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ABSTRACT

This report documents an analysis of the safety-related performance of the reactor protection system (RPS) at U.S. General Electric commercial reactors during the period 1984 through 1995. General Electric RPS designs analyzed in this report include those with relay-based trip systems. The analysis is based on a BWR/4 plant design. RPS operational data were collected for all U.S. General Electric commercial reactors from the Nuclear Plant Reliability Data System and Licensee Event Reports. A risk-based analysis was performed on the data to estimate the observed unavailability of the RPS, based on a fault tree model of the system. An engineering analysis of trends and patterns was also performed on the data to provide additional insights into RPS performance. RPS unavailability results obtained from the data were compared with existing unavailability estimates from Individual Plant Examinations and other reports.

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EXECUTIVE SUMMARY

This report documents an analysis of the safety-related performance of the reactor protection system (RPS) at U.S. General Electric commercial reactors during the period 1984 through 1995. Objectives of the study were the following: (1) to estimate RPS unavailability based on operational experience data and compare the results with models used in probabilistic risk assessments (PRAs) and individual plant examinations (IPEs), and (2) to review the operational data from an engineering perspective to determine trends and patterns, and to gain additional insights into RPS performance. The General Electric RPS designs covered in the unavailability estimation include those with relay-based trip systems. The fault tree developed for this design assumed a BWR/4 plant.

General Electric RPS operational data were collected from Licensee Event Reports as reported in the Sequence Coding and Search System and the Nuclear Plant Reliability Data System. The period covered 1984 through 1995. Data from both sources were evaluated by engineers with operational experience at nuclear power plants. Approximately 7,000 events were evaluated for applicability to this study. Those data not excluded were further characterized as to the type of RPS component, type of failure, failure detection, status of the plant during the failure, etc. Characterized data include both independent component failures and common-cause failures (CCFs) of more than one component. The CCF data were classified as outlined in the report *Common-Cause Failure Data Collection and Analysis System* (NUREG/CR-6268). Component demand counts were obtained from plant reactor trip histories and component test frequency information.

The risk-based analysis of the RPS operational data focused on obtaining failure probabilities for component independent failure and CCF events in the RPS fault tree. The level of detail of the basic events includes the following: channel trip signal sensor/transmitters and associated bistables, process switches, and relays; hydraulic control units (solenoid- and air-operated valves and the scram accumulator); and control rod drives and control rods. CCF events were modeled for all redundant, similar types of components.

Data analysis and subsequent fault tree quantification resulted in an RPS mean unavailability (failure probability upon demand) of 5.8E-6 for the BWR/4 relay-based design. (This unavailability does not include any credit for operator action to actuate the manual scram switches.) An uncertainty analysis resulted in a 5th percentile value of 1.8E-6 and a 95th percentile value of 1.4E-5. Essentially 100% of this unavailability is from CCF events; the combinations of independent failures contribute less than 0.1%. Channel failures contribute 58% to the total unavailability, hydraulic control unit failures contribute 32%, trip system failures contribute 6%, and control rod and control rod drive failures contribute 4%.

CCF events involving the scram pilot solenoid-operated valves (SOVs) and backup scram SOVs contribute 29% to the overall RPS unavailability. The most significant historical event, involving the use of improper seating material and affecting all the scram pilot SOVs, occurred in 1984. Two similar types of SOV CCF events occurred in 1994 but did not affect as many of the components.

Also, problems with the use of liquid thread sealant resulted in several significant CCF events. It is believed that the requirement to test 10% of the control rods each four months helped discover these types of problems (developing over time) before they developed to catastrophic failures.

The backup scram portion of the RPS may be an important contributor to low RPS unavailability, based on the sensitivity study discussed in Appendix G of this report and uncertainties associated with the SOV failure characteristics. (Without the backup scram logic, only two of eight trip system relay failures are needed to fail the RPS, rather than four of eight if the backup scram system is modeled.) The backup scram SOVs are classified as non-safety-related, and these valves are not part of the NPRDS reportable scope for the General Electric RPS. Therefore, no failure data were collected for these valves. Also, it is not clear how often these valves are tested, and what their failure probabilities are. This study assumed these valves are tested every 18 months during shutdown, and that their failure characteristics are similar to the scram pilot SOVs. These assumptions should be verified.

There were significant scram discharge volume (SDV) problems in the early 1980s involving both drainage of SDVs and level instrumentation, dominated by the 1980 Browns Ferry Unit 3 failure of 76 of 185 control rods to insert. Data collected during the period 1984 through 1995 indicate that SDV instrumentation failure probabilities are similar to other RPS trip instrumentation. Also, only one inadvertent filling of the SDV while a plant was at power was identified during the period. Finally, the RPS fault tree quantification indicates that SDV events leading to failure of the RPS contribute less than 1% to the overall RPS unavailability. Therefore, early SDV-related problems in General Electric RPSs are no longer dominant contributors to RPS unavailability.

The RPS fault tree was also quantified allowing credit for manual scram by the operator (with a failure probability of 0.01). The resulting RPS unavailability is 2.6E-6. Operator action reduces the RPS unavailability by approximately 55%. This reduction is limited because a dominant contributor to RPS unavailability is the scram pilot SOV CCF event, which is unaffected by the operator action. Also, the manual scram signal must still pass through the channel and trip system relays, for the configuration analyzed. RPS hydraulic control unit failures (SOVs) contribute 71% to the total unavailability, trip system failures contribute 14%, control rod and control rod drive failures contribute 10%, and channel failures contribute 5%.

The unavailability estimate of 5.8E-6 (allowing no credit for manual scram by the operator) is lower than typically used in the past. Past estimates typically ranged from 1.0E-5 to 3.0E-5 and were usually based on information in NUREG-0460, published in 1978. The individual component failure probabilities per demand, derived from the 1984 through 1995 data, are generally comparable to failure probability estimates listed in previous reports. Therefore, the low RPS unavailability estimate is mostly attributable to lower failure probabilities for the CCF events. The General Electric RPS CCF events collected for this project, covering the period 1984 through 1995, contain few events involving complete failures of many redundant components. Correspondingly, the CCF calculations result in low CCF failure probabilities.

The trends in component failure probabilities and numbers of CCF events are generally flat over the period 1984 through 1995. Therefore, existing RPS surveillance and maintenance practices and industry lessons learned programs have been effective in preventing increasing failure probabilities.

FOREWORD

This report provides information relevant to the reliability of the General Electric reactor protection system (RPS). It summarizes the event data used in the analysis. The results, findings, conclusions, and information contained in this study, the initiating event update study, and related system reliability studies conducted by the Office for Analysis and Evaluation of Operational Data are intended to support several risk-informed regulatory activities. This includes providing information about relevant operating experience that can be used to enhance plant inspections of risk-important systems, and information used to support staff technical reviews of proposed license amendments, including risk-informed applications. In the future, this work will be used in the development of risk-based performance indicators that will be based to a large extent on plant-specific system and equipment performance.

Findings and conclusions from the analyses of the General Electric RPS, which are based on 1984–1995 operating experience, are presented in the Executive Summary. The results of the quantitative analysis and engineering analysis are presented in Sections 3 and 4, respectively. The information to support risk-informed regulatory activities related to the General Electric RPS is summarized in Table F-1. This table provides a condensed index of risk-important data and results presented in discussions, tables, figures, and appendices.

The application of results to plant-specific applications may require a more detailed review of the relevant Licensee Event Report (LER) and Nuclear Plant Reliability Data System (NPRDS) data cited in this report. This review is needed to determine if generic experiences described in this report and specific aspects

Table F-1. Summary of risk-important information specific to General Electric reactor protection system.

1.	General insights and conclusions regarding RPS unavailability	Section 5
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of the RPS events documented in the LER and NPRDS failure records are applicable to the design and operational features at a specific plant or site. Factors such as RPS design, specific components installed in the system, and test and maintenance practices would need to be considered in light of specific information provided in the LER and NPRDS failure records. Other documents such as logs, reports, and inspection reports that contain information about plant-specific experience (e.g., maintenance, operation, or surveillance testing) should be reviewed during plant inspections to supplement the information contained in this report.

Additional insights may be gained about plant-specific performance by examining the specific events in light of the overall industry performance. In addition, a review of recent LERs and plant-specific component failure information in NPRDS or Equipment Performance Information and Exchange System (EPIX) may yield indications of whether performance has undergone any significant change since the last year of this report. A search of the LER database can be conducted through the NRC's Sequence Coding and Search System (SCSS) to identify the RPS events that occurred after the period covered by this report. SCSS contains the full text LERs and is accessible by NRC staff from the SCSS home page (http://scss.ornl.gov/). Nuclear industry organizations and the general public can obtain information from the SCSS on a cost recovery basis by contacting the Oak Ridge National Laboratory directly.

Periodic updates to the information in this report will be performed as additional data become available.

Charles E. Rossi, Director
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ACRONYMS

ACC	hydraulic control unit accumulator
ACRS	Advisory Committee on Reactor Safety (U.S. NRC)
AEOD	Analysis and Evaluation of Operational Data (U.S. NRC Office)
AOV	hydraulic control unit scram inlet or outlet air-operated valve
APRM	average power range monitor
ARI	alternate rod insertion
ATWS	anticipated transient without scram
ATWS-RPT	ATWS recirculation pump trip
BWR	boiling water reactor
BWR/4	design class 4 BWR
CBI	channel bistable (trip unit)
CCF	common-cause failure
CF	complete failure
CPL	channel level sensor/transmitter
CPR	channel pressure sensor/transmitter
CPS	process switch
CRD	control rod drive
FS	fail-safe (component failure not impacting safety function)
HCU	hydraulic control unit
INEEL	Idaho National Engineering and Environmental Laboratory
IPE	Individual Plant Examination
K5, K6, K14	specific General Electric relays
MSW	manual scram switch
NF	no failure
NFS	non-fail-safe (component failure impacting safety function)

NPRDS	Nuclear Plant Reliability Data System
NRC	Nuclear Regulatory Commission (U.S.)
PRA	probabilistic risk assessment
PWR	125 Vdc power to backup scram solenoid-operated valve
RDC	rod and control rod drive
ROD	rod
RPS	reactor protection system
SCSS	Sequence Coding and Search System
SDL	scram discharge volume level switch
SDV	scram discharge volume
SLCS	standby liquid control system
SOV	solenoid-operated valve
TLR	trip logic relay
UC	unknown completeness (unknown if failure was CF or NF)
UKN	unknown (unknown if failure was NFS or FS)

TERMINOLOGY

Channel segment—The portion of the General Electric reactor protection system that includes trip signal sensor/transmitters and associated trip units (bistables), process switches, associated (K1, K5 and K6) relays, and other components distributed throughout the plant, that monitor the state of the plant and generate automatic trip signals. There are four channels in the channel segment.

Common-cause failure—A dependent failure in which two or more similar component fault states exist simultaneously, or within a short time interval, and are a direct result of a shared cause.

Common-cause failure model—A model for classifying and quantifying the probabilities of common-cause failures. The alpha factor model is used in this study.

Hydraulic control unit segment—The set of hydraulic control units (HCU) and associated scram pilot solenoid-operated valves (SOVs), scram inlet and outlet air-operated valves (AOVs), and the scram accumulators. There is one set of HCU equipment for each control rod. The HCU segment also includes the scram discharge volume and two backup scram SOVs controlling instrument air to the common scram AOV air header.

Reactor protection system—The complex control system comprising numerous electronic and mechanical components that provides the ability to produce an automatic or manual rapid shutdown of a nuclear reactor, given plant upset conditions that require a reactor trip.

Rod segment—The portion of the General Electric reactor protection system than includes the control rod drives and the control rods. There are generally 120 to 190 control rods and associated drives in BWR plants.

Scram—Automatic or manual actuation of the reactor protection system, resulting in insertion of control rods into the core and shutdown of the nuclear reaction. Also called a reactor trip.

Trip system segment—The portion of the General Electric reactor protection system that includes the reactor trip (K14) relays housed in cabinets in the control room. There are two trains in the trip system segment. Each train receives signals from two of the four instrument channels and one of the two manual scram switches. Each train energizes one of the two scram pilot solenoid-operated valves for each hydraulic control unit.

Unavailability—The probability that the reactor protection system will not actuate (and result in a reactor trip), given a demand for the system to actuate.

Unreliability—The probability that the reactor protection system will not fulfill its mission, given a demand for the system. Unreliability typically involves both failure to actuate and failure to continue to function for an appropriate mission time. However, the reactor protection system has no mission time. Therefore, for the reactor protection system, unreliability and unavailability are the same.

General Electric Reactor Protection System Unavailability, 1984–1995

1. INTRODUCTION

The U.S. Nuclear Regulatory Commission's (NRC's) Office for Analysis and Evaluation of Operational Data (AEOD) has, in cooperation with other NRC offices, undertaken an effort to ensure that the stated NRC policy to expand the use of probabilistic risk assessment (PRA) within the agency is implemented in a consistent and predictable manner. As part of this effort, the AEOD Safety Programs Division has undertaken to monitor and report upon the functional reliability of risk-important systems in commercial nuclear power plants. The approach is to compare estimates and associated assumptions found in PRAs to actual operating experience. The first phase of the review involves the identification of risk-important systems from a PRA perspective and the performance of reliability and trending analysis on these identified systems. As part of this review, a risk-related performance evaluation of the reactor protection system (RPS) in General Electric boiling water reactors (BWRs) was performed.

An abbreviated U.S. history of regulatory issues related to RPS and anticipated transient without scram (ATWS) begins with a 1969 concern[1] from the Advisory Committee on Reactor Safeguards (ACRS) that RPS common mode failures might result in unreliabilities higher than previously thought. At that time, ATWS events were considered to have frequencies lower than 1E-6/y, based on the levels of redundancy in RPS designs. Therefore, such events were not included in the design basis for U.S. nuclear power plants. This concern was followed by issuance of WASH-1270[2] in 1973, in which the RPS unavailability was estimated to be 6.9E-5 (median value). Based on this information and the fact that increasing numbers of nuclear reactors were being built and operated in the U.S., it was recommended that ATWS events be considered in the safety analysis of nuclear reactors. In 1978, NUREG-0460 [1] was issued. In that report, the RPS unavailability was estimated to be in the range 1E-5 to 1E-4. An unavailability of 3E-5 was recommended, allowing for some improvements in design and performance. In addition, it was recommended that consideration be given to additional systems that would help to mitigate ATWS events, given failure of the RPS. The 1980 BWR Browns Ferry Unit 3 event in which 76 of 185 control rods failed to insert fully and the 1983 pressurized water reactor (PWR) Salem Unit 1 low-power ATWS events (failure of the undervoltage coils to open the reactor trip breakers) led to NUREG-1000[3] and Generic Letter 83-28. [4] These documents discussed actions to improve RPS reliability, including the requirement for functional testing of backup scram systems. Finally, 49FR26036 [5] in 1984, Generic Letter 85-06 [6] in 1985 and 10CFR50.62 [7] in 1986 outlined requirements for diverse ATWS mitigation systems.

The risk-related performance evaluation in this study measures RPS unavailability using actual operating experience. To perform this evaluation, system unavailability was evaluated using two levels of detail: the entire system (without distinguishing components within the system), and the system broken down into components such as sensors, logic modules, and relays. The modeling of components in the RPS was necessary because the U.S. operating experience during the period 1984 through 1995 does not include any RPS system failures. Therefore, unavailability results for the RPS modeled at the system level provide limited information. Additional unavailability information is gained by working at the component level, at which actual failures have occurred. RPS unavailability in this evaluation is concerned with failure of the function of the system to shut down the reactor given a plant upset condition requiring a reactor trip. Component or system failures causing spurious reactor trips or not affecting the shutdown function of the RPS are not considered in this report. However, failures and associated demands that occurred during tests of portions of the RPS are included in the component level evaluation of the RPS unavailability, even though such demands do not model a complete system response for accident mitigation. This is in contrast to previous system studies, in which such partial system tests generally were not used.

It should be noted that the RPS boundary for this study does not include ATWS mitigation systems added or modified in the late 1980s. For General Electric nuclear reactors, these systems include alternate rod insertion (ARI), standby liquid control system (SLCS), and ATWS recirculation pump trip (ATWS-RPT). Also, this study deals mainly with automatic actuation of the RPS. However, RPS unavailability was also determined assuming credit for operator action.

The RPS unavailability study is based on U.S. General Electric RPS operational experience data from the period 1984 through 1995, as reported in both the Nuclear Plant Reliability Data System (NPRDS)[8] and Licensee Event Reports (LERs) found in the Sequence Coding and Search System (SCSS).[9]

The objectives of the study were the following:

1. Estimate RPS unavailability based on operation data, and compare the results with the assumptions, models, and data used in PRAs and Individual Plant Examinations (IPEs).

2. Provide an engineering analysis of the factors affecting system unavailability and determine if trends and patterns are present in the RPS operational data.

The remainder of this report is arranged in five sections. Section 2 describes the scope of the study, including a system description for the RPS, description of the fault tree models used in the analysis, and descriptions of the data collection, characterization, and analysis. Section 3 contains the unavailability results from the operational data and comparisons with PRA/IPE RPS results. Section 4 provides the results of the engineering analysis of the operational data. A summary and conclusions are presented in Section 5. Finally, Section 6 contains the references.

There are also seven appendices in this report. Appendix A provides a detailed explanation of the methods used for data collection, characterization, and analysis. Appendix B gives a summary of the operational data. The detailed statistical analyses are presented in Appendix C. The fault tree model is included in Appendix D. Common-cause failure modeling information is presented in Appendix E. The fault tree quantification results, cut sets and importance rankings, are in Appendix F. Finally, sensitivity analysis results are presented in Appendix G.

2. SCOPE OF STUDY

This study documents an analysis of the operational experience of the General Electric RPS from 1984 through 1995. The analysis focused on the ability of the RPS to automatically shut down the reactor given a plant upset condition requiring a reactor trip while the plant is at full power. The term "reactor trip" refers to a rapid insertion of control rods into the reactor core to inhibit the nuclear reaction. RPS spurious reactor trips or component failures not affecting the automatic shutdown function were not considered. A General Electric RPS description is provided, followed by a description of the RPS fault tree used in the study. The section concludes with a description of the data collection, characterization, and analysis.

2.1 System Description

2.1.1 System Operation

The General Electric RPS is a complex control system comprising numerous electronic components that combine to provide the ability to produce an automatic or manual rapid shutdown of the nuclear reactor, known as a reactor trip or scram. In spite of its complexity, the General Electric RPS components can be roughly divided into four segments—channels, trip systems, hydraulic control units (HCUs), and rods—as shown in Figure 1. The rod segment includes the control rods and associated control rod drives (CRDs). General Electric RPSs typically have 120 to 190 control rods and associated CRDs. The HCU segment includes the HCU components: scram pilot solenoid-operated valves or SOVs, scram inlet and outlet air-operated valves or AOVs, and scram accumulator. There is one HCU for each CRD. Also included in the HCU segment are the scram discharge volume (SDV) and two backup scram SOVs controlling instrument air to the scram air header. Some GE plants have a single, dual-coil SOV rather than two single-coil scram pilot SOVs, and the number of SDVs can be one or two. For the trip system segment, all but one of the GE plants have relay-based trip systems. Clinton, a BWR6 design, is the only GE plant to have a solid-state trip system. (The Clinton RPS design is not covered in this report.)

The analysis of the General Electric RPS is based on a BWR/4 design, with Peach Bottom Unit 2 chosen as the reference plant. This configuration, termed the relay-based RPS, has been used in a General Electric generic analysis of RPSs as representative of BWR RPS designs except for the Clinton solid-state design. A representative integrated system diagram of the RPS is shown in Figure 2. Simplified diagrams of the design, constructed to more clearly show the breakdown of the RPS into segments, are presented in Figures 3 through 6. Note that the relay numbers in Figures 3 through 6 have been chosen to be consistent with the NPRDS GE RPS diagrams.[8]

As shown in Figures 3 through 6, there are two RPS trip systems, A and B. These trip systems receive trip signals from the channels, process the signals, and then open the HCU scram pilot SOVs given appropriate combinations of signals from the channels. Opening the scram pilot SOVs bleeds the

RPS Segments			
Channel	Trip System	HCUs and Related	Rods
4 channels (A – D, sometimes termed A1, A2, B1, and B2)	2 trip systems (A, B); scram logic and backup scram logic	120 to 190 HCUs; 1 or 2 SDVs	120 to 190 CRDs and associated control rods

Figure 1. Segments of General Electric RPS.

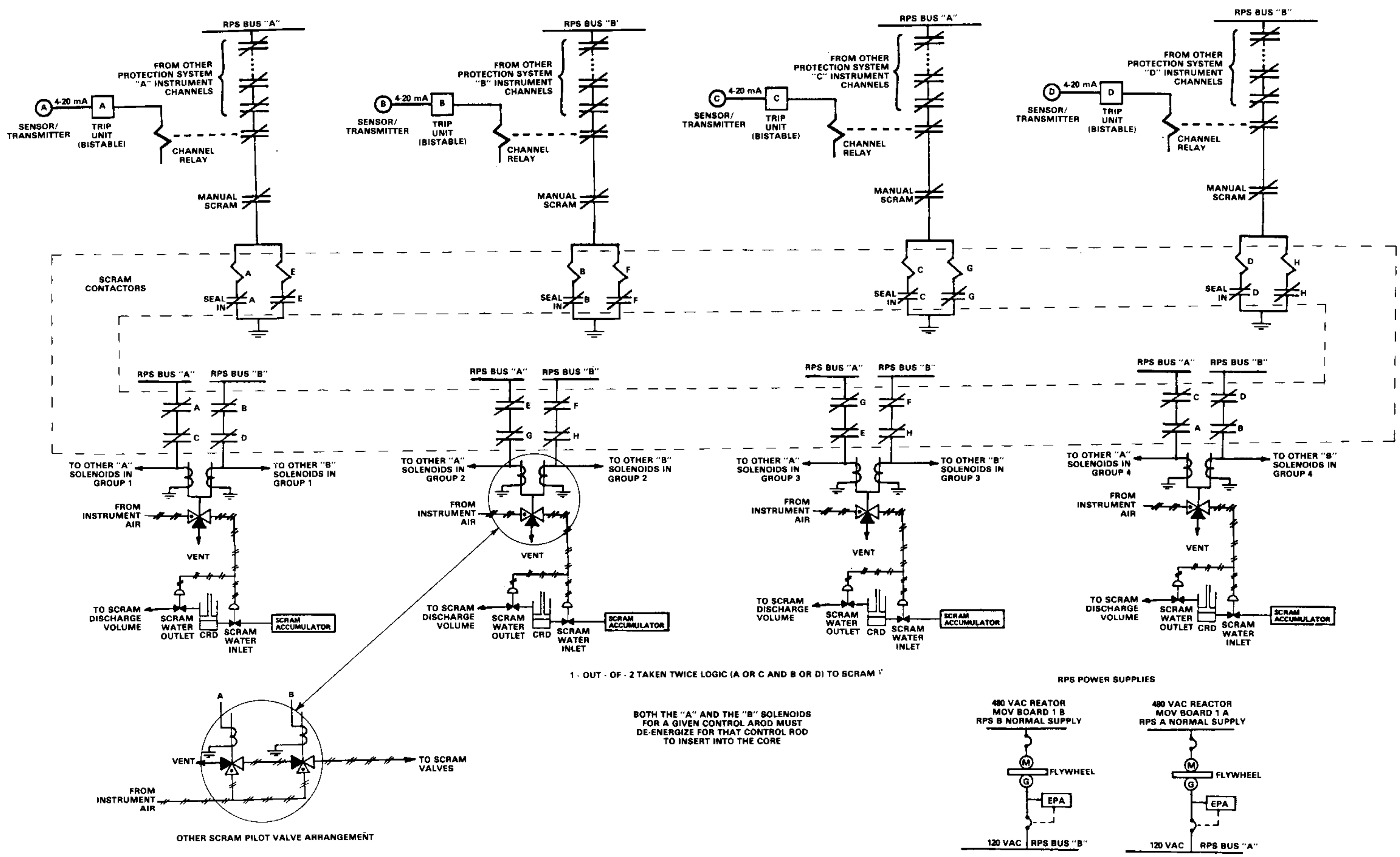


Figure 2. General Electric RPS integrated system diagram.

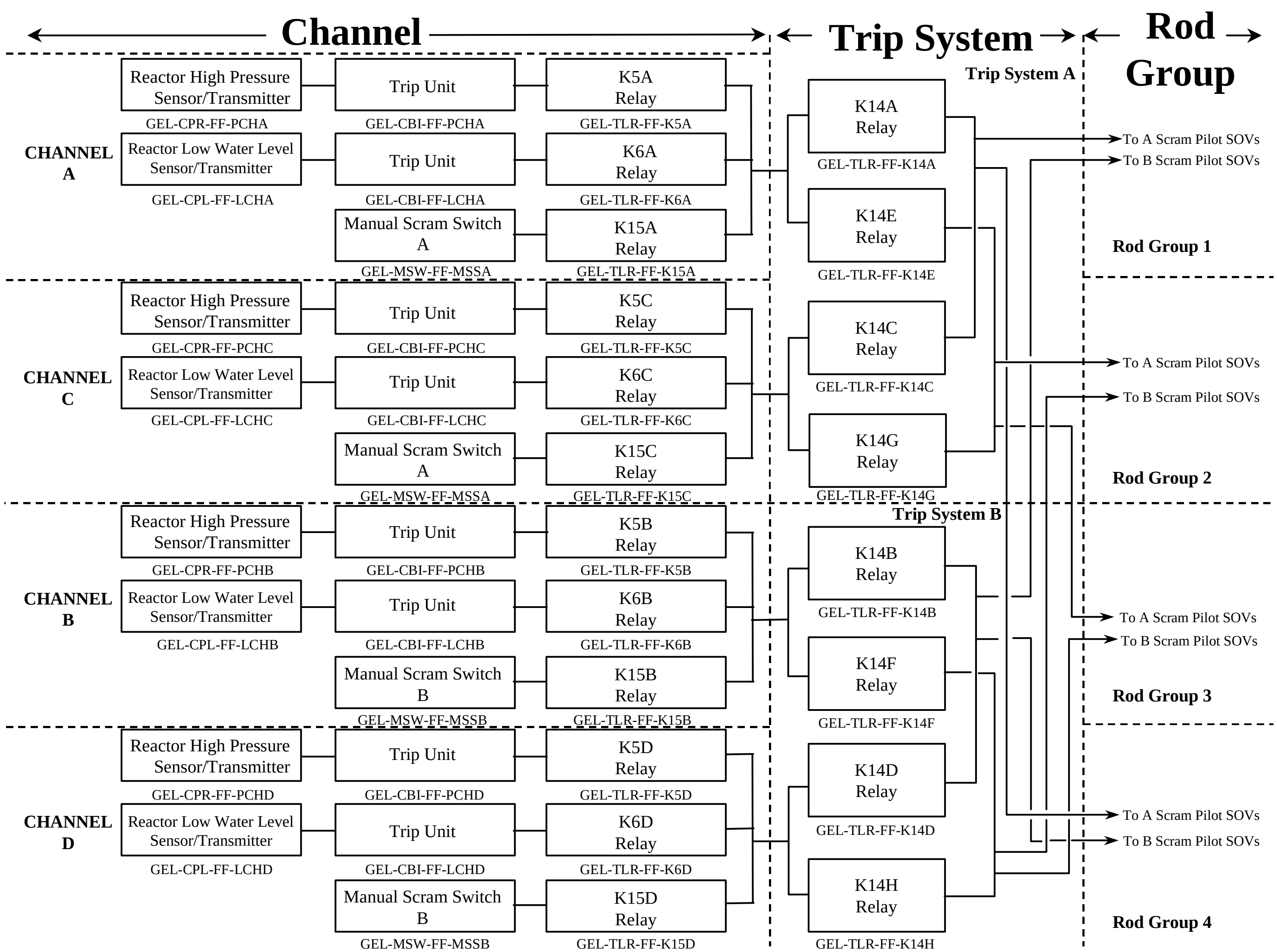


Figure 3. General Electric RPS simplified diagram (scram logic).

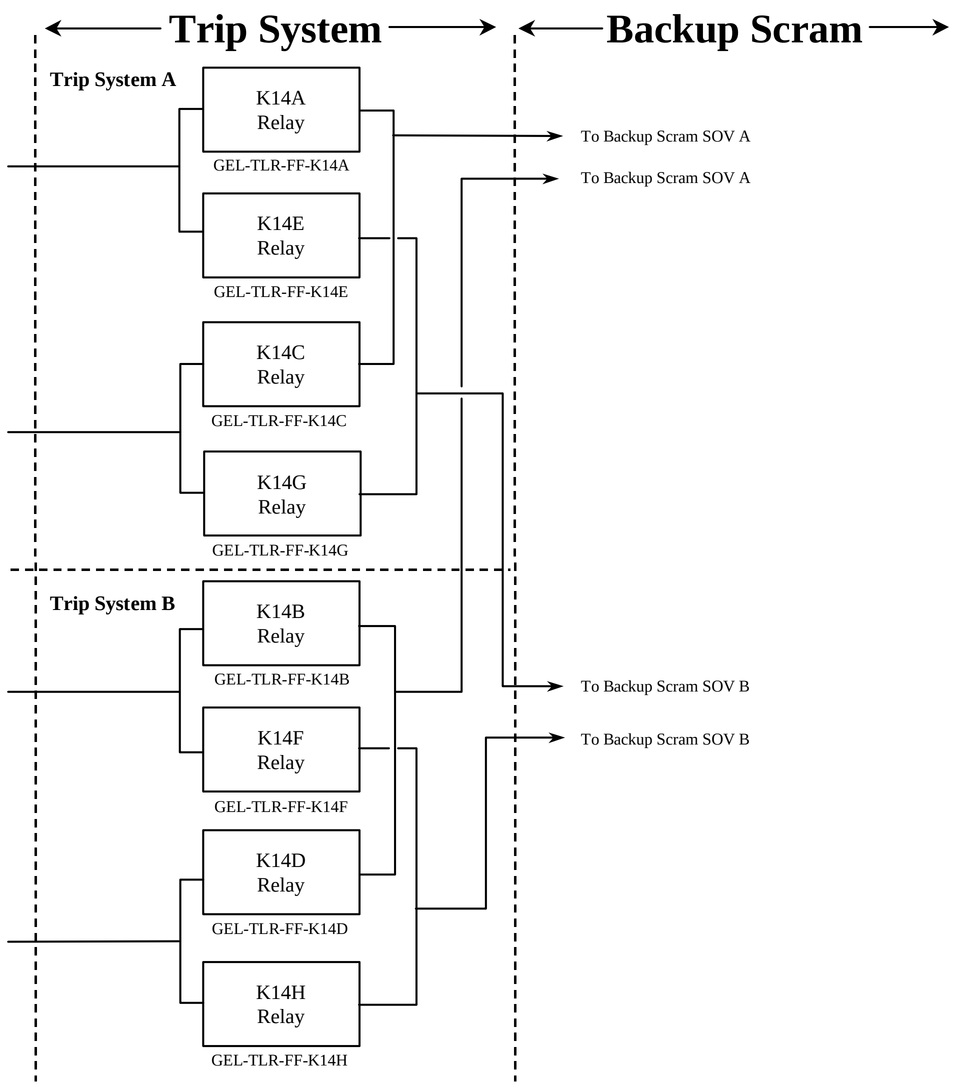


Figure 4. General Electric RPS simplified diagram (backup scram logic).

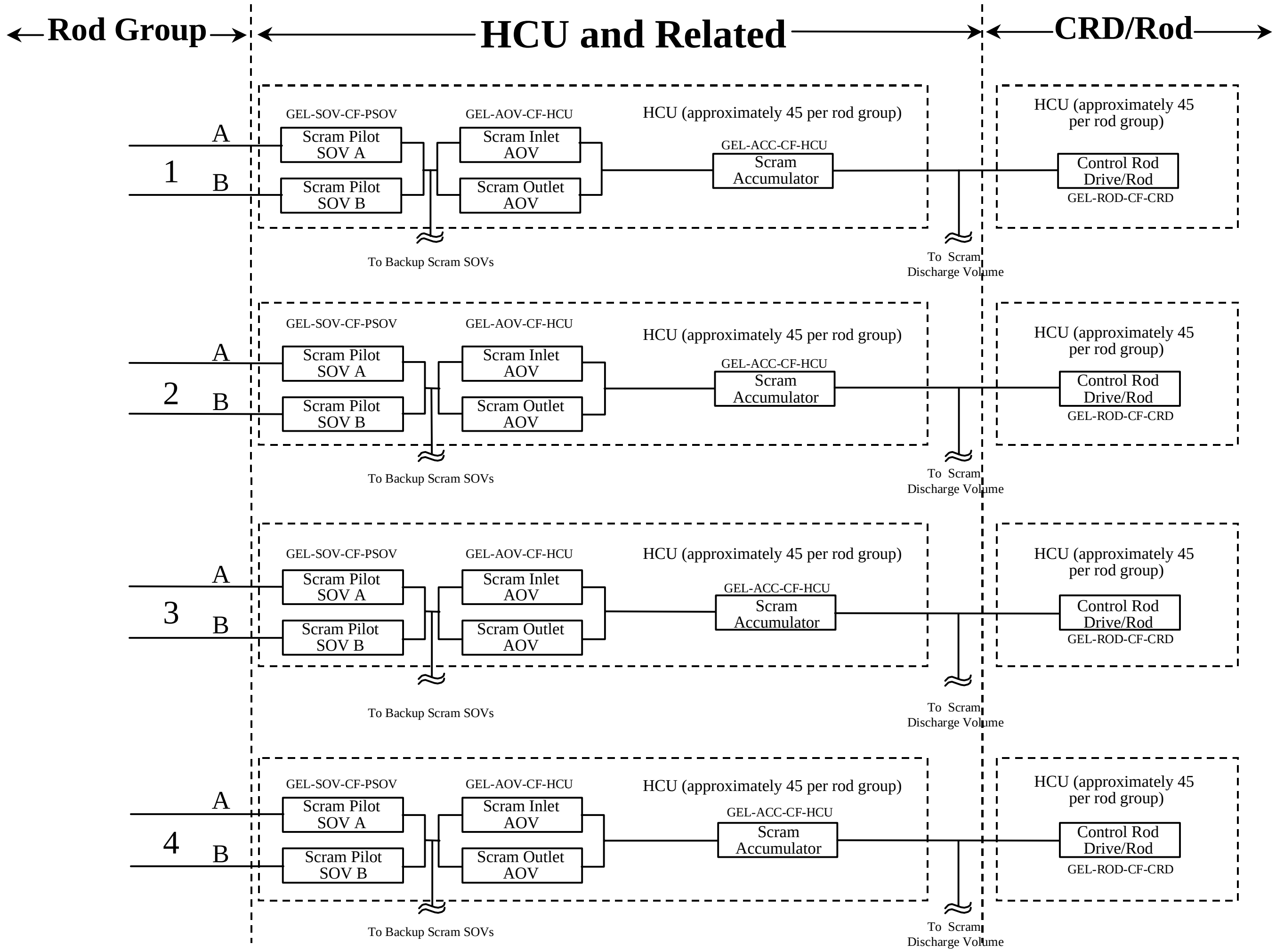


Figure 5. General Electric RPS simplified diagram (mechanical).

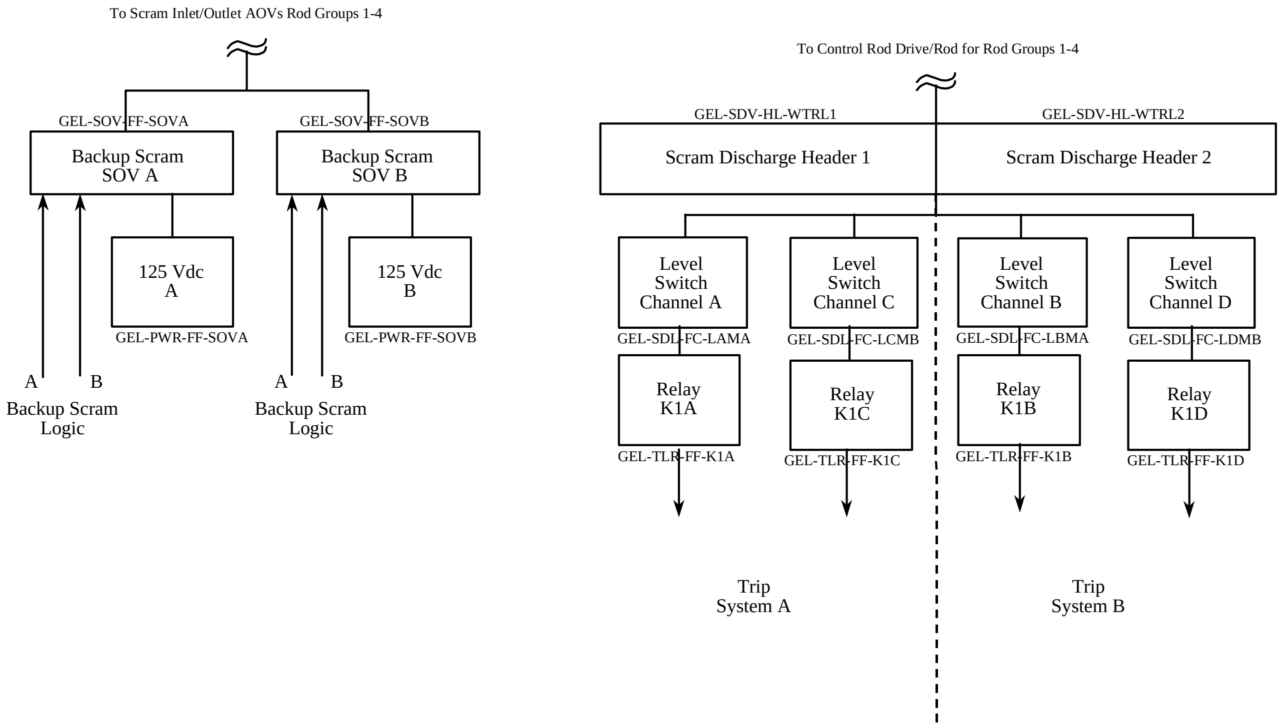


Figure 6. General Electric RPS simplified diagram (SDV and backup scram SOVs).

air from the scram inlet and outlet AOVs, allowing them to open and create a flow path for accumulator water to push the control rods up into the core.

The channel portion of the RPS, channels A through D, includes many different types of trip signals, as shown in Table 1.[10] The trip signals include various neutron flux indications, reactor pressure and level, primary containment pressure, and others. Most of the signals involve four sensor/transmitters (or process switches), with a trip signal being generated if at least one of two measurements associated with each of the two trip systems exceeds a setpoint. This is termed a one-out-of-two-twice logic. Shown in the simplified RPS diagram in Figure 3 are sensor/transmitters and trip units associated with the reactor vessel high pressure and low water level trip signals. (These two signals, along with others, are appropriate for several plant upset conditions, such as main steam line isolation valve or MSIV closure, loss of feedwater, and various losses of electrical loads.) If a trip parameter reaches the trip setting, the trip unit de-energizes the associated relay (shown as relays K5 and K6 in Figure 3). Also shown in the figure are the manual scram switches and associated relays. The sensor/transmitter and trip unit components are located throughout the plant, while the relays are located in the two RPS cabinets in the control room. A loss of electrical power to a sensor/transmitter or trip unit would result in a trip signal.

The trip system portion of the RPS (Figures 3 and 4) includes two systems or trains, A and B. Channels A and C feed into trip system A, and channels B and D feed into trip system B. De-energizing relay K5A or K6A (or the manual scram relay K15A) in channel A results in de-energizing of contactor relays K14A and K14E. The logic is similar for the other three channel inputs to the trip systems. The scram logic (Figure 3) is arranged such that contactor relay K14A or K14C de-energizes the A scram pilot SOVs in rod groups 1 and 4, while contactor relay K14E or K14G de-energizes the A scram pilot SOVs in rod groups 2 and 3. Therefore, trip system A controls all of the A scram pilot SOVs. Similarly, trip system B controls all of the B scram pilot SOVs. Because both A and B scram pilot SOVs in an HCU must de-energize to result in control rod insertion, the main scram logic is one-out-of-two-twice. For example, a reactor vessel high-pressure signal in channel A and a reactor vessel high-pressure signal in channel B would generate a full reactor trip. However high pressure signals in only channels A and C would generate a half trip (only the A scram pilot SOVs would be de-energized). The trip systems are located in the two RPS cabinets in the control room. A loss of electrical power results in a trip signal from the affected trip system.

Figure 3 also shows four rod group circuits. Each rod group circuit controls one-fourth of the control rods. No RPS components are shown as part of the rod group circuits in the simplified diagram. The rod groups are presented to help illustrate how the rod group success criterion (assumed to be three of four) is associated with the scram logic.

Figure 4 shows the backup scram logic of the trip systems. In contrast to the scram logic, which individually controls the scram air supply inside each HCU, the backup scram controls the instrument air supply to the scram air header feeding all of the HCUs. The scram logic and the backup scram logic both utilize the eight K14 contactor relays. However, the backup scram logic uses different contacts in the relays. De-energizing contactor relay K14A or K14C and contactor relay K14B or K14D energizes backup scram SOV A, which cuts off instrument air supply to the scram air header and bleeds off the header air. Similar logic energizes backup scram SOV B, which also performs the same function. If the scram air header is bled off, then all of the HCUs lose air pressure and the control rods insert. Loss of electrical power to the backup scram SOVs results in failure of the backup scram system.

Figure 5 shows most of the mechanical portion of the General Electric RPS. Within each of the 185 HCUs in the Peach Bottom Unit 2 reference plant, there are two scram pilot SOVs, two scram inlet/outlet AOVs, a scram accumulator, and various other components. If both of the scram pilot SOVs in an HCU are de-energized, then the air supply to the AOVs is bled off. Given loss of air, both the scram

Table 1. Peach Bottom Unit 2 RPS trip signals.

Trip Signal	Trip Logic	Purpose of Trip
1. Intermediate range high neutron flux	1 of 2 twice	Prevent an inadvertent power increase at low power
2. Average power range high neutron flux	1 of 2 twice (6 average power range monitors or APRMs, each with 14 to 22 sensors)	Prevent an inadvertent power increase while at power
3. Nuclear system high pressure [a]	1 of 2 twice	Protect the integrity of the reactor vessel and prevent the addition of significant positive reactivity to the core from steam void collapse
4. Primary containment high pressure	1 of 2 twice	Minimize fuel damage and reduce the addition of energy from the core to the coolant (loss-of-coolant accidents)
5. Reactor vessel low water level [a]	1 of 2 twice	Assure there is sufficient water above the reactor core
6. Turbine stop valve closure	1 of 2 twice (3 of 4 valves must close 15% or more)	Anticipate nuclear system high pressure
7. Turbine control valve fast closure	1 of 2 twice (pressure switches in hydraulic control system)	Anticipate nuclear system high pressure
8. Main steam line isolation	1 of 2 twice (3 of 4 steam lines must have a valve close 15% or more)	Anticipate reactor vessel low water level
9. Scram discharge volume high water level [b]	1 of 2 twice	Ensure the scram discharge volume has sufficient capacity to accommodate CRD water discharge resulting from a scram
10. Main steam line high radiation (disabled in some plants)	1 of 2 twice	Limit the fission products released from the core from gross fuel failure
11. Main condenser low vacuum (not in all plants)	1 of 2 twice	Anticipate turbine stop valve closure; protect main condenser from overpressure
12. Manual scram	2 of 2 switches	Provide the operators with a means to quickly shut down the reactor

a. These two signals are modeled in the RPS fault tree used for this study.

b. The scram discharge volume high water level trip signal is included in the fault tree model only as part of a precursor or conditioning event (undetected high scram discharge volume water level when an unrelated demand for the RPS occurs). This trip signal is not included as a third trip signal for the unrelated demand being modeled.

inlet and outlet AOVs open, allowing a path for scram accumulator water to flow to the CRD (forcing the control rod into the core) and CRD water to drain to the SDV. As a sensitivity case, opening of only the scram outlet AOV was analyzed. In such a case, reactor vessel water pressure (rather than accumulator water pressure) forces the control rod into the core. However, the rod insertion time is longer for this type of operation.

Finally, Figure 6 shows the SDV and associated level instrumentation and the backup scram SOVs. As discussed previously, either of the two backup scram SOVs can cut off the instrument air supply to the scram air header and bleed off the header. These SOVs require electrical power to energize to accomplish this.

The CRD water above the hydraulic piston is exhausted to the SDV. All of the 185 CRDs exhaust to this volume. During normal operation, the SDV drain valves are open and the volume contains no water. However, if for some reason, during normal full-power operation, the drain valves were to close and the SDV started to fill with water, level switches (one-out-of-two-twice logic) trip the reactor before enough water collects to impact the CRDs. (If the SDV were full of water before a reactor scram, then none of the CRDs could exhaust water above the hydraulic pistons, and none of the control rods would insert.)

Finally, the CRDs are hydraulic pistons connected to the bottom of the control rods. There is one HCU for each CRD/control rod.

2.1.2 System Testing

Several different types of tests are performed periodically on the General Electric RPS.[11] First, channel checks are performed every 12 hours. These checks ensure that redundant parameter indications, such as reactor vessel pressure and level, agree within certain limits. These channel checks will identify gross failures in the channel sensor/transmitters.

Channel functional checks are generally performed quarterly (every three months) for all of the trip parameters except for neutron flux. These functional tests cover the channel trip units (or switches) up to the contacts for the associated scram pilot SOVs. During such testing, the channel parameter being tested is generally placed in a bypass condition, so it is not available to generate a trip signal if an actual plant upset condition arises during testing. However, the associated K14 contactor relays are not disabled in terms of responding to trip signals from other channel parameters. These channel functional checks also include calibrations of the trip units. It was also assumed that these functional tests cover the transmitter portion of the sensor/transmitter component shown in Figure 3. The neutron flux channels are generally tested weekly.

Weekly manual scram (or automatic actuator) tests cover the trip system logic. Testing one of the manual scram switches actuates the associated trip system (Figure 3), resulting in a half scram signal. Similar testing of the other switch actuates the other trip system. These tests do not actuate the HCU scram pilot SOVs.

The HCUs and CRD/rods are tested every 18 months during refueling. Also, 10% of the CRD/rods are tested every four months. These are termed single rod scram tests. Such tests cover the scram pilot SOVs, the scram inlet and outlet AOVs, and the scram accumulator, as well as operation of the CRD/rod.

Other types of tests every 18 months during refueling include sensor/transmitter calibrations, RPS timing, and logic system function. It was assumed that the backup scram SOVs are tested every 18 months.

2.1.3 System Boundary

The RPS boundary for this study includes the four segments indicated in Figures 3 through 6: channels, trip systems, HCUs and related components, and CRD/rods. Also included is the control room operator who pushes the manual reactor trip buttons. The ATWS mitigation systems—ARI, SLCS, and ATWS-RPT—are not included.

2.2 System Fault Tree

This section contains a brief description of the General Electric RPS fault tree developed for this study. The actual fault tree is presented in Appendix D. The analysis of the General Electric RPS is based on a representative BWR/4 (Peach Bottom Unit 2) design. As mentioned in Section 2.1.1, this general configuration has been used in generic analyses of General Electric RPSs as representative of most of the various designs and configurations. It should be noted that the RPS fault tree development represents a moderate level of detail, reflecting the purpose of this project—to collect actual RPS performance data and assemble the data into overall RPS unavailability estimates. The level of detail of the fault tree reflects the level of detail available from the component failure information in NPRDS and the LERs.

The top event in the RPS fault tree is “Reactor Protection System (RPS) Fails.” RPS failure at this top level is defined as an insufficient number of control rods inserting (upward) into the core to inhibit the nuclear reaction. Various plant upset conditions can result in differing requirements for the minimum number of control rods to be inserted into the core, and the positions of the control rods within the core can also be important. NUREG-0460 (April 1978) indicates one-third of the control rods can fail to insert (in a random pattern) and still result in a shutdown of the nuclear reaction. Also, report NEDC-30851P (May 1985) indicates that 31% of the control rods can fail to insert (in a random pattern).[12] These two estimates agree closely, and both refer to the achievement of hot shutdown. Therefore, the control rod failure criterion was chosen to be one-third (or more) of the control rods fail to insert.

Within the individual HCUs, air must be removed from the scram inlet and outlet AOVs, both AOVs must open, and the scram accumulator must function. Therefore, the one-third (or more) failure criterion for the control rods also applies to these components. Failure to remove air from the AOVs results if either scram pilot SOV fails to de-energize and both backup scram SOVs fail to energize. As a sensitivity case, it was assumed that only the scram outlet AOV was required to open in order for the control rod to insert. Details of this sensitivity case are presented in Appendix G.

Finally, it was assumed that failure of two of four rod group actuations would result in failure of the scram logic. However, for RPS failure, the backup scram logic would also have to fail.

The level of detail in the RPS fault tree includes sensor/transmitters, trip units and switches, relays, SOVs, AOVs, scram accumulators, control rod drives and control rods, and the SDV. Within the channels, two trip parameters are modeled: reactor vessel high pressure and reactor vessel low water level (see Table 1). These are two parameters that would detect several types of plant upset conditions while the plant is at power. In general, at least three RPS parameters are available to initiate a trip signal for any type of plant upset condition requiring a reactor trip.[12] Only two parameters are included to simplify the fault tree. Note that a sensitivity analysis in Appendix G of this report addresses the potential impacts on the results if three trip parameters were included in the fault tree.

Common-cause failures (CCFs) across similar components were explicitly modeled in the RPS fault tree. Examples of such components include the sensor/transmitters, trip units, process switches, relays, SOVs, AOVs, scram accumulators, and CRD/rods. In general, the common-cause modeling in the

RPS fault tree is limited to the events that fail enough components to fail that portion of the RPS. Lower-order CCF events are not modeled in the fault tree. Such events would have to be combined with independent failures to fail the portion of the RPS being modeled. Such combinations of events (not modeled in the fault tree) were reviewed to ensure that they would not have contributed significantly to the overall RPS unavailability.

Test and maintenance outages and associated RPS configurations are modeled for channel outages. For channel outages, the fault tree was developed assuming that a channel out for testing or maintenance is placed into the bypass mode, rather than a tripped mode.

2.3 Operational Data Collection, Characterization, and Analysis

The RPS data collection, characterization, and analysis process is shown in Figure 7. The major tasks include failure data collection and characterization, demand data collection, and data analysis. Each of these major tasks is discussed below. Also discussed is the engineering analysis of the data. A more detailed explanation of the process is presented in Appendix A.

2.3.1 Inoperability Data Collection and Characterization

The RPS is a system required by technical specifications to be operable when the reactor vessel pressure is above 150 psig (some plants have a 90 psig requirement); therefore, all occurrences that result in the system not being operable are required by 10 CFR 50.73(a)(2)(i)(B) to be reported in LERs. In addition, 10 CFR 50.73(a)(2)(vii) requires the licensee to report all common-cause failures resulting in a loss of capability for safe shutdown. Therefore, the SCSS LER database should include all occurrences when the RPS was not operable and all common-cause failures of the RPS. However, the LERs will not normally report RPS component independent failures. Therefore, the LER search was supplemented by the NPRDS data search. NPRDS data were downloaded for all RPS and control rod drive system records for the years 1984 through 1995. The SCSS database was searched for all RPS failures for the same period. In addition, the NRC’s Performance Indicator database was used to obtain a list of unplanned RPS demands (reactor trips).

The NPRDS reportable scope for RPSs and control rod drive systems includes the components modeled in the fault tree described in Section 2.2 and presented in Appendix D, except for the backup scram SOVs. Therefore, the NPRDS data search should identify all RPS component failures except for these SOVs. Failures for control rods, however, are only reported in the NPRDS through March 15, 1994.

In this report, the term inoperability is used to describe any RPS event reported by NPRDS or the LERs. The inoperabilities are classified as fail-safe (FS) or non-fail-safe (NFS) for the purposes of this study. The term NFS is used to identify the subset of inoperabilities for which the safety function of the RPS component was impacted. An example of a NFS event is a failure of the channel trip unit to open given a valid signal to open. The term FS is used to describe the subset of inoperabilities for which the safety function of the RPS component was not impacted. Using the trip unit as an example, a spurious opening of the trip unit is a FS event for the purposes of this study. For some events it was not clear whether the inoperability is FS or NFS. In such cases the event was coded as unknown (UKN).

Inoperability events were further classified with respect to the degree of failure. An event that resulted in complete failure of a component was classified as a complete failure (CF). The failure of a trip unit to open given a valid signal to open is a CF (and NFS) event. Events that indicated some degradation of the component, but with the component still able to function, were classified as no failure (NF). An example

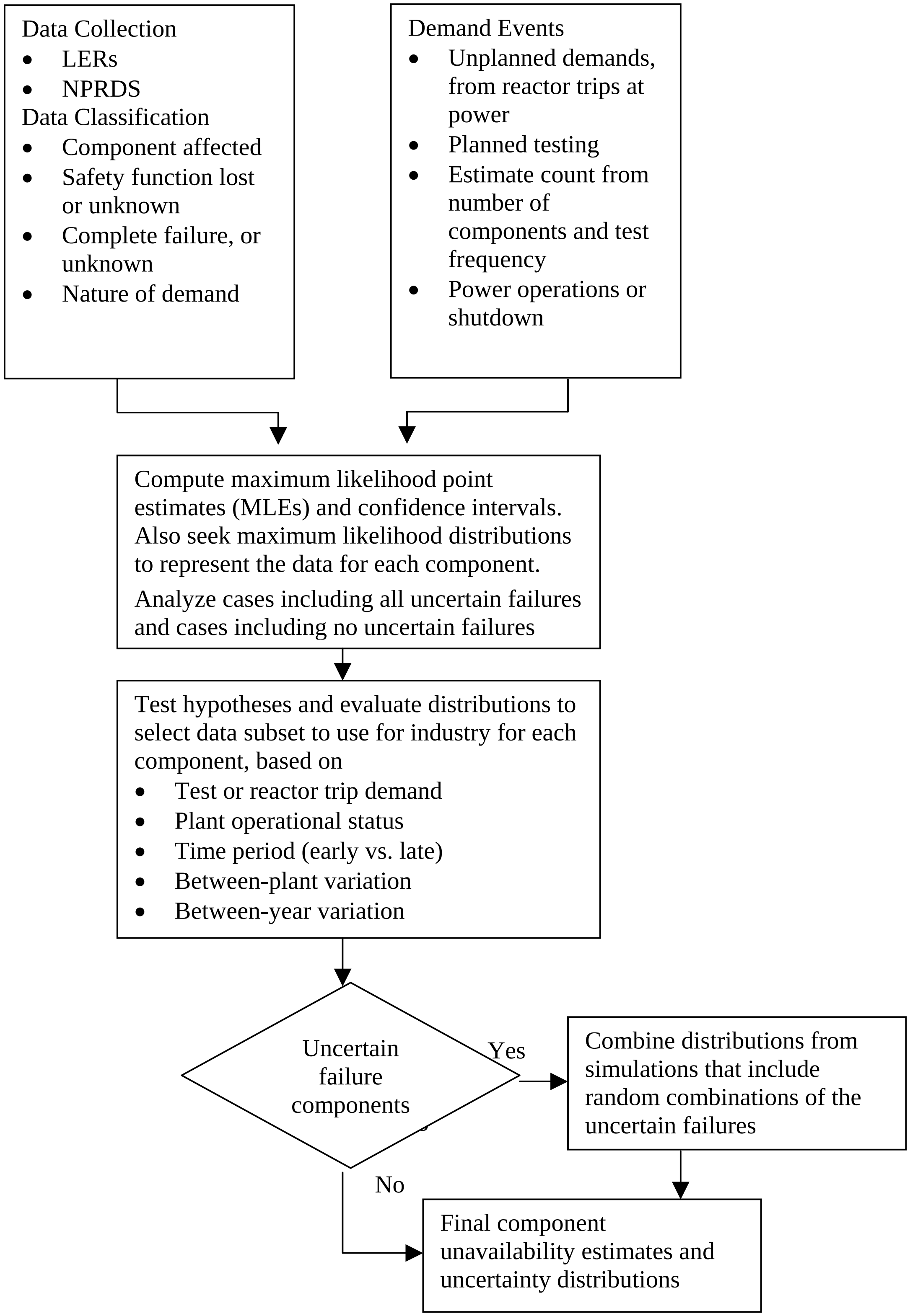


Figure 7. Data collection, characterization, and analysis process.

of a NF event is a trip unit with its trip setting slightly out of specification, but which is still able to open when demanded. For some events is was not clear whether the inoperability was CF or NF. In such cases the event was coded as unknown completeness (UC).

