BL	Neither	No
Blowdown	BD	Neither	No
Boron Recovery	BR	Neither	No
Building Structure	BLD	Neither	No
Chemical & Volume Control	CH	Front line	Yes
Chilled Water	CD	Neither	No
Circulating Water	CW	Support	Yes
Communications	CO	Neither	No
Component Cooling	CC	Support	Yes
Compressed Air	CA	Neither	No
Computer	CM	Neither	No
Condensate	CN	Support	Yes
Condensate Polishing	CP	Neither	No
Containment Access	CE	Neither	No
Containment Vacuum	CV	Neither	No
Control Rod Drive Power Supply	ED	Neither	No
Decay Heat Release	DHR	Neither	No
Decontamination	DC	Neither	No
Demineralizer Drain	WDR	Neither	No
Diesel Air	EB	Support	No
Drains (Aerated)	DA	Neither	No
Drains (Building Services)	DB	Neither	No
Drains (Gaseous)	DG	Neither	No
Domestic Water	DW	Neither	No
Early Warning	EW	Neither	No
Earthquake Reporting	ER	Neither	No
Electrical Calibration	EC	Neither	No
Electrical Equipment	PHP	Neither	No
Electrical Equipment (4KV & Above)	PH	Support	Yes
Electrical Equipment (600V & Below)	PL	Support	Yes
Electrical Hydraulic Control	EH	Neither	No
Electrical Instrumentation	EI	Neither	No
Electrical Power	EP	Support	Yes

TABLE 3.1.1-11 (Continued)
SUMMARY OF NORTH ANNA SYSTEM REVIEW FOR INITIATING EVENTS

System	System Symbol	Front line or Support	Detailed Analysis
Electronic Modules	EM	Neither	No
Emergency Electrical Power	EE	Support	Yes
Emergency Generator & Diesel	EG	Support	No
Emergency Lighting	ELT	Neither	No
Excore Neutron Flux Indication	NFI	Support	No
Extraction Steam	ES	Neither	No
Feedwater	FW	Front line	Yes
Fuel Handling	FH	Neither	No
Fuel Oil	FO	Support	No
Fire Protection	FP	Neither	No
Fuel Pit Cooling	FC	Neither	No
Gaseous Waste (Radioactive)	GW	Neither	No
Generator Breaker	GB	Neither	No
Gland Steam	GS	Neither	No
Hanger	H	Neither	No
Heat Tracing	HT	Neither	No
Heating & Ventilation	HV	Support	Yes
High Pressure Steam Atmospheric Discharge	SAE	Neither	No
High Radiation Sampling	HRS	Neither	No
Incore Instrumentation	IC	Neither	No
Instrument Air	IA	Support	Yes
Laboratory Vacuum	LV	Neither	No
Leakage Monitoring	LM	Neither	No
Liquid & Solid Waste (Radioactive)	LW	Neither	No
Loose Parts Monitoring	LPM	Neither	No
Lubricating Oil (Turbine Motors)	LO	Neither	No
Main Generator (Gas Supplies)	GM	Neither	No
Main Steam	MS	Front line	Yes
Main Steam Piping	SHP	Neither	No
Materials Handling	MH	Neither	No
Meteorological Metering	MM	Neither	No
Miscellaneous	MIS	Neither	No
Neutron Monitoring	NM	Neither	No
Neutron Shield Tank & Cooling	NS	Neither	No
Nuclear Instrumentation	NI	Neither	No
Penetrations Electrical	PE	Neither	No
Penetrations	PEN	Neither	No
Piping	P	Neither	No
Post Accident Hydrogen Removal	HC	Neither	No
Post Accident Monitoring	PAM	Neither	No

TABLE 3.1.1-11 (Continued)
SUMMARY OF NORTH ANNA SYSTEM REVIEW FOR INITIATING EVENTS

System	System Symbol	Front line or Support	Detailed Analysis
Pressurizer Heaters	HS	Neither	No
Primary & Secondary Plant Gas Supplies	GN	Neither	No
Primary Grade Water	PG	Neither	No
Process Instrumentation	PRO	Neither	No
Quench Spray	QS	Front line	Yes
Rad Waste	RW	Neither	No
Radiation Monitoring	RM	Neither	No
Reactor Coolant	RC	Front line	Yes
Reactor Protection	RPS	Front line	No
Recirculation Spray	RS	Front line	Yes
Recirculation Spray/Low Head Area	RSL	Neither	No
Refueling Purification	RP	Neither	No
Relay Testing	RT	Neither	No
Residual Heat Removal	RH	Front line	Yes
Rod Control	RD	Front line	No
Rod Control System	RCS	Front line	No
Rod Position Indication	RPI	Neither	No
S/G Blow Down	WGC	Neither	No
Safety Injection (or ECCS)	SI	Front line	Yes
Safety Valve Discharge	SSV	Neither	No
Sampling System	SS	Neither	No
Sanitary Service	PS	Neither	No
Sanitary Sewage	PB	Neither	No
Secondary Ventilation	SV	Neither	No
Security	SEC	Neither	No
Service Air	SA	Support	Yes
Service Water	SW	Support	Yes
Shock Suppressor	HSS	Neither	No
Solid State Protection System	SSP	Front line	No
Special Metering	SM	Neither	No
Spillway (or Main Dam)	SP	Neither	No
Steam Drains	SD	Neither	No
Steam Generator Drain	SGD	Neither	No
Steam Generator Support	SGS	Neither	No
Structures	STR	Neither	No
Turbine	TM	Neither	No
Vacuum Priming	VP	Neither	No
Valve Monitoring System	VMS	Neither	No
Valve Position Indication	VPI	Neither	No

TABLE 3.1.1-11 (Continued)
SUMMARY OF NORTH ANNA SYSTEM REVIEW FOR INITIATING EVENTS

System	System Symbol	Front line or Support	Detailed Analysis
Valve Service Water	VSW	Support	No
Vents (Aerated)	VA	Neither	No
Vents (Gaseous or Hydrogenated)	VG	Neither	No
Vital Bus	VB	Support	Yes
Water Treatment	WT	Neither	No

TABLE 3.1.1-12
SUMMARY OF LOSS OF SUPPORT SYSTEMS AS INITIATORS

Support System Loss Considered	Impact on Normal Operation	Attendant Important System Failures	Initiating Event Group
4160 V Bus 1H	IRPI loss with, total 4160 1H Bus loss could result in manual Reactor Trip or Shutdown	Charging Pump A ECCS Train A 480 V 1H 480 V 1H1 480 V 1H1-1 480 V 1H1-2S 480 V 1H1-4	Represented by the T9A Initiator. Impact on ESGR cooling also considered in the T8 Initiator.
4160 V Bus 1J	Isolation of RCP CC cooling could result in manual Reactor Trip or Shutdown	Charging Pump B ECCS Train B 480 V 1J 480 V 1J1 480 V 1J1-1 480 V 1J1-2S	Represented by the T9B Initiator. Impact on ESGR cooling also considered in the T8 Initiator.
480 V Bus 1H	IRPI loss with total 480 1H bus loss could result in manual Reactor Trip or Shutdown	Some ECCS Train A 480 V 1H1-1 480 V 1H1-4	Included within the T9A Initiator.
480 V Bus 1H1	No direct impact	Some ECCS Train A 480 V 1H1-2S	Not included as an Initiator. Disables some standby ECCS equipment, but doesn't cause transient or direct reactor trip.

TABLE 3.1.1-12 (Continued)
SUMMARY OF LOSS OF SUPPORT SYSTEMS AS INITIATORS

Support System Loss Considered	Impact on Normal Operation	Attendant Important System Failures	Initiating Event Group
480 V Bus 1H1-1	IRPI loss with total 480 1H1-1 Bus loss could result in manual Reactor	ESGR Chiller Trains A & C	Included within the T9A Initiator.
480 V Bus 1H1-2S	No direct impact	Some ECCS Train A Same as 480 V Bus 1H1.	
480 V Bus 1H1-4	No direct impact	ESGR AHU 6 ESGR Chiller Train C	Included within the T9A Initiator. Impact on ESGR cooling also considered in the T8 Initiator.
480 V Bus 1J	Isolation of RCP cooling could result in manual Reactor Trip or Shutdown	Some ECCS Train B 480 V 1J1-1 480 V 1J1-2S	Included within the T9B Initiator.
480 V Bus 1J1	No direct impact	Some ECCS Train B	Same as 480 V Bus 1H1.

TABLE 3.1.1-12 (Continued)
SUMMARY OF LOSS OF SUPPORT SYSTEMS AS INITIATORS

Support System Loss Considered	Impact on Normal Operation	Attendant Important System Failures	Initiating Event Group
480 V Bus 1J1-1	Isolation of RCP cooling could result in manual Reactor Trip or Shutdown	ESGR AHU 7 ESGR Chiller Train B	Included within the T9B Initiator.
480 V Bus 1J1-2S	No direct impact	Some ECCS Train B	Same as 480 V Bus 1H1.
120 VAC Vital Bus 1-I	Manual Reactor Trip on loss of RCP Cooling	MS Atmospheric Dump Valve A CC to RCP Thermal Barriers isolated	Included within the T3 Initiator.
120 VAC Vital Bus 1-II	No direct impact	MS Atmospheric Dump Valve B	None
120 VAC Vital Bus 1-III	Manual Reactor Trip on loss of RCP Cooling	MS Atmospheric Dump Valve C CC to RCP Thermal Barriers isolated	Included within the T3 Initiator.
125 VDC Bus 1-I	Reactor Trip on loss of MFW	ECCS Train A 4160 V switchgear MS Condenser Dump Valves	Represented by the T5A Initiator.

TABLE 3.1.1-12 (Continued)
SUMMARY OF LOSS OF SUPPORT SYSTEMS AS INITIATORS

Support System Loss Considered	Impact on Normal Operation	Attendant Important System Failures	Initiating Event Group
125 VDC Bus 1-II	No direct impact	MFW Pump B Condensate Pump B	Not included as an Initiator. Standby MFW & Condensate Pumps available with autostart.
125 VDC Bus 1-III	Reactor Trip on loss of MFW	ECCS Train B 4160 V switchgear MS Condenser Dump Valves	Represented by the T5B Initiator.
Service Water	Manual Reactor Trip or Shutdown on loss of CC to RCPs, loss of Instrument Air or loss of ESGR cooling	Charging Pumps A/B/C CC Heat Exchangers ESGR Chillers A/B/C Instrument Air Compressors Recirculation Spray Heat Exchangers RCP Thermal Barriers RHR Pumps and Heat Exchangers cooling for SGTR	Represented by the T6 Initiator.
Component Cooling Water	Manual Reactor Trip or Shutdown on loss of RCP cooling	RCP Thermal Barriers RHR Pumps and Heat Exchangers cooling for SGTR	Impact on RCP Thermal Barriers considered in the T4 Initiator.

TABLE 3.1.1-12 (Continued)
SUMMARY OF LOSS OF SUPPORT SYSTEMS AS INITIATORS

Support System Loss Considered	Impact on Normal Operation	Attendant Important System Failures	Initiating Event Group
Emergency Switchgear Room Cooling	Manual Reactor Trip or Shutdown due to switchgear thermal overload	All AC ECCS switchgear in ESGR	Represented by the T8 Initiator.
Containment Instrument Air	Manual Reactor Trip or Shutdown on loss of RCP cooling	Pressurizer PORV (backup nitrogen supply) RCP Thermal Barriers	Impact on RCP Thermal Barriers considered in the T4 Initiator.
Instrument Air Outside Containment	Reactor Trip on loss of MFW or MS isolation	RCP Thermal Barriers RHR Pump and Heat Exchanger cooling for SGTR MS Condenser Dump Valves MS Atmospheric Dump Valves (backup air (supply)	Included within the T2 Initiator. Impact on RCP Thermal Barriers considered in the T4 Initiator.
Bearing Cooling Water	Reactor Trip on loss of MFW	MFW Pumps Condensate Pumps	Included within the T2 Initiator.

TABLE 3.1.1-13
DEPENDENCY MATRIX OF FRONT LINE SYSTEMS - PUMPS

System:	1-CH			1-CN			1-FW			1-FW			1-QS		1-RH		1-RS		1-RS		1-RS		1-SI	
Component:	P-1			P-1			P-1			P-			P-1		P-1		P-1		P-2		P-3		P-1	
Train:	A	B	C	A	B	C	A	B	C	2	3A	3B	A	B	A	B	A	B	A	B	A	B	A	B
4160 V																								
1A				X			X																	
1B					X			X																
1C						X			X															
1H	X		X								X				X				X				X	
1J		X	+									X				X				X				X
480 V MCC																								
1H1													X				X							
1H1-2S																					X			
1J1														X				X						
1J1-2S																						X		
125 VDC																								
1-I	X		X			X			X	X	X		X		X		X		X				X	
1-II					X			X																
1-III		X	+	X			X			X		X		X		X		X		X				X
Instrument Air										*														
Component Cooling															X	X								
Service Water	X	X	X																					

* Turbine driven pump starts on loss of air
\+ Alternate support

TABLE 3.1.1-13 (Continued)
DEPENDENCY MATRIX OF FRONT LINE SYSTEMS - VALVES

System:	1-FW			1-MS			1-MS	1-MS			1-RC		1-RC	
Component:	FCV-14			PCV-101			TCV-1498	TV-101			MOV-		PCV-	
Train:	78	88	98	A	B	C	A thru H	A	B	C	1536	1535	1455C	1456
Bearing Cooling							X X X	X	X	X				
Main Steam								X						
480 V MCC														
1H1-2S											X			
1J1-2S												X		
120 VAC VB														
1-I				X										
1-II					X									
1-III						X								
125 VDC														
1-I	X	X	X				X	X	X	X			X	
1-III	X	X	X				X	X	X	X				X
Instrument Air	X	X	X	*	*	*	X	X	X	X			*	*

* Backup air supply provided

TABLE 3.1.1-14
DEPENDENCY MATRIX OF SUPPORT SYSTEMS

System: Component: Train:	1-BC P-1 A	1-BC P-1 B	1-CC P-1 A	1-CC P-1 B	1-CC TV-101 A	1-CC TV-101 B	1-CC TV-102 ACE	1-CC TV-102 BDF	1-CC TV-104 A	1-CC TV-104 B	1-CC TV-104 C	1-CC TV-106 A	1-CC TV-106 B	1-CC TV-106 C	1-HV AC- 6	1-HV AC- 7	1-HV E-4 A	1-HV E-4 B	1-HV E-4 C	1-SW P- 1A	1-SW P- 1B	1-SW P- 4A
4160 V																						
1B	X																					
1C		X																				
1H			X																	X		X
1J				X																	X	
480 V MCC																						
1H1-1																	X		X			
1H1-4															X				X			
1J1-1												X	X	X		X		X				
120 VAC VB																						
1-I	X		X		X		X		X	X	X											
1-II																						
1-III		X		X		X		X	X	X	X											
125 VDC																						
1-I		X	X																	X		X
1-II	X																					
1-III				X																	X	
Instrument Air					X	X	X	X	X	X	X	X	X	X								
Service Water																	X	X	X			

TABLE 3.1.1-15
TRANSIENT SUCCESS CRITERIA

Reactivity Control	Core Heat Removal Early	Core Heat Removal Late	Secondary Heat Removal	RCS (Integrity)	Containment Condition
RPS Scram with < 2 rod failure to insert[a]	RCS - Natural Circ.		1/3 MFW pumps[b,f] OR 1/3 AFW pumps to 1/3 SGs[c]	RCS PORV Closure Note 1	Not Required
RPS Scram	1/3 Charging Pumps AND 1 RCS PORV (Feed & Bleed)[e]	Recirc. through 1/3 charging pumps - AND 1/2 Lo Head SI Pumps[d] (Note 3)	Not Required	Note 2	Recirculation through 1/2 IRS OR 1/2 ORS[e]

Notes:

1. Failure of RCS Integrity by failure of RCS PORV to close transfers to S2 event tree.

2. Feed & Bleed operation fails RCS Integrity through continued RCS PORV use.

3. For Transients, RCS depressurization before recirculation is not certain, so only high head safety recirculation is modeled. Also, ORS can be manually aligned to act as a backup for Lo Head Recirc for NAPS Unit 1.

References:

a. WCAP-9691 p. A-11
b. WCAP-9691 p. A-12
c. WCAP-9691 p. A-15
d. WCAP-9744
e. Surry Analysis File 321MAF.1
f. NAPS UFSAR

TABLE 3.1.1-16
LARGE LOCA SUCCESS CRITERIA

Reactivity Control	Core Heat Removal Early	Core Heat Removal Late	Secondary Heat Removal	RCS Integrity	Containment Condition
No Automatic Scram Required But Borated Water Injection Required for Long-Term Subcriticality	1/2 Low (a) Head SI Pumps AND 2/3 Accumulators	1/2 Low (a) Head SI Pumps In Low Pressure Recirculation Mode AND Changeover to hot leg Recirculation (d)	Not Required	Lost as Result of Initiator	1/2 Quench Spray(b) AND 1/2 Inside Recirc Spray OR 1/2 Outside Recirc Spray(c)

References:
(a) North Anna UFSAR
(b) North Anna Analysis File 321MAF.N.1
(c) MAAP analysis
(d) ORS can be manually aligned to act as a backup for Lo Head Recirc for NAPS Unit 1.

TABLE 3.1.1-17
MEDIUM LOCA SUCCESS CRITERIA

Reactivity Control	Core Heat Removal Early	Core Heat Removal Late	Secondary Heat Removal	RCS Integrity	Containment Condition
RPS	1/3 Charging Pumps AND 2/3 Accumulators(a)	1/2 Charging Pumps AND 1/2 Low Head SI Pumps in Recirculation Mode(e)	Not Required	Lost as Result of Initiator	1/2 Outside Recirc Spray OR 1/2 Inside Recirc Spray(c)
RPS	1/3 Charging Pumps	1/3 Charging Pumps AND 1/2 Low Head Safety Injection Pumps in Recirculation Mode(e)	1 AFW Pump to 1/3 SG(f)	Same	Same
RPS	3/3 Accumulators AND 1/2 Low Head SI Pumps(b)	1/2 Low Head SI Pumps In Recirculation Mode (e)	Steam Dump Through 2 SG AOVs with 2 AFW Pumps(d)	Same	Same

References:
(a) WCAP-9601
(b) WCAP-9754
(c) North Anna Analysis File 321MAF.N.1
(d) The AFW arrangement at NAPS requires two steam dump valves and two AFW pumps for success.
(e) ORS can be manually aligned to act as a backup for Lo Head Recirc for NAPS Unit 1.
(f) Beynon, 1988

TABLE 3.1.1-18
SMALL LOCA

Reactivity Control	Core Heat Removal Early	Core Heat Removal Late	Secondary Heat Removal	RCS Integrity	Containment Condition
RPS	1/3 Charging Pumps(a)	1/3 Charg- Pumps AND 1/2 Low Head SI Pumps In Recircu- lation Mode(f)	1/3 AFW pumps to 1/3 SG	Lost as Result of Initiator	1/2 Outside Recirc Spray OR 1/2 Inside Recirc Spray(d)
RPS	1/3 Charg- ing Pumps AND 1 RCS PORV(d)	Same	Not Required	Same	Same
RPS	3/3 Accumu- lators AND 1/2 Low Head SI Pumps(c)	1/2 Low Head SI Pumps in Recircu- lation(f)	Steam Dump Through 2 SG ADVs with 2 AFW Pumps(e)	Same	Same

References:
(a) WCAP-9601
(b) WCAP-9744
(c) WCAP-9754
(d) North Anna Analysis File 321MAF.N.1
(e) The AFW arrangement at NAPS requires two steam dump valves and two AFW pumps for success.
(f) ORS can be manually aligned to act as a backup for Lo Head Recirc for NAPS Unit 1.
(g) For very small breaks no Containment heat removal is required.

TABLE 3.1.1-19
SUCCESS CRITERIA FOR ATWS

Reactivity Control	Core Heat Removal Early	Core Heat Removal Late	Secondary Heat Removal	RCS Integrity	Containment Condition
Reactor Power < 40% (a)					
Manual Rod Insertion OR Deenergize MG Set OR Emergency Boration	RCS	RCS	1 of 3 Aux. Feedwater, OR 1 Main Feedwater Pump	RCS PORV Reclosure	None
Reactor Power > 40% (a) Feedwater Available (1 of 2 Trains)					
Manual Rod Insertion OR Deenergize MG SET OR Emergency Boration	Same		Main Feedwater Continued Operation	RCS PORV Reclosure	None

TABLE 3.1.1-19 (Continued)
SUCCESS CRITERIA FOR ATWS

Reactivity Control	Core Heat Removal Early	Core Heat Removal Late	Secondary Heat Removal	RCS Integrity	Containment Condition
Reactor Power > 40%(a) Feedwater Not Available					
Manual Rod Insertion OR Deenergize MG Set OR Emergency Boration	Same		2 Aux. Feed Pumps to 2 SG(c)	AMSAC(b) AND Adequate Pressure Relief with Subsequent Valve Reclosure	None

References:
(a) WCAP-11993
(b) NAPS UFSAR

TABLE 3.1.1-20
STEAM GENERATOR TUBE RUPTURE SUCCESS CRITERIA

Reactivity Control	Core Heat Removal Early	Core Heat Removal Late	Secondary Heat Removal	RCS Integrity	Containment Condition
RPS	RCS Natural Circulation, (a,f)		1/3 AFW pumps to 1/2 SG	Achieved by cooldown and depress. & isolation of affected SG	Not Required
RPS	1/3 Charging Pumps	1/2 RHR(g) Pumps	1/3 AFW pumps to 1/2 SG	Containment bypassed (core intact)	Same
RPS	1/3 Charging Pumps AND 1 RCS PORV(d)	Recirc.(f) through 1/3 Charging Pumps AND 1/2 Lo Head SI Pumps(h)	Not Required	Lost as a result of induced LOCA	1/2 Outside Recirc Spray OR 1/2 Inside Recirc Spray

TABLE 3.1.1-20 (Continued)
STEAM GENERATOR TUBE RUPTURE SUCCESS CRITERIA

Reactivity Control	Core Heat Removal Early	Core Heat Removal Late	Secondary Heat Removal	RCS Integrity	Containment Condition
RPS	3/3 Accumulators AND 1/2 Low Head SI Pumps(c)	1/2 RHR Pumps	Steam Dump Through 2 SG ADV with 2 AFW Pump(e)	Containment bypassed (core intact)	Not Required

References:
(a) North Anna Analysis File 321MAF.N.1
(b) WCAP-9744
(c) WCAP-9754
(d) North Anna Analysis File 321MAF.N.1
(e) The AFW arrangement at NAPS requires two steam dump valves and two AFW pumps for success.
(f) With Successful Faulted SG Isolation and No Stuck Open Safety Relief Valve
(g) With failure of Faulted SG Isolation and/or Stuck Open Safety Relief Valve
(h) ORS can be manually aligned to act as a backup for Lo Head Recirc for NAPS Unit 1.

TABLE 3.1.2-1
LIST OF INITIATING EVENT CLASSES

INITIATING EVENT GROUP	DESCRIPTIONS	EVENT TREE
T1	Loss of Offsite Power	T1
T1A**	Station Blackout	T1A
T2	Transients with non-recoverable loss of Main Feedwater	T*
T2A	Transients with recoverable loss of Main Feedwater following FW Isolation	T*
T3	Transients with Main Feedwater initially available	T*
T4	Loss of RCP Seal Injection and Thermal Barrier Cooling	T4
T5A	Non-recoverable Loss of DC Bus 1-I	T*
T5B	Non-recoverable loss of DC Bus 1-III	T*
T6	Loss of Service Water	T6
T7	Steam Generator Tube Rupture	T7
T8	Loss Emergency Switchgear Room Cooling	T8
T9A	Loss of 4160 V Emergency Bus 1H	T*
T9B	Loss of 4160 V Emergency Bus 1J	T*

TABLE 3.1.2-1 (Continued)
LIST OF INITIATING EVENT CLASSES

INITIATING EVENT GROUP	DESCRIPTIONS	EVENT TREE
A	Large LOCA 6" - 20"	A
S1	Medium LOCA 2" - 6"	S1
S2	Small LOCA 3/8" - 2"	S2
V	Interfacing System LOCA	Vx
R	Reactor Vessel Rupture	Rx
TL	Transient with failure to Scram at Power < 40 percent	TL
TH	Transient with failure to Scram at Power > 40 percent	TH

* These event trees are discussed in one section of the report, as they are very similar.

** T1A is not a true initiating event, but is a consequential event from T1.

TABLE 3.1.2-2
EVENT TREE HEADINGS

Abbreviation	Headings	Description of Event
A	Large LOCA	Initiating Event-large LOCA
B	Offsite Power Recovery	Failure to recover an ESF bus following station black-out by recovering offsite power.
Ch	Containment Heat Removal	Failure of Service Water to an operable Recirculation Spray heat exchanger.
DG	EDG 1H or 1J Available	Failure of at least one diesel generator to start and run following loss of offsite power leading to station blackout.
Dh	Hot Leg Recirculation	Failure of the operator to switch to hot leg recirculation following a large LOCA.
D1	High Pressure Injection	Failure of Charging Pumps to inject in the appropriate mode.
D2	Accumulators Inject	Failure of Accumulators to inject in the appropriate mode.
D3	Low Head SI	Failure of low head SI pumps to inject.
D4	Emergency Boration	Failure to shutdown following ATWS by boron addition.
Fm	Break Size Partition	Percentage of small breaks not causing a CDA Hi Hi signal.
Hv	ESGR Cooling	Failure to provide HVAC to the ESGR using 1/2 AHUs and 1/3 chillers.
H1	Low Head Recirculation	Failure of low head pumps in the recirculation mode.

TABLE 3.1.2-2 (Continued)
EVENT TREE HEADINGS

Abbreviation	Headings	Description of Event
H2	High Head Recirculation	Failure of low head and charging pumps in the high pressure recirculation mode.
K	Reactor Subcritical	Failure of control rods to insert as result of Reactor Protection System failure.
L	Auxiliary Feedwater System Available	Failure of Auxiliary Feedwater System for transients or small or medium LOCAs with reactor trip.
Lt	Turbine-Driven AFW available	Failure of the Turbine-Driven Auxiliary Feedwater Pump to start and run following station blackout.
M	Main Feedwater System Available	Failure of Main Feedwater.
MS1	Manual Scram	Failure of the operator to remove power from the control rod drive mechanisms.
O	Cooldown and Depressurize	Operator fails to cooldown and depressurize the reactor after a small break or in response to a loss of RCP seal cooling.
O2	Late Cooldown	Failure of operator to cooldown and depressurize in response to a ruptured steam generator.
P	Pressurizer PORVs	Failure of the operator to open 1/2 pressurizer PORVs to cause RCS feed and bleed.
Pr	Pressure Relief	Failure of adequate pressure relief following an ATWS event.

TABLE 3.1.2-2 (Continued)
EVENT TREE HEADINGS

Abbreviation	Headings	Description of Event
Q	RCS Boundary Intact	Failure of pressurizer PORV to close after opening during a transient.
Qs	Quench Spray	Failure of 1/2 trains of Quench Spray.
Rc	Room Cooling Restored	Recovery of ESGR cooling or SW (resulting in reactor trip and loss of emergency power) prior to core uncovery and vessel failure, or containment failure.
Rs	Recirculation Sprays Operable	Failure of at least one train of Recirculation Sprays to remove heat from Containment.
Rv	Reactor Vessel Integrity	Consideration of PTS following a rapid RCS cooldown.
RX	Reactor Vessel Rupture	Initiating event is a Reactor Vessel rupture.
SGI	Steam Generator Isolation	Failure to isolate the ruptured Steam Generator.
Slc	No Potential for RCP Seal Failure	Failure to establish seal cooling from operable Unit 2 CC pumps.
S1	Medium LOCA	Initiating event is a medium LOCA (2" to 6").
S2	Small LOCA	Initiating event is a small LOCA (3/8" to 2").
T	Transients	Representative initiating event for general transient event tree.
Tt	Turbine Trip	Turbine fails to trip.

TABLE 3.1.2-2 (Continued)
EVENT TREE HEADINGS

Abbreviation	Headings	Description of Event
T1	Loss of Offsite Power	Initiating event is Loss of of all Offsite Power.
T1A	Station Blackout	Loss of diesel generators 1H and 1J leading to station blackout at Unit 1.
T1Tr	Loss of ESGR Cooling Transfer from T1 Event Tree	Transfer of T1Hv sequence, Loss of Offsite Power with consequential loss of Emergency Switchgear Room Cooling.
T2	Loss of MFW	Initiating event is non-recoverable loss of Main Feed-water.
T2A	Recoverable Loss of MFW	Initiating event is recoverable loss of Main Feedwater following Feedwater isolation.
T2ATr	Loss of ESGR Cooling Transfer from T2A Event Tree	Transfer of T2AHv sequence, recoverable loss of Main Feedwater with coincidental loss of Emergency Switchgear Room Cooling.
T2Tr	Loss of ESGR Cooling Transfer from T2 Event Tree	Transfer of T2Hv sequence, non-recoverable loss of Main Feedwater with coincidental loss of Emergency Switchgear Room Cooling.
T3	Transient with MFW Available	Initiating event is Transient with Main Feedwater available.
T3Tr	Loss of ESGR Cooling Transfer from T3 Event Tree	Transfer of T3Hv sequence, transient with Main Feedwater available, with coincidental loss of Emergency Switchgear Room Cooling.
T4	Loss of RC Pump Seal Cooling	Initiating event is loss of RCP seal injection and thermal barrier cooling.

TABLE 3.1.2-2 (Continued)
EVENT TREE HEADINGS

Abbreviation	Headings	Description of Event
T5A	Loss of DC Bus I	Initiating event is loss of DC Bus 1-I.
T5B	Loss of DC Bus III	Initiating event is loss of DC Bus 1-III.
T6	Loss of Service Water	Service Water is lost from both the reservoir and Lake Anna.
T7	Steam Generator Tube Rupture	Initiating event is a steam generator tube rupture.
T8	Loss of Emergency Switch-gear Room Cooling	Loss of HVAC to the Emergency Switchgear Room.
T9A	Loss of Power from 4160 V Emergency Bus 1H	Loss of feeder power to or failure of 4160 V emergency bus 1H.
T9ATr	Loss of ESGR Cooling Transfer from T9A Event Tree	Transfer of T9AHv sequence, loss of feeder power to or failure of 4160 V Emergency Bus 1H, with consequential loss of Emergency Switchgear Room Cooling.
T9B	Loss of Power from 4160 V Emergency Bus 1J	Loss of feeder power to or failure of 4160V emergency bus 1J.
T9BTr	Loss of ESGR Cooling Transfer from T9B Event Tree	Transfer of T9BHv sequence, loss of feeder power to or failure of 4160 V Emergency Bus 1J, with consequential loss of Emergency Switchgear Room Cooling.
TL	Low power transients (for ATWS)	Initiating event is all transients at power lower than or equal to 40 percent.
TH	High power transients (for ATWS)	Initiating event is all transients at power greater than or equal to 40 percent.

TABLE 3.1.2-2 (Continued)
EVENT TREE HEADINGS

Abbreviation	Headings	Description of Event
VX	Interfacing System LOCA	Initiating event is an Interfacing System LOCA.
Vi	Isolation of LOCA	Failure to isolate interfacing LOCA.
W	RHR Cooling	Failure of 1/2 Residual Heat Removal Trains.
Y	Core Cooling Recovery	Failure of the operator to use steam to rapidly cooldown and depressurize the RCS as directed by 1-FR-C.1 or C.2.

TABLE 3.3.1-1
DEFINITION OF PROBABILITY MODELS AND THEIR PARAMETERS

Basic Event	Probability Models	Data Required
Initiating Event	Poisson Model $P(r) = \frac{(ft)^r e^{-ft}}{r!}$ f: frequency	Number of events r in time t
Standby component fails on demand	1) Constant probability failure on demand, or $U = \frac{n}{N}$	1) Number of events n in total number of demands N
Standby component fails in time, or component changes state between tests (faults revealed on functional test only)	2) Constant standby failure rate $U = 1 - \frac{1-e^{-\lambda_s T_I}}{\lambda_s T_I}$ T_I : Time between tests λ_s: Standby failure rate	2) Number of events n in total time in standby T_s
Component in operation fails to run, or component changes state during mission (state of component continuously monitored)	Constant failure rate $U = 1-\exp(-\lambda_o T_m)$ $\approx \lambda_o T_m$ T_m: Mission time λ_o: Operating failure rate	Number of events n in total exposure time T_e (Time standby component is operating, or time the component is on line)

TABLE 3.3.1-1 (Continued)
DEFINITION OF PROBABILITY MODELS AND THEIR PARAMETERS

Basic Event	Probability Models	Data Required
Component unavailable due to test	$U = \frac{T_{TD}}{T_I}$ T_{TD}: Test duration (only in the case of no override signal T_I: Time between tests	Average test duration and time between tests
Component unavailable due to corrective maintenance (fault revealed only at periodic test), or preventative maintenance performed at regular intervals	$U = \frac{T_u}{T_T}$ T_u: Average time in the maintenance (out of service). T_T: Total operating time	Total time out of service due to maintenance acts whilst plant is operational. Total operating time.
Component unavailable due to unscheduled maintenance (continuously monitored components)	$U = \frac{\lambda_m T_R}{1+ \lambda_m T_R}$ λ_m: Maintenance rate T_R: Average time of a maintenance outage	Number of maintenance acts r in time T (to estimate λ_m), and average maintenance time.
Standby component which is never tested	$U = \frac{1-e^{-\lambda_s T_P}}{\lambda_s T_P}$ λ_s: Standby failure rate T_P: Exposure time to failure	If the component has never been replaced owing to previous failures, assume T_P = 40 years

TABLE 3.3.1-1 (Continued)
DEFINITION OF PROBABILITY MODELS AND THEIR PARAMETERS

Basic Event	Probability Models	Data Required
Common Cause Failure of k components in a system of m redundant components	$Q_k = \dfrac{n_k}{\binom{m}{k} N_D}$ n_k: Total number of events in the data in which k components failed N_D: Number of demands on the system	1) $n_1, n_2, \ldots, n_k$ where n_k is the number of involving failure of k components due to common cause 2) Total number of system demands N_D, or total operating time T

TABLE 3.3.1-2
LIST OF SOURCES USED TO SUPPLY GENERIC PARAMETER ESTIMATES

1) Poloski, J. P. and Sullivan, W. H., Data Summaries of Licensee Event Reports of Diesel Generators at U.S. Commercial Nuclear Power Plants, January 1, 1976 to December 31, 1978, NUREG/CR-1362, EGG-EA-5092, U.S. Nuclear Regulatory Commission, EG & G Idaho, Inc., 1980.

2) A. McClymont, G. McLagan, Diesel Generator Reliability at Nuclear Power Plants: Data and Preliminary Analysis, Interim Report, June 1982 EPRI-NP-2433. Electric Power Research Institute, Palo Alto, California, 1982.

3) Sharon R. Brown, Data Summaries of Licensee Event Reports of Protective Relays and Circuit Breakers at U.S. Commercial Nuclear Power Plants, January 1, 1976 to December 31, 1983 NUREG/CR-4126, EGG-2370 (Draft), U.S. Nuclear Regulatory Commission EG & G Idaho, Inc., 1985.

4) P. W. Baranowsky, A. M. Kolaczkowski, M. A. Fedeh., A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants, NUREG-0666 RG Division of System and Reliability Research Office of Nuclear Regulatory Research U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, April 1981.

5) M. Trojovsky, S. R. Brown, Data Summaries of Licensee Event Reports of Selected Instrumentation and Control Components at U.S. Commercial Nuclear Power Plants, January 1, 1976 to December 31, 1981, NUREG-CR-1740 Nuclear Regulatory Commission, EG & G Idaho, Inc., 1984.

6) S. R. Brown, M. Trojowsky, Data Summaries of License Event Reports of Inverters at U.S. Commercial Nuclear Power Plants, January 1, 1976 to December 31, 1982, NUREG/CR-3867, EGG-2324, U.S. Nuclear Regulatory Commission, EG & G Idaho, 1984.

7) Hubble, W. H., C. F. Miller, M. Trojovsky, and S. R. Brown, Data Summaries of Licensee Event Reports of Valves of U.S. Commercial Nuclear Power Plants, NUREG/CR-1363, 1982.

8) Trojowsky, M., Data Summaries of Licensee Event Reports on Pumps at U.S. Commercial Nuclear Power Plants, NUREG-CR-1025, Rev. 1, 1982.

9) R. E. Battle, D. J. Campbell, Reliability of Emergency AC Power Systems at Nuclear Power Plants. Prepared for U.S. Nuclear Regulatory Commission. NUREG/CR-2989, 1983.

TABLE 3.3.1-2 (Continued)
LIST OF SOURCES USED TO SUPPLY GENERIC PARAMETER ESTIMATES

10) Reliability Data Book for components in Swedish Nuclear Power Plants, RKS 85-25, prepared for Nuclear Safety Board of the Swedish Utilities and SKI, Swedish Nuclear Power Inspectorate.

11) Drago, J. P., Borkowski, J. R. Fragola, and J. W. Johnson, The In-Plant Reliability Data Base for Nuclear Plant Components: Interim Data Report - The Pump Component, NUREG/CR-2886, 1982.

12) Kahl, W. K. and R. J. Borkowski, The In-Plant Reliability Data Base for Nuclear Plan Components: Interim Report-Diesel Generators, Batteries, Charges and Inverters, NUREG/CR-3831, 1985.

13) Borkowski, R. J., et al, The In-Plant Reliability Data Base for Nuclear Components: Interim Report - The Valve Component, NUREG/CR-3154, 1983.

14) H. Wyckoff, The Reliability of Emergency Diesel Generators at U.S. Nuclear Power Plants. Final Report. NSAC/108. Nuclear Safety Analysis Center, Electric Power Research Institute, Palo Alto, California, September 1986.

15) Interim Reliability Evaluation Program Procedures Guide, NUREG/CR-2728, 1983.

16) Probabilistic Safety Analysis Procedures Guide, NUREG/CR-2815, 1985.

17) U.S. Nuclear Regulatory Commission, Reactor Safety Study: An assessment of accident Risks in U.S. Commercial Nuclear Power Plants, WASH-1400, (NUREG/75/014), Washington, D.C. 1975.

18) Guide to Reliability Data for Nuclear Power Generating Stations, Rev. 1 (IEEE-SRD-500, 1984).

19) D.C. Wood, C. L. Gottshau, Probabilistic Analysis and Operational Data in Response to NUREG-0737, Item II.k.3.2 for Westinghouse NSS Plants, Westinghouse Electric Corporation, Box 355 Pittsburgh, Pennsylvania, 15230, WCAP-9804, 1981.

20) Drouin, M. T., Harper F. T., and Camp, A. L., Analysis of Core Damage Frequency from Internal Events: Methodology Guidelines, Vol. 1, NUREG/CR-4550, September 1987.

21) DeMoss, G., et al, A Risk-Based Review of Instrument Air Systems at Nuclear Power Plants, NUREG/CR-5472, January 1990.

TABLE 3.3.1-2 (Continued)
LIST OF SOURCES USED TO SUPPLY GENERIC PARAMETER ESTIMATES

22) Ericson, D. M., et al, Analysis of Core Damage Frequency: Internal Events Methodology, NUREG/CR-4550, Vol. 1, Rev. 1, January 1990.

TABLE 3.3.1-3
GENERIC DATA FOR INITIATING EVENTS

Identifier	Description	Mean Frequency	Source / Comment
A	Large LOCA	5.0E-4/yr	NUREG/CR-4550 Vol. 1
S1	Medium LOCA	1.0E-3/yr	NUREG/CR-4550 Vol. 1
S2	Small LOCA	2.1E-2/yr	Derived from NUREG/ CR-4550 Vol. 1
T1	Loss of Offsite Power	1.1E-1/yr	NUREG/CR-5032 Common Incidence Rate Model
T5A	Loss of DC Bus 1-I	6.0E-3/yr	NUREG/CR-4550 Vol. 1
T5B	Loss of DC Bus 1-III	6.0E-3/yr	NUREG/CR-4550 Vol. 1
T7	Steam Generator Tube Rupture	1.0E-2/yr	NUREG/CR-4550 Vol. 1
RX	Reactor Vessel Rupture	2.7E-7/yr	WASH-1400 App. V
VX	Interfacing System LOCA	1.6E-6/yr	Derived from NUREG/ CR-4550 Vol. 2

TABLE 3.3.1-4
GENERIC DATA FOR EQUIPMENT FAILURES AND UNAVAILABILITIES

Identifier	Failure Description	Mean Probability	Data Source
Component Failures (/d fail on demand, /hr running or standby failures*):			
ACU-LF	Air handling unit (Air Conditioning) Loss of function (screening value)	8.71E-6	NUREG-4550, Vol 1 (Same as HEX-LU)
AOD-FC	Air operated damper fails closed	1.81E-2	NUREG-1363 (Same as AOV-FC)
AOD-FO	Air operated damper fails open	1.81E-2	NUREG-1363 (Same as AOV-FO)
AOV-FC	Air operated valve fails closed	1.81E-2	NUREG-1363
AOV-FO	Air operated valve fails open	1.81E-2	NUREG-1363
AOV-LF	Air operated valve loss of function	1.81E-2	NUREG-1363 (Same as AOV-FO)
AOV-SC	Air operated valve spurious closing	5.03E-7	NUREG-2728 (Same as MOV-SC)
BAT-LP	Battery fails to supply power	1.25E-6	NUREG-0666
BCH-LP	Battery Charger/ Rectifier fail to supply power	3.50E-6	NUREG-0666
BKR-FC	Breaker fails closed	1.83E-3	NUREG-4126
BKR-FO	Breaker fails open on demand	2.74E-4	Reliability Data Book
BKR-SO	Breaker spurious (inadvert) opening	1.40E-6	NUREG-4126

* interpretation as an operating or a standby rate depends on whether or not the failure would be a revealed fault.

TABLE 3.3.1-4 (Continued)
GENERIC DATA FOR EQUIPMENT FAILURES AND UNAVAILABILITIES

Identifier	Failure Description	Mean Probability	Data Source
BUS-LU	Bus bar loss of use during mission time	5.06E-7	IEEE-STD-500
CBL-LU	Cable (1000 ft) loss of use during mission	5.22E-5	IEEE-STD-500
CHU-FR	Chiller fails to run (screening value)	5.42E-4	IEEE-STD-500 (same as CMP-FR)
CHU-FS	Chiller fails to start on demand (screening value)	3.93E-3	NUREG-1205 (same as MDP)
CKV-FC	Check valve fails closed	6.34E-4	Reliability Data Book
CKV-FO	Check valve fails open	3.44E-3	Reliability Data Book
CKV-PG	Check valve plugged during standby	1.25E-7	NUREG-2728 (Same as MOV-PG, but type 4)
CKV-SO	Check valve spuriously opens during mission	5.03E-7	NUREG-2728
CKV-SP	Check valve spuriously opens during standby	5.03E-7	NUREG-2728
CMP-FR	Compressor fails to run	5.42E-4	IEEE-STD-500
CMP-FS	Compressor fails to start	1.98E-3	NUREG-1205 (Same as PAT-FS)
DDP-FR	Diesel driven pump fails to run	7.99E-4	NUREG-4550 and Industry Survey
DDP-FS	Diesel driven pump fails to start	2.05E-2	Surry Plant Specific Data

TABLE 3.3.1-4 (Continued)
GENERIC DATA FOR EQUIPMENT FAILURES AND UNAVAILABILITIES

Identifier	Failure Description	Mean Probability	Data Source
EDG-FR	Emergency Diesel Generator fails to run	2.23E-3	NUREG-2989
EDG-FS	Emergency Diesel Generator fails to run (screening value)	2.41E-2	NUREG-2989
EHR-LF	Electric heater loss of function during standby	1.18E-5	IEEE-STD-500
FAN-FR	Motor driven ventilation fan fails to run (screening value)	3.30E-5	NUREG-1205 (Same as PSB-FR)
FAN-FS	Motor driven ventilation fan fails to start	3.93E-3	NUREG-1205 (Same as PSB-FS)
FCV-FC	Flow control valve fails closed	1.81E-2	NUREG-1363 (Same as AOV-FC)
FCV-FO	Flow control valve fails open	1.81E-2	NUREG-1363 (Same as AOV-FO)
FCV-LF	Flow control valve loss of function	1.81E-2	NUREG-1363 (Same as AOV-LF)
FCV-LU	Flow control valve loss of function during mission	7.61E-4	Same as AOV-LF assuming 24 hr mission
FCU-SO	Flow control valve spurious opening	5.03E-7	NUREG-2728 (Same as MOV-SO)
FEL-PG	Flow element plugged during standby	1.25E-7	NUREG-2728 (Same as MOV-PG)
FEL-PL	Flow element plugged during mission	1.25E-7	NUREG-2728 (Same as MOV-PG)

TABLE 3.3.1-4 (Continued)
GENERIC DATA FOR EQUIPMENT FAILURES AND UNAVAILABILITIES

Identifier	Failure Description	Mean Probability	Data Source
FIC-LF	Flow channel loss of function during standby	1.29E-5	NUREG-1740
FLT-PG	Filter plugged during standby	2.66E-5	NUREG-2728
FUS-SO	Fuse spurious opening during mission	2.66E-6	NUREG-2728
HCV-FC	Hand control valve fails closed	1.81E-2	NUREG-1363 (Same as AOV-FC)
HCV-FO	Hand control valve fails open	1.81E-2	NUREG-1363 (Same as AOV-FO)
HCV-LF	Hand control valve loss of function	1.81E-2	NUREG-1363 (Same as AOV-LF)
HCV-LU	Hand control valve loss of function during mission	7.61E-4	NUREG-1363 (Same as AOV-LF assuming 24hr mission)
HCV-PG	Normally open hand control valve plugged during standby	1.25E-7	NUREG-2728 (Same as MOV-PG)
HCV-SC	Hand control valve spurious closing	5.03E-7	NUREG-2728 (Same as MOV-SC)
HCV-SO	Hand control valve spurious opening	5.03E-7	NUREG-2728 (Same as MOV-SC)
HEX-LF	Heat exchanger loss of function during standby	8.71E-6	NUREG-4550, Vol 1
HEX-LU	Heat exchanger loss of function during mission	8.71E-6	NUREG-4550, Vol 1
HEX-PG	Heat exchanger plugged during standby	5.70E-6	NUREG-4550, Vol 1

TABLE 3.3.1-4 (Continued)
GENERIC DATA FOR EQUIPMENT FAILURES AND UNAVAILABILITIES

Identifier	Failure Description	Mean Probability	Data Source
HS--LF	Hand switch loss of function	2.66E-5	NUREG-2728
INV-LU	Inverter loss of use during mission	2.56E-5	NUREG-3867
LCV-FC	Level control valve fails closed	1.81E-2	NUREG-1363 (Same as AOV-FC)
LCV-FO	Level control valve fails open	1.81E-2	NUREG-1363 (Same as AOV-FO)
LCV-LF	Level control valve loss of function	1.81E-2	NUREG-1363 (Same as AOV-LF)
LIC-LF	Level channel loss of function during standby	1.29E-5	NUREG-1740
LIC-LU	Level switch loss of use	1.25E-4	NUREG-2728
LMS-LF	Limit switch loss of function	1.25E-4	NUREG-2728
MOD-FC	Motor operated damper fails closed	1.09E-2	NUREG-1363 (Same as MOV-FC)
MOD-FO	Motor operated damper fails open	1.09E-2	NUREG-1363 (Same as MOV-FO)
MOD-SC	Motor operated damper spurious closing	5.03E-7	NUREG-2728 (Same as MOV-SC)
MOV-FC	Motor operated valve fails closed	1.09E-2	NUREG-1363
MOV-FO	Motor operated valve fails open	1.09E-2	NUREG-1363
MOV-PG	Normally open motor operated valve plugged during standby	1.25E-7	NUREG-2728
MOV-SC	Motor operated valve spurious closing	5.03E-7	NUREG-2728 (Same as MOV-SO)

TABLE 3.3.1-4 (Continued)
GENERIC DATA FOR EQUIPMENT FAILURES AND UNAVAILABILITIES

Identifier	Failure Description	Mean Probability	Data Source
MOV-SO	Motor operated valve spurious opening	5.03E-7	NUREG-2728
MV--FC	Manual valve fails closed	1.25E-4	NUREG-2728
MV--FO	Manual valve fails open	1.25E-4	NUREG-2728
MV--PG	Normally open manual valve plugged during standby	1.25E-7	NUREG-2728 (Same as MOV-PG)
MV--SO	Manual valve spuriously opens during mission	5.03E-7	NUREG-2728 (Same as MOV-SO)
MVD-FC	Motor operated vent damper fails closed	1.09E-2	NUREG-1363 (Same as MOV-FC)
MVD-FO	Motor operated vent damper fails open	1.09E-2	NUREG-1363 (Same as MOV-FO)
MVD-SC	Motor operated vent damper spurious closing	1.25E-7	NUREG-2815
MVD-SO	Motor operated vent damper spurious opening	5.03E-7	NUREG-2728 (Same as MOV-SO)
PAT-FR	Motor driven alternating pump fails to run	3.31E-5	NUREG-1205
PAT-FS	Motor driven alternating pump fails to start	1.98E-3	NUREG-1205
PCV-FC	Pressure control valve fails closed	1.81E-2	NUREG-1363 (Same as AOV-FC)
PCV-FO	Pressure control valve fails open	1.81E-2	NUREG-1363 (Same as AOV-FO)

TABLE 3.3.1-4 (Continued)
GENERIC DATA FOR EQUIPMENT FAILURES AND UNAVAILABILITIES

Identifier	Failure Description	Mean Probability	Data Source
PCV-LF	Pressure control valve loss of function	1.81E-2	NUREG-1363 (Same as AOV-LF)
PCV-PG	Normally open pressure control valve plugged during standby	1.25E-7	NUREG-2728 (Same as MOV-PG)
PCV-SC	Pressure control valve spurious closing	5.03E-7	NUREG-2728 (Same as MOV-SC)
PCV-SO	Pressure control valve spurious opening	5.03E-7	NUREG-2728 (Same as MOV-SO)
PIC-LF	Pressure channel loss of function during standby	1.29E-5	NUREG-1740
PSB-FR	Motor driven standby pump fails to run	3.30E-5	NUREG-1205
PSB-FS	Motor driven standby pump fails to start	3.93E-3	NUREG-1205
RLY-LF	Relay loss of function	2.66E-4	NUREG-2728
RLY-SO	Relay spurious operation	2.66E-6	NUREG-2728
RST-LP	Reserve station transformer fails to supply power	7.91E-7	Reliability Data Book
RV--FC	Power operated relief valve fails closed	9.99E-3	WCAP-9804
RV--FO	Power operated relief valve fails open	2.50E-2	NUREG-4550
RV--SO	Power operated relief valve spurious opening	3.89E-6	NUREG-1363
SOD-FC	Solenoid operated damper fails closed	1.81E-2	NUREG-1363 (Same as SOV-FC)

TABLE 3.3.1-4 (Continued)
GENERIC DATA FOR EQUIPMENT FAILURES AND UNAVAILABILITIES

Identifier	Failure Description	Mean Probability	Data Source
SOD-FO	Solenoid operated damper fails open	1.81E-2	NUREG-1363 (Same as SOV-FO)
SOD-SC	Solenoid operated damper spurious closing	1.25E-7	NUREG-2815 (Same as MOV-SC)
SOD-SO	Solenoid operated damper spurious opening	5.03E-7	NUREG-2728 (Same as SOV-SO)
SOV-FC	Solenoid valve fails closed	1.81E-2	NUREG-1363 (Same as AOV-FC)
SOV-FO	Solenoid valve fails closed	1.81E-2	NUREG-1363 (Same as AOV-FO)
SOV-PG	Normally open solenoid valve plugged during standby	1.27E-7	NUREG-2815 (Same as AOV-PG)
SOV-SC	Solenoid operated valve spurious closing	5.03E-7	NUREG-2728 (Same as MOV-SC)
SOV-SO	Solenoid operated valve spurious opening	5.03E-7	NUREG-2728 (Same as MOV-SO)
SST-LP	Station service transformer fails to supply power	7.91E-7	Reliability Data
STR-PG	Strainer plugged during standby	2.66E-5	NUREG-2728
STR-PL	Strainer plugged during mission	2.66E-5	NUREG-2728 (STR-PG assuming 24hr mission)
SV--FC	Safety valve fails closed	1.25E-5	NUREG-2728
SV--FO	Safety valve fails open	1.25E-2	NUREG-2728

TABLE 3.3.1-4 (Continued)
GENERIC DATA FOR EQUIPMENT FAILURES AND UNAVAILABILITIES

Identifier	Failure Description	Mean Probability	Data Source
SV--SO	Safety valve spurious opening	3.89E-6	NUREG-1363 (Same as RV--SO)
TCV-FC	Temperature control valve fails closed	1.81E-2	NUREG-1363 (Same as AOV-FC)
TCV-FO	Temperature control valve fails open	1.81E-2	NUREG-1363 (Same as AOV-FO)
TCV-LF	Temperature control valve loss of function	1.81E-2	NUREG-1363 (Same as AOV-LF)
TCV-SC	Temperature control valve spurious closing	5.03E-7	NUREG-2728 (Same as AOV-SC)
TFM-LP	Transformer fails to supply power during mission	7.91E-7	Reliability Data Book
TIC-LF	Temperature channel loss of function during standby	9.33E-5	NUREG-1740
TMR-LF	Timer loss of function	7.99E-4	NUREG-2728
TNK-LF	Tank loss of function on demand	2.66E-6	NUREG-4550, Vol 3
TNK-LU	Tank loss of function during mission	1.11E-7	NUREG-4550, Vol 3 (Based on TNK-LF with 24 hr mission)
TRB-FR	Turbine driven pump fails to run	4.93E-3	NUREG-4550, Vol 1
TRB-FS	Turbine driven pump fails to start	4.21E-2	NUREG-1205, Rev 1
TRU-LF	Trip unit loss of function	2.66E-6	NUREG-2728

TABLE 3.3.1-4 (Continued)
GENERIC DATA FOR EQUIPMENT FAILURES AND UNAVAILABILITIES

Identifier	Failure Description	Mean Probability	Data Source
Component Unavailabilities Due to Testing and Maintenance:			
ACU-UM	Air Handling Unit (Air Conditioning) unscheduled maintenance (screening value)	4.53E-4	Surry IPE PRA
AOV-TM	Air operated valve scheduled test and maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-TM)
AOV-UM	Air operated valve unscheduled maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-UM)
BUS-UM	Bus bar unscheduled maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-UM)
CHU-UM	Chiller unscheduled maintenance screening value)	3.50E-2	Surry IPE PRA
EDG-TM	Emergency Diesel Generator scheduled test and maintenance (screening value)	1.00E-2	NUREG-4550, Vol 1
EDG-UM	Emergency Diesel Generator unscheduled maintenance (screening value)	1.00E-2	NUREG-4550, Vol 1
FCV-TM	Flow control valve scheduled test and maintenance	2.00E-4	NUREG4550, Vol 1 (Same as MOV-TM)
FCV-UM	Flow control valve unscheduled maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-UM)
HCV-TM	Hand control valve scheduled test and maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-TM)

TABLE 3.3.1-4 (Continued)
GENERIC DATA FOR EQUIPMENT FAILURES AND UNAVAILABILITIES

Identifier	Failure Description	Mean Probability	Data Source
HCV-UM	Hand control valve unscheduled maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-UM)
HEX-UM	Heat exchanger unscheduled maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-UM)
LCV-TM	Level control valve scheduled test and maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-TM)
LCV-UM	Level control valve unscheduled maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-UM)
MOV-TM	Motor operated valve scheduled test and maintenance	2.00E-4	NUREG-4550, Vol 1
MOV-UM	Motor operated valve unscheduled maintenance	2.00E-4	NUREG-4550, VOL 1
PAT-TM	Motor driven alternating pump scheduled test and maintenance	3.75E-3	NUREG-4550, Vol 1
PAT-UM	Motor driven alternating pump unscheduled maintenance	3.75E-3	NUREG-4550, Vol 1
PCV-TM	Pressure control valve scheduled test and maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-TM)
PCV-UM	Pressure control valve unscheduled maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-UM)
PSB-TM	Motor driven standby pump scheduled test and maintenance	3.75E-3	NUREG-4550, Vol 1 (Same as PAT-TM)

TABLE 3.3.1-4 (Continued)
GENERIC DATA FOR EQUIPMENT FAILURES AND UNAVAILABILITIES

Identifier	Failure Description	Mean Probability	Data Source
PSB-UM	Motor driven standby pump unscheduled maintenance	3.75E-3	NUREG-4550, Vol 1 (Same as PAT-UM)
SOV-TM	Solenoid operated valve scheduled test and maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-TM)
SOV-UM	Solenoid operated valve unscheduled maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-UM)
TCV-TM	Temperature control valve scheduled test and maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-TM)
TCV-UM	Temperature control valve unscheduled maintenance	2.00E-4	NUREG-4550, Vol 1 (Same as MOV-UM)
TRB-TM	Turbine driven pump scheduled test and maintenance (screening value)	1.00E-2	NUREG-4550, Vol 1
TRB-UM	Turbine driven pump unscheduled maintenance (screening value)	1.00E-2	NUREG-4550, Vol 1

TABLE 3.3.2-1
DATA REDUCTION FOR INITIATING EVENTS

I.E. Group Name	No. of Events	No. of Years
T2, Loss of MFW	0	10
T2A, Recoverable Loss of MFW	5	10
T3, Transient with MFW Available	13	10
TH, High Power Transients (ATWS)	17[1]	10
TL, Low Power Transients (ATWS)	3	10

1 Talley includes T9-related events not included within T2, T2A, or T3 data.

TABLE 3.3.2-2
FAILURE DATA

Component	Mode	# of failures	Exposure (in # demands or hours)
Demand Faults			
ACU, Air Handling Unit 1-HV-AC-1/2/6/7 2-HV-AC-6/7/8/9	LF, loss of function	0	350,400 hrs[10]
CHU, Air Conditioning Chiller 1-HV-E-4A/B/C 2-HV-E-4A/B/C	FR, fails to run	5	87,600 hrs[8]
CHU, Air Conditioning Chiller 1-HV-E-4A/B/C 2-HV-E-4A/B/C	FS, fails to start	5	120
EDG, Emergency Diesel Gen. 1-EG-EDG-1H/1J 2-EG-EDG-2H/2J	FS, fails to start	5	394
FAN, Air Handling Unit 1-HV-AC-6/7 2-HV-AC-6/7	FR, fails to run	1	175,200 hrs[9]
PAT, Alternating Pump 1-CH-P-1A/B/C 2-CH-P-1A/B/C	FS, fails to start	2	190
PAT, Alternating Pump 1-SW-P-1A/B 2-SW-P-1A/B	FS, fails to start	1	89
PSB, Standby Pump 1-FW-P-3A/B 2-FW-P-3A/B	FS, fails to start	0	264

TABLE 3.3.2-2 (Continued)
FAILURE DATA

Component	Mode	# of failures	Exposure (in # demands or hours)
Running Faults			
PSB, Standby Pump 1-RS-P-1A/1B/2A/2B 1-SI-P-1A/B 2-RS-P-1A/1B/2A/2B 2-SI-P-1A/B	FS, fails to start	1	245
PSB, Standby Pump 1-SW-P-4 2-SW-P-4	FS, fails to start	0	44
TRB, turbine-driven pump 1-FW-P-2 2-FW-P-2	FS, fails to start	2	129

See Table C.5-5 in Appendix C for definition of Operating Exposure

TABLE 3.3.2-3
UNAVAILABILITY DUE TO MAINTENANCE

Component/Train	Total Unavailable Time (hrs.)	Operating Exposure (hrs)
1-CH-P-1A/B/C 2-CH-P-1A/B/C any 1 of 3 any 2 of 3	 23,113 54	 70,727[1] 70,727[1]
1-EG-EDG-1J/1H (assigned EDG)	2,520	141,454[2]
1-EG-EDG-2J/2H (other Unit's EDG)	15,120*	141,454[2]
1-FW-P-2 2-FW-P-2	967	70,727[1]
1-FW-P-3A/B 2-FW-P-3A/B	734	141,454[2]
1-HV-AC-6/7 2-HV-AC-6/7	234	141,454[2]
1-HV-E-4A/B/C 2-HV-E-4A/B/C 1 of 3 2 of 3	 20,034 480	 212,181[3] 212,181[3]
1-RS-P-1A/1B/2A/2B 1-SI-P-1A/B 2-RS-P-1A/1B/2A/2B 2-SI-P-1A/B	1,908	424,362[4]
1-SW-P-1A/B 2-SW-P-1A/B	6,059	162,640[7]
1-SW-P-4 2-SW-P-4	6,742	81,320[6]
SW Supply Discharge Piping Headers (pooled for Unit 1 and Unit 2)	1,614	141,454[2]

See Table C.5-5 in Appendix C for definition of Operating Exposure
*Estimated unavailability from North Anna assigned EDG data, and the Surry IPE PRA ratio of other Unit's EDG to assigned EDG.