A summary of the data classification scheme is presented in Figure 8. In the figure, there are nine bins into which the data can be placed. These nine bins represent combinations of the three types of safety function impact (NFS, UKN, or FS) and the three degrees of failure completeness (CF, UC, or NF). As indicated by the shaded area in Figure 8, the data classification results in one bin containing non-fail-safe complete failures (NFS/CF) and three bins (NFS/UC, UKN/CF, and UKN/UC) that contain events that are potentially NFS/CF. For these three bins, a lack of information in the data event reports did not allow the data analyst to determine whether the events were NFS/CF. The other five bins do not contain NFS/CF events and generally were not used in the data analysis.

The data characterization followed a three-step process: an initial review and classification by personnel with operator level nuclear plant experience, a consistency check by the same personnel (reviewing work performed by others), and a final, focused review by instrumentation and control and RPS experts. This effort involved approximately 7,000 NPRDS and LER records.

2.3.2 Demand Data Collection and Characterization

Demand counts for the RPS include both unplanned system demands or unplanned reactor trips while the plant is at power, and tests of RPS components. These demands meet two necessary criteria: (1) the demands must be identifiable, countable, and associated with specific RPS components, and (2) the demands must reasonably approximate the conditions being considered in this study. Unplanned reactor trips meet these criteria for the following RPS components: trip system relays (K14s), HCU-related components, and the CRD/rods. However, the reactor trips do not meet the first criterion for channel components, because it is not clear what reactor trip signals existed for each unplanned reactor trip. For example, not all unplanned reactor trips might have resulted from a reactor vessel high pressure.

The RPS component tests clearly meet the first criterion, although uncertainty exists in the association of RPS component failures with particular types of testing. For this report, any failures discovered in testing were assumed to be associated with the specific periodic testing described in Section 2.1.2. Because of the types of tests, the test demands also meet the second criterion, i.e., the tests are felt to adequately approximate conditions associated with unplanned reactor trips.

	Safety Function Impact		
Failure Completeness	NFS/CF (safety function impact, complete failure)	UKN/CF (unknown safety function impact, complete failure; potential NFS/CF)	FS/CF (no safety function impact, complete failure)
	NFS/UC (safety function impact, unknown completeness; potential NFS/CF)	UKN/UC (unknown safety function impact, unknown completeness; potential NFS/CF)	FS/UC (no safety function impact, unknown completeness)
	NFS/NF (safety function impact, no failure)	UKN/NF (unknown safety function impact, no failure)	FS/NF (no safety function impact, no failure)

Figure 8. Data classification scheme.

For unplanned demands, the LER Performance Indicator data describe all unplanned reactor trips while plants are critical. The reactor trip LERs were screened to determine whether the reactor trips were automatic or manual, since each type exercises different portions of the RPS. For RPS component tests, demands were counted based on component populations and the testing schedule described in Section 2.1.2. More details on the counting of demands are presented in Appendix A.

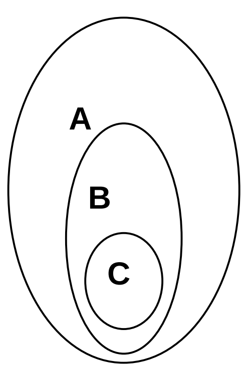
2.3.3 Data Analysis

In Figure 7, the data analysis steps shown cover the risk-based analysis of the operational data, leading to the quantification of RPS unavailability. Not shown in Figure 7 is the engineering analysis of the operational data. The risk-based analysis involves analysis of the data to determine the appropriate subset of data for each component unavailability calculation. Then simulations can be performed to characterize the uncertainty associated with each component unavailability.

The risk-based analysis of the operational data (Section 3) and engineering analysis of the operational data (Sections 4.1 and 4.2) are largely based on two different data sets. The Venn diagram in Figure 9 illustrates the relationship between these data sets. Data set A represents all of the LER and NPRDS events that identified an RPS inoperability. Data set B represents the inoperabilities that resulted in a complete loss of the safety function of the RPS component, or the NFS/CF events (and some fraction of the NFS/UC, UKN/CF, and UKN/UC events). Finally, data set C represents the NFS/CF events (and some fraction of the NFS/UC, UKN/CF, and UKN/UC events) for which the corresponding demands could be counted. Data set C (or a subset of C) is used for the failure upon demand risk-based analysis of the RPS components. Data set C contains all NFS/CF events (and some fraction of the NFS/UC, UKN/CF, and UKN/UC events) that occurred during either an unplanned reactor trip while the plant was critical or a periodic surveillance test.

The purpose of the engineering analysis is to provide qualitative insights into RPS performance. The engineering analysis focused on data set B in Figure 9, which includes data set C as a subset. Data set A was not used for the engineering analysis because the additional FS events in that data set were not judged to be informative with respect to RPS failure to scram, which is the focus of this report.

In contrast to the risk-based analysis of operational data to obtain component failures upon demand, which used data set C, the CCF analysis used data set B. This is appropriate because the CCF analysis is concerned with what fraction of all NFS events involved more than one component. Such an analysis does not require that the failures be matched to demands. The engineering analysis of CCF events, in Section 4, also used data set B.



A RPS inoperabilities identified in NPRDS or LERs

B RPS inoperabilities that are NFS

C RPS NFS events whose demand count could be estimated

Figure 9. RPS data sets.

3. RISK-BASED ANALYSIS OF THE OPERATIONAL DATA

3.1 Unavailability Estimates Based on System Operational Data

If the General Electric RPS is evaluated at the system level, with no consideration of plant-to-plant variations in RPS designs, then a system failure probability can be estimated based on the total system failures and total system demands. For the period 1984 through 1995, there were no total system failures in 1277 demands (unplanned reactor trips). Assuming a Jeffreys noninformative prior and applying a Bayesian update with this evidence results in an RPS mean unavailability of 3.9E-4, with a lower 5^{th} percentile of 1.5E-6 and an upper 95^{th} percentile of 1.5E-3. (See Appendix A for more details on the Bayesian update process. The Jeffreys noninformative prior assumes one-half failure if no failures occurred.) Because no failures occurred, the uncertainty bound on this estimate is broad. Also, the estimate is most likely a conservative upper bound on RPS performance during that period, given previous estimates of RPS unavailabilities (Section 3.3).

This system level, Jeffreys noninformative prior, failure estimate is based on no system failures and a limited number of system demands. Therefore, the unavailability is believed to be conservatively high. In order to obtain a more realistic RPS unavailability estimate with a smaller uncertainty band, an RPS fault tree was developed, as discussed in the following section. That approach could make use of additional RPS component failure data.

3.2 Unavailability Estimates Based on Component Operational Data

3.2.1 Fault Tree Unavailability Results

The General Electric RPS fault tree presented in Appendix D and discussed in Section 2.2 was quantified using the SAPHIRE computer code. [13] Fault tree basic event probabilities are presented in Tables 2 through 4. The basic events are divided into three groups: component independent failure events (Table 2), CCF events (Table 3), and other types of events such as test and maintenance outages and operator errors (Table 4). Failure probabilities for the component independent failures were obtained from the General Electric RPS data as discussed in Sections 2.3 and 2.4. Details of the methodology are discussed in Appendix A, a summary of the data is presented in Appendix B, and the results of the analyses are presented in Appendix C. All of the component independent failure probabilities listed in Table 2 are based on actual General Electric RPS component failure events during the period 1984 through 1995, except for the 125 Vdc power supplies to the backup scram SOVs. However, depending on the results of the data analysis, the failure probabilities may or may not include the following: reactor-trip-related failures and demands, failures while plants are shut down, and 1984 through 1989 data. The component failure probabilities in Table 2 are, in general, comparable to those listed in previous reports listing generic component failure probabilities. [12, 14, and 15] However, the AOV failure probability is significantly lower than previous estimates (obtained from larger size AOV data in other types of safety systems). It is not clear why such a significant difference should exist. However, component boundaries for AOVs sometimes include the associated SOVs that control the air supply to the AOVs. (Inclusion of the SOV within the AOV boundary would significantly increase the AOV failure probability.) In this study those SOVs are modeled separately to more accurately model CCF events.

It should be noted that the backup scram SOVs are not within the reportable scope of the NPRDS database. Therefore, the data search contains no information on these valves. Also, the testing intervals for these valves is uncertain, because they are not classified as safety related. For this report, the backup scram SOVs were assumed to perform comparably to the HCU SOVs in terms of failure probabilities.

Table 2. General Electric RPS fault tree independent failure basic events.

Component Code	Component Type	Fault Tree Basic Event	Number of Failures[a]	Number of Demands or Hours	Modeled Variation[b]	Distribution	Bayes 5%, Mean, 95%	Basic Event Description
ACC	HCU accumulator	None (supports ACC CCF event in fault tree)	1 (0.5)	43883	Sampling	Lognormal	3.3E-6 2.2E-5 6.6E-5	HCU accumulator fails to discharge upon demand to assist the control rods to insert into the core
AOV	HCU scram inlet or outlet air-operated valve	None (supports AOV CCF event in fault tree)	1 (1.0)	522306	Sampling	Lognormal	6.9E-7 2.9E-6 7.2E-6	HCU scram inlet or outlet AOV fails to open upon demand
CBI	Trip unit (bistable)	GEL-CBI-FF-LCHA,B,C,D GEL-CBI-FF-PCHA,B,C,D	7 (4.0)	15026	Year	Lognormal	2.5E-5 2.9E-4 9.7E-4	Channel trip unit (bistable) fails to trip at its setpoint
CPL	Level sensor/ transmitter	GEL-CPL-FF-LCHA,B,C,D	10 (4.9)	6750	Plant	Lognormal	2.4E-5 7.7E-4 2.9E-3	Channel reactor vessel level sensor/ transmitter fails to detect a low level and send a signal to the trip unit
CPR	Pressure sensor/ transmitter	GEL-CPR-FF-PCHA,B,C,D	0 (0.0)	8753	Sampling	Lognormal	5.9E-6 5.7E-5 1.8E-4	Channel reactor vessel pressure sensor/ transmitter fails to detect a high pressure and send a signal to the trip unit
MSW	Manual scram switch	GEL-MSW-FF-MSSA,B	0 (0.0)	38469	Sampling	Lognormal	1.3E-6 1.3E-5 4.2E-5	Manual scram switch fails to operate upon demand
PWR	125 Vdc power to backup scram SOV	GEL-PWR-FF-SOVA,B	NA[c]	NA[c]	NA[c]	Lognormal	2.3E-6 6.0E-5 2.3E-4	125 Vdc power to the backup scram SOV fails (1.0E-5/h*6h repair time)
RDC (ROD and CRD)	Control rod and associated control rod drive	None (supports ROD CCF event in fault tree)	6 (2.7)	62365	Plant	Lognormal	4.6E-6 5.0E-5 1.6E-4	Control rod (or associated control rod drive) fails to insert fully into core upon demand
SDL	Level switch	GEL-SDL-FC-LAMA,LBMA, LCMB,LDMB	4 (3.3)	6075	Plant	Lognormal	5.7E-5 6.1E-4 2.0E-3	Channel (SDV high level) process switch fails to detect a high level and send an appropriate signal to the relay
SDV	Scram discharge volume	GEL-SDV-HL-WTRL1, WTRL2	1 (1.0)	2251	Sampling	Lognormal	1.6E-4 6.7E-4 1.7E-3	As a conditioning event, SDV water level rises too high. If the SDV level instrumentation do not detect this event and cause a scram (modeled separately in the fault tree), then a high SDV water level condition will result.

Table 2. (continued).

Component Code	Component Type	Fault Tree Basic Event	Number of Failures[a]	Number of Demands or Hours	Modeled Variation[b]	Distribution	Bayes 5%, Mean, 95%	Basic Event Description
SOV	HCU scram pilot solenoid-operated valve or backup scram solenoid-operated valve	None (supports SOV CCF event in fault tree) GEL-SOV-FF-SOVA,B	84 (50.1)	77845	Plant	Lognormal	2.4E-5 7.0E-4 2.6E-3	HCU scram pilot SOV (or backup scram SOV) fails to cut off and vent air supply to AOVs
TLR	Relay	GEL-TLR-FF-K1A,B,C,D GEL-TLR-FF-K5A,B,C,D GEL-TLR-FF-K6A,B,C,D GEL-TLR-FF-K14A,B,C,D, E,F,G,H GEL-TLR-FF-K15A,B,C,D	13 (10.8)	579677	Plant	Lognormal	1.7E-6 1.9E-5 6.4E-5	Channel or trip system relay fails to de-energize upon demand

a. Includes uncertain events and CCF events. The number in parentheses is the weighted average number of failures, resulting from the inclusion of uncertain events from data bins NFS/UC, UKN/CF, and UKN/UC (explained in Section 2.3.1).

b. Modeled variation indicates the type of data grouping used to determine the uncertainty bands. For example, for the plant-to-plant variation, data were organized by plant to obtain component failure probabilities per plant. Then the plant failure probabilities were combined to obtain the mean and variance for the component uncertainty distribution. See Appendix A for more details.

c. Power failure data were not analyzed as part of this study. The failure rate per hour was obtained from Reference 14 (Table 4, p. 23). The six-hour repair time was estimated.

Table 3. General Electric RPS fault tree CCF basic events.

Component Code	Component Type	Basic Event(s)	Number of CCF Events	Distribution	Bayes 5%, Mean, 95%	Basic Event Description
ACC	HCU accumulator	GEL-ACC-CF-HCU	3	Lognormal	1.6E-8 1.1E-7 3.2E-7	CCF 33% or more HCU ACCs fail
AOV	HCU scram inlet or outlet air-operated valve	GEL-AOV-CF-HCU	2	Lognormal	6.5E-9 6.9E-9 7.4E-9	CCF 33% or more HCU scram inlet/outlet AOVs fail to open
CBI	Trip unit (bistable)	GEL-CBI-CF-TML3-7	4	Lognormal	1.1E-6 4.2E-6 1.0E-5	CCF specific 3 or more channel CBIs (level T&M)
		GEL-CBI-CF-TMP3-7	4	Lognormal	1.1E-6 4.2E-6 1.0E-5	CCF specific 3 or more channel CBIs (pressure T&M)
		GEL-CBI-CF-TU4-8	4	Lognormal	6.1E-7 3.1E-6 8.2E-6	CCF specific 4 or more channel CBIs
CPL	Level sensor/ transmitter	GEL-CPL-CF-L2-4	16	Lognormal	1.9E-6 7.1E-5 2.7E-4	CCF specific 2 or more CPLs
		GEL-CPL-CF-TML2-3	16	Lognormal	3.2E-6 1.2E-4 4.7E-4	CCF specific 2 or more CPLs (level T&M)
CPR	Pressure sensor/ transmitter	GEL-CPR-CF-P2-4	2	Lognormal	3.4E-7 4.9E-6 1.7E-5	CCF specific 2 or more CPRs
		GEL-CPR-CF-TMP2-3	2	Lognormal	4.5E-7 6.4E-6 2.2E-5	CCF specific 2 or more CPRs (pressure T&M)
MSW	Manual scram switch	GEL-MSW-CF-MSSAB	0	Lognormal	2.3E-8 7.7E-7 2.9E-6	CCF of both MSWs
PWR	125 Vdc power to backup scram SOV	GEL-PWR-CF-PWRAB	NA	Lognormal	2.6E-8 2.1E-6 8.2E-6	CCF 125 Vdc power (SOVs A and B) (No data were collected for this event; the RPS prior was used with no CCF events.)

Table 3. (continued).

Component Code	Component Type	Basic Event(s)	Number of CCF Events	Distribution	Bayes 5%, Mean, 95%	Basic Event Description
SDL	Scram discharge volume level switch	GEL-SDL-CF-HWL2-4	0	Lognormal	1.7E-6 3.1E-5 1.1E-4	CCF specific 2 or more SDV level switches (SDLs)
RDC (ROD and CRD)	Control rod and associated control rod drive	GEL-ROD-CF-CRD	22	Lognormal	2.4E-8 2.5E-7 8.2E-7	CCF 33% or more CRD/rods fail to insert
SOV	HCU scram pilot and backup scram solenoid-operated valves	GEL-SOV-CF-PSOVS	21	Lognormal	5.8E-8 1.7E-6 6.4E-6	CCF 33% or more HCU scram pilot SOVs fail to de-energize and two backup scram SOVs fail to energize
TLR	Relay	GEL-TLR-CF-CH4-8	11	Lognormal	1.2E-8 2.8E-7 1.0E-6	CCF specific 4 or more channel relays (no credit for manual scram by operator)
		GEL-TLR-CF-CHABCD	11	Lognormal	4.6E-9 1.1E-7 4.2E-7	CCF specific 6 or more channel relays (credit for manual scram by operator)
		GEL-TLR-CF-K1-2-4	11	Lognormal	7.5E-8 1.4E-6 4.9E-6	CCF specific 2 or more SDV level relays
		GEL-TLR-CF-TML3-7	11	Lognormal	2.2E-8 3.9E-7 1.4E-6	CCF specific 3 or more channel relays (level T&M) (no credit for manual scram by operator)
		GEL-TLR-CF-TMP3-7	11	Lognormal	2.2E-8 3.9E-7 1.4E-6	CCF specific 3 or more channel relays (pressure T&M) (no credit for manual scram by operator)
		GEL-TLR-CF-TM-LV	11	Lognormal	6.2E-9 1.3E-7 4.9E-7	CCF specific 5 or more channel relays (level T&M) (credit for manual scram by operator)
		GEL-TLR-CF-TM-PR	11	Lognormal	6.2E-9 1.3E-7 4.9E-7	CCF specific 5 or more channel relays (pressure T&M) (credit for manual scram by operator)
		GEL-TLR-CF-TRP4-8	11	Lognormal	1.9E-8 3.8E-7 1.4E-6	CCF specific 4 or more trip system relays

Table 4. General Electric RPS fault tree other basic events.

Basic Event	Distribution	Lower Bound, Mean, Upper Bound	Basic Event Description	Notes
GEL-RPS-TM-ALVL	Uniform	0.0 1.4E-3 2.8E-3	Channel reactor level trip signal bypassed because of testing or maintenance	Assumes 3 hours per quarterly test (outages for each of the four channels combined into channel A).[a] The upper bound assumes 6 hours.
GEL-RPS-TM-APRES	Uniform	0.0 1.4E-3 2.8E-3	Channel reactor pressure trip signal bypassed because of testing or maintenance	Assumes 3 hours per quarterly test (outages for each of the four channels combined into channel A).[a] The upper bound assumes 6 hours.
GEL-XHE-XE-SCRAM	None	1.0 or 1.0E-2	Operator fails to manually actuate RPS	No credit is given for operator action for the base case quantification.

a. From Reference 12, p. 5-17.

The CCF event probabilities in Table 3 are based on the General Electric RPS CCF data during the period 1984 through 1995. However, the CCF event probabilities are also influenced by the prior used in the Bayesian updating of the common cause α parameters. The priors for this study were developed from the overall General Electric RPS CCF database. A summary of the General Electric CCF data is presented in Appendix B, while the actual details of the CCF calculations are in Appendix E. In general, the CCF events reflect multipliers (from the alpha equations) of 0.05 to 0.002 on the component failure probabilities (Q_T's) in Table 2.

The other types of fault tree basic events in Table 4 involve test and maintenance outages and operator error. No credit was taken for operator action to manually actuate the RPS in the base case quantification, so the operator action has a failure probability of 1.0. However, the RPS was also quantified assuming an operator action failure probability of 1.0E-2, which is a typical value used in IPEs.

Using the RPS basic event mean probabilities presented in Tables 2 through 4, the General Electric RPS mean unavailability (failure probability upon demand) is 5.8E-6 with no credit for manual scram by the operator. If credit is taken for manual scram, then the RPS mean unavailability is 2.6E-6. The cut sets from the RPS fault tree quantifications performed using SAPHIRE are presented in Appendix F. Basic event importance rankings are also presented in Appendix F. The dominant failures for the General Electric RPS design involve CCFs of the HCU and backup scram SOVs, channel trip units (bistables), control rods and control rod drives, trip system contactor relays, and channel relays. If credit is taken for manual scram by the operator, then the channel trip unit CCFs are no longer dominant contributors.

RPS segment (HCU, channel, rod, and trip system) contributions to the overall demand unavailability are summarized in Table 5. The channel and HCU failures are dominant.

Another way to segment the General Electric RPS unavailability is to identify the percentage of the total unavailability contributed by independent failures versus CCF events. Such a breakdown is not exact, because RPS cut sets can include combinations of independent failures and CCF events. However, if one assigns all cut sets with one or more CCF events to the CCF category, then the breakdown is clear. The results are presented in Table 6. For the General Electric RPS design, the CCF contribution to overall RPS unavailability is greater than 99.9%. This indicates that the underlying RPS unavailability from independent failures is less than 0.1%, or less than 5.8E-9.

Table 5. General Electric RPS unavailability.

RPS Segment	Unavailability (Point Estimate) with No Credit for Manual Scram by Operator	Unavailability (Point Estimate) with Credit for Manual Scram by Operator
Channel	3.4E-6	1.4E-7
HCU	1.9E-6	1.9E-6
Trip system	3.8E-7	3.8E-7
Rod	2.5E-7	2.5E-7
Total RPS	5.8E-6	2.6E-6

Table 6. General Electric RPS failure contributions (CCF and independent failures).

	No Credit for Manual Scram by Operator		Credit for Manual Scram by Operator	
RPS Segment	Contribution from CCF Events	Contribution from Independent Failures	Contribution from CCF Events	Contribution from Independent Failures
Channel	58%	<0.1%	5%	<0.1%
HCU	32%	<0.1%	71%	<0.1%
Trip system	6%	<0.1%	14%	<0.1%
Rod	4%	<0.1%	10%	<0.1%
Total RPS	>99.9%	<0.1%	>99.9%	<0.1%

Various sensitivity analyses were performed on the RPS fault tree quantification results. These sensitivity analyses are discussed in Appendix G of this report.

3.2.2 Fault Tree Uncertainty Analysis

An uncertainty analysis was performed on the General Electric RPS fault tree cut sets listed in Appendix F. The fault tree uncertainty analysis was performed using the SAPHIRE code. To perform the analysis, uncertainty distributions for each of the fault tree basic events are required. The uncertainty distributions for the basic events involving independent failures of RPS components were obtained from the data statistical analysis presented in Appendix C. The component demand failure probabilities were modeled by lognormal distributions. Note that the component failure rates (per hour) were converted to unavailabilities by multiplying by the repair time (six hours for repair of failure of power to the backup scram SOVs).

Uncertainty distributions for the CCF basic events required additional calculations. Each CCF basic event is represented by an equation involving the component total failure rate, Q_T, and the CCF α's and their coefficients. (See Appendix E for details.) The uncertainty distributions for Q_T were obtained from the statistical analysis results in Appendix C. Uncertainty distributions for the component-specific α's were obtained from the methodology discussed in Appendix E. Each of the α's was assumed to have a beta distribution. The uncertainty distributions for each CCF basic event equation were then evaluated and fit to lognormal distributions. This information was then input to the SAPHIRE calculations.

The results of the uncertainty analysis of the General Electric RPS fault tree model are as follows:

	5%	Median	Mean	95%
No credit for manual scram by operator	1.8E-6	4.6E-6	5.8E-6	1.4E-5
Credit for manual scram by operator	5.2E-7	1.6E-6	2.6E-6	7.7E-6

These results were obtained using a Latin Hypercube simulation with 10,000 samples.

3.3 Comparison with PRAs and Other Sources

Similar to the approaches used in this study, RPS unavailability has been estimated previously from overall system data or from data for individual components within the system. The component approach requires a logic model such as a fault tree to relate component performance to overall system performance. This section summarizes early RPS unavailability estimates using both methods and more recent BWR (General Electric) IPE estimates.

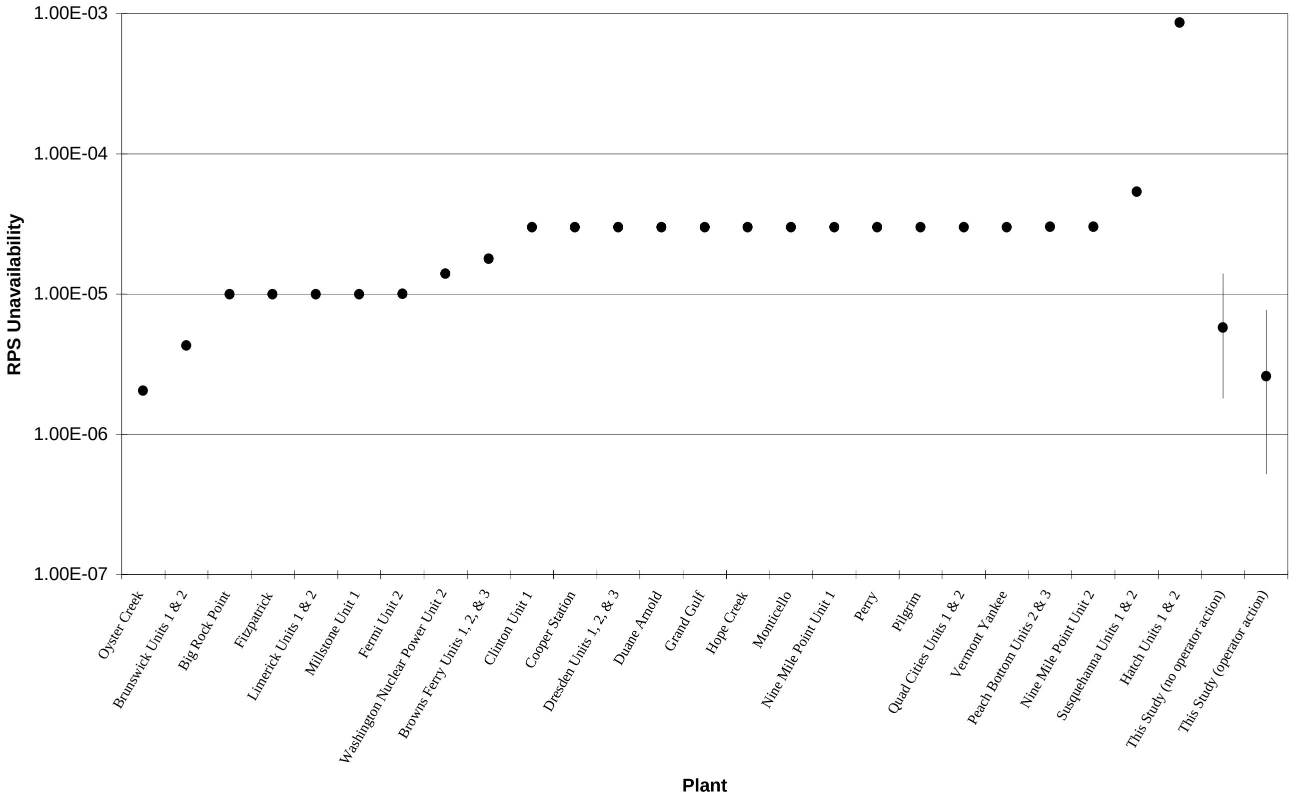
WASH-1270, published in 1973, estimated the RPS unavailability to be 6.9E-5 (median), based on two RPS failures (N-Reactor and German Kahl reactor events) in 1627 reactor-years of operation. Of this combined experience, approximately 1000 reactor-years were from naval reactors. The Electric Power Research Institute (EPRI) ATWS study in 1976 estimated the RPS unavailability to be 7.0E-7 (median), based on no failures in 110,000 reactor trips (75,000 of these were naval reactor trips).[15] Finally, NUREG-0460 in 1978 estimated the RPS unavailability to be 1.1E-4 (median), based on one failure (German Kahl reactor event) in approximately 700 reactor-years. However, that document recommended a value of 3E-5 to account for expected improvements in design and operation, with 1E-5 from the mechanical (rod) portion of the RPS and 2E-5 from the electrical (signal) portion of the RPS. Therefore, early RPS unavailabilities based on system level data ranged from 7.0E-7 (median) to 1.1E-4 (median), depending upon the types of nuclear reactor experience included and the inclusion or exclusion of RPS failure events. Note that these estimates, except for the EPRI study, did not distinguish General Electric RPS designs from pressurized water reactor (PWR) RPS designs.

An early RPS unavailability estimate using component data and fault tree logic models is contained in WASH-1400. WASH-1400 estimated the RPS unavailability to be 1.3E-5 (median). The dominant contributors were rod failures (three or more control rods failing to insert was considered an RPS failure) and channel switch failures. The RPS model used in this report assumed 33% or more of 185 control rods must fail to insert in order to fail to achieve a hot shutdown state, which is a much less conservative failure criterion. This is a major reason why the RPS unavailability presented in this report is much lower than the WASH-1400 result.

Also, General Electric in 1985 analyzed the channel and trip system portion of the RPS (excluding the HCU and control rod portions) and obtained RPS mean unavailabilities of 1.0E-6 to 1.3E-6.[12] The channel and trip system results from the present study indicate an unavailability of 3.8E-6, which is close to the General Electric result. However, the General Electric study did not cover the HCU and control rod portions of the RPS, which contribute 36% to the RPS unavailability in the present study.

Finally, RPS unavailability estimates from the BWR IPEs are presented in Figure 10. The RPS unavailability estimates range from 1.7E-6 (mean) to 8.6E-4 (mean). Details concerning modeling and quantification of the RPS unreliabilities in these IPEs are generally limited. However, most of the IPEs referenced the NUREG-0460 RPS unavailability of 3E-5, consisting of 1E-5 from the mechanical (rod) portion and 2E-5 from the electrical (signal) portion. The IPEs using the 1E-5 unavailability took credit for operator action to bypass failures in the electrical portion of the RPS. The lowest RPS unavailability estimate, 1.7E-6 for Oyster Creek, was obtained from a detailed fault tree model that also took credit for ARI. Rod and scram valve common-cause failures contribute over 99% to the unavailability obtained from that model. The IPE for Hatch does not provide details concerning RPS fault tree models, but the total RPS unavailability of 8.6E-4 is dominated (> 99%) by electrical (signal) failures.

Also shown in Figure 10 are the General Electric RPS unavailability distributions obtained in this study. The mean unavailabilities are 5.8E-6 (no credit for manual scram by the operator) and 2.6E-6 (credit for manual scram). These values lie towards the lower end of the range of the IPE estimates. The control rod and HCU (mechanical portion) of the RPS contribute 2.1E-6 (36% of the total unavailability



Note: The ranges shown for “This Study” are the 5th and 95th percentiles. All other data points are mean values.

Figure 10. BWR IPE RPS unavailabilities.

of 5.8E-6), compared with the NUREG-0460 estimate of 1E-5 (33% of the total of 3E-5). The channel and trip system (electrical portion) contribute 3.8E-6 (64%), compared with the NUREG-0460 estimate of 2E-5 (67%).

3.4 Regulatory Implications

The regulatory history of the RPS can be divided into two distinct areas: general ATWS concerns, and RPS component or segment issues. The general ATWS concerns are covered in NUREG-0460, SECY-83-293, and 10 CFR 50.62. NUREG-0460 outlined the U.S. NRC’s concerns about the potential for ATWS events at U.S. commercial nuclear power plants. That document proposed several alternatives for commercial plants to implement in order to reduce the frequency and consequences of ATWS events. SECY-83-293 included the proposed final ATWS rule, while 10 CFR 50.62 is the final ATWS rule. In those three documents, the assumed General Electric RPS unavailabilities ranged from 1.5E-5 to 6.0E-5. The General Electric RPS unavailability obtained in this report is 5.8E-6, with an upper 95th percentile of 1.4E-5. This value is significantly lower than the values used in the development of the ATWS rule. Because this study did not analyze RPS data from the late 1970s and early 1980s, it is not known what RPS unavailability estimate would have been obtained by this type of study for the ATWS rulemaking period. Therefore, it is not known if the lower RPS unavailability obtained for the period 1984 through 1995 is the result of RPS improvement in performance or a conservatively high RPS estimate in NUREG-0460.

With respect to RPS components or segments, several issues were identified from the document review discussed previously: SDV water level, HCU SOV problems, HFA relay problems, reactor water level instrumentation, and channel test intervals.

The 1980 failure of 76 of 185 control rods to insert at Browns Ferry was the result of too much water in one of the two SDVs during a routine shutdown involving manual scram.[16] A drainage problem existed in the SDVs. Since that event, BWR licensees have been required to review SDV drainage design and to provide for diverse SDV level indication and scram upon high level. A review of the General Electric RPS data indicated that during the period 1984 through 1995, there was only one scram caused by high SDV water level while plants were at power. Therefore, the data indicate few problems with SDV drainage while at power during that period. Also, the data analyses discussed in Section 4 and Appendix C of this report indicate that the SDV level switches (SDLs) have failure probabilities comparable to the other types of RPS process switches (CPSs). Finally, the SDV-related failure contribution to RPS unavailability is less than 1%. All of this information leads to the conclusion that SDV-related RPS problems have not been dominant during the period 1984 through 1995.

Various problems have been identified with SOVs used in General Electric RPS designs, as documented in NUREG-1275.[17] A major problem involved the use of improper seating material, which tended to stick and cause the valves to fail to open or close upon demand. The use of improper liquid thread sealant caused similar problems. Significant CCF events involving the scram pilot SOVs have occurred throughout the period 1984 through 1995, as indicated in Table B-3 in Appendix B. The most important SOV CCF event, in 1984, involved essentially all of the HCU SOVs. This event is the reason that the SOV CCF event contributes 29% to the overall RPS unavailability. The occurrence of significant SOV CCF events throughout the period 1984 through 1995 indicate that this issue has not disappeared.

Various problems with General Electric HFA relays were identified in the 1980s. A summary of problems is presented in NRC Bulletin No. 84-02.[18] Most of the problems involved HFA relays that were normally energized and failed to open when de-energized. This configuration applies to most of the relays in the General Electric RPS design. However, no NRC information notices, bulletins, or generic letters were identified dealing with this problem during the 1990s. Therefore, relay failures have not been dominant contributors to RPS unavailability during the period 1984 through 1995. It should be noted that several types of relays are used in General Electric RPS designs, and the failure data generally did not contain enough information to distinguish the type of relay.

Issues with BWR reactor vessel level instrumentation were discussed in NRC Information Notice 93-89.[19] Most of these issues involved problems with the reference legs used as part of the level instrumentation. Quantification of the General Electric RPS design indicated that level instrumentation failures are not a dominant contributor to RPS unavailability.

Finally, in 1985 General Electric requested approval to change RPS channel testing procedures.[12] In most cases, the channel functional test interval was changed from one month to three months. In addition, during testing the channel could be placed in the bypass mode, rather than the tripped mode. Both of these changes are contained in the current BWR/4 standardized technical specifications.[11] Both of these changes have the potential to increase the unavailability of the RPS. The base case RPS results, obtained with only two trip signals modeled, indicate that the channels contribute approximately 58% to the overall RPS unavailability, assuming no operator action. A sensitivity analysis in Appendix G indicated that if three trip signals had been modeled, the channel contribution would have dropped to approximately 35%. Also, if operator action is credited (failure probability of 1.0E-2), the channels contribute only 5% to the RPS unavailability of 2.6E-6. Although the channel contribution to RPS unavailability is significant, the overall RPS unavailability, 5.8E-6 without operator action and 2.6E-6 with operator action, is low. Therefore, the change from monthly to quarterly testing of the channels does not appear to have adversely impacted RPS unavailability.

4. ENGINEERING ANALYSIS OF THE OPERATIONAL DATA

4.1 System Evaluation

At a system level, the change in RPS performance over time can be roughly characterized by examining the trends with time of component failures and CCFs. A review of the component independent failure counts in Table B-1 of Appendix B indicates a drop in RPS component failures, from a high of 36 failures in 1984 to a low of 10 in 1995. Also, a review of CCF counts in Table B-2 of Appendix B indicates a drop in CCF events over the years, from 23 in 1986 to a low of two in 1995. Both of these trends would seem to support the premise that RPS performance has improved during the period 1984 through 1995. However, detailed analyses of trends with time for component failure probabilities and CCFs, presented in Section 4.3, indicate no trends in events that dominate the RPS unavailability.

The trend in system demands (reactor trips) over time, although not an indicator of RPS unavailability, is one of several indicators of plant safety performance. As indicated in Figure 11, the rate of unplanned reactor trips has dropped approximately 90% over the period 1984 through 1995.

As indicated in Section 3.1, there were no RPS failures during the period 1984 through 1995. This also implies that there were no complete failures of an RPS trip system.

No complete channel failures during unplanned reactor trips were identified during the review of the RPS data. However, because of the complexity and diversity of RPS channels and the uncertainty in determining associated trip signals, it is difficult to determine whether an entire channel failed during an

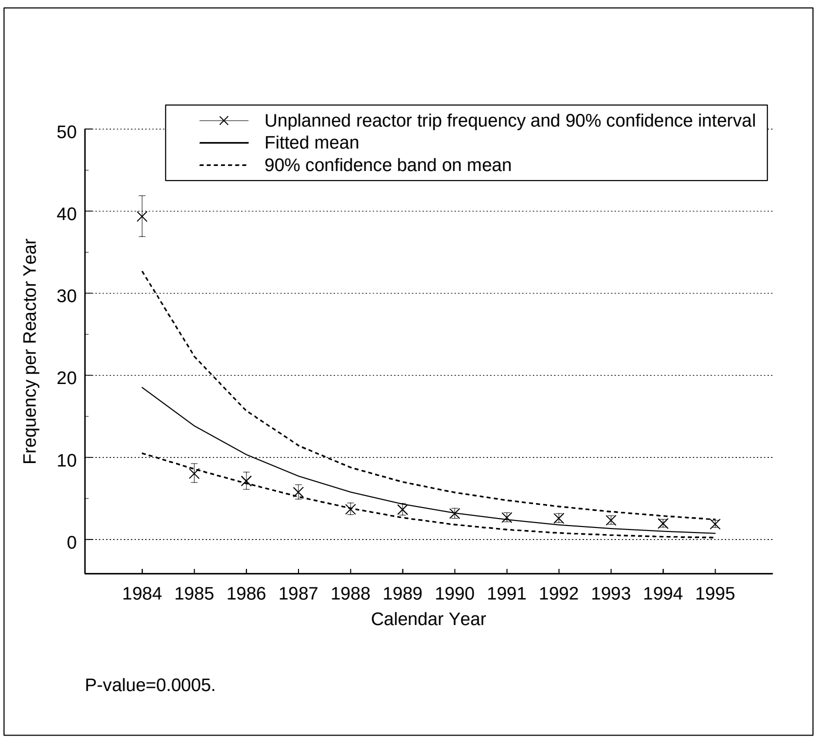


Figure 11. General Electric unplanned reactor trip trend analysis.

unplanned reactor trip. Therefore, it is possible that some complete channel failures might have occurred and were not identified as such in the data review.

4.2 Component Evaluation

Over 7,000 LER and NPRDS records were reviewed for the General Electric RPS study. Data analysts classified these events into the nine bins shown in Figure 8 in Section 2. The highlighted NFS/CF bin contains events involving complete failure of the component's safety function of concern. The other three highlighted bins contain events that may be NFS/CF, but insufficient information prevented the data analysts from classifying the events as NFS/CF. (In the quantification of RPS unavailability discussed in Section 3, a fraction of the events in the three bins was considered to be NFS/CF and was added to the events already in the NFS/CF bin.) General Electric RPS component failure data used in this study are summarized in Table B-1 in Appendix B (independent failures only) and Table C-1 in Appendix C (independent and CCF events).

Approximately 300 to 600 failure events (depending whether CCF events are considered) were identified from the 7,000 events for the period 1984 through 1995. Of this total, approximately 30% are NFS/CF bin events. The remaining 70% are from the three other data bins. The SOVs contribute 30% of the failure events. Other significant components in terms of failure event counts include the RDCs, CPSs, and TLRs. Although none of the component independent failures contribute significantly to the overall RPS unavailability, CCFs of these components are important. Therefore, the independent failures contribute significantly to the RPS unavailability through the associated CCF event probabilities.

The General Electric RPS component data were analyzed for trends with time. The data were analyzed using two sets of data: (1) data from only the NFS/CF bin, and (2) data from all four data bins (with potential NFS/CF events). Results for each year, expressed as frequencies, are the numbers of component failures divided by the numbers of component years. Note that the data analyzed in Section 3 are a subset of the data analyzed in this section. (Section 3 data are generally those associated with countable demands.) Results indicate significant trends over time for only one of the 11 components, RDC. This trend is shown in Figure 12. For these components, the drop from 1984 to 1995 is significant. However, RDC failures are only a minor contributor to RPS unavailability. For the other 10 components, no significant trends were detected.

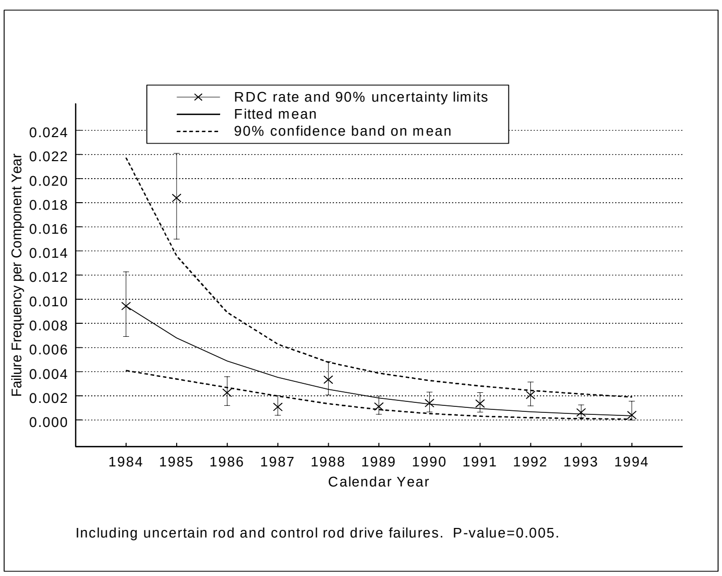


Figure 12. Control rod and control rod drive (combined) failure trend analysis.

4.3 CCF Evaluation

The General Electric RPS CCF data involve CCF and potential CCF events. A complete CCF event involves failure (degradation factor of 1.0) of each of the components in the common cause component group, with additional factors such as shared cause and timing assigned values of 1.0. (See Appendices B and E for additional discussions of the CCF model and failure degradation and other factors.) Additional CCF events involve failure of several (but not all) of the components in the common cause component group. Finally, potential CCFs involve events in which one or more of the degradation or other factors has a value less than 1.0.

General Electric RPS CCF data are summarized in Tables B-2 and B-3 in Appendix B. Approximately 140 CCF and potential CCF events were identified for the period 1984 through 1995. Of that total, approximately 15% are CCF events, with the remaining 85% classified as potential CCF events. However, only one of the CCF events is a complete CCF event, the CPR event in 1987. The rest of the CCF events involve failures ranging from three of four components to 10 of 135. In general, as the size of the component group increases, the significance of the General Electric RPS CCFs decreases.

For the RPS components with large group sizes (ACC, AOV, RDC, and SOV), the most significant CCF events involve potential CCFs of the SOVs. Four of these CCF events involved 183 of 187, 49 of 195, 38 of 195, and 33 of 179 components. All of these events had component failure degradation values of 0.5. For the RPS components with small group sizes, all of the components except the CBIs have significant CCF events.

There are two separate factors contributing to CCF event probabilities: CCF events that are used to calculate the alpha factors; and Q_T, which is the component failure probability due to both independent and common cause factors. In order to identify trends in CCFs, both of these contributors are examined in the following sections. A direct calculation of CCF event probabilities was not performed for each year during the period 1984 through 1995 because the CCF data are generally too sparse for a given year.

4.3.1 CCF Event Trends

All of the CCF events involving the 11 RPS components were analyzed for trends over time. Results for each year, expressed as frequencies, are the number of CCF events divided by the number of reactor years. Two of the component CCF events had decreasing trends with time. The AOV CCF event trend is presented in Figure 13. The other CCF trend is shown in Figure 14, for RDC. Neither of these two components are dominant contributors to overall RPS unavailability, as evaluated in Section 3.2. None of the other component CCF events exhibited statistically significant trends with time over the period 1984 through 1995.

The dominant CCF and potential CCF events with respect to RPS unavailability, as evaluated in Section 3.2, involve the SOVs, CBIs, RDCs, and TLRs. Table B-2 in Appendix B lists 21 SOV CCF and potential CCF events during the period 1984 through 1995. However, only five of these events involved more than 10% of the SOVs. (The failure criterion for the SOVs is 33% or more fail.) The dominant CCF is the 1984 event in which 183 SOVs were affected by improper seating material. Two other similar CCFs occurred in 1994, involving 38 and 33 SOVs. Other problems involved wearout of the SOVs (1991), and the use of improper liquid thread sealant (1994). Although the most significant SOV CCF event occurred in 1984, the other four significant CCFs occurred in the 1990s. Therefore, SOV CCFs are an ongoing concern. Because of the recurrence of several of these failure mechanisms, it is not clear that the lessons learned program is as effective as it should be.

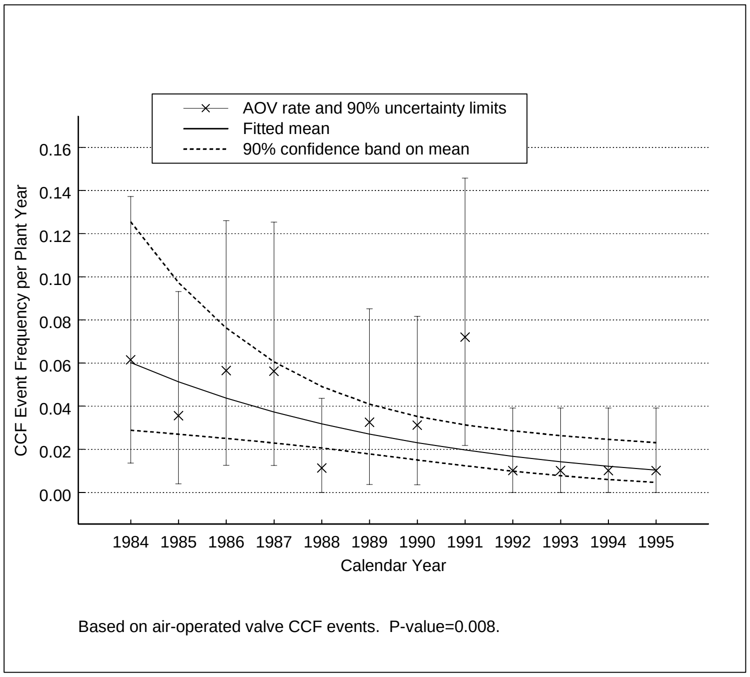


Figure 13. Air-operated valve CCF event trend analysis.

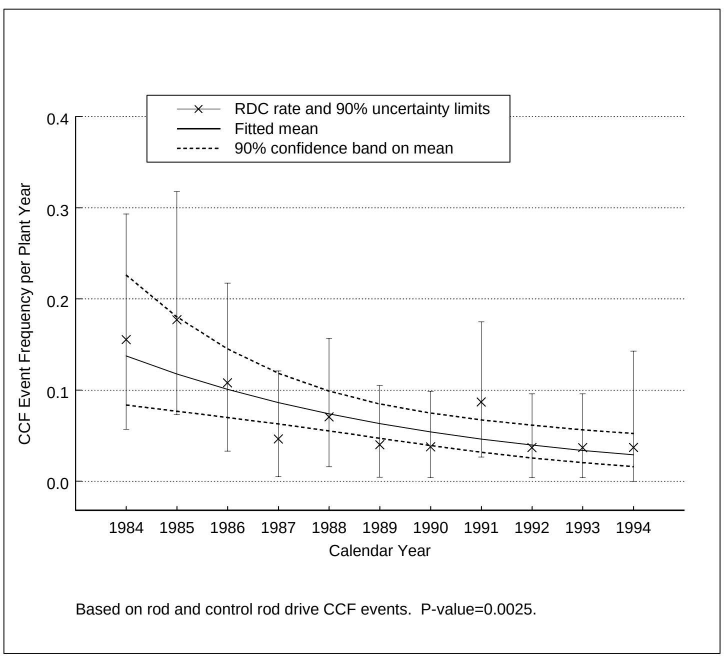


Figure 14. Control rod and control rod drive (combined) CCF event trend analysis.

With respect to the CBIs, only four potential CCF events were identified, as indicated in Table B-2. All four events involved the potential CCF of two trip units. Various trip signals were involved: reactor pressure, reactor level, SDV level, and others. Causes of the potential failures included personnel error, aging and wearout, and corrosion.

Table B-2 in Appendix B lists 22 CCF and potential CCF events for RDC (CRD and ROD). These CCF events included two to 42 components. However, only three events involved more than 10% of the components. These events involving a large number of components had failure completeness values of 0.5 or 0.1, and occurred in 1984 and 1985. Component degradations resulted from aging and wearout. The 1988 event involving 10 control rods had failure completeness values of 1.0. In this event, fuel support plugs were pinching the control rods and causing them to fail to insert. No significant CCF events occurred in the 1990s.

Finally, 11 CCFs and potential CCFs were identified for TLRs, including both channel and trip system relays. These events occurred throughout the period 1984 through 1995, and generally included only two to four relay failures. However, two potential CCF events involved 20 and 58 relays, but with failure completeness values of 0.1. The 58-relay event involved fogging of the relays with an oily substance. The 20-relay event involved improper seating of the sockets. Other relay CCF events included slow opening, cracks, burned coils, smoking, and damage from impact.

4.3.2 Total Failure Probability Trends

Each of the total failure probabilities (Q_T) for the 11 RPS components were analyzed for trends with time. None of the component Q_T's had decreasing trends with time.

As discussed in Section 3.2 of this report, the RPS component failure probabilities obtained from the 1984 through 1995 data are generally comparable to estimates from previous reports. However, the AOV failure probability of 2.9E-6/demand is several orders of magnitude lower than previous estimates. It is not clear why these valves should have a failure probability so much lower than AOVs in other types of safety systems. However, some AOV component boundaries include the associated SOV that controls the air supply to the AOV. Expansion of the AOV boundary to include the associated SOVs would significantly increase the AOV failure probability estimate obtained in this study.

5. SUMMARY AND CONCLUSIONS

A moderately detailed fault tree of the General Electric relay-based RPS was developed and quantified using U.S. General Electric commercial reactor data from the period 1984 through 1995. ATWS mitigation systems such as ARI, SLCS, and ATWS-RPT, were not included in the fault tree model. The RPS fault tree quantification resulted in a mean unavailability of 5.8E-6 (with no credit for manual scram by the operator). The lower 5th percentile value is 1.8E-6 and the upper 95th percentile is 1.4E-5. Channel CCFs contribute 58% to this total unavailability, CCF of the HCU and backup scram SOVs contribute 32%, trip system CCFs contribute 6%, and control rod CCFs contribute 4%. The unavailability estimate of 5.8E-6 is lower than typically used in the past (see Section 3.3). Past estimates typically ranged from 1.0E-5 to 3.0E-5 and were usually based on information in NUREG-0460, published in 1978. The individual component failure probabilities per demand (Table 2), derived from the 1984 through 1995 data, are generally comparable to failure probability estimates listed in previous reports. Therefore, the low RPS unavailability estimate is mostly attributable to lower failure probabilities for the CCF events. The General Electric RPS CCF events collected for this project, covering the period 1984 through 1995, contain few events involving complete failures of many redundant components (Table B-3, Appendix B). Correspondingly, the CCF calculations result in low CCF failure probabilities.

The RPS fault tree was also quantified allowing credit for manual scram by the operator (with a failure probability of 0.01). The resulting RPS unavailability is 2.6E-6. Operator action reduces the RPS unavailability by approximately 55%. This reduction is limited because a dominant contributor to RPS unavailability is the SOV CCF event, which is unaffected by the operator action. Also, the manual scram signal must still pass through the channel and trip system relays, for the configuration analyzed.

Quantification of the CCF events in the RPS fault tree, especially those related to the 33% (or more) failure criterion for the control rods, is complex. The channel and trip system portion of the RPS fault tree contains component group sizes ranging from two to 12, while the control rod and HCU portion contains group sizes of 185 and 370. A prior was developed for each of these two portions of the RPS, based on the overall General Electric RPS data collected. This approach eliminated the need to map failures in a small component group size to much larger group sizes. The prior was then updated using CCF data specific to the component in question. Review of the quantification of the fault tree CCF events indicated that the channel and trip system CCF event probabilities are influenced by many individual CCF events that occurred during the period 1984 through 1995. However, the a dominant CCF event, failure of 33% or more of the HCU and backup scram SOVs, is heavily influenced by a single SOV CCF event that occurred in 1984. (Similar events occurred in the 1990s, but with fewer components affected.)

Several general insights were obtained from this study:

1. CCF events involving the HCU SOVs (and backup scram SOVs) contribute 32% to the overall RPS unavailability. The most significant historical event, involving the use of improper seating material and affecting all of the HCU SOVs, occurred in 1984. Two similar types of SOV CCF events occurred in 1994 but did not affect as many of the components. Several events involving improper use of liquid thread sealant also caused significant CCF events. It is believed that the requirement to test 10% of the control rods each four months helped discover these problems (developing over time) before they developed to catastrophic failures.

2. The backup scram portion of the RPS may be an important contributor to the low RPS unavailability, based on the sensitivity study discussed in Appendix G and uncertainties associated with the SOV failure characteristics. (Without the backup scram logic, only two

of eight trip system relay failures are needed to fail the RPS, rather than four of eight if the backup scram system is modeled.) The backup scram SOVs are classified as non-safety-related, and these valves are not part of the NPRDS reportable scope for the General Electric RPS. Therefore, no failure data were collected for these valves. Also, it is not clear how often these valves are tested, and what their failure probabilities are. This study assumed these valves are tested every 18 months during shutdown, and that their failure characteristics are similar to the HCU SOVs. These assumptions should be verified.

3. The trends in component failure probabilities and numbers of CCF events are generally flat over the period 1984 through 1995, as indicated in Section 4.3 of this report. Therefore, existing RPS surveillance and maintenance practices and industry lessons learned programs have been effective in preventing increasing failure probabilities.

4. There were significant SDV problems in the early 1980s involving both drainage of SDVs and level instrumentation, dominated by the 1980 Browns Ferry Unit 3 failure of 76 of 185 control rods to insert. Data collected during the period 1984 through 1995 indicate that SDV instrumentation failure probabilities are similar to other RPS trip instrumentation (comparison of SDL and CPS in Table C-7, Appendix C of this report). Also, only one inadvertent filling of the SDV while a plant was at power was identified during the period. Finally, the RPS fault tree quantification indicates that SDV events leading to failure of the RPS contribute less than 1% to the overall RPS unavailability. Therefore, early SDV-related problems in General Electric RPSs are no longer dominant contributors to RPS unavailability.

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Appendix A

Data Collection and Analysis Methods

Appendix A

Data Collection and Analysis Methods

To characterize reactor protection system (RPS) performance, operational data pertaining to the RPS from U.S. commercial nuclear power plants from 1984 through 1995 were collected and reviewed. This study, the second in a series, considers the General Electric (GE) boiling water reactor (BWR) plants. Although forty such plants have been licensed, this study excludes three decommissioned plants (Dresden 1, Humboldt Bay 3, and Shoreham), one atypical small plant (Big Rock Point 1), and two plants that were in extended NRC-mandated outages during most of the study period (Browns Ferry 1 and 3).

For the remaining thirty-four plants, reported inoperabilities and unplanned actuations were characterized and studied from the perspective of overall trends and the existence of patterns in the performance of a particular plant. Unlike other operational system studies sponsored by AEOD at the INEEL, the RPS inoperabilities were component failures. Redundancy in the RPS, and interconnections between the RPS channels, trip logic, and hydraulic control units that drive in the control rods, require a more detailed analysis rather than viewing the RPS even at a train level.

Descriptions of the methods for the basic data characterization and the estimation of unavailability are provided below. Situations in which the GE plants were treated differently than the Westinghouse plants in the first report in this series (NUREG/CR-5500, Vol. 2, Reference A-1) are noted. Probabilities coming from the common cause data analysis are explained in Appendix E.

A-1. DATA COLLECTION AND CHARACTERIZATION

In subsections below, methods for acquiring the basic operational data used in this study are described. The data are inoperabilities and the associated demands and exposure time during which the events may occur.

A-1.1 Inoperabilities

Because RPS is a multiple-train system, 10 CFR 50.73 does not require that most failures in RPS components be reported in Licensee Event Reports (LERs). Accordingly, the primary data source for RPS inoperabilities is the Nuclear Plant Reliability Data System (NPRDS). NPRDS failure data were downloaded for components in the RPS and control rod drive systems. Immediate/catastrophic and degraded events were included; incipient events were omitted.

As in the Westinghouse plant study, events prior to 1984 were excluded. The NPRDS failure reporting system changed significantly with the January 1, 1984 institution of the current LER Rule (10 CFR 50.73). The LER rule shifted the emphasis in LER reporting away from single component failures to focus on significant events, leaving NPRDS to cover component failures. Failure reporting to NPRDS is voluntary. As manager of the NPRDS, the Institute for Nuclear Power Operations (INPO) has taken many measures to encourage complete failure reporting to the system since 1984. The NPP industry has relied on the NPRDS for the routine reporting of single component failures since 1984.

To ensure that the failure data set is as complete as possible, the Sequence Coding and Search System (SCSS) LER database was also searched for any RPS inoperabilities reported in LERs.

The NPRDS and SCSS data searches were used to identify events for screening. The major areas of evaluation to support the analysis in this report were as follows:

- What part of the RPS, if any, was affected. Some events pertained to the ATWS mitigation system or to support systems that are not within the scope of the RPS. With one exception, such events were marked as non-failures and were not considered further. The exception is for failures of trip relays for ESF functions. These relays were indistinguishable from failures of the RPS trip relays in the failure data, and were counted both in the failure data set and in the demand data set.

- Do the RPS events affect the performance of the RPS safety function. Failures of indicators and recording devices do not directly affect the ability of the system to provide an automatic trip. Such events were also marked as non-failures and were not considered further.

- For events within the scope of RPS, the specific component affected by the event was indicated. For GE plants, the following distinctions were made (codes for the associated components are in parentheses):

  - Channels (instrumentation rack and bistables): sensors/transmitters and switches [power (CPN), source (CSR) and intermediate range (CIR) neutron detectors, pressure sensor/transmitters (CPR), level sensor/transmitters (CPL), process (CPS) switches, and scram discharge volume level switches (SDL)], radiation detectors (CRA), power supplies (CPW), pressure parameter calculators (CCP), and bistables (BIS).

  - Trains (trip systems): relays (TLR) such as common logic K14 trip relays, and relays from flux trips, manual scram, and channel switches and bistables, the manual scram switch (MSW), and the mode switch (MOD).

  - Control rod drive and control rod components: scram accumulators (ACC), air-operated inlet and outlet valves (AOV), scram pilot and backup scram solenoid-operated valves (SOV), hydraulic control units (HCU), electric protection assembly (EPA), motor-generator set (MGS) and associated 480VAC supply breaker (CB5) and 120VAC output breaker (CB6), control rod drive system filter (FLT) and pump (PMP), and the control rod drive mechanisms (CRD) and control rods (ROD).

- Whether the event contributed to a possible loss of the RPS design safety function of shutting down the reactor. This distinction classifies each inoperability as either a failure or a fault. *Faults* are occurrences that might lead to spurious RPS actuation such as high-pressure set points that have drifted low. *Failures*, on the other hand, are losses at a component level that would contribute to loss of the safety function of RPS (i.e., that would prevent the de-energizing and insertion of the control rods). For the RPS, another way of stating this distinction is that faults are inoperabilities that are fail-safe, while failures are those that are non-fail-safe. The RPS events were flagged as fail-safe (FS), non-fail-safe (NFS), or unknown (UNK). The latter designation applies, for example, when a failure report does not distinguish whether a failed transmitter monitors for high pressure or for low pressure. RPS components such as hydraulic control units (used to finely adjust rod positions), filters, and pumps were found to be fail-safe and thus did not contribute further to the unreliability analysis.

- Whether the event was a common-cause failure (CCF). In this case, several other fields were encoded from the event record: CCF Number, CCF shock type, time delay factor, coupling strength, and a brief event description. These assessments are described further in Appendix B and Appendix E.

- Whether the failure was complete. Completeness is an issue, particularly for failed timing tests and cases where components are out of tolerance but might still perform their safety function if called upon. Completeness is also an issue when component boundary definitions differ and NPRDS reports the complete failure of a component that is a piece part with regard to the RPS fault tree model. The probability of the modeled RPS component functioning given the degradation reported in the LER or NRPDS was assessed 1.0, 0.5, 0.1, or 0.01. These assessments were used in developing impact vectors for the common cause assessment, as discussed in Appendix E. In the basic failure analysis, the 0.5 assessed events were treated as unknown completeness, while the 0.1 and 0.01 assessed events were treated as nonfailures.

- What was the method of discovery of the event [unplanned demand (i.e., reactor trip), surveillance test, other]. For the NPRDS data, "other" includes annunciated events. Failures observed in surveillance tests were sometimes classified according to the test frequency. Unlike data for other safety systems studied in this series of analyses, test frequencies and the corresponding nature of the tests were generally not clear from the event narrative.

  Failures discovered during reactor trips were identified from the LERs and from matching the reactor trip LERs (described in the next section) with the NPRDS failures. Narratives from the few matching records were reviewed. If the failure caused the reactor trip, it was flagged as a fail-safe fault discovered during operations. If it did not cause the reactor trip but was observed during the course of the reactor trip event, it was flagged as being discovered by the reactor trip.

- Whether the plant operational state ("mode") was up or down. All unplanned reactor trip events that are reportable are for critical reactor trips; thus the plant is defined as up for these events. The test events may occur while the plant is up or while it is down. An issue is whether the failure occurrence probabilities (failures per demand) are the same for both situations, and which scenario is the most realistic for the unavailability analysis if they differ. The assessment of plant state for failures during testing and operation was based on the NPRDS and LER narratives, if possible. The data were compared with the outage information used in the NRC Performance Indicator Program to resolve plant state issues in some cases. When the plant state was unknown, it was treated as operating since the plants spend more time in an operating state than shut down.

- The plant and event date for each failure, as presented in the source databases, were preserved and used in the data analysis.

Other attributes were also considered, such as the event cause and failure mode. Some of these fields are described in Appendix B. The screening associated with the common cause analysis is described further in Appendix E.

As with Westinghouse, the GE RPS inoperability evaluation differs from previous NRC system operational unreliability studies (References A-2 through A-7) in several aspects. A greater emphasis on common-cause failure analysis applies due to the many redundant aspects of the system. The system redundancy also leads to the use of NPRDS data, since few unplanned reactor trips reveal problems within the RPS itself. That is, unlike the auxiliary feedwater system, the RPS does not have a sufficient failure data set for analysis from just the LERs from unplanned reactor trips. Given the use of NPRDS data and the focus on components rather than trains or segments, the failure completeness issue is more dynamic for the RPS. The inability to distinguish whether a failure is fail-safe adds uncertainty to the data evaluation. Unlike previous NRC system operational unreliability studies, the failure events were not screened to determine if the events were recoverable, since the RPS performs its mission on demand and has no extended mission time. The lack of a mission time also means that there is no need to evaluate the components based on different failure modes, such as starting and running.

The treatment of maintenance unavailability is also different for the RPS than for the previous system studies. Although the SCSS data search included timing codes such as “actual preexisting” and “potential,” both previously detected and not previously detected, incidents of a channel of the RPS being out-of-service for maintenance or testing when demanded during an unplanned reactor trip are not routinely reported. The primary instances found in the data for such preexisting maintenance were when the maintenance contributed to causing a spurious reactor trip and was thus fail-safe. Thus, neither the NPRDS nor the LER data provide information on planned maintenance unavailabilities. Maintenance unavailabilities were included in the fault tree, with their associated impact on the RPS actuation logic. The fraction of time RPS channels, trains, and hydraulic control units are typically in maintenance was estimated directly from the operating procedures rather than from the failure data.

The data characterization for the events was based on reading the associated NPRDS event narratives and LER abstracts. Nearly 7000 NPRDS and LER RPS events occurred at GE plants. Engineers with commercial power plant experience classified the data and reviewed each other’s work for consistency. A final focused review was performed by instrumentation and control and RPS experts on the failure events for those components that were directly used for the GE unreliability study.

Several additional checks and filters were applied to the RPS failure event data:

- For each plant, the data were constrained to lie between the plant’s commercial operation date and its decommission date (if applicable). NPRDS data reporting for a plant begins with its commercial operation date.

- Events and operating time/demands during NRC-enforced *regulatory outages*, as defined in the NRC Performance Indicator (PI) Program, were excluded as being atypical. Among GE plants, this restriction resulted in the exclusion of data from Browns Ferry Unit 1 and Unit 3. It also omitted Browns Ferry 2 data from 9/15/84 to 5/24/91, Peach Bottom 2 from 3/13/87 to 4/25/89, Peach Bottom 3 from 3/31/87 to 12/10/89, and Pilgrim from 4/12/86 to 3/3/89.

- A date check ensured that no control rod demands or events were counted after March 15, 1994, the date on which the NPRDS reporting scope changed to omit these components (among others) from the NPRDS.

- NPRDS and LER data were matched by plant, event date, and component, and were checked to ensure that no event was counted twice.

Further details of the inoperability characterization and database structure are included in Appendix B.

A-1.2 Demands and Exposure Times

For the reliability estimation process, two models are typically used to estimate unavailability. The first is based simply on failures and demands. The probability of failure on demand is estimated simply as the number of failures divided by the number of demands. The resulting estimate is useful if the demands are complete and unbiased, and the counts of demands and failures are complete. This is the primary model used for the components in the RPS.

For the channel pressure and level sensor/transmitters and scram accumulators, however, failures occur other than the ones routinely monitored by testing. These failures are detected either by annunciators or during periodic walk-throughs by plant operators, and thus are not present during the cyclic surveillance tests. The method of discovery thus distinguishes these failures from the others. The downtime for discovering these failures and repairing them is small, typically eight hours or less. To ensure that this contribution to the unavailability is not overlooked, the non-testing failure rate in time is estimated for these components. For each of the three components, a gamma uncertainty distribution for the rate is combined with an eight-hour downtime to obtain an unavailability. If this unavailability is much greater than the unavailability from the demand events, it is used in the fault model quantification. If, on the other hand, it is much smaller, the unavailability estimated from the failures on demand is used. If the two unavailabilities are comparable, they are summed for the fault model quantification.

In the engineering analysis portion of this study, general failure occurrence frequencies in time are estimated for the assessment of trends. These frequencies are based on all the failures and the associated calendar time for the components.

Estimation of both demands and operating times requires knowledge of the number of each type of RPS component at each plant. Estimates of component counts, demands, and operating times are discussed in the next three sections.

A-1.2.1 Component Counts

For each plant, the number of each type of RPS component used in the fault tree was estimated. These component counts are the exposed population of RPS system components installed at each plant that could fail and for which failures would be reported. Table A-1 summarizes the results for the components used in the fault trees. The plant safety analysis reports were reviewed to identify the number and type (digital switch, or analog transmitter or thermal device) of instrumentation for the following: reactor vessel level and pressure, containment pressure, main steam line radiation, main steam isolation valve closure, turbine control and turbine stop valve closure, high and low levels of the scram discharge volume, power and intermediate range flux, main condenser vacuum, control rod drive pressure, and main steam line break. Configurations of the scram pilot solenoid valve (single or dual coil) and scram discharge volume (single or split) were also sought.

Plant-specific engineering records in the NPRDS are intended to provide a profile of the number of components for which failures are to be reported to the NPRDS system. These records were studied to identify component counts, but they were not directly useful for components other than rods and control rod drives because of differences in the component boundary definitions used for this study.

Note that the scram discharge volume component was not directly a part of the database. The volume must be available to receive the contents of the scram accumulators when scrams occur. The air-operated valve failures were reviewed in order to flag any stuck valves that would prevent the draining of the scram discharge volume. Level switch data were also reviewed. The single failure recorded for this study was located using a special search for scrams caused by a high level in the scram discharge volume.

A-1.2.2 Demands

For RPS, the demand count assessment for unavailability estimates based on failures per demand is more uncertain than in previous NRC system studies. In previous NRC system studies, possible sets of demands were considered, such as demands from unplanned actuations of the system and demands from various types of periodic surveillance tests (monthly, quarterly, or cyclic). Demands at plant startup or

Table A-1. Counts per plant for components in GE fault trees.

Acronym	Definition	Count
Channel parameter monitoring instruments and bistables		
CPR	Pressure sensor/transmitter	Plant-specific (0 to 12)[a]
CPL	Level sensor/transmitter	Plant-specific (0 to 12)[a]
CPS	Process switch	Plant-specific (16 to 48)
SDL [b]	Scram discharge volume level switch	Plant-specific (0 to 8)[a]
CBI	Bistable (one per analog trip unit) [c]	Plant-specific (0 to 24)
Trains (trip systems)		
TLR	Relay	Plant-specific (78 to 100)[d]
MSW	Manual switch	2
Control rod drive & rod components		
SOV	Solenoid-operated valve	(1 or 2)*CRD [e]
AOV	Air-operated valve	2*CRD
ACC	Scram accumulator	CRD
SDV	Scram discharge volume	2 [f]
RDC	Control rod drive and rod (combined)	CRD

a. Zero means that the relevant instruments are of the opposite type (transmitters vs. switches).

b. Scram discharge level switches include float and thermal devices. This specialized process switch was evaluated separately for the fault tree.

c. One per CPR, one per CPL, and one per main steam line radiation monitor. Bistables for flux are excluded because failures for these were not found in the NPRDS database.

d. The totals include an assumed 4 relays for neutron flux trips, 24 for ESF functions, 8 for common logic (excluding resets which are fail-safe), and 4 for manual scrams. The rest are for automatic scrams on parameters other than neutron monitoring.

e. CRD: the number of control rod drives. From 87 to 193, plant-specific. Most plants have at least 130. The multiplier for CRD is 2 for single-coil SOVs and 1 for dual-coil scram pilot SOVs. The two backup scram SOVs at each plant were not counted because their failures were not reportable to the NPRDS.

f. GE plants were assumed to have 2 scram discharge volumes unless known otherwise.

shutdown might also be considered. The selection of the sets of events with particular system demands determines the set of failures to be considered in the reliability estimation (namely, the failures occurring during those demands).

In evaluating the possible sets of demands, the following criteria are sought:

1. An ability to count, or at least estimate, the number of demands

2. An ability to estimate the number of failures. Completeness is sought in the failures, so that they will not be underestimated. Conversely, the failures are to be matched with the

demands, so that failures only on the type of demand being considered are counted. Then the number of successes on the type of demand being considered will not be underestimated.

3. The demands need to be complete and rigorous, like an unplanned demand on the system, so that all the relevant failure modes will be tested.

For RPS, the requirement that the demand event set be *countable* is not always met. Although a fairly accurate count of unplanned reactor trips is available from the LERs since 1984, the reactor trips themselves do not exercise the complete RPS. Particularly for the channel components, different reactor trips come from different out-of-bound parameters. For example, the number of unplanned reactor trips for which the reactor vessel high-pressure setpoint was exceeded is unknown. Unplanned reactor trip demand data are not used in this report for channel data since these demands are not countable. For the same reason, unplanned demands are not used for trip logic relays. Unplanned reactor trip demands are not used for the SOVs and the scram accumulators because undetected failures might occur.

Most of the estimates in this report are therefore based on test data. As indicated in the System Testing section in the main text, quarterly testing is believed to apply for most GE RPS channel and train components. The pressure and level sensor/transmitters are also tested during cyclic refueling outages, when the sensors can be checked. The manual scram switches and flux bistables are tested weekly. (Note, however, that the flux bistables are excluded from the analysis because they were not found among the several bistable failures in the failure data set.) Relay testing depends on the signal. Quarterly testing applies to most of the components but the flux relays and manual scram relays are tested weekly. In addition, the common logic relays receive multiple actuations during each test as test signals pass through different sets of contacts. Each common logic relay coil is assumed to receive three actuations with each weekly test (with the manual scram and high and low flux) and eight actuations in each quarterly test (from the testing for an assumed 8 trip signals). The primary control rod drive and rod components are tested during refueling outages. In addition, ten percent of the control rod drive components of each type (other than the backup scram SOVs) are tested on a triannual basis (every four months). No particular testing applies for the scram discharge volume since it is monitored and significant level changes are annunciated in the control room. Based on calendar time and the number of installed components of each type in each plant, estimates for testing demands are calculated for this report.

The completeness of the failure count for the RPS testing data depends on two attributes. First, the failures need to be reported, either through the LERs or NPRDS. In the August 7, 1991 NRC Policy Issue, SECY-91-244, the NRC staff estimated overall NPRDS completeness at 65 to 70 percent, based on a comparison of 1990 NPRDS failure data and component failures that were reported in LERs. As mentioned previously, the LERs themselves are not expected to be complete for RPS failures since single failures on testing are not required to be reported through the LER system. Thus, the failures may be undercounted.

The second attribute probably leads to an overcounting of the RPS testing failures. This attribute concerns the ability to distinguish whether a failure is detected during testing, or, more specifically, during the type of testing being considered. In this regard, the brief NPRDS failure narratives usually are insufficient to distinguish periodic surveillance tests from post-maintenance tests or other types of testing. Since the testing frequency often is not mentioned, no attempt is made in this study to restrict the set of testing failures to a particular type of test. An example of the influence of this uncertainty in the data is that all failures on testing for temperature sensor/transmitters are used in the unavailability analysis, although the testing (calibration) itself occurs on average only once every eighteen months. No attempt has been made in this study to associate the failure times with the plant refueling outage times. This source of uncertainty is not currently quantified.

The completeness of the periodic surveillance testing for RPS components is believed to be adequate, realistically mimicking the demand that an unplanned reactor trip using this portion of the RPS would place on the system. The demands are believed to be rigorous enough that successes as well as failures provide meaningful component performance information. However, in some of the data, differences have been noted between tests that are conducted while the plant is operating and tests conducted during shutdown periods. The failure probability in some cases is observed to be higher during the shutdown periods. This phenomenon is attributed to the additional complications introduced by the maintenance being done during shutdowns, rather than to an inadequacy in the quarterly and monthly testing that occurs at power.

In the remaining subsections of this section, the methods for estimating the various types of demand counts are described.

A-1.2.2.1 Unplanned Demands. As in the Westinghouse RPS study, the NRC Performance Indicator (PI) databases maintained at the INEEL were used as the source for a list of unplanned actuations of the RPS. Unplanned reactor trips have been a reporting requirement for LERs since the 1984 LER rule. The PI databases have been maintained since 1985 and are a reliable source of LER reactor trip data. The databases include manual as well as automatic reactor trips, although only the latter are currently a performance indicator.

Reactor trip data for 1984 were obtained from the Sequence Coding and Search System. Nine LER number lists with associated event dates for 1984 were obtained. Seven corresponded to each combination of three attributes: required vs. spurious reactor trips, automatic vs. manual reactor trips, and during operation vs. during startup (there were no LERs for the combination of manual spurious reactor trips during startup). The other two files described automatic, spurious reactor trips. The eighth file was for LERs reporting reactor trips at a different unit located at the same site as the unit reporting the LER, and the ninth was for LERs reporting multiple reactor trips. These lists were consolidated, and records for a second unit's reactor trip were added for LERs reporting multiple reactor trips including reactor trips at another unit. The plant identifier field was adjusted to the unit with the reactor trip for LERs with single reactor trips at different units. Finally, records with multiple reactor trips at single units were examined. If multiple records were already present (e.g., reflecting a manual reactor trip and an automatic reactor trip on the same date), no changes were made. If no multiple records were present, the demand field (for number of reactor trips) was changed to two. Although uncertainties are associated with this process, since the SCSS did not provide a simple list of reactor trip dates and counts for each unit, the process is believed to be quite accurate.

The unplanned demands were used for the following components in the fault tree: manual switches (manual scrams only), air-operated valves, hydraulic control units, control rods and control rod drives, and the scram discharge volume. In each of these cases, for each plant and year, the number of reactor trips was multiplied by the assumed number of components to get demand counts.

A-1.2.2.2 Surveillance Tests. Weekly, quarterly, and triannual (every four months) test counts were estimated at a plant-year level by assuming 52 weekly tests, four quarterly tests, and three triannual tests per full plant year. On the year of the plant's commercial service date, and the year of the plant's decommission date (if any), the demands were reduced in proportion to the plant's in-service time. The triannual test counts were multiplied by 0.1 since just ten percent of the associated components are assumed to be tested each time.

Cyclic surveillance test demands at a plant level were counted using the NRC's *OUTINFO* database. This database is based on plant Monthly Operations Reports, and is maintained for the NRC PI program. It lists the starting and ending dates of all periods when the main generator is off-line for a period spanning at least two calendar days. Plausible test dates were estimated based on the ending dates for refueling outages. If the period from the startup after a refueling outage to the beginning of the next refueling outage exceeds 550 days (approximately 18 months), then a plausible date for a mid-cycle test is assigned. The resulting dates are summed by plant and year. For the 1984-1985 period for which the refueling outage information is not available, plausible testing dates are projected back in time from known refuelings.

For each type of periodic surveillance test, the estimated plant counts were pro-rated between plant operation time and plant shutdown time. For each plant and year, the outage time represented in the OUTINFO database, including the days on which outages started and ended, was summed. The down time was summed separately and excluded for the six instances among GE plants in the study period for which a regulatory-imposed outage occurred (three Browns Ferry units, two Peach Bottom units, and Pilgrim, as stated near the end of Section A-1.1 above). The remaining time between a plant's low power license date and its decommission date or the study end date was treated as operational (up) time. The demands were then prorated on a plant and year-specific basis; for example, the operational demands were taken to be the total demand times the fraction of the year the plant was up divided by the sum of the up fraction and the shutdown fraction.

For the current study, the time period covers 1984-1995. Outage data for the period prior to 1986, however, are not readily available. The OUTINFO database has gaps for periods prior to 1986. For periods in 1984 and 1985 between a plant's low power license date and the start of OUTINFO data on the plant, the outage and operational data split was estimated by summing the plant's operational and shutdown time from 1986-1995 and prorating the 1984 and 1985 time to reflect the same percentages.

The plant-year demands were multiplied by the number of components to obtain estimates of component demands. After this multiplication, the estimates for demands during shutdown and demands during operations were rounded up to whole numbers.

A-1.2.3 Operating Time

For failure rate assessments, outage time and operational time were estimated in fractions of calendar years for each plant and year, as discussed in the previous section. These fractions were multiplied by the estimated number of components for which failure data has been reported for each plant and year to obtain exposure times in years for operating and shutdown periods for each component type. As needed, these times were converted to hours.

A-2. ESTIMATION OF UNAVAILABILITY

In subsections below, statistical analysis for each separate component is described, and then the combining of failure modes to characterize the total system unavailability and its uncertainty is addressed.

A-2.1 Estimates for Each Failure Mode

The RPS unavailability assessment is based on a fault tree with three general types of basic events: independent failures, common-cause failures (CCF), and miscellaneous maintenance/operator action events.

The CCF modes tend to contribute the most to the unavailability, because they affect multiple redundant components. With staggered testing, the estimation of each CCF probability is a product of a **total** failure event probability (Q_T), and one or more factors derived from the analysis of the failure events as explained in Appendix E.

Since every RPS component involved in the unavailability analysis is in a train whose function is also provided by at least one more train, every component occurs in the CCF events. Therefore, the focus in the individual component analysis for this report was on total failure probabilities rather than probabilities just for independent events. Separate independent estimates with the common cause events removed were not evaluated. Nor were independent probabilities estimated as $\alpha_1 * Q_T$. The fault tree results were reviewed, and the use of Q_T in place of $\alpha_1 * Q_T$ for the independent events generally introduces less than three percent error.

This section addresses the estimation of the total failure probability and its uncertainty for virtually all of the RPS components appearing in the fault tree. For the RPS basic failure data analysis for the unavailability assessment, fifteen failure modes were identified, one for each of twelve component types, with both a demand probability and an unavailability from short-term events based on rates for three of the components. Each estimate is based on the non-fail-safe failures of a particular type of component. Component failure data from the NPRDS and LERs were not available for just one component, namely the power supply to the backup scram solenoid-operated valves. The power supply failures that were in the databases were fail-safe, tending to cause rather than prevent RPS actuation. Generic data were used for these failure estimates for the fault tree. The failure data also do not address the RPS maintenance unavailabilities.

The contribution of the operator is another aspect of the system operation that tends currently to fall outside the scope of the operational data analysis. At the system level, manual reactor trips are a form of recovery from failure of the automatic reactor trip function.

Table A-2 shows the components for which estimates were obtained. It also indicates which data sets might be applicable for each component. For the components marked in the table as operating, both a probability on demand and a rate were estimated. The demand probability was based on the number of tests and the failures discovered during testing, while the rate was based on the remaining failures that were not discovered during testing.

In subsections below, the processes of selecting particular data sets and estimating probability distributions that reflect uncertainty and variation in the data are described. Finally, a simulation method is described for quantifying the uncertainty in whether certain failures were complete losses of the component's safety function.

A-2.1.1 Data-Based Choice of Data Sets

To determine the most representative set of data for estimating each total failure probability or rate, statistical tests were performed to evaluate differences in the following attributes (as applicable):

- Differences in reactor trip data and testing data

- Differences in test results during operations and during shutdown periods (plant mode differences)

- Differences across time. In particular, the twelve-year time frame of the study was separated into two periods, from 1984-1989 and from 1990 to 1995, and differences were evaluated.

The plant operational mode during testing was considered because the duration of RPS maintenance outages during plant operations is limited by plant technical specifications. During plant outages, the technical specifications are much less restrictive, and the tests might be more detailed.

Table A-2. Possible data sets for components in GE reliability study.

Component	Unplanned Trips	Testing	Operating[a]
Channel parameter monitoring instruments and bistables			
Pressure sensor/transmitter (CPR)	Not used[b]	Cyclic (C) & quarterly (Q)	Yes
Level sensor/transmitter (CPL)	Not used	C and Q	Yes
Process switch (CPS)	Not used	Q	No
Scram discharge volume level sw.	Not used	Q	No
Bistable (CBI)	Not used	Q	No
Trains (trip logic)			
Relay (TLR)	Not used	See note c	No
Manual switch (MSW)	Manual trips	W	No
Control rod drive & rod components			
Solenoid-operated valve (SOV)	Not used	See note d	No
Air-operated valve (AOV)	Applicable	C and T(10%)	No
Scram accumulator (ACC)	Not used	C and T(10%)	Yes
Scram discharge volume (SDV)	Applicable	—	No
Control rod drive and rod (RDC)	Applicable	C and T(10%)	No

a. With failures in time that are annunciated or detected at shift change-overs, rather than by testing.

b. Failures detected in unplanned trips are not counted for components that may not be demanded in these trips.

c. For flux trips and manual scrams, W; for ESF and other parameters, Q; for common logic (K14) relays, both. Each common logic relay coil is assumed to receive 3 actuations with each W test (with the manual scram and high and low flux) and 8 actuations in each Q test (from the testing for an assumed 8 trip signals).

d. In addition to cyclic testing, ten percent of the scram pilot SOVs are also tested every four months. This testing frequency is denoted T, for triannual, followed by the percentage tested in parentheses.

Conversely, failure modes, if any, that can only occur during operations might be revealed in the tests conducted during operations.

All the unplanned demands occurred when the reactor was at power. Reactor trip signals passing through the system when the plant is not at power have not been reportable as LERs since mid-1993, and were never performance indicators. Thus, no analysis with regard to plant operating mode was performed for the unplanned demand data set.

The demand and failure data sets were obtained as described in Section A-1. Unlike other recent NRC system studies (References A-2 through A-7), there was no concern that failures of particular components would preclude demands on other components. The changes in demand counts that the few failures discovered in the unplanned demands might make on other RPS components is negligible compared with the total number of demands. In the testing data, failures of particular components would not preclude demands on other components because the tests are conducted on the components individually and are staggered across channels and hydraulic control units.

To determine which data to use in particular cases, each component failure probability and the associated 90% confidence interval was computed separately in each data set. For failures and demands, the confidence intervals assume binomial distributions for the number of failures observed in a fixed number of demands, with independent trials and a constant probability of failure in each data set. For failures and run times, the confidence intervals assume Poisson distributions for the number of failures observed in a fixed length of time, with a constant failure occurrence rate in each data set.

For each applicable failure mode, the hypothesis that the underlying probabilities were the same between the unplanned demand and testing data was tested. In addition, within the testing data sets the operational and shutdown data were compared. When exactly two groups of data with failures and demands were compared, as with these statistical tests, Fisher's exact test (described in many statistics references) was used. In other cases, chi-square tests were used to evaluate the null hypothesis of equal probabilities for a failure mode across data sets from different types of testing or from unplanned events.

As with Fisher's exact test, a premise for these tests is that variation between subgroups in the data be less than the sampling variation, so that the data can be treated as having constant probabilities of failure across the subgroups. When statistical evidence of differences across a grouping is identified, this hypothesis is not satisfied. For such data sets, confidence intervals based on overall pooled data are too narrow, not reflecting all the variability in the data. However, the additional between-subgroup variation is likely to inflate the likelihood of rejecting the hypothesis of no significant systematic variation between data sets, rather than to mask existing differences.

A further indication of differences among the data sets was whether empirical Bayes distributions were fitted for variation between the testing and unplanned demands or between the two plant modes or the two time periods. This topic is discussed further in the next section.

The following guidelines were used to select the data set for the unavailability analysis when differences were found:

1. Where unplanned demands were listed in Table A-2 for a component, they were used, since they were genuine demands on the RPS. However, when differences were observed, in every case the failure rate or probability associated with the unplanned demands was lower than the estimate associated with testing. Due to concerns about the adequacy of reporting the failures that might have been revealed in the reactor trips, applicable testing data were also used. That is, differences between the unplanned and testing data sets were noted but the data were pooled in spite of such differences.

2. Where differences were seen between the operational and shutdown testing data sets, and both were potentially applicable for the component, the operational data set was used. This is the set that corresponds to the goal of the unavailability analysis, which is to quantify RPS unavailability during operations.

3. When differences were found between the older and more recent data, the more recent data set was selected.

These evaluations were not performed in the common cause analysis. The CCF analysis addresses the probability of multiple failures occurring, given a failure, rather than the incidence of failure itself. The CCF data are too sparse for these distinctions.

A-2.1.2 Estimation of Distributions Showing Variation in the Data

To further characterize the failure probability or rate estimates and their uncertainties, probabilities or rates and confidence bounds were computed in each data set for each year and each plant unit. The hypothesis of no differences across each of these groupings was tested in each data set, using the Pearson chi-square test. Often, the expected cell counts were small enough that the asymptotic chi-square distribution was not a good approximation for the distribution of the test statistic; therefore, the computed p-values were only rough approximations for the likelihood of observing as large a chi-square test statistic when no between-group differences exist. The tests are useful for screening, however. Variation in the rates or probabilities from plant to plant or from year to year is identified in order to describe the resulting variation in the unavailability estimates. Identifying the impact of particular plants or years on the estimates is useful in determining whether the results of the unavailability analysis are influenced by possible outliers. The existence of plant outliers is addressed in this report, although the identity of the plants is not since the NPRDS data are proprietary.

Three methods of modeling the failure/demand or failure in time data for the unavailability calculations were employed. They all use Bayesian tools, with the unknown probability or rate of failure for each failure mode represented by a probability distribution. An updated probability distribution, or *posterior* distribution, is formed by using the observed data to update an assumed *prior* distribution. One important reason for using Bayesian tools is that the resulting distributions for individual failure modes can be propagated easily, yielding an uncertainty distribution for the overall unavailability.

In all three methods, Bayes Theorem provides the mechanics for this process. Details are highlighted for probabilities and for rates in the next two subsections.

A-2.1.2.1 Estimation of Failure Probability Distributions using Demands. The prior distribution describing failure probabilities is taken to be a beta distribution. The beta family of distributions provides a variety of distributions for quantities lying between 0 and 1, ranging from bell-shape distributions to J- and U-shaped distributions. Given a probability (p) sampled from this distribution, the number of failures in a fixed number of demands is taken to be binomially distributed. Use of the beta family of distributions for the prior on p is convenient because, with binomial data, the resulting output distribution is also beta. More specifically, if a and b are the parameters of a prior beta distribution, a plus the number of failures and b plus the number of successes are the parameters of the resulting posterior beta distribution. The posterior distribution thus combines the prior distribution and the observed data, both of which are viewed as relevant for the observed performance.

The three methods differ primarily in the selection of a prior distribution, as described below. After describing the basic methods, a summary section describes additional refinements that are applied in conjunction with these methods.

Simple Bayes Method. Where no significant differences were found between groups (such as plants), the data were pooled, and modeled as arising from a binomial distribution with a failure probability *p*. The assumed prior distribution was taken to be the Jeffreys noninformative prior distribution.[A-8] More specifically, in accordance with the processing of binomially distributed data, the prior distribution was a beta distribution with parameters, a=0.5 and b=0.5. This distribution is diffuse, and has a mean of 0.5. Results from the use of noninformative priors are very similar to traditional confidence bounds. See Atwood[A-9] for further discussion.

In the simple Bayes method, the data were pooled, not because there were no differences between groups (such as years), but because the sampling variability within each group was so much larger than the variability between groups that the between-group variability could not be estimated. The dominant variability was the sampling variability, and this was quantified by the posterior distribution from the pooled data. Therefore, the simple Bayes method used a single posterior distribution for the failure probability. It was used both for any single group and as a generic distribution for industry results.

Empirical Bayes Method. When between-group variability could be estimated, the *empirical Bayes* method was employed.[A-20] Here, the prior beta (a, b) distribution is estimated directly from the data for a failure mode, and it models between-group variation. The model assumes that each group has its own probability of failure, p, drawn from this distribution, and that the number of failures from that group has a binomial distribution governed by the group's p. The likelihood function for the data is based on the observed number of failures and successes in each group and the assumed beta-binomial model. This function of a and b was maximized through an iterative search of the parameter space, using a SAS routine.[A-9] In order to avoid fitting a degenerate, spike-like distribution whose variance is less than the variance of the observed failure counts, the parameter space in this search was restricted to cases where the sum, a plus b, was less than the total number of observed demands. The a and b corresponding to the maximum likelihood were taken as estimates of the generic beta distribution parameters representing the observed data for the failure mode.

The empirical Bayes method uses the empirically estimated distribution for generic results, but it also can yield group-specific results. For this, the generic empirical distribution is used as a prior, which is updated by group-specific data to produce a group-specific posterior distribution. In this process, the generic distribution itself applies for modes and groups, if any, for which no demands occurred (such as plants with no unplanned demands).

A chi-square test was one method used to determine if there were significant differences between the groups. But because of concerns about the appropriateness and power of the chi-square test, discomfort at drawing a fixed line between significant and nonsignificant, and an engineering belief that there were real differences between the groups, an attempt was made for each failure mode to estimate an empirical Bayes prior distribution over years and plants. The fitting of a nondegenerate empirical Bayes distribution was used as the index of whether between-group variability could be estimated. The simple Bayes method was used only if no empirical Bayes distribution could be fitted, or if the empirical Bayes distribution was nearly degenerate, with smaller dispersion than the simple Bayes posterior distribution. Sometimes, an empirical Bayes distribution could be fitted even though the chi-square test did not find a between-group variation that was even close to statistically significant. In such a case, the empirical Bayes method was used, but the numerical results were almost the same as from the simple Bayes method.

If more than one empirical Bayes prior distribution was fitted for a failure mode, such as a distribution describing variation across plants and another one describing variation across years, the general principle was to select the distribution with the largest variability (highest 95th percentile). Exceptions to this rule were based on engineering judgment regarding the most logical and important sources of variation, or the needs of the application.

Alternate Method for Some Group-Specific Investigations. The data for each component were modeled by year to see if trends due to time existed. The above methods tend to mask any such trend. The simple Bayes method pools all the data, and thus yields a single generic posterior distribution. The empirical Bayes method typically does not apply to all of the failure modes, and so masks part of the variation. When empirical Bayes distributions are fitted, and year-specific updated distributions are obtained, the Bayes distribution may smooth the group-specific results and pull them towards the generic fitted distribution, thus masking trends.

It is natural, therefore, to update a prior distribution using only the data from the one group. The Jeffreys noninformative prior is suitably diffuse to allow the data to drive the posterior distribution toward any probability range between 0 and 1, if sufficient data exist. However, when the full data set is split into many groups, the groups often have sparse data and few demands. Any Bayesian update method pulls the posterior distribution toward the mean of the prior distribution. More specifically, with beta distributions and binomial data, the estimated posterior mean is $(a+f)/(a+b+d)$. The Jeffreys prior, with $a = b = 0.5$, thus pulls every failure probability toward 0.5. When the data are sparse, the pull toward 0.5 can be quite strong, and can result in every group having a larger estimated unavailability than the population as a whole. In the worst case of a group and failure mode having no demands, the posterior distribution mean is the same as that of the prior, 0.5, even though the overall industry experience may show that the probability for the particular failure mode is, for example, less than 0.1. Since industry experience is relevant for the performance of a particular group, a more practical prior distribution choice is a diffuse prior whose mean equals the estimated industry mean. Keeping the prior diffuse, and therefore somewhat noninformative, allows the data to strongly affect the posterior distribution; and using the industry mean avoids the bias introduced by the Jeffreys prior distribution when the data are sparse.

To do this, a generalization of the Jeffreys prior called the *constrained noninformative prior* was used. The constrained noninformative prior is defined in Reference A-11 and summarized here. The Jeffreys prior is defined by transforming the binomial data model so that the parameter p is transformed, approximately, to a location parameter, ϕ. The uniform distribution for ϕ is noninformative. The corresponding distribution for p is the Jeffreys noninformative prior. This process is generalized using the maximum entropy distribution[A-12] for ϕ, constrained so that the corresponding mean of p is the industry mean from the pooled data, $(f+0.5)/(d+1)$. The maximum entropy distribution for ϕ is, in a precise sense, as flat as possible subject to the constraint. Therefore, it is quite diffuse. The corresponding distribution for p is found. It does not have a convenient form, so the beta distribution for p having the same mean and variance is found. This beta distribution is referred to here as the constrained noninformative prior. It corresponds to an assumed mean for p but to no other prior information. For various assumed means of p, the noninformative prior beta distributions are tabulated in Reference A-11.

For each failure mode of interest, every group-specific failure probability was found by a Bayesian update of the constrained noninformative prior with the group-specific data. The resulting posterior distributions were pulled toward the industry mean instead of toward 0.5, but they were sensitive to the group-specific data because the prior distribution was so diffuse.

Additional Refinements in the Application of Group-Specific Bayesian Methods. For both the empirical Bayes distribution and the constrained noninformative prior distribution using pooled data, beta distribution parameters are estimated from the data. A minor adjustment[A-13] was made in the posterior beta distribution parameters for particular years to account for the fact that the prior parameters a and b are only estimated, not known. This adjustment increases the group-specific posterior variances somewhat.

Both group-specific failure probability distribution methods use a model, namely, that the failure probability p varies between groups according to a beta distribution. In a second refinement, lack of fit to this model was investigated. Data from the most extreme groups (plants or years) were examined to see if the observed failure counts were consistent with the assumed model, or if they were so far in the tail of the beta-binomial distribution that the assumed model was hard to believe. The test consisted of computing the probability that as many or more than the observed number of failures for the group would occur given the beta posterior distribution and binomial sampling. If this probability was low, the results were flagged for further evaluation of whether the model adequately fitted the data. This test was most important with the empirical Bayes method, since the empirical Bayes prior distribution might not be diffuse. See Atwood[A-9] for more details about this test.

Group-specific updates were not evaluated with the simple Bayes approach because this method is based on the hypothesis that significant differences in the groups do not exist.

Note that, for the RPS study, GE generic distributions were sought rather than distributions updated with plant-specific data. Plant-specific evaluations are not in the scope of this study.

A-2.1.2.2 Estimation of Failure Probability Distributions using Operating Time. Failure rates were estimated for the three operating components using the failures that occurred in time, excluding those detected in testing. Chi-square test statistics were computed and Bayesian methods similar to those described above for probabilities were used to characterize the variation in the rates. The analyses for rates are based on event counts from Poisson distributions, with gamma distributions that reflect the variation in the occurrence rate across subgroups of interest or across the industry. The simple Bayes procedure for rates results in a gamma distribution with shape parameter equal to 0.5+f, where f is the number of failures, and scale parameter 1/T, where T is the total pooled running time. An empirical Bayes method also exists. Here, gamma distribution shape and scale parameters are estimated by identifying the values that maximize the likelihood of the observed data. Finally, the constrained noninformative prior method was applied in a manner similar to the other failure modes but again resulting in a gamma distribution for rates. These methods are described further in References A-11 and A-14.

From the rates, failure probability distributions are estimated in the fault tree software. In addition to the gamma distribution for a rate, the software uses an estimate of the average downtime when a failure occurs. For the RPS components, this time is short since the failures are quickly detected and most corrective actions involve simple replacements and adjustments.

A-2.1.2.3 Estimation of Lognormal Failure Probability Distributions. For simplicity, the uncertainty distributions used in the fault tree analysis were lognormal distributions. These distributions produced more stable results in the fault tree simulations, since the lognormal densities are never J- or U-shaped. For both probabilities and rates, lognormal distributions were identified that had the same means and variances as the original uncertainty distributions.

A-2.1.3 Treatment of Uncertain Failures

In the statistical analysis of Section A-1.2.2, uncertainty is modeled by specifying probability distributions for each input failure probability or rate. These distributions account for known variations. For example, a simple event probability calculated from an observed number of events in an observed number of demands will vary as a result of the random nature of the events. The effect of this sampling variation on the system unavailability is modeled in the simple Bayes method.

For the RPS data, however, the number of events itself was difficult to determine from the often-vague NPRDS failure reports. Uncertain information for two particular aspects of the event records has been flagged. The first is whether the safety function was lost. Many of the failure reports for components such as calculators and sensors do not describe their exact usage. The reports often state how the component failed but not whether the nature of the failure would cause a reactor trip or delay a reactor trip. For example, failing high could have either impact depending on the particular process being monitored. In the failure data, the records were marked as safety function lost, not lost, or unknown.

The second source of uncertainty that has had a significant effect on the data for the RPS is whether the failure represents a total loss of function for the component. In the common-cause methodology, the data analyst assesses his or her confidence in whether a failure represents a total loss. The resulting completeness value represents the probability that, among similar events, the component's function would be completely lost. Assessed values of 1.0, 0.5, 0.1, and 0.01 were used in this field. For the uncertainty analysis, records with 1.0 were treated as complete, those with 0.5 were treated as unknown completeness, and those with lesser values were treated as not complete.

Since they were flagged in the data, these two sources of uncertainty in the RPS failure data were explicitly modeled in the RPS study. This section provides further details on the treatment of these uncertainties.

In the RPS modeling, each assessed common cause fraction (alpha) was multiplied by the corresponding total failure probability for the component. This probability was based on the total number of failures (both independent and common cause) that represent complete losses of the safety function of the component. For each component, potentially nine sets of failures could be identified:

1. Non-fail-safe, complete failure (NFS/CF)
2. Fail-safe, complete failure (FS/CF)
3. Unknown safety function impact, complete failure (UKN/CF)
4. Non-fail-safe, no failure (NFS/NF)
5. Fail-safe, no failure (FS/NF)
6. Unknown safety function impact, no failure (UKN/NF)
7. Non-fail-safe, unknown completeness (NFS/UC)
8. Fail-safe, unknown completeness (FS/UC)
9. Unknown safety function impact, unknown completeness (UKN/UC).

Failures in Categories 3, 7, and 9 were, potentially, non-fail-safe complete failures (NFS/CF). Events in Category 1 are NFS/CF.

In past NRC system studies, uncertainties in data classification or the number of failures or demands have been modeled by explicitly assigning a probability for every possible scenario in the uncertain data. The data set for each scenario was analyzed, and the resulting output distributions were combined as a mixture distribution, weighted according to the assigned probabilities. This process was used to account for uncertain demands for system restart in the High Pressure Core Injection Study (Reference A-2), and to account for whether certain failures to run occurred in the early, middle, or late period in the Emergency Diesel Generator Study (Reference A-3). This method has recently become established in the literature (see References A-15 through A-17).

For each component in the RPS study, too many possible combinations of outcomes exist to separately enumerate each one. There are three types of uncertain data, and in some cases over 100 uncertain events for a component. Therefore, the well-known Monte Carlo simulation method was used to assess the impact of the uncertain failures. Probabilities were assigned for whether to treat each set of uncertain failures as complete failures with the safety function lost. After sampling from probability distributions based on the assigned probabilities, the failure probability or failure rate of the RPS component being studied was characterized as described in Section A-2.1.2. This process was repeated 1000 times, and the variation in the output was used to assess the overall uncertainty for the failure probability or failure rate. As with the previous NRC system uncertainty models, the resulting output distributions were combined as a mixture distribution. Since these distributions arise from simulations, they were weighted equally in forming the final output distribution.

More details on the selection of the probabilities, the nature of the simulations, and the combining of the output distributions are provided in subsections below.

A-2.1.3.1 Selection of Uncertainty Distributions. Three uncertainties were considered, corresponding to Categories 3, 7 and 9 in the list above. Probabilities for these events were developed using engineering judgment, as follows.

The average or best estimate of the probability that the safety function was lost (non-fail-safe) was estimated from the data in each data set. Among complete failures, the ratio of the number of events with known safety function lost, to events with safety function either known to be lost or known to be fail-safe, was used for the probability of counting a complete event with uncertain safety function loss. Similarly, among failures with uncertain completeness, a probability of the safety function actually being lost in questionable cases was estimated by the ratio of the number of events with known safety function lost to events with safety function either known to be lost or known to be fail-safe, among events with uncertain completeness.

For the probability that an event with uncertain completeness would be a complete loss of the safety function of the component, 0.5 was the selected mean value. This choice corresponds to the assessments of the engineers reviewing the failure data. For the uncertain events under consideration, the assessment was that the probability of complete function loss among similar events is closer to 0.5 than to 1.0 or to a value less than or equal to 0.1.

In the simulations, beta distributions were used to model uncertainty in these probabilities. More specifically, the family of constrained noninformative distributions described under Alternate Methods in Section A-2.1.2 was selected. For both the probability of the safety function being lost and the probability of complete losses, the maximum entropy distribution constrained to have the specified mean probability was selected. The maximum entropy property results in a broad distribution; for the probability of an event with uncertain completeness being complete the 5th and 95th percentile bounds are, respectively, 0.006 and 0.994. Thus, these distributions model a range of probabilities for the uncertain data attributes.

For events in Category 9, for which both the safety function status and the completeness were unknown, the probability of complete failures with loss of the safety function was taken to be the product of the two separate probabilities. While the completeness and safety function loss status may not be completely independent among events with both attributes unknown, use of the product ensures that the modeled probability for these events will be as low, or lower, than the probability that the events with only one uncertain factor were complete losses of the safety function.

A-2.1.3.2 Nature of the Simulations. The simulations occurred in the context of the ordinary statistical analysis described in Sections A-2.1.1 and A-2.1.2. The first step in completing the analysis was to identify the best data subset, using the methods of Section A-2.1.1. The variation in the data was bounded by completing the analysis of Section A-2.1.1 using two cases:

- Lower bound case: counting no uncertain failures (using only Category 1 data).

- Upper bound case: counting all uncertain failure (i.e., counting all the failures in Categories 3, 7, and 9 as complete losses of the safety function).

When differences were found between data sets in either of these bounding analyses, the differences were preserved for the simulation. That is, a subset was selected to best represent a RPS component's failure probability or failure rate for GE plants if the rules given in Section A-2.1.1 applied in either the upper bound or the lower bound case.

In the simulation, the selected data subset was analyzed using the simple Bayes method and also the empirical Bayes method for differences between plants and years. In each iteration, the data set itself differs according to the number of uncertain failures included. That is, for each selected set of data, the simulation proceeds as follows. First, a simulated number of failures was calculated for each combination of plant, year, plant mode, and method of discovery present in the data. Then, a simple Bayes or empirical Bayes distribution was sought. The results were saved and combined as described in the next subsection.

The calculation of the simulated number of failures was simple. Suppose a cell of data (plant/year/plant operational mode/method-of-discovery combination) had f failures that were known to be complete losses of the safety function, s failures for which the impact on the safety function was unknown, c failures for which the completeness was unknown, and b failures for which both the safety function impact and completeness were unknown. In the simulation, a p_{sc} for complete failures with unknown safety function status and a p_{su} for unknown completeness failures with unknown safety function status were obtained by sampling from the beta distributions discussed above. A p_c was obtained by sampling from the beta distribution discussed above with mean 0.5. A simulated number of failures with the safety function lost among the s failures with unknown impact was obtained by sampling from a binomial distribution with parameters s and p_{sc}. Here, the first parameter of a binomial distribution is the number of opportunities for an outcome, and the second is the probability of the outcome of interest in each independent trial. Similarly, a simulated number of complete failures among the c failures with unknown completeness was obtained by sampling from a binomial distribution with parameters c and p_c. A simulated number of complete failures with safety function lost was generated from among the b failures with both uncertainties by sampling from a binomial distribution with parameters b and $p_{su}*p_c$. The total number of failures for the cell was f plus the values obtained from sampling from the three binomial distributions. This process was repeated for each cell of data.

A-2.1.3.3 Combining Output Distributions. The resulting beta or gamma distributions from the simulation cases were weighted equally and combined to produce distributions reflecting both the variation between plants or other specifically analyzed data sources, and the underlying uncertainty in the two attributes of the classification of the failure data. Two details of this process bear mention.

In some of the simulated data sets, empirical Bayes distributions were not fitted to the data; the maximum likelihood estimates of the empirical Bayes distribution parameters did not exist. An outcome of the simulation was the percentage of the iterations for which empirical Bayes distributions were found. When no empirical Bayes distribution was fit to the simulated data, the simulated data were treated as being homogenous. The simple Bayes method represented the data using the updated Jeffrey's non-informative prior distribution. The mean was taken to be the number of simulated failures plus 0.5, divided by the number of demands plus 1 (for probabilities) or by the exposure time (for rates). The resulting distribution goes into the mix along with the other distributions computed for the attribute under study in the simulations.

For each studied attribute, the simulation distributions were combined by matching moments. A lognormal distribution was obtained that has the same mean and variance as the mixture distribution arising from the simulation.

An option in the last step of this analysis would be to match the mean and the 95th percentile from the simulation instead of the mean and variance. Two lognormal distributions can generally be found that match a specified mean and upper 95th percentile (the error factors are roots of a quadratic equation). For the RPS data, the 95th percentiles from the simulation were relatively low, and the mean and upper bound match led to unrealistic error factors (generally less than 1.5 or greater than 100). Therefore, lognormal distributions that matched the means and variances of the simulation data were used rather than distributions based on the mean and 95th percentiles.

A-2.2 The Combination of Failure Modes

The failure mode probabilities were combined to obtain the unavailability. The primary tool in this assessment was the SAPHIRE analysis of the two fault trees (for plants with analog channels and for plants with digital channels).

Algebraic methods, described briefly here, were used to compute overall common-cause failure probabilities and their associated uncertainties. The CCF probabilities were linear combinations of selected high-order CCF alpha factors, multiplied by the total failure probability or rate coming from the analysis of Section A-2.1. The CCF alpha factors, described in Appendix E, indicate the probability that, given a failure, a particular number of redundant components will fail by common cause. For example, the probability of 6 of 8 components failing depends on the alpha factors for levels 6, 7, and 8. The linear combination of these terms was multiplied by Q_T, the total failure probability, to get the desired common-cause failure probability.

The following algebraic method is presented in more generality by Martz and Waller.[A-18] The CCF probability was an expression of the form

(aX+bY)*Z,

where X, Y, and Z are events or failure modes or alpha factors that each had an uncertainty distribution, and a and b are positive constants between 0 and 1 that reflect a subset of CCF events of a given order meeting the particular criterion of the RPS fault tree. A combined distribution was obtained by repeatedly rewriting the expression using the facts that

Prob(k*A*) = k Prob(*A*) for the subsetting operation,

Prob(*A***B*) = Prob(*A* and *B*) = Prob(*A*)*Prob(*B*), and

Prob(*A*+*B*) =Prob(*A* or *B*) = 1 - Prob(not *A*)*Prob(*not B*) = 1 - [1 - Prob(*A*)]*[1 - Prob(*B*)],

where *A* and *B* are any independent events. Because the resulting algebraic expressions were linear in each of the failure probabilities, the estimated mean and variance of the combination were obtained by propagating the failure probability means and variances. These means and variances were readily available from the beta distributions. Propagation of the means used the fact that the mean of a product is the product of the means, for independent random variables. Propagation of variances of independent factors was also readily accomplished, based on the fact that the variance of a random variable is the expected value of its square minus the square of its mean.

In practice, estimates were obtained by the following process:

- Compute the mean and variance of each beta distribution.

- Compute the mean and variance of the combination for each case using simple equations for expected values of sums for "or" operations and of products for "and" operations.
- Compute parameters for the lognormal distribution with the same mean and variance.
- Report the mean and the 5th and 95th percentiles of the fitted lognormal distribution.

The means and variances calculated from this process were exact. The 5th and 95th percentiles were only approximate, however, because they assume that the final distribution is a lognormal distribution. Monte Carlo simulation for the percentiles is more accurate than this method if enough Monte Carlo runs are performed, because the output uncertainty distribution is empirical and not required to be lognormal.

A-3. METHODS FOR THE TREND ANALYSIS

In addition to the analyses used to estimate system unavailability, the overall frequencies of unplanned demands (reactor trips), total failures for each component, and common cause events for each component were analyzed by year to identify possible trends. Two specific analyses were performed for the three sets of occurrence frequencies. First, the frequencies were compared to determine whether significant differences exist among the calendar years. Frequencies and confidence bounds were computed for each type of frequency for each year. The hypotheses of simple Poisson distributions for the occurrences with no differences across the year groupings were tested, using the Pearson chi-square test. The computed p-values were approximate since the expected cell counts were often small; however, they were useful for screening.

Regardless of whether particular years were identified as having different occurrence frequencies, the occurrence frequencies were also modeled by year to see if calendar trends exist. Least-squares regression analyses were used to assess the trends. A straight line was fitted to the frequency (shown as dots in the plot), and a straight line was also fitted to log(frequency). Thus, the analysis determined whether either the frequency or the log(frequency) was linear with regard to calendar time. The fit selected was the one that accounted for more of the variation, as measured by R^2, provided that it also produced a plot with regression confidence limits greater than zero. The regression-based confidence band shown as dashed lines on the plots applies to every point of the fitted line simultaneously; it is the band due to Working, Hotelling, and Scheffé, described in statistics books that treat linear regression. The paragraphs below describe certain analysis details associated with the frequency trend analyses.

With sparse data, estimated event frequencies (event counts divided by time) were often zero, and regression trend lines through such data often produced negative frequency estimates for certain groups (years). Since occurrence frequencies cannot be negative, log models were important in this analysis. However, an adjustment was needed in order to include frequencies that are zero in this model.

Using $0.5/t$ as a frequency estimate in such cases is not ideal. Such a method penalizes groups that have no failures, increasing only their estimated frequency. Furthermore, industry performance may show that certain events are very rare, so that $0.5/t$ is an unrealistically high estimate for a frequency. A method that adjusts the frequencies uniformly for all the grouping levels (years) and that uses the overall frequency information contained in the industry mean was needed for sparse data and rare events.

As explained in Section A-2.1.2.2, constrained noninformative priors can be formed for frequencies as well as for probabilities. This method met the requirements identified above. Because it also produced occurrence frequencies for each group (each year) in a way that was very sensitive to the data from that one group, it tended to preserves trends that were present in the unadjusted frequency data.

The mean of the updated posterior distribution was used in the regression trending. This process effectively added 0.5 uniformly to each event count, and *T/(2N+1)* to each group exposure time. The additional refinement explained in Section A-2.1.2.2 that adjusts the posterior gamma distribution parameters for particular years to account for the estimation of the prior distribution scale parameter was also applied.

A final trend analysis was performed on the total failure probabilities (Q_T) used in the risk assessment. Common-cause failure probabilities are largely driven by these probabilities, since the CCF probabilities are estimated by multiplying a function of the estimated alpha parameters (which are too sparse for trend analysis) and Q_T. For each component in the risk assessment, uncertainty distributions were estimated for each year using the constrained noninformative prior method. The failures and demands entering this calculation were from the subset used for the Q_T analysis, with the exception that the entire time period was used even for components for which the unreliability estimates were based on data from the 1990-1995 period. The means of the uncertainty distributions were trended, and significant trends were highlighted and plotted using the same regression methods as for the frequencies.