TABLE 3.3.2-4
INITIATING EVENT FREQUENCIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identified	Description	Mean Frequency	Source/ Comment
T2	Loss of MFW	5.0E-2/yr	Review of LERs
T2A	Recoverable Loss of MFW	5.5E-1/yr	
T3	Transient with MFW Available	1.35E+0/yr	
T4	Loss of RC Pump Seal Cooling	6.0E-7/yr	Fault Tree Model
T6	Loss of Service Water	6.27E-6/yr	Fault Tree Model
T8	Loss of Emergency Switchgear Room Cooling	6.58E-3/yr	Fault Tree Model
T9A	Loss of Power from 4160 V 1H Bus	1.79E-2/yr	Fault Tree Model see Note 1
T9B	Loss of Power from 4160 V 1J Bus	1.79E-2/yr	Fault Tree Model see Note 1
TH	Transient > 40% power	1.75E+0/yr	Derived from T2, T2A, T3 & T9
TL	Transient < 40% power	3.50E-1/yr	

NOTE 1: The quantification values of T9A and T9B are provided as an example only. The actual T9A and T9B fault trees are solved within the T9A and T9B event trees, to further develop system dependencies.

TABLE 3.3.2-5
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
Bayesian Updates of Component Data			
1CHPAT-FS-1CHP1B[1]	Charging Pump 1-CH-P-1B fails to start	4.97E-3	CH-P-1 Plant Specific Data
1CHPAT-FS-1CHP1C[1]	Charging Pump 1-CH-P-1C fails to start	4.97E-3	CH-P-1 Plant Specific Data
1EGEDG-FS-1H	Emergency Diesel Generator 1-EG-EDG-1H fails to start	1.43E-2	EDG Plant Specific Data
1EGEDG-FS-1J	Emergency Diesel Generator 1-EG-EDG-1J fails to start	1.43E-2	EDG Plant Specific Data
1FWPSB-FS-1FWP3A	Motor Driven AFW Pump 1-FW-P-3A fails to start	1.58E-3	FW-P-3 Plant Specific Data
1FWPSB-FS-1FWP3B	Motor Driven AFW Pump 1-FW-P-3B fails to start	1.58E-3	FW-P-3 Plant Specific Data
1FWTRB-FS-1FWP2	Turbine-Driven AFW Pump 1-FW-P-2 fails to start	1.86E-2	FW-P-2 Plant Specific Data
1HVACU-LF-1HVAC6	1-HV-AC-6 AHU loss of function	3.42E-5	1-HV-AC-1/2/6/7 and 2-HV-AC-6/7/8/9 Plant Specific Data for North Anna and Surry

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1HVACU-LF-1HVAC7	1-HV-AC-7 AHU loss of function	3.42E-2	1-HV-AC-1,2,6&7 and 2-HV-AC-6,7,8&9 Plant Specific Data for North Anna and Surry
1HVCHU-FR-1HVE4A	Chiller 1-HV-E-4A fails to run	1.51E-3	HV-E-4 Plant Specific Data
1HVCHU-FR-1HVE4B	Chiller 1-HV-E-4B fails to run	1.51E-3	HV-E-4 Plant Specific Data
1HVCHU-FR-1HVE4C	Chiller 1-HV-E-4C fails to run	1.51E-3	HV-E-4 Plant Specific Data
1HVCHU-FS-1HVE4B	Chiller 1-HV-E-4B fails to start	4.55E-2	HV-E-4 Plant Specific Data
1HVCHU-FS-1HVE4C	Chiller 1-HV-E-4C fails to start	4.55E-2	HV-E-4 Plant Specific Data
1HVFAN-FR-1FM06	1-HV-AC-6 Fan motor fails to run	1.36E-4	1-HV-AC-1/2/6/7 and 2-HV-AC-6/7/8/9 Plant Specific Data with PAT-FR prior
1HVFAN-FR-1FM07	1-HV-AC-7 Fan motor fails to run	1.36E-4	1-HV-AC-1/2/6/7 and 2-HV-AC-6/7/8/9 Plant Specific Data with PAT-FR prior
1RSPSB-FS-1RSP1A	IRS pump 1-RS-P-1A fails to start	3.93E-3	Pooled RS-P-1, RS-P-2 and SI-P-1 Plant Specific Data
1RSPSB-FS-1RSP1B	IRS pump 1-RS-P-1A fails to start	3.93E-3	Pooled RS-P-1, RS-P-2 and SI-P-1 Plant Specific Data

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1RSPSB-FS-1RSP2A	ORS pump 1-RS-P-2A fails to start	3.93E-3	Pooled RS-P-1, RS-P-2 and S1-P1 Plant Specific Data
1RSPSB-FS-1RSP2B	ORS pump 1-RS-P-2B fails to start	3.93E-3	Pooled RS-P-1, RS-P-2 and SI-P1 Plant Specific Data
1SIPSB-FS-1SIP1A	LHSI pump 1-SI-P-1A fails to start	3.93E-3	Pooled RS-P-1, RS-P-2 and SI-P-1 Plant Specific Data
1SIPSB-FS-1RSP1B	LHSI pump 1-SI-P-1B fails to start	3.93E-3	Pooled RS-P-1, RS-P-2 and SI-P-1 Plant Specific Data
1SWPAT-FS-1SWP1B	Service water pump 1-SW-P-1B fails to start	3.84E-3	SW-P-1 Plant Specific Data
1SWPSB-FS-1SWP-4	Auxiliary service water pump 1-SW-P-4 fails to start	3.15E-3	SW-P-4 Plant Specific Data
2EGEDG-FS-2H	Emergency Diesel Generator 2-EG-EDG-2H fails to start	1.43E-2	EDG Plant Specific
2EGEDG-FS-2J	Emergency Diesel Generator 2-EG-EDG-2J	1.43E-2	EDG Plant Specific Data
2HVACU-LF-2HVAC6	2-HV-AC-6 loss of function	3.42E-5	1-HV-AC-1,2,6&7 and 2-HV-AC-6, 7,8&9 Plant Specific Data for North Anna and Surry

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
2HVACU-LF-2HVAC7	2-HV-AC-7 loss of function	3.42E-5	1-HV-AC-1/2/6/7 and 2-HV-AC-6/7/8/9 Plant Specific Data for North Anna and Surry
2HVCHU-FR-2HVE4A	Chiller 2-HV-E-4A fails to run	1.51E-3	HV-E-4 Plant Specific Data
2HVCHU-FR-2HVE4B	Chiller 2-HV-E-4B fails to run	1.51E-3	HV-E-4 Plant Specific Data
2HVCHU-FR-2HVE4C	Chiller 2-HV-E-4C fails to run	1.51E-3	HV-E-4 Plant Specific Data
2HVCHU-FS-2HVE4B	Chiller 2-HV-E-4B fails to run	4.55E-2	HV-E-4 Plant Specific Data
2HVCHU-FS-2HVE4C	Chiller 2-HV-E-4C fails to run	4.55E-2	HV-E-4 Plant Specific Data
2HVFAN-FR-2FM06	2-HV-AC-6 Fan motor fails to run	1.36E-4	1-HV-AC-1/2/6/7 and 2-HV-AC-6/7/8/9 Plant Specific Data with PAT-FR prior
2HVFAN-FR-2FM07	2-HV-AC-7 Fan motor fails to run	1.36E-4	1-HV-AC-1/2/6/7 and 2-HV-AC-6/7/8/9 Plant Specific Data with PAT-FR prior
2SWPAT-FS-2SWP1B	Service Water Pump 2-SW-P-1B fails to start	3.84E-3	SW-P-1 Plant Specific Data

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
T9A-FREQ-500KV-1	Frequency of 500 KV Bus 1 fault causing loss of power to 4160 V Bus 1H and Reactor Trip	1.79E-1	Used for T9A Initiating Event Fault Tree - Plant Specific Data
T9A-FREQ-RSST-C	Frequency of RSST C fault causing loss of power to 4160 V Bus 1H and Reactor Trip	7.14E-2	Used for T9A Initiating Event Fault Tree - Plant Specific Data
T9B-FREQ-500KV-2	Frequency of 500 KV Bus 2 fault causing loss of power to 4160 V Bus 1J and Reactor Trip	1.79E-1	Used for T9B Initiating Event Fault Tree - Plant Specific Data
T9B-FREQ-RSST-A	Frequency of RSST A fault causing loss of power to 4160 V Bus 1J and Reactor Trip	7.14E-2	Used for T9B Initiating Event Fault Tree - Plant Specific Data
Component Unavailabilities Due to Scheduled Testing and Unscheduled Maintenance			
1CHPAT-UM-1CHPBC	Dual Charging Pump 1-CH-P-1B/1C unscheduled maintenance	7.53E-4	All dual charging data within a unit pooled for both units
1CHPAT-UM-1CHP1C	Charging Pump 1-CH-P-1C unscheduled maintenance	3.27E-1	CH-P-1 pooled data

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1EEBUS-UM-1H	4160 V Bus 1H unscheduled maintenance	1.00E-5	Pooled bus data
1EEBUS-UM-1H-480	480 V Bus 1H unscheduled maintenance	1.00E-5	Pooled bus data
1EEBUS-UM-1H1	480 V Bus 1H1 unscheduled maintenance	1.00E-5	Pooled bus data
1EEBUS-UM-1H1-1	480 V Bus 1H1-1 unscheduled maintenance	1.00E-5	Pooled bus data
1EEBUS-UM-1H1-2S	480 V Bus 1H1-2S unscheduled maintenance	1.00E-5	Pooled bus data
1EEBUS-UM-1H1-4	480 V Bus 1H1-4 unscheduled maintenance	1.00E-5	Pooled bus data
1EEBUS-UM-1HSTUB	4160 V Stub Bus 1H unscheduled maintenance	1.00E-5	Pooled bus data
1EEBUS-UM-1J	4160 V Bus 1J unscheduled maintenance	1.00E-5	Pooled bus data
1EEBUS-UM-1J-480	480 V Bus 1J unscheduled maintenance	1.00E-5	Pooled bus data
1EEBUS-UM-1J1	480 V Bus 1J1 unscheduled maintenance	1.00E-5	Pooled bus data
1EEBUS-UM-1J1-1	480 V Bus 1J1-1 unscheduled maintenance	1.00E-5	Pooled bus data

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1EEBUS-UM-1J1-2	480 V Bus 1J1-2N and 2S unscheduled maintenance	1.00E-5	Pooled bus data
1EEBUS-UM-1JSTUB	4160 V Stub Bus 1J unscheduled maintenance	1.00E-5	Pooled bus data
1EGEDG-TM-1H	1-EG-EDG-1H unavailable due to scheduled testing	5.71E-4	Estimate from Plant Procedures
1EGEDG-TM-1J	1-EG-EDG-1J unavailable due to scheduled testing	5.71E-4	Estimate from Plant Procedures
1EGEDG-UM-1H	1EG-EDG-1H unscheduld maintenance (assigned diesel)	1.78E-2	EDG Pooled Data
1EGEDG-UM-1J	1-EG-EDG-1J unscheduled maintenance (assigned diesel)	1.78E-2	EDG Pooled Data
1FWPSB-TM-1FWP3A	Motor Driven AFW Pump 1-FW-P-3A scheduled testing	1.40E-3	Plant Specific Data
1FWPSB-TM-1FWP3B	Motor Driven AFW Pump 1-FW-P-3A scheduled testing	1.40E-3	Plant Specific Data

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1FWPSB-UM-1FWP3A	Motor Driven AFW Pump 1-FW-P-3A unscheduled maintenance	5.18E-3	FW-P-3 pooled data
1FWPSB-UM-1FWP3B	Motor Driven AFW Pump 1-FW-P-3B unscheduled maintenance	5.18E-3	FW-P-3 pooled data
1FWTRB-TM-1FWP2	Turbine-Driven AFW Pump scheduled testing 1-FW-P-2	1.40E-3	Plant Specific Data
1FWTRB-UM-1FWP2	Turbine-Driven AFW Pump 1-FW-P-2 unscheduled maintenance	1.37E-2	FW-P-2 pooled data
1HVACU-UM-1HVAC7	ESGR Air Handling Unit 1-HV-AC-7 unscheduled maintenance	1.65E-3	HV-AC-6/7 pooled data
1HVCHU-UM-1HVE4B	Chiller 1-HV-E-4B unscheduled maintenance	9.44E-2	HV-E-4 pooled data
1HVCHU-UM-1HVE4C	Chiller 1-HV-E-4C unscheduled maintenance	9.44E-2	HV-E-4 pooled data
1HVCHU-UM-HVE4BC	Dual Chiller 1-HV-E-4B & 4C unscheduled maintenance	2.26E-3	HV-E-4 dual chiller data within a unit pooled for both units
1RSPSB-TM-1RSP2A	ORS pump 1-RS-P-2A scheduled testing	4.92E-3	Plant Specific Data

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1RSPSB-TM-1RSP2B	ORS pump 1-RS-P-2B scheduled testing	4.92E-3	Plant Specific Data
1RSPSB-UM-1RSP1A	IRS pump 1-RS-P-1A unscheduled maintenance	4.54E-3	RS-P-1, RS-P-2 and SI-P-1 pooled data
1RSPSB-UM-1RSP1B	IRS pump 1-RS-P-1B unscheduled maintenance	4.54E-3	RS-P-1, RS-P-2 and SI-P-1 pooled data
1RSPSB-UM-1RSP2A	ORS pump 1-RS-P-2A unscheduled maintenance	4.54E-3	RS-P-1, RS-P-2 and SI-P-1 pooled data
1RSPSB-UM-1RSP2B	ORS pump 1-RS-P-2B unscheduled maintenance	4.54E-3	RS-P-1, RS-P-2 and SI-P-1 pooled data
1SIPSB-UM-1SIP1A	LHSI pump 1-SI-P-1A unscheduled maintenance	4.54E-3	RS-P-1, RS-P-2 and SI-P-1 pooled data
1SIPSB-UM-1SIP1B	LHSI pump 1-SI-P-1B unscheduled maintenance	4.54E-3	RS-P-1, RS-P-2 and SI-P-1 pooled data
1SWPAT-UM-1SWP1B	Service Water Pump 1-SW-P-1B unscheduled maintenance	3.73E-2	SW-P-1 pooled data

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1SWPIP-UM-HDRA	Service Water Header A in scheduled maintenance	2.28E-2	Pooled SW header data averaged for Unit 1 and Unit 2 exposure
1SWPIP-UM-HDRB	Service Water Header B in scheduled maintenance	2.28E-2	Pooled SW header data averaged for Unit 1 and Unit 2 exposure
1SWPSB-UM-1SWP-4	Auxiliary Service Water Pump 1-SW-P-4 unscheduled maintenance	8.29E-2	SW-P-4 pooled data
2EGEDG-TM-2H	2-EG-EDG-2H unavailable due to scheduled testing	5.71E-4	Estimate from Plant Procedures
2EGEDG-TM-2J	2-EG-EDG-2J unavailable due to scheduled testing	5.71E-4	Estimate from Plant Procedures
2EGEDG-UM-2H	2-EG-EDG-2H unscheduled maintenance (other Unit's diesel)	1.07E-1	EDG pooled data for assigned EDG multiplied by Surry IPE PRA ratio, other Unit's/assigned EDG

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
2EGEDG-UM-2J	2-EG-EDG-2J unscheduled maintenance (other Unit's diesel)	1.07E-1	EDG pooled data for assigned EDG multiplied by Surry IPE PRA ratio, other Unit's/assigned EDG
2HVACU-UM-2HVAC6	Air handling unit 2-HV-AC-6 unscheduled maintenance	1.65E-3	HV-AC-6/7 pooled data
2HVCHU-UM-2HVE4B	Chiller 2-HV-E-4B unscheduled maintenance	9.44E-2	HV-E-4 pooled data
2HVCHU-UM-2HVE4C	Chiller 2-HV-E-4C unscheduled maintenance	9.44E-2	HV-E-4 pooled data
2HVCHU-UM-HVE4BC	Dual Chiller 2-HV-E-4B & 4C unscheduled maintenance	2.26E-3	Dual chiller data within a unit pooled for both units
2SWPAT-UM-2SWP1B	Service Water Pump 2-SW-P-1B unscheduled maintenance	3.73E-2	SW-P-1 pooled data
Undeveloped Events Modeled as Basic Events			
Simplification to Single Dominant Event:			
1CWSCN-PL-1SWP-4	Auxiliary Service Water 1-SW-P-4 fails due to CW intake screen well plugging	6.39E-4	Simplified to STR-PL

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1CHCKV-FO-1CH254	Check valve fails open, weighted for 1 of 3 charging pumps	1.15E-3	1/3 CKV-FO modeling reduced exposure, since CHP A operated only 1/3 of the time
1FW-ACT-MFWP-A	No Trip Signal to MFW pumps - SI Train A	2.66E-4	Simplified to RLY-LF fault
1FW-ACT-MFWP-B	No Trip Signal to MFW pumps - SI Train B	2.66E-4	Simplified to RLY-LF fault
1FW-CONDHOTWELL	Insufficient inventory Condenser hot well	2.66E-6	Simplified to TNK-LF fault
1FWPAT-OIL1FWP1A	Main FW Pump Lube Oil System fault 1-FW-P-1A	3.31E-5	Simplified to PAT-FR & 24 hr mission
1FWPAT-OIL1FWP1B	Main FW Pump Lube Oil System fault 1-FW-P-1B	3.31E-5	Simplified to PAT-FR & 24 hr mission
1FWPAT-OIL1FWP1C	Standby MFW pump Lube Oil System start and run fault 1-FW-P-1C	2.78E-3	Simplified to PAT-FS and PAT-FR with 24 hr mission
1MS-ACT-SGBDTV	No Signal from AFW pumps or from Containment Isolation to close Stm. Gen. A Blowdown Trip valve	2.66E-4	Simplified to RLY-LF fault
1MS-ACT-TV101A	No Trip Signal to Main Steam Trip Valve	2.66E-4	Simplified to RLY-LF fault

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1RPBKR-CC-MGAMGB	Common cause fault 2/2 CRDM M-G set supply breaker failed closed	1.83E-4	Simplified to 10% Beta Factor model of BKR-FC
1RPBKR-LF-MGA	CRDM M-G set A supply breaker fails closed	1.83E-3	Simplified to BKR-FC
1RPBKR-LF-MGB	CRDM M-G set B supply breaker fails closed	1.83E-3	Simplified to BKR-FC
1SISV--MC-1845A	LHSI pump relief valve fails open	3.75E-5	Simplified to SV--FO with generic HEP for recovery
1SISV--MC-1845C	LHSI pump relief valve fails open	3.75E-5	Simplified to SV--FO with generic HEP for recovery
1SWSCN-PG-1SWP1B	Service Water pump 1-SW-P-1B fails due to SW Reservoir screenwell plugging	2.66E-5	Simplified to STR-PG
1SWSCN-PG-2SWP1B	Service Water pump 2-SW-P-1B fails due to SW Reservoir screenwell plugging	2.66E-5	Simplified to STR-PG
1SWSCN-PL-1SWP1A	Service Water pump 1-SW-P-1A fails during mission due to SW Reservoir screenwell plugging	6.39E-4	Simplified to STR-PL

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
Derived from Industry Documents:			
1CESTR-CC-SUMPPG	Blockage of Containment sump	5.00E-5	NUREG-4550, Vol. 3
1EP-LOOP-24	Loss of Offsite Power within 24 hrs of reactor trip	3.12E-4	Based on NUREG-5032 T1, Common Incidence Rate Model
1FWCKV-LEAKAGE	Steam bound AFW Pumps, undetected leakage thru CKVs 1-FW-68, 100 and 132	1.00E-5	Based on AEOD/CH04, NUREG/CR-4550, and North Anna AFW line alignment
1IAIAS-LF-CONTIA	Containment Instrument Air System loss of function	1.05E-5	NUREG-5472
1IAIAS-LF-OUTIA	Outside Containment Instrument Air System loss of function	1.05E-5	NUREG-5472
1RPBKR-CC-RTARTB	Common cause fault 2/2 reactor trip breaker failed closed	1.30E-5	WCAP-11993
1RPBKR-LF-RTA	Reactor trip breaker fails to open	3.38E-4	WCAP-10271
1RPBKR-LF-RTB	Reactor trip breaker fails to open	3.38E-4	WCAP-10271
1RPROD-LF-CRODS	Control rods fail to insert due to mechanical binding	1.80E-6	WCAP-11993

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1RPRPS-LF-INPUT	No input signal from reactor trip protection	1.40E-6	WCAP-11993
2IAIAS-LF-OUTIA	Outside Containment Instrument Air System loss of function	1.05E-5	NUREG-5472
T9A-FREQ-1460-1H	Frequency of 4160 V Bus 1H fault that also causes Reactor Trip	6.00E-3	Used for T9A Initiating Event Fault Tree - NUREG-4550, Vol 3
T9B-FREQ-4160-1J	Frequency of 4160 V Bus 1J fault that also causes Reactor Trip	6.00E-3	Used for T9B Initiating Event Fault Tree - NUREG-4550, Vol 3
Derived from Plant Data and Expert Opinion:			
1CHPAT-PT-14:2	Charging Pump 1-CH-P-1B PT-14.2 scheduled testing	7.00E-4	Expert opinion and Plant Procedures
1CHPAT-PT-14:3	Charging Pump 1-CH-P-1C PT-14.3 scheduled testing	7.00E-4	Expert opinion and Plant Procedures
1CICDA-TM-HIHI-A	CDA high high train A scheduled testing	1.40E-3	Expert opinion and Plant Procedures
1CICDA-TM-HIHI-B	CDA high high train B scheduled testing	1.40E-3	Expert opinion and Plant Procedures

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1MSMV--LK-1MS168	Condenser dump Train B blocked due to valve leakage	1.00E-2	Expert Opinion
1MSMV--LK-1MS179	Condenser dump Train A blocked due to valve leakage	1.00E-2	Expert Opinion
1MSMV--LK-1MS21	SG A atmospheric dump valve blocked due to leakage	4.00E-2	Expert Opinion
1MSMV--LK-1MS59	SG B atmospheric dump valve blocked due to leakage	4.00E-2	Expert Opinion
1MSMV--LK-1MS97	SG C atmospheric dump valve blocked due to leakage	4.00E-2	Expert Opinion
1QSLEV-TM-RWSTA	RWST Level Protection train A scheduled testing	1.40E-3	Expert opinion and Plant Procedures
1QSLEV-TM-RWSTB	RWST Level Protection train B scheduled testing	1.40E-3	Expert opinion and Plant Procedures
1RCMOV-LK-535536	Both RC PORVs blocked by 1-RC-MOV-1535&1536 due to PORV leakage	0.00E+0	Expert opinion; for dependent event model -- independent event model still allowed

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1RCMOV-LK-1535	RC PORV blocked by 1-RC-MOV-1535 due to leakage	2.50E-2	Expert Opinion
1RCMOV-LK-1536	RC PORV blocked by 1-RC-MOV-1536 due to leakage	2.50E-2	Expert Opinion
Derived from Calculations:			
1EE-BAT-I-2HR	Failure of Battery I at 2 hours	1.00E+0	Modeled as an absolute certainty of occurrence for future development
1EE-BAT-II-2HR	Failure of Battery II at 2 hours	1.00E+0	Modeled as an absolute certainty of occurrence for future development
1EE-BAT-III-2HR	Failure of Battery III at 2 hours	1.00E+0	Modeled as an absolute certainty of occurrence for future development
1EE-BAT-IV-2HR	Failure of Battery IV at 2 hours	1.00E+0	Modeled as an absolute certainty of occurrence for future development
1FWPSB-ACT2A	No signal from lo-lo water level on 2/3 SG train A	1.00E-7	Conservative estimate based upon Surry IPE PRA calculation

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1FWPSB-ACT2B	No signal from lo-lo water level on 2/3 SG train B	1.00E-7	Conservative estimate based upon Surry IPE PRA calculation
1FWPSB-ACT3A	No signal from lo-lo water level on 1/3 SG - RPS A	1.00E-7	Conservative estimate based upon Surry IPE PRA calculation
1FWPSB-ACT3B	No signal from lo-lo water level on 1/3 SG - RPS B	1.00E-7	Conservative estimate based upon Surry IPE PRA calculation
1FW-FIREMAIN	Insufficient water make-up from fire main	1.00E+0	Modeled as an absolute certainty of occurrence for future development. Basic event is no longer used.
1FW-SW-MAKEUP	Insufficient water make-up from Service Water System	1.00E+0	Modeled as an absolute certainty of occurrence for future development. Basic event is no longer used.
1MSPORV-DMDT7	Probability of SG atmospheric dump valve demand - T7	1.00E+0	Modeled as an absolute certainty of occurrence for future development
1MSSRV-DMDT7	Probability of demand of the MS safety valves during T7	4.00E-2	Based on MS atmospheric dump valve blockage

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1RCPORV-DMDSBO	Probability of RC PORV demand during SBO	2.00E-1	Calculation from RCPORV blockage and RETRAN analysis
1RCPORV-T3	Probability of RC PORV demand during T3	6.65E-3	Calculation from NUREG-4550, Vol 3 adjusted with Plant Specific Data
1RCPORV-DMDTWS	Probability of RC PORV demand during ATWS	1.00E+0	Modeled an an absolute certainty of occurrence for future development
1RCSRV-DMDATWS	Probability of all 3 RC safety valves demanded	1.00E+0	Modeled as an absolute certainty of occurrence for future development
1SW-COLDWEA-3MO	Probability of SW Discharge to Reservoir via Bypass	2.50E-1	Estimate of cold weather
1SW-HOTWEA-9MO	Probability of SW Discharge to Reservoir via Spray Arrays	7.50E-1	Estimate of weather not cold (e.g., hot)
2EE-BAT-I-2HR	Failure of Unit 2 Battery I at 2 hours	1.00E+0	Modeled as an absolute certainty of occurrence for future development

TABLE 3.3.2-5 (Continued)
EQUIPMENT FAILURE AND OTHER PROBABILITIES
ESTIMATED UTILIZING PLANT-SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
2EE-BAT-II-2HR	Failure of Unit 2 Battery II at 2 hours	1.00E+0	Modeled as an absolute certainty of occurrence for future development
2EE-BAT-III-2HR	Failure of Unit 2 Battery III at 2 hours	1.00E+0	Modeled as an absolute certainty of occurrence for future development
2EE-BAT-IV-2HR	Failure of Unit 2 Battery IV at 2 hours	1.00E+0	Modeled as an absolute certainty of occurrence for future development

1 1CHPAT-FS-1CHP1B and 1CHP1C basic events are adjusted to include a 1.1E-4 HEP, resulting in a basic event mean probability of 5.08E-3.

TABLE 3.3.3-1
SUMMARY OF OPERATOR ACTIONS
TYPE A HEPs: TEST & MAINTENANCE ACTIONS

Basic Event Name	Point Estimate	Description
1CHHEP-MOV-1270	1.1E-4	CRO STARTS CHARGING PUMP C WHEN SUCTION MOV-1270A IS CLOSED
1CHHEP-MOV-1275A	7.5E-4	CRO LEAVES CHARGING PUMP A RECIRC VALVE MOV-1275A CLOSED___
1CHHEP-MOV-1275B	7.5E-4	CRO LEAVES CHARGING PUMP B RECIRC VALVE MOV-1275B CLOSED___
1CHHEP-MOV-1373	7.5E-4	CRO LEAVES CHARGING PUMP RECIRC VALVE MOV-1373 CLOSED
1FWHEP-HCV-100C_	7.5E-4	AFW_PUMP_3A_NOT____ ALIGNED_TO_SG_C____ HCV_HEADER_HCV-100C
1FWHEP-MOV-100B_	7.5E-4	AFW_PUMP_3B_NOT____ ALIGNED_TO_SG_B____ MOV_HEADER_MOV-100B
1FWHEP-MOV-100D_	7.5E-4	AFW_PUMP_2_NOT_____ ALIGNED_TO_SG_A____ MOV_HEADER_MOV-100D
1FWHEP-1FW543___	7.5E-4	CRO_LEAVES 1-FW-P-2 RECIRC_VALVE_OPEN__ TO_ECST,_1-FW-543__
1FWHEP-1FW546___	7.5E-4	CRO_LEAVES FW-P-3B_ RECIRC_VALVE_OPEN__ TO_ECST,_1-FW-546__
1FWHEP-1FW548___	7.5E-4	CRO_LEAVES FW-P-3A_ RECIRC_VALVE_OPEN__ TO_ECST,_1-FW-548__
1QSHEP-FLANGE___	3.8E-4	CRO_LEAVES_QS_SPRAY HEADER_FLANGE______ INSTALLED_1-PT-63.3

TABLE 3.3.3-1 (Continued)
SUMMARY OF OPERATOR ACTIONS
TYPE A HEPs: TEST & MAINTENANCE ACTIONS

Basic Event Name	Point Estimate	Description
1QSHEP-1QS5_____	7.5E-4	CRO_LEAVES_1-QS-5__ RECIRC_VALVE_OPEN__ AFTER_1-PT-63.1A___
1QSHEP-1QS21____	7.5E-4	CRO_LEAVES_1-QS-21_ RECIRC_VALVE_OPEN__ AFTER_1-PT-63.1B___
1RSHEP-ELBOW____	3.8E-4	CRO_LEAVES_SPRAY___ HEADER_ELBOW_______ REMOVED,_1-PT-64.8_
1RSHEP-FLANGE___	3.8E-4	CRO_LEAVES_SPRAY___ HEADER_FLANGE______ INSTALLED_1-PT-64.3
1RSHEP-MOV-155A_	7.5E-4	CRO_LEAVES_MOV-155A OR_MOV-156A_CLOSED_ OR_DEENERGIZED_____
1RSHEP-MOV-155B_	7.5E-4	CRO_LEAVES_MOV-155B OR_MOV-156B_CLOSED_ OR_DEENERGIZED_____
1RSHEP-1RS12____	7.5E-4	CRO_LEAVES RS-P-2A_ RECIRC_VALVES_OPEN_ 1-RS-12_&_1-RS-95__
1RSHEP-1RS22____	7.5E-4	CRO_LEAVES RS-P-2B_ RECIRC_VALVES_OPEN_ 1-RS-22_&_1-RS-96__

TABLE 3.3.3-2
SUMMARY OF OPERATOR ACTIONS
TYPE C HEPs: POST ACCIDENT ACTIONS

Basic Event Name	Point Estimate	Description
HEP-0AP10_______	5.3E-3	0-AP-10_LOSS_OF____ ELECTRICAL_POWER___ ___________________
HEP-0AP12-10HR__	5.0E-3	0-AP-12_LOSS_OF____ SERVICE_WATER______ RECOVERY_IN_10_HOUR
HEP-0AP12-20HR__	2.6E-4	0-AP-12_LOSS_OF____ SERVICE_WATER______ RECOVERY_IN_20_HOUR
HEP-0AP12-30HR__	6.6E-3	0-AP-12_LOSS_OF____ SERVICE_WATER______ RECOVERY_IN_30_HOUR
HEP-0AP12-40HR__	1.3E-1	0-AP-12_LOSS_OF____ SERVICE_WATER______ RECOVERY_IN_40_HOUR
HEP-0AP12-ATTCH4	1.6E-4	0-AP-12_LOSS_OF_SW_ ATTACHMENT_4:_TWO__ PUMPS_ON_ONE_HEADER
HEP-1AP15-1E____	7.8E-4	1-AP-15_LOSS_OF_CC_ STEP_1E_RESTORE_SW_ TO_CC_HEAT_EXCHANGR
HEP-1AP15-6_____	2.8E-2	1-AP-15_LOSS_OF_CC_ STEP_6_CROSS_TIE_CC IF_UNIT_2_AVAILABLE
HEP-1AP22:5_____	1.8E-4	1-AP-22.5_LOSS_OF__ EMERGENCY_CONDNSATE STORAGE_TANK_______
HEP-1AP33:1_____	3.9E-1	1-AP-33.1_REACTOR__ COOLANT_PUMP_SEAL__ FAILURE____________

TABLE 3.3.3-2 (Continued)
SUMMARY OF OPERATOR ACTIONS
TYPE C HEPs: POST ACCIDENT ACTIONS

Basic Event Name	Point Estimate	Description
HEP-1AP49_______	1.3E-2	1-AP-49_LOSS_OF____ NORMAL_CHARGING____ ________________________
HEP-0AP55-10HR__	5.0E-3	0-AP-55_LOSS_OF____ ESGR/MCR_HVAC______ RECOVERY_IN_10_HOUR
HEP-0AP55-20HR__	2.6E-4	0-AP-55_LOSS_OF____ ESGR/MCR_HVAC______ RECOVERY_IN_20_HOUR
HEP-0AP55-30HR__	6.6E-3	0-AP-55_LOSS_OF____ ESGR/MCR_HVAC______ RECOVERY_IN_30_HOUR
HEP-0AP55-40HR__	1.3E-1	0-AP-55_LOSS_OF____ ESGR/MCR_HVAC______ RECOVERY_IN_40_HOUR
HEP-1E0-1_______	1.4E-3	1-E-0_RX_TRIP_OR_SI STEP_1_VERIFY______ REACTOR_TRIP_______
HEP-1E0-7_______	1.4E-3	1-E-0_RX_TRIP_OR_SI STEP_7_VERIFY_SI___ PUMPS_RUNNING______
HEP-1E0-8_______	1.4E-3	1-E-0_RX_TRIP_OR_SI STEP_8_VERIFY_MAIN_ FEEDWATER_ISOLATION
HEP-1E0-11______	1.4E-3	1-E-0_RX_TRIP_OR_SI STEP_11_VERIFY_SW__ PUMPS_RUNNING______
HEP-1E0-12______	1.4E-3	1-E-0_RX_TRIP_OR_SI STEP_12_MAIN_STEAM_ LINES_ISOLATION____

TABLE 3.3.3-2 (Continued)
SUMMARY OF OPERATOR ACTIONS
TYPE C HEPs: POST ACCIDENT ACTIONS

Basic Event Name	Point Estimate	Description
HEP-1E0-13______	1.4E-3	1-E-0_RX_TRIP_OR_SI STEP_13_CHECK_IF___ CDA_IS_REQUIRED____
HEP-1E0-14______	1.0E+0	1-E-0_RX_TRIP_OR_SI STEP_14_VERIFY_SI__ FLOW_______________
HEP-1E0-15_____	1.1E-3	1-E-0_RX_TRIP_OR_SI STEP_15_VERIFY_AUX_ FEEDWATER_FLOW_____
HEP-1E0-16______	8.0E-3	1-E-0_RX_TRIP_OR_SI STEP_16_CHARGING___ PUMP_ALIGNMENT_____
HEP-1E0-22______	1.9E-2	1-E-0_RX_TRIP_OR_SI STEP_22_PRZR_PORVS_ SPRAY_VALVES_CLOSED
HEP-1E0-ATTACH:1	7.7E-3	1-E-0_RX_TRIP_OR_SI ATTACHMENT_1_VERIFY PHASE_B_ISOLATION__
HEP-1E1-25______	1.2E-2	1-E-1_LOSS_OF_RX_OR 2ND_COOLANT_STEP_25 REDUNDANT_COLD_LEG_
HEP-1E3-3_______	3.7E-3	1-E-3_SGTR_________ STEP_3_ISOLATE_FLOW FROM_RUPTURED_SG___
HEP-1E3-13______	2.2E-2	1-E-3_SGTR_________ STEP_13_INITIATE___ RCS_COOLDOWN_______
HEP-1ECA3:1-16__	3.0E-3	1-ECA-3.1_SGTR_WITH SUBCOOLED_RCS______ STEP_16_COOLDOWN___

TABLE 3.3.3-2 (Continued)
SUMMARY OF OPERATOR ACTIONS
TYPE C HEPs: POST ACCIDENT ACTIONS

Basic Event Name	Point Estimate	Description
HEP-1ECA3:2-5___	7.3E-4	1-ECA-3.2_SGTR_WITH SATURATED_RCS______ STEP_5_COOLDOWN____
HEP-1ECA3:3-27	9.0E-2	1-ECA-3.3_SGTR_&___ NO_PRESSURE_CONTROL STEP_27_COOLDOWN___
HEP-1ECA3:3-35	4.9E-3	1-ECA-3.3_SGTR_&___ NO_PRESSURE_CONTROL STEP_35_LATE_COOLDN
HEP-1ES1:2-S1___	1.0E+0	1-ES-1.2_POST_LOCA_ COOLDOWN_AND_______ DEPRESSURIZATION_S1
HEP-1ES1:2-S2___	8.5E-4	1-ES-1.2_POST_LOCA_ COOLDOWN_AND_______ DEPRESSURIZATION_S2
HEP-1ES1:3______	1.2E-2	1-ES-1.3_TRANSFER__ TO_COLD_LEG________ RECIRCULATION______
HEP-1ES1:4______	8.5E-4	1-ES-1.4_TRANSFER__ TO_HOT_LEG_________ RECIRCULATION______
HEP-1FRC:1-11-S1	1.0E+0	1-FR-C.1_INADEQUATE CORE_COOLING_STEP11 DEPRESSURIZE_SGS___
HEP-1FRC:1-11-S2	8.3E-2	1-FR-C.1_INADEQUATE CORE_COOLING_STEP11 DEPRESSURE_SGS__S2_
HEP-1FRH:1-5____	3.1E-3	1-FR-H.1_LOSS_OF___ HEAT_SINK_STEP_5___ CHECK_SG_LEVELS____

TABLE 3.3.3-2 (Continued)
SUMMARY OF OPERATOR ACTIONS
TYPE C HEPs: POST ACCIDENT ACTIONS

Basic Event Name	Point Estimate	Description
HEP-1FRH:1-11___	4.8E-2	1-FR-H.1_LOSS_OF___ HEAT_SINK_STEP_11__ RCS_FEED_PATH______
HEP-1FRH:1-15___	8.3E-3	1-FR-H.1_LOSS_OF___ HEAT_SINK_STEP_15__ RCS_BLEED_PATH_____
HEP-1FRS:1-4____	7.6E-3	1-FR-S.1_ATWS______ STEP_4_INITIATE____ EMERGENCY_BORATION_
HEP-1FRS:1-5____	3.0E-2	1-FR-S.1_ATWS______ STEP_5_DO_ATTACH_2_ REMOTE_REACTOR_TRIP
HEP-1OP14:1-5:13	4.3E-3	1-OP-14.1 RHR______ STEP 5.13,__OPEN___ MOV-1700_&_MOV-1701
HEP-1OP21:6_____	1.1E-3	1-OP-21.6_MCR_AND__ RELAY_ROOM_AIR_____ CONDITIONING_______
HEP-1OP49:1_____	1.3E-1	0-OP-49.1_STARTUP__ AND_SHUTDOWN_OF_THE SERVICE_WATER_SYSTEM
HEP-0OP49:4A____	6.3E-2	0-OP-49.4A_VALVE___ CHECKOFF_MCR_A/C___ SERVICE_WATER______
HEP-NO-PROCEDURE	1.0E+0	NO_PROCEDURE_FOR___ THIS_OPERATOR______ ACTION_____________

TABLE 3.3.3-3
HEP IMPORTANCE ANALYSIS FROM FINAL QUANTIFICATION

Rank	Event Name	Point Estimate	Fessell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
5	HEP-1FRH:1-11	4.824E-2	1.163E-1	3.29	1.132
16	HEP-0AP55-10HR	4.949E-3	7.078E-2	15.23	1.076
20	HEP-1FRC:1-11-S1	1.000E+0	5.962E-2	1.00	1.063
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
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
42	HEP-1AP22:5	1.750E-4	3.367E-2	193.37	1.035
47	HEP-1OP49:1	1.326E-1	2.497E-2	1.16	1.026
71	HEP-0AP55-40HR	1.250E-1	1.656E-2	1.12	1.017
80	HEP-1FRC:1-11-S2	1.062E-2	1.332E-2	2.24	1.013
97	HEP-1ECA3:1-16	3.025E-3	9.604E-3	4.17	1.010
100	HEP-1ES1:3	1.220E-2	9.378E-3	1.76	1.009
109	HEP-1FRH:1-15	8.249E-3	8.273E-3	1.99	1.008
112	HEP-1E3-3	3.650E-3	7.421E-3	3.03	1.007
118	HEP-1ES1:4	8.499E-4	6.308E-3	8.42	1.006
144	HEP-1ES1:2-S2	8.499E-4	4.624E-3	6.44	1.005
159	HEP-1ECA3:3-27	8.974E-2	3.578E-3	1.04	1.004
178	HEP-1AP15-6	2.815E-2	2.952E-3	1.10	1.003
199	1FWHEP-1FW548	7.499E-4	2.219E-3	3.96	1.002
200	1FWHEP-1FW546	7.499E-4	2.168E-3	3.89	1.002
215	HEP-1OP14:1-5:13	4.259E-3	1.728E-3	1.40	1.002
226	HEP-1E0-7	1.350E-3	1.645E-3	2.22	1.002
230	1FWHEP-1FW543	7.499E-4	1.567E-3	3.09	1.002
252	HEP-1ECA3:2-5	7.249E-4	1.219E-3	2.68	1.001
266	1RSHEP-FLANGE	3.750E-4	9.962E-4	3.66	1.001
316	HEP-1EI-25	1.175E-2	6.087E-4	1.05	1.001
335	HEP-1FRS:1-5	2.970E-2	4.696E-4	1.02	1.000
344	HEP-1FRH:1-5	3.125E-3	3.829E-4	1.12	1.000
357	HEP-0AP55-30HR	6.565E-3	3.173E-4	1.05	1.000
390	HEP-1ECA3:3-35	4.924E-3	2.157E-4	1.04	1.000
398	HEP-1E0-14	1.000E+0	1.952E-4	1.00	1.000
408	HEP-1OP21:6	1.050E-3	1.703E-4	1.16	1.000
409	1QSHEP-FLANGE	3.750E-4	1.700E-4	1.45	1.000
415	HEP-1AP33:1	3.866E-1	1.552E-4	1.00	1.000
433	HEP-1E0-22	1.880E-2	1.340E-4	1.01	1.000
451	HEP-1AP15-1E	7.799E-4	8.620E-5	1.11	1.000
477	HEP-0AP10	5.274E-3	6.366E-5	1.01	1.000

TABLE 3.3.3-3 (Continued)
HEP IMPORTANCE ANALYSIS FROM FINAL QUANTIFICATION

Rank	Event Name	Point Estimate	Fessell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
505	HEP-0AP12-10HR	4.949E-3	4.434E-5	1.01	1.000
539	HEP-0AP12-20HR	2.600E-4	2.177E-5	1.08	1.000
610	1QSHEP-1QS21	7.499E-4	4.557E-6	1.01	1.000
611	1QSHEP-1QS5	7.499E-4	4.522E-6	1.01	1.000
625	HEP-1E0-15	1.075E-3	2.979E-6	1.00	1.000
628	HEP-0AP12-40HR	1.250E-1	2.721E-6	1.00	1.000
676	1FWHEP-MOV-100B	7.499E-4	5.379E-7	1.00	1.000
685	1FWHEP-MOV-100D	7.499E-4	3.866E-7	1.00	1.000
691	1FWHEP-HCV-100C	7.499E-4	3.236E-7	1.00	1.000
700	HEP-0AP12-30HR	6.565E-3	1.723E-7	1.00	1.000

TABLE 3.3.3-4
HUMAN ERROR PROBABILITIES SENSITIVITY RESULTS
HEPs INDIVIDUALLY SET EQUAL TO ONE

Basic Event Name	Point Estimate	Core Damage Frequency When HEPs = 1	% Increase In CDF When HEPs = 1
HEP-1AP22:5	1.8E-4	1.3E-2	19,237
HEP-0AP55-20HR	2.6E-4	9.7E-3	14,142
HEP-0AP55-10HR	5.0E-3	1.0E-3	1423
HEP-1ES1:4	8.5E-4	5.7E-4	742
HEP-1ES1:2-S2	8.5E-4	4.4E-4	544
HEP-1ECA3:1-16	3.0E-3	2.8E-4	317
1FWHEP-1FW548	7.5E-4	2.7E-4	296
1FWHEP-1FW546	7.5E-4	2.6E-4	289
1RSHEP-FLANGE	3.8E-4	2.5E-4	266
HEP-1FRH:1-11	4.8E-2	2.2E-4	229
1FWHEP-1FW543	7.5E-4	2.1E-4	209
HEP-1E3-3	3.7E-3	2.1E-4	203
HEP-1E3-13	2.2E-2	1.9E-4	174
HEP-1ECA3:2-5	7.3E-4	1.8E-4	168
HEP-1FRC:1-11-S2	8.3E-3	1.5E-4	124
HEP-1E0-7	1.4E-3	1.5E-4	122
HEP-1FRH:1-15	8.3E-3	1.4E-4	99
HEP-1ES1:3	1.2E-2	1.2E-4	76
1QSHEP-FLANGE	3.8E-4	9.9E-5	45
HEP-1OP14:1-5:13	4.3E-3	9.5E-5	40
HEP-1OP21:6	1.1E-3	7.9E-5	16
HEP-1OP49:1	1.3E-1	7.9E-5	16

TABLE 3.3.3-4 (Continued)
HUMAN ERROR PROBABILITIES SENSITIVITY RESULTS
HEPs INDIVIDUALLY SET EQUAL TO ONE

Basic Event Name	Point Estimate	Core Damage Frequency When HEPs = 1	% Increase In CDF When HEPs = 1
HEP-1FRH:1-5____	3.1E-3	7.6E-5	12
HEP-0AP55-40HR__	1.3E-1	7.6E-5	12
HEP-1AP15-1E____	7.8E-4	7.5E-5	11
HEP-1AP15-6_____	2.8E-2	7.5E-5	10
HEP-0AP12-20HR__	2.6E-4	7.4E-5	8
HEP-0AP55-30HR__	6.6E-3	7.1E-5	5
HEP-1E1-25______	8.1E-3	7.1E-5	5
HEP-1ECA3:3-27	9.0E-2	7.0E-5	4
HEP-1ECA3:3-35	4.9E-3	7.1E-5	4
HEP-1FRS:1-5____	3.0E-2	6.9E-5	2
1QSHEP-1QS5_____	7.5E-4	6.8E-5	1
1QSHEP-1QS21____	7.5E-4	6.8E-5	1
HEP-0AP10______	5.3E-3	6.9E-5	1
HEP-0AP12-10HR__	5.0E-3	6.8E-5	1
HEP-1E0-22______	1.9E-2	6.8E-5	1
1FWHEP-HCV-100C_	7.5E-4	6.8E-5	0
1FWHEP-MOV-100B_	7.5E-4	6.8E-5	0
1FWHEP-MOV-100D_	7.5E-4	6.8E-5	0
HEP-0AP12-30HR__	6.6E-3	6.8E-5	0
HEP-0AP12-40HR__	1.3E-1	6.8E-5	0
HEP-1AP33:1_____	3.9E-1	6.8E-5	0
HEP-1FRC:1-11-S1	1.0E+0	6.8E-5	0

TABLE 3.3.3-4 (Continued)
HUMAN ERROR PROBABILITIES SENSITIVITY RESULTS
HEPs INDIVIDUALLY SET EQUAL TO ONE

Basic Event Name	Point Estimate	Core Damage Frequency When HEPs = 1	% Increase In CDF When HEPs = 1
HEP-NO-PROCEDURE	1.0E+0	6.8E-5	0
HEP-1ES1:2-S1	1.0E+0	6.8E-5	0
HEP-1E0-14	1.0E+0	6.8E-5	0
HEP-1E0-15	1.1E-3	6.8E-5	0
1CHHEP-MOV-1270A	1.1E-4	**	*
1CHHEP-MOV-1275A	7.5E-4	*	*
1CHHEP-MOV-1275B	7.5E-4	*	*
1CHHEP-MOV-1373	7.5E-4	*	*
1RSHEP-ELBOW	3.8E-4	*	*
1RSHEP-MOV-155A	7.5E-4	*	*
1RSHEP-MOV-155B	7.5E-4	*	*
1RSHEP-1RS12	7.5E-4	*	*
1RSHEP-1RS22	7.5E-4	*	*
HEP-0AP12-ATTCH4	1.6E-4	*	*
HEP-1AP49	1.3E-2	*	*
HEP-1E0-1	1.4E-3	*	*
HEP-1E0-8	1.4E-3	*	*
HEP-1E0-11	1.4E-3	*	*
HEP-1E0-12	1.4E-3	*	*
HEP-1E0-13	1.4E-3	*	*
HEP-1E0-16	8.0E-3	*	*

TABLE 3.3.3-4 (Continued)
HUMAN ERROR PROBABILITIES SENSITIVITY RESULTS
HEPs INDIVIDUALLY SET EQUAL TO ONE

Basic Event Name	Point Estimate	Core Damage Frequency When HEPs = 1	% Increase In CDF When HEPs = 1
HEP-1EO-ATTACH:1	7.7E-3	*	*
HEP-1FRS:1-4	7.6E-3	*	*
HEP-0OP49:4A	6.3E-2	*	*

* = HEP basic events not found in cut sets greater than 1E-11.
** = HEP basic event not included in fault trees.