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Appendix B

Data Summary

Appendix B

Data Summary

This appendix is a summary of the data evaluated in the common-cause failure (CCF) data collection effort in support of the General Electric RPS study. Table B-1 lists independent failure counts by type of component from the source data files and is summarized on a yearly basis. Table B-2 lists the CCF failure event counts by type of component from the CCF file and is again summarized on a yearly basis. Table B-3 gives a detailed summary of the CCF events. The data presented in this appendix represent a subset of the data collected and analyzed for this study. The first screening was to exclude data prior to 1984 and to include only data from General Electric plants. The second screening separated out the components of interest for the RPS study. The following list shows the components that are included in this summary and a short description of each:

Component	Component Description
ACC	Hydraulic control unit (HCU) accumulator
AOV	HCU air operated scram inlet and outlet valves
CBI	Channel bistable
CPL	Channel level sensor/transmitter
CPR	Channel pressure sensor/transmitter
CRD	Control rod drive mechanism (one for each HCU)
CPS	Channel process switch
MSW	Manual scram switch
ROD	Control rod
SDL	Scram discharge volume level switch
SOV	Solenoid-operated scram pilot valve
TLR	Trip logic relay

The third screening was for the safety function significance of the failure. The data collection classified failures into three categories: fail-safe (FS), which represents a failure that does not affect the component's safety function; non-fail-safe (NFS), which represents a failure of the component's safety function; and unknown (UKN), which represents a failure that cannot be classified as FS or NFS because of insufficient information concerning the failure. Only those failures designated as NFS or UKN are included in these attachments.

The fourth screening was for the failure completeness (degradation) value. Events were categorized as complete failures (CF)(P=1.0), non failures (NF)(P=0.1 or lower), or unknown completeness (UC)(P=0.5). Events with failure completeness (degradation) values less than 0.5 are excluded from the counts of independent events in Table B-1.

The Table B-3 headings are listed and described below:

Vendor — The vendor of the plant at which the event occurred. Only General Electric (GE) is considered in this report.

FM — Failure mode. The failure mode is a two-character designator describing the mode of failure. The following list shows the failure modes applicable to this report:

FM	Description
IO	Instrument inoperability
IS	Instrument setpoint drift
CO	Breaker fails to open
FO	Functionally failed (applies to RODs)

Completeness Value — This field indicates the extent of each component failure. The allowable values are decimal numbers from 0.0 to 1.0. Coding guidance for different values follows:

1.0 (CF)	The component has completely failed and will not perform its safety function.
0.5 (UC)	The completeness of the component failure is unknown.
0.1 (NF)	The component is only slightly degraded or failure is incipient.
0.01 (NF)	The component was considered inoperable in the failure report; however, the failure was so slight that failure did not seriously affect component function.
0.0	The component did not fail (given a CCF event).
—	No component exists for this group size.

Failures — The number of failure events included in the data record.

Date — The date of the event.

CCF Number — Unique identifier for each common-cause failure event. For this non-proprietary report, the docket number portion of the CCF number has been replaced with 'XXX'.

Description — The description field for the CCF.

Safety Function — Determination of the type of failure as related to the safety function. Allowable entries are NFS, UKN, or FS.

Shock Type — An indication of whether or not all components in a group can be expected to fail. Allowable entries: 'L' for lethal shock and 'NL' for non-lethal.

Time Delay Factor — The probability that two or more component failures separated in time represent a CCF. Allowable values are between 0.1 and 1.0. (Called the Timing Factor in Appendix E.)

Coupling Strength — The analyst's uncertainty about the existence of coupling among the failures of two or more components. Allowable values are between 0.1 and 1.0. (Called the Shared Cause Factor in Appendix E.)

Appendix B has been compiled from several database files that comprise the RPS study data. The file names and a short description are included here for reference:

RPS Data.mdb	LER, NPRDS, and CCF data files
CCF Analysis Code.mdb	Miscellaneous data tables and programs

Table B-1. General Electric RPS independent failure yearly summary, 1984 to 1995.

SYSTEM	ROD														
	Component	Safety Function	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	Total
	ACC	NFS	1	2			1	2		1	1	4	4		16
	ACC	UKN							1						1
	AOV	NFS				1									1
	AOV	UKN				1									1
	CRD	NFS	3		1	2		2	1	1	2	2	1	1	16
	CRD	UKN	10	7	5	2	3	1	3	1	9	1		4	46
	ROD	NFS			1										1
	ROD	UKN							1						1
	SOV	NFS			3		3	4	2	4	1	4	6		22
	SOV	UKN	2			1									3
Summary for 'SYSTEM' = ROD															
Sum			16	9	10	7	7	9	8	7	13	11	11	5	108

Table B-1. (continued).

SYSTEM RPS

Component	Safety Function	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	Total
CBI	NFS	1			3	2		4	1	2	1			14
CBI	UKN	1			1	1	2	2		1		1		9
CPL	NFS		3	1	1			1	1	2	1	1		11
CPL	UKN	2			2	3	1	1	1			1	1	12
CPR	NFS				1		1	1						3
CPR	UKN			1	1	4	1		1		1		1	10
CPS	NFS	3	6	6	5	4	8	5	3	3	4	1		48
CPS	UKN	7	1	3	3	3	1		1	1	1	1	2	24
SDL	NFS		3		1	1	1	1		1		1		9
SDL	UKN			1	1						1			3
TLR	NFS	2		4		3	2	2	2	2	3	2	1	23
TLR	UKN	4	3	1	2	1	5	3				1		20
Summary for 'SYSTEM' = RPS														
Sum		20	16	17	21	22	22	20	10	12	12	9	5	186
Study Total		36	25	27	28	29	31	28	17	25	23	20	10	294

Table B-2. General Electric RPS common-cause failure yearly summary, 1984 to 1995.

SYSTEM	Component	Safety Function	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	Total
ROD															
	ACC	NFS						1			1	1			3
	AOV	NFS			1					1					2
	CRD	NFS	3	4	1		1			2					11
	CRD	UKN	1	1	2	1		1	1	1		1			9
	ROD	NFS					1				1				2
	SOV	NFS	2	2		1		1		4		5	6		21
Summary for 'SYSTEM' = ROD															
Sum			6	7	4	2	2	3	1	8	2	7	6		48

SYSTEM	Component	Safety Function	1984	1985	1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	Total
RPS															
	CBI	NFS				2		1			1				4
	CPL	NFS	2	2	2			3	1			1	1		12
	CPL	UKN			3			1							4
	CPR	NFS				1									1
	CPR	UKN						1							1
	CPS	NFS	3	5	7	2	4	3	4	2	2	2		2	36
	MSW														0
	SDL														0
	CPS	UKN	3	2	5	1	5	2		2	1				21
	TLR	NFS	1		1			2		1	1	1	1		8
	TLR	UKN	1	1					1						3
Summary for 'SYSTEM' = RPS															
Sum			10	10	18	6	9	13	6	5	5	4	2	2	90
Study Total			16	17	22	8	11	16	7	13	7	11	8	2	138

Table B-3. General Electric RPS common-cause failure detailed summary, 1984 to 1995.

Component	Fail Mode	CCF Number	Event Year	Event Description	Safety Function	TDF	Coupling Strength	CCCG	Shock Type	Date	No. Failures (a)	Degraded Value
ACC	FO	N-XXX-89-1220-FO	1989	CHARGING WATER HEADER BALL CHECK VALVE LEAKED	NFS	1.00	1.00	177	NL	1/30/89 1/30/89	1 1	1.00 1.00
ACC	FO	N-XXX-92-1417-FL	1992	BLOWN SEAL BETWEEN THE N2 ACCUMULATOR AND	NFS	0.10	0.50	145	NL	11/25/92 11/22/92	1 1	1.00 1.00
ACC	FO	N-XXX-93-1418-FO	1993	LEAK IN THE N2 SECTION OF THE ACCUMULATOR, INLEAKAGE OF WATE	NFS	1.00	1.00	145	NL	6/14/93	5	1.00
AOV	VO	N-XXX-86-1470-VO	1986	SCRAM INLET AND OUTLET VALVES, LOSS OF SPRING TENSION	NFS	1.00	1.00	258	NL	8/25/86	2	0.10
AOV	VO	N-XXX-91-1409-FO	1991	HYDRAULIC CONTROL UNIT OUTLET VALVES FOUND OUT OF CALIBRATIO	NFS	1.00	1.00	185	NL	4/10/91	3	0.10
CRD	FO	N-XXX-84-1404-FO	1984	CRDS HIGH BUFFER TIME, WORN SEALS AND A BAD BALL CARD	UKN	1.00	0.50	177	NL	1/20/84	26	0.50
CRD	FO	N-XXX-84-1401-FO	1984	CONTROL ROD DRIVE BROKEN STOP PISTON SEALS AND CRUD BUILDUP	NFS	1.00	0.50	137	NL	9/1/84	3	0.10
CRD	FO	N-XXX-84-1332-FO	1984	CRDS HAD A HIGH STALL FLOW	NFS	1.00	0.50	177	NL	1/19/84	11	0.50
CRD	FO	N-XXX-84-1325-FO	1984	CONTROL ROD DRIVE HIGH STALL FLOWS	NFS	1.00	1.00	129	NL	11/4/84 11/4/84	1 1	0.50 0.50
CRD	FO	N-XXX-85-1400-FO	1985	CONTROL ROD DRIVE FAILED DUE TO WORN INNER DRIVE SEALS	NFS	1.00	0.50	137	NL	12/4/85 12/4/85	1 1	0.10 1.00
CRD	FO	N-XXX-85-1331-FO	1985	THE CONTROL ROD DRIVE WAS DEFECTIVE DUE TO WEAR	NFS	1.00	0.50	183	NL	7/11/85	32	0.50
CRD	FO	N-XXX-85-1326-FO	1985	CONTROL ROD DRIVE INDICATED A HIGH STALL FLOW DURING TESTING	NFS	1.00	0.50	185	NL	3/12/85 3/12/85	1 1	0.50 0.50
CRD	FO	N-XXX-85-1317-FO	1985	(CRD) WOULD NOT OBTAIN A STALL FLOW LT 5 GPM	NFS	1.00	0.50	137	NL	5/16/85	12	0.50

Table B-3. (continued).

Component	Fail Mode	CCF Number	Event Year	Event Description	Safety Function	TDF	Coupling Strength	CCCG	Shock Type	Date	No. Failures (a)	Degraded Value
CRD	FO	N-XXX-85-1318-FO	1985	CONTROL ROD DRIVE (CRD) HAD AN OOS WITHDRAWAL TIME	UKN	1.00	0.50	137	NL	8/4/85	4	0.10
CRD	FO	N-XXX-86-1312-FO	1986	CONTROL ROD DRIVE FAILED TO FULLY INSERT	NFS	1.00	1.00	137	NL	6/18/86 6/17/86	1 1	1.00 1.00
CRD	FO	N-XXX-86-1327-FO	1986	CONTROL ROD DRIVE (CRD) DRIFTED OUT OF POSITION	UKN	1.00	1.00	185	NL	11/24/86	4	0.10
CRD	FO	N-XXX-86-1311-FO	1986	IMPROPER SEATING OF BALL CHECK VALVE AND CRUD BUILD UP IN CY	UKN	1.00	1.00	137	NL	3/15/86	42	0.10
CRD	FO	N-XXX-87-1333-FO	1987	CONTROL ROD DRIFTED OUT OF THE REACTOR	UKN	1.00	1.00	185	NL	4/20/87 4/20/87	1 1	0.10 0.10
CRD	FO	N-XXX-88-1313-FO	1988	CONTROL ROD DRIVE FAILED TO FULLY INSERT	NFS	1.00	1.00	137	NL	1/2/88 1/2/88	1 1	1.00 1.00
CRD	FO	N-XXX-89-1402-FO	1989	CONTROL ROD DRIVE (CRD) HAD ITS UNCOUPLING ROD MISALIGNED	UKN	1.00	1.00	185	NL	5/6/89 5/6/89	1 1	0.50 0.50
CRD	FO	N-XXX-90-1405-FO	1990	CONTROL ROD BLADE 22-39 WOULD NOT LOCK INTO CORRESPONDING CO	UKN	1.00	1.00	185	NL	10/9/90 10/9/90	1 1	1.00 1.00
CRD	FO	N-XXX-91-1302-FO	1991	CRD REQUIRED INCREASED DRIVE WATER PRESSURE TO MOVE THE ROD	NFS	1.00	0.50	145	NL	2/22/91 2/22/91	1 1	0.50 0.50
CRD	FB	N-XXX-91-1304-FB	1991	PREVIOUS REPAIR AND INSTALLATION STATUS DAMAGED THE O-RINGS	UKN	1.00	1.00	185	NL	4/14/91	7	0.01
CRD	FO	N-XXX-91-1303-FO	1991	CRD REQUIRED INCREASED DRIVE WATER PRESSURE TO MOVE THE ROD	NFS	1.00	0.50	145	NL	3/11/91	3	0.50
CRD	FO	N-XXX-93-1314-FO	1993	CONTROL ROD DRIVE WITHDREW TOO FAST	UKN	1.00	0.50	137	NL	1/7/93 1/7/93	1 1	0.10 0.10
ROD	FO	N-XXX-88-1301-FO	1988	SEVERAL CR WERE BEING PINCHED BY FUEL SUPPORT (FS) PLUGS	NFS	1.00	1.00	135	NL	3/13/88	10	1.00

Table B-3. (continued).

Component	Fail Mode	CCF Number	Event Year	Event Description	Safety Function	TDF	Coupling Strength	CCCG	Shock Type	Date	No. Failures (a)	Degraded Value
ROD	FO	N-XXX-92-1406-FO	1992	ROD DID NOT SETTLE AT POSITION 00, WEAR OF THE PISTON SEALS	NFS	1.00	1.00	177	NL	5/14/92	3	0.10
SOV	VO	N-XXX-84-1471-VO	1984	SOVs Failed to open in required time	NFS	1.00	1.00	368	NL	1/24/84	2	1.00
SOV	VO	N-XXX-84-1212-VO	1984	SCRAM PILOT SOLENOID VALVES HAVE IMPROPER SEATING MATERIAL	NFS	1.00	1.00	187	NL	10/12/84	183	0.50
SOV	VO	L-XXX-85-0922-VO	1985	FAILURE OF THE ASSOCIATED SCRAM PILOT SOLENOID VALVES.	NFS	1.00	1.00	292	NL	12/24/85	6	1.00
SOV	VO	N-XXX-85-1190-VO	1985	SCRAM PILOT VALVE WAS FOUND DEFECTIVE, ROD SCRAMED SLOWLY	NFS	1.00	1.00	372	NL	7/24/85 7/24/85	1 1	0.50 0.50
SOV	VO	N-XXX-87-1396-VO	1987	SCRAM OUTLET PILOT VALVES SLOW IN OPENING	NFS	1.00	1.00	372	NL	8/7/87 8/6/87	1 1	0.10 0.10
SOV	VO	L-XXX-89-1029-VO	1989	SEAT MATERIAL IN THE ASSOCIATED SCRAM PILOT SOLENOID VALVES	NFS	1.00	1.00	179	NL	11/25/89	2	1.00
SOV	VO	N-XXX-91-1181-VO	1991	SCRAM PILOT VALVES 117 AND 118 WERE WORN	NFS	1.00	1.00	276	NL	11/26/91	38	0.10
SOV	VO	N-XXX-91-1209-VO	1991	FAULTY SCRAM PILOT SOLENOID VALVE SEAT MATERIAL CONTAMINATIO	NFS	1.00	1.00	179	NL	10/6/91	3	0.10
SOV	VO	N-XXX-91-1201-VO	1991	SCRAM PILOT VALVE WOULD NOT ACTIVATE	NFS	1.00	1.00	372	NL	3/19/91 3/19/91	1 1	0.50 0.50
SOV	VO	L-XXX-91-1004-VO	1991	SLOW VENTING OF AIR FROM THE SCRAM PILOT SOLENOID VALVES	NFS	1.00	1.00	292	NL	8/16/91	16	0.50
SOV	VO	N-XXX-93-1472-VO	1993	Scram Pilot Solenoid Valves were dirty	NFS	1.00	1.00	176	NL	10/15/93 10/15/93	2 2	0.10 0.10
SOV	VO	N-XXX-93-1187-VO	1993	SCRAM SOLENOID PILOT VALVES FAILED TO OPEN	NFS	1.00	1.00	276	NL	1/16/94	6	1.00

Table B-3. (continued).

Component	Fail Mode	CCF Number	Event Year	Event Description	Safety Function	TDF	Coupling Strength	CCCG	Shock Type	Date	No. Failures (a)	Degraded Value
SOV	VO	N-XXX-93-1183-VO	1993	LIQUID THREAD SEALANT PLUGGED SCRAM SOLENOID PILOT VALVES	NFS	1.00	1.00	276	NL	10/3/93	12	0.10
SOV	VO	L-XXX-93-1013-VO	1993	Degraded performance of the Scram Solenoid Pilot Valves	NFS	1.00	1.00	180	NL	4/6/93	7	0.50
SOV	VO	N-XXX-93-1188-VO	1993	FAULTY SCRAM PILOT VALVES	NFS	1.00	1.00	146	NL	12/13/93 12/13/93	1 1	0.50 0.50
SOV	VO	N-XXX-94-1184-VO	1994	LIQUID THREAD SEALANT PLUGGED SCRAM SOLENOID PILOT VALVES	NFS	1.00	1.00	276	NL	1/5/94	10	0.10
SOV	VO	L-XXX-94-1027-VO	1994	Pilot valves disc material from the seating surface inadequa	NFS	1.00	1.00	195	NL	5/28/94	38	0.50
SOV	VO	N-XXX-94-1205-VO	1994	SCRAM PILOT VALVES (SV-117 , SV-118) WERE FOUND TO HAVE DE	NFS	1.00	1.00	292	NL	4/25/94	12	0.50
SOV	VO	N-XXX-94-1204-VO	1994	PREMATURE AGING OF THE SCRAM SOLENOID PILOT VALVE (SSPV)	NFS	1.00	0.50	356	NL	8/29/94	4	0.50
SOV	VO	N-XXX-94-1191-VO	1994	DELAY IN THE INITIAL OPENING OF THE SCRAM SOLENOID PILOT VAL	NFS	1.00	1.00	195	NL	3/27/94	49	0.50
SOV	VO	L-XXX-94-1030-VO	1994	"Slow" control rods exceedied twenty percent of a ten percen	NFS	1.00	1.00	179	NL	12/12/94	33	0.50
CBI	IO	N-XXX-87-1046-IO	1987	TRIP UNIT FAILED A TIME RESPONSE SURVEILLANCE TEST	NFS	1.00	1.00	8	NL	11/15/87 11/15/87	1 1	0.50 0.50
CBI	IO	L-XXX-87-0911-IS	1987	SCRAM SETPOINTS BEING OUT OF SPECIFICATION WAS PERSONNEL ERR	NFS	1.00	1.00	4	NL	3/30/87	2	0.50
CBI	IO	N-XXX-89-1071-IS	1989	TRIP UNIT WAS FOUND OUT OF TOLERANCE	NFS	1.00	0.50	8	NL	4/21/89 3/28/89	1 1	0.10 0.10

Table B-3. (continued).

Component	Fail Mode	CCF Number	Event Year	Event Description	Safety Function	TDF	Coupling Strength	CCCG	Shock Type	Date	No. Failures (a)	Degraded Value
CBI	IS	N-XXX-92-0762-IS	1992	PRESSURE INDICATING SWITCH OOS	NFS	1.00	0.50	4	NL	8/16/92 8/16/92	1 1	0.10 0.10
CPL	IS	N-XXX-84-1064-IS	1984	SCRAM DISCHARGE VOLUME LEVEL TRANSMITTERS OOS	NFS	1.00	0.50	4	NL	11/27/84 11/27/84	1 1	0.10 0.10
CPL	IO	L-XXX-84-0994-IS	1984	3 TRANSMITTERS FOUND TO HAVE BEEN TS DUE TO THE PREVIOUS CAL	NFS	1.00	1.00	4	NL	6/28/84	3	0.10
CPL	IS	N-XXX-85-1065-IS	1985	SCRAM DISCHARGE VOLUME LEVEL TRANSMITTERS OOS	NFS	1.00	0.50	4	NL	1/31/85 1/31/85	1 1	0.10 0.10
CPL	IS	N-XXX-85-0777-IS	1985	LEVEL TRANSMITTTERS FOUND TO HAVE DRIFTED	NFS	1.00	0.50	4	NL	6/28/85 6/27/85	1 1	0.10 0.10
CPL	IO	L-XXX-86-0948-IO	1986	PROCEDURES DID NOT INCLUDE A STEP TO PERTURB WATER LEVEL	UKN	1.00	1.00	4	NL	6/5/86	4	0.10
CPL	IO	N-XXX-86-0783-IO	1986	THE REFERENCE LEG TO THE TRANSMITTER WAS LOW	NFS	1.00	1.00	4	NL	1/2/87 12/28/86	1 1	0.50 0.50
CPL	IS	N-XXX-86-1045-IS	1986	REACTOR VESSEL LEVEL TRANSMITTERS OOS	UKN	1.00	0.50	4	NL	12/10/86 12/10/86	1 1	0.10 0.10
CPL	IO	L-XXX-86-1125-IO	1986	TWO (SDV) LEVEL TRANSMITTERS WERE ISOLATED FROM THE SDV	NFS	1.00	1.00	4	NL	4/10/86	2	1.00
CPL	IS	N-XXX-86-0834-IS	1986	REACTOR VESSEL LEVEL TRANSMITTERS OOS	UKN	1.00	0.50	4	NL	5/6/86 5/6/86	1 1	0.10 0.10
CPL	IO	N-XXX-89-0813-IO	1989	CAPSULE IN THE TRANSMITTER LEAKING DUE TO A MANUFACTURING DE	NFS	1.00	1.00	4	NL	3/31/89 3/31/89	1 1	1.00 1.00
CPL	IO	L-XXX-89-1028-IS	1989	ERROR IN CALIBRATION DATA REACTOR VESSEL LEVEL TRANSMITTERS	NFS	1.00	1.00	4	NL	5/12/89	3	0.10
CPL	IO	L-XXX-89-1376-IO	1989	CALIBRATION DATA DEVELOPED UTILIZING INACCURATE DATA SHEETS	UKN	1.00	1.00	4	NL	6/27/89	4	0.10

Table B-3. (continued).

Component	Fail Mode	CCF Number	Event Year	Event Description	Safety Function	TDF	Coupling Strength	CCCG	Shock Type	Date	No. Failures (a)	Degraded Value
CPL	IS	N-XXX-89-1069-IS	1989	SCRAM DISCHARGE VOLUME LEVEL TRANSMITTERS OOS	NFS	1.00	0.50	4	NL	12/14/89 12/14/89	1 1	0.10 0.10
CPL	IO	N-XXX-90-0753-IO	1990	REACTOR VESSEL LEVEL TRANSMITTER LOSS OF FILL OIL	NFS	0.50	1.00	4	NL	10/20/90 9/24/90	1 1	1.00 1.00
CPL	IO	L-XXX-93-1022-IO	1993	SDV level, column of trapped water in the sensing lines	NFS	1.00	1.00	4	NL	9/1/93	4	0.50
CPL	IS	L-XXX-94-1032-IS	1994	Level transmitter response time data OOS	NFS	1.00	0.50	4	NL	10/3/94	2	0.10
CPR	IO	L-XXX-87-0996-IO	1987	PRESSURE INSTRUMENT ROOT VALVES WERE CLOSED	NFS	1.00	1.00	4	L	6/7/87	4	1.00
CPR	IS	N-XXX-89-0984-IS	1989	PRESSURE TRANSMITTER EXCEEDING THE TECH SPEC LIMITS	UKN	1.00	0.50	4	NL	1/25/89 1/24/89	1 1	0.10 0.10
CPS	IS	N-XXX-84-0868-IS	1984	PRESSURE SWITCHS DRIFTED OUT OF SPECIFICATION HIGH	UKN	1.00	0.50	4	NL	10/26/84	4	0.10
CPS	IS	N-XXX-84-0972-IS	1984	SYSTEM PRESSURE INDICATOR WAS FOUND OUT OF TOLERANCE	UKN	1.00	0.50	4	NL	4/14/84 4/14/84	1 1	0.10 0.10
CPS	IS	N-XXX-84-1033-IS	1984	LEVEL INDICATING SWITCHS FOUND OUT OF	UKN	1.00	0.50	8	NL	5/11/84	7	0.10
CPS	IS	N-XXX-84-0974-IS	1984	SCRAM CONTAINMENT ISOLATION SWITCH WAS FOUND OUT OF SPECIFIC	NFS	1.00	0.50	4	NL	10/4/84 10/4/84	1 1	0.10 0.10
CPS	IS	L-XXX-84-0961-IS	1984	3 OUT OF 4 REACTOR HIGH PRESSURE SWITCHES HAD DRIFTED OOS	NFS	1.00	1.00	4	NL	6/19/84	3	0.10
CPS	IS	N-XXX-84-0976-IS	1984	HIGH DRYWELL PRESSURE & CONTAINMENT ISOLATION SWITCH OOS	NFS	1.00	0.50	4	NL	11/20/84 11/20/84	1 1	0.10 0.10
CPS	IS	L-XXX-85-0928-IS	1985	LEVEL INDICATING SWITCHES OOS	NFS	1.00	0.50	4	NL	2/15/85	2	0.10

Table B-3. (continued).

Component	Fail Mode	CCF Number	Event Year	Event Description	Safety Function	TDF	Coupling Strength	CCCG	Shock Type	Date	No. Failures (a)	Degraded Value
CPS	IS	N-XXX-85-0791-IS	1985	PRESSURE SWITCH WAS OUT OF CALIBRATION	UKN	1.00	0.50	4	NL	10/16/85	3	0.10
CPS	IS	N-XXX-85-1034-IS	1985	SCRAM TRIP SWITCH WAS FOUND LOW OUT OF SPECIFICATION	NFS	1.00	0.50	4	NL	8/5/85 8/5/85	1 1	0.10 0.10
CPS	IS	N-XXX-85-0968-IS	1985	PRESSURE SWITCH WAS FOUND OUT OF TOLERANCE	UKN	1.00	0.50	4	NL	8/27/85 8/27/85	1 1	0.10 0.10
CPS	IS	N-XXX-85-0978-IS	1985	REACTOR VESSEL LOW WATER LEVEL SCRAM SWITCH OOS	NFS	1.00	0.50	4	NL	1/11/85 1/11/85	1 1	0.10 0.10
CPS	IS	L-XXX-85-1126-IS	1985	SETPOINTS FOR TWO REACTOR WATER LEVEL SWITCHES WERE OOS	NFS	1.00	0.50	4	NL	2/12/85	2	0.10
CPS	IS	N-XXX-85-0779-IS	1985	PRESSURE SWITCHES IN NEED OF CALIBRATION	NFS	1.00	0.50	4	NL	10/29/85 10/29/85	1 1	0.10 0.10
CPS	IO	N-XXX-86-0982-IO	1986	MAIN STEAM ISOLATION VALVE LIMIT SWITCH FAILED	NFS	1.00	0.50	8	NL	10/18/86	3	1.00
CPS	IO	L-XXX-86-1124-IO	1986	FAILURE OF THE REACTOR VESSEL LEVEL 3 SWITCHES	NFS	1.00	1.00	4	NL	6/1/86	3	1.00
CPS	IO	N-XXX-86-1035-IO	1986	STATIC-O-RING SWITCHES EXHIBITING EXCESSIVE STATIC SHIFT	NFS	1.00	1.00	4	NL	2/16/86	4	0.10
CPS	IO	N-XXX-86-1037-IO	1986	LIMIT SWITCH DID NOT ENERGIZE WITH THE VALVE NOT FULL OPEN	UKN	1.00	0.50	8	NL	7/25/86 7/25/86	1 1	1.00 1.00
CPS	IS	L-XXX-86-0923-IS	1986	TWO OUT OF FOUR PRESSURE SWITCHES FAILED TO MEET	UKN	1.00	0.50	4	NL	12/30/86	2	0.10
CPS	IS	N-XXX-86-0869-IS	1986	REACTOR LOW WATER LEVEL SCRAM SENSOR	NFS	1.00	0.50	4	NL	1/17/86	5	0.10
CPS	IS	L-XXX-86-0909-IS	1986	THREE OUT OF FOUR REACTOR LOW LEVEL SCRAM SENSORS OOS	NFS	1.00	0.50	4	NL	1/17/86	3	0.10

Table B-3. (continued).

Component	Fail Mode	CCF Number	Event Year	Event Description	Safety Function	TDF	Coupling Strength	CCCG	Shock Type	Date	No. Failures (a)	Degraded Value
CPS	IO	N-XXX-86-0886-IO	1986	PRESSURE SWITCH WAS ERRATIC IN ITS OPERATION (DRIFT)	UKN	1.00	0.50	4	NL	4/2/86 3/26/86	1 1	0.10 0.10
CPS	IO	N-XXX-86-0839-IO	1986	LIMIT SWITCHES FAILED TO TRIP THE (RPS) AS	NFS	1.00	1.00	8	NL	8/16/86 8/16/86	1 1	1.00 1.00
CPS	IS	N-XXX-86-0838-IS	1986	SWITCH ' AS FOUND ' DATA EXCEEDED THE TECH . SPEC .	NFS	1.00	0.50	4	NL	1/28/86 1/28/86	1 1	0.10 0.10
CPS	IS	N-XXX-86-0849-IS	1986	LEVEL SWITCH WAS FOUND TO BE OUT OF ACCEPTANCE CRITERIA	UKN	1.00	0.50	4	NL	12/17/86	1	0.10
CPS	IS	L-XXX-86-0910-IS	1986	REPEATABILITY/DRIFT PROBLEMS WITH LOW WATER LEVEL SCRAM	UKN	1.00	0.50	4	NL	3/27/86	4	0.10
CPS	IS	N-XXX-87-0888-IS	1987	FIRST STAGE PRESSURE SWITCHES OOS	NFS	1.00	0.50	4	NL	8/11/87	4	0.10
CPS	IO	N-XXX-87-0890-IO	1987	ACTUATING PLATE WAS LOOSE AND TILTED	NFS	1.00	1.00	8	NL	11/14/87 10/16/87	1 1	1.00 1.00
CPS	IS	L-XXX-87-0925-IS	1987	ALL FOUR PRESSURE SWITCHES OOS	UKN	1.00	1.00	4	NL	9/15/87	4	0.10
CPS	IO	N-XXX-88-0856-IO	1988	PRESSURE SWITCH HAD A GAS BUBBLE IN THE ' KAPTON ' DIAPHRAGM	NFS	1.00	1.00	4	NL	1/25/88 1/25/88	1 1	0.50 0.50
CPS	IS	N-XXX-88-0983-IS	1988	PRESSURE SWITCH EXCEEDED TECHNICAL	NFS	1.00	0.50	4	NL	7/3/88 7/3/88	1 1	0.10 0.10
CPS	IS	N-XXX-88-0873-IS	1988	REACTOR HIGH PRESSURE SCRAM SWITCHS OOS	UKN	0.50	0.50	4	NL	9/17/88 8/22/88	1 1	0.10 0.10
CPS	IS	N-XXX-88-0851-IS	1988	LEVEL SWITCH FAILED SURVEILLANCE ACCEPTANCE	UKN	1.00	0.50	4	NL	12/20/88 12/20/88	3 1	0.10 0.10
CPS	IS	N-XXX-88-0850-IS	1988	REACTOR VESSEL LEVEL TRIP SWITCHES OOS	UKN	1.00	0.50	4	NL	1/6/88	4	0.10
CPS	IO	N-XXX-88-0872-IO	1988	REACTOR HIGH PRESSURE SWITCHS FOUND TO HAVE NO REPEATABILITY	UKN	1.00	0.50	4	NL	3/3/88 2/19/88	1 1	0.50 0.50

Table B-3. (continued).

Component	Fail Mode	CCF Number	Event Year	Event Description	Safety Function	TDF	Coupling Strength	CCCG	Shock Type	Date	No. Failures (a)	Degraded Value
CPS	IS	N-XXX-88-0811-IS	1988	PRESSURE SWITCHES TRIP OUTSIDE OF THE TECH SPEC LIMITS	NFS	1.00	0.50	4	NL	7/22/88 7/22/88	1 1	0.10 0.10
CPS	IS	L-XXX-88-1001-IS	1988	SCRAM INITIATION LIMIT SWITCHES WERE CALIBRATED IN EXCESS OF	NFS	1.00	1.00	4	NL	7/7/88	4	0.10
CPS	IO	N-XXX-88-0871-IO	1988	REACTOR HIGH PRESSURE SWITCHS FOUND TO HAVE NO REPEATABILITY	UKN	0.10	0.50	4	NL	2/18/88 1/17/88	1 1	0.50 0.50
CPS	IS	L-XXX-89-0998-IS	1989	FOUR OUT OF SIX CONDENSER LOW VACCUUM SCRAM SETPOINTS OOS	NFS	1.00	0.50	6	NL	12/9/89	4	0.10
CPS	IS	N-XXX-89-0819-IS	1989	LEVEL INDICATING SWITCHES WERE OUT OF TECH SPEC LIMITS	NFS	1.00	0.50	4	NL	1/12/89 1/12/89	1 1	0.10 0.10
CPS	IS	N-XXX-89-1371-IS	1989	REACTOR VESSEL LEVEL SWITCHS OOS	NFS	1.00	0.50	4	NL	7/25/89 7/11/89	1 1	0.50 0.50
CPS	IS	N-XXX-89-0874-IS	1989	CONDENSER LOW VACUUM SWITCHS OOS	UKN	1.00	0.50	4	NL	2/25/89 2/25/89	1 1	0.10 0.10
CPS	IS	N-XXX-89-1039-IS	1989	LOW PRESSURE SCRAM BYPASS SWITCH OOS	UKN	1.00	0.50	4	NL	8/26/89 8/26/89	1 1	0.10 0.10
CPS	IS	N-XXX-90-0892-IS	1990	REACTOR HIGH PRESSURE SCRAM SETPOINT FOR PRESSURE SWITCHES O	NFS	1.00	0.50	4	NL	4/8/90 4/6/90	1 1	0.10 0.10
CPS	IO	N-XXX-90-1041-IO	1990	LIMIT SWITCHES WERE OUT OF ADJUSTMENT	NFS	1.00	0.50	8	NL	7/11/90	3	1.00
CPS	IO	N-XXX-90-1070-IO	1990	POSITION SWITCH FAILED TO ACTIVATE RELAYS	NFS	1.00	1.00	8	NL	8/29/90 8/14/90	1 1	0.50 1.00
CPS	IS	L-XXX-90-1003-IS	1990	All four of the condenser low vacuum scram switches OOS	NFS	1.00	1.00	4	NL	12/12/90	4	0.10
CPS	IS	N-XXX-91-0845-IS	1991	LEVEL 3 TRIP SWITCH WAS OUT OF ACCEPTANCE CRITERIA ON THE HI	UKN	1.00	0.50	4	NL	11/5/91 11/5/91	1 1	0.10 0.10

Table B-3. (continued).

Component	Fail Mode	CCF Number	Event Year	Event Description	Safety Function	TDF	Coupling Strength	CCCG	Shock Type	Date	No. Failures (a)	Degraded Value
CPS	IS	N-XXX-91-0843-IS	1991	LEVEL SWITCH WAS OUT OF CALIBRATION	UKN	1.00	0.50	4	NL	1/29/91 1/29/91	1 1	0.10 0.10
CPS	IO	N-XXX-91-0830-IO	1991	POSITION SWITCHES WOULD NOT ACTUATE	NFS	1.00	0.50	8	NL	11/26/91 11/26/91	1 1	0.50 1.00
CPS	IS	N-XXX-91-0794-IS	1991	PRESSURE SWITCH WAS FOUND OUT OF TOLERANCE	NFS	1.00	0.50	4	NL	5/24/91	3	0.10
CPS	IS	N-XXX-92-0858-IS	1992	POSITION SWITCHES ON THE TURBINE VALVES WERE NOT SET PROPERL	UKN	1.00	0.50	8	NL	7/20/92 7/20/92	1 1	0.10 0.10
CPS	IS	N-XXX-92-0895-IS	1992	PRESSURE SWITCHS OUT OF TECHNICAL SPECIFICATION REQUIREMENTS	NFS	1.00	0.50	4	NL	2/29/92	4	0.10
CPS	IS	N-XXX-92-0896-IS	1992	PRESSURE SWITCHS LOW OUT OF TOLERANCE	NFS	0.10	0.50	8	NL	4/22/92 3/9/92	1 1	0.10 0.10
CPS	IS	L-XXX-93-1006-IS	1993	Three out of four Main Turbine Pressure Switches OOS	NFS	1.00	0.50	4	NL	1/17/93	3	0.10
CPS	IS	N-XXX-93-0797-IS	1993	ARMING SWITCHS FOUND OUT OF TOLERANCE	NFS	1.00	0.50	4	NL	7/6/93 7/6/93	1 1	0.10 0.10
CPS	IS	L-XXX-95-1012-IS	1995	All four (4) low condenser vacuum scram switches OOS	NFS	1.00	1.00	4	NL	7/26/95	4	0.10
CPS	IS	L-XXX-95-1000-IS	1995	Anticipatory scram bypass pressure switches OOS	NFS	1.00	1.00	8	NL	8/25/95	8	0.10
TLR	RO	L-XXX-84-0908-RX	1984	(58) RELAYS (OUT OF 68) FOGGED UP WITH AN OILY VAPOR	UKN	1.00	1.00	68	NL	6/7/84	58	0.10
TLR	RO	L-XXX-84-1122-RO	1984	TIME DELAYS FOR RELAYS K101A THROUGH K101D WERE OOS	NFS	1.00	0.50	4	NL	8/28/84	4	0.10
TLR	RC	N-XXX-85-0823-RC	1985	REACTOR PROTECTION SYSTEM RELAY SOCKETS DEFECTIVE	UKN	1.00	1.00	92	NL	2/5/85 2/5/85	1 1	0.10 0.10

Table B-3. (continued).

Component	Fail Mode	CCF Number	Event Year	Event Description	Safety Function	TDF	Coupling Strength	CCCG	Shock Type	Date	No. Failures (a)	Degraded Value
TLR	RO	L-XXX-86-0993-RO	1986	RELAY WAS IMPROPERLY SEATED IN THE SOCKET	NFS	1.00	1.00	92	NL	12/23/86	20	0.10
TLR	RO	N-XXX-89-1040-RO	1989	LOW PRESSURE TIME DELAY RELAY OOS	NFS	0.10	0.50	4	NL	11/10/89 9/11/89	1 1	0.10 0.10
TLR	RO	N-XXX-89-0853-RO	1989	FAILURE TO MEET REQUIRED RESPONSE TIME, LACK OF LUBRICATION	NFS	1.00	1.00	8	NL	11/1/89	3	0.10
TLR	RC	N-XXX-90-1370-RC	1990	RELAY WAS DAMAGED DURING REPAIR	UKN	1.00	1.00	4	NL	10/12/90 10/12/90	1 1	1.00 1.00
TLR	RO	N-XXX-91-1043-RO	1991	TIME DELAY RELAY WAS FOUND OUT OF CALIBRATION	NFS	1.00	0.50	4	NL	11/7/91 11/6/91	1 1	0.10 0.10
TLR	RO	N-XXX-92-0987-RO	1992	RELAY K80A CONTACTS WERE OPENING SLOWER THAN TECH SPECS ALLOW	NFS	1.00	0.50	8	NL	6/14/92	4	0.10
TLR	RO	N-XXX-93-0985-RO	1993	TRIP RELAY EXCEEDED THE TECHNICAL SPECIFICATION RESPONSE TIM	NFS	0.10	1.00	84	NL	2/20/93	3	0.10
TLR	RO	N-XXX-94-0986-RO	1994	RELAY DID NOT DROP OUT AS EXPECTED	NFS	0.50	1.00	84	NL	3/12/94 2/14/94	1 1	1.00 1.00

a. This value represents the summarized number of failures in the CCF event.

Appendix C

Quantitative Results of Basic Component Operational Data Analysis

Appendix C

Quantitative Results of Basic Component Operational Data Analysis

This appendix displays relevant RPS component counts and the estimated probability or rate for each failure mode, including distributions that characterize any variation observed between portions of the data. The analysis is based exclusively on data from General Electric plants during the period 1984 through 1995.

The quantitative analysis of the RPS failure data was at each stage influenced by the uncertainty in the number of complete failures for which the safety function of the associated component was lost. Table C-1 provides a breakdown of the component data, showing the number of events fully classified as known, complete failures, and the number of uncertain events within various subsets of the data. The table lists the failure modes in sequence across the RPS, beginning with the channel sensor/transmitters, then the channel process switches and bistables, trip logic relays, solenoid and air-operated valves, accumulators, and rod drives and rods.

Within each component grouping, subsets in Table C-1 are based on the assessed method of discovery and the plant status (operations or shutdown) for each event (note that uncertainty in these two attributes of the data was not quantified in the data assessment). In addition, rows in Table C-1 show breakdowns for whether the failures occurred during the first half of the study period (1984-1989) or during the second half (1990-1995).

The choice of the most representative subset of data to use for each component for the fault tree was a major part of the statistical data analysis. Where operations and shutdown data differ significantly, the subset of operations data was selected since the risk assessment describes risk during operations. Similarly, when the newer data differed significantly from the data earlier in the study period, the newer data were used for the analysis. The analysis also considered whether the test data and data from unplanned scrams differ, for the limited number of components that are always demanded in a scram and whose failures would be detected. Rules for subset selection are discussed further in Section 2.1.1.

Table C-1 shows that the observed number of failures for each component potentially lies between two bounds: a lower bound that excludes all the uncertain failures, and an upper bound that includes them. The initial analysis of the RPS failure data, to select the subsets, was based on these two extreme cases. The next four tables provide information on how the subsets were selected using these two sets of data. Figure C-1 is an overview of the selection process and how the results feed into these tables.

As shown in Figure C-1, the analysis first considered the lower bound (LOB) case of no uncertain failures. These data correspond to the first failure count column in Table C-1. Table C-2 provides these counts for several subsets, along with the associated denominators and simple calculated probabilities or rates. It also gives confidence bounds for the estimates. Note that the confidence bounds do not consider any special sources of variation (e.g., year or plant). The maximum likelihood estimates and bounds are provided for simple comparisons. They are not used directly in the risk assessment.

Table C-3 summarizes the results from testing the hypothesis of constant probabilities or, as applicable, constant rates, across groupings for each basic component failure mode in the RPS fault trees having data. The table provides probability values (p-values) for the hypothesis tests, rounded to the nearest 0.001. When the hypothesis is rejected, the data show evidence of variation. The tests are for possible differences based on method of discovery or data source (unplanned reactor trips or testing), on

Table C-1. Summary of General Electric RPS total failure counts and weighted average total failures (independent and common-cause failures).

Basic Event (component)	Data Set[a]	Lower Bound: Known Failures Only (NFS/CF)	Uncertain Failure Counts: Uncertain Loss of Safety Function (UKN/CF)	Uncertain Failure Counts: Uncertain Completeness (NFS/UC)	Uncertain Failure Counts: Both Uncertainties (UKN/UC)	Upper Bound: All Failures Counted	Total Failure Weighted Average[b]
Pressure sensor/ transmitter (CPR)	Cyc. & qtr. tests	0	1	2	0	3	1.5
	—(1984-1989 s/d)	0	1	1	0	2	1.0
	—(1990-1995 s/d)	0	0	1	0	1	0.5
	Occurrences in time	1	2	0	3	6	1.6
	—(op)	0	1	0	1	2	0.2
	—(s/d)	1	1	0	2	4	1.6
	(1984-1989)	1	1	0	1	3	1.6
	—(1984-1989 op)	0	0	0	1	1	0.3
	—(1984-1989 s/d)	1	1	0	0	2	1.4
	(1990-1995)	0	1	0	2	3	0.3
	—(1990-1995 op)	0	1	0	0	1	0.2
	—(1990-1995 s/d)	0	0	0	2	2	0.3
Level sensor/ transmitter (CPL)	Cyc. & qtr. tests	6	3	8	7	24	15.7
	—(op)	0	3	6	1	10	5.0
	—(s/d)	6	0	2	6	14	9.5
	(1984-1989)	3	3	1	3	10	7.3
	—(1984-1989 op)	0	3	1	1	5	2.4
	—(1984-1989 s/d)	3	0	0	2	5	3.5
	(1990-1995)	3	0	7	4	14	8.4
	—(1990-1995 op)	0	0	5	0	5	2.5
	—(1990-1995 s/d)	3	0	2	4	9	5.7
	Trips (op) (1990-1995) (not used)[c]	1	0	0	0	1	1.0

Table C-1. (continued)

Basic Event (component)	Data Set[a]	Lower Bound: Known Failures Only (NFS/CF)	Uncertain Failure Counts			Upper Bound: All Failures Counted	Total Failure Weighted Average[b]
			Uncertain Loss of Safety Function (UKN/CF)	Uncertain Completeness (NFS/UC)	Both Uncertainties (UKN/UC)		
Level sensor/ transmitter (CPL) (continued)	Occurrences in time	0	1	6	1	8	3.3
	—(op)	0	1	4	1	6	2.3
	—(s/d)	0	0	2	0	2	1.0
	(1984-1989)	0	1	5	1	7	2.8
	—(1984-1989 op)	0	1	4	1	6	2.3
	—(1984-1989 s/d)	0	0	1	0	1	0.5
	(1990-1995) (s/d)	0	0	1	0	1	0.5
Process switch (CPS)	Qtr. Tests	42	8	19	19	88	63.5
	—(op)	26	4	10	15	55	38.5
	—(s/d)	16	4	9	4	33	24.5
	(1984-1989)	26	8	12	13	59	43.0
	—(1984-1989 op)	18	4	7	13	42	30.1
	—(1984-1989 s/d)	8	4	5	0	17	12.6
	(1990-1995)	16	0	7	6	29	21.4
	—(1990-1995 op)	8	0	3	2	13	9.9
	—(1990-1995 s/d)	8	0	4	4	16	11.8
	Occur. in time (not used)	6	2	2	0	10	7.4
Scr. disch. vol. level sw. (SDL)	Qtr. Tests	7	1	2	2	12	9.0
	—(op)	3	0	0	1	4	3.3
	—(s/d)	4	1	2	1	8	5.9
	(1984-1989)	4	1	2	1	8	5.8
	—(1984-1989 op)	1	0	0	1	2	1.3
	—(1984-1989 s/d)	3	1	2	0	6	4.6
	(1990-1995)	3	0	0	1	4	3.1
	—(1990-1995 op)	2	0	0	0	2	2.0
	—(1990-1995 s/d)	1	0	0	1	2	1.1

Table C-1. (continued).

Basic Event (component)	Data Set[a]	Lower Bound: Known Failures Only (NFS/CF)	Uncertain Failure Counts: Uncertain Loss of Safety Function (UKN/CF)	Uncertain Failure Counts: Uncertain Completeness (NFS/UC)	Uncertain Failure Counts: Both Uncertainties (UKN/UC)	Upper Bound: All Failures Counted	Total Failure Weighted Average[b]
Bistable (CBI)	Qtr. Tests	4	3	7	3	17	11.2
	—(op)	1	1	4	1	7	4.0
	—(s/d)	3	2	3	2	10	7.1
	(1984-1989)	1	0	7	3	11	5.9
	—(1984-1989 op)	0	0	4	1	5	2.5
	—(1984-1989 s/d)	1	0	3	2	6	3.4
	(1990-1995)	3	3	0	0	6	5.1
	—(1990-1995 op)	1	1	0	0	2	1.5
	—(1990-1995 s/d)	2	2	0	0	4	3.7
	—Trips (op) (not used)	0	0	0	0	0	0.0
	Occur. in time (not used)	5	3	0	0	8	6.0
Relay (TLR)	Weekly tests	18	9	3	2	32	23.4
	—(op)	9	2	2	0	13	10.8
	—(s/d)	9	7	1	2	19	12.5
	(1984-1989)	8	6	1	2	17	11.9
	—(1984-1989 op)	5	1	0	0	6	5.6
	—(1984-1989 s/d)	3	5	1	2	11	6.0
	(1990-1995)	10	3	2	0	15	11.9
	—(1990-1995 op)	4	1	2	0	7	5.3
	—(1990-1995 s/d)	6	2	0	0	8	6.6
	Occur. in time (not used)	4	4	0	6	14	5.1
Manual switch (MSW)	Unpl. reactor trips & weekly tests	0	0	0	0	0	0.0

Table C-1. (continued)

Basic Event (component)	Data Set[a]	Lower Bound: Known Failures Only (NFS/CF)	Uncertain Failure Counts: Uncertain Loss of Safety Function (UKN/CF)	Uncertain Completeness (NFS/UC)	Both Uncertainties (UKN/UC)	Upper Bound: All Failures Counted	Total Failure Weighted Average[b]
Solenoid-operated valve (SOV)	3x10% & cyc. tests [d]	38	2	138	0	178	107.8
	—(op)	16	1	67	0	84	49.8
	—(s/d)	22	1	71	0	94	58.2
	(1984-1989)	17	2	22	0	41	28.6
	—(1984-1989 op)	8	1	21	0	30	18.7
	—(1984-1989 s/d)	9	1	1	0	11	10.2
	(1990-1995)	21	0	116	0	137	79.0
	—(1990-1995 op)	8	0	46	0	54	31.0
	—(1990-1995 s/d)	13	0	70	0	83	48.0
	Occur. in time (not used)	5	2	0	0	7	5.3
Air-operated valve (AOV)	3x10% & cyc. tests (1984-1989, s/d)	1	0	0	0	1	1.0
	Occur. in time (not used)	0	1	0	2	3	0.2
Scram accumulator (ACC)	3x10% & cyc. tests	4	1	1	0	6	5.4
	—(op)	0	0	1	0	1	0.5
	—(s/d)	4	1	0	0	5	4.9
	(1984-1989)	3	0	1	0	4	3.5
	—(1984-1989 op)	0	0	1	0	1	0.5
	—(1984-1989 s/d)	3	0	0	0	3	3.0
	(1990-1995) (s/d)	1	1	0	0	2	1.8
	Occurrences in time	10	0	15	0	25	17.5
	—(op)	2	0	4	0	6	4.0
	—(s/d)	8	0	11	0	19	13.5
	(1984-1989)	2	0	2	0	4	3.0
	—(1984-1989 op)	1	0	1	0	2	1.5
	—(1984-1989 s/d)	1	0	1	0	2	1.5
	(1990-1995)	8	0	13	0	21	14.5
	—(1990-1995 op)	1	0	3	0	4	2.5
	—(1990-1995 s/d)	7	0	10	0	17	12.0

Table C-1. (continued).

Basic Event (component)	Data Set[a]	Lower Bound: Known Failures Only (NFS/CF)	Uncertain Failure Counts: Uncertain Loss of Safety Function (UKN/CF)	Uncertain Failure Counts: Uncertain Completeness (NFS/UC)	Uncertain Failure Counts: Both Uncertainties (UKN/UC)	Upper Bound: All Failures Counted	Total Failure Weighted Average[b]
Scram discharge volume (SDV)	Unpl. Reactor trips (1984-1989)	1	0	0	0	1	1.0
Rod and control rod drive (RDC	Unpl. Reactor trips (1984-1989)	3	0	1	0	4	3.5
	3x10% & cyc. tests [d]	14	6	77	29	126	71.7
	—(op)	3	2	47	10	62	32.4
	—(s/d)	11	4	30	19	64	39.2
	(1984-1989)	13	4	74	19	110	62.8
	—(1984-1989 op)	3	2	44	7	56	29.7
	—(1984-1989 s/d)	10	2	30	12	54	32.8
	(1990-1995)	1	2	3	10	16	7.9
	—(1990-1995 op)	0	0	3	3	6	2.8
	—(1990-1995 s/d)	1	2	0	7	10	4.3
	Occur. in time (not used)	2	1	14	19	36	18.6

a. Testing frequency abbreviations: weekly, weekly; qtr., quarterly; cyclic, cyclic. The frequency of testing applies to the demand count estimations. The failure data are classified as being discovered on testing, unplanned demands, or observation (occurrences in time). Plant status abbreviations: op, operating; s/d, shut down.

b. The tabulated values are the means or weighted averages of the data. The uncertain events are analyzed using a simulation that in each iteration either counts or does not count them. In this column, 0.5 is the probability of events with uncertain completeness being counted. The ratio of the number of events with known safety function lost to events with safety function either known to be lost or known to be fail-safe, among complete events, was used for the probability of counting a complete event with uncertain safety function loss. For events with both uncertainties, 0.5 times the ratio of the number of events with known safety function lost to events with safety function either known to be lost or known to be fail-safe, among events with uncertain completeness, was used for the probability of counting an event.

c. Not used in the RPS fault tree unavailability analysis.

d. The uncertain failure counts from testing for SOV and RDC components are lower than from the values cited in Appendix B. The counts were reduced in order to exclude failures from demands that are not part of the testing scheme modeled in the analysis. When certain common-cause failures (CCF) were found, the entire set of components was tested rather than the modeled ten percent. These extra demands, and associated failures, are part of the CCF assessment but are not in the set of independent demands considered for estimation of the overall failure probability.

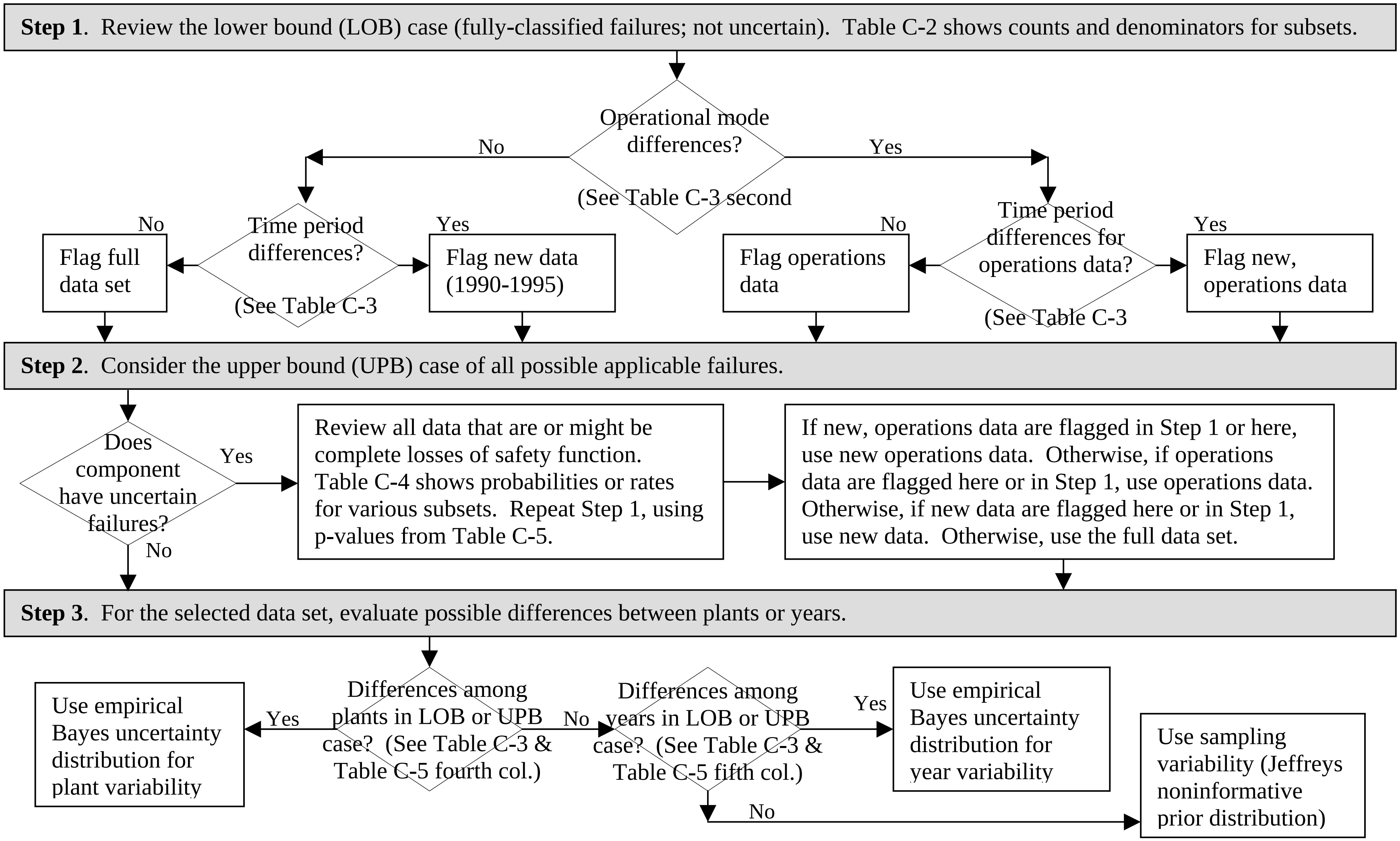


Figure C-1. Decision algorithm for uncertainty distribution selection (applied for each component).

Table C-2. Point estimates and confidence bounds for RPS total failure probabilities and rates (NFS/CF only).

Failure Mode (component)	Data Set	Failures f	Denominator d or T	Probability or Rate[a] and 90% Confidence Interval
Channel parameter monitoring instruments				
Pressure sensor/ transmitter (CPR)	Quarterly & cyclic tests	0	8753	(0.0E+00, 0.0E+00, 3.4E-04)
	Occurrences in time	1	1857.6 [c]	(2.8E-05, 5.4E-04, 2.6E-03)
Level sensor/ transmitter (CPL)	Quarterly & cyclic tests	6	9153	(2.9E-04, 6.6E-04, 1.3E-03)
	Quarterly & cyclic tests (op)	**0**	**6750**	**(0.0E+00, 0.0E+00, 4.4E-04)**
	Quarterly & cyclic tests (s/d)	6	2403	(1.1E-03, 2.5E-03, 4.9E-03)
	Occurrences in time	0	1939.7 [c]	(0.0E+00, 0.0E+00, 1.5E-03)
Process switch (CPS)	Quarterly tests	42	38237	(8.4E-04, 1.1E-03, 1.4E-03)
	Quarterly tests, 1984-1989	26	17108	(1.1E-03, 1.5E-03, 2.1E-03)
	Quarterly tests, 1990-1995	16	21129	(4.8E-04, 7.6E-04, 1.1E-03)
Scr. disch. vol. level sw. (SDL)	Quarterly tests	7	8323	(3.9E-04, 8.4E-04, 1.6E-03)
Bistable (CBI)	Quarterly tests	4	20612	(6.6E-05, 1.9E-04, 4.4E-04)
	Quarterly tests (op)	**1**	**15026**	**(3.4E-06, 6.7E-05, 3.2E-04)**
	Quarterly tests (s/d)	3	5586	(1.5E-04, 5.4E-04, 1.4E-03)
Trains (trip systems)				
Relay (TLR)	Weekly tests	18	792801	(1.5E-05, 2.3E-05, 3.4E-05)
	Weekly tests (op)	**9**	**579677**	**(8.1E-06, 1.6E-05, 2.7E-05)**
	Weekly tests (s/d)	9	213124	(2.2E-05, 4.2E-05, 7.4E-05)
Manual switch (MSW)	Unplanned trips	0	1034	(0.0E+00, 0.0E+00, 2.9E-03)
	Weekly tests	0	37435	(0.0E+00, 0.0E+00, 8.0E-05)
	Pooled trips & tests	**0**	**38469**	**(0.0E+00, 0.0E+00, 7.8E-05)**
Control rod drive and rod components				
Solenoid-operated valve (SOV)	Triannual (10%) & cyclic	38	104218	(2.7E-04, 3.6E-04, 4.8E-04)
	Triannual (10%) & cyclic (op)	**16**	**77845**	**(1.3E-04, 2.1E-04, 3.1E-04)**
	Triannual (10%) & cyclic (s/d)	22	26373	(5.6E-04, 8.3E-04, 1.2E-03)
Air-operated valve (AOV)	Unplanned trips	0	405616	(0.0E+00, 0.0E+00, 7.4E-06)
	Triannual (10%) & cyclic	1	116690	(4.4E-07, 8.6E-06, 4.1E-05)
	Pooled trips & tests	**1**	**522306**	**(9.8E-08, 1.9E-06, 9.1E-06)**
Scram accumulator (ACC)	Triannual (10%) & cyclic	4	58346	(2.3E-05, 6.9E-05, 1.6E-04)
	Triannual (10%) & cyclic (op)	**0**	**43883**	**(0.0E+00, 0.0E+00, 6.8E-05)**
	Triannual (10%) & cyclic (s/d)	4	14463	(9.4E-05, 2.8E-04, 6.3E-04)
	Occurrences in time	10	56980.0	(9.5E-05, 1.8E-04, 3.0E-04)
	Occurrences in time (op)	2	41617.5 [c]	(8.5E-06, 4.8E-05, 1.5E-04)
	Occurrences in time (s/d)	8	15362.5 [c]	(2.6E-04, 5.2E-04, 9.4E-04)
	Occur. in time, 1984-1989 (s/d)	1	7633.5 [c]	(6.7E-06, 1.3E-04, 6.2E-04)
	Occur. in time, 1990-1995 (s/d)	7	7729.0 [c]	(4.3E-04, 9.1E-04, 1.7E-03)

Table C-2. (continued).

Failure Mode (component)	Data Set	Failures f	Denominator d or T	Probability or Rate[a] and 90% Confidence Interval
Scram discharge volume (SDV)	**Unplanned trips**	**1**	**2251**	**(2.3E-05, 4.4E-04, 2.1E-03)**
Rod and control rod drive (RDC)	Unplanned trips	3	186939	(4.4E-06, 1.6E-05, 4.1E-05)
	Triannual (10%) & cyclic	14	47863	(1.8E-04, 2.9E-04, 4.6E-04)
	Triannual (10%) & cyclic (op)	3	35535	(2.3E-05, 8.4E-05, 2.2E-04)
	—, 1984-1989 (op)	3	16759	(4.9E-05, 1.8E-04, 4.6E-04)
	—, 1990-1995 (op)	0	18776	(0.0E+00, 0.0E+00, 1.6E-04)
	Triannual (10%) & cyclic (s/d)	11	12328	(5.0E-04, 8.9E-04, 1.5E-03)
	—, 1984-1989 (s/d)	10	6675	(8.1E-04, 1.5E-03, 2.5E-03)
	—, 1990-1995 (s/d)	1	5653	(9.1E-06, 1.8E-04, 8.4E-04)
	Pooled trips & tests	17	234802	(4.6E-05, 7.2E-05, 1.1E-04)
	Pooled trips & tests (op)	6	222474	(1.2E-05, 2.7E-05, 5.3E-05)
	Pooled trips & tests, 1984-1989	16	166784	(6.0E-05, 9.6E-05, 1.5E-04)
	Pooled trips & tests, 1990-1995	1	68018	(7.5E-07, 1.5E-05, 7.0E-05)

a. The middle number is the point estimate, *f/d*, or f/T, and the two end numbers form a 90% confidence interval. For demands, the interval is based on a binomial distribution for the occurrence of failures, while it is based on a Poisson distribution for the rates. Rates are identified from the "occurrences in time" data set, and a footnote in the denominator column. Note that these maximum likelihood estimates may be zero, and are not used directly in the risk assessment.

b. Highlighted rows show the data sets selected for the unavailability analysis. In sections where no row is highlighted, see Table C-4.

c. Component years. The associated rates are failures per component year.

Table C-3. Evaluation of differences between groups for RPS failure modes (NFS/CF only).[a]

Failure Mode (component)	Data Set [b]	P-Values for Test of Variation [c]: Rx. Trip vs. Tests	In Plant Modes	In Time Periods	In Plant Units	In Years
Channel parameter monitoring instruments and bistables						
Pressure sensor/ transmitter (CPR)	Quarterly & cyclic tests	—	0 F	0 F	0 F	0 F
	Occurrences in time	—	0.102	0.220	0.442	0.190
Level sensor/ transmitter (CPL)	Quarterly & cyclic tests	—	**0.000** (E)	0.696	**0.003** (E)	**0.007** (E)
	Quarterly & cyclic tests (op)	—	—	0 F	0 F	0 F
	Quarterly & cyclic tests (s/d)	—	—	1.000	**0.001** (E)	0.057 (E)
	Occurrences in time	—	0 F	0 F	0 F	0 F
Process switch (CPS)	Quarterly tests	—	0.115	**0.029** (E)	**0.042** (E)	**0.029** (E)
	Quarterly tests, 1984-1989	—		—	0.416 (E)	0.678
	Quarterly tests, 1990-1995	—		—	0.126 (E)	**0.004** (E)
Scr. disch. vol. level sw. (SDL)	Quarterly tests	—	0.091	0.707	0.458	0.567
Bistable (CBI)	Quarterly tests	—	0.063 (E)	0.645	**0.049** (E)	0.802
	Quarterly tests (op)	—	—	1.000	0.552	0.645
	Quarterly tests (s/d)	—	—	0.623	0.429	0.698
Trains (trip systems)						
Relay (TLR)	Weekly tests	—	**0.034** (E)	1.000	**0.003** (E)	0.477
	Weekly tests (op)	—	—	0.503	**0.011** (E)	0.293 (E)
	Weekly tests (s/d)	—	—	0.508	**0.020** (E)	0.479
Manual switch (MSW)	Unplanned trips	—	0 F	0 F	0 F	0 F
	Weekly tests	—	0 F	0 F	0 F	0 F
	Pooled trips & tests	0 F	0 F	0 F	0 F	0 F
Control rod drive and rod components						
Solenoid-operated valve (SOV)	Triannual (10%) & cyclic	—	**0.000** (E)	0.622	**0.001** (E)	**0.007** (E)
	Triannual (10%) & cyclic (op)	—	—	0.441	**0.001** (E)	**0.001** (E)
	Triannual (10%) & cyclic (s/d)	—	—	0.673	**0.001** (E)	0.052 (E)
Air-operated valve (AOV)	Unplanned trips	—	0 F	0 F	0 F	0 F
	Triannual (10%) & cyclic	—	0.248	0.402	0.586	0.360
	Pooled trips & tests	0.062	0.055	1.000	**0.001**	0.301
Scram accumulator (ACC)	Triannual (10%) & cyclic	—	**0.004** (E)	0.309	**0.041** (E)	0.317 (E)
	Triannual (10%) & cyclic (op)	—	—	0 F	0 F	0 F
	Triannual (10%) & cyclic (s/d)	—	—	0.342	0.268 (E)	0.348 (E)
	Occurrences in time	—	**0.000** (E)	0.138	**0.000** (E)	**0.003** (E)
	Occurrences in time (op)	—	—	0.792	0.075	0.371
	Occurrences in time (s/d)	—	—	**0.035** (E)	**0.000** (E)	**0.003** (E)
	Occur. in time, 1984-1989 (s/d)	—	—	—	**0.007**	0.516
	Occur. in time, 1990-1995 (s/d)	—	—	—	**0.000** (E)	**0.022** (E)
Scram discharge volume (SDV)	Unplanned trips	—	—	1.000	0.036	0.080

Table C-3. (continued).

Failure Mode (component)	Data Set [b]	P-Values for Test of Variation [c]: Rx. Trip vs. Tests	In Plant Modes	In Time Periods	In Plant Units	In Years
Rod and control rod drive (RDC)	Unplanned trips	—	—	1.000	**0.012** (E)	**0.035** (E)
	Triannual (10%) & cyclic	—	**0.000** (E)	**0.001** (E)	**0.001** (E)	**0.001** (E)
	Triannual (10%) & cyclic (op)	—	—	0.105 (E)	**0.001** (E)	**0.020** (E)
	—, 1984-1989 (op)	—	—	—	**0.013** (E)	0.136 (E)
	—, 1990-1995 (op)	—	—	—	0 F	0 F
	Triannual (10%) & cyclic (s/d)	—	—	**0.015** (E)	**0.001** (E)	**0.001** (E)
	—, 1984-1989 (s/d)	—	—	—	**0.001**	**0.001** (E)
	—, 1990-1995 (s/d)	—	—	—	0.464	0.564
	Pooled trips & tests	**0.001**	**0.000** (E)	**0.033** (E)	**0.001** (E)	**0.001** (E)
	Pooled trips & tests (op)	**0.023**	—	0.195	**0.001** (E)	0.142 (E)
	Pooled trips & tests, 1984-1989	**0.001**		—	**0.001** (E)	**0.001** (E)
	Pooled trips & tests, 1990-1995	0.182		—	0.304	0.526

a. This table describes components in the fault tree whose failure probability or rate was estimated from the RPS data. Unplanned demands are considered for some components as indicated in Table A-2. Additional rows for subsets based on plant status or time period appear if significant differences in these attributes were found in the larger groups of data.

b. "—", a subset of the test data for the component based on plant state (operating or shut down) and/or year.

c. "—", not applicable; 0 F, no failures (thus, no test); All F, no successes (thus, no test); **0.000**, less than 5E-4; NE, not evaluated. P-values less than or equal to 0.05 are in a bold font. For the evaluation columns other than "Rx. trip vs. tests," an "E" is in parentheses after the p-value if and only if an empirical Bayes distribution was found accounting for variations in groupings. Low p-values and the fitting of empirical Bayes distributions are indications of variability between the groupings considered in the column.

plant mode (operations or shutdown), on the time period (1984-1989 versus 1990-1995), on different plant units, and on different calendar years. Like Table C-2, Table C-3 applies to the LOB data. The results in every case are subdivided according to the method of discovery, if applicable. In the table, finding empirical Bayes distributions for differences in plant mode resulted in the generation of lines describing the operational and shutdown data separately. Similarly, a finding of an empirical Bayes distribution in the time period data groupings produced additional separate evaluations of the older and more recent data.

In Table C-3, low p-values point to variation and lack of homogeneity in the associated data groupings. For example, in Table C-3 the 0.000 p-value for level sensor/transmitter differences in quarterly and cyclic tests by plant mode shows that, when the operational failures and demands are pooled and compared with the corresponding total failures and demands during shutdowns, the likelihood of the observed difference or a more extreme difference if the groups did have the same failure probability is less than 0.0005. Either a “rare” (probability less than 0.0005) situation occurred, or the two pooled sets of failures and demands have different failure probabilities. Throughout these tables, p-values that are less than or equal to 0.05 are highlighted. The tables show many cases where differences in plant unit reporting were observed.

In each of the first three evaluation columns in Table C-3, two entities or data groupings are being compared (reactor trips versus tests, operational versus shutdown, and older versus more recent). In the first column, where applicable, the testing versus reactor trip data were compared. This evaluation is for information only; both sets of data were pooled for the risk assessment. In Table C-3, the rod and control rod drive component shows a higher probability from testing failures than from trips (the same number of failures but fewer demands among the operations testing data). The trip data are directly relevant to the study of operational reliability, but confidence in the detection of all failures detected during trips is not as high as for the periodic testing failures. The test data are also believed to be complete. Pooling the two data sets is conservative.

The second and third evaluations in Table C-3 also reflect the comparison of pairs of attributes. "Step 1" in Figure C-1 shows how the plant operating mode and time period evaluations are used in the selection of a subset of data for analysis. The selections were also dictated by the allowed component combinations listed in Table A-2.

Step 2 in the data selection process is to repeat Step 1 using the upper bound (UPB) data from the fifth data column in Table C-1. Table C-4 is similar to Table C-2, and gives denominators, probabilities or rates, and confidence intervals. Table C-5 shows the p-values computed for the tests of differences in groups for the UPB data.

The subset selection results for the LOB and UPB cases agreed for several of the components. In the overall analysis described below, subsets were used if either of the bounding analyses showed a need for them. This point is explained in the last Step 2 box in Figure C-1. In both Tables C-2 and C-4, lines are highlighted corresponding to the subsets selected. Table C-6 provides a concise summary of the data in the selected subsets.

Within each selected subset, the next evaluation focused on the two remaining attributes for study of data variation, namely differences between plants and between calendar years. Tables C-3 and C-5 include results from these evaluations in the last two columns. These evaluations are used in Step 3 in Figure 1. In nearly every instance where a significant p-value appears in these columns, empirical Bayes distributions reflect the associated variability. The two exception to this finding are for plant differences among the older shutdown data for rod/control rod drive (RDC) failure probabilities and for accumulator (ACC) failure rates in Table C-3. In the RDC case, all ten failures were at one plant; and the plant had an

Table C-4. Point estimates and confidence bounds for RPS total failure probabilities and rates (NFS/CF, NFS/UC, UKN/CF, and UKN/UC).

Failure Mode (component)	Data Set	Failures f	Denominator d or T	Probability or Rate[a] and 90% Confidence Interval
Channel parameter monitoring instruments				
Pressure sensor/ transmitter (CPR)	Quarterly & cyclic tests	3	8753	(9.3E-05, 3.4E-04, 8.9E-04)
	Quarterly & cyclic tests (op) [b]	**0**	**6424**	**(0.0E+00, 0.0E+00, 4.7E-04)**
	Quarterly & cyclic tests (s/d)	3	2329	(3.5E-04, 1.3E-03, 3.3E-03)
	Occurrences in time	6	1857.6 [c]	(1.4E-03, 3.2E-03, 6.4E-03)
	Occurrences in time (op)	2	1351.9 [c]	(2.6E-04, 1.5E-03, 4.6E-03)
	Occurrences in time (s/d)	4	505.7 [c]	(2.7E-03, 7.9E-03, 1.8E-02)
Level sensor/ transmitter (CPL)	Quarterly & cyclic tests	24	9153 [c]	(1.8E-03, 2.6E-03, 3.7E-03)
	Quarterly & cyclic tests (op)	**10**	**6750**	**(8.0E-04, 1.5E-03, 2.5E-03)**
	Quarterly & cyclic tests (s/d)	14	2403	(3.5E-03, 5.8E-03, 9.1E-03)
	Occurrences in time	8	1939.7 [c]	(2.1E-03, 4.1E-03, 7.4E-03)
	Occurrences in time, 1984-1989	7	817.4 [c]	(4.0E-03, 8.6E-03, 1.6E-02)
	Occurrences in time, 1990-1995	1	1122.3 [c]	(4.6E-05, 8.9E-04, 4.2E-03)
Process switch (CPS)	Quarterly tests	88	38237	(1.9E-03, 2.3E-03, 2.7E-03)
	Quarterly tests (op)	55	28022	(1.5E-03, 2.0E-03, 2.5E-03)
	Qtr. tests, 1984-1989 (op)	42	12024	(2.7E-03, 3.5E-03, 4.5E-03)
	Qtr. tests, 1990-1995 (op)	**13**	**15998**	**(4.8E-04, 8.1E-04, 1.3E-03)**
	Quarterly tests (s/d)	33	10215	(2.4E-03, 3.2E-03, 4.3E-03)
Scr. disch. vol. level sw. (SDL)	Quarterly tests	12	8323	(8.3E-04, 1.4E-03, 2.3E-03)
	Quarterly tests (op)	**4**	**6075**	**(2.2E-04, 6.6E-04, 1.5E-03)**
	Quarterly tests (s/d)	8	2248	(1.8E-03, 3.6E-03, 6.4E-03)
Bistable (CBI)	Quarterly tests	17	20612	(5.3E-04, 8.2E-04, 1.2E-03)
	Quarterly tests (op)	**7**	**15026**	**(2.2E-04, 4.7E-04, 8.7E-04)**
	Quarterly tests (s/d)	10	5586	(9.7E-04, 1.8E-03, 3.0E-03)
	Quarterly tests, 1984-1989	11	8607	(7.2E-04, 1.3E-03, 2.1E-03)
	Quarterly tests, 1990-1995	6	12005	(2.2E-04, 5.0E-04, 9.9E-04)
Trains (trip systems)[d]				
Relay (TLR)	Weekly tests	32	792801	(2.9E-05, 4.0E-05, 5.4E-05)
	Weekly tests (op)	**13**	**579677**	**(1.3E-05, 2.2E-05, 3.6E-05)**
	Weekly tests (s/d)	19	213124	(5.8E-05, 8.9E-05, 1.3E-04)
Manual switch (MSW)	See Note d	—	—	—
Control rod drive and rod components[e]				
Solenoid-operated valve (SOV)	Triannual (10%) & cyclic	178	104218	(1.5E-03, 1.7E-03, 1.9E-03)
	Triannual (10%) & cyclic (op)	**84**	**77845**	**(8.9E-04, 1.1E-03, 1.3E-03)**
	Triannual (10%) & cyclic (s/d)	94	26373	(3.0E-03, 3.6E-03, 4.2E-03)
	—, 1984-1989 (s/d)	11	12242	(5.0E-04, 9.0E-04, 1.5E-03)
	—, 1990-1995 (s/d)	83	14131	(4.9E-03, 5.9E-03, 7.0E-03)

Table C-4. (continued).

Failure Mode (component)	Data Set	Failures f	Denominator d or T	Probability or Rate[a] and 90% Confidence Interval
Air-operated valve (AOV)	See Note d	—	—	—
Scram accumulator (ACC)	Triannual (10%) & cyclic	6	58346	(4.5E-05, 1.0E-04, 2.0E-04)
	Triannual (10%) & cyclic (op)	**1**	**43883**	**(1.2E-06, 2.3E-05, 1.1E-04)**
	Triannual (10%) & cyclic (s/d)	5	14463	(1.4E-04, 3.5E-04, 7.3E-04)
	Occurrences in time	25	56980.0	(3.1E-04, 4.4E-04, 6.1E-04)
	Occurrences in time (op)	6	41617.5 [c]	(6.3E-05, 1.4E-04, 2.8E-04)
	Occurrences in time (s/d)	19	15362.5 [c]	(8.1E-04, 1.2E-03, 1.8E-03)
	Occur. in time, 1984-1989 (s/d)	2	7633.5 [c]	(4.7E-05, 2.6E-04, 8.2E-04)
	Occur. in time, 1990-1995 (s/d)	17	7729.0 [c]	(1.4E-03, 2.2E-03, 3.3E-03)
Scram discharge volume (SDV)	See Note d	—	—	—
Rod and control rod drive (RDC)	Unplanned trips	4	186939	(7.3E-06, 2.1E-05, 4.9E-05)
	Triannual (10%) & cyclic	126	47863	(2.3E-03, 2.6E-03, 3.1E-03)
	Triannual (10%) & cyclic (op)	62	35535	(1.4E-03, 1.7E-03, 2.2E-03)
	—, 1984-1989 (op)	56	16759	(2.6E-03, 3.3E-03, 4.2E-03)
	—, 1990-1995 (op)	6	18776	(1.4E-04, 3.2E-04, 6.3E-04)
	Triannual (10%) & cyclic (s/d)	64	12328	(4.2E-03, 5.2E-03, 6.4E-03)
	—, 1984-1989 (s/d)	54	6675	(6.4E-03, 8.1E-03, 1.0E-02)
	—, 1990-1995 (s/d)	10	5653	(9.6E-04, 1.8E-03, 3.0E-03)
	Pooled trips & tests	130	234802	(4.8E-04, 5.5E-04, 6.4E-04)
	Pooled trips & tests (op)	66	222474	(2.4E-04, 3.0E-04, 3.6E-04)
	Pooled trips & tests, 1984-1989 (op)	60	160109	(3.0E-04, 3.7E-04, 4.6E-04)
	Pooled trips & tests, 1990-1995 (op)	**6**	**62365**	**(4.2E-05, 9.6E-05, 1.9E-04)**

a. The middle number is the point estimate, *f/d*, or f/T, and the two end numbers form a 90% confidence interval. For demands, the interval is based on a binomial distribution for the occurrence of failures, while it is based on a Poisson distribution for the rates. Rates are identified from the “occurrences in time” data set, and a footnote in the denominator column. Note that these maximum likelihood estimates may be zero, and are not used directly in the risk assessment.

b. Highlighted rows show the data sets selected for the unavailability analysis. No rows are highlighted among the occurrences in time because the unavailability associated with each rate and an 8-hour per year down time is two orders of magnitude lower than the unavailability computed from the test data.

c. Component years. The associated rates are failures per component year.

d. See Table C-2. There were no uncertain failures for these components.

Table C-5. Evaluation of differences between groups for RPS failure modes (NFS/CF, NFS/UC, UKN/CF, and UKN/UC).[a]

Failure Mode (component)	Data Set [b]	P-Values for Test of Variation [c]: Rx. Trip vs. Tests	In Plant Modes	In Time Periods	In Plant Units	In Years
Channel parameter monitoring instruments						
Pressure sensor/ transmitter (CPR)	Quarterly & cyclic tests	—	**0.019** (E)	0.564	0.521	0.132 (E)
	Quarterly & cyclic tests (op)	—	—	0 F	0 F	0 F
	Quarterly & cyclic tests (s/d)	—	—	0.623	0.363	0.295 (E)
	Occurrences in time	—	**0.030** (E)	0.613	**0.036** (E)	0.844
	Occurrences in time (op)	—	—	0.677	0.164	0.628
	Occurrences in time (s/d)	—	—	0.981	0.377	0.623
Level sensor/ transmitter (CPL)	Quarterly & cyclic tests	—	**0.002** (E)	1.000	**0.036** (E)	0.965
	Quarterly & cyclic tests (op)	—	—	0.522	**0.001** (E)	0.132 (E)
	Quarterly & cyclic tests (s/d)	—	—	0.423	0.061 (E)	0.295 (E)
	Occurrences in time	—	0.911	**0.009** (E)	**0.000** (E)	0.220
	Occurrences in time, 1984-1989	—		—	**0.000** (E)	0.688
	Occurrences in time, 1990-1995	—		—	0.995	0.403
Process switch (CPS)	Quarterly tests	—	**0.029** (E)	**0.000** (E)	**0.001** (E)	**0.008** (E)
	Quarterly tests (op)	—	—	**0.000** (E)	**0.001** (E)	**0.001** (E)
	Qtr. tests, 1984-1989 (op)	—	—	—	**0.001** (E)	0.510
	Qtr. tests, 1990-1995 (op)	—	—	—	0.181 (E)	0.107 (E)
	Quarterly tests (s/d)	—	—	0.863	**0.035** (E)	0.587
Scr. disch. vol. level sw. (SDL)	Quarterly tests	—	**0.005** (E)	0.151	**0.028** (E)	0.519
	Quarterly tests (op)	—	—	1.000	**0.003** (E)	0.603
	Quarterly tests (s/d)	—	—	0.171	0.109	0.442
Bistable (CBI)	Quarterly tests	—	**0.006** (E)	0.082 (E)	0.705	**0.001** (E)
	Quarterly tests (op)	—	—	0.116	0.416	**0.001** (E)
	Quarterly tests (s/d)	—	—	0.754	0.524	0.056 (E)
	Quarterly tests, 1984-1989	—		—	0.350 (E)	**0.001** (E)
	Quarterly tests, 1990-1995	—		—	0.561 (E)	0.286
Trains (trip systems)						
Relay (TLR)	Weekly tests	—	**0.000** (E)	0.291	**0.004** (E)	0.523
	Weekly tests (op)	—	—	0.782	**0.011** (E)	0.435
	Weekly tests (s/d)	—	—	0.503	**0.001** (E)	0.865
Manual switch (MSW)	See Note d	—	—	—	—	—
Control rod drive and rod components[e]						
Solenoid-operated valve (SOV)	Triannual (10%) & cyclic	—	**0.000** (E)	**0.000** (E)	**0.001** (E)	**0.001** (E)
	Triannual (10%) & cyclic (op)	—	—	0.655	**0.001** (E)	**0.001** (E)
	Triannual (10%) & cyclic (s/d)	—	—	**0.000** (E)	**0.001** (E)	**0.001** (E)
	—, 1984-1989 (s/d)	—	—	—	**0.001** (E)	0.270
	—, 1990-1995 (s/d)	—	—	—	**0.001** (E)	**0.001** (E)

Table C-5. (continued).

Failure Mode (component)	Data Set [b]	P-Values for Test of Variation [c] Rx. Trip vs. Tests	In Plant Modes	In Time Periods	In Plant Units	In Years
Air-operated valve (AOV)	See Note d	—	—	—	—	—
Scram accumulator (ACC)	Triannual (10%) & cyclic	—	**0.004** (E)	0.227	0.226 (E)	0.561
	Triannual (10%) & cyclic (op)	—	—	0.381	0.742	0.103
	Triannual (10%) & cyclic (s/d)	—	—	0.668	0.313 (E)	0.487
	Occurrences in time	—	**0.000** (E)	**0.006** (E)	**0.000** (E)	**0.001** (E)
	Occurrences in time (op)	—	—	0.709	**0.000** (E)	**0.005** (E)
	Occurrences in time (s/d)	—	—	**0.001** (E)	**0.000** (E)	**0.000** (E)
	Occur. in time, 1984-1989 (s/d)	—	—	—	0.134	0.132 (E)
	Occur. in time, 1990-1995 (s/d)	—	—	—	**0.000** (E)	**0.034** (E)
Scram discharge volume (SDV)	See Note d	—	—	—	—	—
Rod and control rod drive (RDC)	Unplanned trips	—	—	0.579	0.080 (E)	0.083 (E)
	Triannual (10%) & cyclic	—	**0.000** (E)	**0.000** (E)	**0.001** (E)	**0.001** (E)
	Triannual (10%) & cyclic (op)	—	—	**0.000** (E)	**0.001** (E)	**0.001** (E)
	—, 1984-1989 (op)	—	—	—	**0.001** (E)	**0.001** (E)
	—, 1990-1995 (op)	—	—	—	**0.024** (E)	0.455
	Triannual (10%) & cyclic (s/d)	—	—	**0.000** (E)	**0.001** (E)	**0.001** (E)
	—, 1984-1989 (s/d)	—	—	—	**0.001** (E)	**0.001** (E)
	—, 1990-1995 (s/d)	—	—	—	**0.006** (E)	0.616
	Pooled trips & tests	**0.001**	**0.000** (E)	**0.000** (E)	**0.001 (E)**	**0.001 (E)**
	Pooled trips & tests (op)	**0.001**	—	**0.000** (E)	**0.001 (E)**	**0.001 (E)**
	Pooled trips & tests, 1984-1989 (op)	**0.001**	—	—	**0.001 (E)**	**0.001 (E)**
	Pooled trips & tests, 1990-1995 (op)	**0.001**	—	—	0.207 (E)	0.468

a. This table describes components in the fault tree whose failure probability or rate was estimated from the RPS data including uncertain failures. Unplanned demands are considered for some components as indicated in Table A-2. Additional rows for subsets based on plant status or time period appear if significant differences in these attributes were found in the larger groups of data.

b. "—", a subset of the test data for the component based on plant state (operating or shut down) and/or year.

c. "—", not applicable; 0 F, no failures (thus, no test); All F, no successes (thus, no test); **0.000**, less than 5E-4, NE, not evaluated. P-values less than or equal to 0.05 are in a bold font. For the evaluation columns other than "Rx. trip vs. tests," an "E" is in parentheses after the p-value if and only if an empirical Bayes distribution was found accounting for variations in groupings. Low p-values and the fitting of empirical Bayes distributions are indications of variability between the groupings considered in the column.

d. See Table C-3. There were no failures with unknown completeness and/or unknown loss of safety function for this component.

Table C-6. Point estimates of failure probabilities and rates for RPS risk assessment.

Basic Event (Component)	Data Set (General Electric Data Only)	No Uncertain Failures	Failure Count with Uncertain Failures Included	Probability Applied to Uncertainty in Whether the Safety Function is Lost[b]: Among Complete Failures	Probability Applied to Uncertainty in Whether the Safety Function is Lost[b]: Among Uncertain Completeness Failures	Weighted Average Total Failures	Denominator (Demands or Hours)	Failures per Demand or Hour	Update of Jeffreys Noninformative Prior [a]
Channel parameter monitoring instruments									
Pressure sensor/ transmitter (CPR)	Cyc. & qtr. tests (op)	0	0	—	—	0.0	6424	0.0E+00	7.8E-05
	Occurrences in time (op)	0	2	0.125	0.167	0.2	11842574	1.8E-08	6.0E-08
Level sensor/ transmitter (CPL)	Cyc. & qtr. tests (op)	0	10	0.500	0.929	5.0	6750	7.4E-04	8.1E-04
	Occurrences in time, 1990-1995	0	1	—	—	0.5	9831068	5.1E-08	1.0E-07
Process switch (CPS)	Qtr. tests, 1990-1995 (op)	8	13	—	0.438	9.9	15998	6.2E-04	6.5E-04
Scr. disch. vol. level sw. (SDL)	Qtr. tests (op)	3	4	—	0.500	3.3	6075	5.3E-04	6.2E-04
Bistable (CBI)	Qtr. tests (op)	1	7	0.500	0.900	4.0	15026	2.6E-04	3.0E-04
Trains (trip systems)									
Relay (TLR)	Certain weekly & qtr. tests (op)	9	13	0.413	—	10.8	579677	1.9E-05	2.0E-05
Manual switch (MSW)	Unpl. scrams & weekly tests	0	0	—	—	0.0	38469	0.0E+00	1.3E-05

Table C-6. (continued).

Basic Event (Component)	Data Set (General Electric Data Only)	No Uncertain Failures	Failure Count with Uncertain Failures Included	Probability Applied to Uncertainty in Whether the Safety Function is Lost[b]		Weighted Average Total Failures	Denominator (Demands or Hours)	Failures per Demand or Hour	Update of Jeffreys Noninformative Prior [a]
				Among Complete Failures	Among Uncertain Completeness Failures				
Control rod drive and rod components									
Solenoid-operated valve (SOV)	Cyc. & 3x10% tests (op)	16	84	0.254	—	49.8	77845	6.4E-04	6.5E-04
Air-operated valve (AOV)	Unpl. scr.& 3x10%/cyc. tests	1	1	—	—	1.0	522306	1.9E-06	2.9E-06
Scram accumulator (ACC)	3x10% & cyc. tests (op)	0	1	—	—	0.5	43883	1.1E-05	2.3E-05
	Occurrences in time (op)	2	6	—	—	4.0	364568871	1.1E-08	1.2E-08
Scram discharge volume (SDV)	Unplanned scrams	1	1	—	—	1.0	2251	4.4E-04	6.7E-04
Rod and control rod drive (RDC)	Unpl. scr. & 3x10%/cyc. tests, 1990-1995 (op)	0	6	—	0.875	2.8	62365	4.5E-05	5.3E-05

a. (Failures + 0.5)/(Denominator + 1) for probabilities; (Failures + 0.5)/Denominator for rates.

b. "—" when there were no applicable uncertain events. The probability applied for uncertainty in completeness is 0.5.

average number of demands. In the ACC case, just one failure occurred in the group but the associated plant had fewer operating years during the study period than most plants. Neither of these data sets were used in the risk assessment.

The upper and lower bound empirical Bayes analyses included tests of goodness of fit for the resulting beta-binomial model for probabilities or the associated gamma-Poisson model for rates. Each grouping (plant or year) was evaluated to see if it was a high outlier compared with the fitted GE model for each component. For the subsets of data used in the unreliability analysis, no outliers were found.

For the three components (pressure and level sensors/transmitters, and scram accumulators) that were modeled both for failures detected on demands and for minor unavailabilities that are annunciated or detected and easily fixed during inspections at the start of each shift, the unavailability contribution from the rate data in Table C-6 was evaluated using an 8-hour downtime. Since the resulting unavailability was two orders of magnitude lower than the unavailability estimated from the failures on demand, these data were dropped from the risk assessment.

Within each selected subset for which differences exist in the remaining LOB and UPB data, a simulation was conducted to observe the variation in the composite data which includes the fully classified failures and a fraction of the uncertain failures. This evaluation focused on the two remaining attributes for study of data variation, namely differences between plants and between calendar years.

In the simulation, the probability of being complete failures for events whose completeness was unknown was determined by a fixed distribution with a mean of 0.5. The probability that events with unknown safety function status were losses of the safety function was estimated based on the failure data within each subset, including the events (not shown in Table C-1) that were assessed as fail-safe. For the data sets used in the analysis, these probabilities are cited in Table C-6. The last column of Table C-1 shows the weighted average of the events that would be complete losses of the safety function. This average can differ slightly for rows that have the same failure counts in Table C-1. Such a difference would be caused by the fact that subsets that included more events have the possibility of including more non-fail-safe events, and thus have the possibility of having a different assessed probability of counting the events with unknown loss of safety function.

Table C-7 gives the final results of the basic quantitative component data analysis, most of which come from the simulation. It describes the Bayes distributions initially selected to describe the statistical variability in the data used to model the basic RPS events. Table C-7 differs from Tables C-3 and C-4 because it gives Bayes distributions and intervals, not confidence intervals. This choice allows the results for the failure modes to be combined to give an uncertainty distribution on the unavailability. When distributions were fit for both plant variation and year variation, the distribution for differences between plants had greater variability and was selected. Where empirical Bayes distributions were not found, the simple Bayes method was used to obtain uncertainty distributions.

For the unreliability analysis, the means and variances of the generic Bayes distributions were fitted to lognormal distributions, listed in Table C-8. As applicable, these distributions describe the total failure probabilities (Q_T) associated with the common-cause fault tree events.

One additional evaluation was performed: the process switch (CPS) and scram discharge volume level (SDL) switch data were combined. Both of these components are process switches. They were originally distinguished due to the importance of the SDL and the possibility of different environments that might affect the unavailabilities. The results provided similar probabilities for the two components. When the data are pooled, the mean is 6.1E-4 and the upper and lower lognormal uncertainty bounds are 1.6E-4 and 1.5E-3, respectively.

Table C-7. Results of uncertainty analysis.

Failure Mode (Component)	Failures[a]	Denominator[b]	Modeled Variation[c]	Distribution[d]	Bayes Mean and interval[e]
Channel parameter monitoring instruments					
Pressure sensor/ transmitter (CPR)	0	8753	Sampling (only)	Beta(0.5,8753.5)	(2.25E-07,5.71E-05,2.19E-04)
Level sensor/ transmitter (CPL)	4.9	6750	Between plant	Beta(0.1,173.0)	(1.00E-09,7.72E-04,4.34E-03)
Process switch (CPS)	10.0	15998	Between plant	Beta(0.9,1484.2)	(2.35E-05,6.06E-04,1.88E-03)
Scr. Disch. vol. level sw. (SDL)	3.3	6075	Between plant	Beta(0.4,716.6)	(1.17E-06,6.13E-04,2.46E-03)
Bistable (CBI)	4.0	15026	Between year	Beta(0.4,1406.8)	(3.38E-07,2.89E-04,1.19E-03)
Trains (trip systems)					
Relay (TLR)	10.8	579677	Between plant	Beta(0.4,21972)	(2.92E-08,1.93E-05,7.86E-05)
Manual switch (MSW)	0	38469	Sampling (only)	Beta(0.5,38470)	(5.11E-08,1.30E-05,4.99E-05)
Control rod drive and rod components					
Solenoid-operated valve (SOV)	50.1	77845	Between plant	Beta(0.1,214.6)	(1.00E-09,6.97E-04,3.84E-03)
Air-operated valve (AOV)	1	522306	Sampling (only)	Beta(1.5,522306)	(3.37E-07,2.87E-06,7.48E-06)
Scram accumulator (ACC)	0.5	43883	Sampling	Beta(0.8,34963)	(5.61E-07,2.23E-05,7.30E-05)
Scram discharge volume (SDV)	1	2251	Sampling (only)	Beta(1.5,2250.5)	(7.82E-05,6.66E-04,1.73E-03)
Rod and control rod drive (RDC)	2.7	62365	Between plant	Beta(0.4,8931.0)	(9.89E-08,4.96E-05,1.99E-04)

a. Average number of failures, averaged over the 1000 simulation iterations, each of which had an integral number of failures.

b. Estimated number of demands, based on the selected data sets or subsets shown in Table C-6. The three rate estimates in Table C-6 are not listed here because, with an 8-hr mean time to detect and repair, the unavailabilities were less than 1E-7 and were not significant compared with the unavailability estimated from the failures detected during testing.

c. In addition to variation from unknown completeness and/or from unknown loss of safety function.

d. Beta distributions for probabilities and gamma distributions for rates. The simple and empirical Bayes distributions are initially either beta or gamma distributions. Lognormal bounds from distributions with the same mean and variance as these distributions are in Table C-8.

e. Aggregate of Bayes distributions from simulation, unless otherwise noted. Obtained by matching the mean and variance of the simulation output distribution. If the variation is not just sampling, empirical Bayes (EB) distributions were found in each simulated iteration, with the following exceptions: for level sensor/transmitters rates, EB distributions were found 77.1% of the time; for process switches, 97.8%; for bistables, 58.3%; for scram accumulator rates, 70.2% of the time. Sampling variation (from the simple Bayes method) entered the simulation mixture when EB distributions were not found.

f. Simple Bayes distribution not based on the simulations. No uncertain events were in the subsets selected for the analysis.

g. Component years rather than demands. Also, the rates in the Bayes mean column are per year.

Table C-8. Lognormal uncertainty distributions.

Failure Mode (Component)	Median	Error Factor[a]	Lognormal Distribution Mean and Interval[b]
Channel parameter monitoring instruments			
Pressure sensor/transmitter (CPR)	3.3E-05	5.6	(5.9E-06, 5.7E-05, 1.8E-04)
Level sensor/transmitter (CPL)	2.7E-04	11.0	(2.4E-05, 7.7E-04, 2.9E-03)
Process switch (CPS)	4.2E-04	4.1	(1.0E-04, 6.1E-04, 1.7E-03)
Scr. disch. vol. level sw. (SDL)	3.4E-04	6.0	(5.7E-05, 6.1E-04, 2.0E-03)
Bistable (CBI)	1.6E-04	6.2	(2.5E-05, 2.9E-04, 9.7E-04)
Trains (trip systems)			
Relay (TLR)	1.1E-05	6.1	(1.7E-06, 1.9E-05, 6.4E-05)
Manual switch (MSW)	7.5E-06	5.6	(1.3E-06, 1.3E-05, 4.2E-05)
Control rod drive and rod components			
Solenoid-operated valve (SOV)	2.5E-04	10.4	(2.4E-05, 7.0E-04, 2.6E-03)
Air-operated valve (AOV)	2.2E-06	3.2	(6.9E-07, 2.9E-06, 7.2E-06)
Scram accumulator (ACC)	1.5E-05	4.5	(3.3E-06, 2.2E-05, 6.6E-05)
Scram discharge volume (SDV)	5.2E-04	3.2	(1.6E-04, 6.7E-04, 1.7E-03)
Rod and control rod drive (RDC)	2.8E-05	6.0	(4.6E-06, 5.0E-05, 1.6E-04)

a. Lognormal error factor corresponding to 5% and 95% bounds.

b. Mean and lognormal distribution 5th and 95th percentiles.

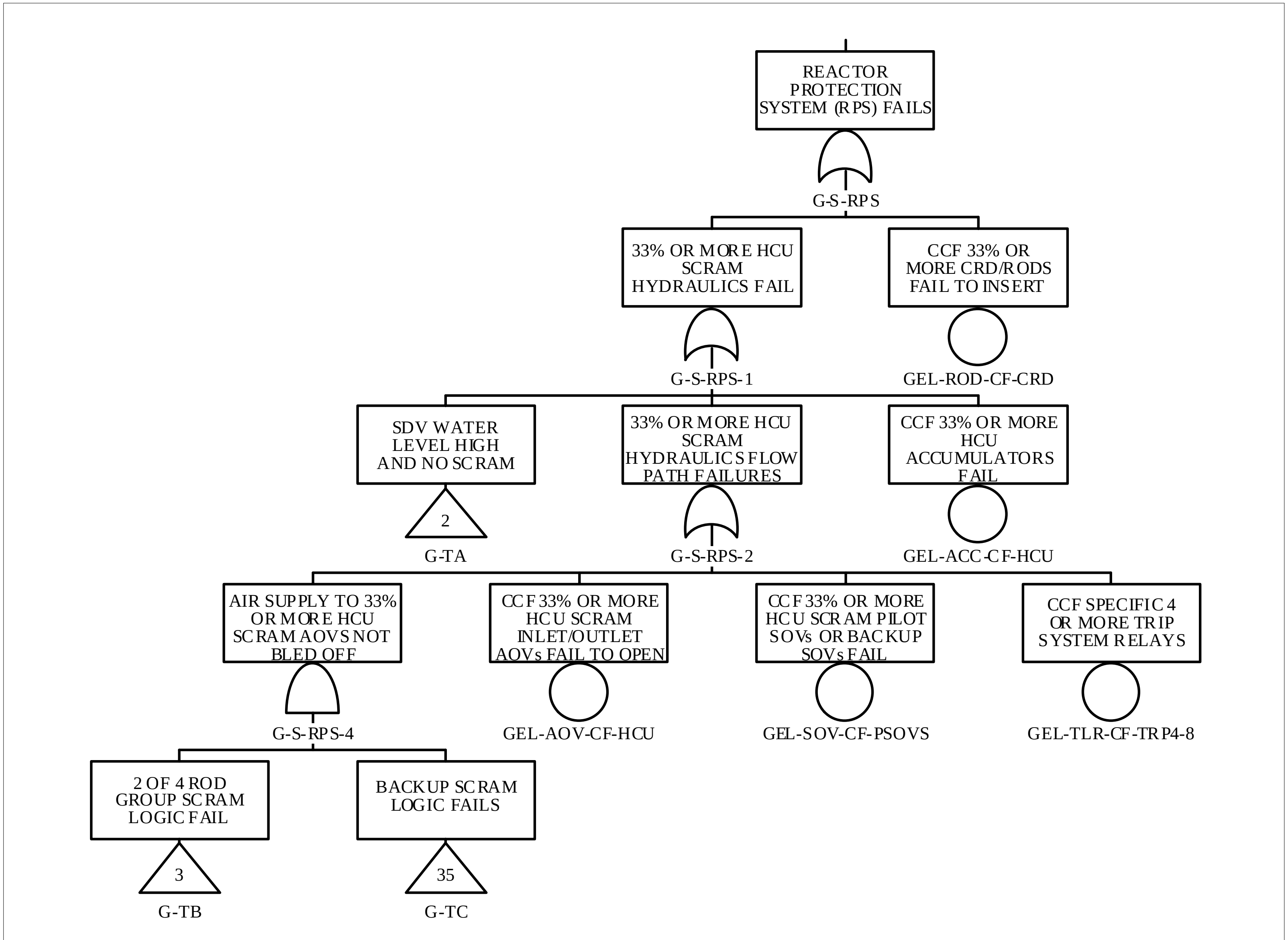
Appendix D

Fault Tree

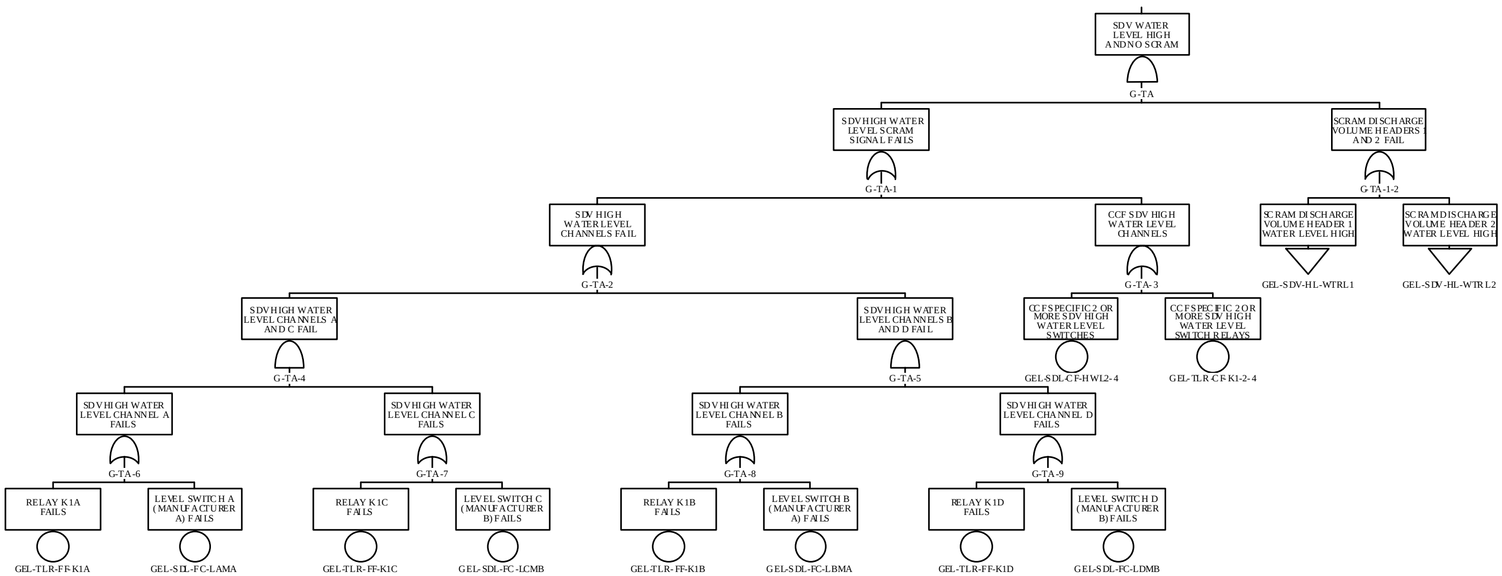
Appendix D

Fault Tree

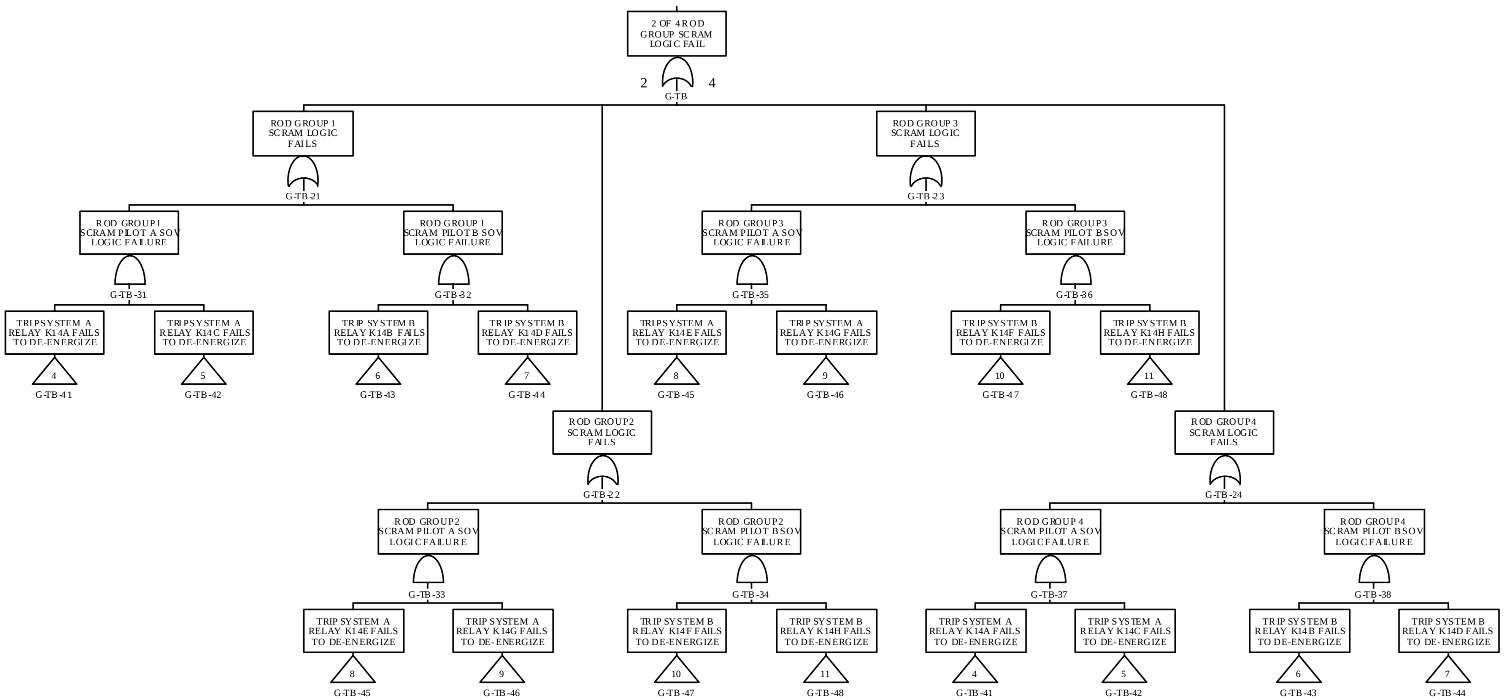
This appendix contains the reactor protection system (RPS) fault tree representing the General Electric RPS design. The number near the bottom of transfer gates indicates the fault tree page number (shown in the lower right corner of the fault tree border) where the logic is transferred.