TABLE 3.3.4-1
COMMON CAUSE FAILURE PROBABILITIES
UTILIZING GENERIC AND PLANT SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
CCFs Estimated Using NUREG/CR-4780 method:			
MOV-CC	Motor operated valve CCF fails open/closed	2.50E-4	NUREG-4780 method
1EGEDG-CC-1H-1J	CCF of EDG 1H & 1J fail to start & run	2.66E-4	NUREG-4780 method
1EGEDG-CC-1H-2H	CCF of EDG 1H & 2H fail to start & run	2.66E-4	NUREG-4780 method
1EGEDG-CC-1H-2J	CCF of EDG 1H & 2J fail to start & run	2.66E-4	NUREG-4780 method
1EGEDG-CC-1H1J2H	CCF of EDG 1H, 1J & 2H fail to start & run	9.58E-5	NUREG-4780 method
1EGEDG-CC-1H1J2J	CCF of EDG 1H, 1J & 2J fail to start & run	9.58E-5	NUREG-4780 method
1EGEDG-CC-1H2H2J	CCF of EDG 1H, 2H & 2J fail to start & run	9.58E-5	NUREG-4780 method
1EGEDG-CC-1J-2H	CCF of EDG 1J & 2H fail to start & run	2.66E-4	NUREG-4780 method
1EGEDG-CC-1J-2J	CCF of EDG 1J & 2J fail to start & run	2.66E-4	NUREG-4780 method
1EGEDG-CC-1J2H2J	CCF of EDG 1J, 2H & 2J fail to start & run	9.58E-5	NUREG-4780 method

TABLE 3.3.4-1 (Continued)
COMMON CAUSE FAILURE PROBABILITIES
UTILIZING GENERIC AND PLANT SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1EGEDG-CC-ALL	CCF of EDG 1H, 1J, 2H & 2J fail to start & run	6.09E-5	NUREG-4780 method
1FWPSB-CC-MDP3AB	Motor Driven AFW pumps 1-FW-P-3A/B CCF fail to start & run	1.42E-4	NUREG-4780 method
1SIPSB-CC-FS1A1B	LHSI pumps 1-SI-P-1A/1B CCF fail to start & run	4.40E-4	NUREG-4780 method
2EGEDG-CC-2H-2J	CCF of EDG 2H & 2J fail to start & run	2.66E-4	NUREG-4780 method
CCF Estimated with Beta Factor Model:			
AOV-CC	Air operated valve CCF fails closed	1.81E-3	10% AOV-FC
BAT-CC	CCF batteries fail to supply power	1.05E-6	Beta of 1% on BAT-LP
CKV-CC	Check valve CCF fails closed	6.34E-5	10% CKV-FC
FCV-CC	Flow control valve CCF fails closed	1.81E-3	10% AOV-FC
FIC-CC	Flow instrument channel, 30 day test interval	4.64E-4	Based on 10% FIC-LF
LCV-CC	Level control valve CCF fails closed	1.81E-3	10% AOV-FC

TABLE 3.3.4-1 (Continued)
COMMON CAUSE FAILURE PROBABILITIES
UTILIZING GENERIC AND PLANT SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
LIC-CC	Level instrument channel, 30 day test interval	4.64E-4	Based on 10% FIC-LF
MV--CC	Manual valve CCF fails closed	1.25E-5	10% MV--FC
PAT-CC	Motor driven alternating pump CCF fails to start from standby	3.31E-5	10% PAT-FS
PCV-CC	Pressure control valve CCF fails closed	1.81E-3	10% AOV-FC
PIC-CC	Pressure instrument channel, 30 day test interval	4.64E-4	Based on 10% FIC-LF
PSB-CC	Motor driven standby pump CCF fails to start	3.93E-4	10% PSB-FS
RV--CC	Pressure operated relief valve fails closed	9.99E-4	10% RV--FC
SV--CC	CCF safety valve fails closed	1.25E-6	10% SV--FC
TCV-CC	Temperature control valve CCF fails closed	1.81E-3	10% AOV-FC
TIC-CC	Temperature instrument channel, 30 day test interval	4.64E-4	Based on 10% FIC-LF
1CHPAT-CC-FS1B1C	CCF FS 1-CH-P-1B/1C fail to start	4.97E-4	Based on 10% 1CHPAT-FS

TABLE 3.3.4-1 (Continued)
COMMON CAUSE FAILURE PROBABILITIES
UTILIZING GENERIC AND PLANT SPECIFIC DATA

Identifier	Failure Description	Mean Probability	Comments
1CHPAT-CC-FS1ABC	CCF FS 1-CH-P-1A/ 1B/1C fail to start	4.97E-4	Based on 10% 1CHPAT-FS
1FWCKV-CC-ALLAFW	CCF 3 of 3 AFW pumps failed by CKV-FC events	6.34E-5	Based on 10% CKV-FC
1FWPCV-CC-159AB	CCF PCVs plugged, 1-FW-PCV-159A/159B	1.37E-5	Based on 10% PCV-PG
1HVCHU-CC-HVE4	Chillers CCF fail to start 1-HV-E-4B/4C	6.20E-3	Based on 10% CHU-FS-NUE4
1RSPSB-CC-1A-1B	IRS pumps 1-RS-P-1A/1B CCF fail to start	4.02E-4	Based on 10% 1RSPSB-FS-1RSP1
1RSPSB-CC-2A-2B	IRS pumps 1-RS-P-2A/2B CCF fail to start	4.02E-4	Based on 10% 1RSPSB-FS-1RSP2
1SWPSB-CC-SWP1B	Normal SW pumps 1-SW-P-1B & 2-SW-P-1B CCF fail to start	3.84E-4	Based on 10% SWPAT-FS-SWP1
1SWSCN-CC-SWRES	Normal SW pumps 1-SW-P-1A/1B and 2-SW-P-1B CCF due to plugged intake screens at SW reservoir	6.93E-5	Based on 10% 1SWSCN-PL 1SWP1A with 24 hr mission
2HVCHU-CC-HVE4	Chillers CCF fail to start 2-HU-E-4B/4C	6.20E-3	10% CHU-FS

TABLE 3.3.5-1
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
B01	Failure to recover offsite power in 1 hour with RCS boundary not intact.	4.8E-1
B02	Failure to recover offsite power in 1 hour with turbine driven pump failure.	3.4E-1
B10	Failure to recover offsite power in 10 hours assuming RCP seal failure due to lack of CC from Unit 2.	2.0E-2
B16	Failure to recover offsite power in 11 hours assuming AFW failure after battery depletion.	7.5E-3
B102	Failure to recover offsite power before vessel failure with RCS boundary not intact.	6.8E-1
B103	Failure to recover offsite power before vessel failure with failure of turbine driven AFW pump.	6.8E-1
B111	Failure to recover offsite power before vessel failure with seal LOCA.	6.8E-1
B117	Failure to recover offsite power before vessel failure with eventual AFW failure.	6.8E-1
B220	Failure to recovver offsite power before containment failure with RCS boundary not intact.	9.0E-4
B221	Failure to recover offsite power before containment failure with failure of the turbine driven AFW pump.	9.0E-4
B229	Failure to recover offsite power before containment failure with seal LOCA.	9.0E-4
B235	Failure to recover offsite power before containment failure with eventual AFW failure.	9.0E-4
Ch01	Containment heat removal by 1 of 4 IRS or ORS trains with SW and Casing Cooling assuming QS success.	2.7E-3

TABLE 3.3.5-1 (Continued)
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
Ch02	Containment heat removal by 1 of 2 ORS trains with SW and Casing Cooling assuming QS failure.	1.8E-2
Ch03	Containment heat removal by 1 of 4 IRS or ORS trains with SW but without Casing Cooling.	2.6E-3
Ch04	Containment heat removal for T1 LOOP by 1 of 4 IRS or ORS trains with SW but without Casing Cooling.	5.5E-3
Ch05	Containment heat removal for T1 LOOP by 1 of 4 IRS or ORS trains with SW and Casing Cooling assuming QS success.	6.3E-3
Ch06	Containment heat removal for T5A by 1 of 4 IRS or ORS trains with SW and Casing Cooling assuming QS success.	1.3E-2
Ch07	Containment heat removal for T5A by 1 of 4 IRS or ORS trains with SW but without Casing Cooling.	4.5E-3
Ch08	Containment heat removal for T5B by 1 of 4 IRS or ORS trains with SW and Casing Cooling assuming QS success.	1.3E-2
Ch09	Containment heat removal for T5B by 1 of 4 IRS or ORS trains with SW but without Casing Cooling.	4.5E-3
Ch10	Containment heat removal for T9A by 1 of 4 IRS or ORS trains with SW and Casing Cooling assuming QS success.	1.6E-2
Ch11	Containment heat removal for T9A by 1 of 4 IRS or ORS trains with SW but without Casing Cooling.	6.6E-3
Ch12	Containment heat removal for T9B by 1 of 4 IRS or ORS trains with SW and Casing Cooling assuming QS success.	1.4E-2

TABLE 3.3.5-1 (Continued)
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
Ch13	Containment heat removal for T9B by 1 of of 4 IRS or ORS trains with SW and Casing Cooling assuming QS success.	5.3E-3
D101	Failure to provide 1/3 HHSI pumps in injection mode from RWST.	4.6E-3
D102	Failure to provide 1/3 HHSI pumps in injection mode from RWST for T1.	6.0E-2
D103	Failure to provide 1/3 HHSI pumps in injection mode from RWST for T1 with RCS boundary not intact.	1.2E-2
D104	Failure to provide 1/3 HHSI pumps in injection mode from RWST after recovery of AC Power with a seal LOCA.	2.7E-4
D105	Failure to provide 1/3 HHSI pumps in injection mode from RWST assuming manual start for feed and bleed.	5.3E-2
D106	Failure to provide 1/3 HHSI pumps in injection mode from RWST assuming auto-start with RCS boundary not intact for T5A.	3.3E-2
D107	Failure to provide 1/3 HHSI pumps in injection mode from RWST assuming auto-start with RCS boundary not intact for T5B.	6.0E-3
D108	Failure to provide 1/3 HHSI pumps in injection mode from RWST assuming auto-start with RCS boundary not intact for T9A.	5.7E-2
D109	Failure to provide 1/3 HHSI pumps in injection mode from RWST assuming auto-start with RCS boundary not intact for T9B.	5.1E-2
D201	Failure of 2/2 Accumulators to inject following a large LOCA.	4.2E-3

TABLE 3.3.5-1 (Continued)
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
D202	Failure of 2/3 Accumulators to inject following a medium LOCA.	7.7E-5
D203	Failure of 3/3 Accumulators to inject for core cooling recovery.	6.3E-3
D301	Failure of 1/2 LHSI pumps to inject from RWST for large LOCA. Auto initiation on SI signal.	1.2E-3
D302	Failure of 1/2 LHSI pumps to inject from RWST for core cooling recovery.	2.6E-2
D303	Not used.	------
D304	Failure 2/2 LHSI pumps to inject from RWST following Core Cooling recovery success.	1.2E-1
D401	Emergency boration for ATWS events using 1/2 Charging Pumps, and 1 boric acid transfer pump. Initiated by operator action.	9.0E-3
D402	Emergency boration for ATWS events through Safety Injection from the BIT with a stuck open pressurizer relief valve, using 1/3 HHSI pumps. Manual SI initiation.	4.0E-3
DG01	Emergency bus power from 1 of 2 EDGs (1H or 1J).	2.7E-3
DH01	Failure to provide hot leg recirculation using 1/2 LHSI pumps injecting to hot legs from Containment sump after a large LOCA.	3.9E-3
Fm01	Probability of a very small LOCA given that S2 or an ISLOCA has occurred.	9.5E-1
H101	Failure to provide low head recirculation using 1/2 LHSI pumps taking suction from Containment sump, for large LOCA.	2.5E-3

TABLE 3.3.5-1 (Continued)
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
H102	Failure to provide low head recirculation using 1/2 LHSI pumps taking suction from Containment sump, for small breaks and transients where LHSI pumps could not inject in early phase due to high pressure in RCS.	2.6E-3
H103	Failure to provide low head recirculation by 1 of 2 LHSI trains assuming auto-start of SI signals for T5A.	3.9E-2
H104	Failure to provide low head recirculation by 1 of 2 LHSI trains assuming auto-start of SI signals for T5B.	3.8E-2
H105	Failure to provide low head recirculation by 1 of 2 LHSI trains assuming auto-start of SI signals for T9A.	5.1E-2
H106	Failure to provide low head recirculation by 1 of 2 LHSI trains assuming auto-start of SI signals for T9B.	5.1E-2
H107	Failure to provide low head recirculation by 1 of 2 LHSI trains assuming auto-start of SI signals for T1.	1.0E-2
H201	Failure to provide high head recirculation using 1/3 HHSI taking suction from 1/2 LHSI, taking suction from the Containment sump. Used in intermediate breaks and feed and bleed sequences when offsite power available.	5.4E-3
H202	Failure to provide high head recirculation using 1/3 HHSI taking suction from 1/2 LHSI, taking suction from the Containment Sump. Used for T1 sequences.	1.4E-2
H203	Failure to provide high head recirculation 1/3 HHSI taking suction from 1/2 LHSI, taking suction from Containment sump. Used for T5A initiator.	6.8E-2

TABLE 3.3.5-1 (Continued)
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
H204	Failure to provide high head recirculation using 1/3 HHSI taking suction from 1/2 LHSI, taking suction from the Containment sump. Used for T5B initiator.	4.0E-2
H205	Failure to provide high head recirculation using 1/3 HHSI taking suction from 1/2 LHSI, taking suction from the Containment sump. Used for T9A initiator.	9.6E-2
H206	Failure to provide high head recirculation using 1/3 HHSI taking suction from 1/2 LHSI, taking suction from the Containment sump. Used for T9B initiator.	9.1E-2
HV01	Emergency Switchgear Room (ESGR) cooling by 1 of 2 air handling units and 1 of 3 chiller trains for LOCAs and transients with electric power.	1.3E-3
HV02	ESGR cooling by 1 of 2 air handling units and 1 of 3 chiller trains for the T1 initiator.	1.6E-2
HV03	ESGR cooling by 1 of 2 air handling units and 1 of 3 chiller trains for the T5A initiator.	1.4E-3
HV04	ESGR cooling by 1 of 2 air handling units and 1 of 3 chiller trains for the T5B initiator.	1.3E-3
HV05	ESGR cooling by 1 of 2 air handling units and 1 of 3 chiller trains for the T9A initiator.	2.5E-1
HV06	ESGR cooling by 1 of 2 air handling units and 1 of 3 chiller trains for the T9B initiator.	2.6E-3
K01	Failure to make the reactor subcritical by automatic RPS or manual scram within one minute.	1.9E-6

TABLE 3.3.5-1 (Continued)
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
L01	Failure to provide AFW to 1/3 Steam Generators from 1/3 AFW Pumps, with offsite power nominally available.	2.8E-4
L02	Failure to provide AFW to 2/3 SG's for small and medium LOCAs.	3.5E-3
L03	Failure to provide AFW to 1/3 SG's for the loss of offsite power initiator.	8.1E-4
L04	Failure to provide AFW to 1/3 SG's with 1/2 MD AFW Pumps after power restoration.	4.9E-4
L05	Failure to provide AFW to 1/3 SG's from 1/3 AFW Pumps for the T5A initiator.	1.7E-3
L06	Failure to provide AFW to 1/3 SG's from 1/3 AFW Pumps for T5B sequences.	1.7E-3
L07	Failure to provide AFW to 1/2 SG's for SG tube rupture.	2.9E-4
L08	Failure to provide AFW to 2/2 SG's for T7 assuming HPI failure.	1.6E-1
L09	Failure to provide AFW to 1/3 SG's from 1/3 AFW Pumps for the T9A initiator.	1.7E-3
L10	Failure to provide AFW to 1/3 SG's from 1/3 AFW Pumps for the T9B initiator.	1.7E-3
L11	Failure to provide auxiliary feedwater to 2 of 3 SG's for the high power ATWS.	1.8E-3
L12	Failure to provide auxiliary feedwater to 2 of 3 SG-s for the low power ATWS.	5.5E-4
Lt01	Failure to provide AFW to 1/3 SG's from the Turbine-Driven Pump for 12 hours used in the T1A event tree.	9.3E-2

TABLE 3.3.5-1 (Continued)
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
M01	Failure to restore Main Feedwater and provide FW to 1/3 SG's from 1/2 MFW Pumps for 24 hours. Used after T2A initiators.	7.1E-3
M02	Failure to provide FW to 1/3 SG's from 1/2 MFW pumps for 24 hours. Used in T3 initiators.	3.2E-3
M03	Portion of ATWS initiators where MFW is failed as part of the initiator. Used for high power ATWS sequences.	2.9E-1
MS101	Manual scram late for ATWS events.	3.0E-2
O01	Failure of the operator to cooldown and depressurize the RCS using 1/3 SG's after an intermediate size break. Currently modeled to not occur.	1.0E+0
O02	Failure of the operator to cooldown and depressurize the RCS using 1/3 SG's after a small break LOCA.	2.2E-3
O03	Failure of the operator to cooldown and depressurize the RCS, using 1/3 SG's for very small LOCA's.	2.2E-3
O04	Failure of the operator to cooldown and depressurize using 1/3 SG's for the T1 initiator.	1.0E-2
O05	Failure of the operator to cooldown and depressurize the RCS using 1/3 SG's for the T4 initiator.	3.9E-1
O06	Failure of the operator to cooldown and depressurize, using 1/2 SG's for the T7 initiator with successful HPI and AFW.	2.2E-2
O07	Failure of the operator to cooldown and depressurize, using 1/2 SG's, for the T7 initiator with HPI failure.	9.2E-2

TABLE 3.3.5-1 (Continued)
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
008	Failure of the operator to cooldown and depressurize, using 1/3 SG's for the T6 and T8 initiators.	4.5E-3
0201	Failure of the operator to cooldown and depressurize the RCS to atmospheric conditions after initial failure to cooldown and depressurize within 45 min of an SG tube rupture.	3.5E-3
0202	Failure of the operator to cooldown and depressurize the RCS to atmospheric conditions after failure to isolate the ruptured Steam Generator in a SG tube rupture sequence.	1.2E-3
0203	Failure of the operator to cooldown and depressurize the RCS to atmospheric conditions after failure to initiate early cooldown, for SG tube rupture sequences with failure of HHSI injection.	5.4E-3
P01	Failure to provide feed and bleed cooling with 1/3 HHSI pumps drawing suction from the RWST with 1/2 PORV's opening. Operator initiated.	9.4E-3
P02	Failure to provide feed and bleed cooling with 1/3 HHSI pumps drawing suction from the RWST with 1/2 PORV's opening. Operator initiated. Used in the T1 tree for loss of offsite power.	1.3E-2
P03	Failure to provide feed and bleed cooling with 1/3 HHSI pumps drawing suction from the RWST and injecting through 1/2 PORV's. Operator initiation of PORV's only with one sticking open.	1.3E-2
Pr01	Failure to provide adequate pressure relief in the RCS for high power ATWS sequences.	2.8E-1

TABLE 3.3.5-1 (Continued)
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
Q01	Failure of RCS boundary integrity due to a non-isolable PORV after transients with loss of offsite power.	2.6E-5
Q02	Failure of RCS boundary integrity due to a non-isolable PORV during a station blackout.	1.0E-2
Q03	Failure of RCS boundary integrity due to a non-isolable PORV after a transient.	1.0E-5
Q04	Failure of RCS boundary integrity due to a non-isolable PORV after a T5A initiator (loss of DC Bus A).	1.0E-5
Q05	Failure of RCS boundary integrity due to a non-isolable PORV after a T5B initiator (loss of DC Bus B).	1.0E-5
Q06	Failure of RCS boundary integrity due to a non-isolable PORV after a T9A initiator.	1.7E-4
Q07	Failure of RCS boundary integrity due to a non-isolable PORV after a T9B initiator.	1.7E-4
Q08	Leakage due to RCS pressure in excess of 3200 psi during a high power ATWS.	1.0E+0
Q09	Failure of RCS boundary integrity due to a non-isolable PORV during an ATWS.	3.9E-2
Qs01	Failure to provide quench spray from 1/2 trains during LOCA's and transients with power available.	4.0E-3
Qs02	Failure to provide quench spray from 1/2 trains following loss of offsite power.	1.2E-2
Qs03	Failure to provide quench spray from 1/2 trains for the T5A initiator.	5.4E-2

TABLE 3.3.5-1 (Continued)
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
Qs04	Failure to provide quench spray from 1/2 trains for the T5B initiator.	5.4E-2
Qs05	Failure to provide quench spray from 1/2 trains for the T9A initiator.	5.4E-2
Qs06	Failure to provide quench spray from 1/2 trains for the T9B initiator.	5.4E-2
RC101	Probability of non-recovery of SW in 10 hours for the loss of SW initiator.	5.0E-3
RC102	Probability of non-recovery of ESGR cooling in 10 hours for the loss of ESGR cooling initiator.	6.5E-3
RC103	Probability of non-recovery of ESGR cooling in 10 hours for T1, Loss of Offsite Power initiator.	1.2E-3
RC201	Probability of non-recovery of SW in 20 hours for the loss of SW initiator.	2.6E-4
RC202	Probability of non-recovery of ESGR cooling in 20 hours for the loss of ESGR cooling initiator.	1.8E-3
RC203	Probability of non-recovery of ESGR cooling in 20 hours for T1, Loss of Offsite Power initiator.	1.1E-5
RC301	Probability of non-recovery of SW between 20 and 30 hours after a loss of SW initiating event.	1.3E-1
RC302	Probability of non-recovery of SW between 10 and 30 hours following a loss of SW initiating event.	6.6E-3
RC303	Probability of non-recovery of SW between 20 and 30 hours following a loss of ESGR cooling initiator.	1.3E-1
RC304	Probability of non-recovery of SW between 10 and 30 hours after a loss of ESGR cooling.	6.6E-3

TABLE 3.3.5-1 (Continued)
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
Rs01	Failure to provide flow from 1/4 recirculation spray trains for LOCA's and transients with offsite power and QS nominally available.	4.5E-4
Rs02	Failure to provide flow from 1/2 ORS trains for the large LOCA initiator.	1.1E-2
Rs03	Failure to provide flow from 1/4 recirculation spray trains with no QS available. This function is used for transients and LOCA's with power available.	4.3E-4
Rs04	Failure to provide flow from 1/4 recirculation spray trains without QS available for the loss of offsite power initiator.	3.2E-3
Rs05	Failure to provide flow from 1/4 IRS or ORS trains for the T1 initiator without QS required.	3.8E-3
Rs06	Failure to provide flow from 1/4 IRS or ORS trains with QS available for the T5A initiator.	7.2E-3
Rs07	Failure to provide flow from 1/4 IRS or ORS trains without QS required for the T5A initiator.	1.1E-3
Rs08	Failure to provide flow from 1/4 IRS of ORS trains with QS available for the T5B initiator.	7.2E-3
Rs09	Failure to provide flow from 1/4 IRS or ORS trains without QS required for the T5B initiator.	1.0E-3
Rs10	Failure to provide flow from 1/4 IRS or ORS trains with QS available for the T9A initiator.	7.2E-3
Rs11	Failure to provide flow from 1/4 IRS or ORS trains without QS required for the T9A initiator.	1.1E-3

TABLE 3.3.5-1 (Continued)
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
Rs12	Failure to provide flow from 1/4 IRS or ORS trains with QS available for the T9B initiator.	7.2E-3
Rs13	Failure to provide flow from 1/4 IRS or ORS trains without QS required for the T9B initiator.	1.1E-3
SGI01	Failure of SG isolation following a SGTR.	1.1E-2
Slc01	Failure of RCP seal cooling from 1/3 charging pumps of 1/2 CC pumps during a T1 initiator.	4.8E-3
Slc02	Failure of RCP seal cooling, following a SBO, by 1/3 charing pumps from the opposite unit.	2.8E-2
Tt01	Turbine trip actuated by AMSAC following ATWS event at high power.	2.0E-1
Vi01	Failure to isolate interfacing LOCA prior to depletion of RWST. Current model assumes guaranteed failure to isolate.	1.0E+0
W01	Failure to provide RHR cooling of the Reactor Coolant System with 1/2 RHR pumps and 1/2 RHR heat exchangers with Component Cooling Water supplied to the heat exchangers.	5.0E-2
Y01	Failure to provide core cooling recovery in accordance with FRP C.1, by rapid cooldown of the Steam Generators, accumulator injection, and injection from 1/2 LHSI pumps. Used for medium LOCA's. Operator initiated action to provide AFW to 2/3 SG's and 2/3 SG ADV success.	1.0E+0

TABLE 3.3.5-1 (Continued)
SUMMARY OF FUNCTIONAL UNAVAILABILITIES

Function	Description	Unavailability
Y02	Failure to provide core cooling recovery in accordance with FRP C.1, by rapid cooldown of the Steam Generators, accumulator injection, and injection from 1/2 LHSI pumps. Used for SI initiators that lead to RCP seal LOCA. Operator initiated action to provide AFW to 2/3 SG's and 2/3 SG ADV success.	2.0E-2
Y03	Failure to provide core cooling recovery in accordance with 1-FR-C.1, by rapid cooldown of the Steam Generators, accumulator injection, and injection from 1/2 LHSI pumps. Used for T1 initiators. Operator initiated action to provide AFW to 2/3 SG's and 2/3 SG ADV success.	1.0E-1
Y04	Failure to provide core cooling recovery in accordance with 1-FR-C.1, by rapid cooldown of the Steam Generators, accumulator injection, and injection from 1/2 LHSI pumps. Used for loss of seal cooling initiators. Operator initiated action to provide AFW to 2/3 SG's and 2/3 SG ADV success.	1.5E-2

TABLE 3.3.7-1
SUMMARY OF POTENTIAL FLOOD DAMAGE IN EACH FLOOD AREA

Flood Area	Worst Case Mitigating System Damage	Worst Case Initiating Event
FLA1-1: Unit 1 Containment	Loss of inside Recirculation Spray pumps and primary PORVs	T2 Unit 1 Loss of Feedwater
FLA2: Unit 1 and 2 Control Room	Loss of control to all safety and non safety systems (Unit 1 and 2)	Not Defined
FLA3-1: Unit 1 Cable Vault and Tunnel	1-EP-MC-19 (1H1-2N MCC) 1-EP-MC-20 (1H1-2S MCC) 1-EP-MC-21 (1J1-2H MCC) 1-EP-MC-22 (1J1-2S MCC) (Consequence of damage to this equipment has not been analyzed due to the elevation difference between the flood source and the equipment.)	No initiating event is postulated since flooding in this area is unlikely.
FLA5-1: Unit 1 Air Conditioning and Chiller Room	Loss of Unit 1 Air Conditioning Chillers	T8, Unit 1 & 2 ESGR HVAC
FLA8A: Unit 1 Turbine Building	Loss of Unit 1 & 2 Systems: - Feedwater System	T2 Unit 1 & 2 Loss of Feedwater
FLA8B: Unit 1 Switchgear Rooms	Not analyzed due to the improbability of flooding in this area	
FLA8C: Unit 1 Cable Spreading Room	None	None
FLA9A-1: 1H Diesel Generator Room	Loss of DG 1H (Unit 1)	None
FLA9B-1: 1J Diesel Generator	Loss of DG 1J (Unit 1)	None

TABLE 3.3.7-1 (Continued)
SUMMARY OF POTENTIAL FLOOD DAMAGE IN EACH FLOOD AREA

Flood Area	Worst Case Mitigating System Damage	Worst Case Initiating Event
FLA11A: Auxiliary Building	Loss of the following Unit 1 and 2 systems: - HHSI Suction - HHSI Pumps - HHSI Discharge - HHSI Recirculation - Component Cooling Water	T4, Unit 1 & 2 Loss of RCP Seal Cooling
FLA11B: Fuel and Decontamination	None	None
FLA14A-1: Unit 1 Turbine Driven Auxiliary Feedwater Pump Room	Loss of Unit 1 Systems: - TDAFW Pump	None
FLA14B-1: Unit 1 Motor Driven Auxiliary Feedwater Pump Room	Loss of Unit 1 Systems: - MDAFW Pumps	None
FLA15-1: Unit 1 Quench Spray Pump House	Loss of Unit 1 Systems: - QS Pumps - QS Discharge - LHSI Pumps - LHSI Recirculation - Outside RS Pumps	None

TABLE 3.3.7-2
FREQUENCY OF MANUALLY ISOLABLE SW FLOODING EVENTS IN THE AUXILIARY BUILDING

CAT. 1 (25000 gpm < FLOW RATE< 53000 gpm)	Frequency/ Year
1 - Two 24" Diameter MOV on the SW Supply Headers	5.9E-6
2 - 24" Diameter Piping SEC. on the SW Headers	7.3E-6
CAT. 1 TOTAL CONTRIBUTION =	1.3E-5

CAT. 2 (8000 gpm < FLOW RATE< 25000 gpm)	Frequency/ Year
1 - Two 24" Diameter MOV on the SW Supply Headers	1.7E-5
2 - 24" Diameter Piping SEC. on the SW Headers	1.1E-5
CAT. 2 TOTAL CONTRIBUTION =	2.8E-5

CAT. 3 (3800 gpm < FLOW RATE< 8000 gpm)	Frequency/ Year
1 - Two 24" Diameter MOV on the SW Supply Headers	1.1E-5
2 - 24" Diameter Piping SEC. on the SW Headers	7.2E-6
3 - 10" Diameter MOV on the SW Supply Header to the CCW Pit Coolers	1.4E-6
4 - 10" Diameter Piping on the SW Supply Header to the CCW Pit Cooler	1.7E-5
CAT. 3 TOTAL CONTRIBUTION =	3.7E-5

TABLE 3.3.7-3
FREQUENCY OF REMOTELY ISOLABLE SW FLOODING EVENTS IN THE AUXILIARY BUILDING

CAT. 1 (25000 gpm < FLOW RATE< 53000 gpm)	Frequency/ Year
1 - Two 24" Diameter EJ on the SW Supply Headers	3.1E-5
2 - Four 20" Diameter EJ on the SW Supply Headers	3.4E-5
3 - Two 24" Diameter MOV on the SW Supply Headers	5.9E-6
4 - Two 24" Diameter Piping SEC. on the SW Supply Headers	1.6E-5
5 - One 20" Diameter Piping Sec. on the SW Supply Headers	4.0E-6
6 - Two 18" Diameter Piping Sec. on the SW Supply Headers	2.6E-6
CAT. 1 TOTAL CONTRIBUTION =	9.3E-5

TABLE 3.3.7-3 (Continued)
FREQUENCY OF REMOTELY ISOLABLE SW FLOODING EVENTS IN THE AUXILIARY BUILDING

CAT. 2 (8000 gpm < FLOW RATE< 25000 gpm)	Frequency/ Year
1 - Two 24" Diameter EJ on the SW Supply Headers	4.7E-5
2 - Two 24" Diameter EJ on the SW Return Headers	7.7E-5
3 - Four 20" Diameter EJ on the SW Supply Headers	9.9E-5
4 - Four 20" Diameter EJ on the SW Return Headers	1.3E-4
5 - Two 24" Diameter MOV on the SW Supply Headers	1.7E-5
6 - Two 24" Diameter MAN. VLVs on the SW Return Headers	2.3E-5
7 - Eight 18" Diameter MAN. VLVs on the SW Supply Headers	5.8E-5
8 - 24" Diameter Piping SEC. on the SW Supply Headers	2.4E-5
9 - 24" Diameter Piping SEC. on the SW Return Headers	3.9E-5
10- 20" Diameter Piping SEC. on the SW Supply Headers	1.2E-5
11- 20" Diameter Piping SEC. on the SW Return Headers	1.6E-5
12- 18" Diameter Piping SEC. on the SW Supply Headers	1.6E-5
13- 18" Diameter Piping SEC. on the SW Supply Headers	1.9E-5
CAT. 2 TOTAL CONTRIBUTION =	5.8E-4

TABLE 3.3.7-3 (Continued)
FREQUENCY OF REMOTELY ISOLABLE SW FLOODING EVENTS IN THE AUXILIARY BUILDING

CAT. 3 (3800 gpm < FLOW RATE< 8000 gpm)	Frequency/ Year
1 - Two 24" Diameter EJ on the SW Supply Headers	3.0E-5
2 - Two 24" Diameter EJ on the SW Return Headers	3.0E-5
3 - Four 20" Diameter EJ on the SW Supply Headers	6.5E-5
4 - Four 20" Diameter EJ on the SW Return Headers	6.5E-4
5 - Two 24" Diameter MOV on the SW Supply Headers	1.1E-5
6 - Two 24" Diameter MAN. VLVs on the SW Return Headers	4.6E-5
7 - Eight 18" Diameter MAN. VLVs on the SW Supply Headers	4.5E-5
8 - Eight 18" Diameter MAN. VLVs on the SW Return Headers	7.2E-6
9 - 24" Diameter Piping SEC. on the SW Supply Headers	1.6E-5
10- 24" Diameter Piping SEC. on the SW Return Headers	1.6E-5
11- 20" Diameter Piping SEC. on the SW Return Headers	7.7E-6
12- 20" Diameter Piping SEC. on the SW Return Headers	7.7E-6
13- 18" Diameter Piping SEC. on the SW Supply Headers	1.1E-5
14- 18" Diameter Piping SEC. on the SW Return Headers	1.1E-5
CAT. 3 TOTAL CONTRIBUTION =	3.7E-4

TABLE 3.3.7-4
CONTRIBUTION OF MANUALLY ISOLABLE FLOODING
TO THE FREQUENCY OF 1AB2 FDS

Contributor	Frequency	Probability of Failure to Isolate	Contribution to the Frequency of of 1AB2 FDS
1 - CAT. 1	1.3E-5	1.0E+0	1.3E-5
2 - CAT. 2	2.8E-5	1.0E+0	2.8E-5
3 - CAT. 3	3.7E-5	5.0E-2	1.9E-6
TOTAL CONTRIBUTION	=		4.3E-5

TABLE 3.3.7-5
CONTRIBUTION OF REMOTELY ISOLABLE FLOODING
TO THE FREQUENCY OF 1AB2 FDS

Contributor	Frequency	Probability of Failure to Isolate	Contribution to the Frequency of of 1AB2 FDS
1 - CAT. 1	9.3E-5	1.0E-1	9.3E-6
2 - CAT. 2	5.8E-4	8.0E-2	4.6E-5
3 - CAT. 3	3.7E-4	5.0E-3	1.9E-6
TOTAL CONTRIBUTION =			5.7E-5

TABLE 3.3.7-6
FREQUENCY OF FIRE PROTECTION SYSTEM FLOODING EVENTS IN THE AUXILIARY BUILDING

CAT. 1 (2700 < FLOW RATE< 8000)	FREQUENCY (per year)
1 - Most severe rupture of 4" pipe	7.0E-6
2 - Most severe rupture of 6" pipe	1.7E-6
3 - Most severe rupture of any one of four manual valves	2.5E-6
TOTAL FREQUENCY FOR CAT. 1 =	1.1E-5

CAT. 2 (900 < FLOW RATE< 2700)	FREQUENCY (per year)
1 - Severe rupture of 4" pipe	2.1E-5
2 - Severe rupture of 6" pipe	5.1E-6
3 - Severe rupture of any one of four manual valves	7.5E-6
TOTAL FREQUENCY FOR CAT. 2 =	3.4E-5

CAT. 3 (FLOW RATE< 900)	FREQUENCY (per year)
1 - Least severe rupture of 4" pipe	4.2E-5
2 - Least severe rupture of 6" pipe	1.0E-5
3 - Least severe rupture of any one of four manual valves	1.5E-5
TOTAL FREQUENCY FOR CAT. 3 =	6.7E-5

TABLE 3.3.7-7
CONTRIBUTION OF FIRE PROTECTION SYSTEM FLOODING
TO THE FREQUENCY OF 1AB4

Contributor	Frequency	Probability of Failure to Isolate	Contribution to the Frequency of of 1AB2 FDS
1 - CAT. 1	1.1E-5	3.6E-2	4.0E-7
2 - CAT. 2	3.4E-5	4.7E-3	1.6E-7
3 - CAT. 3	6.7E-5	8.8E-4	5.9E-8
TOTAL CONTRIBUTION	=		6.2E-7

TABLE 3.3.7-8
FREQUENCY OF SW FLOOD CATEGORIES
IN THE AIR CONDITIONING AND CHILLER ROOM

CAT. 1 (1000 gpm < FLOW RATE< 1500 gpm)	Frequency/ Year
1 - Most Severe Rupture Of 4" Diameter Piping	9.1E-6
2 - Most Severe Rupture of Any One of 3 MOVs	8.3E-6
3 - Most Severe Rupture of Any One of 6, 4" MVs	3.8E-6
4 - Most Severe Rupture of Any One of 9, 4" EJ	2.3E-4
5 - Most Severe Rupture of Any 1 of 3, 4" CK Valves	7.6E-6
CAT. 1 TOTAL CONTRIBUTION =	2.6E-4
CAT. 2 (500 gpm < FLOW RATE< 1000 gpm)	
1 - Severe Rupture Of 4" Diameter Piping	2.7E-5
2 - Severe Rupture of Any One of 3 MOVs	2.5E-5
3 - Severe Rupture of Any One of 6, 4" MVs	1.1E-5
4 - Severe Rupture of Any One of 9, 4" EJ	6.8E-4
5 - Severe Rupture of Any 1 of 3, 4" CK Valves	2.3E-5
6 - Most Severe Rupture of Any One of 6 AOVs (3"DIA)	5.8E-6
7 - Most Severe Rupture of 3" Piping Section	1.9E-6
8 - Most Severe Rupture of Any 1 of 3, 3" CK Valves	7.6E-6
CAT. 2 TOTAL CONTRIBUTION =	7.8E-4

CAT. 3 (200 gpm < FLOW RATE< 500 gpm)	Frequency/ Year
1 - Severe Rupture of Any One of 6 AOVs (3"DIA)	5.7E-6
2 - Severe Rupture of 3" Piping Section	1.7E-5
3 - Severe Rupture of Any 1 of 3, 3" CK Valves	2.3E-5
CAT. 3 TOTAL CONTRIBUTION =	4.6E-5

TABLE 3.3.7-9
CONTRIBUTION OF SW FLOODING
TO THE FREQUENCY OF 1AC1 FDS

Contributor	Frequency	Probability of Failure to Isolate	Contribution to the Frequency of of 1AC1 FDS
1 - CAT. 1	2.6E-4	1.0E+0	2.6E-4
2 - CAT. 2	7.8E-4	3.9E-1	3.0E-4
3 - CAT. 3	4.6E-5	2.8E-2	1.3E-6
TOTAL CONTRIBUTION =			5.6E-4

TABLE 3.4.1-1
SUMMARY OF CORE DAMAGE FREQUENCY BY INITIATING EVENT

Initiating Event	Core Damage Frequency (per year)	Fraction of Total
S2 Small Loca	1.01E-5	14.8%
T1A	7.98E-6	11.7%
T1Tr	7.27E-6	10.7%
T7 SGTR	7.02E-6	10.3%
S1 Med Loca	6.64E-6	9.8%
T8 ESGR	6.56E-6	9.7%
T1 LOSP	4.60E-6	6.8%
A Large LOCA	4.09E-6	6.0%
T3Tr	4.06E-6	6.0%
T9ATr	3.26E-6	4.8%
T2ATr	1.65E-6	2.4%
VX ISLOCA	1.60E-6	2.4%
T2	8.86E-7	1.3%
T9B	5.81E-7	0.9%
TH	4.20E-7	0.6%
T9A	4.15E-7	0.6%
RX	2.68E-7	0.4%
T2Tr	1.44E-7	0.2%
T5A	1.11E-7	0.2%
T5B	1.09E-7	0.2%
T3	7.61E-8	0.1%
T9BTr	6.78E-8	0.1%
T2A	6.11E-8	0.1%
T4	1.07E-8	0.0%
T6	4.52E-9	0.0%
TL	0.00E+0	0.0%
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TOTAL	6.80E-5	100.0%

TABLE 3.3.7-10
SUMMARY OF NORTH ANNA INTERNAL FLOOD DAMAGE STATES

Flood Source	Flood Damage State	Frequency	Internal Events Model
	AUXILIARY BUILDING		
SW (Unit 1)	1AB2 - Loss of All Charging Pumps - Loss of All CC Pumps - Loss of SW To RS Heat Exchangers	1.0E-4 Table E.3-9/10	Loss of Seal Cooling (T4)
Fire Protection System (Unit 1)	1AB4 - Loss of All Charging Pumps - Loss of All CC Pumps	6.2E-7 Table E.3-13	Loss of Seal Cooling (T4) Conservatively Quantified=1AB2
	AIR CONDITIONING AND CHILLER ROOM		
SW Supply to HVAC	1AC1 - Loss of Control/Relay Room HVAC Chillers	5.6E-4 Table E.3-19	Loss of HVAC (Units 1 & 2)

TABLE 3.4.1-2
SUMMARY OF CORE DAMAGE SEQUENCES GREATER THAN 1E-7/YEAR, GROUPED BY INITIATOR

Initiator	Sequence	Core Damage Frequency (per year)	Fraction of Total	Functional Failures
S2	Sum =	1.01E-5	14.8%	
	S2P35	5.15E-6	7.6%	S2D1D3
	S2P04	2.45E-6	3.6%	S2H1
	S2P43	1.19E-6	1.8%	S2D1Y
	S2P39	5.20E-7	0.8%	S2D1D2
	S2P47	3.27E-7	0.5%	S2D1L
T1A	Sum =	7.98E-6	11.7%	
	T1AP51	2.99E-6	4.4%	T1ALtBB1
	T1AP46	1.41E-6	2.1%	T1ALtB
	T1AP07	1.38E-6	2.0%	T1ABB1
	T1AP67	8.86E-7	1.3%	T1AQBB1
	T1AP02	6.51E-7	1.0%	T1AB
	T1AP58	4.17E-7	0.6%	T1AQB
	T1AP26	1.04E-7	0.2%	T1AS1cBB1
T1Tr	Sum =	7.27E-6	10.7%	
	T1TrP17	4.00E-6	5.9%	T1TrOD1
	T1TrP21	2.22E-6	3.3%	T1TrOD1Qs
	T1TrP14	1.01E-6	1.5%	T1TrOH1
T7	Sum =	7.02E-6	10.3%	
	T7P04	2.98E-6	4.4%	T7OO2
	T7P03	1.98E-6	2.9%	T7OW
	T7P06	1.10E-6	1.6%	T7SGIW
	T7P26	3.85E-7	0.6%	T7D1SGI
	T7P23	1.80E-7	0.3%	T7D1OD3
	T7P07	1.10E-7	0.2%	T7SGIO2
S1	Sum =	6.64E-6	9.8%	
	S1P38	4.04E-6	5.9%	S1D1Y
	S1P10	2.45E-6	3.6%	S1OH2
T8	Sum =	6.56E-6	9.7%	
	T8P22	3.17E-6	4.7%	T8LtRC1
	T8P02	2.52E-6	3.7%	T8RC2
	T8P06	6.06E-7	0.9%	T8RC2RC3

TABLE 3.4.1-2 (Continued)
SUMMARY OF CORE DAMAGE SEQUENCES GREATER THAN 1E-7/YEAR, GROUPED BY INITIATOR

Initiator	Sequence	Core Damage Frequency (per year)	Fraction of Total	Functional Failures
T1	Sum =	4.60E-6	6.8%	
	T1P10	2.71E-6	4.0%	T1LD1
	T1P07	5.66E-7	0.8%	T1LH2H1
	T1P15	5.16E-7	0.8%	T1LP
	T1P14	2.07E-7	0.3%	T1LD1Qs
	T1P19	1.91E-7	0.3%	T1LPQs
	T1P06	1.69E-7	0.2%	T1LH2
A	Sum =	4.09E-6	6.0%	
	AP15	2.12E-6	3.1%	AD2
	AP03	8.26E-7	1.2%	AH1
	AP11	5.88E-7	0.9%	AD3
	AP02	5.17E-7	0.8%	ADh
T3Tr	Sum =	4.06E-6	6.0%	
	T3TrP11	1.67E-6	2.5%	T3TrOD1
	T3TrP03	1.57E-6	2.3%	T3TrRC2Ch
	T3TrP06	2.83E-7	0.4%	T3TrRC2RC3
	T3TrP23	1.84E-7	0.3%	T3TrLtRC1Ch
	T3TrP22	1.21E-7	0.2%	T3TrLtRC1
T9ATr	Sum =	3.26E-6	4.8%	
	T9ATrP08	1.53E-6	2.2%	T9ATrLtRC1
	T9ATrP02	8.33E-7	1.2%	T9ATrRC2
	T9ATrP14	3.88E-7	0.6%	T9ATrOH1
	T9ATrP17	3.07E-7	0.5%	T9ATrOD1
	T9ATrP06	1.07E-7	0.2%	T9ATrRC2RC3
T2ATr	Sum =	1.65E-6	2.4%	
	T2ATrP11	6.78E-7	1.0%	T2ATrOD1
	T2ATrP03	6.40E-7	0.9%	T2ATrRC2Ch
	T2ATrP06	1.15E-7	0.2%	T2ATrRC2RC3
VX	Sum =	1.60E-6	2.4%	
	VXP07	1.52E-6	2.2%	VXFm
T2	Sum =	8.86E-7	1.3%	
	T2P09	7.22E-7	1.1%	T2LD1
	T2P14	1.30E-7	0.2%	T2LP

TABLE 3.4.1-2 (Continued)
SUMMARY OF CORE DAMAGE SEQUENCES GREATER THAN 1E-7/YEAR, GROUPED BY INITIATOR

Initiator	Sequence	Core Damage Frequency (per year)	Fraction of Total	Functional Failures
T9B	Sum =	5.81E-7	0.9%	
	T9BP02	2.37E-7	0.3%	T9BL
	T9BP10	1.79E-7	0.3%	T9BQH2
	T9BP13	1.30E-7	0.2%	T9BQD1
TH	Sum =	4.20E-7	0.6%	
	THP46	2.14E-7	0.3%	THKMTtQ
	THP30	2.06E-7	0.3%	THKMPr
T9A	Sum =	4.15E-7	0.6%	
	T9AP02	1.72E-7	0.3%	T9AL
	T9AP10	1.31E-7	0.2%	T9AQH2
	T9AP13	1.02E-7	0.2%	T9AQD1
RX	Sum =	2.68E-7	0.4%	
	RXP01	2.66E-7	0.4%	RX
T2Tr	Sum =	1.44E-7	0.2%	
T5A	Sum =	1.11E-7	0.2%	
T5B	Sum =	1.09E-7	0.2%	
T3	Sum =	7.61E-8	0.1%	
T9BTr	Sum =	6.78E-8	0.1%	
T2A	Sum =	6.11E-8	0.1%	
T4	Sum =	1.07E-8	0.0%	
T6	Sum =	4.52E-9	0.0%	
TL	Sum =	0.00E+0	0.0%	

TABLE 3.4.1-3
SUMMARY OF CORE DAMAGE SEQUENCES WHICH CONTRIBUTE TO THE UPPER 95% OF THE TOTAL CORE DAMAGE FREQUENCY

Sequence	Core Damage Frequency (per year)	Fraction of Total	Functional Failures
S2P35	5.15E-6	7.6%	S2D1D3
S1P38	4.04E-6	5.9%	S1D1Y
T1TrP17	4.00E-6	5.9%	T1TrOD1
T8P22	3.17E-6	4.7%	T8LtRC1
T1AP51	2.99E-6	4.4%	T1ALtBB1
T7P04	2.98E-6	4.4%	T7OO2
T1P10	2.71E-6	4.0%	T1LD1
T8P02	2.52E-6	3.7%	T8RC2
S1P10	2.45E-6	3.6%	S1OH2
S2P04	2.45E-6	3.6%	S2H1
T1TrP21	2.22E-6	3.3%	T1TrOD1Qs
AP15	2.12E-6	3.1%	AD2
T7P03	1.98E-6	2.9%	T7OW
T3TrP11	1.67E-6	2.5%	T3TrOD1
T3TrP03	1.57E-6	2.3%	T3TrRC2Ch
T9ATrP08	1.53E-6	2.2%	T9ATrLtRC1
VXP07	1.52E-6	2.2%	VXFm
T1AP46	1.41E-6	2.1%	T1ALtB
T1AP07	1.38E-6	2.0%	T1ABB1
S2P43	1.19E-6	1.8%	S2D1Y
T7P06	1.10E-6	1.6%	T7SGIW
T1TrP14	1.01E-6	1.5%	T1TrOH1
T1AP67	8.86E-7	1.3%	T1AQBB1
T9ATrP02	8.33E-7	1.2%	T9ATrRC2
AP03	8.26E-7	1.2%	AH1
T2P09	7.22E-7	1.1%	T2LD1
T2ATrP11	6.78E-7	1.0%	T2ATrOD1
T1AP02	6.51E-7	1.0%	T1AB
T2ATrP03	6.40E-7	0.9%	T2ATrRC2Ch
T8P06	6.06E-7	0.9%	T8RC2RC3
AP11	5.88E-7	0.9%	AD3
T1P07	5.66E-7	0.8%	T1LH2H1
S2P39	5.20E-7	0.8%	S2D1D2
AP02	5.17E-7	0.8%	ADh
T1P15	5.16E-7	0.8%	T1LP
T1AP58	4.17E-7	0.6%	T1AQB
T9ATrP14	3.88E-7	0.6%	T9ATrOH1
T7P26	3.85E-7	0.6%	T7D1SGI
S2P47	3.27E-7	0.5%	S2D1L
T9ATrP17	3.07E-7	0.5%	T9ATrOD1
T3TrP06	2.83E-7	0.4%	T3TrRC2RC3
RXP01	2.66E-7	0.4%	RX

TABLE 3.4.1-3 (Continued)
SUMMARY OF CORE DAMAGE SEQUENCES WHICH CONTRIBUTE TO THE UPPER 95% OF THE TOTAL CORE DAMAGE FREQUENCY

Sequence	Core Damage Frequency (per year)	Fraction of Total	Functional Failures
T9BP02	2.37E-7	0.3%	T9BL
THP46	2.14E-7	0.3%	THKMTtQ
T1P14	2.07E-7	0.3%	T1LD1Qs
THP30	2.06E-7	0.3%	THKMPr
T1P19	1.91E-7	0.3%	T1LPQs
T3TrP23	1.84E-7	0.3%	T3TrLtRC1Ch
T7P23	1.80E-7	0.3%	T7D1OD3
T9BP10	1.79E-7	0.3%	T9BQH2
T9AP02	1.72E-7	0.3%	T9AL
T1P06	1.69E-7	0.2%	T1LH2
T9AP10	1.31E-7	0.2%	T9AQH2
T2P14	1.30E-7	0.2%	T2LP
T9BP13	1.30E-7	0.2%	T9BQD1
T3TrP22	1.21E-7	0.2%	T3TrLtRC1
T2ATrP06	1.15E-7	0.2%	T2ATrRC2RC3
T7P07	1.10E-7	0.2%	T7SGIO2
T9ATrP06	1.07E-7	0.2%	T9ATrRC2RC3
T1AP26	1.04E-7	0.2%	T1AS1cBB1
T9AP13	1.02E-7	0.2%	T9AQD1

TABLE 3.4.1-4
SUMMARY OF SEQUENCES THAT CONTRIBUTE TO A CONTAINMENT BYPASS FREQUENCY IN EXCESS OF 1E-8 PER REACTOR YEAR

Sequence	Frequency (per year)
Primary to Secondary Leakage (Steam Generator Tube Rupture)	
T7P04	2.98E-6
T7P03	1.98E-6
T7P06	1.10E-6
T7P26	3.85E-7
T7P23	1.80E-7
T7P07	1.10E-7
T7P25	8.44E-8
T7P27	7.20E-8
T7P14	3.15E-8
T7P22	3.10E-8
T7P15	2.59E-8
T7P24	1.86E-8
Interfacing LOCA	
VXP07	1.52E-6
VXP03	7.68E-8
Containment Failed at Time of Core Damage	
S2P02	3.22E-8
S1P08	3.19E-8
AP04	1.60E-8
Containment Bypassed	
ISP01	2.85E-8

TABLE 3.4.1-5
DOMINANT ACCIDENT SEQUENCES BY INITIATING EVENT TYPE

Initiating Event Type	Point Estimate Frequency Per Year	Percentage of Total
LOCA	2.1E-5	31%
Loss of Offsite Power	2.0E-5	29%
Transient	1.8E-5	27%
Steam Generators Tube Rupture	7.0E-6	10%
Interfacing System LOCA	1.6E-6	2%
ATWS	4.2E-7	1%
Total	6.8E-5	100%

TABLE 3.4.1-6
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

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
42	HEP-1AP22:5	1.750E-4	3.367E-2	193.37	1.035
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-B1Q3	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

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
83	1SIMOV-CC-1115BD	3.903E-4	1.323E-2	34.88	1.013
84	NON-REC-B102	6.799E-1	1.313E-2	1.01	1.013
85	1SIPSB-UM-1SIP1B	4.536E-3	1.246E-2	3.73	1.013
86	1FWCKV-CC-ALLAFW	6.339E-5	1.210E-2	191.93	1.012
87	1SIPSB-FS-1SIP1B	4.018E-3	1.210E-2	4.00	1.012
88	1SIPSB-UM-1SIP1A	4.536E-3	1.166E-2	3.56	1.012
89	1SIMOV-FO-1115E	1.090E-2	1.134E-2	2.03	1.011
90	1SIMOV-FC-1115B	1.090E-2	1.134E-2	2.03	1.011
91	1RCPORV-T3	6.651E-3	1.133E-2	2.69	1.011
92	1SIPSB-FS-1SIP1A	4.018E-3	1.122E-2	3.78	1.011
93	REC-1FRH:1-4	1.131E-2	1.065E-2	1.93	1.011
94	1HVPCV-FC-1235B1	1.812E-2	1.051E-2	1.57	1.011
95	1HVTCV-FC-TCV167	1.812E-2	1.040E-2	1.56	1.011
96	C-L08	8.410E-1	1.027E-2	1.00	1.010
97	HEP-1ECA3:1-16	3.025E-3	9.604E-3	4.17	1.010
98	C-B117	3.200E-1	9.597E-3	1.02	1.010
99	1EEBKR-SO-15H8	3.356E-5	9.561E-3	285.88	1.010
100	HEP-1ES1:3	1.220E-2	9.378E-3	1.76	1.009
101	1EEBKR-SO-14H1	3.356E-5	9.332E-3	279.06	1.009
102	1FWPSB-UM-1FWP3A	5.183E-3	9.273E-3	2.78	1.009
103	1RHHCV-FC-1758	1.812E-2	9.258E-3	1.50	1.009
104	1RCRV--CC-RCPORV	9.988E-4	9.040E-3	10.04	1.009
105	1FWPSB-UM-1FWP3B	5.183E-3	9.027E-3	2.73	1.009
106	C-H105	9.490E-1	8.554E-3	1.00	1.009
107	1EEBKR-SO-14H2	3.356E-5	8.525E-3	255.01	1.009
108	T9A-FREQ-RSST-C	7.143E-2	8.311E-3	1.11	1.008
109	HEP-1FRH:1-15	8.249E-3	8.273E-3	1.99	1.008
110	1SIMOV-FC-1863B	1.090E-2	7.688E-3	1.70	1.008
111	1MSRV--CC-101ABC	9.988E-4	7.573E-3	8.57	1.008
112	HEP-1E3-3	3.650E-3	7.421E-3	3.03	1.007
113	1SIMOV-PG-1865C	8.207E-4	7.166E-3	9.72	1.007
114	1SIMOV-PG-1865A	8.207E-4	7.166E-3	9.72	1.007
115	1EEBUS-UM-DC-III	2.000E-4	7.079E-3	36.39	1.007
116	1MSCKV-FO-1MS58	3.442E-3	6.998E-3	3.03	1.007
117	1MSCKV-FO-1MS19	3.442E-3	6.998E-3	3.03	1.007
118	HEP-1ES1:4	8.499E-4	6.308E-3	8.42	1.006
119	1SIMOV-FO-1115C	1.090E-2	6.306E-3	1.57	1.006
120	1SIMOV-FC-1115D	1.090E-2	6.306E-3	1.57	1.006
121	PROB-M03	2.942E-1	6.187E-3	1.01	1.006
122	IE-TH	1.750E+0	6.187E-3	1.00	1.006
123	C-B102	3.200E-1	6.155E-3	1.01	1.006

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
124	C-HV05	7.490E-1	6.120E-3	1.00	1.006
125	C-QS06	9.460E-1	6.112E-3	1.00	1.006
126	1RPROD-LF-CRODS	1.800E-6	5.718E-3	3178.03	1.006
127	1RHHEX-LF-1RHE2B	2.807E-2	5.675E-3	1.20	1.006
128	C-H106	9.356E-1	5.643E-3	1.00	1.006
129	1RHHEX-LF-1RHE2A	2.807E-2	5.640E-3	1.20	1.006
130	1RHMOV-FC-1700	1.090E-2	5.570E-3	1.51	1.006
131	1RHMOV-FC-1701	1.090E-2	5.570E-3	1.51	1.006
132	1SICKV-FC-1SI161	6.339E-4	5.540E-3	9.73	1.006
133	1SICKV-FC-1SI159	6.339E-4	5.534E-3	9.72	1.006
134	1SICKV-FC-1SI127	6.339E-4	5.534E-3	9.72	1.006
135	1SICKV-FC-1SI125	6.339E-4	5.534E-3	9.72	1.006
136	1EEBUS-UM-DC-I	2.000E-4	5.508E-3	28.54	1.006
137	1EGEDG-CC-1H1J2J	9.576E-5	5.252E-3	55.84	1.005
138	1EETFM-LP-1H	1.899E-5	5.134E-3	271.32	1.005
139	1EGEDG-CC-1H1J2H	9.576E-5	5.043E-3	53.66	1.005
140	1HVMOV-FC-111B	1.090E-2	4.928E-3	1.45	1.005
141	1HVMOV-FC-113B	1.090E-2	4.928E-3	1.45	1.005
142	1HVMOD-FO-MOD137	1.090E-2	4.870E-3	1.44	1.005
143	1HVMOD-FC-MOD138	1.090E-2	4.870E-3	1.44	1.005
144	HEP-1ES1:2-S2	8.499E-4	4.624E-3	6.44	1.005
145	1RHHEX-LF-1RHE1B	2.807E-2	4.620E-3	1.16	1.005
146	1RHHEX-LF-1RHE1A	2.807E-2	4.546E-3	1.16	1.005
147	1FWPSB-FS-1FWP3A	1.583E-3	4.395E-3	3.77	1.004
148	T9B-FREQ-500KV-2	1.786E-1	4.317E-3	1.02	1.004
149	1FWPSB-FS-1FWP3B	1.583E-3	4.280E-3	3.70	1.004
150	1MSAOV-CC-111AB	1.812E-3	4.077E-3	3.25	1.004
151	2IAIAS-LF-OUTIA	2.520E-4	3.980E-3	16.79	1.004
152	IE-RX	2.664E-7	3.946E-3	14814.19	1.004
153	1EP-LOOP-24	3.120E-4	3.850E-3	13.34	1.004
154	1FWPSB-CC-MDP3AB	1.418E-4	3.839E-3	28.07	1.004
155	REC-2AP28	1.017E-1	3.809E-3	1.03	1.004
156	1SIMOV-FC-1863A	1.090E-2	3.673E-3	1.33	1.004
157	1EEBKR-SO-15J8	3.356E-5	3.646E-3	109.63	1.004
158	1EEBKR-SO-14J1	3.356E-5	3.646E-3	109.63	1.004
159	HEP-1ECA3:3-27	8.974E-2	3.578E-3	1.04	1.004
160	2HVCHU-UM-2HVE4B	9.440E-2	3.517E-3	1.03	1.004
161	T9B-FREQ-4160-1J	5.999E-3	3.464E-3	1.57	1.003
162	1EGEDG-CC-ALL	6.090E-5	3.334E-3	55.74	1.003
163	1EEBUS-LU-1H-480	1.215E-5	3.334E-3	275.48	1.003
164	1EEBUS-LU-1H1	1.215E-5	3.334E-3	275.48	1.003

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
165	1EEBUS-LU-1H	1.215E-5	3.330E-3	275.14	1.003
166	1CHPAT-FS-1CHP1B	5.078E-3	3.306E-3	1.65	1.003
167	C-D105	9.470E-1	3.255E-3	1.00	1.003
168	PROB-Q08	1.000E+0	3.154E-3	1.00	1.003
169	1RCMOV-LK-1536	2.500E-2	3.094E-3	1.12	1.003
170	REC-MMP-C-MR-2	2.510E-1	3.072E-3	1.01	1.003
171	1RCMOV-LK-1535	2.500E-2	3.060E-3	1.12	1.003
172	C-TT01	8.000E-1	3.033E-3	1.00	1.003
173	PROB-PR01	2.776E-1	3.033E-3	1.01	1.003
174	1EEBUS-LU-1H1-4	1.215E-5	2.993E-3	247.47	1.003
175	1SWPIP-UM-HDRA	2.281E-2	2.987E-3	1.13	1.003
176	2EGEDG-UM-2J	1.069E-1	2.985E-3	1.02	1.003
177	1RCPCV-FC-1455A	1.812E-2	2.968E-3	1.16	1.003
178	HEP-1AP15-6	2.815E-2	2.952E-3	1.10	1.003
179	1SIMOV-CC-1890CD	3.903E-4	2.902E-3	8.43	1.003
180	1SW-HOTWEA-9MO	7.500E-1	2.874E-3	1.00	1.003
181	2HVSTR-PG-2HVS1B	9.528E-3	2.873E-3	1.30	1.003
182	1SICKV-FO-1SI47	3.442E-3	2.817E-3	1.82	1.003
183	1EEBKR-SO-14J4	3.356E-5	2.807E-3	84.64	1.003
184	2HVCHU-FS-2HVE4B	4.545E-2	2.800E-3	1.06	1.003
185	2HVPCV-FC-2235B1	1.812E-2	2.666E-3	1.14	1.003
186	1SICKV-CC-FC926	6.339E-5	2.574E-3	41.60	1.003
187	1SICKV-CC-FC116	6.339E-5	2.571E-3	41.55	1.003
188	1RCRV--FC-1455C	9.988E-3	2.569E-3	1.25	1.003
189	1EEBUS-UM-1H	1.000E-5	2.556E-3	256.64	1.003
190	1EEBUS-UM-1H-480	1.000E-5	2.498E-3	250.86	1.003
191	1SIMOV-FC-1867C	1.090E-2	2.411E-3	1.22	1.002
192	1SIMOV-FC-1867A	1.090E-2	2.411E-3	1.22	1.002
193	2EEBUS-UM-2H1-1	2.000E-4	2.323E-3	12.62	1.002
194	2EEBUS-UM-2H-480	2.000E-4	2.323E-3	12.62	1.002
195	2EEBUS-UM-2H	2.000E-4	2.323E-3	12.62	1.002
196	NON-REC-B10	2.000E-2	2.285E-3	1.11	1.002
197	1EEBUS-UM-1H1-4	1.000E-5	2.280E-3	229.03	1.002
198	1SIMOV-PG-1860B	1.357E-3	2.221E-3	2.63	1.002
199	1FWHEP-1FW548	7.499E-4	2.219E-3	3.96	1.002
200	1FWHEP-1FW546	7.499E-4	2.168E-3	3.89	1.002
201	1SICKV-CC-79185	6.339E-5	2.121E-3	34.46	1.002
202	1SIMOV-PG-1860A	1.357E-3	2.117E-3	2.56	1.002
203	2HVTCV-FC-TCV266	1.812E-2	2.108E-3	1.11	1.002
204	1FWPSB-FR-24HP3A	7.927E-4	2.081E-3	3.62	1.002
205	1EE-BAT-IV-2HR	1.000E+0	2.039E-3	1.00	1.002

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
206	1SWMOV-FC-1SW117	1.090E-2	2.030E-3	1.18	1.002
207	1RHFEL-PG-1605	4.105E-4	2.010E-3	5.90	1.002
208	1FWPSB-FR-24HP3B	7.927E-4	2.001E-3	3.52	1.002
209	1EETFM-LP-1J	1.899E-5	1.974E-3	104.94	1.002
210	1RHPSB-CC-1RHP1	3.933E-4	1.926E-3	5.90	1.002
211	2HVPAT-FR-HVP22A	7.930E-4	1.901E-3	3.39	1.002
212	2HVPAT-FR-HVP20A	7.930E-4	1.901E-3	3.39	1.002
213	1FWCKV-LEAKAGE	1.000E-5	1.802E-3	181.25	1.002
214	2HVCHU-FR-2HVE4A	1.506E-3	1.778E-3	2.18	1.002
215	HEP-1OP14:1-5:13	4.259E-3	1.728E-3	1.40	1.002
216	1SICKV-FC-1SI9	6.339E-4	1.703E-3	3.69	1.002
217	1SIMOV-PG-1864A	8.207E-4	1.697E-3	3.07	1.002
218	1SIMOV-PG-1864B	8.207E-4	1.688E-3	3.06	1.002
219	1HVCHU-CC-HVE4	4.547E-3	1.684E-3	1.37	1.002
220	1SICKV-FC-1SI18	6.339E-4	1.681E-3	3.65	1.002
221	1SICKV-FC-1SI26	6.339E-4	1.681E-3	3.65	1.002
222	T9B-FREQ-RSST-A	7.143E-2	1.678E-3	1.02	1.002
223	1CESTR-CC-SUMPPG	5.000E-5	1.670E-3	34.39	1.002
224	1QSSTR-PG-1FL1B	2.822E-2	1.657E-3	1.06	1.002
225	1FWCKV-FC-1FW165	6.339E-4	1.648E-3	3.60	1.002
226	HEP-1E0-7	1.350E-3	1.645E-3	2.22	1.002
227	IE-T5A	5.999E-3	1.613E-3	1.27	1.002
228	IE-T5B	5.999E-3	1.589E-3	1.26	1.002
229	1FWCKV-FC-1FW183	6.339E-4	1.580E-3	3.49	1.002
230	1FWHEP-1FW543	7.499E-4	1.567E-3	3.09	1.002
231	REC-CONTAINMENT	2.000E-2	1.552E-3	1.08	1.002
232	NON-REC-B111	6.799E-1	1.530E-3	1.00	1.002
233	2HVCHU-CC-HVE4	4.547E-3	1.515E-3	1.33	1.002
234	1SIMV--PG-1SI46	4.499E-5	1.497E-3	34.26	1.001
235	2HVMOV-FC-211B	1.090E-2	1.484E-3	1.13	1.001
236	2HVMOV-FC-213B	1.090E-2	1.484E-3	1.13	1.001
237	C-QS03	9.460E-1	1.446E-3	1.00	1.001
238	2EEBUS-UM-2H1-4	2.000E-4	1.445E-3	8.22	1.001
239	C-QS04	9.460E-1	1.426E-3	1.00	1.001
240	C-H103	9.610E-1	1.401E-3	1.00	1.001
241	1HVCHU-FR-1HVE4A	1.506E-3	1.390E-3	1.92	1.001
242	2HVCHU-UM-2HVE4C	9.440E-2	1.388E-3	1.01	1.001
243	C-H104	9.620E-1	1.381E-3	1.00	1.001
244	1MSPIC-LF-1447	8.022E-2	1.330E-3	1.02	1.001
245	1MSPIC-LF-1446	8.022E-2	1.330E-3	1.02	1.001
246	1FWCKV-FC-1FW148	6.339E-4	1.299E-3	3.05	1.001

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
247	1EEBUS-LU-1J-480	1.215E-5	1.258E-3	104.55	1.001
248	1EEBUS-LU-1J1	1.215E-5	1.258E-3	104.55	1.001
249	1EEBUS-LU-1J	1.215E-5	1.248E-3	103.72	1.001
250	1SIPSB-FR-24HP1B	7.927E-4	1.243E-3	2.57	1.001
251	1MSAOV-FO-TV101C	1.812E-2	1.240E-3	1.07	1.001
252	HEP-1ECA3:2-5	7.249E-4	1.219E-3	2.68	1.001
253	2HVMOD-FO-MOD238	1.090E-2	1.219E-3	1.11	1.001
254	2HVMOD-FC-MOD237	1.090E-2	1.219E-3	1.11	1.001
255	1SIPSB-FR-24HP1A	7.927E-4	1.203E-3	2.52	1.001
256	C-Y03	8.980E-1	1.147E-3	1.00	1.001
257	1SICKV-FC-1SI12	6.339E-4	1.134E-3	2.79	1.001
258	1SICKV-FC-1SI29	6.339E-4	1.129E-3	2.78	1.001
259	1SICKV-CC-FC1229	6.339E-5	1.081E-3	18.06	1.001
260	1SIMOV-PG-1865B	8.207E-4	1.080E-3	2.31	1.001
261	1HVFAN-FS-1FMO7	3.933E-3	1.040E-3	1.26	1.001
262	2EGEDG-UM-2H	1.069E-1	1.036E-3	1.01	1.001
263	1QSSTR-PG-1FL1A	2.822E-2	1.009E-3	1.03	1.001
264	1MSSV--FO-101C	1.250E-2	1.006E-3	1.08	1.001
265	1MSSRV-DMDT7	3.999E-2	1.006E-3	1.02	1.001
266	1RSHEP-FLANGE	3.750E-4	9.962E-4	3.66	1.001
267	1EEBUS-LU-1J1-1	1.215E-5	9.851E-4	82.11	1.001
268	1SICKV-FC-1SI1	6.339E-4	9.476E-4	2.49	1.001
269	1SW-COLDWEA-3MO	2.500E-1	9.437E-4	1.00	1.001
270	1SWPAT-FS-1SWP1B	3.842E-3	9.411E-4	1.24	1.001
271	1SICKV-FC-1SI16	6.339E-4	9.261E-4	2.46	1.001
272	1SWPAT-UM-1SWP1B	3.750E-3	9.185E-4	1.24	1.001
273	1SWMOV-CC-103A-D	3.903E-4	9.136E-4	3.34	1.001
274	1SWMOV-CC-105A-D	3.903E-4	9.136E-4	3.34	1.001
275	1SWMOV-CC-101A-D	3.903E-4	9.136E-4	3.34	1.001
276	1SWMOV-CC-104A-D	3.903E-4	9.136E-4	3.34	1.001
277	1CHPAT-UM-1CHP1C	3.267E-1	9.008E-4	1.00	1.001
278	1CCAOV-FC-TV103B	1.812E-2	9.002E-4	1.05	1.001
279	2HVCHU-FS-2HVE4C	4.545E-2	8.969E-4	1.02	1.001
280	1CCAOV-FC-TV103A	1.812E-2	8.953E-4	1.05	1.001
281	1SWTCV-CC-102BC	1.812E-3	8.842E-4	1.49	1.001
282	1EGEDG-TM-1H	5.708E-4	8.783E-4	2.54	1.001
283	1RCPIC-LF-PC402	4.123E-2	8.598E-4	1.02	1.001
284	1RCPIC-LF-PC403	4.123E-2	8.598E-4	1.02	1.001
285	1RHPSB-FS-1RHP1B	3.933E-3	8.524E-4	1.22	1.001
286	1RHPSB-FS-1RHP1A	3.933E-3	8.487E-4	1.21	1.001
287	1SICKV-FC-1SI144	6.339E-4	8.387E-4	2.32	1.001

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
288	1SICKV-FC-1SI142	6.339E-4	8.334E-4	2.31	1.001
289	1SWPIP-UM-HDRB	2.281E-2	8.311E-4	1.04	1.001
290	1EGEDG-CC-1H-2J	2.663E-4	8.302E-4	4.12	1.001
291	1SWPAT-CC-SWP1B	3.842E-4	8.204E-4	3.13	1.001
292	1CHPAT-FS-1CHP1A	1.983E-3	8.171E-4	1.41	1.001
293	1EGEDG-TM-1J	5.708E-4	8.046E-4	2.41	1.001
294	1EEBUS-UM-1J-480	1.000E-5	8.015E-4	81.16	1.001
295	1EEBUS-UM-1J	1.000E-5	7.929E-4	80.30	1.001
296	1EEBKR-FO-15H2	2.735E-4	7.754E-4	3.83	1.001
297	1EGEDG-CC-1H-2H	2.663E-4	7.427E-4	3.79	1.001
298	C-B111	3.200E-1	7.197E-4	1.00	1.001
299	1EEBKR-FO-15J2	2.735E-4	7.151E-4	3.61	1.001
300	1HVCHU-UM-1HVE4C	9.440E-2	7.027E-4	1.01	1.001
301	2SWPAT-UM-2SWP1B	3.725E-2	6.937E-4	1.02	1.001
302	1EGEDG-CC-1J-2J	2.663E-4	6.922E-4	3.60	1.001
303	1EGEDG-CC-1J-2H	2.663E-4	6.911E-4	3.59	1.001
304	1HVCHU-FS-1HVE4C	4.545E-2	6.827E-4	1.01	1.001
305	1HVPAT-FR-HVP20A	7.930E-4	6.749E-4	1.85	1.001
306	1HVPAT-FR-HVP22A	7.930E-4	6.749E-4	1.85	1.001
307	1RHPSB-UM-1RHP1B	3.750E-3	6.745E-4	1.18	1.001
308	1RHPSB-UM-1RHP1A	3.750E-3	6.710E-4	1.18	1.001
309	2EEBKR-SO-25H8	3.356E-5	6.664E-4	20.86	1.001
310	2EEBKR-SO-24H4	3.356E-5	6.664E-4	20.86	1.001
311	2EEBKR-SO-24H1	3.356E-5	6.664E-4	20.86	1.001
312	1MSRV--FC-101B	9.988E-3	6.552E-4	1.06	1.001
313	1MSAOV-FC-TV111A	1.812E-2	6.497E-4	1.04	1.001
314	1MSAOV-FC-TV111B	1.812E-2	6.497E-4	1.04	1.001
315	1HVSTR-PG-1HVS1B	9.528E-3	6.418E-4	1.07	1.001
316	HEP-1E1-25	1.175E-2	6.087E-4	1.05	1.001
317	2HVSTR-PL-2HVS1A	6.390E-4	6.079E-4	1.95	1.001
318	2HVPCV-FC-2235C1	1.812E-2	6.053E-4	1.03	1.001
319	1SIMOV-FO-1115B	1.090E-2	6.050E-4	1.05	1.001
320	1SIMOV-FO-1115D	1.090E-2	6.050E-4	1.05	1.001
321	2HVFAN-FR-2FM07	1.357E-4	6.000E-4	5.42	1.001
322	1SWPSB-FS-1SWP-4	3.152E-3	5.783E-4	1.18	1.001
323	1EEBUS-UM-1J1-1	1.000E-5	5.544E-4	56.44	1.001
324	1SICKV-CC-ACCCKV	6.339E-5	5.510E-4	9.69	1.001
325	1SIMOV-FC-1836	1.090E-2	5.418E-4	1.05	1.001
326	2HVPCV-CC-2235	1.812E-3	5.396E-4	1.30	1.001
327	1SICKV-FC-1SI79	6.339E-4	5.352E-4	1.84	1.001
328	2EEBKR-SO-24H2	3.356E-5	5.242E-4	16.62	1.001

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
329	1FWFCV-CC-788898	1.812E-3	5.205E-4	1.29	1.001
330	2HVCHU-UM-HVE4BC	2.259E-3	5.044E-4	1.22	1.001
331	1HVCHU-UM-HVE4BC	2.259E-3	4.818E-4	1.21	1.000
332	1HVPCV-CC-1235	1.812E-3	4.769E-4	1.26	1.000
333	1SICKV-CC-959903	6.339E-5	4.697E-4	8.41	1.000
334	1SICKV-CC-206207	6.339E-5	4.697E-4	8.41	1.000
335	HEP-1FRS:1-5	2.970E-2	4.696E-4	1.02	1.000
336	1RPBKR-CC-RTARTB	1.300E-5	4.696E-4	37.13	1.000
337	1EEBUS-UM-1H1-2N	2.000E-4	4.505E-4	3.25	1.000
338	1HVSTR-PL-1HVS1A	6.390E-4	4.459E-4	1.70	1.000
339	2HVSV--SO-2200	9.333E-5	4.113E-4	5.41	1.000
340	REC-1ES1:4-1	1.039E-1	3.923E-4	1.00	1.000
341	1HVPAT-FS-HVP22B	1.983E-3	3.884E-4	1.20	1.000
342	1HVPAT-FS-HVP20B	1.983E-3	3.884E-4	1.20	1.000
343	1MSMOV-FO-NRV101	1.090E-2	3.856E-4	1.03	1.000
344	HEP-1FRH:1-5	3.125E-3	3.829E-4	1.12	1.000
345	1QSMOV-FC-101B	1.090E-2	3.808E-4	1.03	1.000
346	2HVFAN-FS-2FM06	3.933E-3	3.787E-4	1.10	1.000
347	1MSRV--FC-101A	9.988E-3	3.634E-4	1.04	1.000
348	2EETFM-LP-2H	1.899E-5	3.484E-4	19.35	1.000
349	1SIMOV-PG-1862A	1.350E-4	3.338E-4	3.47	1.000
350	1SIMV--PG-1SI305	1.350E-4	3.338E-4	3.47	1.000
351	1SIMV--PG-1SI306	1.350E-4	3.308E-4	3.45	1.000
352	1SIMOV-PG-1862B	1.350E-4	3.308E-4	3.45	1.000
353	1EEBUS-LU-DC-I	1.215E-5	3.301E-4	28.17	1.000
354	1EEBUS-LU-DC-III	1.215E-5	3.285E-4	28.05	1.000
355	1CCMOV-FC-CC100B	1.090E-2	3.273E-4	1.03	1.000
356	1CCMOV-FC-CC100A	1.090E-2	3.248E-4	1.03	1.000
357	HEP-0AP55-30HR	6.565E-3	3.173E-4	1.05	1.000
358	1FWPCV-CC-159AB	1.369E-5	3.099E-4	23.64	1.000
359	1SICKV-CC-144161	6.339E-5	3.097E-4	5.89	1.000
360	1RHCKV-CC-1RH715	6.339E-5	3.097E-4	5.89	1.000
361	2HVMOV-FC-211C	1.090E-2	3.085E-4	1.03	1.000
362	2HVMOV-FC-213C	1.090E-2	3.085E-4	1.03	1.000
363	1SIMOV-CC-1867CD	3.903E-4	3.079E-4	1.79	1.000
364	1SIMOV-CC-1867AB	3.903E-4	3.079E-4	1.79	1.000
365	1SWSCN-PG-1SWP1B	9.528E-3	3.053E-4	1.03	1.000
366	1MSTCV-CC-1408AB	1.812E-3	2.958E-4	1.16	1.000
367	1RCPCV-CC-1455AB	1.812E-3	2.950E-4	1.16	1.000
368	1QSMOV-FC-101A	1.090E-2	2.942E-4	1.03	1.000
369	1EPBUS-UM-2	2.000E-4	2.886E-4	2.44	1.000

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
370	1EPBUS-UM-4	2.000E-4	2.886E-4	2.44	1.000
371	1EEBKR-SO-14J5	3.356E-5	2.723E-4	9.11	1.000
372	1CHPAT-FR-24HP1A	7.930E-4	2.683E-4	1.34	1.000
373	1HVCHU-FR-1HVE4B	1.506E-3	2.655E-4	1.18	1.000
374	1CHPAT-FR-24HP1B	7.930E-4	2.577E-4	1.32	1.000
375	1MSMV--FO-1MS95	1.250E-4	2.479E-4	2.98	1.000
376	1TMSOV-FC-20-ET	1.812E-2	2.474E-4	1.01	1.000
377	1TMSOV-FC-ASO	1.812E-2	2.474E-4	1.01	1.000
378	1EGEDG-CC-1H2H2J	9.576E-5	2.458E-4	3.57	1.000
379	1HVPCV-FC-1235C1	1.812E-2	2.454E-4	1.01	1.000
380	1SIMOV-PG-1885D	1.350E-4	2.291E-4	2.70	1.000
381	1SIMOV-PG-1885C	1.350E-4	2.291E-4	2.70	1.000
382	1SIMOV-PG-1885A	1.350E-4	2.291E-4	2.70	1.000
383	1SIMOV-PG-1885B	1.350E-4	2.291E-4	2.70	1.000
384	1EGEDG-CC-1J2H2J	9.576E-5	2.250E-4	3.35	1.000
385	1SILIC-CC-RWST	4.644E-4	2.235E-4	1.48	1.000
386	2EEBUS-LU-2H-480	1.215E-5	2.225E-4	19.32	1.000
387	2EEBUS-LU-2H1-1	1.215E-5	2.225E-4	19.32	1.000
388	2EEBUS-LU-2H	1.215E-5	2.225E-4	19.32	1.000
389	2EEBUS-LU-2H1	1.215E-5	2.225E-4	19.32	1.000
390	HEP-1ECA3:3-35	4.924E-3	2.157E-4	1.04	1.000
391	1FWFCV-CC-798999	1.812E-3	2.120E-4	1.12	1.000
392	1HVACU-UM-1HVAC7	1.654E-3	2.074E-4	1.13	1.000
393	1EPBUS-UM-1E	2.000E-4	2.043E-4	2.02	1.000
394	1EEBKR-SO-14H1-7	3.356E-5	2.017E-4	7.01	1.000
395	1EEBKR-SO-14H1-1	3.356E-5	2.017E-4	7.01	1.000
396	1CCMOV-CC-100AB	3.903E-4	1.989E-4	1.51	1.000
397	1RHMOV-CC-1720	3.903E-4	1.989E-4	1.51	1.000
398	HEP-1E0-14	1.000E+0	1.952E-4	1.00	1.000
399	2HVPAT-FS-HVP22B	1.983E-3	1.951E-4	1.10	1.000
400	2HVPAT-FS-HVP20B	1.983E-3	1.951E-4	1.10	1.000
401	1CHPAT-PT-14:2	6.999E-4	1.947E-4	1.28	1.000
402	1SIMOV-FC-1890B	1.090E-2	1.899E-4	1.02	1.000
403	1SIMOV-FO-1864B	1.090E-2	1.899E-4	1.02	1.000
404	1SIMOV-FO-1864A	1.090E-2	1.882E-4	1.02	1.000
405	1SIMOV-FC-1890A	1.090E-2	1.882E-4	1.02	1.000
406	1CHPAT-UM-1CHPBC	7.529E-4	1.848E-4	1.25	1.000
407	2EEBUS-LU-2H1-4	1.215E-5	1.757E-4	15.46	1.000
408	HEP-1OP21:6	1.050E-3	1.703E-4	1.16	1.000
409	1QSHEP-FLANGE	3.750E-4	1.700E-4	1.45	1.000
410	2EGEDG-FS-2J	1.434E-2	1.638E-4	1.01	1.000

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
411	IE-T4	6.001E-7	1.573E-4	263.09	1.000
412	1QSPSB-CC-P1A-1B	3.933E-4	1.568E-4	1.40	1.000
413	1SILMS-LF-1860A	1.250E-4	1.566E-4	2.25	1.000
414	1QSMOV-CC-101A-B	3.903E-4	1.556E-4	1.40	1.000
415	HEP-1AP33:1	3.866E-1	1.552E-4	1.00	1.000
416	1HVFAN-FR-1FMO6	1.357E-4	1.550E-4	2.14	1.000
417	1RHPSB-FR-1RHP1B	7.927E-4	1.546E-4	1.19	1.000
418	1RHPSB-FR-1RHP1A	7.927E-4	1.546E-4	1.19	1.000
419	1SILMS-LF-1860B	1.250E-4	1.538E-4	2.23	1.000
420	1EEBKR-SO-14H4	3.356E-5	1.537E-4	5.58	1.000
421	2EGEDG-FR-2J	1.330E-2	1.517E-4	1.01	1.000
422	2HVCHU-FR-2HVE4B	1.506E-3	1.480E-4	1.10	1.000
423	1CHCKV-FC-1CH267	6.339E-4	1.472E-4	1.23	1.000
424	2HVACU-LF-2HVAC7	3.425E-5	1.450E-4	5.24	1.000
425	1SIMOV-FO-1885C	1.090E-2	1.450E-4	1.01	1.000
426	1SIMOV-FO-1885A	1.090E-2	1.450E-4	1.01	1.000
427	1SIMOV-FO-1885B	1.090E-2	1.441E-4	1.01	1.000
428	1SIMOV-FO-1885D	1.090E-2	1.441E-4	1.01	1.000
429	2HVSV--SO-2205A	9.333E-5	1.403E-4	2.50	1.000
430	2HVSV--SO-2202A	9.333E-5	1.403E-4	2.50	1.000
431	1CCMV--PG-1CC199	4.105E-4	1.349E-4	1.33	1.000
432	1CCMV--PG-1CC194	4.105E-4	1.349E-4	1.33	1.000
433	HEP-1E0-22	1.880E-2	1.340E-4	1.01	1.000
434	1HVMOV-FC-113C	1.090E-2	1.322E-4	1.01	1.000
435	1HVMOV-FC-111C	1.090E-2	1.322E-4	1.01	1.000
436	1SWPAT-FR-1SWP1B	7.930E-4	1.291E-4	1.16	1.000
437	1RCPAT-FR-1RCP1A	7.930E-4	1.290E-4	1.16	1.000
438	1FWCKV-FC-1FW100	6.339E-4	1.240E-4	1.20	1.000
439	1RHCKV-FC-1RH15	6.339E-4	1.223E-4	1.19	1.000
440	1RHCKV-FC-1RH7	6.339E-4	1.223E-4	1.19	1.000
441	1FWCKV-FC-1FW132	6.339E-4	1.185E-4	1.19	1.000
442	C-Y04	9.850E-1	1.104E-4	1.00	1.000
443	1EETFM-LP-1H1	1.899E-5	1.091E-4	6.74	1.000
444	1EEBUS-UM-VB-III	2.000E-4	1.089E-4	1.54	1.000
445	2EGEDG-FS-2H	1.434E-2	1.057E-4	1.01	1.000
446	1HVSV--SO-1200	9.333E-5	1.035E-4	2.11	1.000
447	1SWCKV-FC-1SW10	6.339E-4	1.032E-4	1.16	1.000
448	2EGEDG-FR-2H	1.330E-2	9.813E-5	1.01	1.000
449	1RCMOV-FC-1535	1.090E-2	9.288E-5	1.01	1.000
450	1EEBUS-LU-1J1-2	1.215E-5	9.022E-5	8.43	1.000
451	HEP-1AP15-1E	7.799E-4	8.620E-5	1.11	1.000

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
452	1MSTCV-FC-1408B	1.812E-2	8.251E-5	1.00	1.000
453	1MSTCV-FC-1408A	1.812E-2	8.251E-5	1.00	1.000
454	1SIMOV-FC-1867D	1.090E-2	8.154E-5	1.01	1.000
455	1SIMOV-FC-1867B	1.090E-2	8.154E-5	1.01	1.000
456	2HVMOV-CC-HV213	3.903E-4	7.886E-5	1.20	1.000
457	2HVMOV-CC-HV211	3.903E-4	7.886E-5	1.20	1.000
458	REC-1ES1:2	2.660E-3	7.839E-5	1.03	1.000
459	1RHMV--PG-1RH9	4.105E-4	7.680E-5	1.19	1.000
460	1RHMV--PG-1RH16	4.105E-4	7.680E-5	1.19	1.000
461	1RHMV--PG-1RH1	4.105E-4	7.680E-5	1.19	1.000
462	1RHMV--PG-1RH8	4.105E-4	7.680E-5	1.19	1.000
463	1EEBUS-UM-1J1-2	1.000E-5	7.666E-5	8.67	1.000
464	1QSPSB-FS-1QSP1B	3.933E-3	7.146E-5	1.02	1.000
465	1QSPSB-FS-1QSP1A	3.933E-3	6.974E-5	1.02	1.000
466	1EEBUS-LU-1H1-2S	1.215E-5	6.940E-5	6.71	1.000
467	1EEBKR-SO-14H3	3.356E-5	6.837E-5	3.04	1.000
468	1EEBUS-UM-VB-I	2.000E-4	6.721E-5	1.34	1.000
469	IE-T6	6.270E-6	6.611E-5	11.54	1.000
470	1RHCKV-FO-1RH7	3.442E-3	6.580E-5	1.02	1.000
471	1RHCKV-FO-1RH15	3.442E-3	6.580E-5	1.02	1.000
472	2HVPAT-FR-HVP22B	7.930E-4	6.482E-5	1.08	1.000
473	2HVPAT-FR-HVP20B	7.930E-4	6.482E-5	1.08	1.000
474	1CCAOV-UM-TV103A	2.000E-4	6.400E-5	1.32	1.000
475	1RHMOV-FC-1720B	1.090E-2	6.367E-5	1.01	1.000
476	1RHMOV-FC-1720A	1.090E-2	6.367E-5	1.01	1.000
477	HEP-0AP10	5.274E-3	6.366E-5	1.01	1.000
478	1RHMV--PG-1RH19	4.105E-4	6.174E-5	1.15	1.000
479	1CCMV--PG-1CC762	4.105E-4	6.174E-5	1.15	1.000
480	1CCMV--PG-1CC785	4.105E-4	6.174E-5	1.15	1.000
481	1RHMV--PG-1RH30	4.105E-4	6.174E-5	1.15	1.000
482	1RHMV--PG-1RH24	4.105E-4	6.174E-5	1.15	1.000
483	1QSPSB-TM-1QSP1A	3.750E-3	6.088E-5	1.02	1.000
484	1QSPSB-UM-1QSP1A	3.750E-3	6.088E-5	1.02	1.000
485	1RHFCV-SO-1605	1.208E-5	5.871E-5	5.86	1.000
486	1EEBUS-UM-1H1	1.000E-5	5.841E-5	6.84	1.000
487	1EEBUS-UM-1H1-2S	1.000E-5	5.841E-5	6.84	1.000
488	1QSPSB-UM-1QSP1B	3.750E-3	5.638E-5	1.01	1.000
489	1QSPSB-TM-1QSP1B	3.750E-3	5.638E-5	1.01	1.000
490	1CHHEX-LU-1CHE5A	2.090E-4	5.608E-5	1.27	1.000
491	1EEBAT-CC-I-III	1.050E-6	5.426E-5	52.68	1.000
492	1MSAOV-FC-TV101A	1.812E-2	5.327E-5	1.00	1.000

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
493	1MSAOV-FC-TV101B	1.812E-2	5.327E-5	1.00	1.000
494	REC-0OP21:6	1.687E-3	5.122E-5	1.03	1.000
495	1EEBUS-LU-1H1-1	1.215E-5	5.016E-5	5.13	1.000
496	2SWCKV-FC-2SW337	6.339E-4	4.896E-5	1.08	1.000
497	2SWCKV-FC-2SW353	6.339E-4	4.896E-5	1.08	1.000
498	2CDCKV-FC-2CD211	6.339E-4	4.896E-5	1.08	1.000
499	1CCPAT-FR-1CCP1A	7.930E-4	4.770E-5	1.06	1.000
500	2HVMOD-SC-MOD238	1.208E-5	4.657E-5	4.86	1.000
501	2HVTCV-SC-TCV267	1.208E-5	4.657E-5	4.86	1.000
502	1EPBUS-UM-1B1	2.000E-4	4.648E-5	1.23	1.000
503	1MSMV--LK-1MS179	1.000E-2	4.500E-5	1.00	1.000
504	1MSMV--LK-1MS168	1.000E-2	4.500E-5	1.00	1.000
505	HEP-0AP12-10HR	4.949E-3	4.434E-5	1.01	1.000
506	1HVSV--SO-1202A	9.333E-5	4.359E-5	1.47	1.000
507	1HVSV--SO-1205A	9.333E-5	4.359E-5	1.47	1.000
508	1FWMOV-CC-150ABC	3.903E-4	4.259E-5	1.11	1.000
509	2HVPAT-FS-HVP22C	1.983E-3	4.190E-5	1.02	1.000
510	2HVPAT-FS-HVP20C	1.983E-3	4.190E-5	1.02	1.000
511	1SIPSB-FR-1HRP1A	3.304E-5	4.142E-5	2.25	1.000
512	1SIPSB-FR-1HRP1B	3.304E-5	4.127E-5	2.25	1.000
513	2HVPAT-CC-HVP20	1.983E-4	3.983E-5	1.20	1.000
514	2HVPAT-CC-HVP22	1.983E-4	3.983E-5	1.20	1.000
515	1BDAOV-FO-TV100E	1.812E-2	3.799E-5	1.00	1.000
516	1BDSOV-FO-100J	1.812E-2	3.799E-5	1.00	1.000
517	1BDSOV-FO-100F	1.812E-2	3.799E-5	1.00	1.000
518	1BDAOV-FO-TV100J	1.812E-2	3.799E-5	1.00	1.000
519	1BDSOV-FO-100E	1.812E-2	3.799E-5	1.00	1.000
520	1BDAOV-FO-TV100F	1.812E-2	3.799E-5	1.00	1.000
521	2HVACU-UM-2HVAC6	1.654E-3	3.656E-5	1.02	1.000
522	NON-REC-B221	8.999E-4	3.506E-5	1.04	1.000
523	1RCMOV-FC-1536	1.090E-2	3.432E-5	1.00	1.000
524	1HVACU-LF-1HVAC6	3.425E-5	3.340E-5	1.98	1.000
525	2HVCHU-FR-2HVE4C	1.506E-3	3.167E-5	1.02	1.000
526	1EEBAT-LP-III	1.500E-5	3.035E-5	3.02	1.000
527	1EEBAT-LP-I	1.500E-5	3.012E-5	3.01	1.000
528	1EPBUS-UM-1A1	2.000E-4	2.962E-5	1.15	1.000
529	1QSLIC-LF-100A	4.633E-3	2.902E-5	1.01	1.000
530	1QSLIC-LF-100D	4.633E-3	2.902E-5	1.01	1.000
531	1QSLIC-LF-100B	4.633E-3	2.902E-5	1.01	1.000
532	1QSLIC-LF-100C	4.633E-3	2.902E-5	1.01	1.000
533	1SWTCV-FC-SW102C	1.812E-2	2.847E-5	1.00	1.000

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
534	1HVMOV-CC-HV111	3.903E-4	2.713E-5	1.07	1.000
535	1HVMOV-CC-HV113	3.903E-4	2.713E-5	1.07	1.000
536	1SWCKV-FC-SW-116	6.339E-4	2.362E-5	1.04	1.000
537	1SWCKV-FC-SW-114	6.339E-4	2.362E-5	1.04	1.000
538	1SICKV-FC-1SI185	6.339E-4	2.234E-5	1.04	1.000
539	HEP-0AP12-20HR	2.600E-4	2.177E-5	1.08	1.000
540	1QSLEV-TM-RWSTA	1.400E-3	2.171E-5	1.02	1.000
541	1QSLEV-TM-RWSTB	1.400E-3	2.171E-5	1.02	1.000
542	PROB-D104A	5.999E-2	2.085E-5	1.00	1.000
543	1HVPAT-FR-HVP22B	7.930E-4	2.074E-5	1.03	1.000
544	1HVPAT-FR-HVP20B	7.930E-4	2.074E-5	1.03	1.000
545	1SISV--MC-1845A	3.750E-5	2.066E-5	1.55	1.000
546	1SISV--MC-1845C	3.750E-5	2.031E-5	1.54	1.000
547	C-RC301	8.750E-1	1.905E-5	1.00	1.000
548	1CCPSB-FS-1CCP1B	3.933E-3	1.869E-5	1.00	1.000
549	1EEBUS-LU-1H1-2N	1.215E-5	1.866E-5	2.54	1.000
550	1RHMV--FC-1RH25	1.250E-4	1.804E-5	1.14	1.000
551	1CCPSB-UM-1CCP1B	3.750E-3	1.782E-5	1.00	1.000
552	1EEINV-LU-II	6.136E-4	1.772E-5	1.03	1.000
553	1CCSV--SO-RV131A	9.333E-5	1.653E-5	1.18	1.000
554	1CCSV--SO-RV131B	9.333E-5	1.653E-5	1.18	1.000
555	1RHSV--SO-1721B	9.333E-5	1.653E-5	1.18	1.000
556	1RHSV--SO-1721A	9.333E-5	1.653E-5	1.18	1.000
557	1EEBUS-UM-1H1-1	1.000E-5	1.652E-5	2.65	1.000
558	1RCMOV-FO-1536	1.090E-2	1.609E-5	1.00	1.000
559	1RCMOV-FO-1535	1.090E-2	1.609E-5	1.00	1.000
560	1MSMV--LK-1MS59	3.999E-2	1.608E-5	1.00	1.000
561	1EEBKR-SO-15J2	8.390E-6	1.576E-5	2.88	1.000
562	1EEBKR-SO-15H2	8.390E-6	1.576E-5	2.88	1.000
563	NON-REC-B235	8.999E-4	1.551E-5	1.02	1.000
564	1EPBKR-SO-L202	3.356E-5	1.512E-5	1.45	1.000
565	1SICKV-FC-1SI207	6.339E-4	1.447E-5	1.02	1.000
566	1FWCKV-FC-1FW68	6.339E-4	1.416E-5	1.02	1.000
567	1SICKV-FC-1SI206	6.339E-4	1.354E-5	1.02	1.000
568	1CCSV--SO-RV128B	9.333E-5	1.347E-5	1.14	1.000
569	1CCSV--SO-RV128A	9.333E-5	1.347E-5	1.14	1.000
570	1HVPAT-FS-HVP22C	1.983E-3	1.345E-5	1.01	1.000
571	1HVPAT-FS-HVP20C	1.983E-3	1.345E-5	1.01	1.000
572	1QSCKV-CC-V19-11	6.339E-5	1.331E-5	1.21	1.000
573	1CCCKV-FO-1CC24	3.442E-3	1.310E-5	1.00	1.000
574	1MSMV--LK-1MS21	3.999E-2	1.301E-5	1.00	1.000

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
575	1CHHEX-LU-1CHE5B	2.090E-4	1.297E-5	1.06	1.000
576	1CCTNK-LF-1CCTK1	2.664E-6	1.279E-5	5.80	1.000
577	1HVPAT-CC-HVP22	1.983E-4	1.182E-5	1.06	1.000
578	1HVPAT-CC-HVP20	1.983E-4	1.182E-5	1.06	1.000
579	1SIMOV-PG-1267B	8.207E-4	1.134E-5	1.01	1.000
580	NON-REC-B220	8.999E-4	1.117E-5	1.01	1.000
581	1HVCHU-FR-1HVE4C	1.506E-3	9.717E-6	1.01	1.000
582	C-B02	6.600E-1	9.476E-6	1.00	1.000
583	1CCHEX-LU-1CCE1A	2.090E-4	9.363E-6	1.04	1.000
584	1CCHEX-LF-1CCE1B	9.477E-3	9.363E-6	1.00	1.000
585	1EEBKR-SO-VB1-35	3.356E-5	9.108E-6	1.27	1.000
586	1EEBKR-SO-VB3-35	3.356E-5	9.108E-6	1.27	1.000
587	1HVTCV-SC-TCV166	1.208E-5	8.854E-6	1.73	1.000
588	1HVMOD-SC-MOD137	1.208E-5	8.854E-6	1.73	1.000
589	1EEBUS-UM-VB-II	2.000E-4	8.503E-6	1.04	1.000
590	1FWCKV-CC-9395	6.339E-5	8.267E-6	1.13	1.000
591	1FWCKV-CC-125127	6.339E-5	8.267E-6	1.13	1.000
592	1FWMV--FO-1FW128	1.250E-4	7.779E-6	1.06	1.000
593	1EEHS--LF-I	2.664E-5	7.229E-6	1.27	1.000
594	1EEHS--LF-III	2.664E-5	7.229E-6	1.27	1.000
595	1EPBUS-UM-1	2.000E-4	6.983E-6	1.03	1.000
596	1EPBUS-UM-1F	2.000E-4	6.983E-6	1.03	1.000
597	1EPBUS-UM-3	2.000E-4	6.983E-6	1.03	1.000
598	1EEBUS-UM-DC-II	2.000E-4	5.775E-6	1.03	1.000
599	1CHPAT-FS-1CHP1C	5.078E-3	5.701E-6	1.00	1.000
600	1EPBUS-UM-1B3	2.000E-4	5.428E-6	1.03	1.000
601	1EPBKR-SO-15E3	3.356E-5	5.165E-6	1.15	1.000
602	1EPBKR-SO-15E1	3.356E-5	5.165E-6	1.15	1.000
603	1EPBKR-SO-242	3.356E-5	5.165E-6	1.15	1.000
604	2EEBKR-SO-25H11	3.356E-5	5.165E-6	1.15	1.000
605	1MSMV--FO-1MS97	1.250E-4	4.680E-6	1.04	1.000
606	1MSPORV-DMDT7	1.000E+0	4.680E-6	1.00	1.000
607	1MSRV--FO-101C	2.500E-2	4.680E-6	1.00	1.000
608	1EEBKR-SO-15H12	3.356E-5	4.655E-6	1.14	1.000
609	1EPBKR-SO-15J12	3.356E-5	4.655E-6	1.14	1.000
610	1QSHEP-1QS21	7.499E-4	4.557E-6	1.01	1.000
611	1QSHEP-1QS5	7.499E-4	4.522E-6	1.01	1.000
612	1EPTFM-LP-2	1.899E-5	4.521E-6	1.24	1.000
613	1EPBKR-FC-15F1	1.834E-3	3.789E-6	1.00	1.000
614	2HVPAT-FR-HVP22C	7.930E-4	3.571E-6	1.00	1.000
615	2HVPAT-FR-HVP20C	7.930E-4	3.571E-6	1.00	1.000

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
616	1QSCKV-FC-1QS-11	6.339E-4	3.441E-6	1.01	1.000
617	1CHCKV-CC-267279	6.339E-5	3.387E-6	1.05	1.000
618	1QSCKV-FC-1QS-19	6.339E-4	3.268E-6	1.01	1.000
619	1EEBUS-LU-VB-I	1.215E-5	3.223E-6	1.27	1.000
620	1EEBUS-LU-VB-III	1.215E-5	3.223E-6	1.27	1.000
621	1MSMV--PG-1MS268	1.368E-4	3.127E-6	1.02	1.000
622	1MSMV--PG-1MS270	1.368E-4	3.127E-6	1.02	1.000
623	1MSMV--PG-1MS269	1.368E-4	3.127E-6	1.02	1.000
624	1MSMV--PG-1MS271	1.368E-4	3.127E-6	1.02	1.000
625	HEP-1E0-15	1.075E-3	2.979E-6	1.00	1.000
626	1CCPSB-FR-1CCP1B	7.927E-4	2.971E-6	1.00	1.000
627	1EPBKR-FC-15A2	1.834E-3	2.760E-6	1.00	1.000
628	HEP-0AP12-40HR	1.250E-1	2.721E-6	1.00	1.000
629	1FWHCV-FO-100C	1.812E-2	2.657E-6	1.00	1.000
630	1SWMOV-FC-SW101B	1.090E-2	2.412E-6	1.00	1.000
631	1SWMOV-FC-SW101D	1.090E-2	2.412E-6	1.00	1.000
632	1SWMOV-FC-SW101A	1.090E-2	2.412E-6	1.00	1.000
633	1SWMOV-FC-SW101C	1.090E-2	2.412E-6	1.00	1.000
634	1CCCKV-FC-1CC47	6.339E-4	2.376E-6	1.00	1.000
635	1EPBUS-LU-1A	1.215E-5	1.944E-6	1.16	1.000
636	1HVPAT-FR-HVP20C	7.930E-4	1.904E-6	1.00	1.000
637	1HVPAT-FR-HVP22C	7.930E-4	1.904E-6	1.00	1.000
638	1EPBUS-LU-1B1	1.215E-5	1.612E-6	1.13	1.000
639	1EEBUS-LU-1JSTUB	1.215E-5	1.612E-6	1.13	1.000
640	1EEBUS-LU-1HSTUB	1.215E-5	1.612E-6	1.13	1.000
641	1EPBUS-LU-1A1	1.215E-5	1.612E-6	1.13	1.000
642	1EPBKR-FC-15B2	1.834E-3	1.587E-6	1.00	1.000
643	1EPBUS-LU-4	1.215E-5	1.490E-6	1.12	1.000
644	1EPBUS-LU-2	1.215E-5	1.490E-6	1.12	1.000
645	1RCMOV-CC-535536	3.903E-4	1.443E-6	1.00	1.000
646	2EGEDG-TM-2J	5.708E-4	1.179E-6	1.00	1.000
647	1EPBKR-SO-332	3.356E-5	1.172E-6	1.03	1.000
648	1EEBKR-SO-15H11	3.356E-5	1.172E-6	1.03	1.000
649	1EPBKR-SO-15F3	3.356E-5	1.172E-6	1.03	1.000
650	1EPBKR-SO-15F1	3.356E-5	1.172E-6	1.03	1.000
651	1EPBKR-SO-L102	3.356E-5	1.172E-6	1.03	1.000
652	1EEBUS-UM-1JSTUB	1.000E-5	1.042E-6	1.10	1.000
653	1EEBUS-UM-1HSTUB	1.000E-5	1.042E-6	1.10	1.000
654	1SWCKV-CC-647648	6.339E-5	1.034E-6	1.02	1.000
655	1RSSTR-PG-TEMPB	2.822E-2	1.012E-6	1.00	1.000
656	1FWHCV-FC-100B	1.812E-2	9.925E-7	1.00	1.000

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
657	1SITNK-LF-1SITK2	2.664E-6	8.886E-7	1.33	1.000
658	1EEBCH-LP-I	8.399E-5	7.986E-7	1.01	1.000
659	1EEBCH-LP-III	8.399E-5	7.986E-7	1.01	1.000
660	1FWCKV-CC-477911	6.339E-5	7.413E-7	1.01	1.000
661	1RSSTR-PG-TEMPA	2.822E-2	6.727E-7	1.00	1.000
662	1EPTFM-LP-RSST-C	1.899E-5	6.631E-7	1.03	1.000
663	1EPTFM-LP-1	1.899E-5	6.631E-7	1.03	1.000
664	1SWMOV-SC-SW208A	1.208E-5	6.561E-7	1.05	1.000
665	1SWMOV-SC-SW108A	1.208E-5	6.561E-7	1.05	1.000
666	1SWMOV-SC-SW208B	1.208E-5	6.561E-7	1.05	1.000
667	1SWMOV-SC-SW108B	1.208E-5	6.561E-7	1.05	1.000
668	1FWMOV-FC-100C	1.090E-2	5.972E-7	1.00	1.000
669	1CHPAT-FR-24HP1C	7.930E-4	5.877E-7	1.00	1.000
670	2EEBKR-FO-25J11	2.735E-4	5.649E-7	1.00	1.000
671	1EEBKR-FO-15H11	2.735E-4	5.649E-7	1.00	1.000
672	1EPBKR-FO-15F4	2.735E-4	5.649E-7	1.00	1.000
673	1EPBKR-FO-15F3	2.735E-4	5.649E-7	1.00	1.000
674	2EEBKR-FO-25J2	2.735E-4	5.649E-7	1.00	1.000
675	2EGEDG-CC-2H-2J	2.663E-4	5.500E-7	1.00	1.000
676	1FWHEP-MOV-100B	7.499E-4	5.379E-7	1.00	1.000
677	1CHPAT-PT-14:3	6.999E-4	5.187E-7	1.00	1.000
678	1SWMOV-FC-SW103B	1.090E-2	5.062E-7	1.00	1.000
679	1SWMOV-FC-SW104B	1.090E-2	5.062E-7	1.00	1.000
680	1EEBKR-SO-II-14	3.356E-5	4.922E-7	1.01	1.000
681	1EEBKR-SO-VB2-35	3.356E-5	4.922E-7	1.01	1.000
682	1EPBKR-FC-G12	1.834E-3	4.896E-7	1.00	1.000
683	1FWCKV-FC-1FW93	6.339E-4	4.547E-7	1.00	1.000
684	NON-REC-B229	8.999E-4	4.489E-7	1.00	1.000
685	1FWHEP-MOV-100D	7.499E-4	3.866E-7	1.00	1.000
686	1CHMOV-FO-1286A	1.090E-2	3.654E-7	1.00	1.000
687	1SWMOV-FC-SW103A	1.090E-2	3.364E-7	1.00	1.000
688	1SWMOV-FC-SW104A	1.090E-2	3.364E-7	1.00	1.000
689	1CHCKV-FC-1CH279	6.339E-4	3.356E-7	1.00	1.000
690	1FWCKV-FC-1FW279	6.339E-4	3.268E-7	1.00	1.000
691	1FWHEP-HCV-100C	7.499E-4	3.236E-7	1.00	1.000
692	1FWCKV-FC-1FW127	6.339E-4	2.736E-7	1.00	1.000
693	1EEBKR-SO-III-11	3.356E-5	2.575E-7	1.01	1.000
694	1EEBKR-SO-J1-B1L	3.356E-5	2.575E-7	1.01	1.000
695	1EEBKR-SO-I-11	3.356E-5	2.575E-7	1.01	1.000
696	1EEBKR-SO-H4-D2L	3.356E-5	2.575E-7	1.01	1.000
697	1MSMV--PG-1MS179	9.123E-5	2.406E-7	1.00	1.000