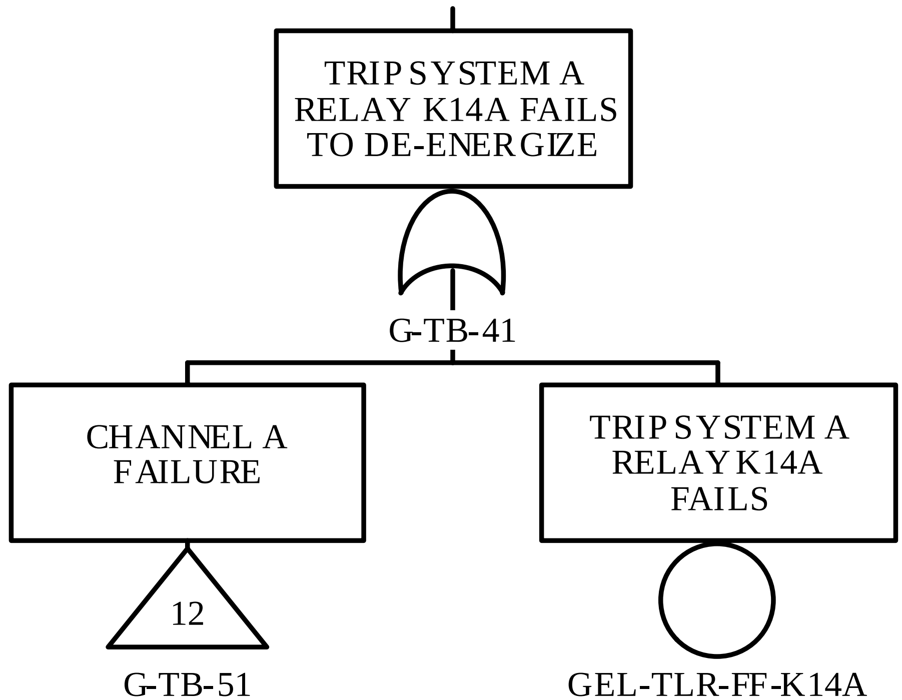
G-S-RPS - REACTOR PROTECTION SYSTEM (RPS) FAILS 1998/10/20 Page 1



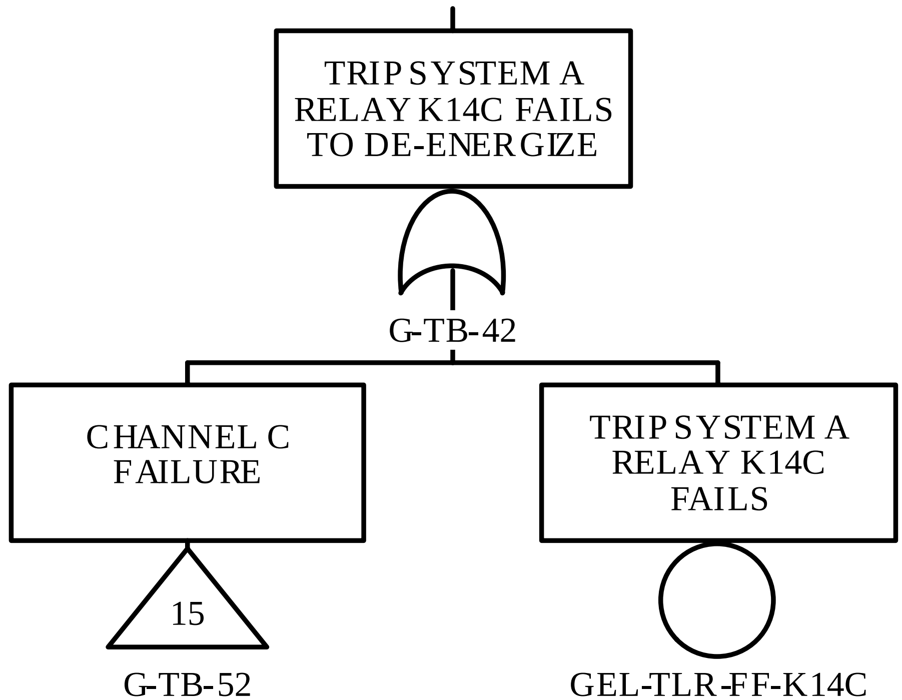
G-TA - SDV WATER LEVEL HIGH AND NO SCRAM 1998/10/20 Page 2



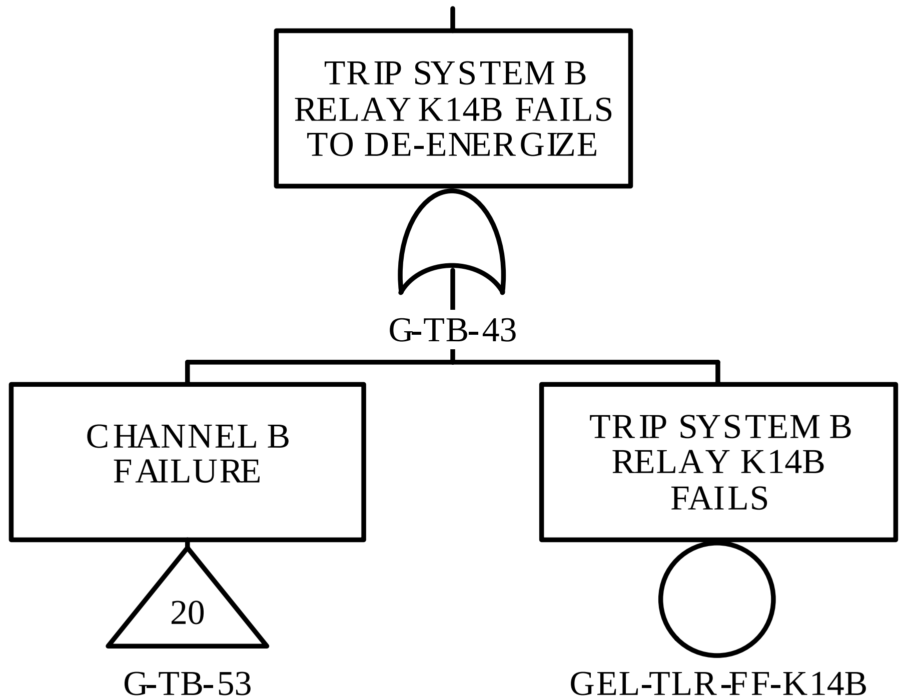
G-TB - 2 OF 4 ROD GROUP SCRAM LOGIC FAIL 1998/10/20 Page 3



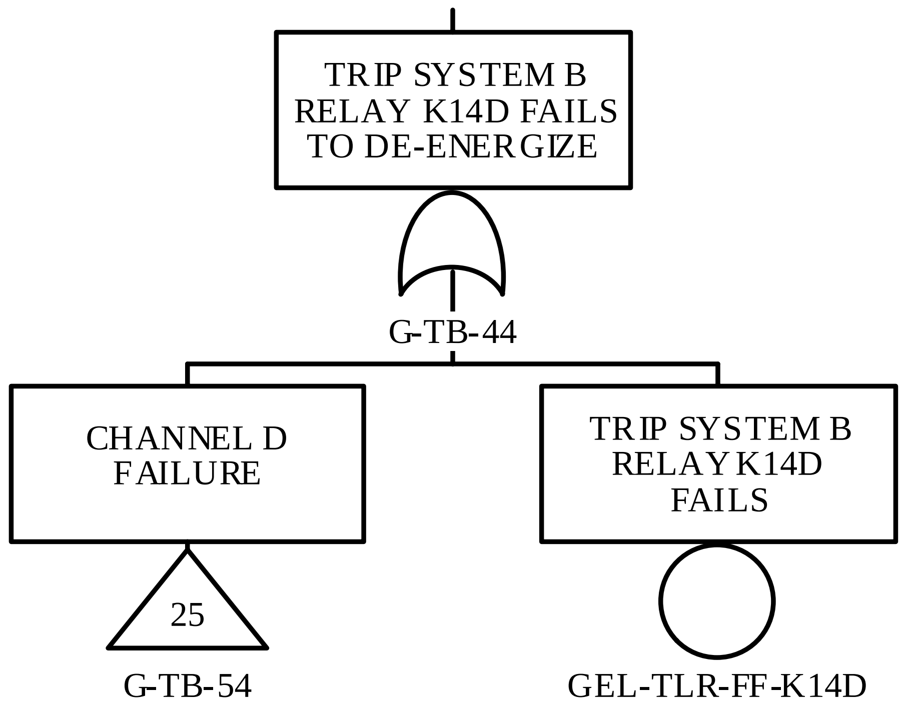
G-TB-41 - TRIP SYSTEM A RELAY K14A FAILS TO DE-ENERGIZE 1998/08/19 Page 4



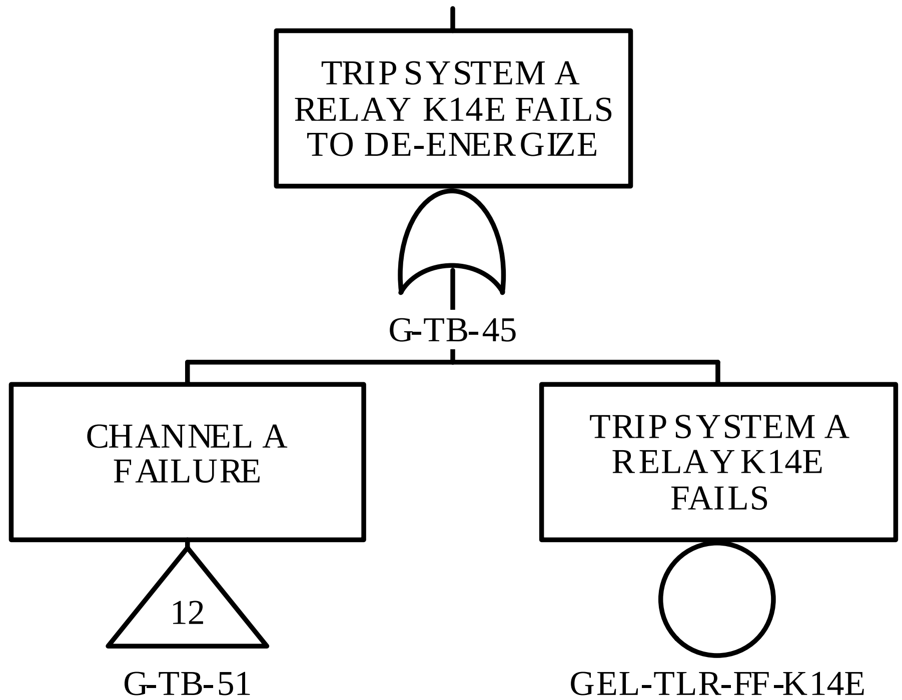
G-TB-42 - TRIP SYSTEM A RELAY K14C FAILS TO DE-ENERGIZE 1998/08/19 Page 5



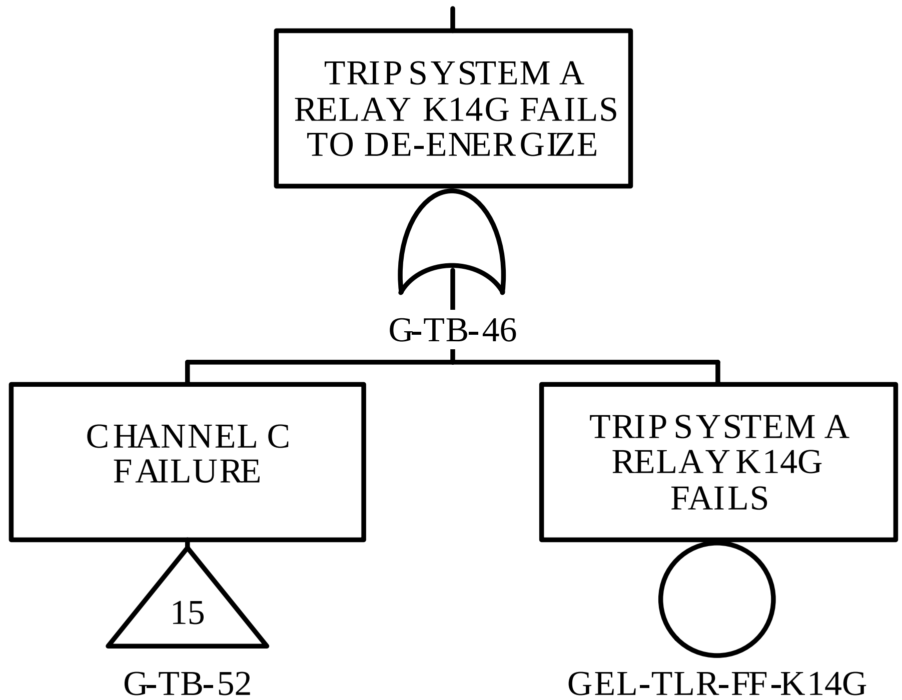
G-TB-43 - TRIP SYSTEM B RELAY K14B FAILS TO DE-ENERGIZE 1998/08/19 Page 6



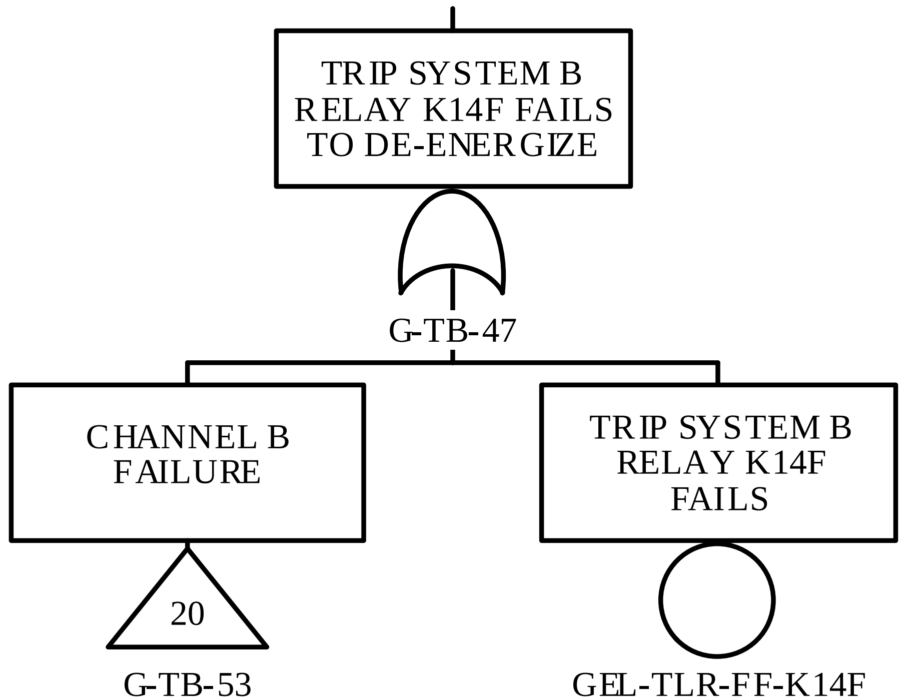
G-TB-44 - TRIP SYSTEM B RELAY K14D FAILS TO DE-ENERGIZE 1998/08/19 Page 7



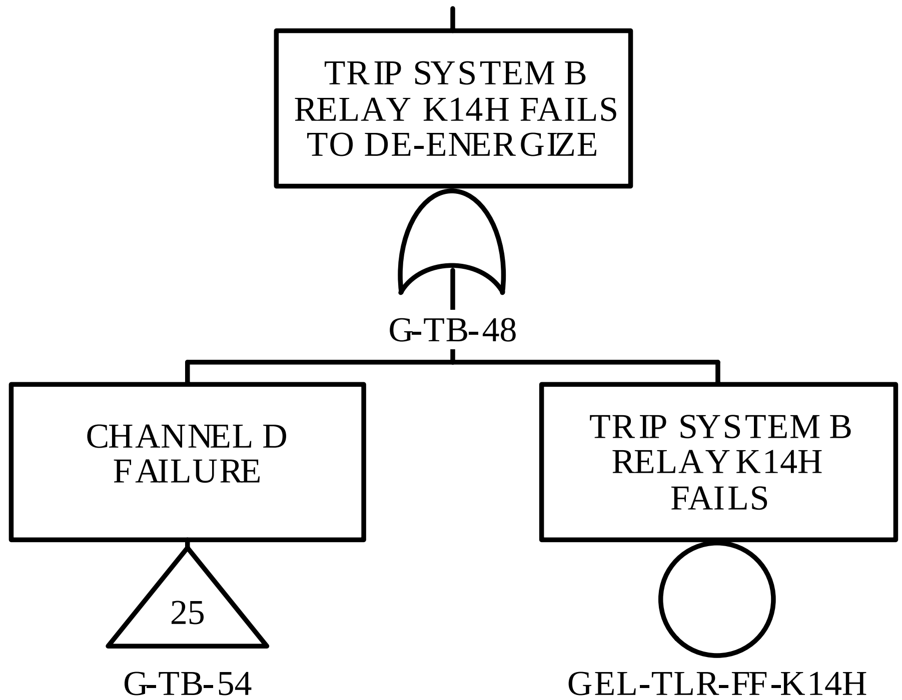
G-TB-45 - TRIP SYSTEM A RELAY K14E FAILS TO DE-ENERGIZE 1998/08/19 Page 8



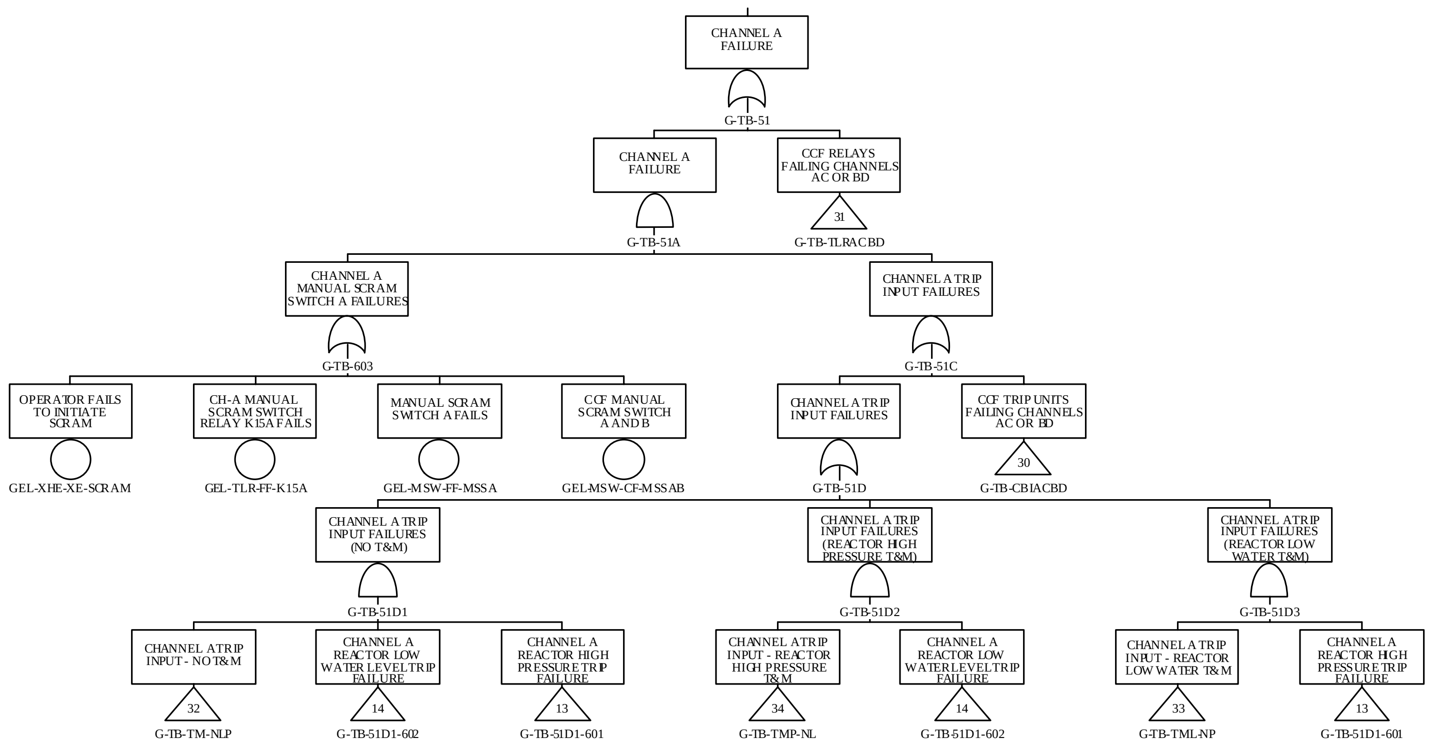
G-TB-46 - TRIP SYSTEM A RELAY K14G FAILS TO DE-ENERGIZE 1998/08/19 Page 9



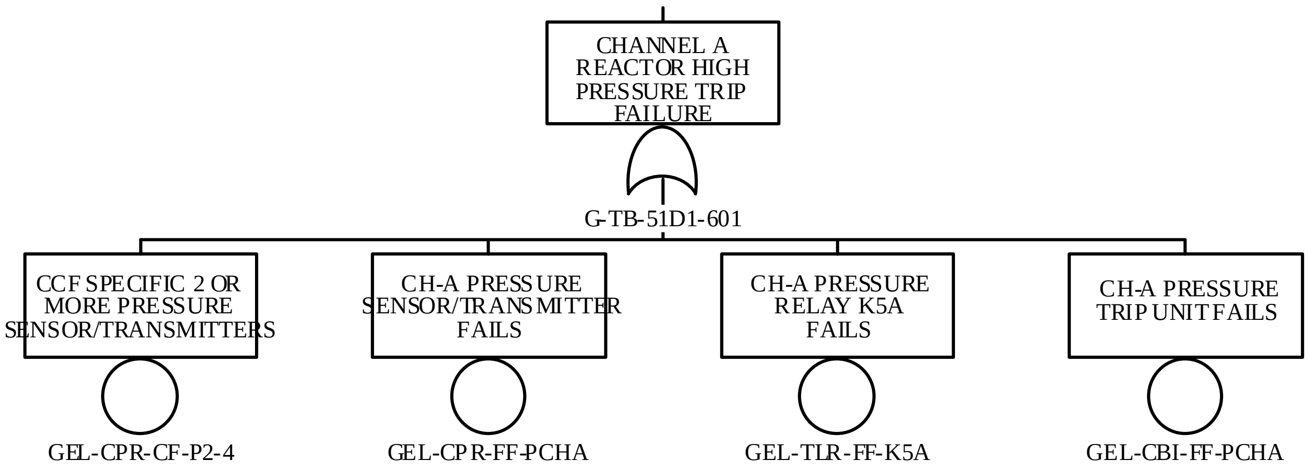
G-TB-47 - TRIP SYSTEM B RELAY K14F FAILS TO DE-ENERGIZE 1998/08/19 Page 10



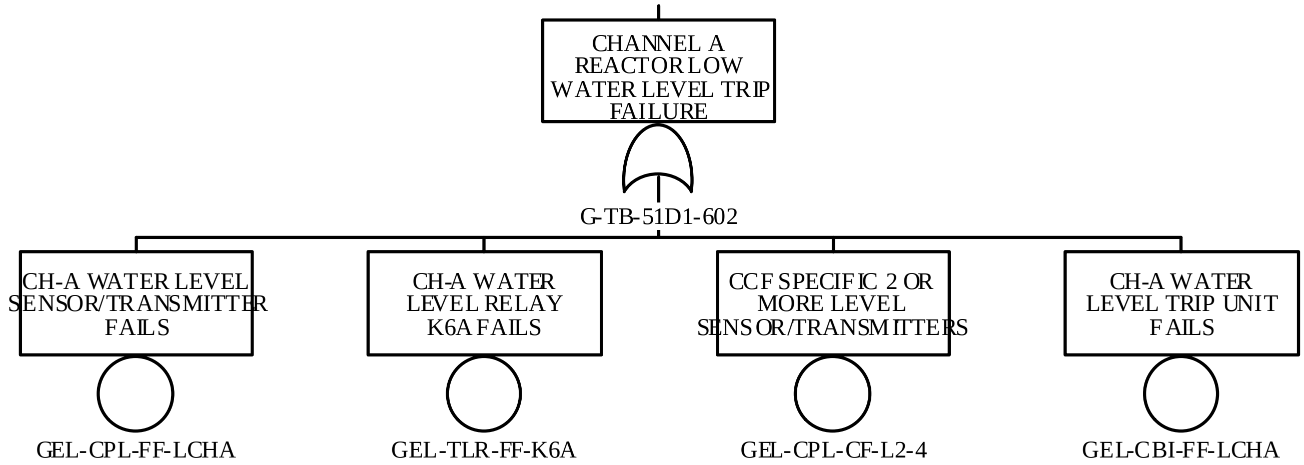
G-TB-48 - TRIP SYSTEM B RELAY K14H FAILS TO DE-ENERGIZE 1998/08/19 Page 11



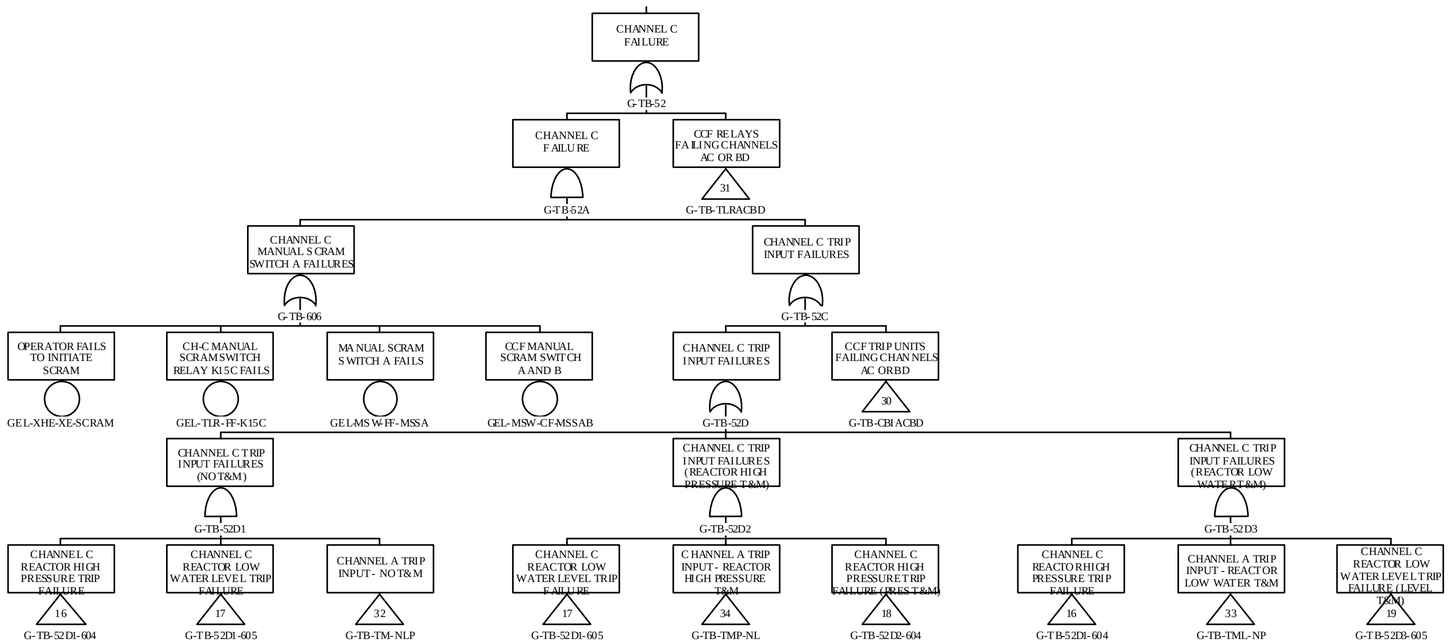
G-TB-51 - CHANNEL A FAILURE 1998/08/20 Page 12



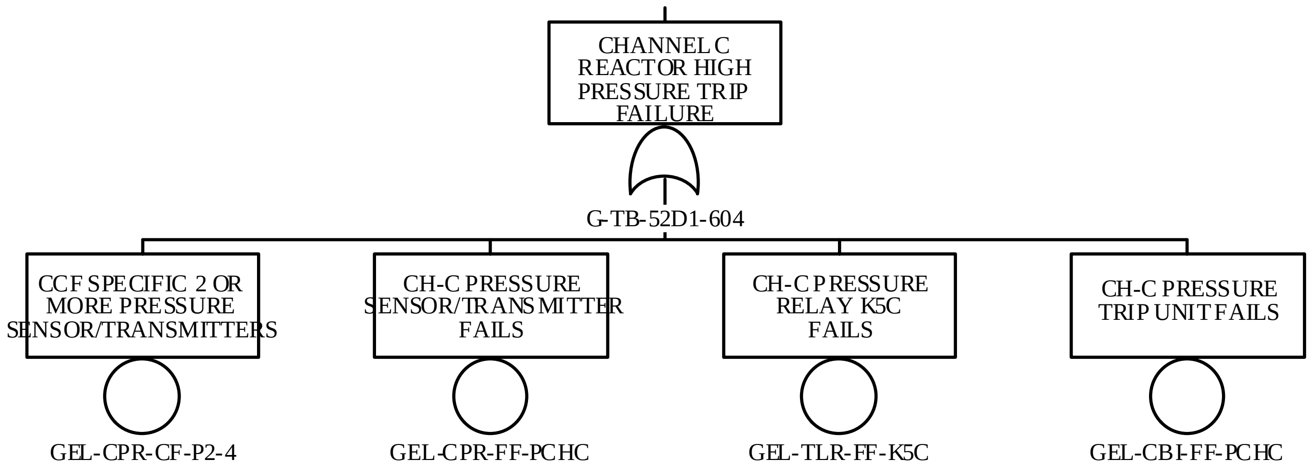
G-TB-51D1-601 - CHANNEL A REACTOR HIGH PRESSURE TRIP FAILURE 1998/08/19 Page 13



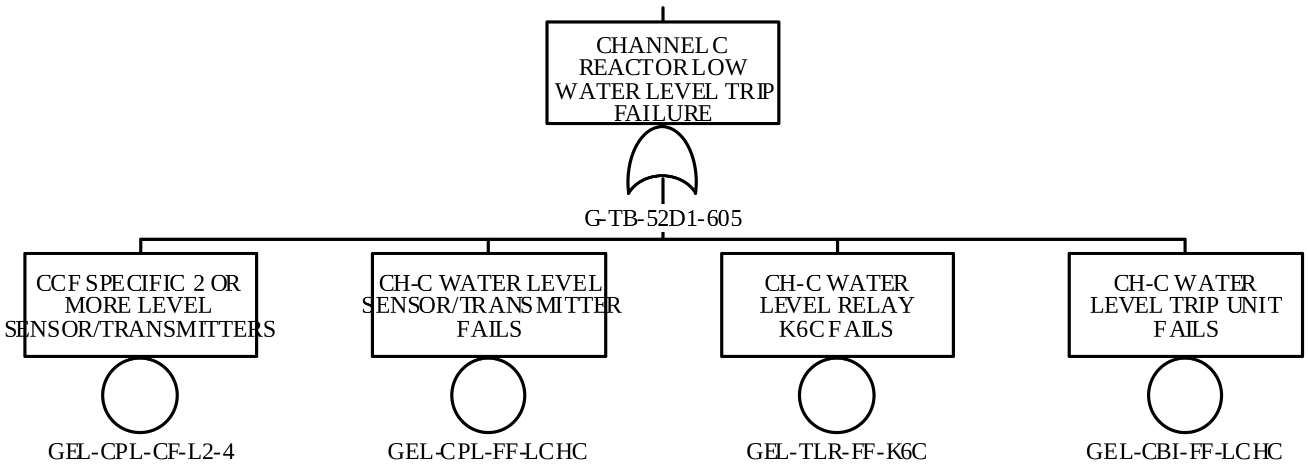
G-TB-51D1-602 - CHANNEL A REACTOR LOW WATER LEVEL TRIP FAILURE 1998/08/19 Page 14



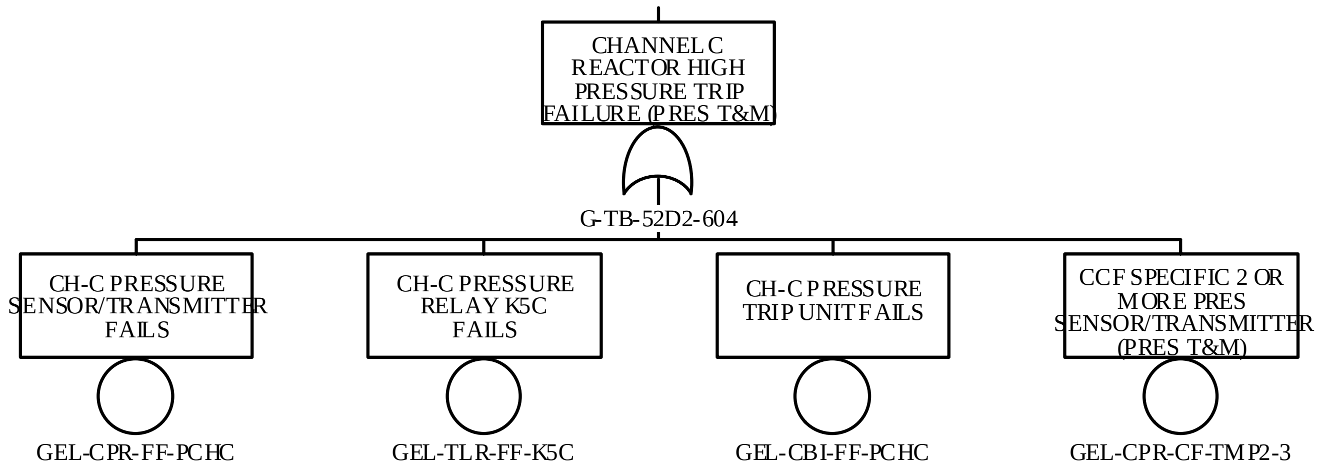
G-TB-52 - CHANNEL C FAILURE 1998/08/20 Page 15



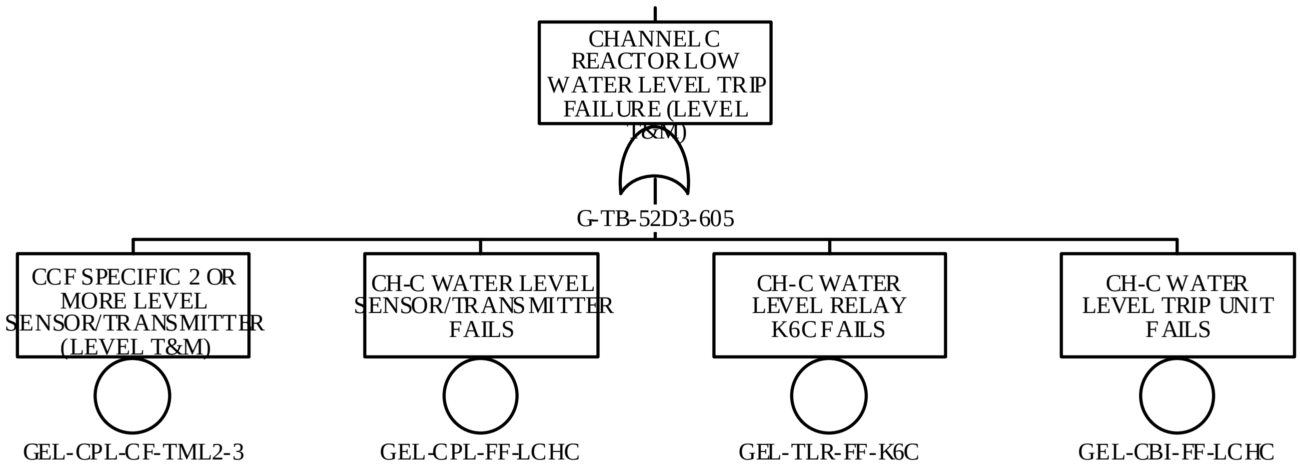
G-TB-52D1-604 - CHANNEL C REACTOR HIGH PRESSURE TRIP FAILURE 1998/08/19 Page 16



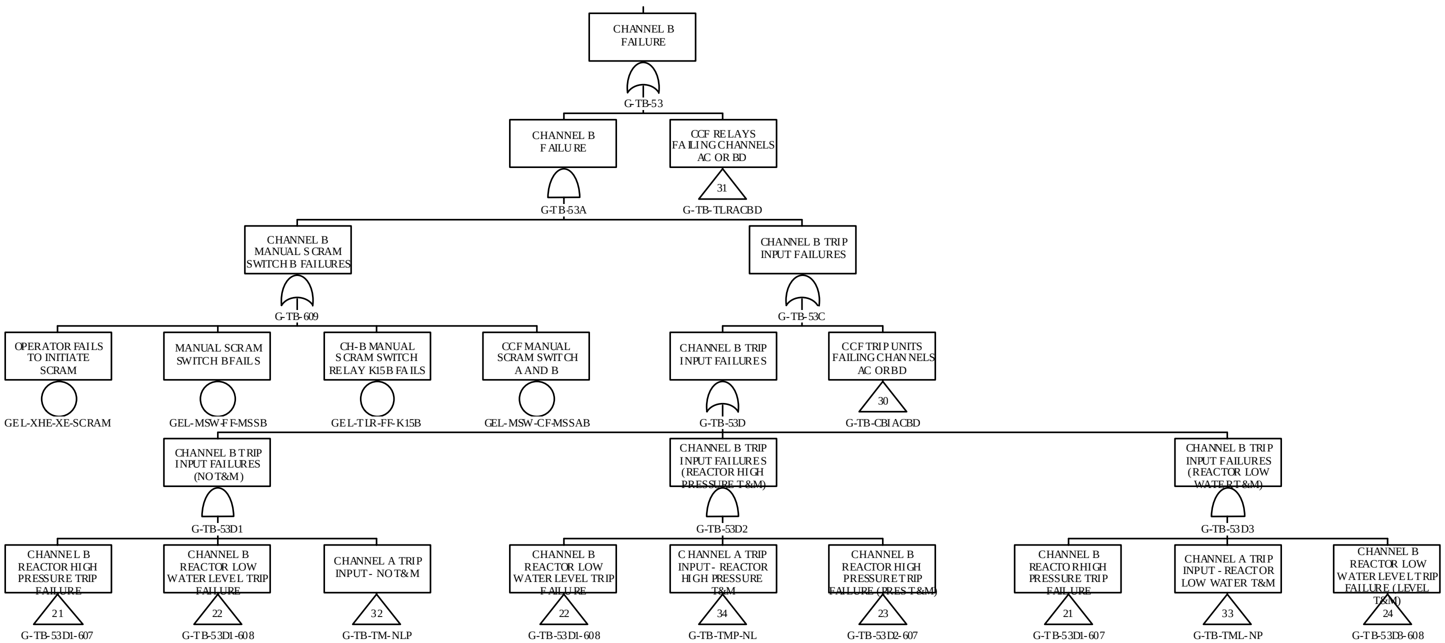
G-TB-52D1-605 - CHANNEL C REACTOR LOW WATER LEVEL TRIP FAILURE 1998/08/19 Page 17



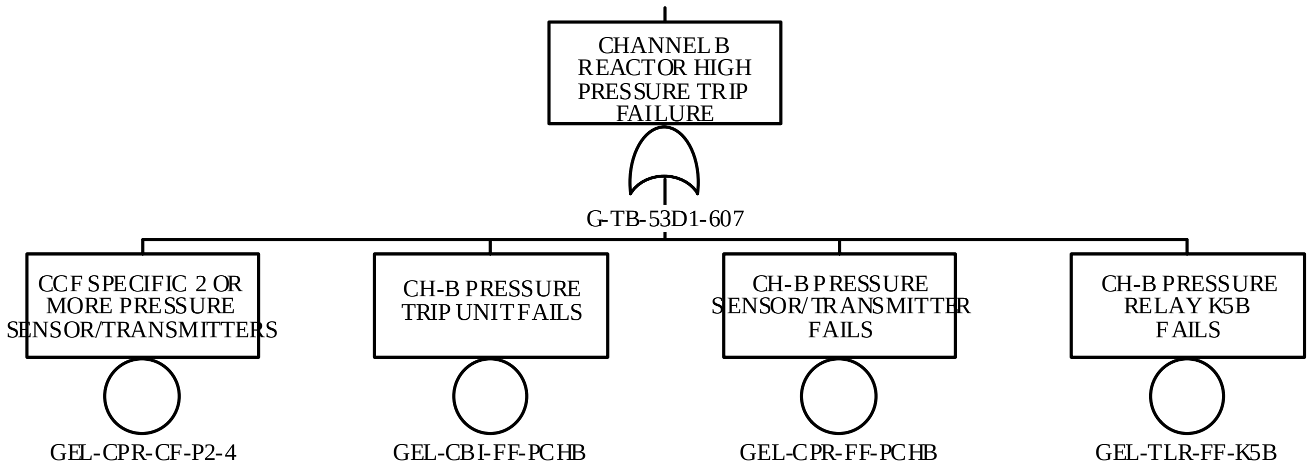
G-TB-52D2-604 - CHANNEL C REACTOR HIGH PRESSURE TRIP FAILURE (PRES T&M) 1998/08/24 Page 18



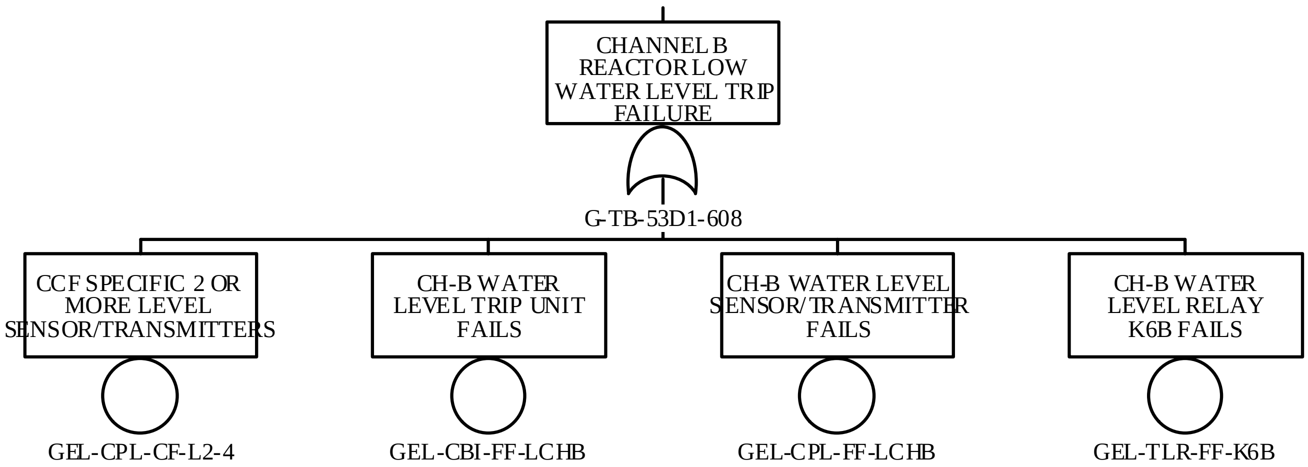
G-TB-52D3-605 - CHANNEL C REACTOR LOW WATER LEVEL TRIP FAILURE (LEVEL T&M) 1998/08/24 Page 19



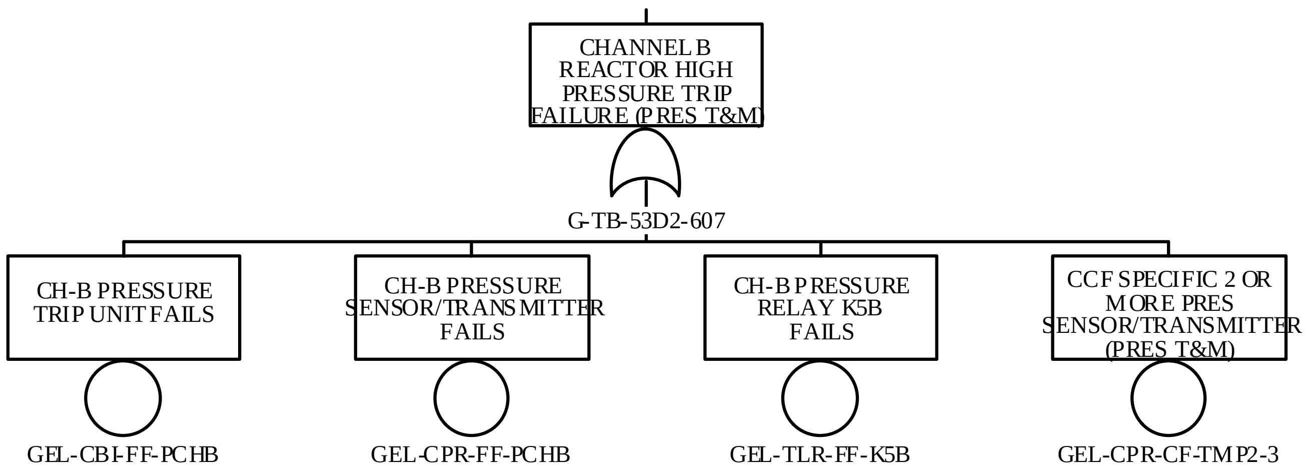
G-TB-53 - CHANNEL B FAILURE 1998/08/20 Page 20



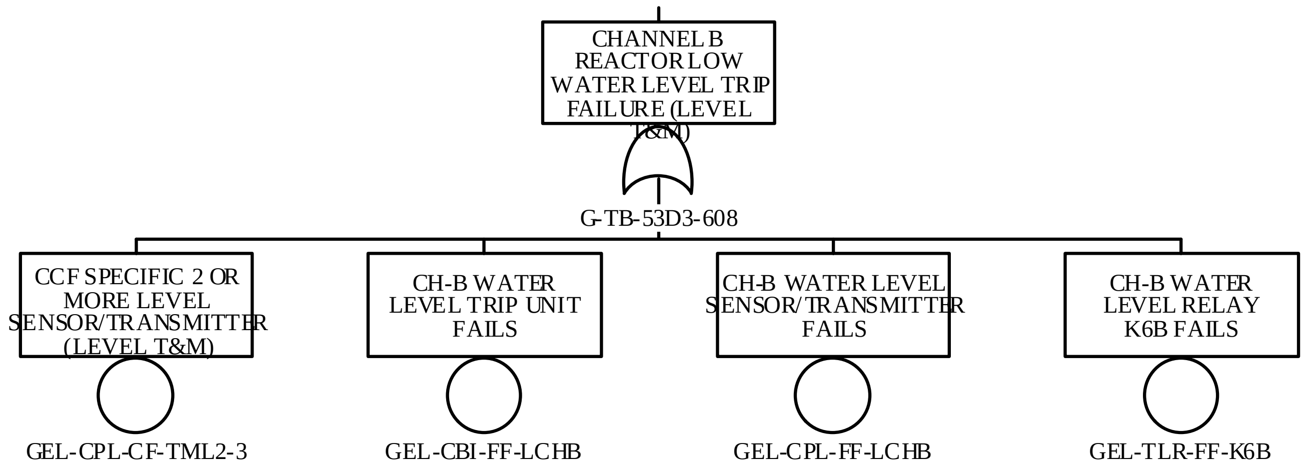
G-TB-53D1-607 - CHANNEL B REACTOR HIGH PRESSURE TRIP FAILURE 1998/08/19 Page 21



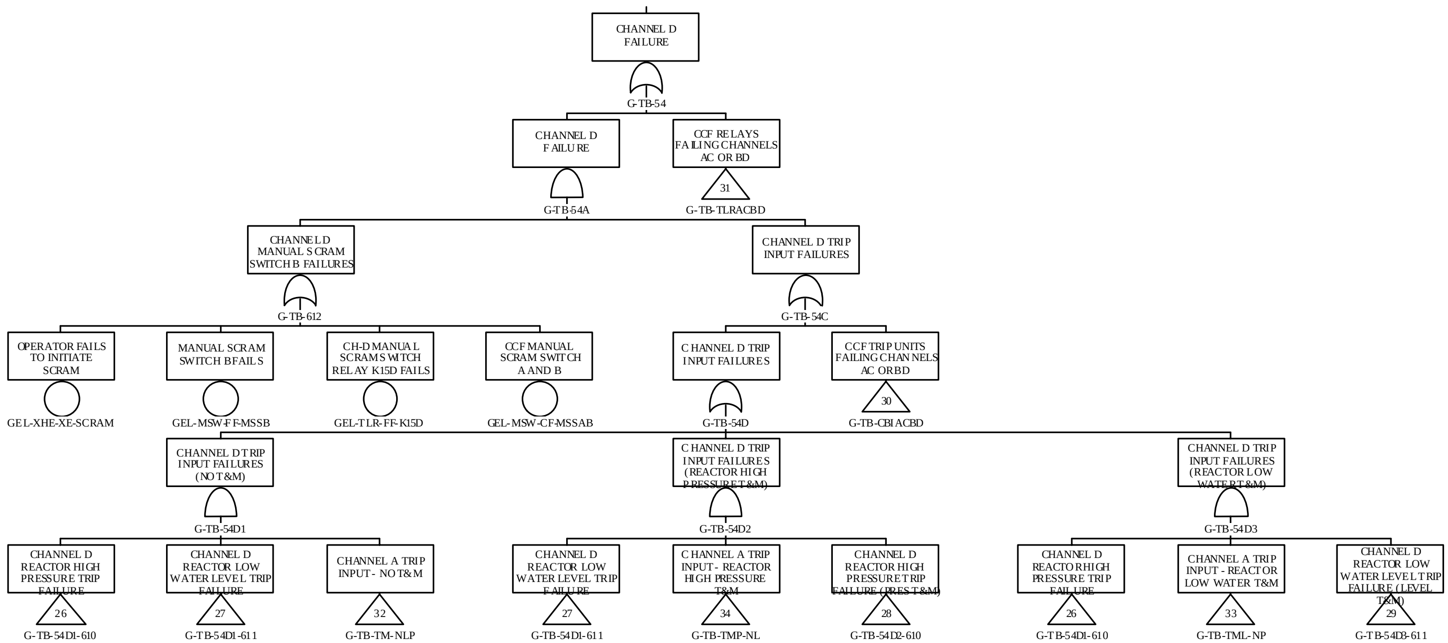
G-TB-53D1-608 - CHANNEL B REACTOR LOW WATER LEVEL TRIP FAILURE 1998/08/19 Page 22



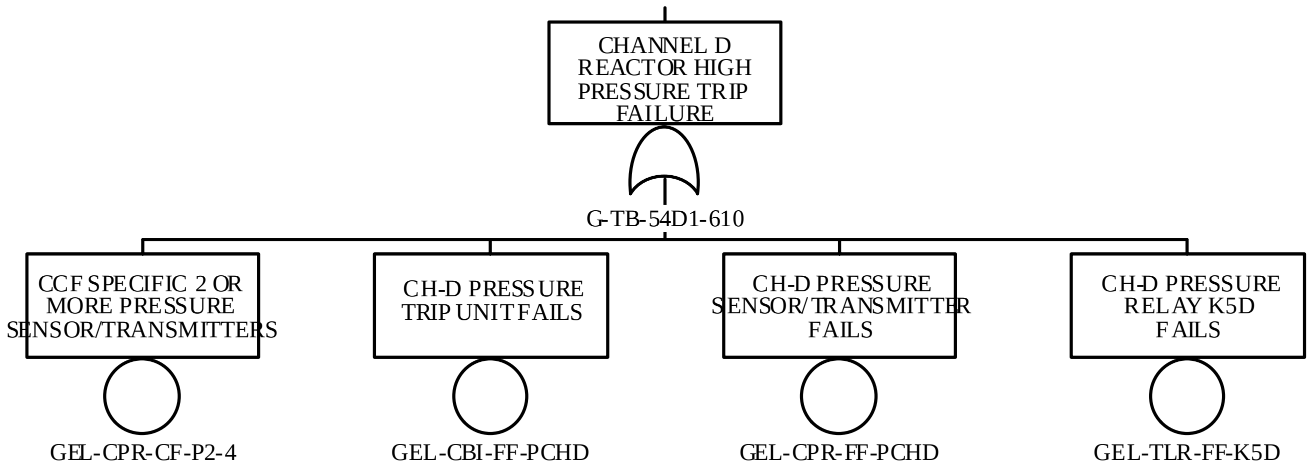
G-TB-53D2-607 - CHANNEL B REACTOR HIGH PRESSURE TRIP FAILURE (PRES T&M) 1998/08/24 Page 23



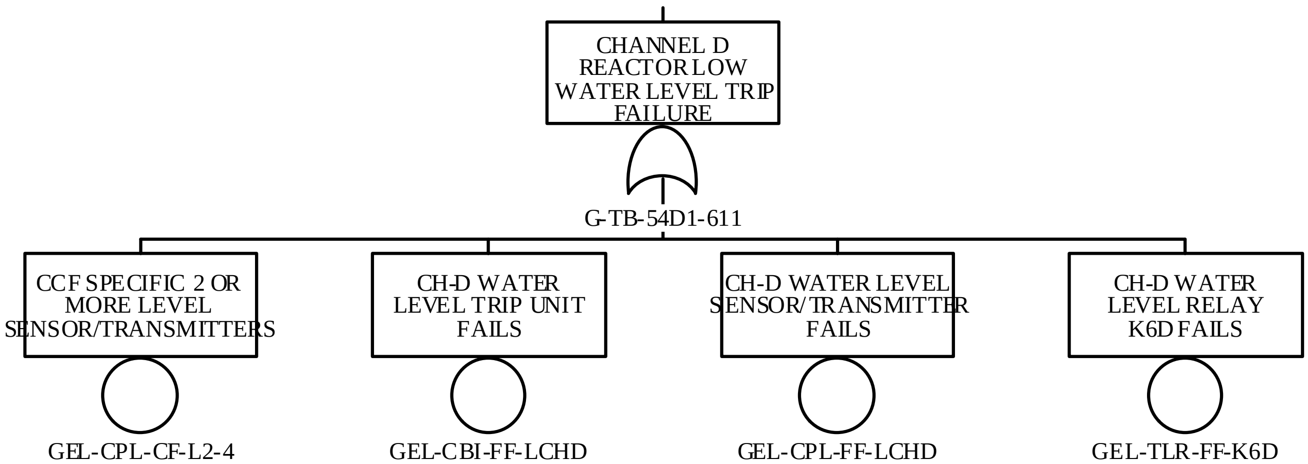
G-TB-53D3-608 - CHANNEL B REACTOR LOW WATER LEVEL TRIP FAILURE (LEVEL T&M) 1998/08/24 Page 24



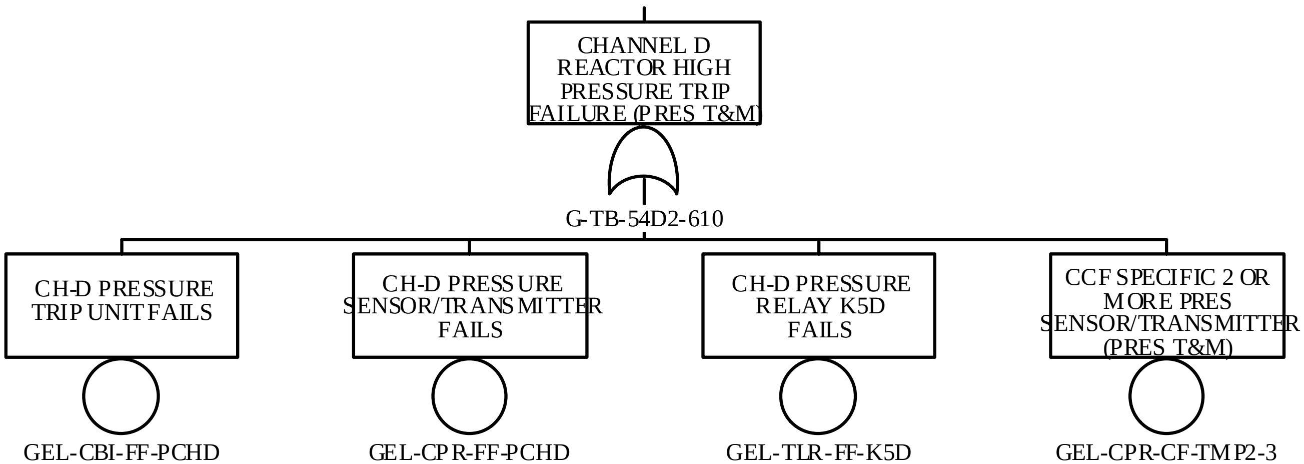
G-TB-54 - CHANNEL D FAILURE 1998/08/20 Page 25



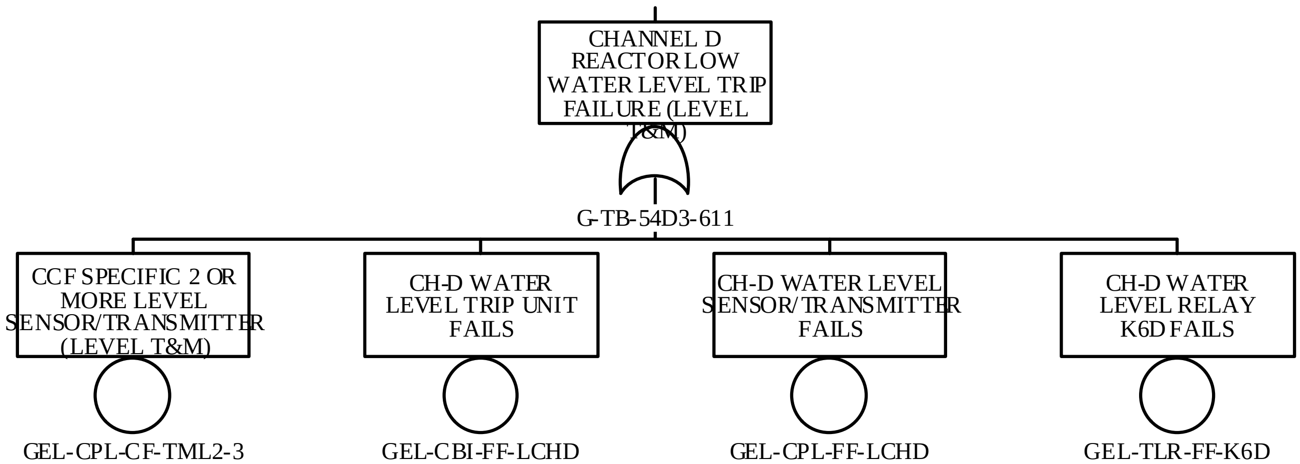
G-TB-54D1-610 - CHANNEL D REACTOR HIGH PRESSURE TRIP FAILURE 1998/08/19 Page 26



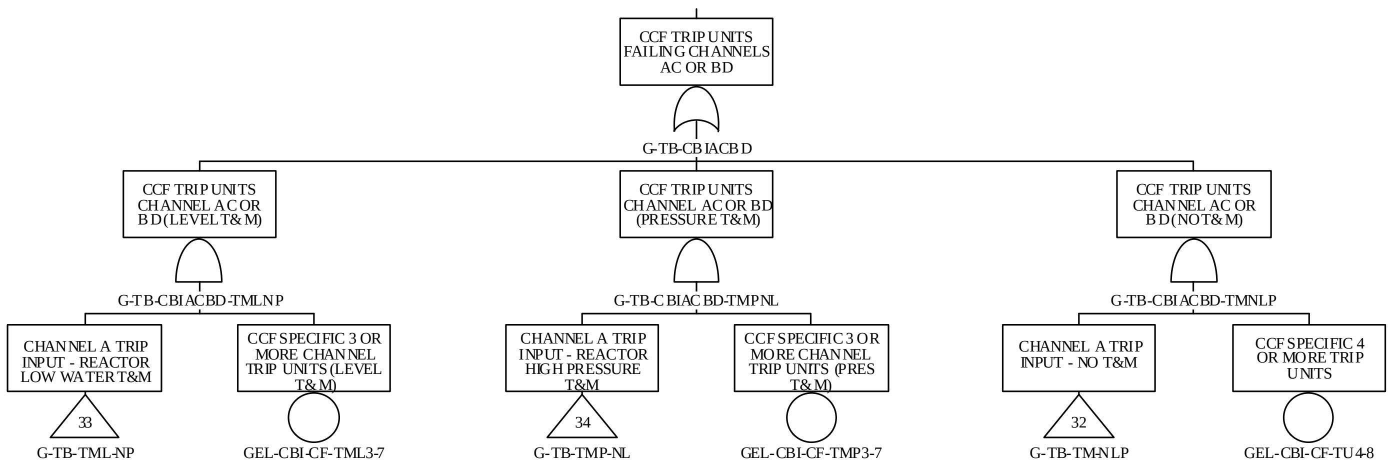
G-TB-54D1-611 - CHANNEL D REACTOR LOW WATER LEVEL TRIP FAILURE 1998/08/19 Page 27