TABLE 3.4.1-6 (Continued)
BASIC EVENTS RANKED BY THE
FUSSELL-VESELY IMPORTANCE MEASURES

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achievement Worth	Risk Reduction Worth
698	1MSMV--PG-1MS168	9.123E-5	2.406E-7	1.00	1.000
699	1EPBKR-FO-15A1	2.735E-4	2.099E-7	1.00	1.000
700	HEP-0AP12-30HR	6.565E-3	1.723E-7	1.00	1.000
701	1EPBKR-SO-15B10	3.356E-5	1.693E-7	1.01	1.000
702	1EPBKR-SO-14B3-1	3.356E-5	1.693E-7	1.01	1.000
703	1EPBUS-LU-1	1.215E-5	1.662E-7	1.01	1.000
704	1EPBUS-LU-1F	1.215E-5	1.662E-7	1.01	1.000
705	1EPBUS-LU-3	1.215E-5	1.662E-7	1.01	1.000
706	2EEBUS-UM-2J	2.000E-4	1.619E-7	1.00	1.000
707	1FWBKR-FC-15C5	1.834E-3	1.546E-7	1.00	1.000
708	1FWBKR-FC-15A5	1.834E-3	1.546E-7	1.00	1.000
709	1FWBKR-FC-15A6	1.834E-3	1.546E-7	1.00	1.000
710	1FWBKR-FC-15B5	1.834E-3	1.546E-7	1.00	1.000

TABLE 3.4.1-7
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	14814.19	1.004
2	IE-VX	1.600E-6	2.358E-2	14737.41	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

TABLE 3.4.1-7 (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
43	1SIMOV-CC-1860AB	3.903E-4	1.598E-2	41.92	1.016
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

TABLE 3.4.1-7 (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
85	1SICKV-FC-1SI127	6.339E-4	5.534E-3	9.72	1.006
86	1SICKV-FC-1SI159	6.339E-4	5.534E-3	9.72	1.006
87	1SIMOV-PG-1865A	8.207E-4	7.166E-3	9.72	1.007
88	1SIMOV-PG-1865C	8.207E-4	7.166E-3	9.72	1.007
89	1SICKV-CC-ACCCKV	6.339E-5	5.510E-4	9.69	1.001
90	1EEBKR-SO-14J5	3.356E-5	2.723E-4	9.11	1.000
91	1EEBUS-UM-1J1-2	1.000E-5	7.666E-5	8.67	1.000
92	1MSRV--CC-101ABC	9.988E-4	7.573E-3	8.57	1.008
93	1SIMOV-CC-1890CD	3.903E-4	2.902E-3	8.43	1.003
94	1EEBUS-LU-1J1-2	1.215E-5	9.022E-5	8.43	1.000
95	HEP-1ES1:4	8.499E-4	6.308E-3	8.42	1.006
96	1SICKV-CC-959903	6.339E-5	4.697E-4	8.41	1.000
97	1SICKV-CC-206207	6.339E-5	4.697E-4	8.41	1.000
98	2EEBUS-UM-2H1-4	2.000E-4	1.445E-3	8.22	1.001
99	IE-S2	2.100E-2	1.479E-1	7.90	1.174
100	1EEBKR-SO-14H1-7	3.356E-5	2.017E-4	7.01	1.000
101	1EEBKR-SO-14H1-1	3.356E-5	2.017E-4	7.01	1.000
102	1EGEDG-FS-1H	1.434E-2	8.702E-2	6.98	1.095
103	1EGEDG-FR-1H	1.330E-2	8.029E-2	6.96	1.087
104	1EEBUS-UM-1H1	1.000E-5	5.841E-5	6.84	1.000
105	1EEBUS-UM-1H1-2S	1.000E-5	5.841E-5	6.84	1.000
106	1EETFM-LP-1H1	1.899E-5	1.091E-4	6.74	1.000
107	1EEBUS-LU-1H1-2S	1.215E-5	6.940E-5	6.71	1.000
108	HEP-1ES1:2-S2	8.499E-4	4.624E-3	6.44	1.005
109	1RHPSB-CC-1RHP1	3.933E-4	1.926E-3	5.90	1.002
110	1RHFEL-PG-1605	4.105E-4	2.010E-3	5.90	1.002
111	1SICKV-CC-144161	6.339E-5	3.097E-4	5.89	1.000
112	1RHCKV-CC-1RH715	6.339E-5	3.097E-4	5.89	1.000
113	1RHFCV-SO-1605	1.208E-5	5.871E-5	5.86	1.000
114	1CCTNK-LF-1CCTK1	2.664E-6	1.279E-5	5.80	1.000
115	1EEBKR-SO-14H4	3.356E-5	1.537E-4	5.58	1.000
116	2HVFAN-FR-2FMO7	1.357E-4	6.000E-4	5.42	1.001
117	2HVSV--SO-2200	9.333E-5	4.113E-4	5.41	1.000
118	2HVACU-LF-2HVAC7	3.425E-5	1.450E-4	5.24	1.000
119	1EEBUS-LU-1H1-1	1.215E-5	5.016E-5	5.13	1.000
120	NON-REC-B16	7.499E-3	3.005E-2	4.98	1.031
121	2HVMOD-SC-MOD238	1.208E-5	4.657E-5	4.86	1.000
122	2HVTCV-SC-TCV267	1.208E-5	4.657E-5	4.86	1.000
123	T9A-FREQ-4160-1H	5.999E-3	2.248E-2	4.73	1.023
124	1EGEDG-UM-1H	1.781E-2	6.081E-2	4.35	1.065
125	1EGEDG-FR-1J	1.330E-2	4.455E-2	4.30	1.047
126	1EGEDG-FS-1J	1.434E-2	4.804E-2	4.30	1.050

TABLE 3.4.1-7 (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
127	HEP-1ECA3:1-16	3.025E-3	9.604E-3	4.17	1.010
128	1EGEDG-CC-1H-2J	2.663E-4	8.302E-4	4.12	1.001
129	1SIPSB-FS-1SIP1B	4.018E-3	1.210E-2	4.00	1.012
130	1FWHEP-1FW548	7.499E-4	2.219E-3	3.96	1.002
131	1FWHEP-1FW546	7.499E-4	2.168E-3	3.89	1.002
132	1EEBKR-FO-15H2	2.735E-4	7.754E-4	3.83	1.001
133	1EGEDG-CC-1H-2H	2.663E-4	7.427E-4	3.79	1.001
134	1SIPSB-FS-1SIP1A	4.018E-3	1.122E-2	3.78	1.011
135	1FWPSB-FS-1FWP3A	1.583E-3	4.395E-3	3.77	1.004
136	1SIPSB-UM-1SIP1B	4.536E-3	1.246E-2	3.73	1.013
137	1FWPSB-FS-1FWP3B	1.583E-3	4.280E-3	3.70	1.004
138	1SICKV-FC-1SI9	6.339E-4	1.703E-3	3.69	1.002
139	1RSHEP-FLANGE	3.750E-4	9.962E-4	3.66	1.001
140	1SICKV-FC-1SI18	6.339E-4	1.681E-3	3.65	1.002
141	1SICKV-FC-1SI26	6.339E-4	1.681E-3	3.65	1.002
142	1FWPSB-FR-24HP3A	7.927E-4	2.081E-3	3.62	1.002
143	1EEBKR-FO-15J2	2.735E-4	7.151E-4	3.61	1.001
144	1EGEDG-CC-1J-2J	2.663E-4	6.922E-4	3.60	1.001
145	1FWCKV-FC-1FW165	6.339E-4	1.648E-3	3.60	1.002
146	1EGEDG-CC-1J-2H	2.663E-4	6.911E-4	3.59	1.001
147	1EGEDG-CC-1H2H2J	9.576E-5	2.458E-4	3.57	1.000
148	1SIPSB-UM-1SIP1A	4.536E-3	1.166E-2	3.56	1.012
149	1FWPSB-FR-24HP3B	7.927E-4	2.001E-3	3.52	1.002
150	1FWCKV-FC-1FW183	6.339E-4	1.580E-3	3.49	1.002
151	1FWTRB-FS-1FWP2	1.854E-2	4.678E-2	3.48	1.049
152	1SIMV--PG-1SI305	1.350E-4	3.338E-4	3.47	1.000
153	1SIMOV-PG-1862A	1.350E-4	3.338E-4	3.47	1.000
154	1RCRV--FC-1456	9.988E-3	2.474E-2	3.45	1.025
155	1SIMOV-PG-1862B	1.350E-4	3.308E-4	3.45	1.000
156	1SIMV--PG-1SI306	1.350E-4	3.308E-4	3.45	1.000
157	2HVPAT-FR-HVP22A	7.930E-4	1.901E-3	3.39	1.002
158	2HVPAT-FR-HVP20A	7.930E-4	1.901E-3	3.39	1.002
159	1EGEDG-CC-1J2H2J	9.576E-5	2.250E-4	3.35	1.000
160	1SWMOV-CC-104A-D	3.903E-4	9.136E-4	3.34	1.001
161	1SWMOV-CC-105A-D	3.903E-4	9.136E-4	3.34	1.001
162	1SWMOV-CC-103A-D	3.903E-4	9.136E-4	3.34	1.001
163	1SWMOV-CC-101A-D	3.903E-4	9.136E-4	3.34	1.001
164	HEP-1FRH:1-11	4.824E-2	1.163E-1	3.29	1.132
165	1MSRV--FC-101C	9.988E-3	2.310E-2	3.29	1.024
166	IE-T1	1.139E-1	2.923E-1	3.27	1.413
167	1EEBUS-UM-1H1-2N	2.000E-4	4.505E-4	3.25	1.000
168	1MSAOV-CC-111AB	1.812E-3	4.077E-3	3.25	1.004

TABLE 3.4.1-7 (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
169	1SWPAT-CC-SWP1B	3.842E-4	8.204E-4	3.13	1.001
170	1FWHEP-1FW543	7.499E-4	1.567E-3	3.09	1.002
171	1SIMOV-PG-1864A	8.207E-4	1.697E-3	3.07	1.002
172	1SIMOV-PG-1864B	8.207E-4	1.688E-3	3.06	1.002
173	1FWCKV-FC-1FW148	6.339E-4	1.299E-3	3.05	1.001
174	1EEBKR-SO-14H3	3.356E-5	6.837E-5	3.04	1.000
175	HEP-1E3-3	3.650E-3	7.421E-3	3.03	1.007
176	1MSCKV-FO-1MS19	3.442E-3	6.998E-3	3.03	1.007
177	1MSCKV-FO-1MS58	3.442E-3	6.998E-3	3.03	1.007
178	1EEBAT-LP-III	1.500E-5	3.035E-5	3.02	1.000
179	1EEBAT-LP-I	1.500E-5	3.012E-5	3.01	1.000
180	1MSMV--FO-1MS95	1.250E-4	2.479E-4	2.98	1.000
181	1SIMOV-FO-1862B	1.090E-2	2.153E-2	2.95	1.022
182	1SIMOV-FC-1860B	1.090E-2	2.153E-2	2.95	1.022
183	1EEBKR-SO-15J2	8.390E-6	1.576E-5	2.88	1.000
184	1EEBKR-SO-15H2	8.390E-6	1.576E-5	2.88	1.000
185	1EGEDG-UM-1J	1.781E-2	3.391E-2	2.87	1.035
186	1SICKV-FC-1SI12	6.339E-4	1.134E-3	2.79	1.001
187	1SICKV-FC-1SI29	6.339E-4	1.129E-3	2.78	1.001
188	1FWPSB-UM-1FWP3A	5.183E-3	9.273E-3	2.78	1.009
189	HEP-1E3-13	2.180E-2	3.881E-2	2.74	1.040
190	1FWPSB-UM-1FWP3B	5.183E-3	9.027E-3	2.73	1.009
191	1SIMOV-PG-1885A	1.350E-4	2.291E-4	2.70	1.000
192	1SIMOV-PG-1885D	1.350E-4	2.291E-4	2.70	1.000
193	1SIMOV-PG-1885C	1.350E-4	2.291E-4	2.70	1.000
194	1SIMOV-PG-1885B	1.350E-4	2.291E-4	2.70	1.000
195	1RCPORV-T3	6.651E-3	1.133E-2	2.69	1.011
196	HEP-1ECA3:2-5	7.249E-4	1.219E-3	2.68	1.001
197	1EEBUS-UM-1H1-1	1.000E-5	1.652E-5	2.65	1.000
198	1FWTRB-UM-1FWP2	1.366E-2	2.267E-2	2.64	1.023
199	1SIMOV-PG-1860B	1.357E-3	2.221E-3	2.63	1.002
200	1SIMOV-FO-1862A	1.090E-2	1.771E-2	2.61	1.018
201	1SIMOV-FC-1860A	1.090E-2	1.771E-2	2.61	1.018
202	1SIPSB-FR-24HP1B	7.927E-4	1.243E-3	2.57	1.001
203	1SIMOV-PG-1860A	1.357E-3	2.117E-3	2.56	1.002
204	1EGEDG-TM-1H	5.708E-4	8.783E-4	2.54	1.001
205	1EEBUS-LU-1H1-2N	1.215E-5	1.866E-5	2.54	1.000
206	1SIPSB-FR-24HP1A	7.927E-4	1.203E-3	2.52	1.001
207	2HVSV--SO-2205A	9.333E-5	1.403E-4	2.50	1.000
208	2HVSV--SO-2202A	9.333E-5	1.403E-4	2.50	1.000
209	1SICKV-FC-1SI1	6.339E-4	9.476E-4	2.49	1.001
210	1SICKV-FC-1SI16	6.339E-4	9.261E-4	2.46	1.001

TABLE 3.4.1-7 (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
211	1EPBUS-UM-4	2.000E-4	2.886E-4	2.44	1.000
212	1EPBUS-UM-2	2.000E-4	2.886E-4	2.44	1.000
213	1EGEDG-TM-1J	5.708E-4	8.046E-4	2.41	1.001
214	1SICKV-FC-1SI144	6.339E-4	8.387E-4	2.32	1.001
215	1SIMOV-PG-1865B	8.207E-4	1.080E-3	2.31	1.001
216	1SICKV-FC-1SI142	6.339E-4	8.334E-4	2.31	1.001
217	1SIPSB-FR-1HRP1A	3.304E-5	4.142E-5	2.25	1.000
218	1SILMS-LF-1860A	1.250E-4	1.566E-4	2.25	1.000
219	1SIPSB-FR-1HRP1B	3.304E-5	4.127E-5	2.25	1.000
220	HEP-1FRC:1-11-S2	1.062E-2	1.332E-2	2.24	1.013
221	1SILMS-LF-1860B	1.250E-4	1.538E-4	2.23	1.000
222	HEP-1E0-7	1.350E-3	1.645E-3	2.22	1.002
223	1FWTRB-FR-12HP2	5.742E-2	7.282E-2	2.20	1.079
224	2HVCHU-FR-2HVE4A	1.506E-3	1.778E-3	2.18	1.002
225	1HVFAN-FR-1FMO6	1.357E-4	1.550E-4	2.14	1.000
226	1HVSV--SO-1200	9.333E-5	1.035E-4	2.11	1.000
227	1SIMOV-FO-1115E	1.090E-2	1.134E-2	2.03	1.011
228	1SIMOV-FC-1115B	1.090E-2	1.134E-2	2.03	1.011
229	1EPBUS-UM-1E	2.000E-4	2.043E-4	2.02	1.000
230	HEP-1FRH:1-15	8.249E-3	8.273E-3	1.99	1.008
231	1HVACU-LF-1HVAC6	3.425E-5	3.340E-5	1.98	1.000
232	2HVSTR-PL-2HVS1A	6.390E-4	6.079E-4	1.95	1.001
233	REC-1FRH:1-4	1.131E-2	1.065E-2	1.93	1.011
234	1HVCHU-FR-1HVE4A	1.506E-3	1.390E-3	1.92	1.001
235	1SWTCV-FC-SW102B	1.812E-2	1.589E-2	1.86	1.016
236	1HVPAT-FR-HVP20A	7.930E-4	6.749E-4	1.85	1.001
237	1HVPAT-FR-HVP22A	7.930E-4	6.749E-4	1.85	1.001
238	1SICKV-FC-1SI79	6.339E-4	5.352E-4	1.84	1.001
239	1SICKV-FO-1SI47	3.442E-3	2.817E-3	1.82	1.003
240	1SIMOV-CC-1867AB	3.903E-4	3.079E-4	1.79	1.000
241	1SIMOV-CC-1867CD	3.903E-4	3.079E-4	1.79	1.000
242	C-FM01	4.800E-2	3.866E-2	1.77	1.040
243	HEP-1ES1:3	1.220E-2	9.378E-3	1.76	1.009
244	1HVTCV-SC-TCV166	1.208E-5	8.854E-6	1.73	1.000
245	1HVMOD-SC-MOD137	1.208E-5	8.854E-6	1.73	1.000
246	1HVSTR-PL-1HVS1A	6.390E-4	4.459E-4	1.70	1.000
247	1SIMOV-FC-1863B	1.090E-2	7.688E-3	1.70	1.008
248	1CHPAT-FS-1CHP1B	5.078E-3	3.306E-3	1.65	1.003
249	1HVCHU-FS-1HVE4B	4.545E-2	3.016E-2	1.63	1.031
250	1RCRV--FO-1456	2.500E-2	1.548E-2	1.60	1.016
251	1RCRV--FO-1455C	2.500E-2	1.514E-2	1.59	1.015
252	T9B-FREQ-4160-1J	5.999E-3	3.464E-3	1.57	1.003

TABLE 3.4.1-7 (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
253	1SIMOV-FC-1115D	1.090E-2	6.306E-3	1.57	1.006
254	1SIMOV-FO-1115C	1.090E-2	6.306E-3	1.57	1.006
255	1HVPCV-FC-1235B1	1.812E-2	1.051E-2	1.57	1.011
256	1HVTCV-FC-TCV167	1.812E-2	1.040E-2	1.56	1.011
257	1MSMV--LK-1MS97	3.999E-2	2.305E-2	1.55	1.024
258	1SISV--MC-1845A	3.750E-5	2.066E-5	1.55	1.000
259	1EEBUS-UM-VB-III	2.000E-4	1.089E-4	1.54	1.000
260	1SISV--MC-1845C	3.750E-5	2.031E-5	1.54	1.000
261	1CCMOV-CC-100AB	3.903E-4	1.989E-4	1.51	1.000
262	1RHMOV-CC-1720	3.903E-4	1.989E-4	1.51	1.000
263	1RHMOV-FC-1701	1.090E-2	5.570E-3	1.51	1.006
264	1RHMOV-FC-1700	1.090E-2	5.570E-3	1.51	1.006
265	1RHHCV-FC-1758	1.812E-2	9.258E-3	1.50	1.009
266	1SWTCV-CC-102BC	1.812E-3	8.842E-4	1.49	1.001
267	1SILIC-CC-RWST	4.644E-4	2.235E-4	1.48	1.000
268	1HVSV--SO-1202A	9.333E-5	4.359E-5	1.47	1.000
269	1HVSV--SO-1205A	9.333E-5	4.359E-5	1.47	1.000
270	1QSHEP-FLANGE	3.750E-4	1.700E-4	1.45	1.000
271	1EPBKR-SO-L202	3.356E-5	1.512E-5	1.45	1.000
272	1HVMOV-FC-111B	1.090E-2	4.928E-3	1.45	1.005
273	1HVMOV-FC-113B	1.090E-2	4.928E-3	1.45	1.005
274	1HVMOD-FO-MOD137	1.090E-2	4.870E-3	1.44	1.005
275	1HVMOD-FC-MOD138	1.090E-2	4.870E-3	1.44	1.005
276	1HVCHU-UM-1HVE4B	9.440E-2	4.579E-2	1.44	1.048
277	1CHPAT-FS-1CHP1A	1.983E-3	8.171E-4	1.41	1.001
278	HEP-1OP14:1-5:13	4.259E-3	1.728E-3	1.40	1.002
279	1QSMOV-CC-101A-B	3.903E-4	1.556E-4	1.40	1.000
280	1QSPSB-CC-P1A-1B	3.933E-4	1.568E-4	1.40	1.000
281	REC-SCREEN-TURNS	1.000E-1	4.348E-2	1.39	1.045
282	REC-1AP28	1.017E-1	4.257E-2	1.38	1.044
283	1HVCHU-CC-HVE4	4.547E-3	1.684E-3	1.37	1.002
284	1CHPAT-FR-24HP1A	7.930E-4	2.683E-4	1.34	1.000
285	1EEBUS-UM-VB-I	2.000E-4	6.721E-5	1.34	1.000
286	1SITNK-LF-1SITK2	2.664E-6	8.886E-7	1.33	1.000
287	1SIMOV-FC-1863A	1.090E-2	3.673E-3	1.33	1.004
288	2HVCHU-CC-HVE4	4.547E-3	1.515E-3	1.33	1.002
289	1FWTRB-FR-24HP2	1.115E-1	4.127E-2	1.33	1.043
290	1CCMV--PG-1CC199	4.105E-4	1.349E-4	1.33	1.000
291	1CCMV--PG-1CC194	4.105E-4	1.349E-4	1.33	1.000
292	1CHPAT-FR-24HP1B	7.930E-4	2.577E-4	1.32	1.000
293	1CCAOV-UM-TV103A	2.000E-4	6.400E-5	1.32	1.000
294	2HVSTR-PG-2HVS1B	9.528E-3	2.873E-3	1.30	1.003

TABLE 3.4.1-7 (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
295	2HVPCV-CC-2235	1.812E-3	5.396E-4	1.30	1.001
296	IE-T2	5.000E-2	1.515E-2	1.29	1.015
297	1FWFCV-CC-788898	1.812E-3	5.205E-4	1.29	1.001
298	1CHPAT-PT-14:2	6.999E-4	1.947E-4	1.28	1.000
299	1EEHS--LF-III	2.664E-5	7.229E-6	1.27	1.000
300	1EEHS--LF-I	2.664E-5	7.229E-6	1.27	1.000
301	1EEBKR-SO-VB1-35	3.356E-5	9.108E-6	1.27	1.000
302	1EEBKR-SO-VB3-35	3.356E-5	9.108E-6	1.27	1.000
303	1CHHEX-LU-1CHE5A	2.090E-4	5.608E-5	1.27	1.000
304	IE-T5A	5.999E-3	1.613E-3	1.27	1.002
305	1EEBUS-LU-VB-III	1.215E-5	3.223E-6	1.27	1.000
306	1EEBUS-LU-VB-I	1.215E-5	3.223E-6	1.27	1.000
307	1HVFAN-FS-1FMO7	3.933E-3	1.040E-3	1.26	1.001
308	IE-T5B	5.999E-3	1.589E-3	1.26	1.002
309	1HVPCV-CC-1235	1.812E-3	4.769E-4	1.26	1.000
310	1RCRV--FC-1455C	9.988E-3	2.569E-3	1.25	1.003
311	1CHPAT-UM-1CHPBC	7.529E-4	1.848E-4	1.25	1.000
312	1SWPAT-UM-1SWP1B	3.750E-3	9.185E-4	1.24	1.001
313	1SWPAT-FS-1SWP1B	3.842E-3	9.411E-4	1.24	1.001
314	1EPTFM-LP-2	1.899E-5	4.521E-6	1.24	1.000
315	1EPBUS-UM-1B1	2.000E-4	4.648E-5	1.23	1.000
316	1CHCKV-FC-1CH267	6.339E-4	1.472E-4	1.23	1.000
317	2HVCHU-UM-HVE4BC	2.259E-3	5.044E-4	1.22	1.001
318	1SIMOV-FC-1867A	1.090E-2	2.411E-3	1.22	1.002
319	1SIMOV-FC-1867C	1.090E-2	2.411E-3	1.22	1.002
320	1RHPSB-FS-1RHP1B	3.933E-3	8.524E-4	1.22	1.001
321	1RHPSB-FS-1RHP1A	3.933E-3	8.487E-4	1.21	1.001
322	1HVCHU-UM-HVE4BC	2.259E-3	4.818E-4	1.21	1.000
323	1QSCKV-CC-V19-11	6.339E-5	1.331E-5	1.21	1.000
324	REC-B12AVE	1.056E-1	2.441E-2	1.21	1.025
325	2HVMOV-CC-HV213	3.903E-4	7.886E-5	1.20	1.000
326	2HVMOV-CC-HV211	3.903E-4	7.886E-5	1.20	1.000
327	2HVPAT-CC-HVP20	1.983E-4	3.983E-5	1.20	1.000
328	2HVPAT-CC-HVP22	1.983E-4	3.983E-5	1.20	1.000
329	REC-1OP14:1	1.043E-1	2.288E-2	1.20	1.023
330	1RHHEX-LF-1RHE2B	2.807E-2	5.675E-3	1.20	1.006
331	1HVPAT-FS-HVP20B	1.983E-3	3.884E-4	1.20	1.000
332	1HVPAT-FS-HVP22B	1.983E-3	3.884E-4	1.20	1.000
333	1FWCKV-FC-1FW100	6.339E-4	1.240E-4	1.20	1.000
334	1RHHEX-LF-1RHE2A	2.807E-2	5.640E-3	1.20	1.006
335	1RHPSB-FR-1RHP1A	7.927E-4	1.546E-4	1.19	1.000
336	1RHPSB-FR-1RHP1B	7.927E-4	1.546E-4	1.19	1.000

TABLE 3.4.1-7 (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
337	1RHCKV-FC-1RH7	6.339E-4	1.223E-4	1.19	1.000
338	1RHCKV-FC-1RH15	6.339E-4	1.223E-4	1.19	1.000
339	1RHMV--PG-1RH8	4.105E-4	7.680E-5	1.19	1.000
340	1RHMV--PG-1RH9	4.105E-4	7.680E-5	1.19	1.000
341	1RHMV--PG-1RH1	4.105E-4	7.680E-5	1.19	1.000
342	1RHMV--PG-1RH16	4.105E-4	7.680E-5	1.19	1.000
343	1FWCKV-FC-1FW132	6.339E-4	1.185E-4	1.19	1.000
344	1SWMOV-FC-1SW117	1.090E-2	2.030E-3	1.18	1.002
345	1SWPSB-FS-1SWP-4	3.152E-3	5.783E-4	1.18	1.001
346	1RHPSB-UM-1RHP1B	3.750E-3	6.745E-4	1.18	1.001
347	1RHPSB-UM-1RHP1A	3.750E-3	6.710E-4	1.18	1.001
348	1RHSV--SO-1721A	9.333E-5	1.653E-5	1.18	1.000
349	1CCSV--SO-RV131A	9.333E-5	1.653E-5	1.18	1.000
350	1RHSV--SO-1721B	9.333E-5	1.653E-5	1.18	1.000
351	1CCSV--SO-RV131B	9.333E-5	1.653E-5	1.18	1.000
352	1HVCHU-FR-1HVE4B	1.506E-3	2.655E-4	1.18	1.000
353	1SWPSB-UM-1SWP-4	8.290E-2	1.560E-2	1.17	1.016
354	HEP-1OP49:1	1.326E-1	2.497E-2	1.16	1.026
355	1MSTCV-CC-1408AB	1.812E-3	2.958E-4	1.16	1.000
356	1SWCKV-FC-1SW10	6.339E-4	1.032E-4	1.16	1.000
357	1SWPAT-FR-1SWP1B	7.930E-4	1.291E-4	1.16	1.000
358	1RCPAT-FR-1RCP1A	7.930E-4	1.290E-4	1.16	1.000
359	1RCPCV-CC-1455AB	1.812E-3	2.950E-4	1.16	1.000
360	HEP-1OP21:6	1.050E-3	1.703E-4	1.16	1.000
361	1RCPCV-FC-1455A	1.812E-2	2.968E-3	1.16	1.003
362	1EPBUS-LU-1A	1.215E-5	1.944E-6	1.16	1.000
363	1RHHEX-LF-1RHE1B	2.807E-2	4.620E-3	1.16	1.005
364	1RHHEX-LF-1RHE1A	2.807E-2	4.546E-3	1.16	1.005
365	1EPBKR-SO-242	3.356E-5	5.165E-6	1.15	1.000
366	1EPBKR-SO-15E3	3.356E-5	5.165E-6	1.15	1.000
367	1EPBKR-SO-15E1	3.356E-5	5.165E-6	1.15	1.000
368	2EEBKR-SO-25H11	3.356E-5	5.165E-6	1.15	1.000
369	1RHMV--PG-1RH30	4.105E-4	6.174E-5	1.15	1.000
370	1RHMV--PG-1RH19	4.105E-4	6.174E-5	1.15	1.000
371	1CCMV--PG-1CC762	4.105E-4	6.174E-5	1.15	1.000
372	1RHMV--PG-1RH24	4.105E-4	6.174E-5	1.15	1.000
373	1CCMV--PG-1CC785	4.105E-4	6.174E-5	1.15	1.000
374	1EPBUS-UM-1A1	2.000E-4	2.962E-5	1.15	1.000
375	2HVPCV-FC-2235B1	1.812E-2	2.666E-3	1.14	1.003
376	1CCSV--SO-RV128A	9.333E-5	1.347E-5	1.14	1.000
377	1CCSV--SO-RV128B	9.333E-5	1.347E-5	1.14	1.000
378	1RHMV--FC-1RH25	1.250E-4	1.804E-5	1.14	1.000

TABLE 3.4.1-7 (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
379	1EPBKR-SO-15J12	3.356E-5	4.655E-6	1.14	1.000
380	1EEBKR-SO-15H12	3.356E-5	4.655E-6	1.14	1.000
381	2HVMOV-FC-213B	1.090E-2	1.484E-3	1.13	1.001
382	2HVMOV-FC-211B	1.090E-2	1.484E-3	1.13	1.001
383	1EPBUS-LU-1B1	1.215E-5	1.612E-6	1.13	1.000
384	1EEBUS-LU-1HSTUB	1.215E-5	1.612E-6	1.13	1.000
385	1EEBUS-LU-1JSTUB	1.215E-5	1.612E-6	1.13	1.000
386	1EPBUS-LU-1A1	1.215E-5	1.612E-6	1.13	1.000
387	1FWCKV-CC-125127	6.339E-5	8.267E-6	1.13	1.000
388	1FWCKV-CC-9395	6.339E-5	8.267E-6	1.13	1.000
389	1SWPIP-UM-HDRA	2.281E-2	2.987E-3	1.13	1.003
390	1HVACU-UM-1HVAC7	1.654E-3	2.074E-4	1.13	1.000
391	1EPBUS-LU-2	1.215E-5	1.490E-6	1.12	1.000
392	1EPBUS-LU-4	1.215E-5	1.490E-6	1.12	1.000
393	HEP-1FRH:1-5	3.125E-3	3.829E-4	1.12	1.000
394	1RCMOV-LK-1536	2.500E-2	3.094E-3	1.12	1.003
395	1RCMOV-LK-1535	2.500E-2	3.060E-3	1.12	1.003
396	1FWFCV-CC-798999	1.812E-3	2.120E-4	1.12	1.000
397	HEP-0AP55-40HR	1.250E-1	1.656E-2	1.12	1.017
398	2HVTCV-FC-TCV266	1.812E-2	2.108E-3	1.11	1.002
399	NON-REC-B10	2.000E-2	2.285E-3	1.11	1.002
400	2HVMOD-FC-MOD237	1.090E-2	1.219E-3	1.11	1.001
401	2HVMOD-FO-MOD238	1.090E-2	1.219E-3	1.11	1.001
402	HEP-1AP15-1E	7.799E-4	8.620E-5	1.11	1.000
403	1FWMOV-CC-150ABC	3.903E-4	4.259E-5	1.11	1.000
404	T9A-FREQ-RSST-C	7.143E-2	8.311E-3	1.11	1.008
405	T9A-FREQ-500KV-1	1.786E-1	2.330E-2	1.11	1.024
406	1EEBUS-UM-1JSTUB	1.000E-5	1.042E-6	1.10	1.000
407	1EEBUS-UM-1HSTUB	1.000E-5	1.042E-6	1.10	1.000
408	HEP-1AP15-6	2.815E-2	2.952E-3	1.10	1.003
409	2HVPAT-FS-HVP20B	1.983E-3	1.951E-4	1.10	1.000
410	2HVPAT-FS-HVP22B	1.983E-3	1.951E-4	1.10	1.000
411	2HVCHU-FR-2HVE4B	1.506E-3	1.480E-4	1.10	1.000
412	2HVFAN-FS-2FMO6	3.933E-3	3.787E-4	1.10	1.000
413	HEP-0AP12-20HR	2.600E-4	2.177E-5	1.08	1.000
414	2HVPAT-FR-HVP20B	7.930E-4	6.482E-5	1.08	1.000
415	2HVPAT-FR-HVP22B	7.930E-4	6.482E-5	1.08	1.000
416	1MSSV--FO-101C	1.250E-2	1.006E-3	1.08	1.001
417	NON-REC-B02	3.400E-1	4.072E-2	1.08	1.042
418	2CDCKV-FC-2CD211	6.339E-4	4.896E-5	1.08	1.000
419	2SWCKV-FC-2SW337	6.339E-4	4.896E-5	1.08	1.000
420	2SWCKV-FC-2SW353	6.339E-4	4.896E-5	1.08	1.000

TABLE 3.4.1-7 (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
421	1RCPORV-DMDSBO	2.000E-1	1.928E-2	1.08	1.020
422	REC-CONTAINMENT	2.000E-2	1.552E-3	1.08	1.002
423	1HVMOV-CC-HV113	3.903E-4	2.713E-5	1.07	1.000
424	1HVMOV-CC-HV111	3.903E-4	2.713E-5	1.07	1.000
425	1MSAOV-FO-TV101C	1.812E-2	1.240E-3	1.07	1.001
426	1HVSTR-PG-1HVS1B	9.528E-3	6.418E-4	1.07	1.001
427	1MSRV--FC-101B	9.988E-3	6.552E-4	1.06	1.001
428	1FWMV--FO-1FW128	1.250E-4	7.779E-6	1.06	1.000
429	1CHHEX-LU-1CHE5B	2.090E-4	1.297E-5	1.06	1.000
430	1CCPAT-FR-1CCP1A	7.930E-4	4.770E-5	1.06	1.000
431	1HVPAT-CC-HVP22	1.983E-4	1.182E-5	1.06	1.000
432	1HVPAT-CC-HVP20	1.983E-4	1.182E-5	1.06	1.000
433	2HVCHU-FS-2HVE4B	4.545E-2	2.800E-3	1.06	1.003
434	1QSSTR-PG-1FL1B	2.822E-2	1.657E-3	1.06	1.002
435	1SIMOV-FO-1115D	1.090E-2	6.050E-4	1.05	1.001
436	1SIMOV-FO-1115B	1.090E-2	6.050E-4	1.05	1.001
437	1SWMOV-SC-SW208A	1.208E-5	6.561E-7	1.05	1.000
438	1SWMOV-SC-SW208B	1.208E-5	6.561E-7	1.05	1.000
439	1SWMOV-SC-SW108A	1.208E-5	6.561E-7	1.05	1.000
440	1SWMOV-SC-SW108B	1.208E-5	6.561E-7	1.05	1.000
441	1CHCKV-CC-267279	6.339E-5	3.387E-6	1.05	1.000
442	HEP-1E1-25	1.175E-2	6.087E-4	1.05	1.001
443	1SIMOV-FC-1836	1.090E-2	5.418E-4	1.05	1.001
444	1CCAOV-FC-TV103B	1.812E-2	9.002E-4	1.05	1.001
445	1CCAOV-FC-TV103A	1.812E-2	8.953E-4	1.05	1.001
446	HEP-0AP55-30HR	6.565E-3	3.173E-4	1.05	1.000
447	1CCHEX-LU-1CCE1A	2.090E-4	9.363E-6	1.04	1.000
448	C-B103	3.200E-1	2.081E-2	1.04	1.021
449	HEP-1ECA3:3-35	4.924E-3	2.157E-4	1.04	1.000
450	1EEBUS-UM-VB-II	2.000E-4	8.503E-6	1.04	1.000
451	NON-REC-B221	8.999E-4	3.506E-5	1.04	1.000
452	1MSMV--FO-1MS97	1.250E-4	4.680E-6	1.04	1.000
453	1SWCKV-FC-SW-114	6.339E-4	2.362E-5	1.04	1.000
454	1SWCKV-FC-SW-116	6.339E-4	2.362E-5	1.04	1.000
455	HEP-1ECA3:3-27	8.974E-2	3.578E-3	1.04	1.004
456	1MSRV--FC-101A	9.988E-3	3.634E-4	1.04	1.000
457	1SWPIP-UM-HDRB	2.281E-2	8.311E-4	1.04	1.001
458	1SICKV-FC-1SI185	6.339E-4	2.234E-5	1.04	1.000
459	1MSAOV-FC-TV111A	1.812E-2	6.497E-4	1.04	1.001
460	1MSAOV-FC-TV111B	1.812E-2	6.497E-4	1.04	1.001
461	1MSMOV-FO-NRV101	1.090E-2	3.856E-4	1.03	1.000
462	1EPTFM-LP-RSST-C	1.899E-5	6.631E-7	1.03	1.000

TABLE 3.4.1-7 (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
463	1EPTFM-LP-1	1.899E-5	6.631E-7	1.03	1.000
464	1EPBKR-SO-15F3	3.356E-5	1.172E-6	1.03	1.000
465	1EPBKR-SO-15F1	3.356E-5	1.172E-6	1.03	1.000
466	1EPBKR-SO-L102	3.356E-5	1.172E-6	1.03	1.000
467	1EPBKR-SO-332	3.356E-5	1.172E-6	1.03	1.000
468	1EEBKR-SO-15H11	3.356E-5	1.172E-6	1.03	1.000
469	1EPBUS-UM-1F	2.000E-4	6.983E-6	1.03	1.000
470	1EPBUS-UM-1	2.000E-4	6.983E-6	1.03	1.000
471	1EPBUS-UM-3	2.000E-4	6.983E-6	1.03	1.000
472	1QSSTR-PG-1FL1A	2.822E-2	1.009E-3	1.03	1.001
473	1QSMOV-FC-101B	1.090E-2	3.808E-4	1.03	1.000
474	2HVCHU-UM-2HVE4B	9.440E-2	3.517E-3	1.03	1.004
475	REC-2AP28	1.017E-1	3.809E-3	1.03	1.004
476	2HVPCV-FC-2235C1	1.812E-2	6.053E-4	1.03	1.001
477	1SWSCN-PG-1SWP1B	9.528E-3	3.053E-4	1.03	1.000
478	REC-0OP21:6	1.687E-3	5.122E-5	1.03	1.000
479	1CCMOV-FC-CC100B	1.090E-2	3.273E-4	1.03	1.000
480	1CCMOV-FC-CC100A	1.090E-2	3.248E-4	1.03	1.000
481	REC-1ES1:2	2.660E-3	7.839E-5	1.03	1.000
482	1EEBUS-UM-DC-II	2.000E-4	5.775E-6	1.03	1.000
483	1EEINV-LU-II	6.136E-4	1.772E-5	1.03	1.000
484	2HVMOV-FC-213C	1.090E-2	3.085E-4	1.03	1.000
485	2HVMOV-FC-211C	1.090E-2	3.085E-4	1.03	1.000
486	1EPBUS-UM-1B3	2.000E-4	5.428E-6	1.03	1.000
487	1QSMOV-FC-101A	1.090E-2	2.942E-4	1.03	1.000
488	1HVPAT-FR-HVP20B	7.930E-4	2.074E-5	1.03	1.000
489	1HVPAT-FR-HVP22B	7.930E-4	2.074E-5	1.03	1.000
490	2EGEDG-UM-2J	1.069E-1	2.985E-3	1.02	1.003
491	1MSSRV-DMDT7	3.999E-2	1.006E-3	1.02	1.001
492	1MSMV--PG-1MS269	1.368E-4	3.127E-6	1.02	1.000
493	1MSMV--PG-1MS271	1.368E-4	3.127E-6	1.02	1.000
494	1MSMV--PG-1MS270	1.368E-4	3.127E-6	1.02	1.000
495	1MSMV--PG-1MS268	1.368E-4	3.127E-6	1.02	1.000
496	1SICKV-FC-1SI207	6.339E-4	1.447E-5	1.02	1.000
497	1FWCKV-FC-1FW68	6.339E-4	1.416E-5	1.02	1.000
498	2HVACU-UM-2HVAC6	1.654E-3	3.656E-5	1.02	1.000
499	T9B-FREQ-RSST-A	7.143E-2	1.678E-3	1.02	1.002
500	1SICKV-FC-1SI206	6.339E-4	1.354E-5	1.02	1.000
501	2HVPAT-FS-HVP22C	1.983E-3	4.190E-5	1.02	1.000
502	2HVPAT-FS-HVP20C	1.983E-3	4.190E-5	1.02	1.000
503	2HVCHU-FR-2HVE4C	1.506E-3	3.167E-5	1.02	1.000
504	NON-REC-B01	4.799E-1	1.928E-2	1.02	1.020

TABLE 3.4.1-7 (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
505	NON-REC-B103	6.799E-1	4.431E-2	1.02	1.046
506	IE-T2A	5.500E-1	2.510E-2	1.02	1.026
507	C-B117	3.200E-1	9.597E-3	1.02	1.010
508	1RCPIC-LF-PC403	4.123E-2	8.598E-4	1.02	1.001
509	1RCPIC-LF-PC402	4.123E-2	8.598E-4	1.02	1.001
510	T9B-FREQ-500KV-2	1.786E-1	4.317E-3	1.02	1.004
511	1RHCKV-FO-1RH7	3.442E-3	6.580E-5	1.02	1.000
512	1RHCKV-FO-1RH15	3.442E-3	6.580E-5	1.02	1.000
513	2HVCHU-FS-2HVE4C	4.545E-2	8.969E-4	1.02	1.001
514	1QSPSB-FS-1QSP1B	3.933E-3	7.146E-5	1.02	1.000
515	2SWPAT-UM-2SWP1B	3.725E-2	6.937E-4	1.02	1.001
516	1QSPSB-FS-1QSP1A	3.933E-3	6.974E-5	1.02	1.000
517	1SIMOV-FO-1864B	1.090E-2	1.899E-4	1.02	1.000
518	1SIMOV-FC-1890B	1.090E-2	1.899E-4	1.02	1.000
519	NON-REC-B235	8.999E-4	1.551E-5	1.02	1.000
520	1SIMOV-FO-1864A	1.090E-2	1.882E-4	1.02	1.000
521	1SIMOV-FC-1890A	1.090E-2	1.882E-4	1.02	1.000
522	1SWCKV-CC-647648	6.339E-5	1.034E-6	1.02	1.000
523	1QSPSB-TM-1QSP1A	3.750E-3	6.088E-5	1.02	1.000
524	1QSPSB-UM-1QSP1A	3.750E-3	6.088E-5	1.02	1.000
525	1QSLEV-TM-RWSTB	1.400E-3	2.171E-5	1.02	1.000
526	1QSLEV-TM-RWSTA	1.400E-3	2.171E-5	1.02	1.000
527	HEP-1FRS:1-5	2.970E-2	4.696E-4	1.02	1.000
528	1MSPIC-LF-1447	8.022E-2	1.330E-3	1.02	1.001
529	1MSPIC-LF-1446	8.022E-2	1.330E-3	1.02	1.001
530	1QSPSB-UM-1QSP1B	3.750E-3	5.638E-5	1.01	1.000
531	1QSPSB-TM-1QSP1B	3.750E-3	5.638E-5	1.01	1.000
532	PROB-M03	2.942E-1	6.187E-3	1.01	1.006
533	C-LT01	9.068E-1	1.436E-1	1.01	1.168
534	1EEBKR-SO-II-14	3.356E-5	4.922E-7	1.01	1.000
535	1EEBKR-SO-VB2-35	3.356E-5	4.922E-7	1.01	1.000
536	1HVCHU-FS-1HVE4C	4.545E-2	6.827E-4	1.01	1.001
537	1SIMOV-PG-1267B	8.207E-4	1.134E-5	1.01	1.000
538	1EPBUS-LU-1F	1.215E-5	1.662E-7	1.01	1.000
539	1EPBUS-LU-1	1.215E-5	1.662E-7	1.01	1.000
540	1EPBUS-LU-3	1.215E-5	1.662E-7	1.01	1.000
541	1TMSOV-FC-20-ET	1.812E-2	2.474E-4	1.01	1.000
542	1TMSOV-FC-ASO	1.812E-2	2.474E-4	1.01	1.000
543	2HVCHU-UM-2HVE4C	9.440E-2	1.388E-3	1.01	1.001
544	1HVPCV-FC-1235C1	1.812E-2	2.454E-4	1.01	1.000
545	1SIMOV-FO-1885A	1.090E-2	1.450E-4	1.01	1.000
546	1SIMOV-FO-1885C	1.090E-2	1.450E-4	1.01	1.000

TABLE 3.4.1-7 (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
547	C-B102	3.200E-1	6.155E-3	1.01	1.006
548	1SIMOV-FO-1885B	1.090E-2	1.441E-4	1.01	1.000
549	1SIMOV-FO-1885D	1.090E-2	1.441E-4	1.01	1.000
550	NON-REC-B220	8.999E-4	1.117E-5	1.01	1.000
551	C-RC303	8.750E-1	8.554E-2	1.01	1.094
552	HEP-0AP10	5.274E-3	6.366E-5	1.01	1.000
553	1HVMOV-FC-111C	1.090E-2	1.322E-4	1.01	1.000
554	1HVMOV-FC-113C	1.090E-2	1.322E-4	1.01	1.000
555	1FWCKV-CC-477911	6.339E-5	7.413E-7	1.01	1.000
556	2EGEDG-FS-2J	1.434E-2	1.638E-4	1.01	1.000
557	2EGEDG-FR-2J	1.330E-2	1.517E-4	1.01	1.000
558	NON-REC-B117	6.799E-1	2.045E-2	1.01	1.021
559	1EEBCH-LP-III	8.399E-5	7.986E-7	1.01	1.000
560	1EEBCH-LP-I	8.399E-5	7.986E-7	1.01	1.000
561	REC-MMP-C-MR-2	2.510E-1	3.072E-3	1.01	1.003
562	HEP-0AP12-10HR	4.949E-3	4.434E-5	1.01	1.000
563	2EGEDG-UM-2H	1.069E-1	1.036E-3	1.01	1.001
564	1RCMOV-FC-1535	1.090E-2	9.288E-5	1.01	1.000
565	PROB-PR01	2.776E-1	3.033E-3	1.01	1.003
566	1EEBKR-SO-III-11	3.356E-5	2.575E-7	1.01	1.000
567	1EEBKR-SO-I-11	3.356E-5	2.575E-7	1.01	1.000
568	1EEBKR-SO-J1-B1L	3.356E-5	2.575E-7	1.01	1.000
569	1EEBKR-SO-H4-D2L	3.356E-5	2.575E-7	1.01	1.000
570	1SIMOV-FC-1867D	1.090E-2	8.154E-5	1.01	1.000
571	1SIMOV-FC-1867B	1.090E-2	8.154E-5	1.01	1.000
572	2EGEDG-FR-2H	1.330E-2	9.813E-5	1.01	1.000
573	2EGEDG-FS-2H	1.434E-2	1.057E-4	1.01	1.000
574	HEP-1E0-22	1.880E-2	1.340E-4	1.01	1.000
575	1HVPAT-FS-HVP20C	1.983E-3	1.345E-5	1.01	1.000
576	1HVPAT-FS-HVP22C	1.983E-3	1.345E-5	1.01	1.000
577	1HVCHU-UM-1HVE4C	9.440E-2	7.027E-4	1.01	1.001
578	1HVCHU-FR-1HVE4C	1.506E-3	9.717E-6	1.01	1.000
579	1QSLIC-LF-100B	4.633E-3	2.902E-5	1.01	1.000
580	1QSLIC-LF-100C	4.633E-3	2.902E-5	1.01	1.000
581	1QSLIC-LF-100D	4.633E-3	2.902E-5	1.01	1.000
582	1QSLIC-LF-100A	4.633E-3	2.902E-5	1.01	1.000
583	NON-REC-B102	6.799E-1	1.313E-2	1.01	1.013
584	1QSHEP-1QS21	7.499E-4	4.557E-6	1.01	1.000
585	1QSHEP-1QS5	7.499E-4	4.522E-6	1.01	1.000
586	1RHMOV-FC-1720A	1.090E-2	6.367E-5	1.01	1.000
587	1RHMOV-FC-1720B	1.090E-2	6.367E-5	1.01	1.000
588	1QSCKV-FC-1QS-11	6.339E-4	3.441E-6	1.01	1.000

TABLE 3.4.1-7 (Continued)
BASIC EVENTS RANKED BY THE
RISK ACHIEVEMENT WORTH IMPORTANCE MEASURE

Rank	Event Name	Point Estimate	Fussell-Vesely Importance	Risk Achieve-ment Worth	Risk Reduc-tion Worth
589	1QSCKV-FC-1QS-19	6.339E-4	3.268E-6	1.01	1.000
590	1EPBKR-SO-15B10	3.356E-5	1.693E-7	1.01	1.000
591	1EPBKR-SO-14B3-1	3.356E-5	1.693E-7	1.01	1.000
592	1CCPSB-UM-1CCP1B	3.750E-3	1.782E-5	1.00	1.000
593	1CCPSB-FS-1CCP1B	3.933E-3	1.869E-5	1.00	1.000
594	2HVPAT-FR-HVP22C	7.930E-4	3.571E-6	1.00	1.000
595	2HVPAT-FR-HVP20C	7.930E-4	3.571E-6	1.00	1.000
596	1MSTCV-FC-1408A	1.812E-2	8.251E-5	1.00	1.000
597	1MSTCV-FC-1408B	1.812E-2	8.251E-5	1.00	1.000
598	1MSMV--LK-1MS168	1.000E-2	4.500E-5	1.00	1.000
599	1MSMV--LK-1MS179	1.000E-2	4.500E-5	1.00	1.000
600	1CCCKV-FO-1CC24	3.442E-3	1.310E-5	1.00	1.000
601	1CCCKV-FC-1CC47	6.339E-4	2.376E-6	1.00	1.000
602	1CCPSB-FR-1CCP1B	7.927E-4	2.971E-6	1.00	1.000
603	1RCMOV-CC-535536	3.903E-4	1.443E-6	1.00	1.000
604	REC-1ES1:4-1	1.039E-1	3.923E-4	1.00	1.000
605	1RCMOV-FC-1536	1.090E-2	3.432E-5	1.00	1.000
606	1MSAOV-FC-TV101A	1.812E-2	5.327E-5	1.00	1.000
607	1MSAOV-FC-TV101B	1.812E-2	5.327E-5	1.00	1.000
608	1SW-COLDWEA-3MO	2.500E-1	9.437E-4	1.00	1.001
609	HEP-1E0-15	1.075E-3	2.979E-6	1.00	1.000
610	1MSMV--PG-1MS168	9.123E-5	2.406E-7	1.00	1.000
611	1MSMV--PG-1MS179	9.123E-5	2.406E-7	1.00	1.000
612	C-QS05	9.460E-1	4.416E-2	1.00	1.046
613	1HVPAT-FR-HVP20C	7.930E-4	1.904E-6	1.00	1.000
614	1HVPAT-FR-HVP22C	7.930E-4	1.904E-6	1.00	1.000
615	2EEBKR-FO-25J2	2.735E-4	5.649E-7	1.00	1.000
616	1EPBKR-FO-15F3	2.735E-4	5.649E-7	1.00	1.000
617	2EEBKR-FO-25J11	2.735E-4	5.649E-7	1.00	1.000
618	1EEBKR-FO-15H11	2.735E-4	5.649E-7	1.00	1.000
619	2EGEDG-CC-2H-2J	2.663E-4	5.500E-7	1.00	1.000
620	1EPBKR-FO-15F4	2.735E-4	5.649E-7	1.00	1.000
621	2EGEDG-TM-2J	5.708E-4	1.179E-6	1.00	1.000
622	1EPBKR-FC-15F1	1.834E-3	3.789E-6	1.00	1.000
623	1BDAOV-FO-TV100F	1.812E-2	3.799E-5	1.00	1.000
624	1BDSOV-FO-100E	1.812E-2	3.799E-5	1.00	1.000
625	1BDAOV-FO-TV100J	1.812E-2	3.799E-5	1.00	1.000
626	1BDSOV-FO-100F	1.812E-2	3.799E-5	1.00	1.000
627	1BDAOV-FO-TV100E	1.812E-2	3.799E-5	1.00	1.000
628	1BDSOV-FO-100J	1.812E-2	3.799E-5	1.00	1.000
629	C-HV05	7.490E-1	6.120E-3	1.00	1.006
630	C-L08	8.410E-1	1.027E-2	1.00	1.010

TABLE 3.4.1-7 (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
631	1CHPAT-UM-1CHP1C	3.267E-1	9.008E-4	1.00	1.001
632	C-Y02	9.800E-1	8.556E-2	1.00	1.094
633	C-D102	9.400E-1	2.601E-2	1.00	1.027
634	1SWTCV-FC-SW102C	1.812E-2	2.847E-5	1.00	1.000
635	C-B111	3.200E-1	7.197E-4	1.00	1.001
636	1EPBKR-FC-15A2	1.834E-3	2.760E-6	1.00	1.000
637	1RCMOV-FO-1536	1.090E-2	1.609E-5	1.00	1.000
638	1RCMOV-FO-1535	1.090E-2	1.609E-5	1.00	1.000
639	PROB-FM01	9.522E-1	2.383E-2	1.00	1.024
640	1CHPAT-FS-1CHP1C	5.078E-3	5.701E-6	1.00	1.000
641	1CCHEX-LF-1CCE1B	9.477E-3	9.363E-6	1.00	1.000
642	1SW-HOTWEA-9MO	7.500E-1	2.874E-3	1.00	1.003
643	C-SGI01	9.890E-1	7.934E-2	1.00	1.086
644	1EPBKR-FC-15B2	1.834E-3	1.587E-6	1.00	1.000
645	2EEBUS-UM-2J	2.000E-4	1.619E-7	1.00	1.000
646	1EPBKR-FO-15A1	2.735E-4	2.099E-7	1.00	1.000
647	C-TT01	8.000E-1	3.033E-3	1.00	1.003
648	1CHPAT-PT-14:3	6.999E-4	5.187E-7	1.00	1.000
649	1CHPAT-FR-24HP1C	7.930E-4	5.877E-7	1.00	1.000
650	NON-REC-B111	6.799E-1	1.530E-3	1.00	1.002
651	1FWCKV-FC-1FW93	6.339E-4	4.547E-7	1.00	1.000
652	1FWHEP-MOV-100B	7.499E-4	5.379E-7	1.00	1.000
653	C-P02	9.870E-1	5.411E-2	1.00	1.057
654	1CHCKV-FC-1CH279	6.339E-4	3.356E-7	1.00	1.000
655	1FWCKV-FC-1FW279	6.339E-4	3.268E-7	1.00	1.000
656	1FWHEP-MOV-100D	7.499E-4	3.866E-7	1.00	1.000
657	NON-REC-B229	8.999E-4	4.489E-7	1.00	1.000
658	C-H105	9.490E-1	8.554E-3	1.00	1.009
659	1FWCKV-FC-1FW127	6.339E-4	2.736E-7	1.00	1.000
660	1FWHEP-HCV-100C	7.499E-4	3.236E-7	1.00	1.000
661	C-H106	9.356E-1	5.643E-3	1.00	1.006
662	1MSMV--LK-1MS59	3.999E-2	1.608E-5	1.00	1.000
663	C-QS06	9.460E-1	6.112E-3	1.00	1.006
664	PROB-D104A	5.999E-2	2.085E-5	1.00	1.000
665	1MSMV--LK-1MS21	3.999E-2	1.301E-5	1.00	1.000
666	1EPBKR-FC-G12	1.834E-3	4.896E-7	1.00	1.000
667	HEP-1AP33:1	3.866E-1	1.552E-4	1.00	1.000
668	1SWMOV-FC-SW101D	1.090E-2	2.412E-6	1.00	1.000
669	1SWMOV-FC-SW101B	1.090E-2	2.412E-6	1.00	1.000
670	1SWMOV-FC-SW101A	1.090E-2	2.412E-6	1.00	1.000
671	1SWMOV-FC-SW101C	1.090E-2	2.412E-6	1.00	1.000
672	1MSRV--FO-101C	2.500E-2	4.680E-6	1.00	1.000

TABLE 3.4.1-7 (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
673	C-D105	9.470E-1	3.255E-3	1.00	1.003
674	1FWHCV-FO-100C	1.812E-2	2.657E-6	1.00	1.000
675	C-Y03	8.980E-1	1.147E-3	1.00	1.001
676	1FWBKR-FC-15A6	1.834E-3	1.546E-7	1.00	1.000
677	1FWBKR-FC-15C5	1.834E-3	1.546E-7	1.00	1.000
678	1FWBKR-FC-15A5	1.834E-3	1.546E-7	1.00	1.000
679	1FWBKR-FC-15B5	1.834E-3	1.546E-7	1.00	1.000
680	C-QS03	9.460E-1	1.446E-3	1.00	1.001
681	C-QS04	9.460E-1	1.426E-3	1.00	1.001
682	C-H103	9.610E-1	1.401E-3	1.00	1.001
683	C-H104	9.620E-1	1.381E-3	1.00	1.001
684	1FWMOV-FC-100C	1.090E-2	5.972E-7	1.00	1.000
685	1FWHCV-FC-100B	1.812E-2	9.925E-7	1.00	1.000
686	1SWMOV-FC-SW103B	1.090E-2	5.062E-7	1.00	1.000
687	1SWMOV-FC-SW104B	1.090E-2	5.062E-7	1.00	1.000
688	1RSSTR-PG-TEMPB	2.822E-2	1.012E-6	1.00	1.000
689	1CHMOV-FO-1286A	1.090E-2	3.654E-7	1.00	1.000
690	1SWMOV-FC-SW103A	1.090E-2	3.364E-7	1.00	1.000
691	1SWMOV-FC-SW104A	1.090E-2	3.364E-7	1.00	1.000
692	HEP-0AP12-30HR	6.565E-3	1.723E-7	1.00	1.000
693	1RSSTR-PG-TEMPA	2.822E-2	6.727E-7	1.00	1.000
694	HEP-0AP12-40HR	1.250E-1	2.721E-6	1.00	1.000
695	C-B02	6.600E-1	9.476E-6	1.00	1.000
696	C-RC301	8.750E-1	1.905E-5	1.00	1.000
697	C-Y04	9.850E-1	1.104E-4	1.00	1.000
698	1MSPORV-DMDT7	1.000E+0	4.680E-6	1.00	1.000
699	PROB-Q08	1.000E+0	3.154E-3	1.00	1.003
700	1EE-BAT-II-2HR	1.000E+0	8.499E-2	1.00	1.093
701	1EE-BAT-IV-2HR	1.000E+0	2.039E-3	1.00	1.002
702	1EE-BAT-I-2HR	1.000E+0	1.442E-1	1.00	1.169
703	1EE-BAT-III-2HR	1.000E+0	3.614E-2	1.00	1.037
704	C-P01	1.000E+0	1.347E-2	1.00	1.014
705	HEP-NO-PROCEDURE	1.000E+0	3.910E-2	1.00	1.041
706	HEP-1ES1:2-S1	1.000E+0	3.860E-2	1.00	1.040
707	HEP-1E0-14	1.000E+0	1.952E-4	1.00	1.000
708	HEP-1FRC:1-11-S1	1.000E+0	5.962E-2	1.00	1.063
709	IE-TH	1.750E+0	6.187E-3	1.00	1.006
710	IE-T3	1.350E+0	6.078E-2	0.98	1.065

TABLE 3.4.1-8
CONTRIBUTION TO CORE DAMAGE FREQUENCY OF FUNCTIONAL FAILURES

Function	Contribution to CDF
Failure of Injection (D1,D2,D3)	42%
Failure to Cooldown and Depressurize (O,Y)	36%
Failure of Emergency Switchgear Room Cooling (T8,HV)	34%
Failure of Auxiliary Feedwater (L,Lt)	24%
Failure of Recirculation (H1,H2)	13%
Failure to Recover Offsite Power (B)	12%
Failure of Feed and Bleed (P)	1%
Seal LOCA (T4,Slc)	<1

TABLE 3.4.1-9
CONTRIBUTION OF PLANT DAMAGE STATES TO CONTAINMENT FAILURE

Plant Damage State	Core Damage Frequency (per year)	% Contribution
21	1.93E-5	28.3%
4	1.27E-5	18.7%
20	8.22E-6	12.1%
25	7.01E-6	10.3%
14	2.92E-6	4.3%
12	2.92E-6	4.3%
23	2.65E-6	3.9%
5	2.60E-6	3.8%
3	2.07E-6	3.0%
24	1.60E-6	2.4%
13	1.42E-6	2.1%
8	1.27E-6	1.9%
7	1.15E-6	1.7%
18	1.06E-6	1.6%
15	2.48E-7	0.4%
11	2.35E-7	0.3%
9	2.31E-7	0.3%
16	1.22E-7	0.2%
1	8.98E-8	0.1%
6	6.82E-8	0.1%
22	5.44E-8	0.1%
17	4.05E-8	0.1%
2	2.19E-8	0.0%
19	4.31E-9	0.0%
10	1.62E-9	0.0%

TABLE 3.4.1-10
CONTRIBUTION OF SEQUENCES TO PLANT DAMAGE STATES WHICH LEAD TO CONTAINMENT FAILURE

Plant Damage Category	Sequence	Core Damage Frequency (per year)	Fraction of Total	Functional Failures
21	Sum =	1.93E-5	28.3%	
	S2P35	5.15E-6	7.6%	S2D1D3
	S1P38	4.04E-6	5.9%	S1D1Y
	S1P10	2.45E-6	3.6%	S1OH2
	S2P04	2.45E-6	3.6%	S2H1
	S2P43	1.19E-6	1.8%	S2D1Y
	T1TrP14	1.01E-6	1.5%	T1TrOH1
	T1P07	5.66E-7	0.8%	T1LH2H1
	S2P39	5.20E-7	0.8%	S2D1D2
	T9ATrP14	3.88E-7	0.6%	T9ATrOH1
	S2P47	3.27E-7	0.5%	S2D1L
	THP46	2.14E-7	0.3%	THKMTtQ
	T9BP10	1.79E-7	0.3%	T9BQH2
	T1P06	1.69E-7	0.2%	T1LH2
	T9AP10	1.31E-7	0.2%	T9AQH2
	S2P32	9.12E-8	0.1%	S2D1H1
	S2P17	9.02E-8	0.1%	S2FmOH2
	T1P21	8.17E-8	0.1%	T1QH1
4	Sum =	1.27E-5	18.7%	
	T8P22	3.17E-6	4.7%	T8LtRC1
	T1AP51	2.99E-6	4.4%	T1ALtBB1
	T8P02	2.52E-6	3.7%	T8RC2
	T9ATrP08	1.53E-6	2.2%	T9ATrLtRC1
	T1AP07	1.38E-6	2.0%	T1ABB1
	T9ATrP02	8.33E-7	1.2%	T9ATrRC2
	T3TrP22	1.21E-7	0.2%	T3TrLtRC1
20	Sum =	8.22E-6	12.1%	
	T1TrP17	4.00E-6	5.9%	T1TrOD1
	T1P10	2.71E-6	4.0%	T1LD1
	T2P09	7.22E-7	1.1%	T2LD1
	T9ATrP17	3.07E-7	0.5%	T9ATrOD1
	T9BP13	1.30E-7	0.2%	T9BQD1
	T9AP13	1.02E-7	0.2%	T9AQD1
	T1P36	7.75E-8	0.1%	T1QD1D3

TABLE 3.4.1-10 (Continued)
CONTRIBUTION OF SEQUENCE TO PLANT DAMAGE STATE

Plant Damage Category	Sequence	Core Damage Frequency (per year)	Fraction of Total	Functional Failures
25	Sum =	7.01E-6	10.3%	
	T7P04	2.98E-6	4.4%	T7OO2
	T7P03	1.98E-6	2.9%	T7OW
	T7P06	1.10E-6	1.6%	T7SGIW
	T7P26	3.85E-7	0.6%	T7D1SGI
	T7P23	1.80E-7	0.3%	T7D1OD3
	T7P07	1.10E-7	0.2%	T7SGIO2
	T7P25	8.44E-8	0.1%	T7D1OO2
	T7P27	7.20E-8	0.1%	T7D1L
14	Sum =	2.92E-6	4.3%	
	T3TrP11	1.67E-6	2.5%	T3TrOD1
	T2ATrP11	6.78E-7	1.0%	T2ATrOD1
	T1AP58	4.17E-7	0.6%	T1AQB
	T8P11	9.57E-8	0.1%	T8OD1
12	Sum =	2.92E-6	4.3%	
	AP15	2.12E-6	3.1%	AD2
	AP02	5.17E-7	0.8%	ADh
	RXP01	2.66E-7	0.4%	RX
23	Sum =	2.65E-6	3.9%	
	T1TrP21	2.22E-6	3.3%	T1TrOD1Qs
	T1P14	2.07E-7	0.3%	T1LD1Qs
5	Sum =	2.60E-6	3.8%	
	T3TrP03	1.57E-6	2.3%	T3TrRC2Ch
	T2ATrP03	6.40E-7	0.9%	T2ATrRC2Ch
	T3TrP23	1.84E-7	0.3%	T3TrLtRC1Ch
	T2ATrP23	7.51E-8	0.1%	T2ATrLtRC1Ch
3	Sum =	2.07E-6	3.0%	
	T1AP46	1.41E-6	2.1%	T1ALtB
	T1AP02	6.51E-7	1.0%	T1AB
24	Sum =	1.60E-6	2.4%	
	VXP07	1.52E-6	2.2%	VXFm
	VXP03	7.68E-8	0.1%	VXO
13	Sum =	1.42E-6	2.1%	
	AP03	8.26E-7	1.2%	AH1
	AP11	5.88E-7	0.9%	AD3

TABLE 3.4.1-10 (Continued)
CONTRIBUTION OF SEQUENCE TO PLANT DAMAGE STATE

Plant Damage Category	Sequence	Core Damage Frequency (per year)	Fraction of Total	Functional Failures
8	Sum =	1.27E-6	1.9%	
	T1P15	5.16E-7	0.8%	T1LP
	T9BP02	2.37E-7	0.3%	T9BL
	T9AP02	1.72E-7	0.3%	T9AL
	T2P14	1.30E-7	0.2%	T2LP
	T5BP02	9.37E-8	0.1%	T5BL
	T5AP02	9.36E-8	0.1%	T5AL
7	Sum =	1.15E-6	1.7%	
	T8P06	6.06E-7	0.9%	T8RC2RC3
	T3TrP06	2.83E-7	0.4%	T3TrRC2RC3
	T2ATrP06	1.15E-7	0.2%	T2ATrRC2RC3
	T9ATrP06	1.07E-7	0.2%	T9ATrRC2RC3
18	Sum =	1.06E-6	1.6%	
	T1AP67	8.86E-7	1.3%	T1AQBB1
	T1AP26	1.04E-7	0.2%	T1AS1cBB1
15	Sum =	2.48E-7	0.4%	
11	Sum =	2.35E-7	0.3%	
	T1P19	1.91E-7	0.3%	T1LPQs
9	Sum =	2.31E-7	0.3%	
	THP30	2.06E-7	0.3%	THKMPr
16	Sum =	1.22E-7	0.2%	
1	Sum =	8.98E-8	0.1%	
6	Sum =	6.82E-8	0.1%	
22	Sum =	5.44E-8	0.1%	
17	Sum =	4.05E-8	0.1%	
2	Sum =	2.19E-8	0.0%	
19	Sum =	4.31E-9	0.0%	
10	Sum =	1.62E-9	0.0%	

TABLE 3.4.1-11
SUMMARY OF RECOVERY BASIC EVENTS

Basic Event Name	Equip-ment	Human Error	Total	Description
REC-CONTAINMENT	2E-2	0	2E-2	RECOVER SEQUENCES CONTAINMENT HAS FAILED NO CORE DAMAGE
REC-1ES1:2	2E-3	9E-4	3E-3	1-ES-1.2 POST LOCA COOLDOWN AND DEPRESSURIZATION
REC-0OP21:6	6E-4	1E-3	2E-3	0-OP-21.6 MCR AND RELAY ROOM AIR CONDITIONING
REC-MMP-C-MR-2	2E-3	2E-1	3E-1	MMP-C-MR-2 TROUBLE SHOOTING & REPAIR MCR CHILLER UNITS
REC-SCREEN-TURNS	1E-1	0	1E-1	SW RESERVOIR TRAVELING SCREEN AUTO ROTATES & WASH
REC-1AP28	1E-1	2E-3	1E-1	1-AP-28 LOSS OF INSTRUMENT AIR
REC-2AP28	1E-1	2E-3	1E-1	2-AP-28 LOSS OF INSTRUMENT AIR
REC-1FRH:1-4	8E-3	3E-3	1E-2	1-FR-H.1 LOSS OF HEAT SINK STEP 4 MAIN FEEDWATER
REC-1OP14:1	1E-1	4E-3	1E-1	1-OP-14.1 RHR RECOVERY
REC-1ES1:4-1	1E-1	4E-3	1E-1	1-ES-1.4 HOT LEG RECIRC STEP 1 OPEN 1-SI-MOV-1890A & B
REC-B12AVE	1E-1	0	1E-1	TIME AVERAGED NON RECOVERY OF AC POWER IN 12 HOURS

TABLE 3.4.1-12 SENSITIVITY RESULTS RECOVERY BASIC EVENTS			
Basic Event Name	Mean	Description	CDF With REC=1 (% increase)
REC-CONTAINMENT	2E-2	RECOVER SEQUENCES CONTAINMENT HAS FAILED NO CORE DAMAGE	7.31E-5 (7.6)
REC-1ES1:2	3E-3	1-ES-1.2 POST LOCA COOLDOWN AND DEPRESSURIZATION	6.99E-5 (2.9)
REC-0OP21:6	2E-3	0-OP-21.6 MCR AND RELAY ROOM AIR CONDITIONING	6.99E-5 (3.0)
REC-MMP-C-MR-2	3E-1	MMP-C-MR-2 TROUBLE SHOOTING & REPAIR MCR CHILLER UNITS	6.85E-5 (0.9)
REC-SCREEN-TURNS	1E-1	SW RESERVOIR TRAVELING SCREEN AUTO ROTATES & WASH	9.45E-5 (39.4)
REC-1AP28	1E-1	1-AP-28 LOSS OF INSTRUMENT AIR	9.34E-5 (37.6)
REC-2AP28	1E-1	2-AP-28 LOSS OF INSTRUMENT AIR	7.02E-5 (3.4)
REC-1FRH:1-4	1E-2	1-FR-H.1 LOSS OF HEAT SINK STEP 4 MAIN FEEDWATER	1.31E-4 (93.1)
REC-1OP14:1	1E-1	1-OP-14.1 RHR RECOVERY	8.12E-5 (19.7)
REC-1ES1:4-1	1E-1	1-ES-1.4 HOT LEG RECIRC STEP 1 OPEN 1-SI-MOV-1890A & B	6.81E-5 (0.3)
REC-B12AVE	1E-1	TIME AVERAGED NON RECOVERY OF AC POWER IN 12 HOURS	8.19E-5 (20.7)

TABLE 3.4.3-1

COMPARISONS OF INITIATING EVENTS IN THE NORTH ANNA IPE AND SHUTDOWN DECAY HEAT REMOVAL ANALYSIS WESTINGHOUSE 3 LOOP PWR CASE STUDY

North Anna IPE	Case Study
Small LOCA	Small LOCA
Loss of Offsite Power	Loss of Offsite Power
Transients Resulting From Initial Loss of Power Conversion System	Transients Resulting from Initial Loss of Power Conversion System
Transients with Offsite Power and Power Conversion System Available	Transients with Offsite Power and Power Conversion System Available
Transients Resulting from Loss of AC and DC Bus	Transients Resulting from Loss of AC or DC Bus
Medium LOCA	
Large LOCA	
Steam Generator Tube Rupture	
Interfacing LOCA	
Loss of RC Pump Seal Injection and Cooling	
Loss of Emergency Switchgear Room Cooling	
Anticipated Transients without Scram.	

TABLE 3.4.3-2
COMPARISON OF FRONT LINE AND SUPPORT SYSTEMS
IN THE NORTH ANNA IPE AND SHUTDOWN DECAY HEAT REMOVAL ANALYSIS
WESTINGHOUSE 3 LOOP PWR CASE STUDY

North Anna IPE	Case Study
Frontline	
Auxiliary Feedwater (L)	Auxiliary Feedwater (L)
Power Conversion System (M)	Power Conversion System (M)
High Pressure Injection (D1) and Recirculation (H1)	High Pressure Injection (D1) and Recirculation (H1)
Low Pressure Injection (D2) and Recirculation (H2)	Low Pressure Injection (D2) and Recirculation (H2)
Pressurizer Safety and Relief Valves (P)	Pressurizer Safety and Relief Valves (P)
Secondary Safety and Relief Valves and Steam Dump (Y)	Secondary Safety and Relief Valves (Q)

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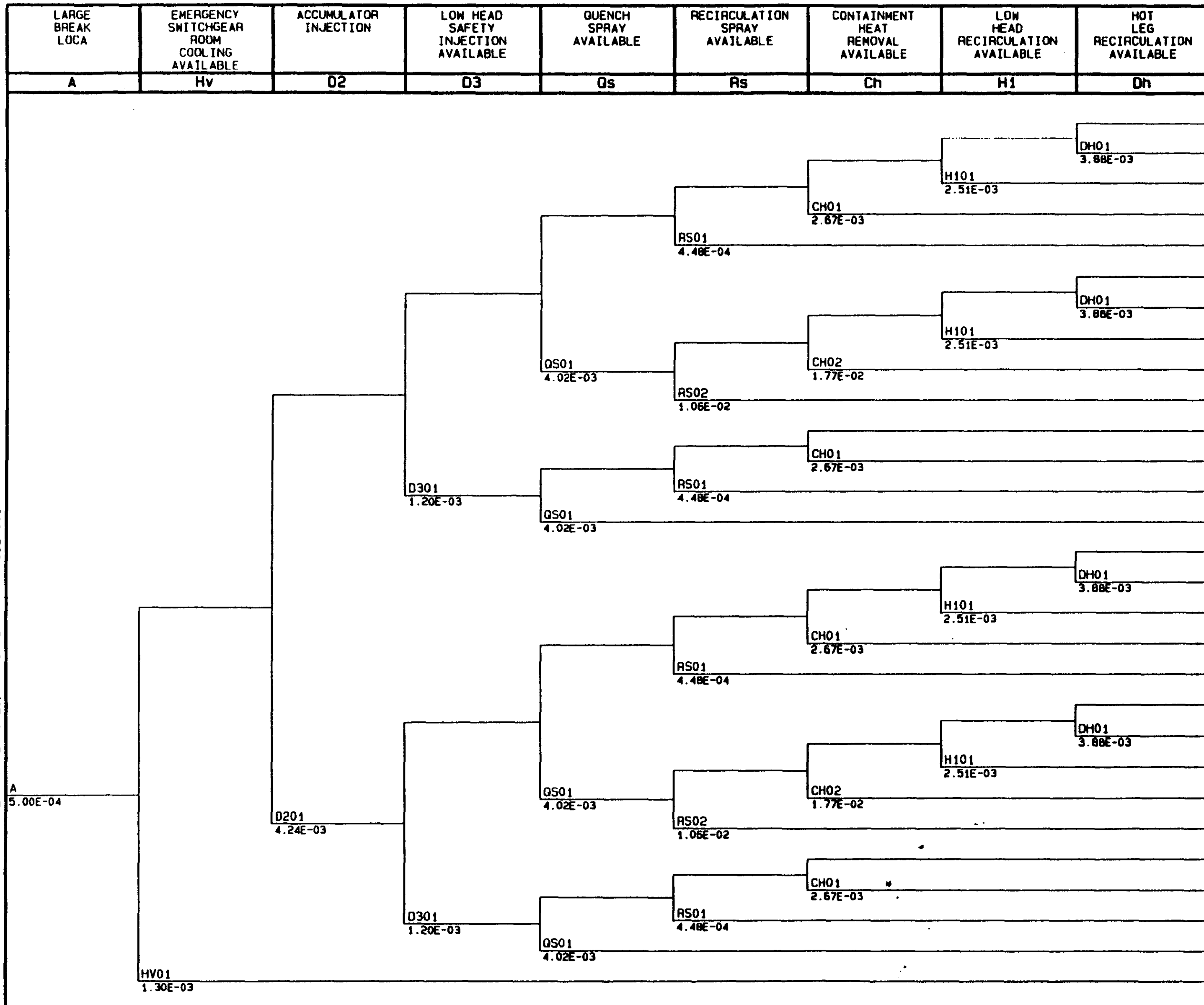
C:\NAPS10\ETREES\A.EVT 2:48:14pm 9-27-92 NUPRA 2.0 VPWR
Quantification Date: 9-27-92 2:45:25pm TOTAL CMF = 4.09E-006



SEQ #	SEQUENCE DESCRIPTOR	POS #	FREQUENCY
P01	A	OK	
P02	ADh	12	5.17E-07
P03	AH1	13	8.26E-07
P04	ACh	1	1.60E-08
P05	ARs	2	4.28E-09
P06	AQs	OK	
P07	AQsDh	12	1.69E-09
P08	AQsH1	13	1.90E-09
P09	AQsCh	1	7.54E-11
P10	AQsRs	2	1.21E-10
P11	AD3	13	5.88E-07
P12	AD3Ch	13	8.41E-10
P13	AD3Rs	13	2.24E-10
P14	AD3Qs	13	1.81E-09
P15	AD2	12	2.12E-06
P16	AD2Dh	12	2.07E-09
P17	AD2H1	13	2.73E-09
P18	AD2Ch	1	6.52E-11
P19	AD2Rs	2	1.57E-11
P20	AD2Qs	12	7.68E-09
P21	AD2QsDh	12	0.00E+00
P22	AD2QsH1	13	0.00E+00
P23	AD2QsCh	1	0.00E+00
P24	AD2QsRs	2	0.00E+00
P25	AD2D3	13	2.28E-09
P26	AD2D3Ch	13	0.00E+00
P27	AD2D3Rs	13	0.00E+00
P28	AD2D3Qs	13	0.00E+00
P29	AHv	TR	3.38E-07 T8

FIGURE 3.1-A

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

A: LARGE BREAK LOSS OF COOLANT ACCIDENT EVENT TREE

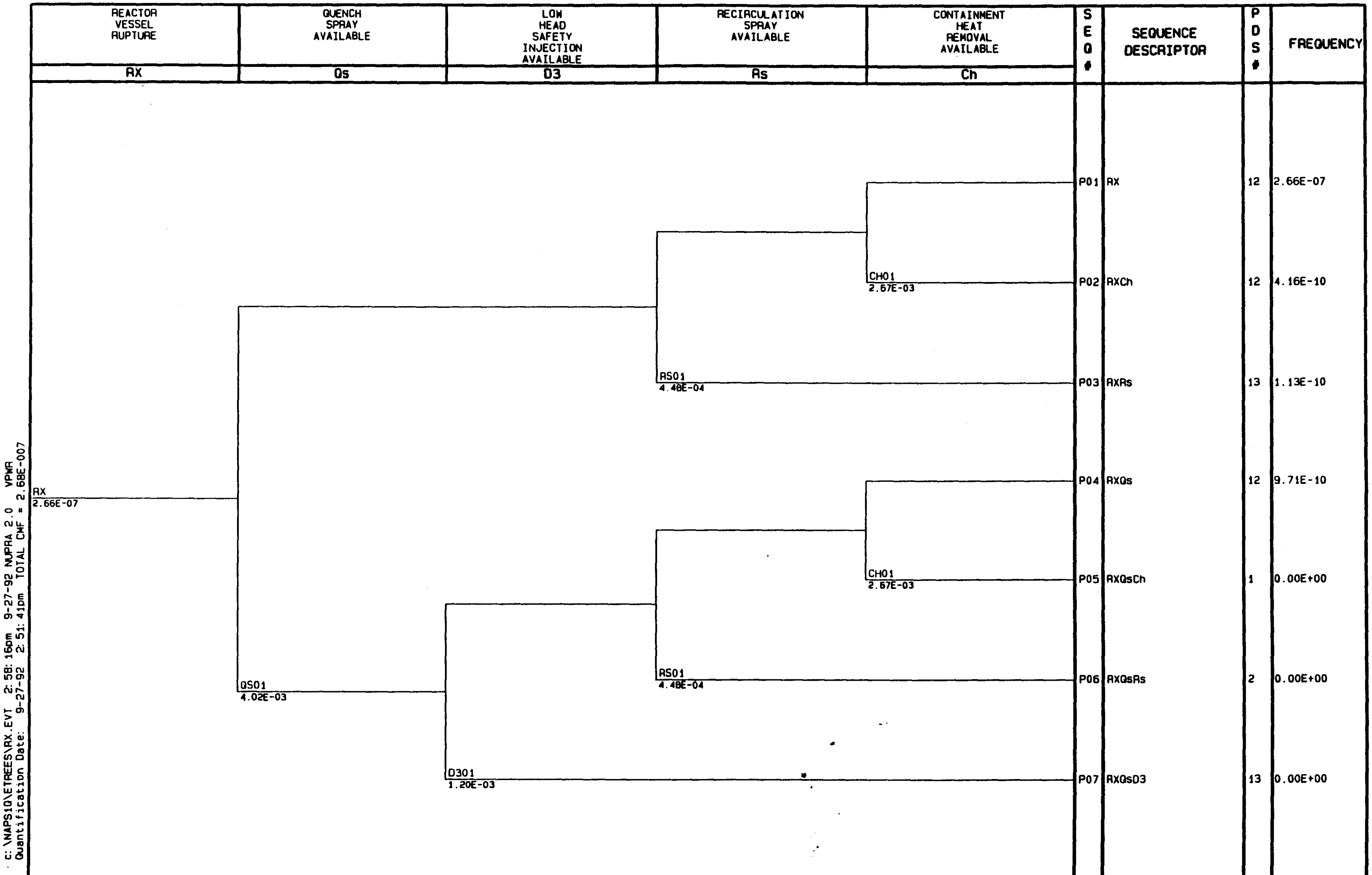
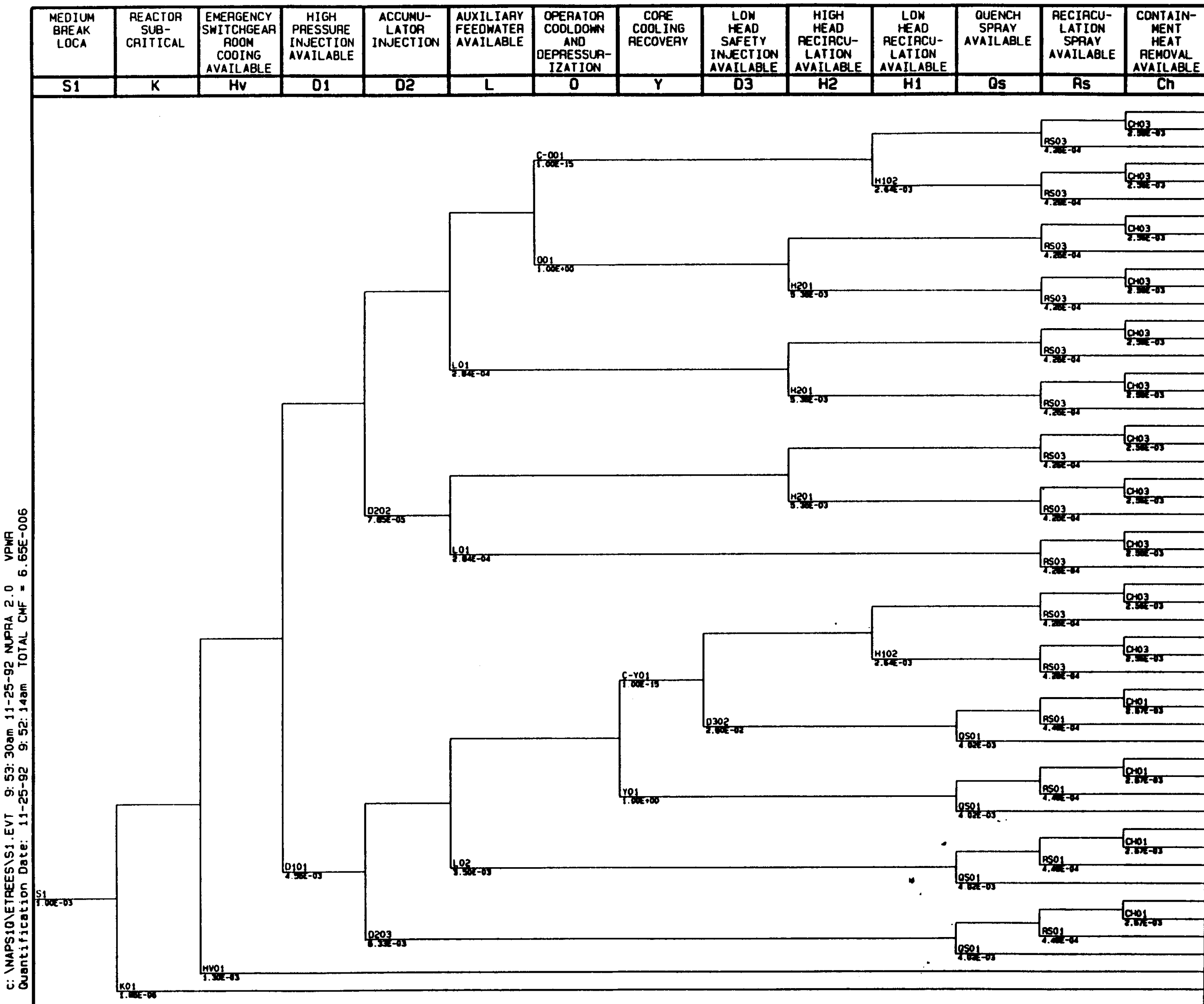


FIGURE 3.1-RX

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

RX: REACTOR VESSEL RUPTURE EVENT TREE

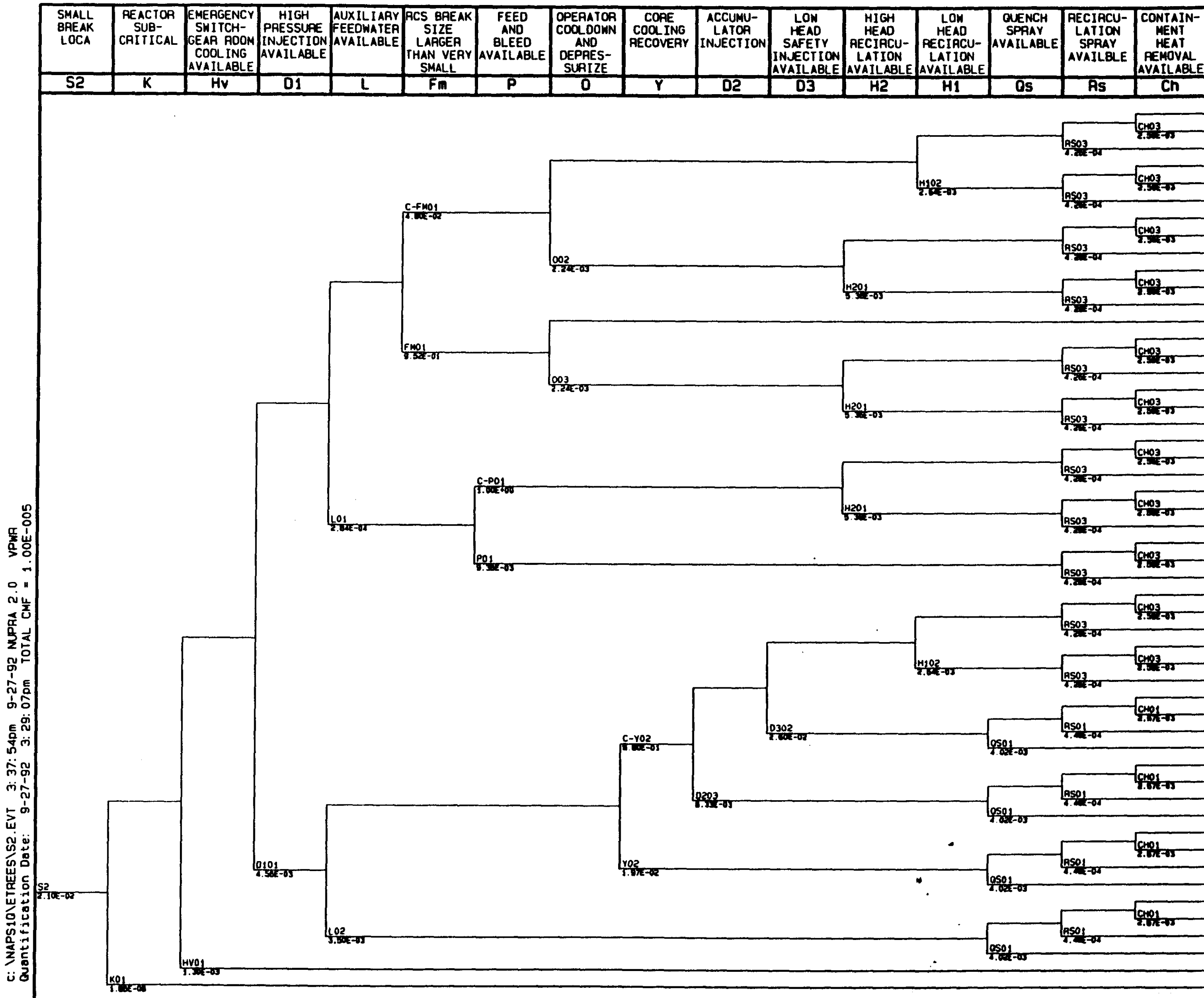


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	S1	OK	
P02	S1Ch	1	0.00E+00
P03	S1Rs	2	0.00E+00
P04	S1H1	21	0.00E+00
P05	S1H1Ch	22	0.00E+00
P06	S1H1Rs	23	0.00E+00
P07	S1O	OK	
P08	S1OCh	1	3.19E-08
P09	S1ORs	2	7.50E-09
P10	S1OH2	21	2.45E-06
P11	S1OH2Ch	22	3.21E-08
P12	S1OH2Rs	23	5.09E-08
P13	S1L	OK	
P14	S1LCh	1	5.46E-12
P15	S1LRs	2	1.31E-12
P16	S1LH2	21	3.37E-10
P17	S1LH2Ch	22	0.00E+00
P18	S1LH2Rs	23	0.00E+00
P19	S1D2	OK	
P20	S1D2Ch	1	0.00E+00
P21	S1D2Rs	2	0.00E+00
P22	S1D2H2	21	5.80E-11
P23	S1D2H2Ch	22	0.00E+00
P24	S1D2H2Rs	23	0.00E+00
P25	S1D2L	21	1.11E-11
P26	S1D2LCh	22	0.00E+00
P27	S1D2LRs	23	0.00E+00
P28	S1D1	OK	
P29	S1D1Ch	1	0.00E+00
P30	S1D1Rs	2	0.00E+00
P31	S1D1H1	21	0.00E+00
P32	S1D1H1Ch	22	0.00E+00
P33	S1D1H1Rs	23	0.00E+00
P34	S1D1D3	21	0.00E+00
P35	S1D1D3Ch	22	0.00E+00
P36	S1D1D3Rs	23	0.00E+00
P37	S1D1D3Qs	23	0.00E+00
P38	S1D1Y	21	4.04E-06
P39	S1D1YCh	22	6.19E-09
P40	S1D1YRs	23	1.67E-09
P41	S1D1YQs	23	1.53E-08
P42	S1D1L	21	1.36E-06
P43	S1D1LCh	22	0.00E+00
P44	S1D1LRs	23	0.00E+00
P45	S1D1LQs	23	1.14E-11
P46	S1D1D2	21	2.49E-06
P47	S1D1D2Ch	22	0.00E+00
P48	S1D1D2Rs	23	0.00E+00
P49	S1D1D2Qs	23	0.00E+00
P50	S1Hv	TR	8.78E-07T8
P51	S1K	TR	1.00E-09TH

FIGURE 3.1-S1

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

S1: MEDIUM BREAK LOSS OF COOLANT ACCIDENT EVENT TREE

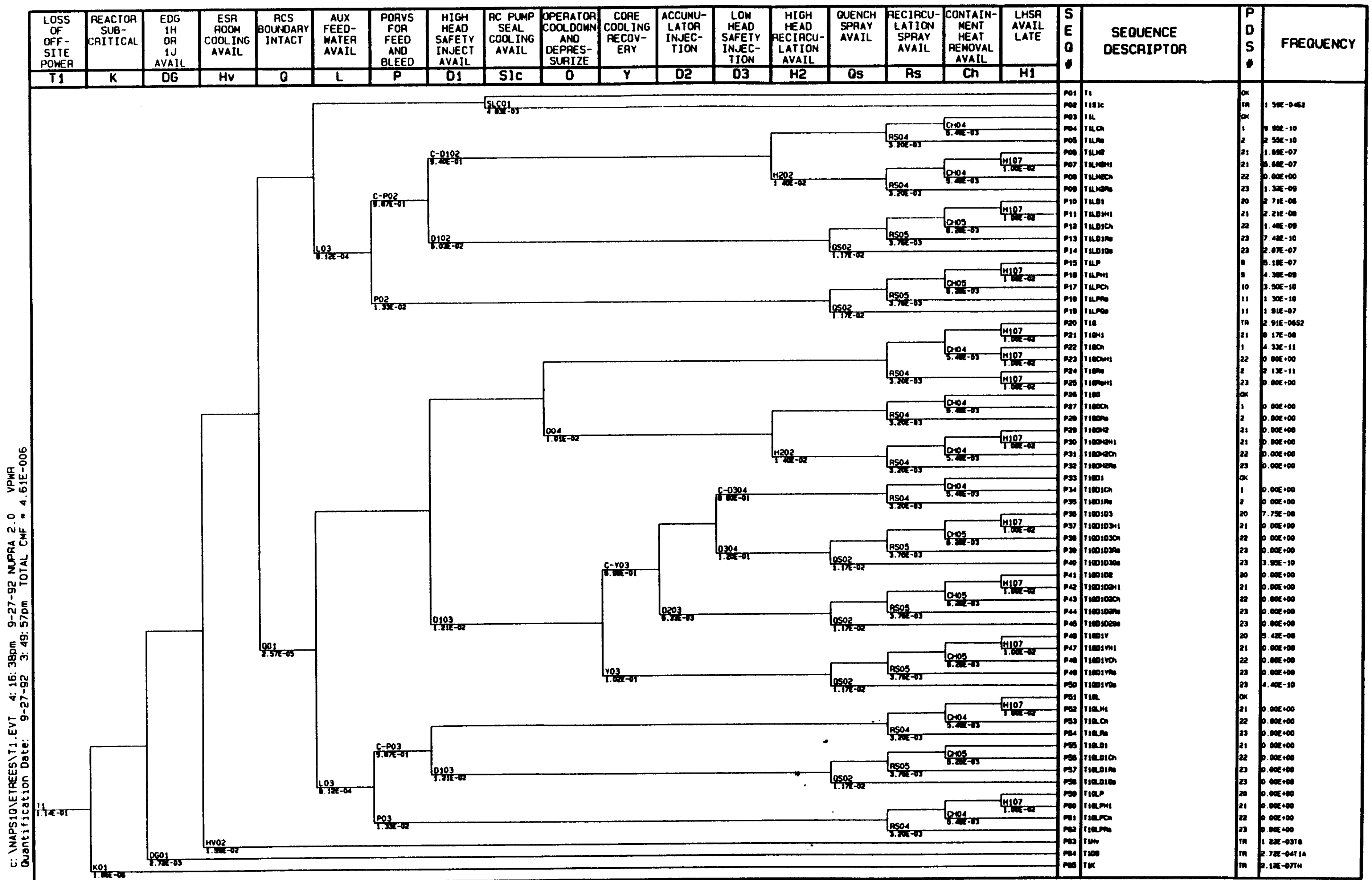


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	S2	OK	
P02	S2Ch	1	3.22E-08
P03	S2Rs	2	7.58E-09
P04	S2H1	21	2.45E-06
P05	S2H1Ch	22	2.44E-09
P06	S2H1Rs	23	5.13E-08
P07	S2O	OK	
P08	S2OCh	1	5.82E-11
P09	S2ORs	2	1.40E-11
P10	S2OH2	21	3.84E-09
P11	S2OH2Ch	22	0.00E+00
P12	S2OH2Rs	23	1.23E-10
P13	S2Fm	OK	
P14	S2FmO	OK	
P15	S2FmOCh	1	1.10E-09
P16	S2FmORs	2	3.50E-10
P17	S2FmOH2	21	9.02E-08
P18	S2FmOH2Ch	22	0.00E+00
P19	S2FmOH2Rs	23	2.85E-09
P20	S2L	OK	
P21	S2LCh	1	1.75E-10
P22	S2LRs	2	4.37E-11
P23	S2LH2	21	1.20E-08
P24	S2LH2Ch	22	0.00E+00
P25	S2LH2Rs	23	3.80E-10
P26	S2LP	21	5.45E-08
P27	S2LPCh	22	6.45E-11
P28	S2LPRs	23	1.34E-10
P29	S2D1	OK	
P30	S2D1Ch	1	2.52E-09
P31	S2D1Rs	2	5.99E-10
P32	S2D1H1	21	9.12E-08
P33	S2D1H1Ch	22	0.00E+00
P34	S2D1H1Rs	23	3.94E-09
P35	S2D1D3	21	5.15E-06
P36	S2D1D3Ch	22	5.38E-09
P37	S2D1D3Rs	23	2.02E-09
P38	S2D1D3Qs	23	3.28E-06
P39	S2D1D2	21	5.20E-07
P40	S2D1D2Ch	22	9.00E-11
P41	S2D1D2Rs	23	1.82E-10
P42	S2D1D2Qs	23	3.48E-10
P43	S2D1Y	21	1.19E-06
P44	S2D1YCh	22	1.28E-09
P45	S2D1YRs	23	4.41E-10
P46	S2D1YQs	23	9.83E-09
P47	S2D1L	21	3.27E-07
P48	S2D1LCh	22	6.29E-12
P49	S2D1LRs	23	7.28E-11
P50	S2D1LQs	23	1.30E-08
P51	S2Hv	TR	1.42E-05TB
P52	S2K	TR	3.91E-08TH

FIGURE 3.1-S2

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

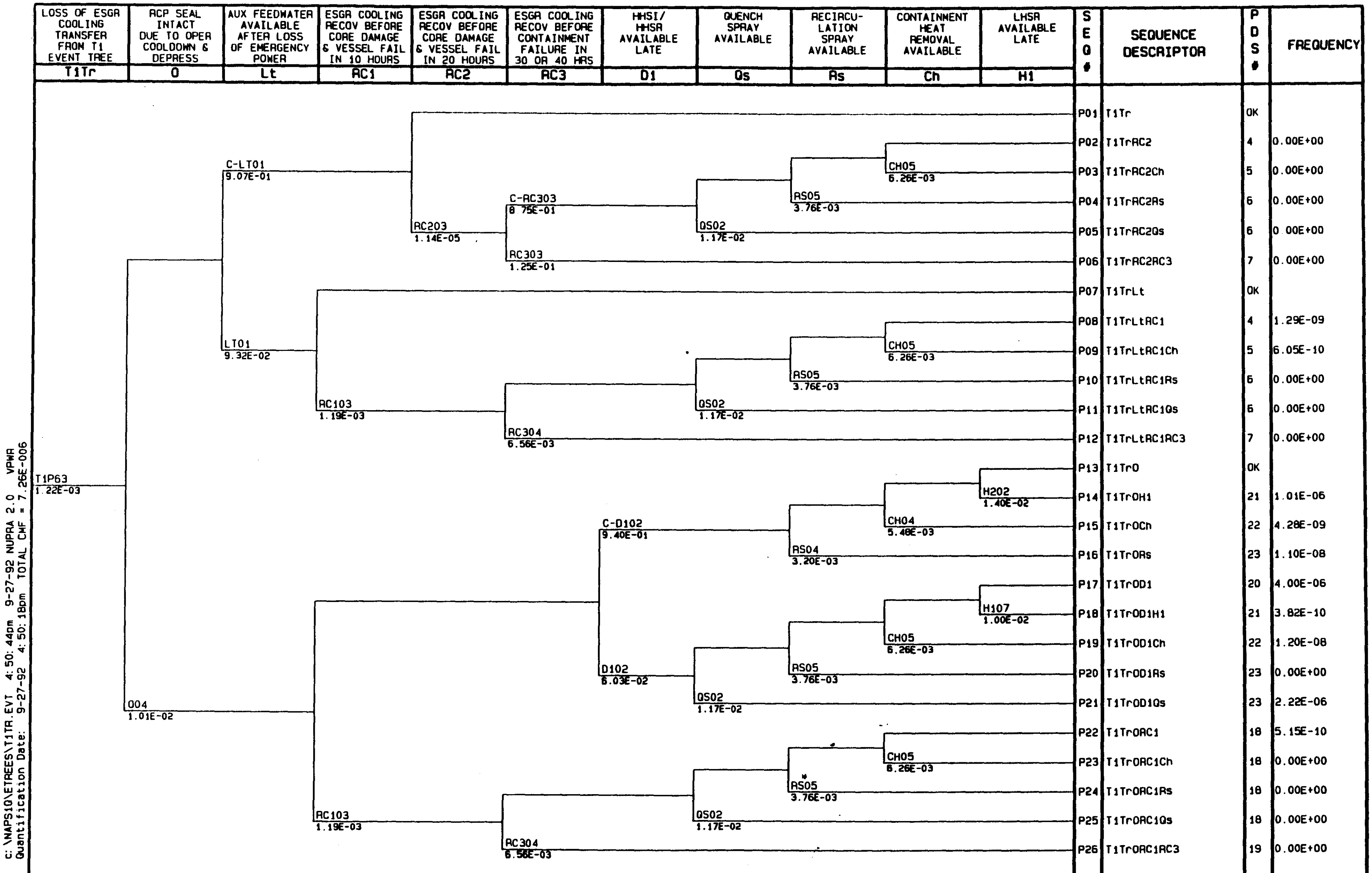
S2: SMALL BREAK LOSS OF COOLANT ACCIDENT EVENT TREE



NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T1: LOSS OF OFFSITE POWER EVENT TREE

FIGURE 3.1-T1

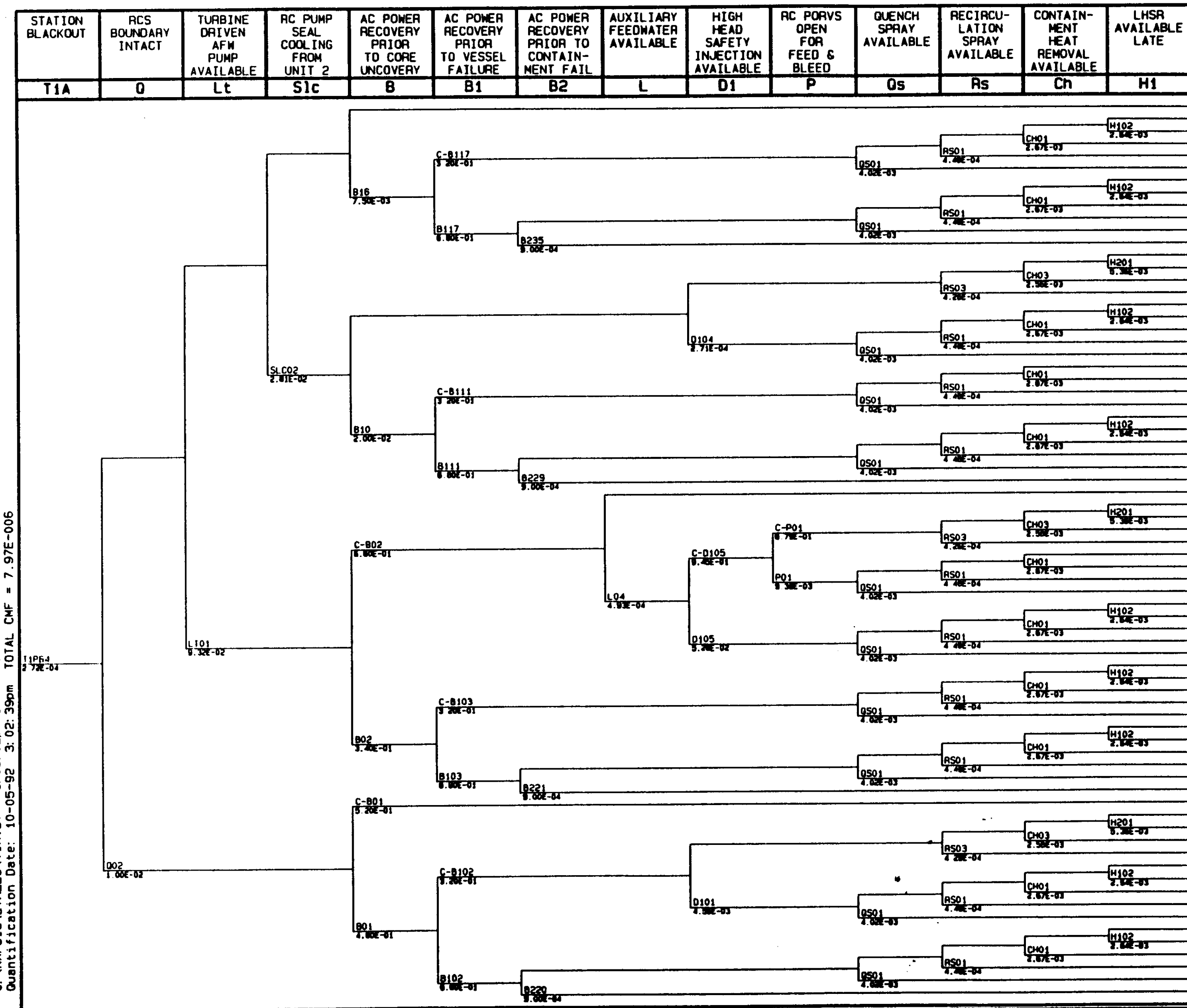


NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T1Tr: LOSS OF EMERGENCY SWITCHGEAR ROOM COOLING TRANSFER FROM T1 LOSS OF OFFSITE POWER EVENT TREE

FIGURE 3.1-T1Tr

c:\NAPS10\ETREES\T1A.EVT 3:09:02pm 10-05-92 NUPRA 2.0 VPWR
Quantification Date: 10-05-92 3:02:39pm TOTAL CMF = 7.97E-006



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T1A	OK	
P02	T1AB	3	8.51E-07
P03	T1ABH1	3	0.00E+00
P04	T1ABCh	3	0.00E+00
P05	T1ABRs	3	2.28E-10
P06	T1ABQs	3	2.28E-10
P07	T1ABB1	4	1.38E-06
P08	T1ABB1H1	4	3.58E-10
P09	T1ABB1Ch	5	0.00E+00
P10	T1ABB1Rs	6	4.94E-10
P11	T1ABB1Qs	6	1.43E-09
P12	T1ABB1B2	7	1.09E-09
P13	T1ASlc	OK	
P14	T1ASlcH1	15	3.18E-08
P15	T1ASlcCh	16	1.14E-08
P16	T1ASlcRs	17	3.22E-09
P17	T1ASlcD1	14	1.42E-09
P18	T1ASlcD1H1	15	0.00E+00
P19	T1ASlcD1Ch	16	0.00E+00
P20	T1ASlcD1Rs	17	0.00E+00
P21	T1ASlcD1Qs	17	0.00E+00
P22	T1ASlcB	15	4.88E-08
P23	T1ASlcBCh	16	0.00E+00
P24	T1ASlcBRs	17	1.50E-11
P25	T1ASlcBQs	17	1.55E-11
P26	T1ASlcBB1	18	1.04E-07
P27	T1ASlcBB1H1	18	0.00E+00
P28	T1ASlcBB1Ch	18	0.00E+00
P29	T1ASlcBB1Rs	18	3.61E-11
P30	T1ASlcBB1Qs	18	3.61E-11
P31	T1ASlcBB1B2	19	3.05E-11
P32	T1ALt	OK	
P33	T1ALtL	OK	
P34	T1ALtLH1	15	0.00E+00
P35	T1ALtLCh	16	0.00E+00
P36	T1ALtLRs	17	0.00E+00
P37	T1ALtLP	3	2.38E-10
P38	T1ALtLPCh	3	0.00E+00
P39	T1ALtLPRs	3	0.00E+00
P40	T1ALtLPQs	3	0.00E+00
P41	T1ALtLD1	3	5.41E-10
P42	T1ALtLD1H1	3	0.00E+00
P43	T1ALtLD1Ch	3	0.00E+00
P44	T1ALtLD1Rs	3	0.00E+00
P45	T1ALtLD1Qs	3	0.00E+00
P46	T1ALtB	3	1.41E-08
P47	T1ALtBH1	3	1.29E-09
P48	T1ALtBCh	3	1.78E-09
P49	T1ALtBRs	3	4.81E-10
P50	T1ALtBQs	3	3.27E-09
P51	T1ALtBB1	4	2.98E-06
P52	T1ALtBB1H1	4	3.87E-09
P53	T1ALtBB1Ch	5	4.08E-09
P54	T1ALtBB1Rs	6	1.11E-09
P55	T1ALtBB1Qs	6	8.08E-09
P56	T1ALtBB1B2	7	2.38E-08
P57	T1AQ	TR	1.41E-0652
P58	T1AQB	14	4.17E-07
P59	T1AQBH1	15	3.11E-10
P60	T1AQBCh	16	5.50E-10
P61	T1AQBRs	17	1.49E-10
P62	T1AQBD1	14	1.29E-09
P63	T1AQBD1H1	15	0.00E+00
P64	T1AQBD1Ch	16	0.00E+00
P65	T1AQBD1Rs	17	0.00E+00
P66	T1AQBD1Qs	17	0.00E+00
P67	T1AQBB1	18	8.88E-07
P68	T1AQBB1H1	18	8.93E-10
P69	T1AQBB1Ch	18	1.28E-09
P70	T1AQBB1Rs	18	3.18E-10
P71	T1AQBB1Qs	18	2.37E-08
P72	T1AQBB1B2	18	7.58E-10

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T1A: STATION BLACKOUT EVENT TREE
TRANSFER FROM T1 LOSS OF OFFSITE POWER