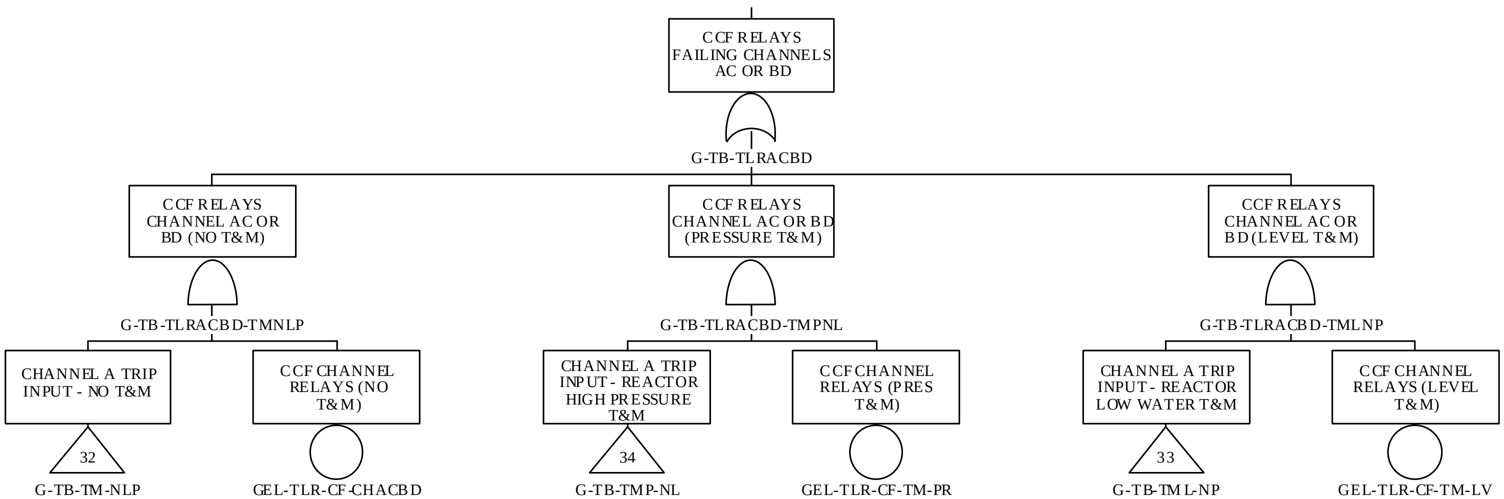
G-TB-54D2-610 - CHANNEL D REACTOR HIGH PRESSURE TRIP FAILURE (PRES T&M) 1998/08/24 Page 28



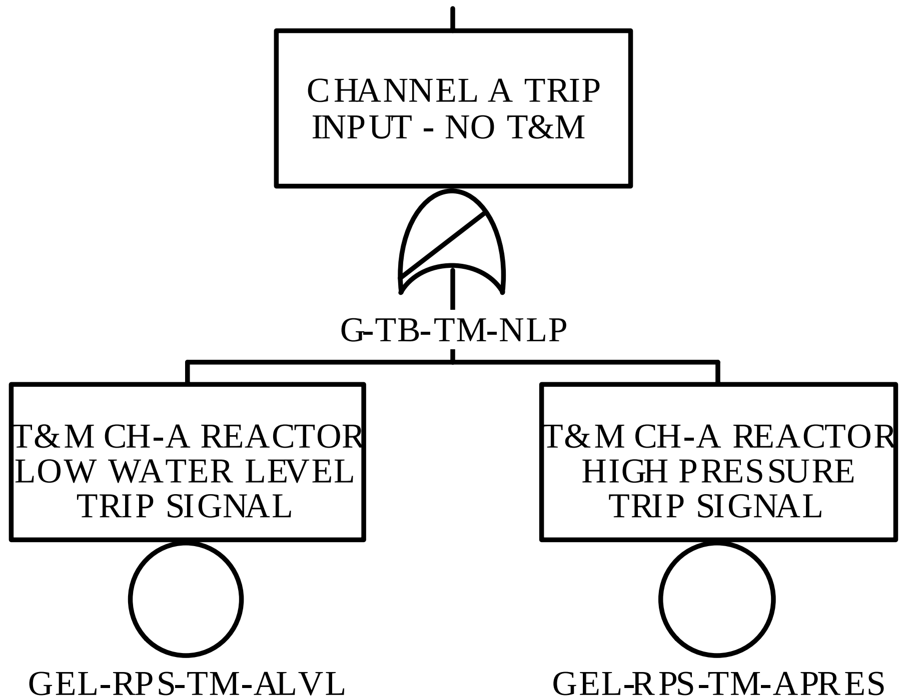
G-TB-54D3-611 - CHANNEL D REACTOR LOW WATER LEVEL TRIP FAILURE (LEVEL T&M) 1998/08/24 Page 29



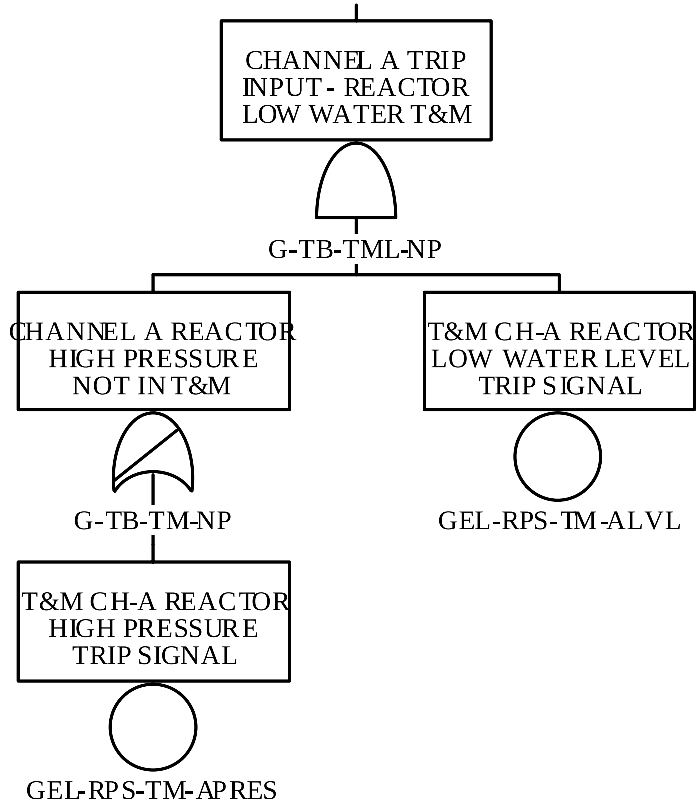
G-TB-CBIACBD - CCF TRIP UNITS FAILING CHANNELS AC OR BD 1998/08/20 Page 30



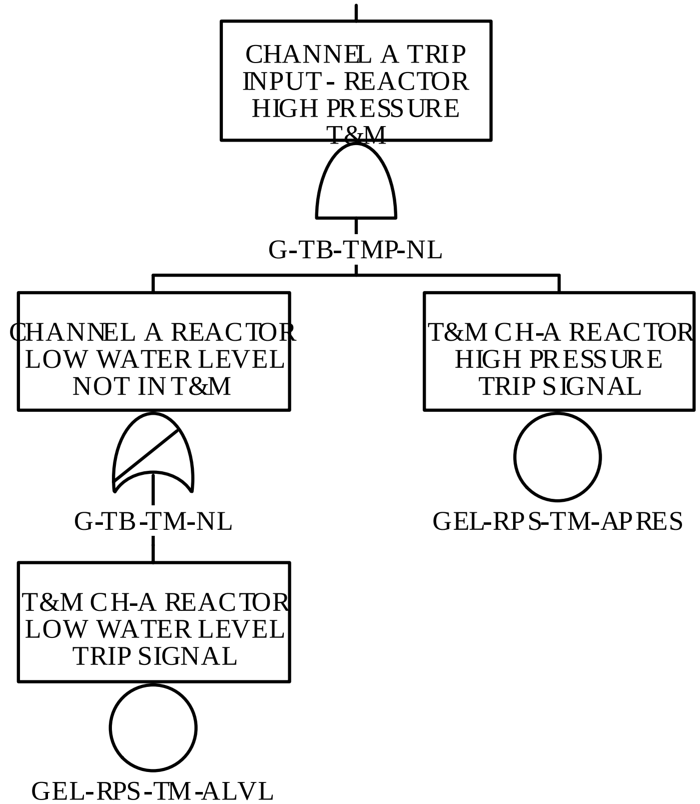
G-TB-TLRACBD - CCF RELAYS FAILING CHANNELS AC OR BD 1998/10/20 Page 31



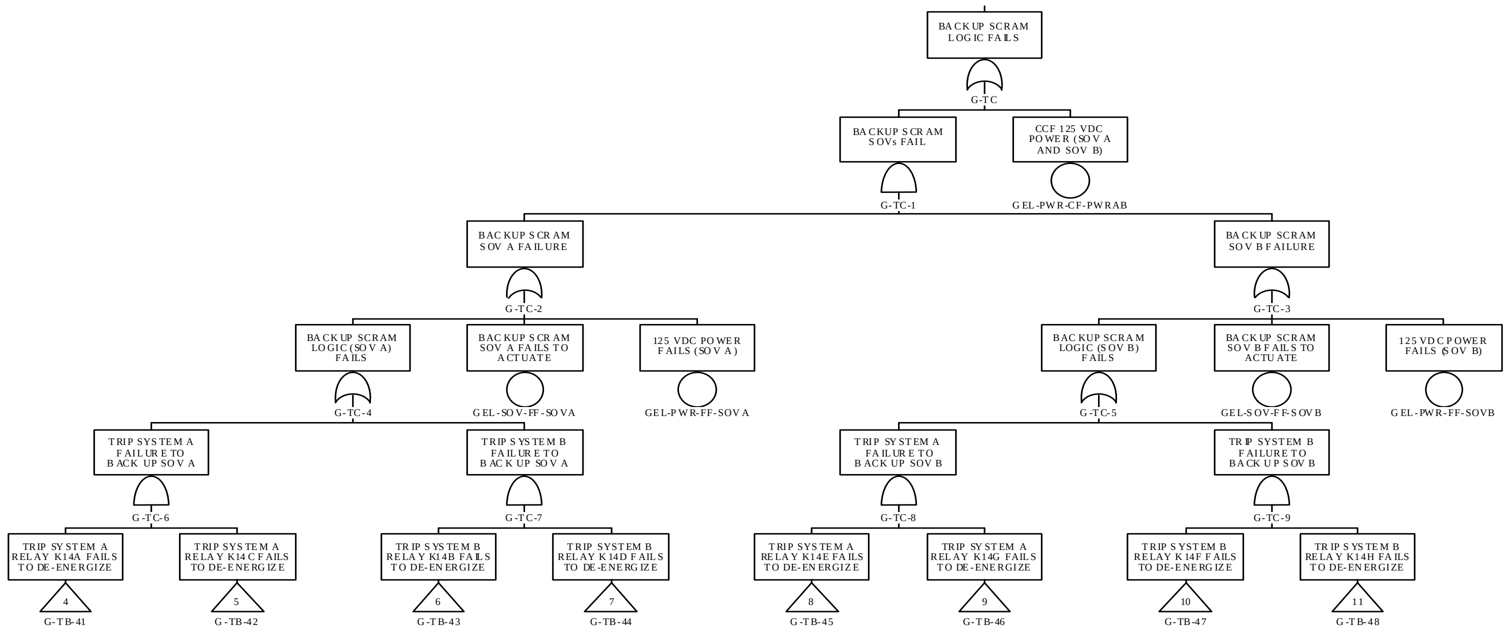
G-TB-TM-NLP - CHANNEL A TRIP INPUT - NO T&M 1998/08/19 Page 32



G-TB-TML-NP - CHANNEL A TRIP INPUT - REACTOR LOW WATER T&M 1998/08/19 Page 33



G-TB-TMP-NL - CHANNEL A TRIP INPUT - REACTOR HIGH PRESSURE T&M 1998/08/19 Page 34



G-TC - BACKUP SCRAM LOGIC FAILS 1998/10/20 Page 35

Appendix E

Common-Cause Failure Analysis

Appendix E

Common-Cause Failure Analysis

E-1. INTRODUCTION

This appendix presents general information on the subject of common-cause failure (CCF) and special techniques developed for the reactor protection system (RPS) study. Included are sections that discuss background, methodology, the RPS CCF database, the prior, special software developed for this study, calculation of CCF basic event (BE) probabilities, and sensitivities. Throughout this section, component codes (e.g., CPR) are used when referring to components used in the RPS study. These codes are defined in the acronym list at the beginning of this report.

E-1.1 CCF Event Definition

A CCF event consists of component failures that meet four criteria: (1) two or more individual components fail or are degraded, including failures during demand, in-service testing, or deficiencies that would have resulted in a failure if a demand signal had been received; (2) components fail within a selected period of time, such that success of the probabilistic risk assessment (PRA) mission would be uncertain; (3) component failures result from a single shared cause and coupling mechanism; and (4) component failures are not due to failures of equipment outside the established component boundary.

Two data sources are used to select equipment failure reports to be reviewed for CCF event identification. The first is the Nuclear Plant Reliability Data System (NPRDS), which contains component failure information. The second one is the Sequence Coding and Search System (SCSS), which contains Licensee Event Reports (LERs).

The CCF event identification process includes a review of failure data to identify CCF events and independent failure event counts. The identification process allows the analyst to consistently screen failures and identify CCF events. The CCF event coding process provides guidance for the analyst to consistently code CCF events. Sufficient information is recorded to ensure accuracy and consistency. Additionally, the CCF events are stored in a format that allows PRA analysts to review the events and develop an understanding of CCF phenomenology.

E-1.2 Approach

The calculation of a CCF BE probability is a multi-step process. The fault trees developed for the RPS study identified CCF events that contributed to the possible failure of the RPS to successfully initiate a reactor scram. The data review and calculation of those CCF BE probabilities was driven by those needs. Figure E-1 shows a process flow diagram outlining the steps necessary to calculate a CCF BE probability. The step involving analysis of failure events is discussed in Appendices A and C. Fault tree development, defining CCF BE criteria, and component boundary definitions are discussed in Section 2 of the main body of this report.

A brief review of the CCF calculations is presented in this appendix to familiarize the reader with the terminology. More information can be found in the report *Common-Cause Failure Database and Analysis System: Event Definition and Classification*.[E-1]

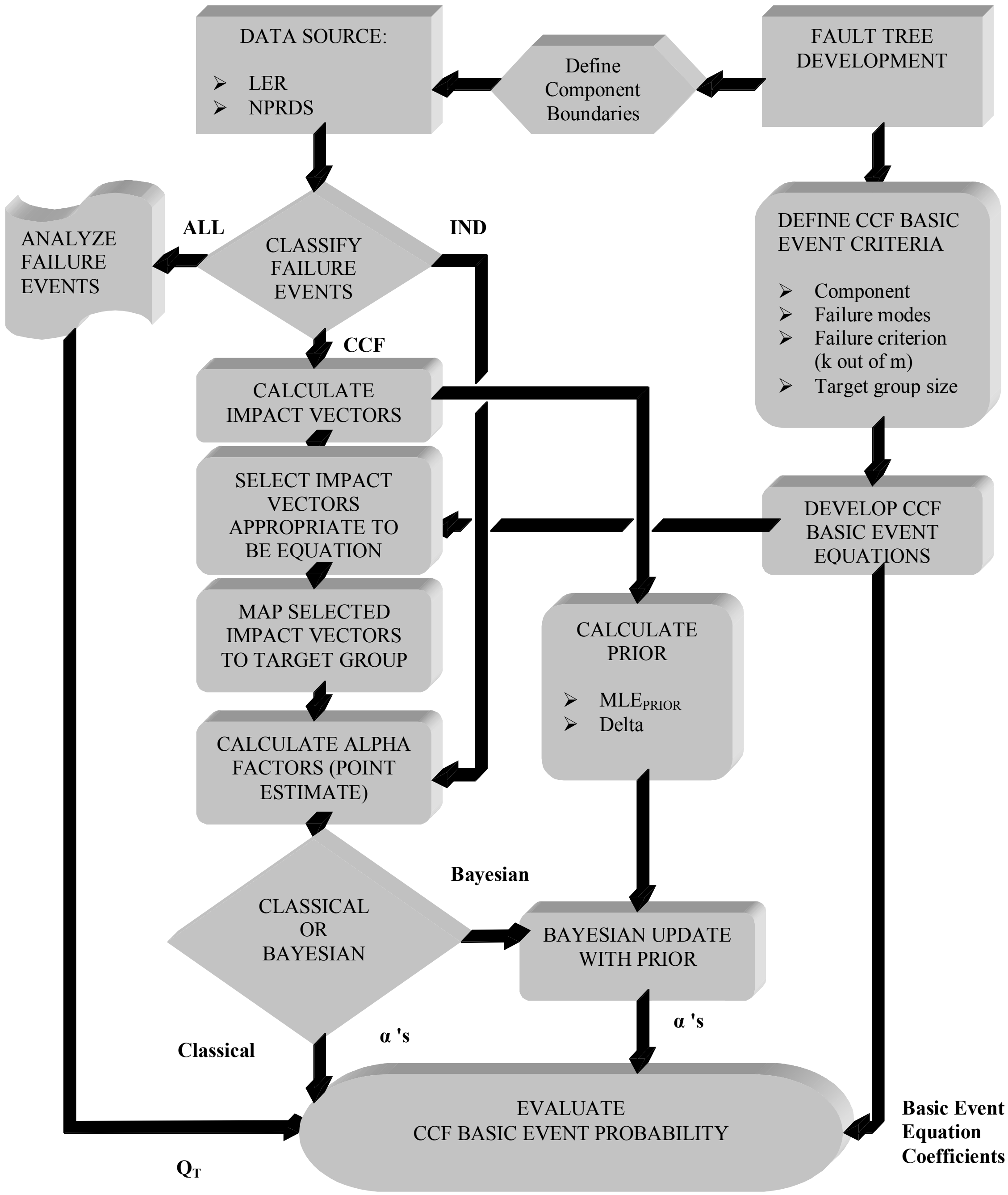


Figure E-1. CCF process flow diagram.

E-2. CCF MODEL

This section provides information on the type of CCF model used in this study and describes the process of developing the CCF BE equation.

E-2.1 Alpha Model

In order to provide estimates of the probability of a common-cause event involving *k* specific components in a common-cause component group (CCCG) of size *m*, a model needed to be selected from among the available models. The available models included the Basic Parameter model, the Beta model, the Multiple Greek Letter (MGL) model, and the Alpha Factor model.

The parametric Alpha Factor model was chosen. Reasons for this choice are that the alpha factor model (1) is a multi-parameter model, which can handle any redundancy level, (2) is based on ratios of failure rates which makes the assessment of its parameters easier when no statistical data are available, and (3) has a simpler statistical model, and produces more accurate point estimates as well as uncertainty distributions compared to other parametric models which have the above two properties.

The alpha factor model estimates CCF frequencies from a set of ratios of failures and the total component failure rate. The parameters of the model are:

Q_T ≡ total failure frequency of each component (includes independent and common-cause events)

$\alpha^{(m)}_k$ ≡ fraction of the total frequency of failure events that occur in the system involving the failure of *k* components in a system of *m* components due to a common-cause.

E-2.2 CCF Basic Event Equation Development

Two types of failure criterion are used in the GE RPS study. The first is one-out-of-two-twice logic. This type of logic is used throughout the RPS instrumentation logic. The second type is any k of m combinations. This type is used in the ROD model.

E-2.2.1 One-Out-of-Two-Twice Logic

In terms of the alpha factor model, the BE probability for a specific *k* failures out of a system of *m* components (assuming a staggered testing scheme) is shown in Equation E-1.

$$BE_{CCF} = Q_T \sum_{i=k}^{m} C_i \frac{(m-i)!(i-1)!}{(m-1)!} \alpha_i^{(m)} \qquad \text{E-1}$$

where:

C_i ≡ number of combinations of *k* component failures that will fail the system

A *specific failure criterion* is represented by the C_i term in Equation E-1. An example of a one-out-of-two-twice logic failure criterion is shown in Figure E-2. This example applies to the 4/8 CBI CCF event used in the fault trees. In this example, the failure criterion is described in shorthand as 4/8. This is based on failure of two of two components to fail a channel and specific failure of two of four channels to fail a train. Some of the combinations of four component failures will fail two channels, but no trains

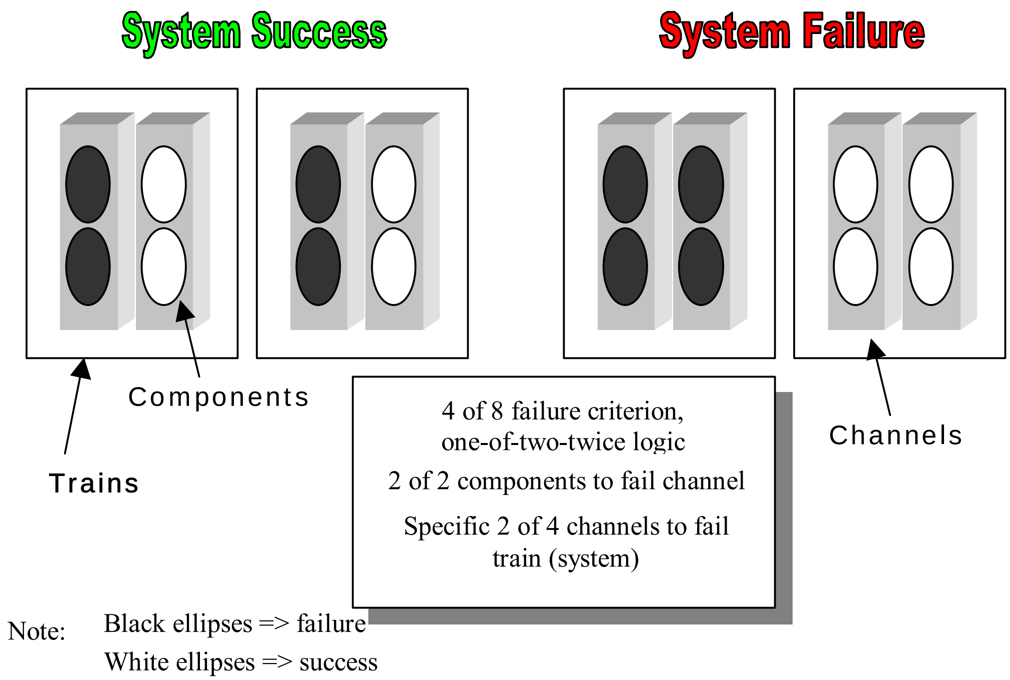


Figure E-2. Example of a one-of-two-twice logic failure criterion for a 4-out-of-8 system.

(e.g., those combinations where two failures are in each of two trains). Some combinations of four will fail an entire train, an example is shown in the failure side of Figure E-2. The valid failure combinations are counted and the sum becomes the C_i term in Equation E-1. When a component is taken out of service for maintenance, it is placed in a non-tripped (bypassed) status. The possible combinations are counted with the component always failed. This maintenance event is described in shorthand as 3/7 |8.

E-2.2.2 Any k of m Combinations

The form of the CCF BE equation for <u>any</u> *k* out of *m* components failing is given by Equation E-2 for staggered testing:

$$Q_{CCF} = Q_T \sum_{i=k}^{m} \frac{\binom{m}{i}}{\binom{m-1}{i-1}} \alpha_i = Q_T \sum_{i=k}^{m} \frac{m}{i} \alpha_i \qquad \text{E-2}$$

where:

α_i = the ratio of *i* and only *i* CCF failures to total failures

m = the number of total rods in the component group

k = the failure criteria for a number of rod failures in the component group

Q_T = the random failure rate (total)

Q_{CCF} = the failure probability of k and greater than k components due to CCF

E-2.2.3 Special Equations

Certain events required the development of special equations due to the nature of the failure criteria. The K14 trip relay 4 of 8 equation was manually developed with the specific criterion that both the backup scram solenoids as well as two of the four rod groups must fail. This equation is shown in Table E-1 and is denoted as K14Relay in the shorthand column.

The second event that needed special treatment is the event with both backup scram solenoids failed as well as 123 of 370 scram pilot solenoids failed. To estimate the basic event equation, we performed some algebraic manipulation of the all combinations equation with specific 2 of 2 failed. The result is shown in Equation E-3.

$$BE_{123/370}^{2/2} = Q_T * \sum_{i=123}^{370} \frac{m-1}{i-1} * \alpha_i \qquad \text{E-3}$$

E-2.2.4 CCF BE Probability Equations

Table E-1 shows the CCF BE probability equations used in the GE RPS study. All of the equations are based on staggered testing.

Table E-1. Failure criteria and basic event equation table.

Failure Criteria: Channel or Train Level	Failure Criteria: Component (within channel or train)	Shorthand Criterion[a]	Basic Event Probability Equations
2/2	1/1	2/2	$\alpha_2 * Q_T$
2/4	1/1	2/4	$(\alpha_4 + 4\alpha_3/3 + 2/3\alpha_2) * Q_T$
2/4	1/1	1/3|4[b]	$(\alpha_4 + 4/3\ \alpha_3 + 4/3\ \alpha_2) * Q_T$
2/4	2/2	4/8	$(\alpha_8 + 8\alpha_7/7 + 12\alpha_6/21 + 8\alpha_5/35 + 2\alpha_4/35) * Q_T$
2/4	2/2	3/7 |8	$(\alpha_8 + 8\alpha_7/7 + 16\alpha_6/21 + 14\alpha_5/35 + 6\alpha_4/35 + \alpha_3/21) * Q_T$
2/4	3/3	6/12	$(\alpha_{12} + 12\alpha_{11}/11 + 30\alpha_{10}/55 + 40\alpha_9/165 + 30\alpha_8/330 + 12\alpha_7/462 + 2\alpha_6/462) * Q_T$
2/4	3/3	5/11 |12	$(\alpha_{12} + 12\alpha_{11}/11 + 36\alpha_{10}/55 + 55\alpha_9/165 + 50\alpha_8/330 + 27\alpha_7/462 + 8\alpha_6/462 + \alpha_5/330) * Q_T$
2/4	2/2	K14Relay	$(\alpha_8 + 8\alpha_7/7 + 20\alpha_6/21 + 16\alpha_5/35 + 4\alpha_4/35) * Q_T$
61/185	1/1	61/185	Equation E-1
125/372	1/1	125/372	Equation E-3
123/370	1/1	123/370	Equation E-2

a. Shorthand criteria with the form x/y |z are maintenance events involving one component taken out of service due to maintenance.

b. This particular event equation is modified to include the independent failure explicitly in the fault tree. The α_1 term has been deleted from this equation to accommodate this modeling technique.

E-3. CCF PARAMETER DEVELOPMENT

This section provides detailed discussions of the parameters, tools, and treatments developed specifically for the RPS study. Specifically, it describes the development of a General Electric RPS-specific prior and ROD-specific prior, how CCF BE probabilities are calculated, application of the safety function knowledge, and special application of the Bayesian update process.

E-3.1 CCF Calculation Methodology

Three techniques are discussed in this section. These techniques are used to facilitate the estimation of plant-specific CCF probabilities from industry experience. One technique is the *impact vector* method, which is used to classify events according to the level of impact of common-cause events and the associated uncertainties in numerical terms. The second is impact vector specialization, in which impact vectors are modified to reflect the likelihood of the occurrence of the event in the specific system of interest. This technique is called *mapping*. The third technique is the estimation of alpha factors from the mapped impact vectors. Each technique is described briefly.

E-3.1.1 Impact Vector

An impact vector is a numerical representation of a CCF event. For a CCCG of size m, an impact vector has m+1 elements. The k+1 element, denoted by F_k, equals one if failure of exactly k components occurred, and zero otherwise. This applies to those situations where the component degradation values equal 1.0 and the time delay and coupling strength are 1.0. For those cases where these parameters are less than 1.0, the following techniques are used to develop an impact vector.

E-3.1.1.1 Impact Vector Equations. The values of the different elements (F_k) of the impact vector can be calculated based on the possible combinations of failures and non-failures. Equation E-4 shows, in general, how an element of the impact vector is calculated based on a degraded component state.

$$F_k^{(m)} = \sum_{l=0}^{\binom{m}{k}} \prod_{i=0}^{k} (p_i) \prod_{j=0}^{m-k} (1 - p_j) \qquad \text{E-4}$$

where:

m	=	the number of elements in the group
k	=	the number of failures out of the group of m
i	=	the failure elements of the $l^{\underline{th}}$ combination of k out of m failures
j	=	the non-failure elements of the $l^{\underline{th}}$ combination of k out of m failures
p	=	the weight or probability of the failure of each component (component degradation value)

Two additional parameters are coded with each CCF event: q represents the timing factor, and c represents the shared cause factor. The impact vector is then modified to reflect these parameters in the following manner:

$$
\begin{aligned}
I_{CCF} &= \left[cqF_0^{(m)}, cqF_1^{(m)}, \ldots, cqF_m^{(m)} \right] \\
I_{c_1} &= \left[(1-cq)(1-p_1), (1-cq)p_1, 0, \ldots, 0 \right] \\
&\vdots \\
I_{c_m} &= \left[(1-cq)(1-p_m), (1-cq)p_m, 0, \ldots, 0 \right]
\end{aligned}
\qquad \text{E-5}
$$

where:

c = shared cause factor

q = timing factor

Finally, the average impact vector is obtained by adding I_{CCF} and the I_c's, element by element.

E-3.1.1.2 Treatment of Uncertainty in Determining the Loss of Component Safety Function. During the review of the NPRDS and LER data for the RPS study there was some uncertainty about whether the safety function of the piece of equipment under scrutiny was compromised due to the failure mechanism. The uncertainty in this judgement is due to either: (1) unclear text in the event narrative, or (2) the component could be required to perform in different modes in the fault trees. For example, if a temperature detector fails high, it could either cause a spurious scram or contribute to preventing a scram depending on the parameter being measured.

To document the safety function impact, an additional field (FM2) was added to the database. When the analyst was uncertain about the status of the safety function, UKN (unknown) was entered in this field. Otherwise the field was coded FS for a fail-safe failure mode or NFS for a non-fail-safe failure mode.

This information was used in estimating component failure rates or Q_T's in Appendix C. The method is to calculate a ratio (NFS Ratio) of the failures identified as NFS to those that are identified as either FS or NFS. The NFS ratio was then applied by multiplying the count of UKN events by the NFS ratio and adding that to the NFS count.

The CCF data were treated in a similar manner. The method chosen to implement this treatment is to multiply each element of the average impact vector (for those CCF events designated as UKN) by the NFS ratio the same as the treatment of coupling strength and time delay. This effectively provides consistency between the CCF alpha parameter calculation and the Q_T calculation. A list of the component-specific ratios is given in Table E-2.

Table E-2. Component NFS ratios.

Component	FS Count	NFS Count	NFS Ratio
ACC	5	605	0.99
AOV	39	17	0.31
CBI	20	43	0.68
CPL	51	61	0.54
CPR	20	71	0.78
CPS	133	361	0.73
CRD	5	288	0.98
ROD	2	16	0.87
SDL	70	15	0.18
SOV	76	446	0.85
TLR	120	62	0.34

E-3.1.2 Mapping of Data

E-3.1.2.1 Exposed Population versus Component Group Size. There is a difference between the concepts of exposed population and the CCCG size. The exposed population is a data analysis concept, and CCCG size is a modeling concept. An example of the difference is provided in the context of the RPS study.

BWR plants contain from 0 to 24 bistables in the RPS. The actual number of bistables in a particular plant represents the exposed population and remains the same for a given plant. Table A-1 shows the exposed population counts used in this study. For a given scram scenario, one or more bistables are required to function in each channel. The CCCG size is the number of bistables required per channel times the number of channels. This varies as the number of modeled scram parameters changes, depending upon the channel design. Therefore, it is possible to have events with in-plant populations of up to 24 components, and the modeled events have a CCCG from two to the exposed population. In the case of a maintenance event, one channel's worth of components is removed from the CCCG.

An impact vector represents a CCF in a specific group of components of exposed population size m. A collection of impact vectors used to calculate the CCF BE probability for a particular component may contain impact vectors of many different exposed population sizes (e.g., events that occur in different plants or different systems). In this case, the impact vectors are mapped to the CCCG size of interest.

E-3.1.2.2 Mapping Techniques. An impact vector will be mapped up, mapped down, or unchanged depending upon the relationship between the original system and the target system CCCG. The process for determining the equations for mapping has been written into a program to allow mapping from any size system to any other size system. The equations that describe the mapping process are discussed below.

There are three general routines for mapping, depending on the relationship between the original impact vectors and the system of interest. Mapping down is performed when the impact vector exposed population size is larger than the target group size, and mapping up is performed when the impact vector exposed population size is smaller than the target group size. In the special case where the impact vector has been coded as a "lethal shock," the impact vector for the new system of m components contains a 1.0 in the F_m position. To illustrate the mapping process, mapping down and mapping up equations are presented for CCCGs of three and five in Equations E-6 and E-7.

Mapping Down (5 ⇒ 3)

$$\begin{aligned} F_1^{(3)} &= 3/5F_1^{(5)} + 3/5F_2^{(5)} + 3/10F_3^{(5)} \\ F_2^{(3)} &= 3/10F_2^{(5)} + 3/5F_3^{(5)} + 3/5F_4^{(5)} \\ F_3^{(3)} &= 1/10F_3^{(5)} + 2/5F_4^{(5)} + F_5^{(5)} \end{aligned} \quad \text{E-6}$$

Mapping Up (3 ⇒ 5)

$$\begin{aligned} F_1^{(5)} &= 5/3(1-\rho)^2 F_1^{(3)} \\ F_2^{(5)} &= 7/3\rho(1-\rho)^1 F_1^{(3)} + (1-\rho)^2 F_2^{(3)} \\ F_3^{(5)} &= \rho^2 F_1^{(3)} + 2\rho(1-\rho)^1 F_2^{(3)} + (1-\rho)^2 F_3^{(3)} \\ F_4^{(5)} &= \rho^2 F_2^{(3)} + 2\rho(1-\rho)^1 F_3^{(3)} \\ F_5^{(5)} &= \rho^2 F_3^{(3)} \end{aligned} \quad \text{E-7}$$

The parameter ρ in Equation E-7 is called the *mapping up* parameter. It is the probability that the non-lethal shock or cause would have failed a single component added to the system. One method of estimating ρ is given in Equation E-8.

$$\rho = \sum_{i=1}^{m} \frac{i}{m} f_i \qquad \text{E-8}$$

and:

m = the number of elements in the group (CCCG)

f_i = the i^{th} element of the generic impact vector

This method works well when the system sizes are close to one another (e.g., mapping from size 2 to size 3 or 4) or when at least one of the component degradation values is less than 1.0. When all of the component degradation values are equal to 1.0, ρ is also equal to 1.0. When used in the *mapping up* equations for the RPS data, this method tends to overestimate the probability that additional components added to a system will exhibit the same lethal shock-like behavior. Examination of trends in the unmapped RPS data shows that as the number of components in a system increases, the likelihood of lethal behavior in that group of components decreases rapidly. Based on these observed trends, a limit of 0.85 was established for ρ.

E-3.1.3 Estimation of CCF Alpha Factors

Once the impact vectors are calculated for the target group, the number of events in each impact category (n_k, Equation E-9), can be calculated by adding the corresponding elements of the impact vectors. That is, with n CCF events,

$$\mathrm{n_k} = \sum_{\mathrm{j=1}}^{\mathrm{n}} \mathrm{F_k(j)} \qquad \text{E-9}$$

where:

$F_k(i)$ = the k^{th} element of the impact vector for event i

The parameters of the alpha-factor model, Equation E-10, can be estimated using the following maximum likelihood estimators (MLE):

$$\alpha_k = \frac{n_k}{\sum_{k=1}^{m} n_k} \qquad \text{E-10}$$

E-3.2 Development of an RPS-Specific Prior

E-3.2.1 Background

The Bayesian approach utilizes the concept of a prior distribution. The prior reflects the analyst's degree of belief about the parameter before the evidence. This prior distribution is based on a generic data source, and updating the prior with a specific data set has the effect of specializing the prior to the specific application. The updated data set is known as the posterior distribution. The posterior represents the degree of belief about the parameter after incorporating the evidence.

E-3.2.2 RPS and ROD CCF Prior Event Population

For this study, priors were developed based on the common-cause data created during the course of the study. The resultant priors represent vendor and system generic data, which are updated with component specific evidence in the Bayesian update. Pooling of data from the RPS and ROD systems is not considered justifiable given the disparity in the number of components between the systems.

The problem with the disparity in the group sizes is that the required vector mapping to pool the data does not have any basis in an engineering sense. Component groups with significantly different sizes do not behave similarly in the sense of common-cause. Therefore, to eliminate the problem, we created two priors, RPS and ROD.

The General Electric RPS CCF events comprise a suitably large volume of data to use as the prior population for this study. The RPS prior data set contains 128 CCF events and the ROD prior data set contains 153 events.

E-3.2.3 Prior Results

The General Electric CCF data were repeatedly mapped to CCCGs of 2 to 16 (for the RPS Prior) and 185 and 370 (for the ROD Prior), and a data set representing a generic distribution for each CCCG was created. The results are shown in Table E-4 through Table E-7. Table E-4 and Table E-6 show the sums of each element (n_k) of the impact vectors for each CCCG, which are the results of the mapping. and Table E-7 show the prior maximum likelihood estimators (MLE) for each component CCCG. The MLE is represented by Equation E-11:

$$MLE_i = \frac{n_i}{\sum_{j=1}^{m} n_j} \qquad \text{E-11}$$

where:

m	=	CCCG
n_k	=	the sum of the k^{th} element of the impact vector, over all events
n_1	=	sum of the first element and the Adjusted Independent
Adjusted Independent	=	(Ind. Event Count * Mapped CCCG)/Average CCCG

The CCF prior distributions for RPS and ROD systems, derived from the complete set of General Electric RPS data, provide initial estimates for each $\alpha^{(m)}_k$ by mapping the data to each CCCG of interest, summing the impact vector elements for each CCF event, adding the number of independent events for the CCCG being considered to the $\alpha^{(m)}_1$ term, and normalizing across the alphas for the CCCG so that they add up to one. These estimates are taken to be the prior distribution mean values for each uncertainty distribution.

The RPS and ROD priors were examined to determine which events, if any, were influencing the results. In order to evaluate the importance of individual events, the sections of the prior that are used in representative calculations were examined. To evaluate the RPS prior, the mapped vectors of group size 8 were ranked based on the values of the vector elements 8 to 4 (this corresponds to the failure criterion, 4 out of 8). Out of 128 events in the RPS prior, 45 events contribute significantly to the values of the prior MLE elements. Table E-3 shows the breakdown of the significant events. It is interesting to note that the number of significant events increases as the redundancy (element number) decreases. No single cause or component dominates the results. Based on this analysis, a diffuse prior has been developed for the RPS system.

The ROD prior was evaluated similarly. The mapped vectors of group size 185 were ranked based on the values of the vector elements 185 to 61 (this corresponds to the failure criterion 61 out of 185). Out of 153 CCF events used to generate the ROD prior, one event contributes significantly to the values of the prior MLE elements. The event is N-XXX-84-1212-VO, a partial failure in 1984 of all of the scram pilot solenoid valves in the ROD system (original group size of 187). The event was due to a design error in the selection of the valve seat material. The other mapped vectors in the ROD prior are comprised of events with failures of 2 out of 370 to 49 out of 185. The reason these events are not important is that the failure criterion does not include these MLE elements.

The ROD prior MLEs show an oscillating behavior from MLE_1 to MLE_{50} (group size 185) and from MLE_1 to MLE_{100} (group size 370). This is due to insufficient events to fill in all the gaps of the summed vector. If more events were available (in suitable group size) with varying k-out-of-n failures, the MLEs would decrease smoothly similar to the RPS prior. However, this does not affect the current analysis since the elements of interest occur after the oscillating portion of the vectors.

Table E-3. RPS prior significant event summary.

MLE Element	Count of Significant Events
8	2
7	6
6	6
5	14
4	17
3	19
2	59

Table E-4. Sums of impact vector elements for General Electric RPS prior.

Group Size	Adjusted Independent	Prior Σn_k Vector
2	70.33	[2.94e+01, 6.29e+00]
3	105.50	[3.47e+01, 9.37e+00, 3.17e+00]
4	140.67	[3.82e+01, 1.21e+01, 4.41e+00, 2.06e+00]
5	175.83	[4.04e+01, 1.36e+01, 5.64e+00, 3.47e+00, 1.73e+00]
6	211.00	[4.18e+01, 1.54e+01, 6.59e+00, 3.71e+00, 2.85e+00, 1.46e+00]
7	246.16	[4.26e+01, 1.74e+01, 7.09e+00, 4.32e+00, 2.90e+00, 2.39e+00, 1.24e+00]
8	281.33	[4.28e+01, 1.96e+01, 7.47e+00, 4.74e+00, 3.21e+00, 2.46e+00, 2.03e+00, 1.05e+00]
9	316.50	[4.24e+01, 2.05e+01, 8.84e+00, 5.22e+00, 3.65e+00, 2.54e+00, 2.16e+00, 1.75e+00, 8.89e-01]
10	351.66	[4.21e+01, 2.10e+01, 1.02e+01, 5.64e+00, 4.07e+00, 2.85e+00, 2.13e+00, 1.94e+00, 1.52e+00, 7.55e-01]
11	386.83	[4.19e+01, 2.12e+01, 1.12e+01, 6.18e+00, 4.35e+00, 3.26e+00, 2.29e+00, 1.88e+00, 1.75e+00, 1.33e+00, 6.42e-01]
12	422.00	[4.17e+01, 2.14e+01, 1.19e+01, 6.79e+00, 4.57e+00, 3.57e+00, 2.61e+00, 1.90e+00, 1.71e+00, 1.59e+00, 1.17e+00, 5.45e-01]
13	457.16	[4.10e+01, 2.20e+01, 1.24e+01, 7.40e+00, 4.82e+00, 3.76e+00, 2.95e+00, 2.11e+00, 1.66e+00, 1.59e+00, 1.44e+00, 1.04e+00, 4.63e-01]
14	492.33	[4.04e+01, 2.24e+01, 1.28e+01, 7.93e+00, 5.14e+00, 3.88e+00, 3.20e+00, 2.42e+00, 1.75e+00, 1.50e+00, 1.48e+00, 1.32e+00, 9.21e-01, 3.94e-01]
15	527.49	[3.97e+01, 2.28e+01, 1.32e+01, 8.37e+00, 5.49e+00, 4.01e+00, 3.34e+00, 2.71e+00, 1.98e+00, 1.49e+00, 1.40e+00, 1.39e+00, 1.20e+00, 8.19e-01, 3.35e-01]
16	562.66	[3.91e+01, 2.31e+01, 1.34e+01, 8.72e+00, 5.85e+00, 4.16e+00, 3.41e+00, 2.91e+00, 2.27e+00, 1.63e+00, 1.33e+00, 1.33e+00, 1.30e+00, 1.10e+00, 7.29e-01, 2.84e-01]

Table E-5. Maximum likelihood estimators of α_k for General Electric RPS prior.

Group Size	MLE Vector
2	[9.41e-01, 5.93e-02]
3	[9.18e-01, 6.13e-02, 2.07e-02]
4	[9.06e-01, 6.14e-02, 2.24e-02, 1.05e-02]
5	[8.98e-01, 5.66e-02, 2.34e-02, 1.44e-02, 7.19e-03]
6	[8.94e-01, 5.44e-02, 2.33e-02, 1.31e-02, 1.01e-02, 5.16e-03]
7	[8.91e-01, 5.36e-02, 2.19e-02, 1.33e-02, 8.96e-03, 7.38e-03, 3.81e-03]
8	[8.89e-01, 5.37e-02, 2.05e-02, 1.30e-02, 8.79e-03, 6.73e-03, 5.58e-03, 2.87e-03]
9	[8.87e-01, 5.08e-02, 2.18e-02, 1.29e-02, 9.01e-03, 6.28e-03, 5.34e-03, 4.33e-03, 2.20e-03]
10	[8.87e-01, 4.73e-02, 2.29e-02, 1.27e-02, 9.17e-03, 6.43e-03, 4.81e-03, 4.36e-03, 3.43e-03, 1.70e-03]
11	[8.88e-01, 4.40e-02, 2.32e-02, 1.28e-02, 9.00e-03, 6.74e-03, 4.74e-03, 3.89e-03, 3.62e-03, 2.76e-03, 1.33e-03]
12	[8.89e-01, 4.10e-02, 2.29e-02, 1.30e-02, 8.76e-03, 6.84e-03, 5.01e-03, 3.65e-03, 3.28e-03, 3.05e-03, 2.25e-03, 1.05e-03]
13	[8.90e-01, 3.92e-02, 2.22e-02, 1.32e-02, 8.61e-03, 6.72e-03, 5.27e-03, 3.77e-03, 2.96e-03, 2.83e-03, 2.58e-03, 1.85e-03, 8.28e-04]
14	[8.91e-01, 3.75e-02, 2.14e-02, 1.33e-02, 8.59e-03, 6.50e-03, 5.35e-03, 4.05e-03, 2.92e-03, 2.51e-03, 2.48e-03, 2.20e-03, 1.54e-03, 6.59e-04]
15	[8.92e-01, 3.58e-02, 2.07e-02, 1.32e-02, 8.63e-03, 6.30e-03, 5.25e-03, 4.26e-03, 3.11e-03, 2.35e-03, 2.20e-03, 2.19e-03, 1.89e-03, 1.29e-03, 5.27e-04]
16	[8.94e-01, 3.43e-02, 1.99e-02, 1.29e-02, 8.69e-03, 6.18e-03, 5.06e-03, 4.33e-03, 3.37e-03, 2.42e-03, 1.98e-03, 1.97e-03, 1.94e-03, 1.63e-03, 1.08e-03, 4.23e-04]

Table E-6. Sums of impact vector elements for General Electric ROD prior.

Group Size	Adjusted Independent	Prior Σn_k Vector
185	834.16	[6.05e+01, 2.13e+01, 7.77e+00, 4.81e+00, 2.22e+00, 1.22e+00, 6.21e-01, 3.19e-01, 1.46e-01, 8.34e-02, 1.11e-01, 1.77e-01, 2.18e-01, 2.03e-01, 1.49e-01, 8.90e-02, 4.47e-02, 1.92e-02, 7.17e-03, 2.35e-03, 6.95e-04, 3.36e-04, 1.61e-03, 1.06e-02, 4.87e-02, 1.50e-01, 2.94e-01, 3.32e-01, 1.63e-01, 1.16e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, 2.10e-09, 3.01e-08, 2.82e-07, 1.95e-06, 1.05e-05, 4.66e-05, 1.73e-04, 5.49e-04, 1.52e-03, 3.70e-03, 8.01e-03, 1.55e-02, 2.72e-02, 4.31e-02, 6.21e-02, 8.20e-02, 9.91e-02, 1.10e-01, 1.13e-01, 1.07e-01, 9.39e-02, 7.63e-02, 5.75e-02, 4.04e-02, 2.65e-02, 1.65e-02, 9.97e-03, 6.33e-03, 4.76e-03, 4.64e-03, 5.51e-03, 7.15e-03, 9.43e-03, 1.23e-02, 1.58e-02, 1.99e-02, 2.44e-02, 2.93e-02, 3.45e-02, 3.98e-02, 4.48e-02, 4.94e-02, 5.33e-02, 5.62e-02, 5.81e-02, 5.87e-02, 5.81e-02, 5.63e-02, 5.33e-02, 4.94e-02, 4.49e-02, 3.98e-02, 3.46e-02, 2.94e-02, 2.45e-02, 1.99e-02, 1.59e-02, 1.24e-02, 9.42e-03, 7.02e-03, 5.12e-03, 3.65e-03, 2.55e-03, 1.74e-03, 1.16e-03, 7.55e-04, 4.81e-04, 3.00e-04, 1.82e-04, 1.09e-04, 6.31e-05, 3.59e-05, 1.99e-05, 1.08e-05, 5.70e-06, 2.95e-06, 1.49e-06, 7.31e-07, 3.51e-07, 1.64e-07, 7.51e-08, 3.35e-08, 1.45e-08, 6.15e-09, 2.54e-09, 1.02e-09, 3.99e-10, ...the rest of the vector elements are < 1.0e-10]
370	1668.31	[4.66e+01, 2.77e+01, 1.76e+01, 1.21e+01, 8.63e+00, 6.46e+00, 4.96e+00, 3.66e+00, 2.54e+00, 1.71e+00, 1.15e+00, 7.75e-01, 5.24e-01, 3.48e-01, 2.25e-01, 1.41e-01, 8.52e-02, 5.18e-02, 3.50e-02, 3.05e-02, 3.50e-02, 4.58e-02, 6.00e-02, 7.50e-02, 8.80e-02, 9.66e-02, 9.95e-02, 9.64e-02, 8.80e-02, 7.61e-02, 6.23e-02, 4.85e-02, 3.59e-02, 2.54e-02, 1.71e-02, 1.11e-02, 6.87e-03, 4.11e-03, 2.41e-03, 1.51e-03, 1.22e-03, 1.55e-03, 2.60e-03, 4.63e-03, 7.99e-03, 1.30e-02, 1.99e-02, 2.88e-02, 3.92e-02, 5.05e-02, 6.18e-02, 7.18e-02, 7.95e-02, 8.39e-02, 8.45e-02, 8.15e-02, 7.52e-02, 6.66e-02, 5.66e-02, 4.63e-02, 3.64e-02, 2.75e-02, 2.01e-02, 1.41e-02, 9.61e-03, 6.31e-03, 4.00e-03, 2.46e-03, 1.46e-03, 8.39e-04, 4.68e-04, 2.53e-04, 1.32e-04, 6.73e-05, 3.33e-05, 1.59e-05, 7.43e-06, 3.36e-06, 1.48e-06, 6.34e-07, 2.64e-07, 1.07e-07, 4.23e-08, 1.63e-08, 6.13e-09, 2.36e-09, 1.17e-09, 1.24e-09, 2.49e-09, 5.80e-09, 1.35e-08, 3.09e-08, 6.86e-08, 1.49e-07, 3.13e-07, 6.44e-07, 1.29e-06, 2.53e-06, 4.83e-06, 9.00e-06, 1.64e-05, 2.91e-05, 5.06e-05, 8.60e-05, 1.43e-04, 2.32e-04, 3.68e-04, 5.72e-04, 8.69e-04, 1.29e-03, 1.88e-03, 2.69e-03, 3.76e-03, 5.14e-03, 6.90e-03, 9.07e-03, 1.17e-02, 1.47e-02, 1.83e-02, 2.22e-02, 2.64e-02, 3.08e-02, 3.53e-02, 3.97e-02, 4.38e-02, 4.74e-02, 5.03e-02, 5.25e-02, 5.37e-02, 5.40e-02, 5.33e-02, 5.16e-02, 4.91e-02, 4.59e-02, 4.22e-02, 3.80e-02, 3.37e-02, 2.93e-02, 2.51e-02, 2.11e-02, 1.74e-02, 1.41e-02, 1.13e-02, 8.85e-03, 6.83e-03, 5.18e-03, 3.87e-03, 2.84e-03, 2.07e-03, 1.49e-03, 1.08e-03, 7.97e-04, 6.21e-04, 5.32e-04, 5.15e-04, 5.63e-04, 6.75e-04, 8.52e-04, 1.10e-03, 1.43e-03, 1.85e-03, 2.38e-03, 3.04e-03, 3.83e-03, 4.78e-03, 5.90e-03, 7.21e-03, 8.71e-03, 1.04e-02, 1.23e-02, 1.44e-02, 1.67e-02, 1.91e-02, 2.16e-02, 2.42e-02, 2.68e-02, 2.94e-02, 3.18e-02, 3.41e-02, 3.62e-02, 3.80e-02, 3.95e-02, 4.06e-02, 4.12e-02, 4.15e-02, 4.12e-02, 4.06e-02, 3.95e-02, 3.80e-02, 3.62e-02, 3.41e-02, 3.18e-02, 2.94e-02, 2.68e-02, 2.42e-02, 2.16e-02, 1.91e-02, 1.67e-02, 1.44e-02, 1.23e-02, 1.04e-02, 8.71e-03, 7.21e-03, 5.90e-03, 4.78e-03, 3.83e-03, 3.03e-03, 2.38e-03, 1.84e-03, 1.41e-03, 1.07e-03, 8.03e-04, 5.96e-04, 4.37e-04, 3.17e-04, 2.28e-04, 1.61e-04, 1.13e-04, 7.87e-05, 5.40e-05, 3.66e-05, 2.46e-05, 1.63e-05, 1.07e-05, 6.95e-06, 4.46e-06, 2.83e-06, 1.77e-06, 1.10e-06, 6.75e-07, 4.09e-07, 2.45e-07, 1.45e-07, 8.49e-08, 4.92e-08, 2.81e-08, 1.59e-08, 8.88e-09, 4.91e-09, 2.68e-09, 1.44e-09, 7.70e-10, 4.06e-10, 2.11e-10, ...the rest of the vector elements are < 1.0e-10]

Table E-7. Maximum likelihood estimators of α_k for General Electric ROD prior.

Group Size	MLE Vector
185	[9.55e-01, 2.27e-02, 8.30e-03, 5.14e-03, 2.37e-03, 1.30e-03, 6.62e-04, 3.41e-04, 1.56e-04, 8.90e-05, 1.18e-04, 1.89e-04, 2.33e-04, 2.16e-04, 1.59e-04, 9.50e-05, 4.77e-05, 2.05e-05, 7.66e-06, 2.51e-06, 7.42e-07, 3.59e-07, 1.72e-06, 1.13e-05, 5.20e-05, 1.60e-04, 3.14e-04, 3.54e-04, 1.74e-04, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, 3.01e-10, 2.08e-09, 1.12e-08, 4.97e-08, 1.84e-07, 5.86e-07, 1.62e-06, 3.95e-06, 8.55e-06, 1.66e-05, 2.90e-05, 4.59e-05, 6.63e-05, 8.75e-05, 1.06e-04, 1.18e-04, 1.21e-04, 1.14e-04, 1.00e-04, 8.14e-05, 6.14e-05, 4.31e-05, 2.83e-05, 1.76e-05, 1.06e-05, 6.75e-06, 5.08e-06, 4.95e-06, 5.88e-06, 7.63e-06, 1.01e-05, 1.32e-05, 1.69e-05, 2.12e-05, 2.60e-05, 3.13e-05, 3.68e-05, 4.24e-05, 4.78e-05, 5.27e-05, 5.68e-05, 6.00e-05, 6.20e-05, 6.27e-05, 6.20e-05, 6.00e-05, 5.69e-05, 5.28e-05, 4.79e-05, 4.25e-05, 3.69e-05, 3.14e-05, 2.61e-05, 2.13e-05, 1.69e-05, 1.32e-05, 1.00e-05, 7.49e-06, 5.46e-06, 3.90e-06, 2.72e-06, 1.85e-06, 1.24e-06, 8.06e-07, 5.13e-07, 3.20e-07, 1.95e-07, 1.16e-07, 6.74e-08, 3.83e-08, 2.12e-08, 1.15e-08, 6.09e-09, 3.14e-09, 1.59e-09, 7.80e-10, 3.75e-10, ...the rest of the vector elements are < 1.0e-10]
370	[9.49e-01, 1.53e-02, 9.74e-03, 6.71e-03, 4.77e-03, 3.58e-03, 2.75e-03, 2.02e-03, 1.41e-03, 9.47e-04, 6.35e-04, 4.29e-04, 2.90e-04, 1.93e-04, 1.25e-04, 7.78e-05, 4.71e-05, 2.87e-05, 1.94e-05, 1.69e-05, 1.94e-05, 2.53e-05, 3.32e-05, 4.15e-05, 4.87e-05, 5.34e-05, 5.50e-05, 5.33e-05, 4.87e-05, 4.21e-05, 3.45e-05, 2.68e-05, 1.99e-05, 1.40e-05, 9.48e-06, 6.13e-06, 3.80e-06, 2.27e-06, 1.34e-06, 8.34e-07, 6.77e-07, 8.55e-07, 1.44e-06, 2.56e-06, 4.42e-06, 7.20e-06, 1.10e-05, 1.59e-05, 2.17e-05, 2.80e-05, 3.42e-05, 3.97e-05, 4.40e-05, 4.64e-05, 4.68e-05, 4.51e-05, 4.16e-05, 3.69e-05, 3.13e-05, 2.56e-05, 2.01e-05, 1.52e-05, 1.11e-05, 7.83e-06, 5.32e-06, 3.49e-06, 2.21e-06, 1.36e-06, 8.07e-07, 4.64e-07, 2.59e-07, 1.40e-07, 7.33e-08, 3.73e-08, 1.84e-08, 8.82e-09, 4.11e-09, 1.86e-09, 8.19e-10, 3.51e-10, 1.46e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, 1.73e-10, 3.57e-10, 7.15e-10, 1.40e-09, 2.67e-09, 4.98e-09, 9.06e-09, 1.61e-08, 2.80e-08, 4.76e-08, 7.90e-08, 1.28e-07, 2.04e-07, 3.16e-07, 4.81e-07, 7.16e-07, 1.04e-06, 1.49e-06, 2.08e-06, 2.85e-06, 3.82e-06, 5.02e-06, 6.46e-06, 8.16e-06, 1.01e-05, 1.23e-05, 1.46e-05, 1.70e-05, 1.95e-05, 2.20e-05, 2.42e-05, 2.62e-05, 2.78e-05, 2.90e-05, 2.97e-05, 2.99e-05, 2.95e-05, 2.86e-05, 2.72e-05, 2.54e-05, 2.33e-05, 2.10e-05, 1.86e-05, 1.62e-05, 1.39e-05, 1.17e-05, 9.64e-06, 7.83e-06, 6.25e-06, 4.90e-06, 3.78e-06, 2.87e-06, 2.14e-06, 1.57e-06, 1.14e-06, 8.24e-07, 5.96e-07, 4.41e-07, 3.44e-07, 2.94e-07, 2.85e-07, 3.12e-07, 3.74e-07, 4.71e-07, 6.09e-07, 7.91e-07, 1.02e-06, 1.32e-06, 1.68e-06, 2.12e-06, 2.64e-06, 3.26e-06, 3.99e-06, 4.82e-06, 5.76e-06, 6.81e-06, 7.96e-06, 9.21e-06, 1.05e-05, 1.19e-05, 1.34e-05, 1.48e-05, 1.62e-05, 1.76e-05, 1.89e-05, 2.00e-05, 2.10e-05, 2.18e-05, 2.24e-05, 2.28e-05, 2.29e-05, 2.28e-05, 2.24e-05, 2.18e-05, 2.10e-05, 2.00e-05, 1.89e-05, 1.76e-05, 1.62e-05, 1.48e-05, 1.34e-05, 1.19e-05, 1.05e-05, 9.21e-06, 7.96e-06, 6.81e-06, 5.76e-06, 4.82e-06, 3.99e-06, 3.26e-06, 2.64e-06, 2.12e-06, 1.68e-06, 1.31e-06, 1.02e-06, 7.81e-07, 5.92e-07, 4.44e-07, 3.30e-07, 2.42e-07, 1.75e-07, 1.26e-07, 8.93e-08, 6.27e-08, 4.35e-08, 2.99e-08, 2.03e-08, 1.36e-08, 9.03e-09, 5.93e-09, 3.85e-09, 2.47e-09, 1.57e-09, 9.82e-10, 6.09e-10, 3.73e-10, 2.26e-10, ...the rest of the vector elements are < 1.0e-10]

E-3.3 Bayesian Update Process

This section presents specific methods taken to complete the Bayesian update calculation of CCF BEs in the GE RPS study.