FIGURE 3.1-T1A

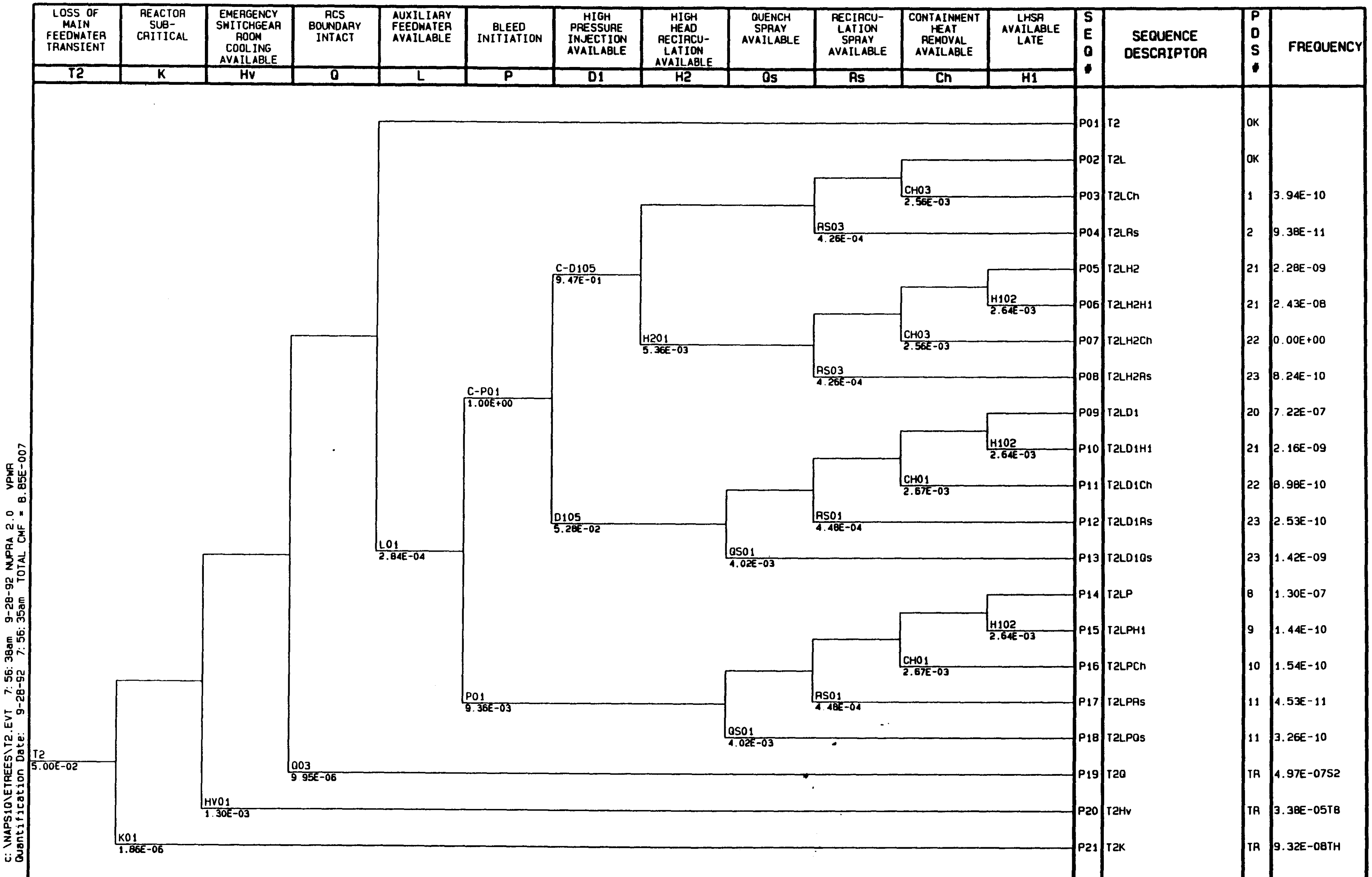
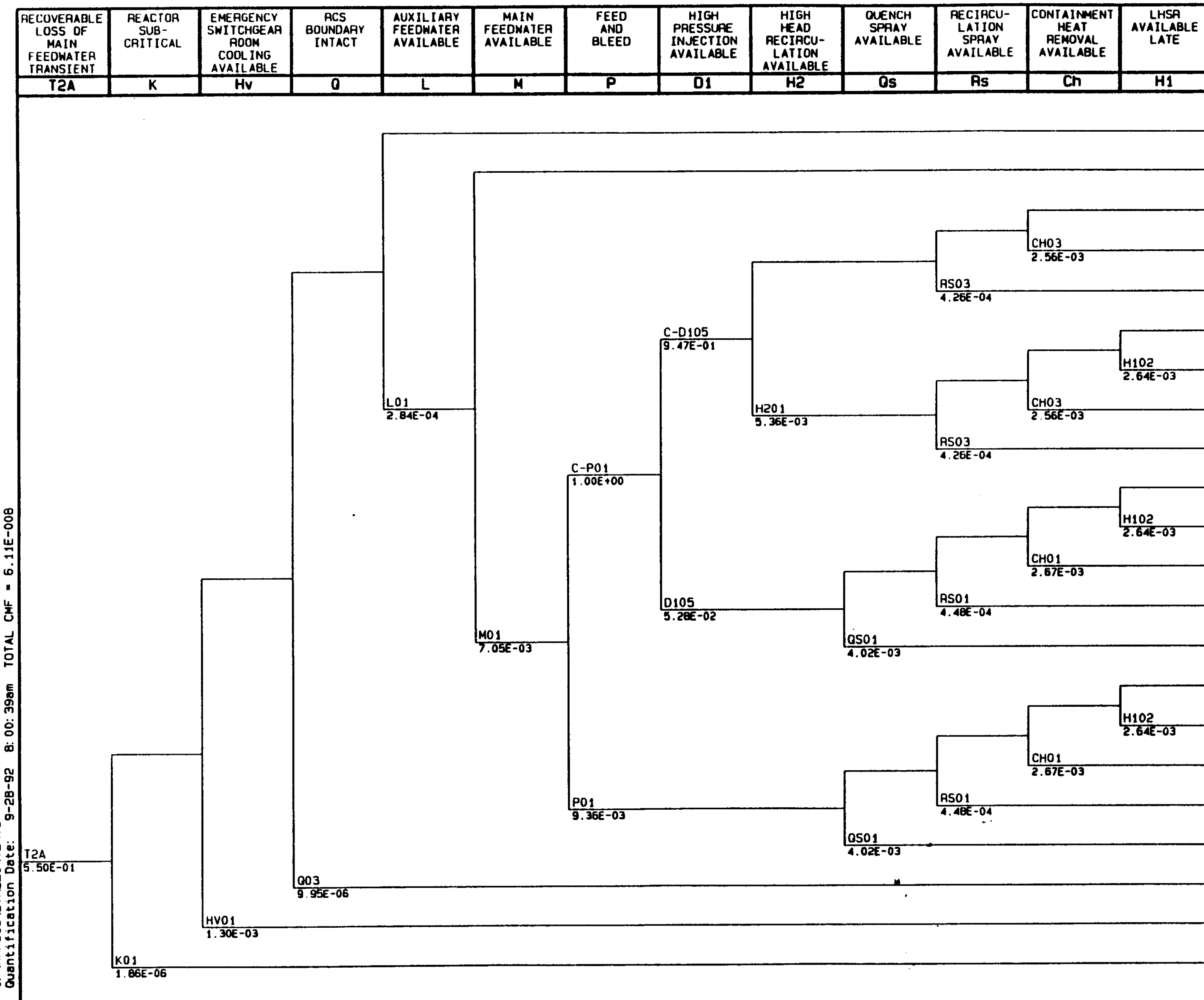


FIGURE 3.1-T2

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T2: LOSS OF MAIN FEEDWATER EVENT TREE

C:\NAPS10\ETREES\T2A.EVT 8:00:40am 9-28-92 NUPRA 2.0 VPWR
Quantification Date: 9-28-92 8:00:39am TOTAL CMF = 6.11E-008

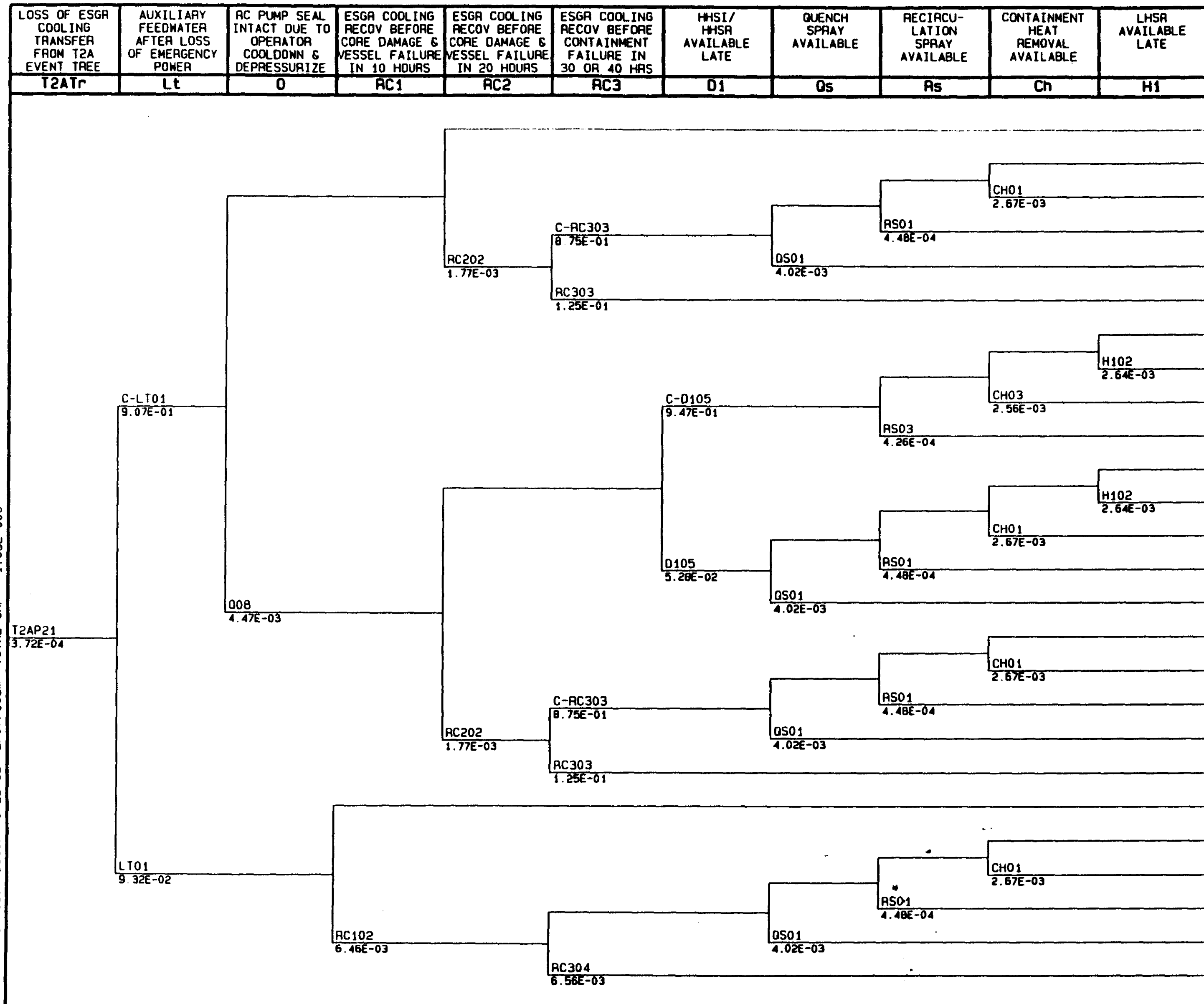


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T2A	OK	
P02	T2AL	OK	
P03	T2ALM	OK	
P04	T2ALMCh	1	0.00E+00
P05	T2ALMRs	2	3.37E-12
P06	T2ALMH2	21	1.90E-11
P07	T2ALMH2H1	21	4.69E-10
P08	T2ALMH2Ch	22	0.00E+00
P09	T2ALMH2Rs	23	2.84E-09
P10	T2ALMD1	20	4.55E-08
P11	T2ALMD1H1	21	0.00E+00
P12	T2ALMD1Ch	22	0.00E+00
P13	T2ALMD1Rs	23	0.00E+00
P14	T2ALMD1Qs	23	0.00E+00
P15	T2ALMP	8	1.23E-08
P16	T2ALMPH1	9	0.00E+00
P17	T2ALMPCh	10	0.00E+00
P18	T2ALMPRs	11	0.00E+00
P19	T2ALMPQs	11	0.00E+00
P20	T2AQ	TR	5.47E-06S2
P21	T2AHv	TR	3.72E-04T8
P22	T2AK	TR	1.03E-06TH

FIGURE 3.1-T2A

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T2A: RECOVERABLE LOSS OF MAIN FEEDWATER EVENT TREE

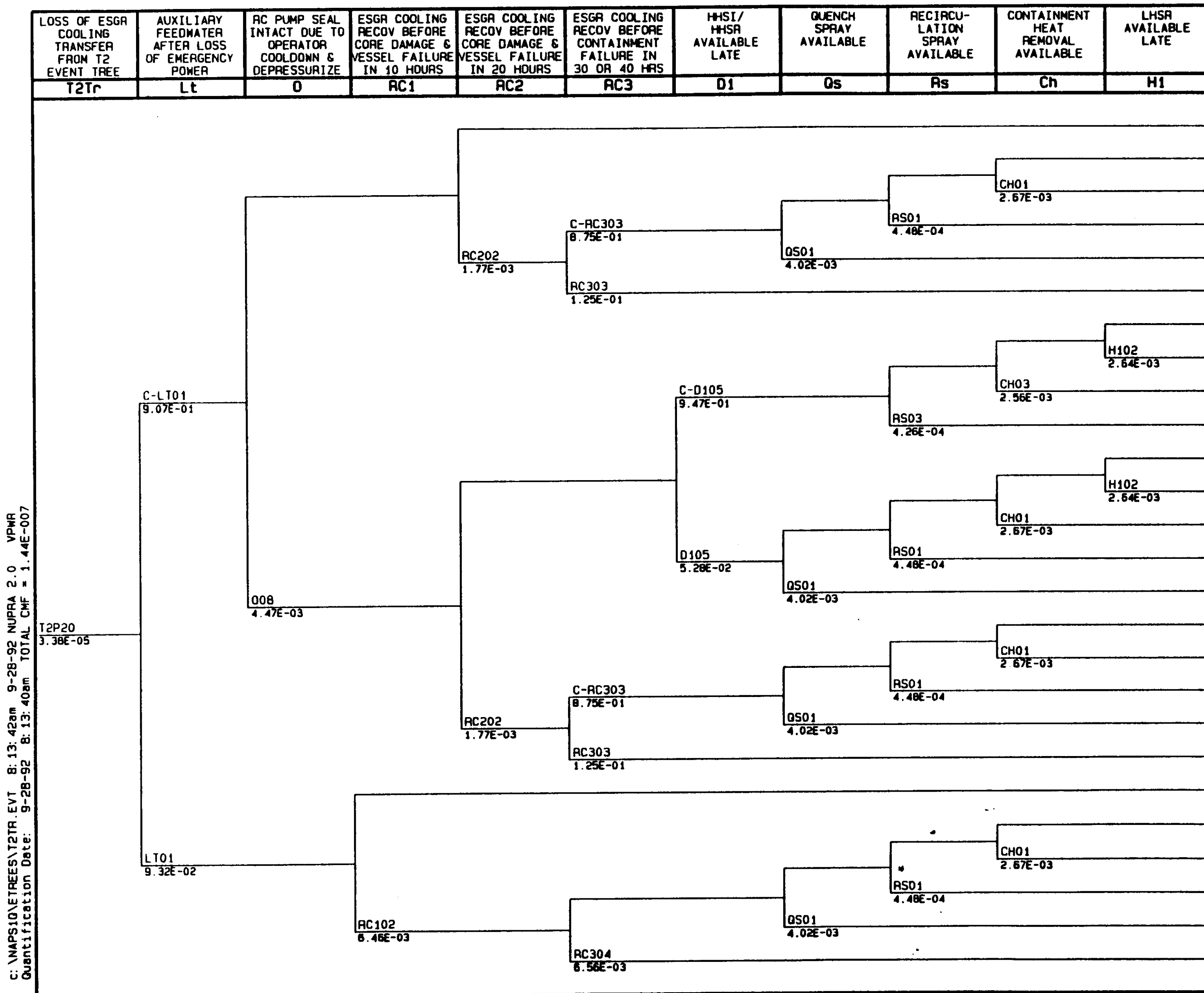


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T2ATr	OK	
P02	T2ATrRC2	4	2.19E-08
P03	T2ATrRC2Ch	5	6.40E-07
P04	T2ATrRC2Rs	6	2.40E-10
P05	T2ATrRC2Qs	6	1.56E-09
P06	T2ATrRC2RC3	7	1.15E-07
P07	T2ATrO	OK	
P08	T2ATrOH1	15	2.71E-08
P09	T2ATrOCh	16	1.92E-08
P10	T2ATrORs	17	5.15E-09
P11	T2ATrOD1	14	6.78E-07
P12	T2ATrOD1H1	15	2.51E-09
P13	T2ATrOD1Ch	16	9.63E-10
P14	T2ATrOD1Rs	17	2.48E-10
P15	T2ATrOD1Qs	17	2.07E-09
P16	T2ATrORC2	18	5.89E-09
P17	T2ATrORC2Ch	18	2.40E-09
P18	T2ATrORC2Rs	18	0.00E+00
P19	T2ATrORC2Qs	18	0.00E+00
P20	T2ATrORC2RC3	19	8.18E-10
P21	T2ATrLt	OK	
P22	T2ATrLtRC1	4	4.75E-08
P23	T2ATrLtRC1Ch	5	7.51E-08
P24	T2ATrLtRC1Rs	6	2.70E-11
P25	T2ATrLtRC1Qs	6	4.08E-11
P26	T2ATrLtRC1RC3	7	3.83E-10

FIGURE 3.1-T2ATr

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T2ATr: LOSS OF EMERGENCY SWITCHGEAR ROOM COOLING TRANSFER FROM T2A RECOVERABLE LOSS OF MAIN FW EVENT TREE



SEQ #	SEQUENCE DESCRIPTOR	POS #	FREQUENCY
P01	T2Tr	OK	
P02	T2TrRC2	4	1.29E-09
P03	T2TrRC2Ch	5	5.82E-08
P04	T2TrRC2Rs	6	2.15E-11
P05	T2TrRC2Qs	6	1.42E-10
P06	T2TrRC2RC3	7	8.32E-09
P07	T2TrO	OK	
P08	T2TrOH1	15	2.45E-09
P09	T2TrOCh	16	1.74E-09
P10	T2TrORs	17	4.67E-10
P11	T2TrOD1	14	6.06E-08
P12	T2TrOD1H1	15	2.28E-10
P13	T2TrOD1Ch	16	8.75E-11
P14	T2TrOD1Rs	17	2.10E-11
P15	T2TrOD1Qs	17	1.87E-10
P16	T2TrORC2	18	5.31E-10
P17	T2TrORC2Ch	18	2.17E-10
P18	T2TrORC2Rs	18	0.00E+00
P19	T2TrORC2Qs	18	0.00E+00
P20	T2TrORC2RC3	19	7.43E-11
P21	T2TrLt	OK	
P22	T2TrLtRC1	4	2.91E-09
P23	T2TrLtRC1Ch	5	6.82E-09
P24	T2TrLtRC1Rs	6	0.00E+00
P25	T2TrLtRC1Qs	6	0.00E+00
P26	T2TrLtRC1RC3	7	3.35E-11

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T2Tr: LOSS OF EMERGENCY SWITCHGEAR ROOM COOLING TRANSFER FROM T2 LOSS OF MAIN FEEDWATER EVENT TREE

FIGURE 3.1-T2Tr

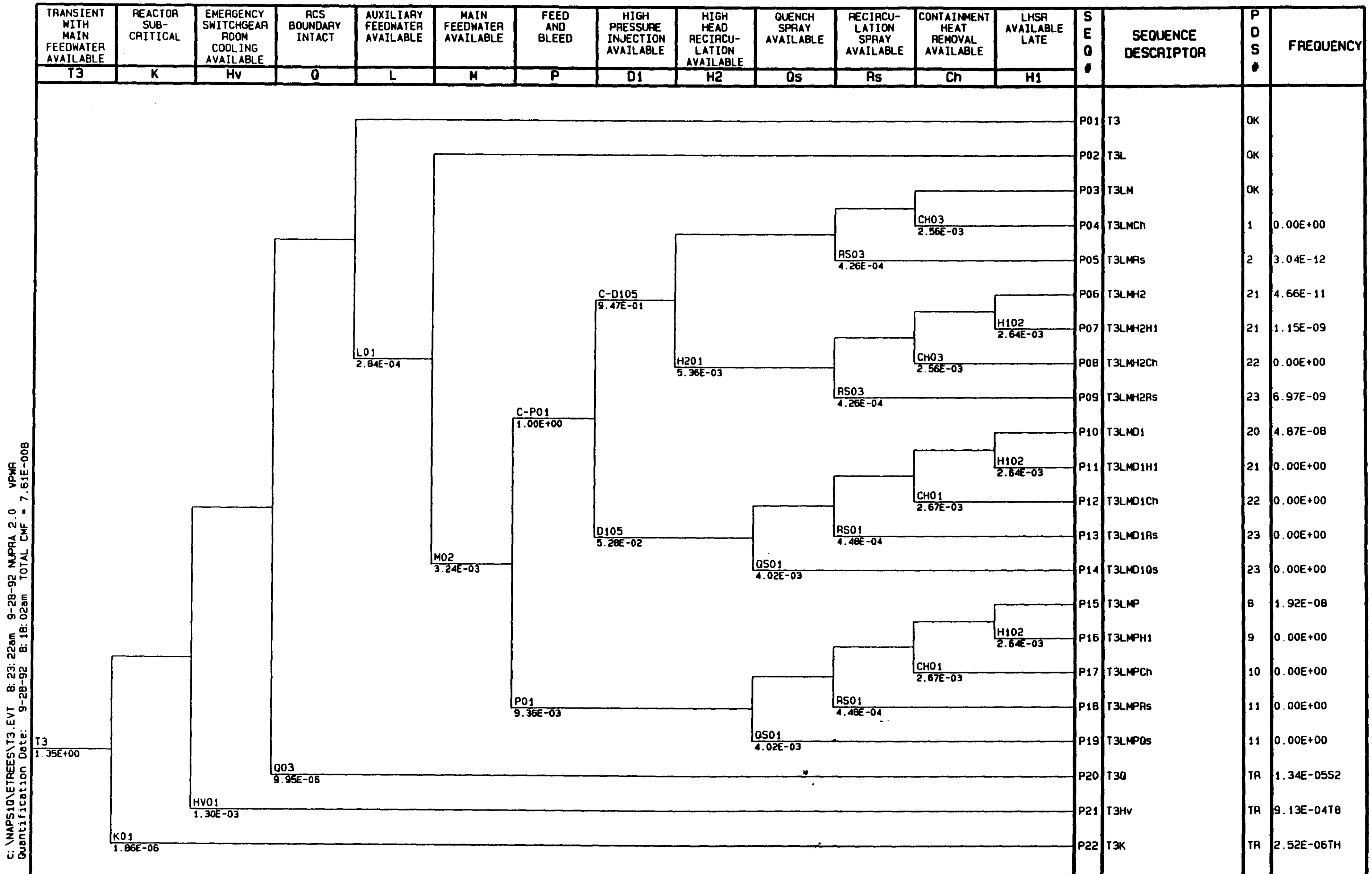
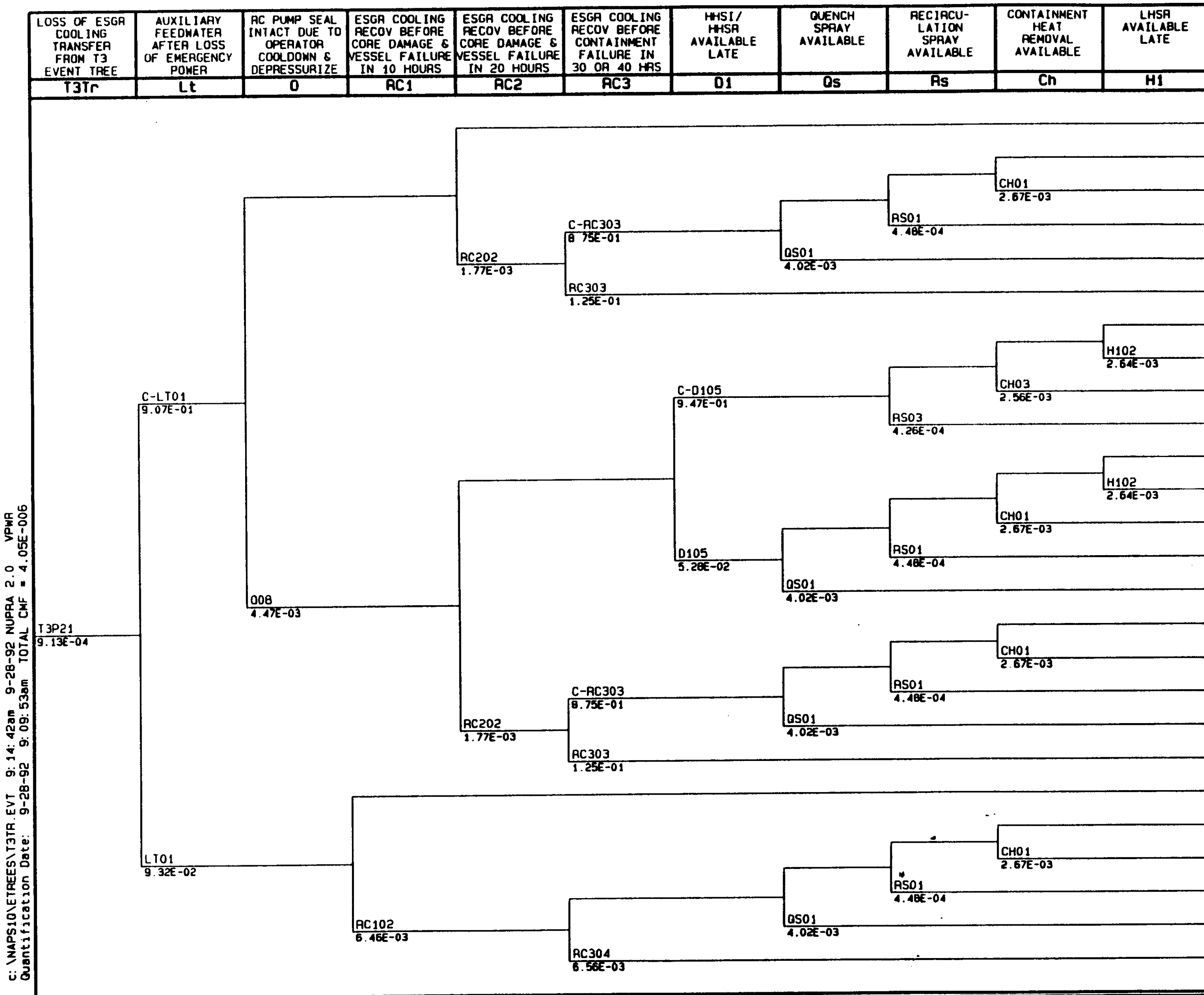


FIGURE 3.1-T3

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T3: TRANSIENT WITH MAIN FEEDWATER AVAILABLE EVENT TREE

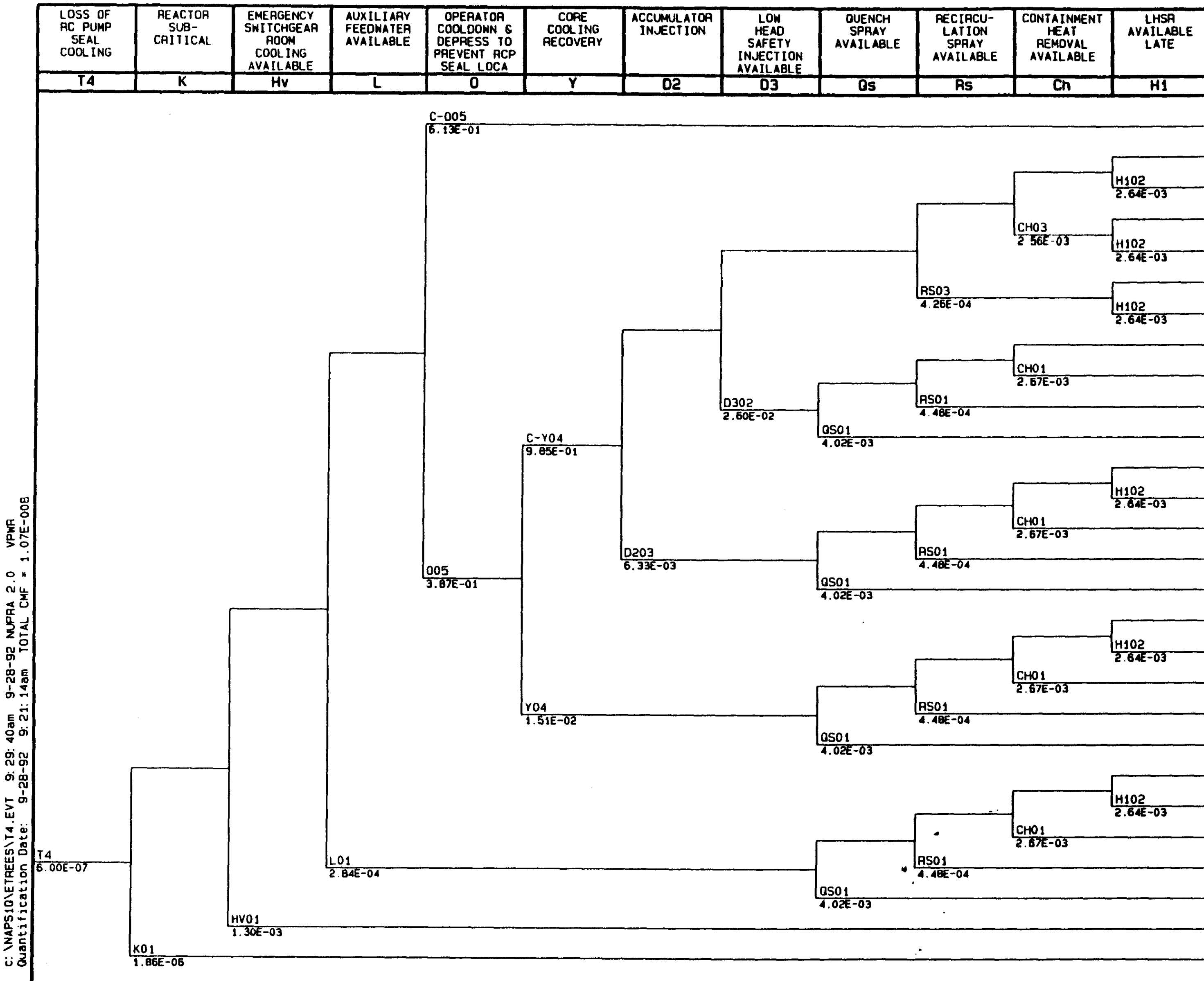


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T3Tr	OK	
P02	T3TrRC2	4	5.83E-08
P03	T3TrRC2Ch	5	1.57E-06
P04	T3TrRC2Rs	6	5.89E-10
P05	T3TrRC2Qs	6	3.84E-09
P06	T3TrRC2RC3	7	2.83E-07
P07	T3TrO	OK	
P08	T3TrOH1	15	6.67E-08
P09	T3TrOCh	16	4.70E-08
P10	T3TrORs	17	1.27E-08
P11	T3TrOD1	14	1.67E-06
P12	T3TrOD1H1	15	6.16E-09
P13	T3TrOD1Ch	16	2.36E-09
P14	T3TrOD1Rs	17	6.08E-10
P15	T3TrOD1Qs	17	5.09E-09
P16	T3TrORC2	18	1.44E-08
P17	T3TrORC2Ch	18	5.90E-09
P18	T3TrORC2Rs	18	0.00E+00
P19	T3TrORC2Qs	18	0.00E+00
P20	T3TrORC2RC3	19	1.76E-09
P21	T3TrLt	OK	
P22	T3TrLtRC1	4	1.21E-07
P23	T3TrLtRC1Ch	5	1.84E-07
P24	T3TrLtRC1Rs	6	6.74E-11
P25	T3TrLtRC1Qs	6	1.35E-10
P26	T3TrLtRC1RC3	7	9.96E-10

FIGURE 3.1-T3Tr

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T3Tr: LOSS OF EMERGENCY SWITCHGEAR ROOM COOLING TRANSFER FROM T3 TRANSIENT WITH MFW AVAILABLE EVENT TREE

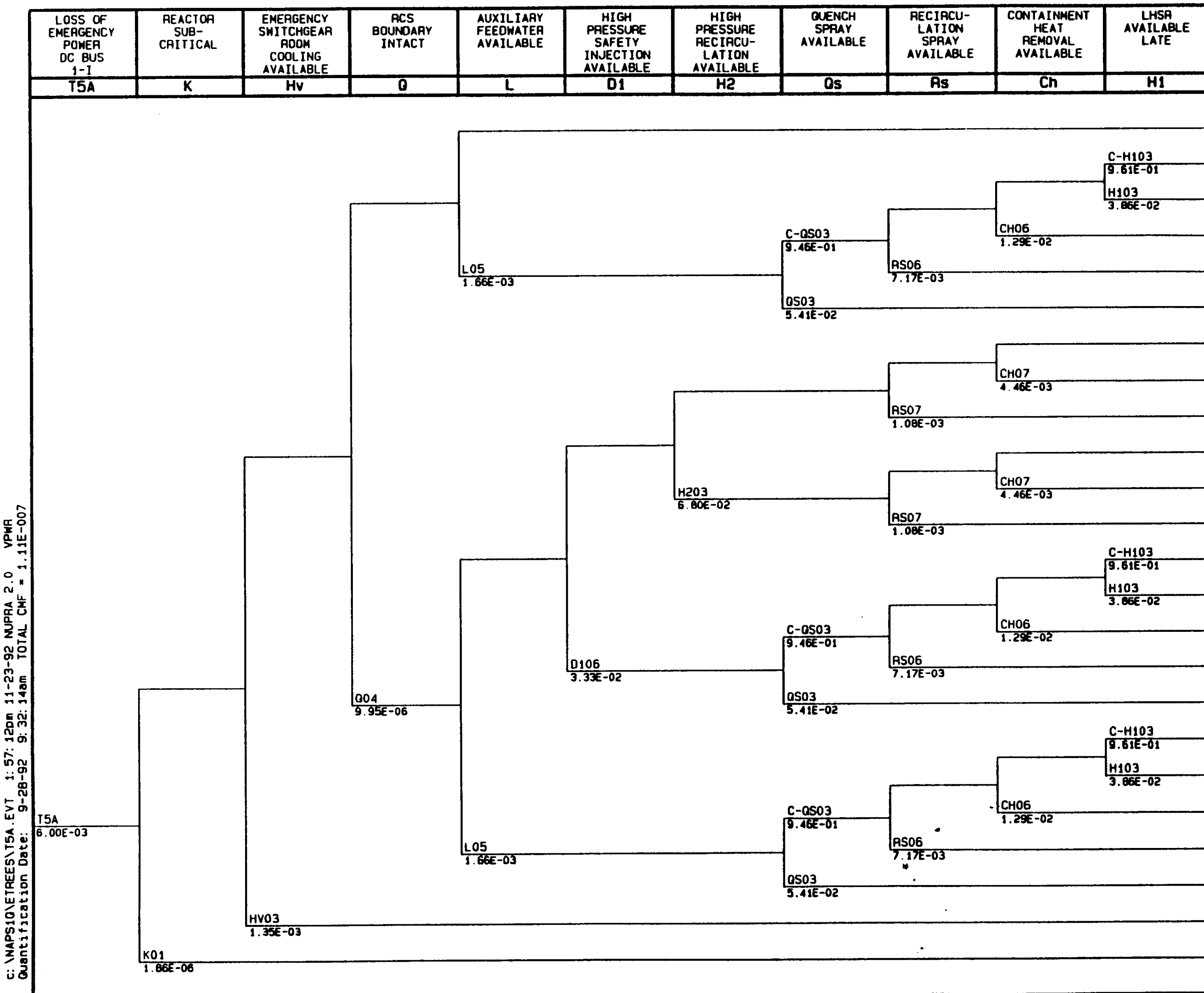


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T4	OK	
P02	T4O	OK	
P03	T4OH1	21	2.12E-10
P04	T4OCh	1	7.13E-12
P05	T4OChH1	22	0.00E+00
P06	T4ORs	2	1.71E-12
P07	T4ORsH1	23	1.14E-11
P08	T4OD3	21	5.84E-09
P09	T4OD3Ch	22	0.00E+00
P10	T4OD3Rs	23	0.00E+00
P11	T4OD3Qs	23	0.00E+00
P12	T4OD2	20	1.45E-09
P13	T4OD2H1	21	0.00E+00
P14	T4OD2Ch	22	0.00E+00
P15	T4OD2Rs	23	0.00E+00
P16	T4OD2Qs	23	0.00E+00
P17	T4OY	20	3.06E-09
P18	T4OYH1	21	0.00E+00
P19	T4OYCh	22	0.00E+00
P20	T4OYRs	23	0.00E+00
P21	T4OYQs	23	0.00E+00
P22	T4L	8	1.43E-10
P23	T4LH1	9	0.00E+00
P24	T4LCh	10	0.00E+00
P25	T4LRs	11	0.00E+00
P26	T4LQs	11	0.00E+00
P27	T4Hv	TR	2.82E-10TR
P28	T4K	TR	0.00E+00TR

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T4: LOSS OF REACTOR COOLANT PUMP SEAL COOLING EVENT TREE

FIGURE 3.1-T4

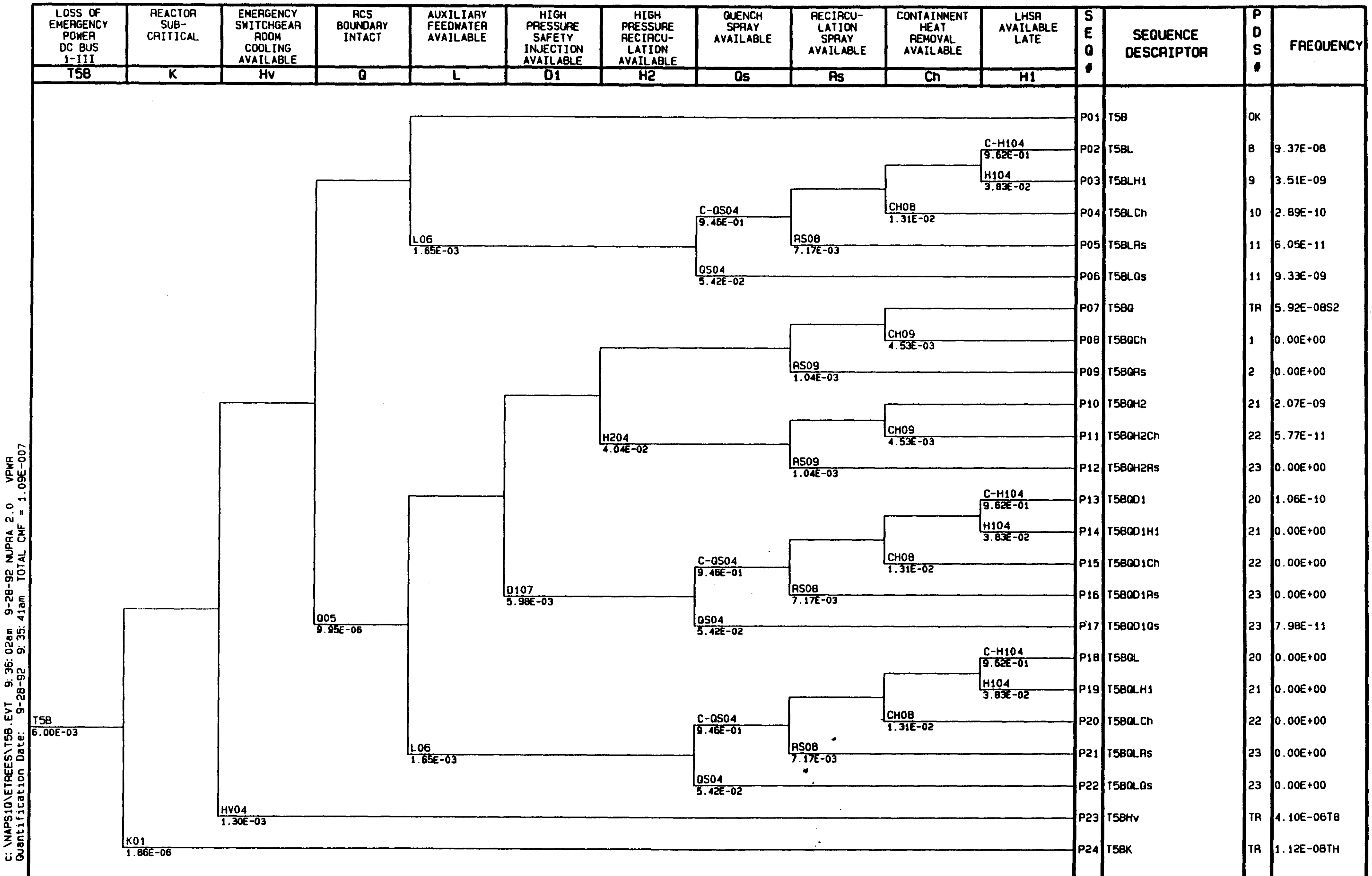


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T5A	OK	
P02	T5AL	8	9.36E-08
P03	T5ALH1	9	3.56E-09
P04	T5ALCh	10	2.89E-10
P05	T5ALRs	11	6.05E-11
P06	T5ALQs	11	9.63E-09
P07	T5AQ	TR	5.92E-08S2
P08	T5AQCh	1	0.00E+00
P09	T5AQRs	2	0.00E+00
P10	T5AQH2	21	1.98E-09
P11	T5AQH2Ch	22	1.76E-10
P12	T5AQH2Rs	23	0.00E+00
P13	T5AQD1	20	1.53E-09
P14	T5AQD1H1	21	0.00E+00
P15	T5AQD1Ch	22	0.00E+00
P16	T5AQD1Rs	23	0.00E+00
P17	T5AQD1Qs	23	1.42E-10
P18	T5AQL	20	0.00E+00
P19	T5AQLH1	21	0.00E+00
P20	T5AQLCh	22	0.00E+00
P21	T5AQLRs	23	0.00E+00
P22	T5AQLQs	23	0.00E+00
P23	T5AHv	TR	4.36E-06T8
P24	T5AK	TR	1.12E-08TH

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T5A: LOSS OF EMERGENCY POWER DC BUS 1-I EVENT TREE

FIGURE 3.1-T5A



NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T5B: LOSS OF EMERGENCY POWER DC BUS 1-III EVENT TREE

FIGURE 3.1-T5B

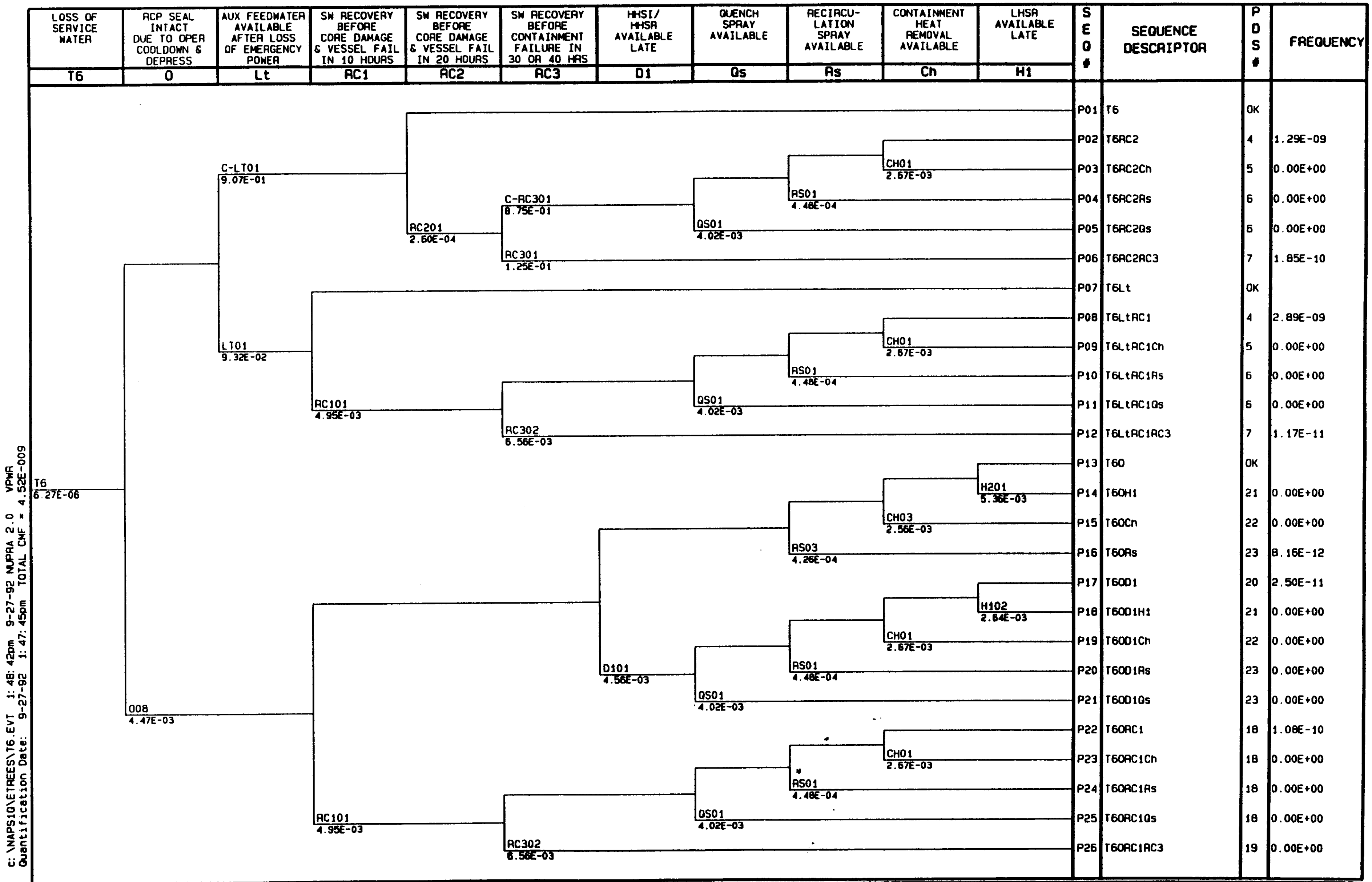


FIGURE 3.1-T6

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T6: LOSS OF SERVICE WATER EVENT TREE

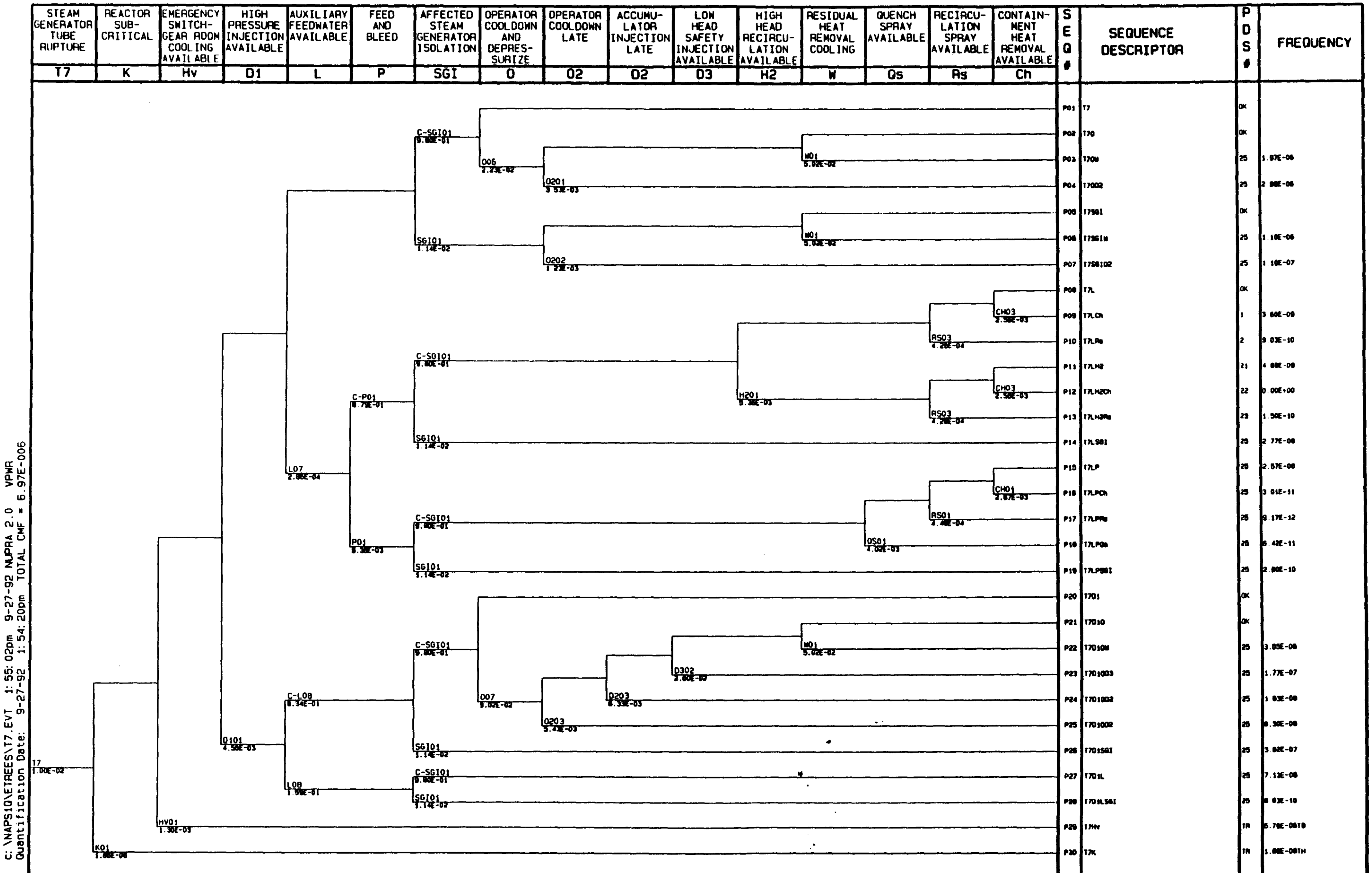


FIGURE 3.1-T7

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T7: STEAM GENERATOR TUBE RUPTURE EVENT TREE

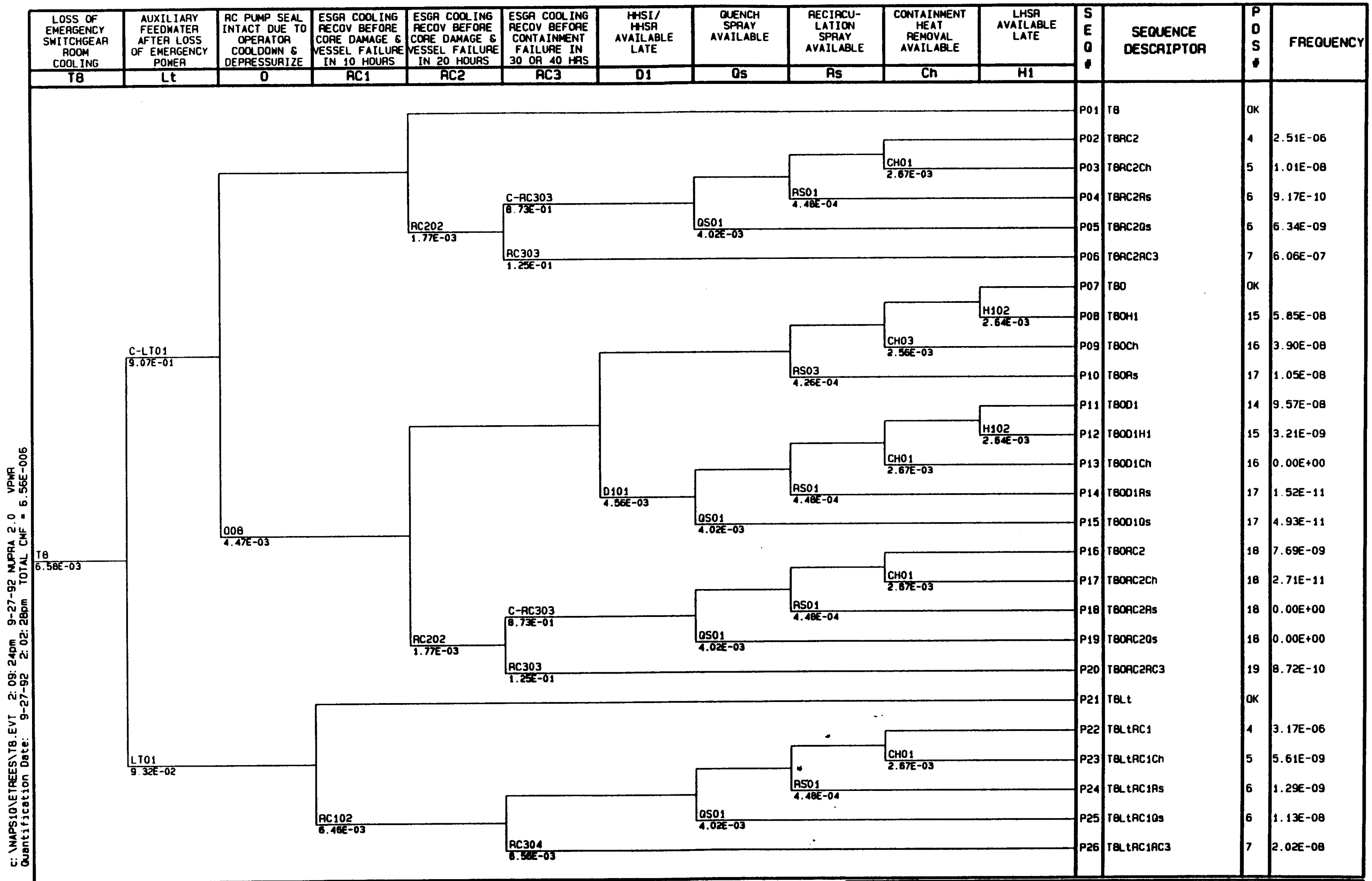
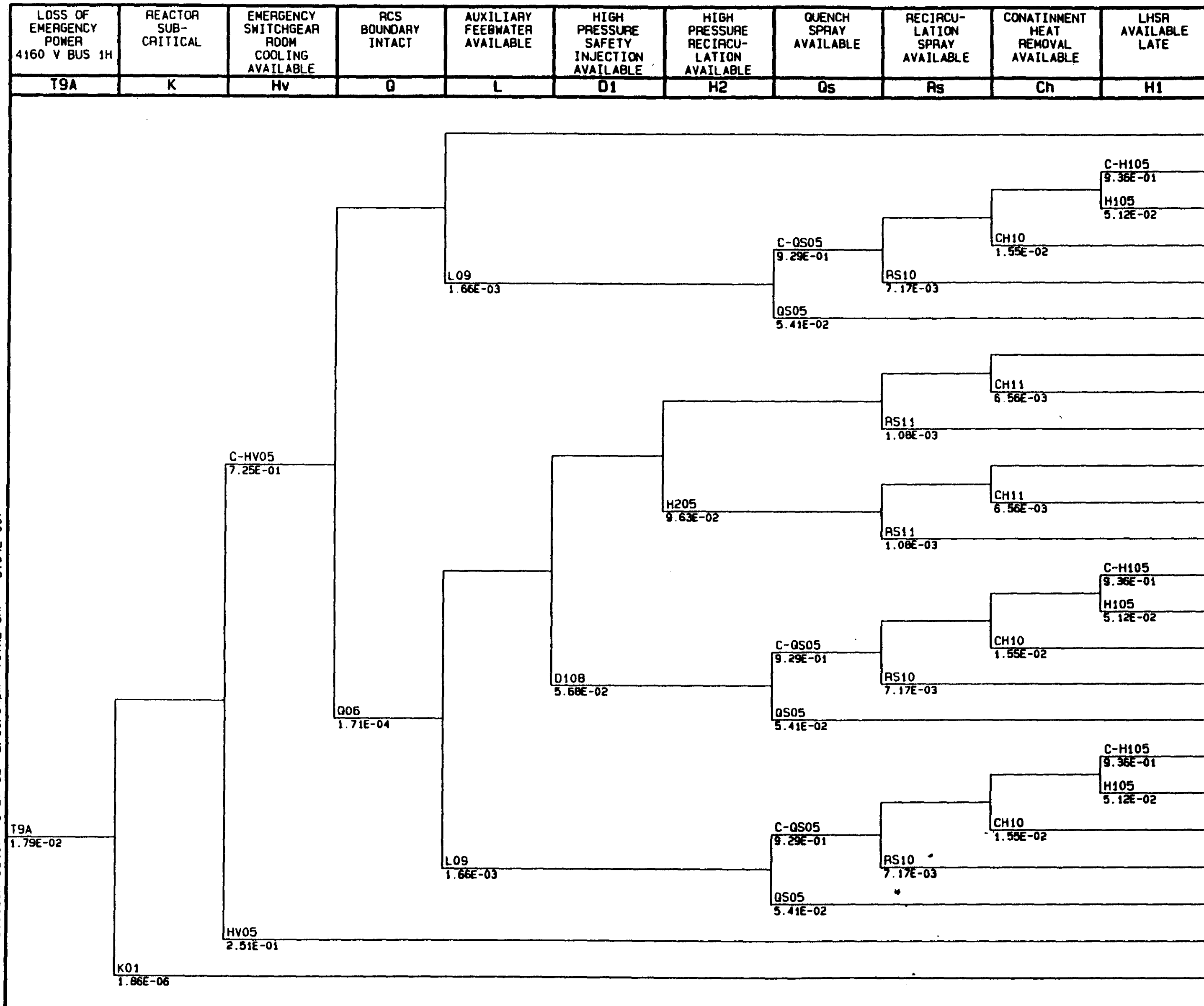


FIGURE 3.1-T8

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T8: LOSS OF EMERGENCY SWITCHGEAR ROOM COOLING EVENT TREE

C:\NAPS10\ETREES\T9A.EVT 2:09:58pm 11-23-92 NUPRA 2.0 VPWR
Quantification Date: 9-27-92 2:11:54pm TOTAL CMF = 3.94E-007

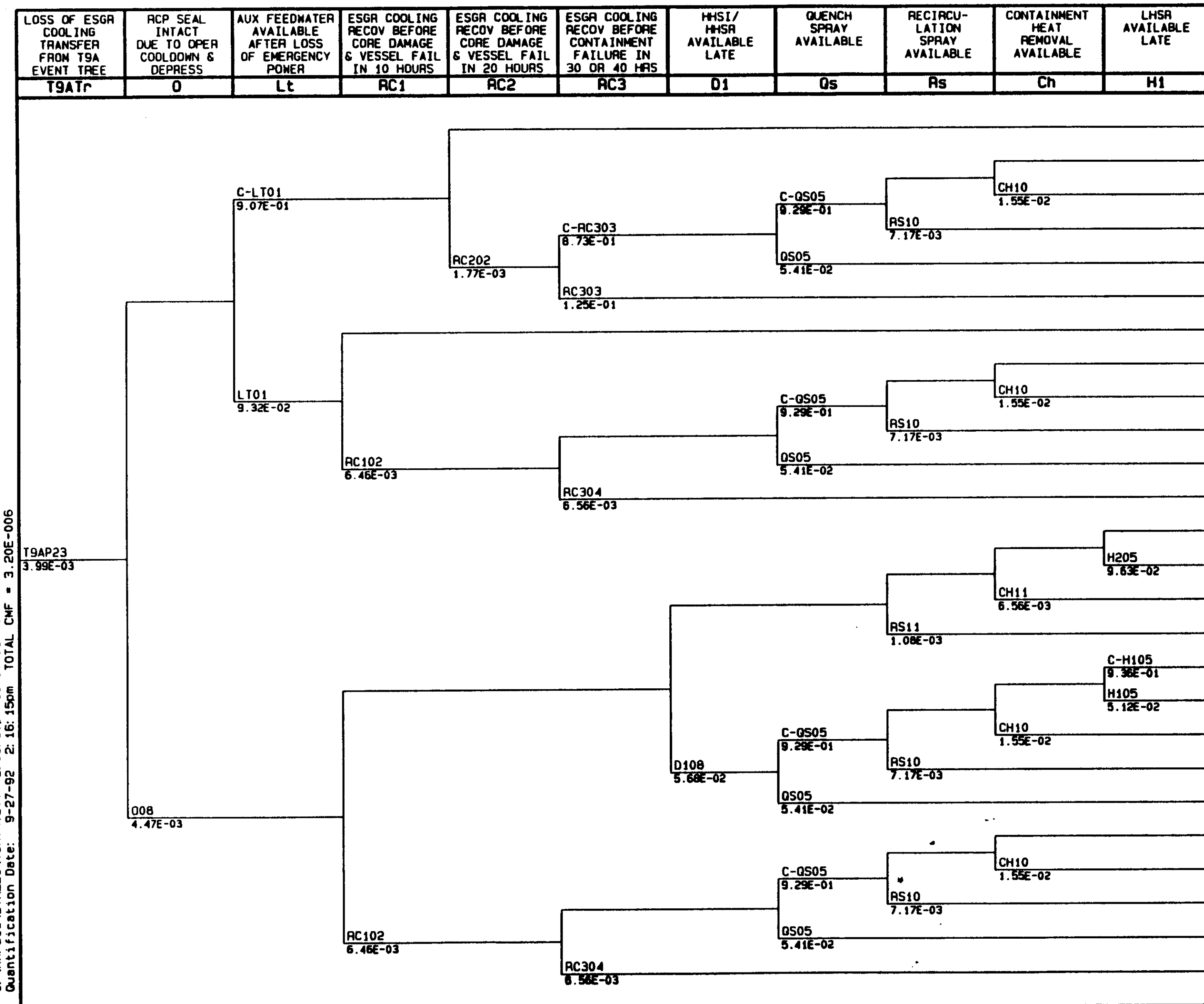


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T9A	OK	
P02	T9AL	8	1.61E-07
P03	T9ALH1	9	1.64E-09
P04	T9ALCh	10	0.00E+00
P05	T9ALRs	11	0.00E+00
P06	T9ALQs	11	6.05E-09
P07	T9AQ	TR	2.22E-06S2
P08	T9AQCh	1	0.00E+00
P09	T9AQRs	2	0.00E+00
P10	T9AQH2	21	1.27E-07
P11	T9AQH2Ch	22	0.00E+00
P12	T9AQH2Rs	23	0.00E+00
P13	T9AQD1	20	9.57E-08
P14	T9AQD1H1	21	4.14E-10
P15	T9AQD1Ch	22	0.00E+00
P16	T9AQD1Rs	23	0.00E+00
P17	T9AQD1Qs	23	1.84E-09
P18	T9AQL	20	0.00E+00
P19	T9AQLH1	21	0.00E+00
P20	T9AQLCh	22	0.00E+00
P21	T9AQLRs	23	0.00E+00
P22	T9AQLQs	23	0.00E+00
P23	T9AHv	TR	4.00E-03T8
P24	T9AK	TR	3.18E-08TH

FIGURE 3.1-T9A

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T9A: LOSS OF EMERGENCY POWER 4160 V BUS 1H EVENT TREE



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T9ATr	OK	
P02	T9ATrRC2	4	8.16E-07
P03	T9ATrRC2Ch	5	1.83E-08
P04	T9ATrRC2Rs	6	0.00E+00
P05	T9ATrRC2Qs	6	1.16E-08
P06	T9ATrRC2RC3	7	1.07E-07
P07	T9ATrLt	OK	
P08	T9ATrLtRC1	4	1.50E-06
P09	T9ATrLtRC1Ch	5	8.08E-10
P10	T9ATrLtRC1Rs	6	0.00E+00
P11	T9ATrLtRC1Qs	6	1.89E-08
P12	T9ATrLtRC1RC3	7	0.00E+00
P13	T9ATrO	OK	
P14	T9ATrOH1	21	3.88E-07
P15	T9ATrOCh	22	0.00E+00
P16	T9ATrORs	23	9.42E-10
P17	T9ATrOD1	20	2.97E-07
P18	T9ATrOD1H1	21	0.00E+00
P19	T9ATrOD1Ch	22	1.66E-08
P20	T9ATrOD1Rs	23	0.00E+00
P21	T9ATrOD1Qs	23	8.77E-09
P22	T9ATrORC1	18	1.97E-08
P23	T9ATrORC1Ch	18	0.00E+00
P24	T9ATrORC1Rs	18	0.00E+00
P25	T9ATrORC1Qs	18	0.00E+00
P26	T9ATrORC1RC3	19	0.00E+00

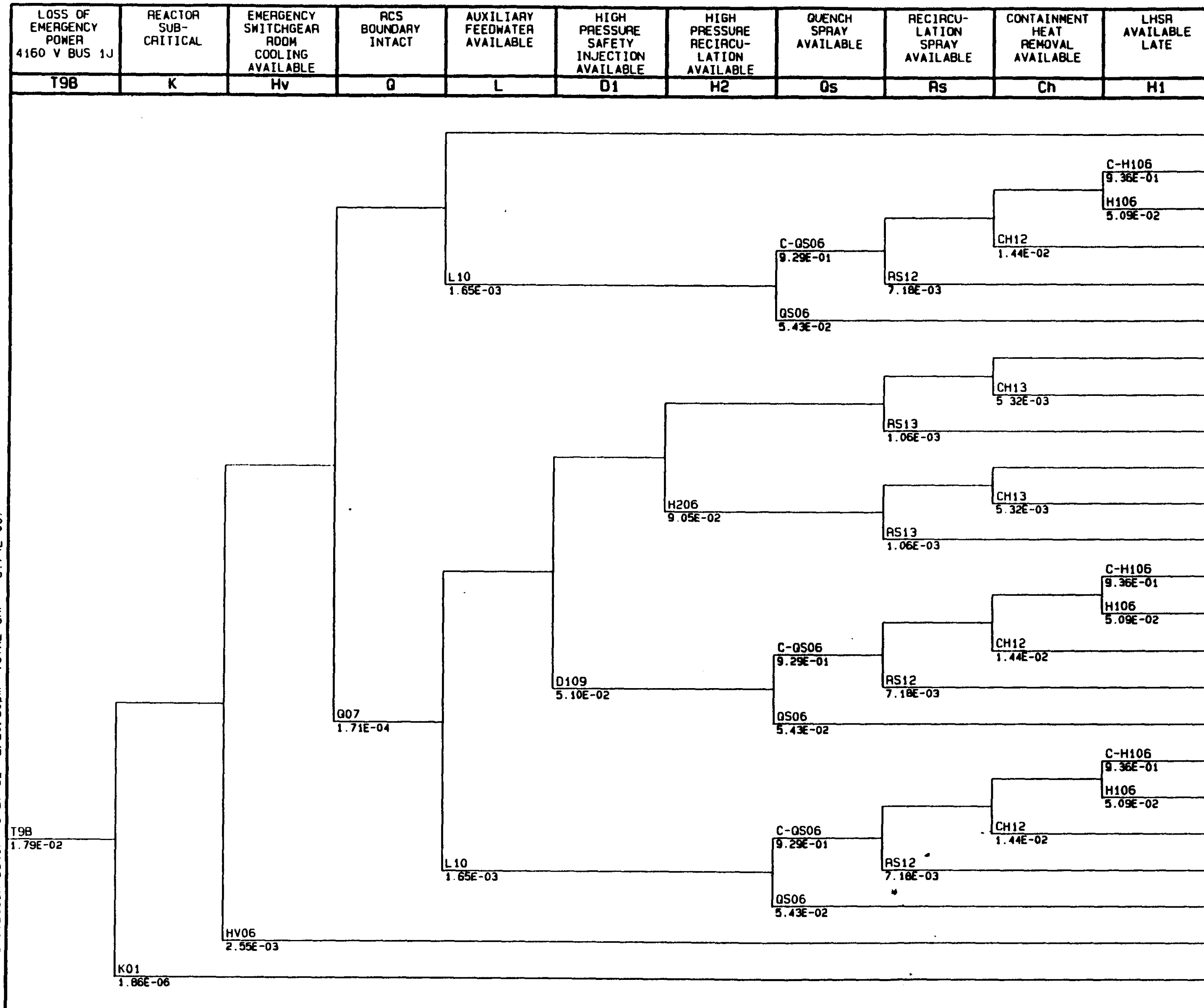
c:\NAPS1O\ETREES\T9ATR.EVT 2:01:10pm 11-23-92 NUPRA 2.0 VPWR
Quantification Date: 9-27-92 2:16:15pm TOTAL CMF = 3.20E-006

FIGURE 3.1-T9ATr

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T9ATr: LOSS OF EMERGENCY SWITCHGEAR ROOM COOLING TRANSFER FROM T9A LOSS OF 4160 V BUS 1H EVENT TREE

C:\NAPS10\ETREES\T9B.EVT 7:19:28am 11-24-92 NUPRA 2.0 VPWR
Quantification Date: 9-27-92 2:20:30pm TOTAL CHF = 5.74E-007

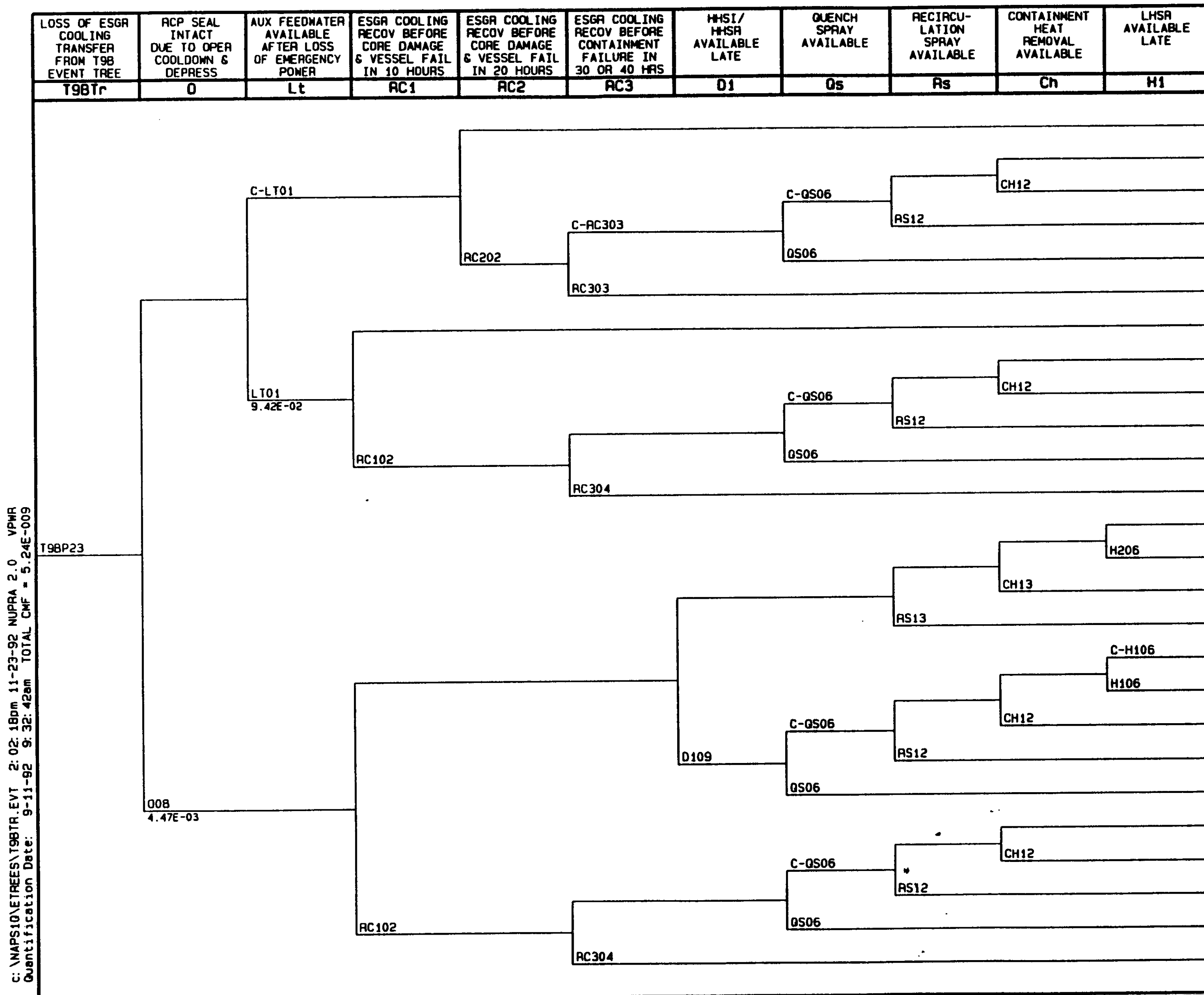


SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T9B	OK	
P02	T9BL	8	2.33E-07
P03	T9BLH1	9	1.13E-08
P04	T9BLCh	10	5.34E-10
P05	T9BLRs	11	1.24E-10
P06	T9BLQs	11	1.78E-08
P07	T9BQ	TR	3.06E-06S2
P08	T9BQCh	1	1.06E-10
P09	T9BQRs	2	2.16E-11
P10	T9BQH2	21	1.79E-07
P11	T9BQH2Ch	22	0.00E+00
P12	T9BQH2Rs	23	3.47E-11
P13	T9BQD1	20	1.27E-07
P14	T9BQD1H1	21	5.56E-10
P15	T9BQD1Ch	22	0.00E+00
P16	T9BQD1Rs	23	2.32E-11
P17	T9BQD1Qs	23	4.31E-09
P18	T9BQL	20	1.38E-11
P19	T9BQLH1	21	0.00E+00
P20	T9BQLCh	22	0.00E+00
P21	T9BQLRs	23	0.00E+00
P22	T9BQLQs	23	0.00E+00
P23	T9BHv	TR	3.09E-05T8
P24	T9BK	TR	3.18E-08TH

FIGURE 3.1-T9B

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T9B: LOSS OF EMERGENCY POWER 4160 V BUS 1J EVENT TREE



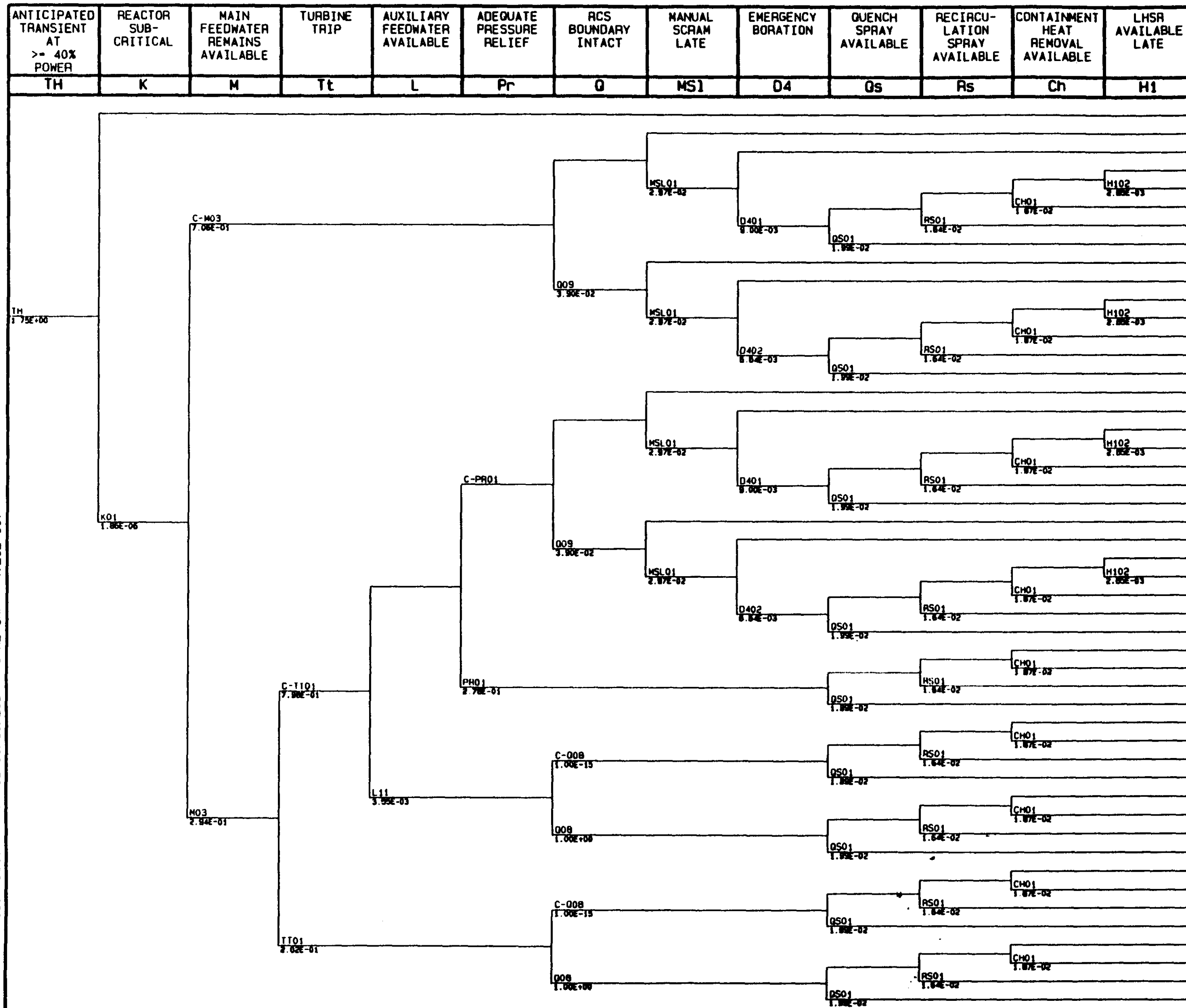
SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	T9BTr	OK	
P02	T9BTrRC2	4	1.43E-09
P03	T9BTrRC2Ch	5	0.00E+00
P04	T9BTrRC2Rs	6	2.30E-11
P05	T9BTrRC2Qs	6	2.30E-11
P06	T9BTrRC2RC3	7	2.04E-10
P07	T9BTrLt	OK	
P08	T9BTrLtRC1	4	2.71E-09
P09	T9BTrLtRC1Ch	5	0.00E+00
P10	T9BTrLtRC1Rs	6	2.69E-11
P11	T9BTrLtRC1Qs	6	2.69E-11
P12	T9BTrLtRC1RC3	7	1.83E-10
P13	T9BTrO	OK	
P14	T9BTrOH1	21	0.00E+00
P15	T9BTrOCh	22	0.00E+00
P16	T9BTrORs	23	4.37E-10
P17	T9BTrOD1	20	7.49E-11
P18	T9BTrOD1H1	21	0.00E+00
P19	T9BTrOD1Ch	22	0.00E+00
P20	T9BTrOD1Rs	23	0.00E+00
P21	T9BTrOD1Qs	23	0.00E+00
P22	T9BTrORC1	18	1.01E-10
P23	T9BTrORC1Ch	18	0.00E+00
P24	T9BTrORC1Rs	18	0.00E+00
P25	T9BTrORC1Qs	18	0.00E+00
P26	T9BTrORC1RC3	19	0.00E+00

FIGURE 3.1-T9BTr

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

T9BTr: LOSS OF EMERGENCY SWITCHGEAR ROOM COOLING TRANSFER FROM T9B LOSS OF 4160 V BUS 1J EVENT TREE

C:\NAPS1Q\ETREES\TH.EVT 10:20:48am 9-14-92 NUPRA 2.0 VPWR
Quantification Date: 9-11-92 10:10:21am TOTAL CMF = 4.26E-007



SEQ #	SEQUENCE DESCRIPTOR	PDS #	FREQUENCY
P01	TH	OK	
P02	THK	OK	
P03	THKMS1	OK	
P04	THKMS1D4	8	0.00E+00
P05	THKMS1D4H1	9	0.00E+00
P06	THKMS1D4Ch	10	0.00E+00
P07	THKMS1D4Rs	11	0.00E+00
P08	THKMS1D4Qs	11	0.00E+00
P09	THKQ	TR	8.34E-08S2
P10	THKQMS1	TR	0.00E+00S2
P11	THKQMS1D4	20	0.00E+00
P12	THKQMS1D4H1	21	0.00E+00
P13	THKQMS1D4Ch	22	0.00E+00
P14	THKQMS1D4Rs	23	0.00E+00
P15	THKQMS1D4Qs	23	0.00E+00
P16	THKM	OK	
P17	THKMMS1	OK	
P18	THKMMS1D4	8	0.00E+00
P19	THKMMS1D4H1	9	0.00E+00
P20	THKMMS1D4Ch	10	0.00E+00
P21	THKMMS1D4Rs	11	0.00E+00
P22	THKMMS1D4Qs	11	0.00E+00
P23	THKMQ	TR	2.77E-08S2
P24	THKMQMS1	TR	0.00E+00S2
P25	THKMQMS1D4	20	0.00E+00
P26	THKMQMS1D4H1	21	0.00E+00
P27	THKMQMS1D4Ch	22	0.00E+00
P28	THKMQMS1D4Rs	23	0.00E+00
P29	THKMQMS1D4Qs	23	0.00E+00
P30	THKMPr	9	2.05E-07
P31	THKMPrCh	10	0.00E+00
P32	THKMPrRs	11	3.31E-08
P33	THKMPrQs	11	3.31E-09
P34	THKML	9	0.00E+00
P35	THKMLCh	10	0.00E+00
P36	THKMLRs	11	0.00E+00
P37	THKMLQs	11	0.00E+00
P38	THKMLQ	21	0.00E+00
P39	THKMLQCh	22	0.00E+00
P40	THKMLQRs	23	0.00E+00
P41	THKMLQQs	23	0.00E+00
P42	THKMTt	9	0.00E+00
P43	THKMTtCh	10	0.00E+00
P44	THKMTtRs	11	0.00E+00
P45	THKMTtQs	11	0.00E+00
P46	THKMTtQ	21	2.14E-07
P47	THKMTtQCh	22	0.00E+00
P48	THKMTtQRs	23	0.00E+00
P49	THKMTtQQs	23	0.00E+00

FIGURE 3.1-TH

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

TH: HIGH POWER ATWS EVENT TREE
(ANTICIPATED TRANSIENT WITHOUT SCRAM AT 40% OR MORE POWER)

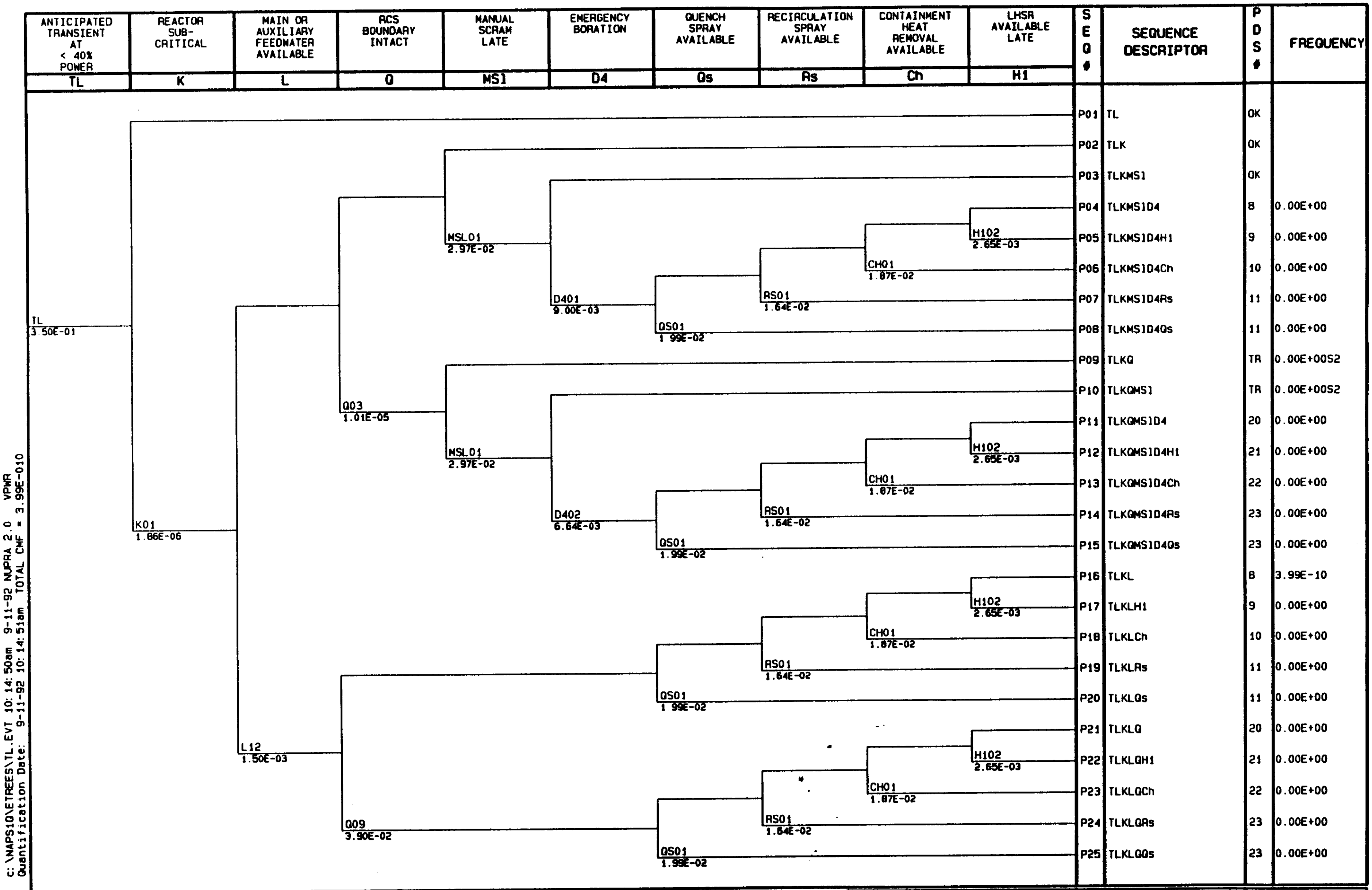


FIGURE 3.1-TL

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

TL: LOW POWER ATWS EVENT TREE
(ANTICIPATED TRANSIENT WITHOUT SCRAM, LESS THAN 40% POWER)

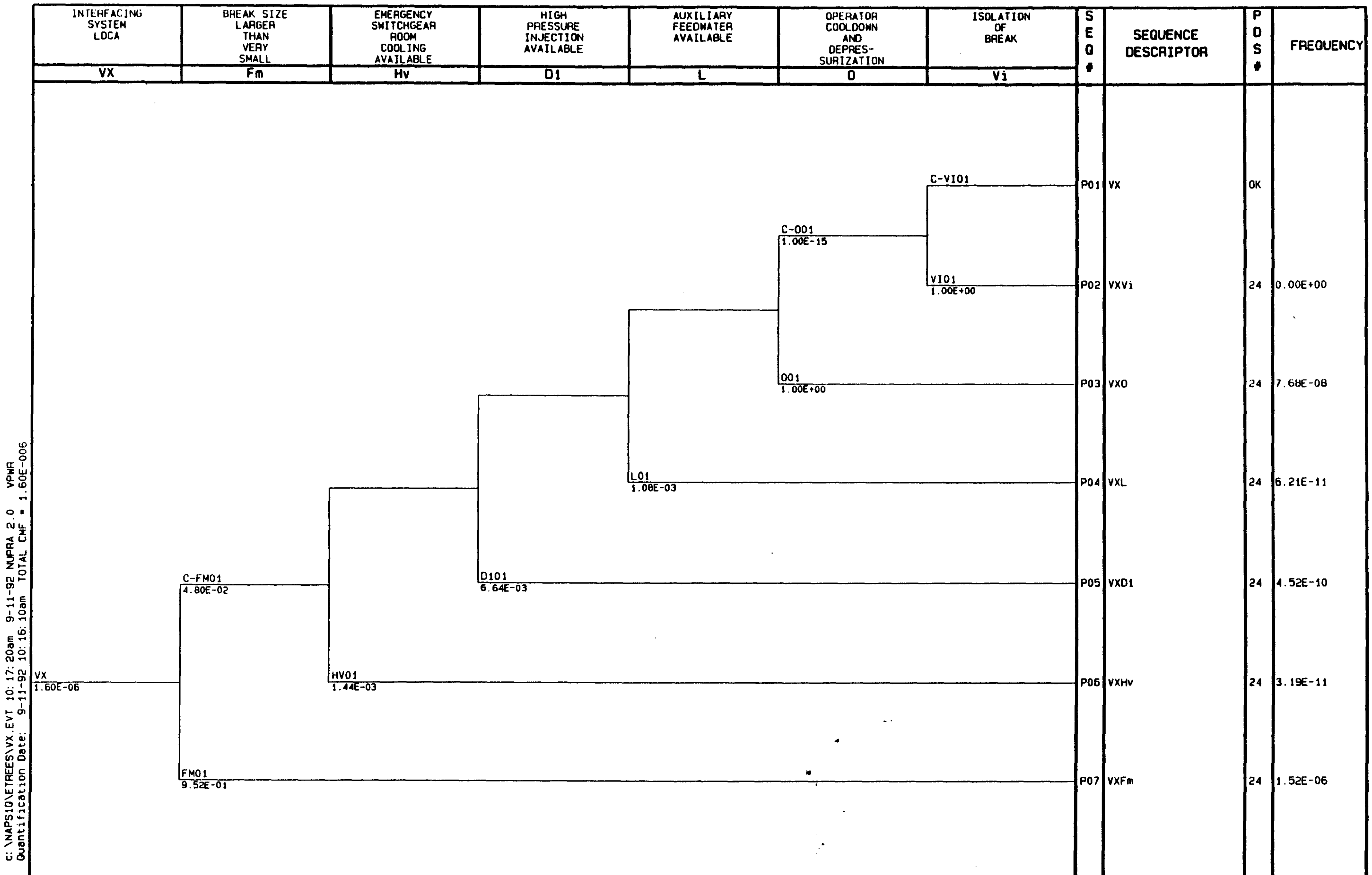


FIGURE 3.1-VX

NORTH ANNA INDIVIDUAL PLANT EXAMINATION

VX: INTERFACING SYSTEM LOCA EVENT TREE

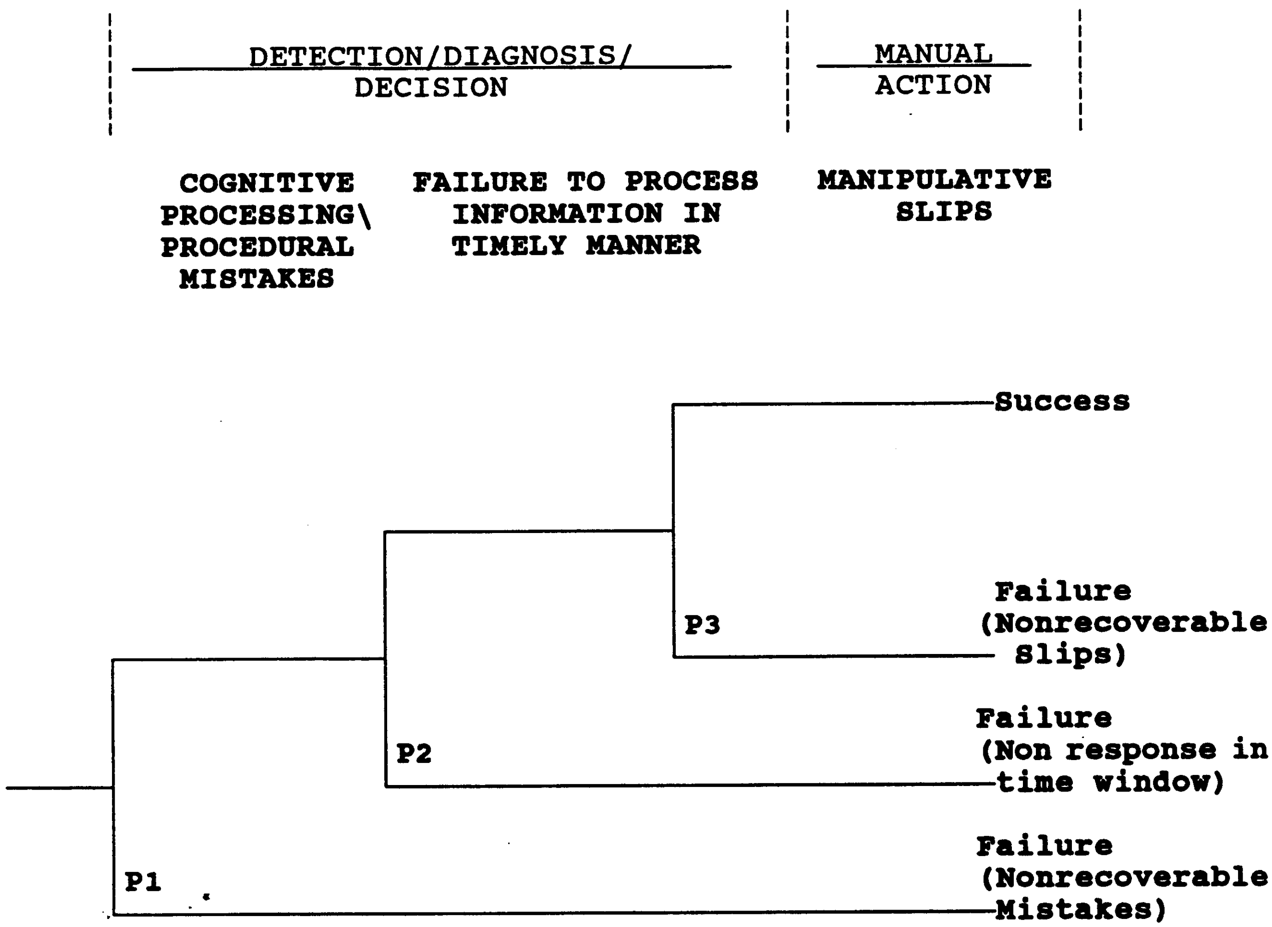


FIGURE 3.3.3-1

GENERALIZED EVENT TREE REPRESENTATION OF TYPE CP HUMAN INTERACTIONS

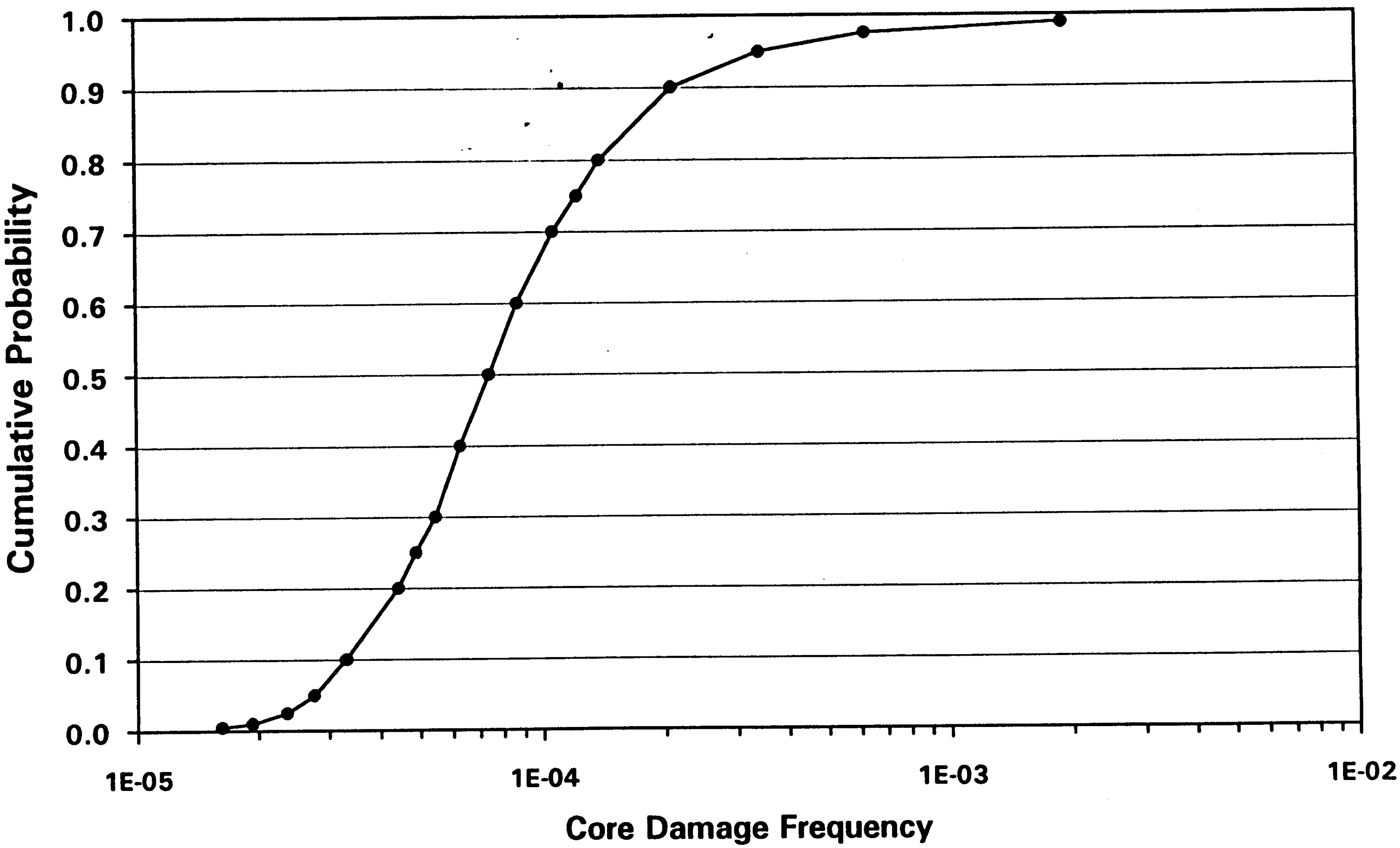


Figure 3.4.1-1
North Anna Core Damage Frequency Distribution

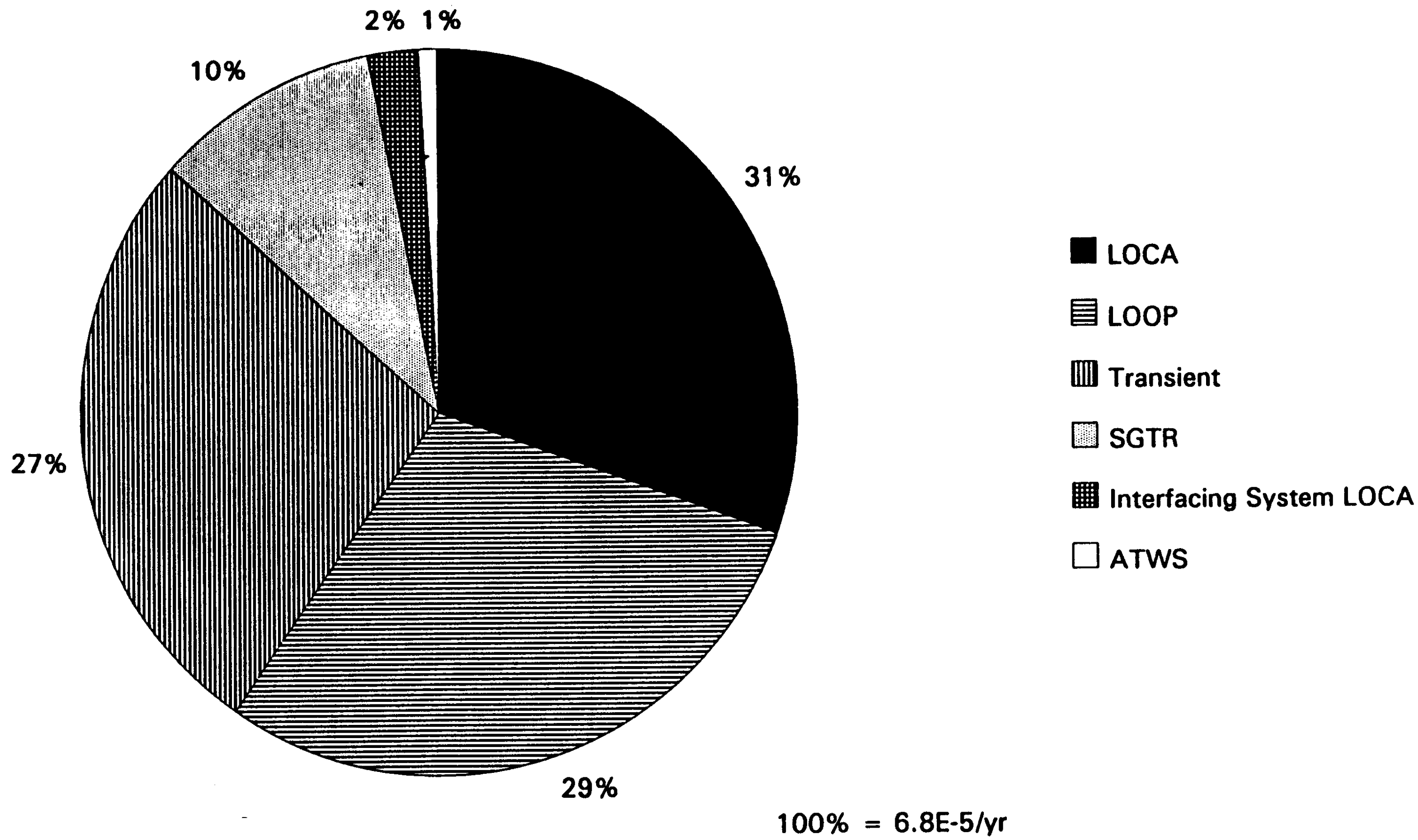


Figure 3.4.1-2
Contribution of Accident Groups to Core Damage Frequency

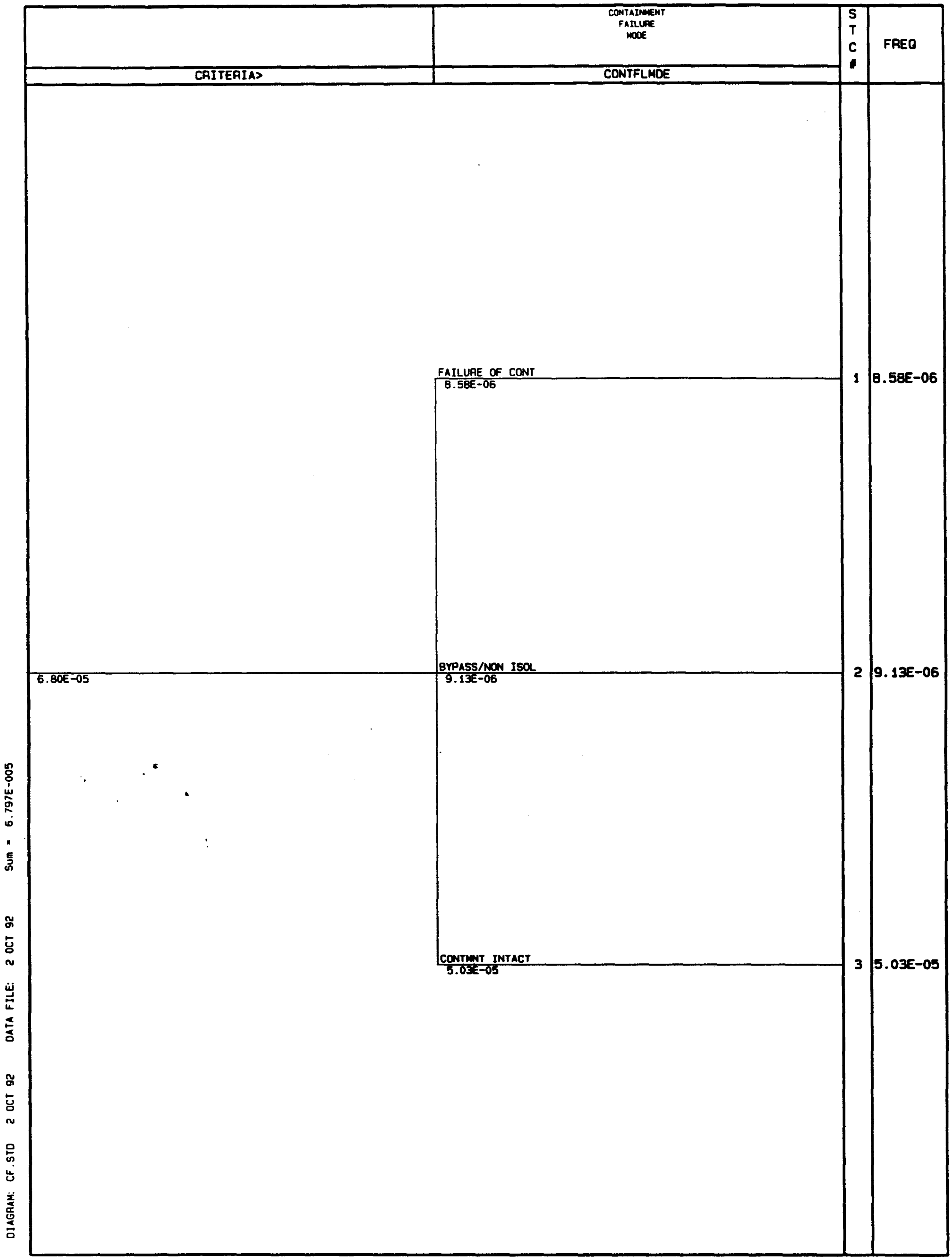


FIGURE 3.4.1-3
Classification of CD Sequences by Containment Status

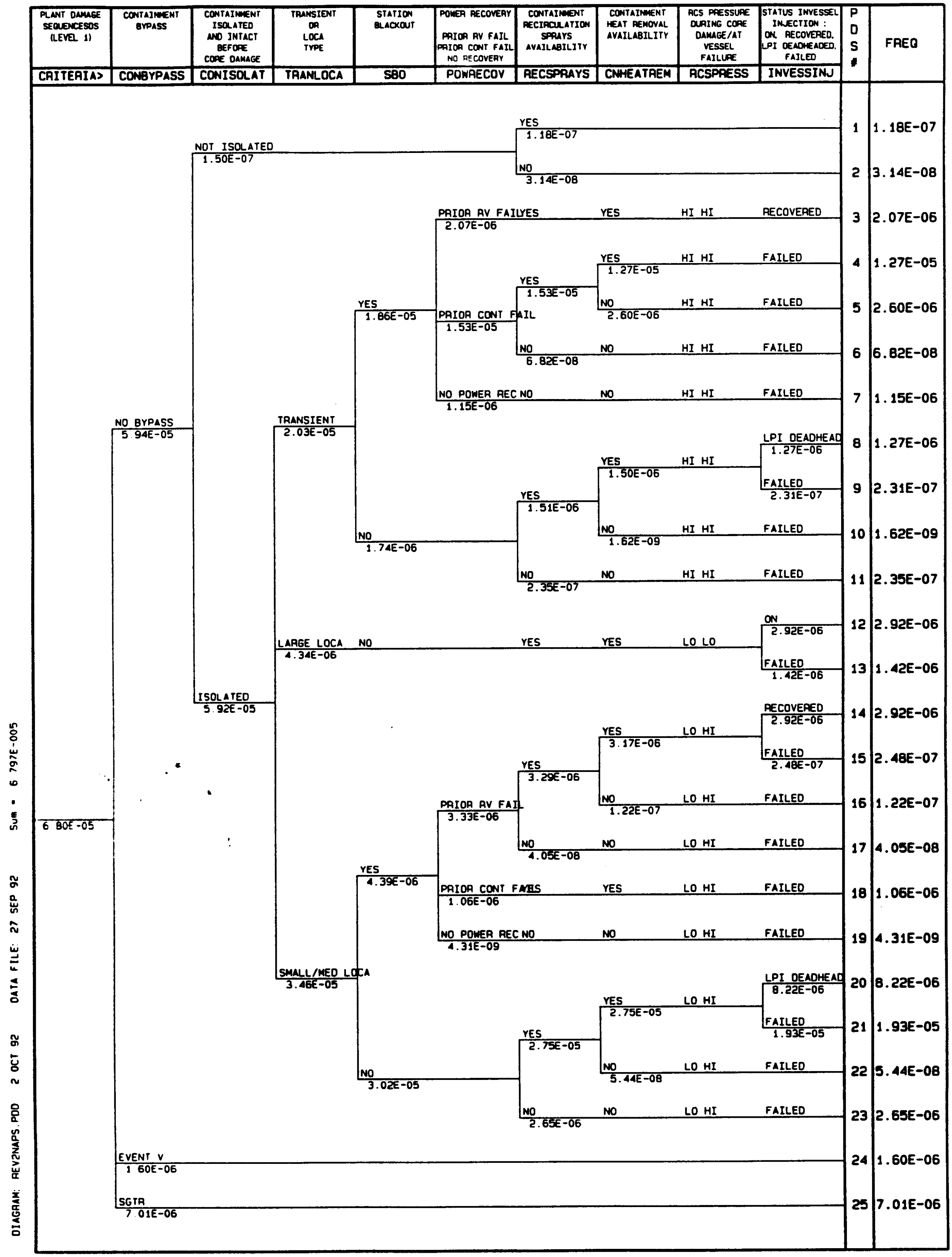


FIGURE 3.4.1-4 NAPS Plant Damage State Logic Diagram

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