E-3.3.1 Bayesian Update Methodology

In accordance with the methods explained in Section A-2.1.2.1, the distributions of the prior α_k are assumed to have a beta distribution form. When the prior α_k has a beta distribution for the probability of an occurrence, and occurrence data are generated from a binomial distribution with this probability, the posterior distribution from a Bayesian update is also a beta distribution. Thus, beta distributions are conjugate prior distributions for binomial data, and are a natural choice for the uncertainty in the CCF alpha parameters. The mean of the posterior uncertainty distribution (E-12) that results from updating a beta prior distribution with the observed data is a weighted average of the mean of the prior distribution and the maximum likelihood estimate from the data, as follows:

$$\alpha_{CCF} = \alpha_{\mathrm{Pr}ior} * \frac{\delta}{\delta + d} + \frac{f}{d} * \frac{d}{d + \delta} \qquad \text{E-12}$$

where:

α_{CCF} = posterior alpha

α_{Prior} = prior alpha

δ ≡ $\alpha + \beta$, parameters of the beta distribution of the prior

f = the sum of the i^{th} impact vector elements for the component, CCCG, and degree of CCF loss under consideration

d = the sum of all the impact vector elements for the CCCG and component under consideration

E-3.3.2 Uncertainty in the Prior Alpha Factors

To characterize the uncertainty in the common-cause alpha factors for the RPS, a distribution was associated with each alpha factor in the equation used to estimate each CCF probability (Table E-1). To complete the uncertainty analysis, distributions were needed for the alpha factors, $\alpha^{(m)}_k, \ldots \alpha^{(m)}_m$.

The particular beta distribution for each alpha parameter remains to be determined. With the means based on estimates from the data, just a single beta distribution parameter remains to be determined. The δ in Equation E-12 is a convenient choice. As δ increases, the variance of the uncertainty distribution decreases. Two basic approaches were used to estimate the prior distribution delta parameter, as discussed in subsections below.

E-3.3.2.1 Constrained Noninformative Distributions for CCF Factors. The first approach was to fit a constrained noninformative (CN) prior distribution for each $\alpha^{(m)}_i$, for i = 2, …, m. In this approach, the variance of the selected beta distribution maximizes the entropy, subject to the constraint that the mean matches the estimated probability of loss of i of m components by common-cause. In practice, knowledge of the constrained mean leads to an estimate of the alpha parameter of the desired beta distribution. When the fixed mean is very small (i.e., less than 0.001), the alpha parameter of the fitted CN distribution is approximately 0.50. The beta parameter is selected so that $\alpha/(\alpha + \beta)=\alpha/\delta$, which equals the mean. Further details of the method are found in the "Alternate Method" subsection of Section A-2.1.2.1. Figure E-3 shows the relationship between the fixed mean and the alpha parameter of the beta distribution about the mean.

Application of the CN method treats each $\alpha^{(m)}_k$ independently. It results in a generally different prior distribution delta for each CCF $\alpha^{(m)}_k$. As a result, the sum of the $\alpha^{(m)}_k$ from 1 to m does not equal 1.0. Since the sum of the CCF $\alpha^{(m)}_k$ from 1 to m must equal 1.0, the independent failure probability term, $\alpha^{(m)}_1$, would be obtained by subtraction if it were needed (note that it is not needed in any of the Table E-1 equations).

Also, since the prior δ parameters differ, the weighting between the prior distribution and the data for a particular component (Equation E-2) differs as the level of loss of redundancy (k in the subscript $\alpha^{(m)}_k$) changes across a CCCG. The results of the calculation of the prior δ are shown in Table E-8 through Table E-9. Rod prior, constrained noninformative δ and the geometric mean of δ (continued).

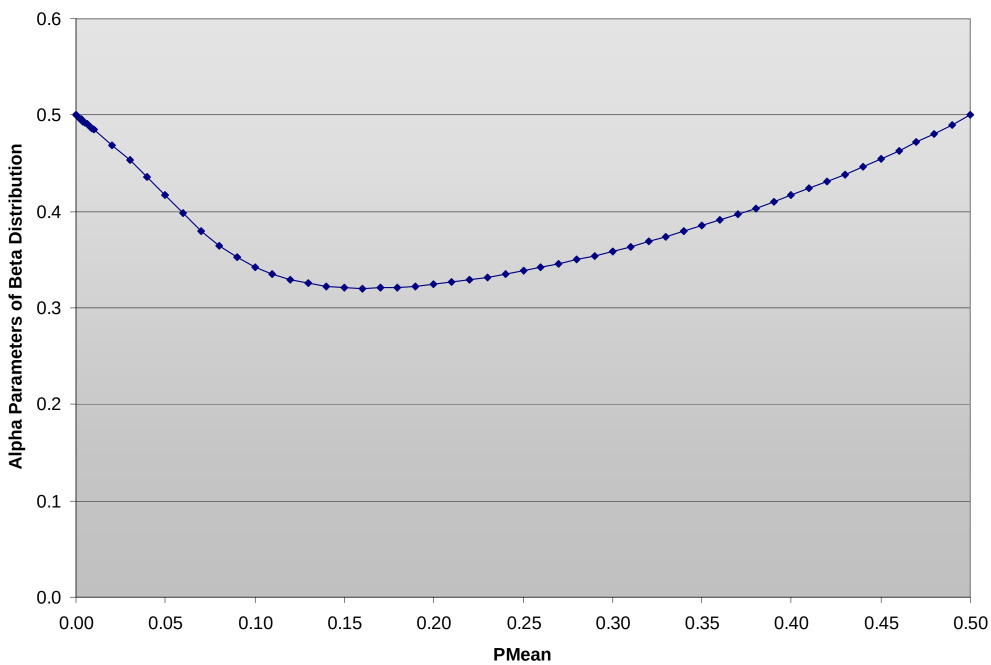


Figure E-3. Constrained non-informative prior alpha calculation.

Table E-8. RPS prior, constrained noninformative δ and the average of δ.

Group Size	Delta Vector	Average
2	[7.01e+00, 7.01e+00]	7.01
3	[4.32e+00, 6.24e+00, 2.19e+01]	10.82
4	[3.68e+00, 6.24e+00, 2.04e+01, 4.49e+01]	18.83
5	[3.31e+00, 7.26e+00, 1.96e+01, 3.30e+01, 6.79e+01]	26.21
6	[3.20e+00, 7.48e+00, 1.97e+01, 3.61e+01, 4.65e+01, 9.51e+01]	34.68
7	[3.13e+00, 7.55e+00, 2.08e+01, 3.56e+01, 5.45e+01, 6.62e+01, 1.30e+02]	45.37
8	[2.97e+00, 7.55e+00, 2.22e+01, 3.65e+01, 5.55e+01, 7.28e+01, 8.82e+01, 1.73e+02]	57.30
9	[2.94e+00, 7.87e+00, 2.09e+01, 3.67e+01, 5.38e+01, 7.80e+01, 9.21e+01, 1.14e+02, 2.25e+02]	70.18
10	[2.93e+00, 9.11e+00, 2.00e+01, 3.73e+01, 5.29e+01, 7.62e+01, 1.03e+02, 1.13e+02, 1.44e+02, 2.93e+02]	85.13
11	[2.95e+00, 9.65e+00, 1.98e+01, 3.70e+01, 5.39e+01, 7.28e+01, 1.04e+02, 1.27e+02, 1.37e+02, 1.80e+02, 3.74e+02]	101.68
12	[2.98e+00, 1.02e+01, 2.00e+01, 3.64e+01, 5.56e+01, 7.17e+01, 9.80e+01, 1.35e+02, 1.51e+02, 1.62e+02, 2.20e+02, 4.75e+02]	119.94
13	[2.99e+00, 1.15e+01, 2.06e+01, 3.59e+01, 5.66e+01, 7.30e+01, 9.32e+01, 1.31e+02, 1.68e+02, 1.75e+02, 1.92e+02, 2.69e+02, 6.04e+02]	141.02
14	[3.14e+00, 1.20e+01, 2.12e+01, 3.58e+01, 5.67e+01, 7.54e+01, 9.19e+01, 1.22e+02, 1.70e+02, 1.98e+02, 2.00e+02, 2.25e+02, 3.23e+02, 7.58e+02]	163.76
15	[3.16e+00, 1.25e+01, 2.20e+01, 3.60e+01, 5.64e+01, 7.77e+01, 9.35e+01, 1.16e+02, 1.59e+02, 2.11e+02, 2.26e+02, 2.27e+02, 2.64e+02, 3.86e+02, 9.48e+02]	189.14
16	[3.20e+00, 1.29e+01, 2.43e+01, 3.66e+01, 5.61e+01, 7.92e+01, 9.70e+01, 1.14e+02, 1.47e+02, 2.05e+02, 2.52e+02, 2.53e+02, 2.57e+02, 3.06e+02, 4.59e+02, 1.18e+03]	217.73

Appendix E

Table E-9. ROD prior, constrained noninformative δ and the geometric mean of δ.

Group Size	Delta Vector	Average
185	[9.42e+00, 2.01e+01, 5.87e+01, 9.56e+01, 2.10e+02, 3.83e+02, 7.54e+02, 1.47e+03, 3.20e+03, 5.61e+03, 4.22e+03, 2.63e+03, 2.14e+03, 2.31e+03, 3.15e+03, 5.25e+03, 1.04e+04, 2.43e+04, 6.51e+04, 1.98e+05, 6.72e+05, 1.00e+06, 2.90e+05, 4.41e+04, 9.60e+03, 3.12e+03, 1.59e+03, 1.41e+03, 2.86e+03, 1.00e+06, 8.50e+05, 3.07e+05, 1.26e+05, 5.83e+04, 3.01e+04, 1.72e+04, 1.09e+04, 7.52e+03, 5.70e+03, 4.71e+03, 4.24e+03, 4.13e+03, 4.36e+03, 4.98e+03, 6.13e+03, 8.12e+03, 1.16e+04, 1.76e+04, 2.84e+04, 4.69e+04, 7.38e+04, 9.81e+04, 1.01e+05, 8.47e+04, 6.53e+04, 4.95e+04, 3.79e+04, 2.95e+04, 2.35e+04, 1.91e+04, 1.59e+04, 1.35e+04, 1.18e+04, 1.04e+04, 9.46e+03, 8.77e+03, 8.31e+03, 8.04e+03, 7.95e+03, 8.04e+03, 8.30e+03, 8.76e+03, 9.45e+03, 1.04e+04, 1.17e+04, 1.35e+04, 1.59e+04, 1.91e+04, 2.35e+04, 2.94e+04, 3.78e+04, 4.96e+04, 6.65e+04, 9.13e+04, 1.28e+05, 1.83e+05, 2.69e+05, 4.03e+05, 6.19e+05, 9.71e+05, ...the rest of the vector elements are > 1.0e+06]	559799.31
370	[7.81e+00, 3.12e+01, 4.99e+01, 7.31e+01, 1.03e+02, 1.38e+02, 1.81e+02, 2.45e+02, 3.54e+02, 5.28e+02, 7.86e+02, 1.16e+03, 1.72e+03, 2.59e+03, 4.00e+03, 6.41e+03, 1.06e+04, 1.74e+04, 2.57e+04, 2.95e+04, 2.57e+04, 1.97e+04, 1.50e+04, 1.20e+04, 1.02e+04, 9.33e+03, 9.06e+03, 9.35e+03, 1.02e+04, 1.18e+04, 1.45e+04, 1.86e+04, 2.51e+04, 3.55e+04, 5.26e+04, 8.13e+04, 1.31e+05, 2.19e+05, 3.73e+05, 5.98e+05, 7.36e+05, 5.83e+05, 3.47e+05, 1.95e+05, 1.13e+05, 6.93e+04, 4.52e+04, 3.13e+04, 2.30e+04, 1.78e+04, 1.46e+04, 1.25e+04, 1.13e+04, 1.07e+04, 1.07e+04, 1.11e+04, 1.20e+04, 1.35e+04, 1.59e+04, 1.95e+04, 2.48e+04, 3.27e+04, 4.48e+04, 6.37e+04, 9.38e+04, 1.43e+05, 2.25e+05, 3.67e+05, 6.18e+05, 1.00e+06, 6.97e+05, 4.78e+05, 3.35e+05, 2.40e+05, 1.75e+05, 1.31e+05, 9.94e+04, 7.72e+04, 6.11e+04, 4.94e+04, 4.07e+04, 3.41e+04, 2.92e+04, 2.55e+04, 2.27e+04, 2.06e+04, 1.90e+04, 1.79e+04, 1.72e+04, 1.68e+04, 1.67e+04, 1.69e+04, 1.75e+04, 1.83e+04, 1.96e+04, 2.14e+04, 2.37e+04, 2.67e+04, 3.07e+04, 3.59e+04, 4.27e+04, 5.17e+04, 6.37e+04, 7.98e+04, 1.02e+05, 1.32e+05, 1.74e+05, 2.33e+05, 3.17e+05, 4.36e+05, 6.05e+05, 8.36e+05, 1.00e+06, 1.00e+06, 1.00e+06, 1.00e+06, 1.00e+06, 1.00e+06, 1.00e+06, 8.19e+05, 6.30e+05, 4.86e+05, 3.78e+05, 2.97e+05, 2.35e+05, 1.89e+05, 1.53e+05, 1.25e+05, 1.03e+05, 8.66e+04, 7.32e+04, 6.26e+04, 5.41e+04, 4.73e+04, 4.18e+04, 3.73e+04, 3.36e+04, 3.07e+04, 2.83e+04, 2.64e+04, 2.49e+04, 2.37e+04, 2.28e+04, 2.22e+04, 2.19e+04, 2.17e+04, 2.19e+04, 2.22e+04, 2.28e+04, 2.37e+04, 2.49e+04, 2.64e+04, 2.83e+04, 3.07e+04, 3.36e+04, 3.73e+04, 4.18e+04, 4.73e+04, 5.41e+04, 6.26e+04, 7.32e+04, 8.66e+04, 1.03e+05, 1.25e+05, 1.53e+05, 1.89e+05, 2.35e+05, 2.97e+05, 3.79e+05, 4.89e+05, 6.38e+05, 8.42e+05, ...the rest of the vector elements are > 1.0e+06]	610096.50

E-3.3.2.2 Dirichlet Distributions for CCF Factors. In the CCF analysis methodology, an underlying assumption is that, among the failure events, the number (k_1) of events with just one failure and no CCF loss, together with the number (k_2) of events with exactly two components lost by CCF, and the number (k_3) with exactly 3 components lost, and so forth, up to m components lost by CCF (k_m), form a joint multinomial probability distribution. Each event independently provides an increment for one of the k_i. The CCF $\alpha^{(m)}_k$ are the conditional probabilities that describe the likelihood for each level of component loss. The Dirichlet distribution is the multi nominal counterpart to a beta distribution function in which the parameters ($\alpha_1, \ldots, \alpha_m$) sum to one and represent the probability of exactly k failures out of m components in one event. Equation E-13 shows the Dirichlet distribution function:

$$\pi(\alpha_1, \ldots, \alpha_m) = \frac{\Gamma(A_1 + A_2 + \ldots, + A_m)}{\Gamma(A_1) \ldots \Gamma(A_m)} \alpha_1^{A_1 - 1} \alpha_2^{A_2 - 1} \ldots \alpha_m^{A_m - 1} \qquad \text{E-13}$$

The A_k's [$k = 1, \ldots, m$] are the parameters of the distribution and act like the count of events with k failures in the data.

When the set of alpha parameters { $\alpha^{(m)}_k$ }, for $k = 1, \ldots, m$ has a joint Dirichlet distribution, the marginal distributions are beta distributions with a common $\delta = (\alpha + \beta)$ parameter. That is, the mean of each common-cause α parameter is expressed as α_i/δ, for an appropriate alpha parameter α_i, and the corresponding beta parameter of the marginal beta distribution for each common-cause alpha is $\delta - \alpha_i$. Given the mean values and δ, the marginal beta distributions are fixed: $a_i = \delta$ * *the mean*, and $b_i = \delta - \alpha_i$. The Dirichlet uncertainty distribution depends on just the choice of the common δ, given the basic CCF alpha estimates.

When the Dirichlet prior distributions are updated with component-specific data, the posterior common-cause parameters will automatically sum to one. This is shown in Equation E-12, where both d from the data and δ from the prior distribution remain constant as the level of redundancy lost increases from 1 to m. In addition, with the Dirichlet distribution choice, the weighting between the prior and the data shown in Equation E-12 no longer depends on the level of redundancy of the alpha parameter. The treatment is thus more even-handed.

A reasonable choice for the δ is the geometric mean of the δ parameters computed in the CN distribution method. If the orders of magnitude between the estimated CCF alphas are not large, this average will result in uncertainty distributions that are not too skewed. Since the prior common-cause mean is α_i/δ, the beta distribution alpha parameter α_i is the mean times δ. From Figure E-3, low mean values lead to α_i parameters around 0.5. Since the chosen δ was calculated from the CN δ's, the resulting α_i parameters will center around 0.5, which is generally not too small. Small values for the alpha parameter of a beta distribution must be avoided, since they result in extremely skewed distributions.

E-3.3.3 Data

Data were selected from the RPS CCF database to match the criteria of each defined CCF BE used in the fault trees. Data for the component of interest included events in which the Safety Function is either NFS or UKN. The associated component independent failure count was extracted from the database and was selected using the same criteria as the CCF data.

E-3.4 CCF Basic Event Probability Results

E-3.4.1 Bayesian Update Results

Table E-10 shows the results of the CCF BE calculations with the Dirichlet prior for those components modeled in the fault trees. The Failure Criterion designation for each component points to an equation in Table E-1.

Table E-11 shows the lognormal uncertainty parameters for the CCF BEs. Error propagation using the beta distributions described in Section E-3.3.2 leads to uncertainty distributions on the estimated BE probabilities. The process, leading to lognormal distributions, is explained in Section A-2.2.

E-3.4.2 Classical Results

The classical or no prior influence results are shown in Table E-12. The results of the classical method show that, in general, the CCF results updated with a prior are higher. This method does not produce uncertainty distributions.

Table E-10. Bayesian update CCF basic event results.

Basic Event Name	Failure Criterion	QT Mean	CCF Basic Event Failure Probability	Alpha Vector	Event Description
GEL-ACC-CF-HCU	61/185	2.23E-05	1.09E-07	[The first 60 array elements are not used and not shown here... 4.59e-05, 6.63e-05, 8.75e-05, 1.06e-04, 1.18e-04, 1.21e-04, 1.14e-04, 1.00e-04, 8.14e-05, 6.14e-05, 4.31e-05, 2.83e-05, 1.76e-05, 1.06e-05, 6.75e-06, 5.08e-06, 4.95e-06, 5.88e-06, 7.63e-06, 1.01e-05, 1.32e-05, 1.69e-05, 2.12e-05, 2.60e-05, 3.13e-05, 3.68e-05, 4.24e-05, 4.78e-05, 5.27e-05, 5.68e-05, 6.00e-05, 6.20e-05, 6.27e-05, 6.20e-05, 6.00e-05, 5.69e-05, 5.28e-05, 4.79e-05, 4.25e-05, 3.69e-05, 3.14e-05, 2.61e-05, 2.13e-05, 1.69e-05, 1.32e-05, 1.00e-05, 7.49e-06, 5.46e-06, 3.90e-06, 2.72e-06, 1.85e-06, 1.24e-06, 8.05e-07, 5.13e-07, 3.20e-07, 1.95e-07, 1.16e-07, 6.74e-08, 3.83e-08, 2.12e-08, 1.15e-08, 6.09e-09, 3.14e-09, 1.59e-09, 7.80e-10, 3.75e-10, ...the rest of the vector elements are < 1.0e-10]	CCF 33% OR MORE HCU ACCUMULATORS FAIL
GEL-AOV-CF-HCU	123/370	2.87E-06	6.94E-09	[The first 122 array elements are not used and not shown here... 1.70e-05, 1.95e-05, 2.20e-05, 2.42e-05, 2.62e-05, 2.78e-05, 2.90e-05, 2.97e-05, 2.99e-05, 2.95e-05, 2.86e-05, 2.72e-05, 2.54e-05, 2.33e-05, 2.10e-05, 1.86e-05, 1.62e-05, 1.39e-05, 1.17e-05, 9.64e-06, 7.83e-06, 6.25e-06, 4.90e-06, 3.78e-06, 2.87e-06, 2.14e-06, 1.57e-06, 1.14e-06, 8.24e-07, 5.96e-07, 4.41e-07, 3.44e-07, 2.94e-07, 2.85e-07, 3.12e-07, 3.74e-07, 4.71e-07, 6.09e-07, 7.91e-07, 1.02e-06, 1.32e-06, 1.68e-06, 2.12e-06, 2.64e-06, 3.26e-06, 3.99e-06, 4.82e-06, 5.76e-06, 6.81e-06, 7.96e-06, 9.21e-06, 1.05e-05, 1.19e-05, 1.34e-05, 1.48e-05, 1.62e-05, 1.76e-05, 1.89e-05, 2.00e-05, 2.10e-05, 2.18e-05, 2.24e-05, 2.28e-05, 2.29e-05, 2.28e-05, 2.24e-05, 2.18e-05, 2.10e-05, 2.00e-05, 1.89e-05, 1.76e-05, 1.62e-05, 1.48e-05, 1.34e-05, 1.19e-05, 1.05e-05, 9.21e-06, 7.96e-06, 6.81e-06, 5.76e-06, 4.82e-06, 3.99e-06, 3.26e-06, 2.64e-06, 2.12e-06, 1.68e-06, 1.31e-06, 1.02e-06, 7.81e-07, 5.92e-07, 4.44e-07, 3.30e-07, 2.42e-07, 1.75e-07, 1.26e-07, 8.93e-08, 6.27e-08, 4.35e-08, 2.99e-08, 2.03e-08, 1.36e-08, 9.03e-09, 5.93e-09, 3.85e-09, 2.47e-09, 1.57e-09, 9.82e-10, 6.09e-10, 3.73e-10, 2.26e-10, ...the rest of the vector elements are < 1.0e-10]	CCF 33% OR MORE HCU SCRAM INLET/OUTLET AOVs FAIL TO OPEN
GEL-CBI-CF-TML3-7	3/7 |8	2.89E-04	4.15E-06	[9.15e-01, 4.34e-02, 1.63e-02, 9.53e-03, 6.00e-03, 4.48e-03, 3.70e-03, 1.91e-03]	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (LEVEL T&M)
GEL-CBI-CF-TMP3-7	3/7 |8	2.89E-04	4.15E-06	[9.15e-01, 4.34e-02, 1.63e-02, 9.53e-03, 6.00e-03, 4.48e-03, 3.70e-03, 1.91e-03]	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (PRES T&M)
GEL-CBI-CF-TU4-8	4/8	2.89E-04	3.07E-06	[9.15e-01, 4.34e-02, 1.63e-02, 9.53e-03, 6.00e-03, 4.48e-03, 3.70e-03, 1.91e-03]	CCF SPECIFIC 4 OR MORE TRIP UNITS
GEL-CPL-CF-L2-4	2/4	7.72E-04	7.10E-05	[8.80e-01, 9.89e-02, 1.52e-02, 5.81e-03]	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/ TRANSMITTERS
GEL-CPL-CF-TML2-3	2/3 |4	7.72E-04	1.22E-04	[8.80e-01, 9.89e-02, 1.52e-02, 5.81e-03]	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTER (LEVEL T&M)

Table E-10. (continued).

Basic Event Name	Failure Criterion	QT Mean	CCF Basic Event Failure Probability	Alpha Vector	Event Description
GEL-CPR-CF-P2-4	2/4	5.71E-05	4.86E-06	[9.07e-01, 3.90e-02, 1.42e-02, 4.03e-02]	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS
GEL-CPR-CF-TMP2-3	2/3 |4	5.71E-05	6.35E-06	[9.07e-01, 3.90e-02, 1.42e-02, 4.03e-02]	CCF SPECIFIC 2 OR MORE PRES SENSOR/TRANSMITTER (PRES T&M)
GEL-CPS-CF-HWL2-4	2/4	6.06E-04	2.84E-05	[9.47e-01, 3.53e-02, 1.59e-02, 2.06e-03]	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCHES
GEL-MSW-CF-MSSAB	2/2	1.30E-05	7.72E-07	[9.41e-01, 5.93e-02]	Common-Cause Failure of Both Manual Scram Switches
GEL-ROD-CF-CRD	61/185	5.13E-05	2.50E-07	[The first 60 array elements are not used and not shown here... 4.59e-05, 6.63e-05, 8.74e-05, 1.06e-04, 1.18e-04, 1.21e-04, 1.14e-04, 1.00e-04, 8.14e-05, 6.14e-05, 4.31e-05, 2.83e-05, 1.76e-05, 1.06e-05, 6.75e-06, 5.08e-06, 4.95e-06, 5.88e-06, 7.63e-06, 1.01e-05, 1.32e-05, 1.69e-05, 2.12e-05, 2.60e-05, 3.13e-05, 3.68e-05, 4.24e-05, 4.78e-05, 5.27e-05, 5.68e-05, 6.00e-05, 6.20e-05, 6.27e-05, 6.20e-05, 6.00e-05, 5.69e-05, 5.27e-05, 4.79e-05, 4.25e-05, 3.69e-05, 3.14e-05, 2.61e-05, 2.13e-05, 1.69e-05, 1.32e-05, 1.00e-05, 7.49e-06, 5.46e-06, 3.89e-06, 2.72e-06, 1.85e-06, 1.24e-06, 8.05e-07, 5.13e-07, 3.20e-07, 1.95e-07, 1.16e-07, 6.74e-08, 3.83e-08, 2.12e-08, 1.15e-08, 6.09e-09, 3.14e-09, 1.59e-09, 7.80e-10, 3.75e-10, ...the rest of the vector elements are < 1.0e-10]	CCF 33% OR MORE CRD/RODS FAIL TO INSERT
GEL-SDL-CF-2-4	2/4	6.13E-04	3.09E-05	[9.42e-01, 3.81e-02, 1.39e-02, 6.49e-03]	SCRAM DISCHARGE VOLUME LEVEL DETECTION 2 OF 4
GEL-SOV-CF-PSOVS	123/370	6.97E-04	1.69E-06	[The first 122 array elements are not used and not shown here... 1.70e-05, 1.95e-05, 2.20e-05, 2.42e-05, 2.62e-05, 2.78e-05, 2.90e-05, 2.97e-05, 2.99e-05, 2.95e-05, 2.86e-05, 2.72e-05, 2.54e-05, 2.33e-05, 2.10e-05, 1.86e-05, 1.62e-05, 1.39e-05, 1.17e-05, 9.64e-06, 7.83e-06, 6.24e-06, 4.90e-06, 3.78e-06, 2.86e-06, 2.14e-06, 1.57e-06, 1.14e-06, 8.24e-07, 5.96e-07, 4.41e-07, 3.44e-07, 2.95e-07, 2.86e-07, 3.13e-07, 3.76e-07, 4.74e-07, 6.13e-07, 7.96e-07, 1.03e-06, 1.33e-06, 1.69e-06, 2.13e-06, 2.66e-06, 3.28e-06, 4.01e-06, 4.84e-06, 5.79e-06, 6.84e-06, 8.00e-06, 9.25e-06, 1.06e-05, 1.20e-05, 1.34e-05, 1.49e-05, 1.63e-05, 1.77e-05, 1.89e-05, 2.01e-05, 2.11e-05, 2.19e-05, 2.25e-05, 2.29e-05, 2.30e-05, 2.29e-05, 2.25e-05, 2.19e-05, 2.11e-05, 2.01e-05, 1.89e-05, 1.76e-05, 1.63e-05, 1.48e-05, 1.34e-05, 1.20e-05, 1.06e-05, 9.23e-06, 7.98e-06, 6.82e-06, 5.77e-06, 4.83e-06, 3.99e-06, 3.27e-06, 2.65e-06, 2.12e-06, 1.68e-06, 1.32e-06, 1.02e-06, 7.82e-07, 5.93e-07, 4.45e-07, 3.30e-07, 2.42e-07, 1.76e-07, 1.26e-07, 8.94e-08, 6.28e-08, 4.36e-08, 2.99e-08, 2.03e-08, 1.36e-08, 9.04e-09, 5.93e-09, 3.85e-09, 2.47e-09, 1.57e-09, 9.82e-10, 6.09e-10, 3.73e-10, 2.26e-10, ...the rest of the vector elements are < 1.0e-10]	CCF 33% OR MORE HCU SCRAM PILOT SOVs FAIL
GEL-TLR-CF-2OF8	2/8	1.93E-05	2.35E-06	[8.76e-01, 5.65e-02, 2.72e-02, 1.77e-02, 9.70e-03, 5.88e-03, 4.62e-03, 2.38e-03]	CCF of 2 of 8 Trip Logic Relays for the Sensitivity of the Backup Scram Valves

Table E-10. (continued).

Basic Event Name	Failure Criterion	QT Mean	CCF Basic Event Failure Probability	Alpha Vector	Event Description
GEL-TLR-CF-CH4-8	4/8	1.93E-05	2.75E-07	[8.76e-01, 5.65e-02, 2.72e-02, 1.77e-02, 9.70e-03, 5.88e-03, 4.62e-03, 2.38e-03]	CCF SPECIFIC 4 OR MORE TRIP SYSTEM RELAYS
GEL-TLR-CF-CHABCD	6/12	1.93E-05	1.12E-07	[8.82e-01, 4.19e-02, 2.29e-02, 1.47e-02, 1.14e-02, 9.07e-03, 6.08e-03, 3.79e-03, 3.00e-03, 2.68e-03, 1.98e-03, 9.18e-04]	CCF SPECIFIC 6 OR MORE CHANNEL RELAYS
GEL-TLR-CF-K1-2-4	2/4	1.93E-05	1.36E-06	[9.15e-01, 6.08e-02, 1.66e-02, 7.70e-03]	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCH RELAYS
GEL-TLR-CF-TM-LV	5/11 |12	1.93E-05	1.34E-07	[8.82e-01, 4.19e-02, 2.29e-02, 1.47e-02, 1.14e-02, 9.07e-03, 6.08e-03, 3.79e-03, 3.00e-03, 2.68e-03, 1.98e-03, 9.18e-04]	CCF SPECIFIC 5 OR MORE CHANNEL RELAYS (LEVEL T&M)
GEL-TLR-CF-TM-PR	5/11 |12	1.93E-05	1.34E-07	[8.82e-01, 4.19e-02, 2.29e-02, 1.47e-02, 1.14e-02, 9.07e-03, 6.08e-03, 3.79e-03, 3.00e-03, 2.68e-03, 1.98e-03, 9.18e-04]	CCF SPECIFIC 5 OR MORE CHANNEL RELAYS (PRES T&M)
GEL-TLR-CF-TML3-7	3/7 |8	1.93E-05	3.93E-07	[8.76e-01, 5.65e-02, 2.72e-02, 1.77e-02, 9.70e-03, 5.88e-03, 4.62e-03, 2.38e-03]	CCF SPECIFIC 3 OR MORE TRIP SYSTEM RELAYS (LEVEL T&M)
GEL-TLR-CF-TMP3-7	3/7 |8	1.93E-05	3.93E-07	[8.76e-01, 5.65e-02, 2.72e-02, 1.77e-02, 9.70e-03, 5.88e-03, 4.62e-03, 2.38e-03]	CCF SPECIFIC 3 OR MORE TRIP SYSTEM RELAYS (PRES T&M)
GEL-TLR-CF-TRP4-8	K14Relays	1.93E-05	3.80E-07	[8.76e-01, 5.65e-02, 2.72e-02, 1.77e-02, 9.70e-03, 5.88e-03, 4.62e-03, 2.38e-03]	CCF SPECIFIC 4 OR MORE TRIP SYSTEM RELAYS

Table E-11. Lognormal uncertainty distributions for CCF events.

Basic Event Name	Median	EF	CCF Failure Rate Low [a]	CCF Failure Rate Mean [a]	CCF Failure Rate Upper [a]
GEL-ACC-CF-HCU	7.23E-08	4.41	1.64E-08	1.09E-07	3.19E-07
GEL-AOV-CF-HCU	6.94E-09	1.07	6.49E-09	6.94E-09	7.42E-09
GEL-CBI-CF-TML3-7	3.30E-06	3.06	1.08E-06	4.15E-06	1.01E-05
GEL-CBI-CF-TMP3-7	3.30E-06	3.06	1.08E-06	4.15E-06	1.01E-05
GEL-CBI-CF-TU4-8	2.25E-06	3.67	6.13E-07	3.07E-06	8.24E-06
GEL-CPL-CF-L2-4	2.25E-05	12.11	1.86E-06	7.10E-05	2.72E-04
GEL-CPL-CF-TML2-3	3.89E-05	12.01	3.24E-06	1.22E-04	4.67E-04
GEL-CPR-CF-P2-4	2.40E-06	7.05	3.41E-07	4.86E-06	1.69E-05
GEL-CPR-CF-TMP2-3	3.16E-06	6.99	4.51E-07	6.35E-06	2.21E-05
GEL-CPS-CF-HWL2-4	1.77E-05	4.93	3.60E-06	2.84E-05	8.74E-05
GEL-MSW-CF-MSSAB	2.62E-07	11.23	2.33E-08	7.71E-07	2.94E-06
GEL-ROD-CF-CRD	1.41E-07	5.81	2.43E-08	2.50E-07	8.19E-07
GEL-SDL-CF-2-4	1.36E-05	8.24	1.65E-06	3.09E-05	1.12E-04
GEL-SOV-CF-PSOVS	6.09E-07	10.47	5.82E-08	1.69E-06	6.38E-06
GEL-TLR-CF-2OF8	1.19E-06	6.77	1.76E-07	2.35E-06	8.09E-06
GEL-TLR-CF-CH4-8	1.12E-07	9.05	1.24E-08	2.75E-07	1.01E-06
GEL-TLR-CF-CHABCD	4.37E-08	9.55	4.58E-09	1.12E-07	4.18E-07
GEL-TLR-CF-K1-2-4	6.05E-07	8.09	7.48E-08	1.36E-06	4.89E-06
GEL-TLR-CF-TM-LV	5.52E-08	8.96	6.16E-09	1.34E-07	4.94E-07
GEL-TLR-CF-TM-PR	5.52E-08	8.96	6.16E-09	1.34E-07	4.94E-07
GEL-TLR-CF-TML3-7	1.75E-07	8.11	2.16E-08	3.93E-07	1.42E-06
GEL-TLR-CF-TMP3-7	1.75E-07	8.11	2.16E-08	3.93E-07	1.42E-06
GEL-TLR-CF-TRP4-8	1.64E-07	8.47	1.93E-08	3.80E-07	1.39E-06

a. Fifth percentile, mean, and 95th percentile of lognormal distribution found by propagating the means and variances of the Bayesian updated alpha terms from Table E-10 through the equations in Table E-1. The means and variances of the Q_T terms used in this calculation are the means and variances of the distributions listed in Table C-7.

Table E-12. Classical CCF basic event results.

Basic Event Name	Failure Criterion	QT Mean	CCF Basic Event Failure Probability	Alpha Vector	Event Description
GEL-ACC-CF-HCU	61/185	2.23E-05	<1.0e-10	[The first 60 array elements are not used and not shown here... ...the rest of the vector elements are < 1.0e-10]	CCF 33% OR MORE HCU ACCUMULATORS FAIL
GEL-AOV-CF-HCU	123/370	2.87E-06	<1.0e-10	[The first 122 array elements are not used and not shown here... ...the rest of the vector elements are < 1.0e-10]	CCF 33% OR MORE HCU SCRAM INLET/OUTLET AOVs FAIL TO OPEN
GEL-CBI-CF-TML3-7	3/7 |8	2.89E-04	3.05E-07	[9.66e-01, 2.32e-02, 8.01e-03, 2.68e-03, 4.72e-04, ...the rest of the vector elements are < 1.0e-10]	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (LEVEL T&M)
GEL-CBI-CF-TMP3-7	3/7 |8	2.89E-04	3.05E-07	[9.66e-01, 2.32e-02, 8.01e-03, 2.68e-03, 4.72e-04, ...the rest of the vector elements are < 1.0e-10]	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (PRES T&M)
GEL-CBI-CF-TU4-8	4/8	2.89E-04	8.10E-08	[9.66e-01, 2.32e-02, 8.01e-03, 2.68e-03, 4.72e-04, ...the rest of the vector elements are < 1.0e-10]	CCF SPECIFIC 4 OR MORE TRIP UNITS
GEL-CPL-CF-L2-4	2/4	7.72E-04	7.70E-05	[8.61e-01, 1.26e-01, 9.90e-03, 2.42e-03]	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS
GEL-CPL-CF-TML2-3	2/3 |4	7.72E-04	1.42E-04	[8.61e-01, 1.26e-01, 9.90e-03, 2.42e-03]	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTER (LEVEL T&M)
GEL-CPR-CF-P2-4	2/4	5.71E-05	5.25E-06	[9.08e-01, 3.57e-04, <1.0e-10, 9.18e-02]	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS
GEL-CPR-CF-TMP2-3	2/3 |4	5.71E-05	5.27E-06	[9.08e-01, 3.57e-04, <1.0e-10, 9.18e-02]	CCF SPECIFIC 2 OR MORE PRES SENSOR/TRANSMITTER (PRES T&M)
GEL-CPS-CF-HWL2-4	2/4	6.06E-04	2.33E-05	[9.57e-01, 2.90e-02, 1.44e-02, 1.25e-05]	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCHES
GEL-MSW-CF-MSSAB	2/2	1.30E-05	<1.0e-10	[...the rest of the vector elements are < 1.0e-10]	Common-Cause Failure of Both Manual Scram Switches
GEL-ROD-CF-CRD	61/185	5.13E-05	<1.0e-10	[The first 60 array elements are not used and not shown here... ...the rest of the vector elements are < 1.0e-10]	CCF 33% OR MORE CRD/RODS FAIL TO INSERT
GEL-SDL-CF-2-4	2/4	6.13E-04	<1.0e-10	[...the rest of the vector elements are < 1.0e-10]	SCRAM DISCHARGE VOLUME LEVEL DETECTION 2 OF 4

Table E-12. (continued).

Basic Event Name	Failure Criterion	QT Mean	CCF Basic Event Failure Probability	Alpha Vector	Event Description
GEL-SOV-CF-PSOVS	123/370	6.97E-04	2.34E-05	[The first 122 array elements are not used and not shown here... <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, <1.0e-10, 1.24e-10, 2.23e-10, 3.97e-10, 6.99e-10, 1.22e-09, 2.09e-09, 3.55e-09, 5.96e-09, 9.89e-09, 1.62e-08, 2.63e-08, 4.21e-08, 6.68e-08, 1.05e-07, 1.62e-07, 2.48e-07, 3.75e-07, 5.62e-07, 8.31e-07, 1.22e-06, 1.76e-06, 2.52e-06, 3.56e-06, 4.98e-06, 6.88e-06, 9.41e-06, 1.27e-05, 1.70e-05, 2.25e-05, 2.94e-05, 3.81e-05, 4.87e-05, 6.17e-05, 7.72e-05, 9.55e-05, 1.17e-04, 1.42e-04, 1.70e-04, 2.01e-04, 2.35e-04, 2.73e-04, 3.13e-04, 3.54e-04, 3.97e-04, 4.41e-04, 4.84e-04, 5.25e-04, 5.64e-04, 5.99e-04, 6.29e-04, 6.53e-04, 6.72e-04, 6.83e-04, 6.87e-04, 6.83e-04, 6.72e-04, 6.55e-04, 6.30e-04, 6.00e-04, 5.66e-04, 5.27e-04, 4.86e-04, 4.43e-04, 4.00e-04, 3.57e-04, 3.15e-04, 2.75e-04, 2.38e-04, 2.03e-04, 1.71e-04, 1.43e-04, 1.18e-04, 9.69e-05, 7.83e-05, 6.27e-05, 4.96e-05, 3.88e-05, 3.00e-05, 2.30e-05, 1.74e-05, 1.30e-05, 9.64e-06, 7.06e-06, 5.11e-06, 3.66e-06, 2.59e-06, 1.82e-06, 1.26e-06, 8.61e-07, 5.83e-07, 3.90e-07, 2.58e-07, 1.69e-07, 1.10e-07, 7.01e-08, 4.43e-08, 2.77e-08, 1.71e-08, 1.05e-08, 6.33e-09, 3.78e-09, 2.23e-09, 1.30e-09, 7.52e-10, 4.29e-10, 2.42e-10, ...the rest of the vector elements are < 1.0e-10]	CCF 33% OR MORE HCU SCRAM PILOT SOVs FAIL
GEL-TLR-CF-2OF8	2/8	1.93E-05	3.76E-06	[8.15e-01, 6.98e-02, 5.92e-02, 4.04e-02, 1.41e-02, 1.79e-03, 3.66e-08, 4.93e-10]	CCF of 2 of 8 Trip Logic Relays for the Sensitivity of the Backup Scram Valves
GEL-TLR-CF-CH4-8	4/8	1.93E-05	1.26E-07	[8.15e-01, 6.98e-02, 5.92e-02, 4.04e-02, 1.41e-02, 1.79e-03, 3.66e-08, 4.93e-10]	CCF SPECIFIC 4 OR MORE TRIP SYSTEM RELAYS
GEL-TLR-CF-CHABCD	6/12	1.93E-05	2.27E-08	[8.26e-01, 4.83e-02, 2.32e-02, 2.67e-02, 3.07e-02, 2.52e-02, 1.38e-02, 4.80e-03, 9.52e-04, ...the rest of the vector elements are < 1.0e-10]	CCF SPECIFIC 6 OR MORE CHANNEL RELAYS
GEL-TLR-CF-K1-2-4	2/4	1.93E-05	7.72E-07	[9.40e-01, 5.91e-02, 4.17e-04, 1.02e-05]	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCH RELAYS
GEL-TLR-CF-TM-LV	5/11 |12	1.93E-05	4.69E-08	[8.26e-01, 4.83e-02, 2.32e-02, 2.67e-02, 3.07e-02, 2.52e-02, 1.38e-02, 4.80e-03, 9.52e-04, ...the rest of the vector elements are < 1.0e-10]	CCF SPECIFIC 5 OR MORE CHANNEL RELAYS (LEVEL T&M)
GEL-TLR-CF-TM-PR	5/11 |12	1.93E-05	4.69E-08	[8.26e-01, 4.83e-02, 2.32e-02, 2.67e-02, 3.07e-02, 2.52e-02, 1.38e-02, 4.80e-03, 9.52e-04, ...the rest of the vector elements are < 1.0e-10]	CCF SPECIFIC 5 OR MORE CHANNEL RELAYS (PRES T&M)

Table E-12. (continued).

Basic Event Name	Failure Criterion	QT Mean	CCF Basic Event Failure Probability	Alpha Vector	Event Description
GEL-TLR-CF-TML3-7	3/7 |8	1.93E-05	3.23E-07	[8.15e-01, 6.98e-02, 5.92e-02, 4.04e-02, 1.41e-02, 1.79e-03, 3.66e-08, 4.93e-10]	CCF SPECIFIC 3 OR MORE TRIP SYSTEM RELAYS (LEVEL T&M)
GEL-TLR-CF-TMP3-7	3/7 |8	1.93E-05	3.23E-07	[8.15e-01, 6.98e-02, 5.92e-02, 4.04e-02, 1.41e-02, 1.79e-03, 3.66e-08, 4.93e-10]	CCF SPECIFIC 3 OR MORE TRIP SYSTEM RELAYS (PRES T&M)
GEL-TLR-CF-TRP4-8	K14Relays	1.93E-05	2.46E-07	[8.15e-01, 6.98e-02, 5.92e-02, 4.04e-02, 1.41e-02, 1.79e-03, 3.66e-08, 4.93e-10]	CCF SPECIFIC 4 OR MORE TRIP SYSTEM RELAYS

E-4. REFERENCES

E-1. F. M. Marshall et. al., *Common-Cause Failure Database and Analysis System: Event Definition and Classification*, NUREG/CR-6268, Vol. 2, June 1998.

Appendix F

Fault Tree Quantification Results

Appendix F

Fault Tree Quantification Results

This appendix contains the SAPHIRE cut sets, importance rankings, and basic event reports from the quantification of the General Electric RPS fault tree. Two separate cases of results are presented in this appendix. The first case of results presented assumes that the basic event value for the operator failing to initiate a scram (GEL-XHE-HE-SCRAM) is TRUE (i.e., failure probability is 1.0). Tables F-1, F-2, F-3, and F-4 contain the cut sets, importance measures sorted by Fussell-Vesely, Risk Increase Ratio, and Birnbaum, respectively, for this case. The RPS fault tree cut sets were generated with no truncation level specified. Table F-5 provides a listing of the basic events used in the GE RPS fault tree along with their respective failure probability, uncertainty data, and description.

The second case of results presented assumes that the basic event value for the operator failing to initiate a scram (GEL-XHE-HE-SCRAM) is 0.01. Tables F-6, F-7, F-8, and F-9 contain the cut sets, importance measures sorted by Fussell-Vesely, Risk Increase Ratio, and Birnbaum, respectively, for this case. The RPS fault tree cut sets were generated with no truncation level specified. Table F-10 provides a listing of the basic events that are affected by the assumption that the basic event value for the operator failing to initiate a scram is 0.01.

The cut sets that are shown in Tables F-1 and F-6 contain some basic events with a "/" in front of them. A "/" as the first character in a basic event name indicates a complemented event (Success = 1 - Failure). For example, the basic event for reactor low water level trip signal channel A in test and maintenance (T&M) is GEL-RPS-TM-ALVL (Failure = 1.40E-03). Thus, the basic event name for reactor low water level trip signal channel A not in T&M is /GEL-RPS-TM-ALVL (Success = 9.986E-01). The event description for complemented events remains the same as the description used for the failure event.

Table F-1. Top 100 cut sets (operator fails to initiate scram = TRUE) RPS mincut = 5.8E-06.

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
1	52.5	3.1E-06	GEL-CBI-CF-TU4-8	CCF SPECIFIC 4 OR MORE TRIP UNITS	3.1E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
2	29.0	1.7E-06	GEL-SOV-CF-PSOVS	CCF 33% OR MORE HCU SCRAM PILOT SOVS OR BACKUP SOVS FAIL	1.7E-6
3	6.5	3.8E-07	GEL-TLR-CF-TRP4-8	CCF SPECIFIC 4 OR MORE TRIP SYSTEM RELAYS	3.8E-7
4	4.7	2.7E-07	/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-CF-CHACBD	CCF CHANNEL RELAYS (NO T&M)	2.7E-7
5	4.3	2.5E-07	GEL-ROD-CF-CRD	CCF 33% OR MORE CRD/RODS FAIL TO INSERT	2.5E-7
6	1.9	1.1E-07	GEL-ACC-CF-HCU	CCF 33% OR MORE HCU ACCUMULATORS FAIL	1.1E-7
7	0.4	2.1E-08	GEL-SDL-CF-HWL2-4	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCHES	3.1E-5
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
8	0.4	2.1E-08	GEL-SDL-CF-HWL2-4	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCHES	3.1E-5
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
9	0.1	6.9E-09	GEL-AOV-CF-HCU	CCF 33% OR MORE HCU SCRAM INLET/OUTLET AOVS FAIL TO OPEN	6.9E-9
10	0.1	5.8E-09	GEL-CBI-CF-TML3-7	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (LEVEL T&M)	4.2E-6
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
11	0.1	5.8E-09	GEL-CBI-CF-TMP3-7	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (PRES T&M)	4.2E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
12	0.0	9.1E-10	GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-CF-K1-2-4	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCH RELAYS	1.4E-6
13	0.0	9.1E-10	GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-CF-K1-2-4	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCH RELAYS	1.4E-6
14	0.0	5.5E-10	GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-CF-TM-LV	CCF CHANNEL RELAYS (LEVEL T&M)	3.9E-7
15	0.0	5.5E-10	/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
			GEL-TLR-CF-TM-PR	CCF CHANNEL RELAYS (PRES T&M)	3.9E-7
16	0.0	3.4E-10	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
17	0.0	2.5E-10	GEL-SDL-FC-LAMA	LEVEL SWITCH A (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDL-FC-LCMB	LEVEL SWITCH C (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
18	0.0	2.5E-10	GEL-SDL-FC-LAMA	LEVEL SWITCH A (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDL-FC-LCMB	LEVEL SWITCH C (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
19	0.0	2.5E-10	GEL-SDL-FC-LBMA	LEVEL SWITCH B (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDL-FC-LDMB	LEVEL SWITCH D (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
20	0.0	2.5E-10	GEL-SDL-FC-LBMA	LEVEL SWITCH B (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDL-FC-LDMB	LEVEL SWITCH D (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
21	0.0	2.9E-11	GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
22	0.0	7.9E-12	GEL-SDL-FC-LAMA	LEVEL SWITCH A (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1C	RELAY K1C FAILS	1.9E-5
23	0.0	7.9E-12	GEL-SDL-FC-LAMA	LEVEL SWITCH A (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1C	RELAY K1C FAILS	1.9E-5
24	0.0	7.9E-12	GEL-SDL-FC-LBMA	LEVEL SWITCH B (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1D	RELAY K1D FAILS	1.9E-5
25	0.0	7.9E-12	GEL-SDL-FC-LBMA	LEVEL SWITCH B (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1D	RELAY K1D FAILS	1.9E-5

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
26	0.0	7.9E-12	GEL-SDL-FC-LCMB	LEVEL SWITCH C (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1A	RELAY K1A FAILS	1.9E-5
27	0.0	7.9E-12	GEL-SDL-FC-LCMB	LEVEL SWITCH C (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1A	RELAY K1A FAILS	1.9E-5
28	0.0	7.9E-12	GEL-SDL-FC-LDMB	LEVEL SWITCH D (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1B	RELAY K1B FAILS	1.9E-5
29	0.0	7.9E-12	GEL-SDL-FC-LDMB	LEVEL SWITCH D (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1B	RELAY K1B FAILS	1.9E-5
30	0.0	5.9E-12	GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
31	0.0	5.9E-12	GEL-CBI-FF-PCHB	CH-B PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHD	CH-D PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
32	0.0	5.7E-12	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHC	CH-C PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
33	0.0	5.3E-12	GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
34	0.0	2.9E-12	GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
35	0.0	2.9E-12	GEL-CPL-FF-LCHB	CH-B WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPL-FF-LCHD	CH-D WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
36	0.0	2.0E-12	GEL-CBI-FF-LCHC	CH-C WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
37	0.0	1.9E-12	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
			GEL-TLR-FF-K5C	CH-C PRESSURE RELAY K5C FAILS	1.9E-5
38	0.0	1.2E-12	GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHC	CH-C PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
39	0.0	1.2E-12	GEL-CBI-FF-PCHB	CH-B PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHD	CH-D PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
40	0.0	1.2E-12	GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHA	CH-A PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
41	0.0	1.2E-12	GEL-CBI-FF-PCHD	CH-D PRESSURE TRIP UNIT FAILS	2.9E-4

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHB	CH-B PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
42	0.0	1.1E-12	GEL-CBI-FF-LCHA	CH-A WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
43	0.0	1.1E-12	GEL-CBI-FF-LCHB	CH-B WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHD	CH-D WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
44	0.0	1.1E-12	GEL-CBI-FF-LCHC	CH-C WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
45	0.0	1.1E-12	GEL-CBI-FF-LCHD	CH-D WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHB	CH-B WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
46	0.0	8.3E-13	GEL-CPL-CF-TML2-3	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTER (LEVEL T&M)	1.2E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
47	0.0	6.3E-13	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-CF-TMP2-3	CCF SPECIFIC 2 OR MORE PRES SENSOR/TRANSMITTER (PRES T&M)	6.4E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
48	0.0	4.1E-13	GEL-CBI-FF-LCHA	CH-A WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-LCHC	CH-C WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
49	0.0	4.1E-13	GEL-CBI-FF-LCHB	CH-B WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-LCHD	CH-D WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
50	0.0	4.0E-13	GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5C	CH-C PRESSURE RELAY K5C FAILS	1.9E-5
51	0.0	4.0E-13	GEL-CBI-FF-PCHB	CH-B PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5D	CH-D PRESSURE RELAY K5D FAILS	1.9E-5
52	0.0	4.0E-13	GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5A	CH-A PRESSURE RELAY K5A FAILS	1.9E-5
53	0.0	4.0E-13	GEL-CBI-FF-PCHD	CH-D PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5B	CH-B PRESSURE RELAY K5B FAILS	1.9E-5
54	0.0	2.6E-13	GEL-SOV-FF-SOVA	BACKUP SCRAM SOV A FAILS TO ACTUATE	7.0E-4
			GEL-TLR-FF-K14E	TRIP SYSTEM A RELAY K14E FAILS	1.9E-5

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
			GEL-TLR-FF-K14G	TRIP SYSTEM A RELAY K14G FAILS	1.9E-5
55	0.0	2.6E-13	GEL-SOV-FF-SOVA	BACKUP SCRAM SOV A FAILS TO ACTUATE	7.0E-4
			GEL-TLR-FF-K14F	TRIP SYSTEM B RELAY K14F FAILS	1.9E-5
			GEL-TLR-FF-K14H	TRIP SYSTEM B RELAY K14H FAILS	1.9E-5
56	0.0	2.6E-13	GEL-SOV-FF-SOVB	BACKUP SCRAM SOV B FAILS TO ACTUATE	7.0E-4
			GEL-TLR-FF-K14A	TRIP SYSTEM A RELAY K14A FAILS	1.9E-5
			GEL-TLR-FF-K14C	TRIP SYSTEM A RELAY K14C FAILS	1.9E-5
57	0.0	2.6E-13	GEL-SOV-FF-SOVB	BACKUP SCRAM SOV B FAILS TO ACTUATE	7.0E-4
			GEL-TLR-FF-K14B	TRIP SYSTEM B RELAY K14B FAILS	1.9E-5
			GEL-TLR-FF-K14D	TRIP SYSTEM B RELAY K14D FAILS	1.9E-5
58	0.0	2.5E-13	GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1A	RELAY K1A FAILS	1.9E-5
			GEL-TLR-FF-K1C	RELAY K1C FAILS	1.9E-5
59	0.0	2.5E-13	GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1B	RELAY K1B FAILS	1.9E-5
			GEL-TLR-FF-K1D	RELAY K1D FAILS	1.9E-5
60	0.0	2.5E-13	GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1A	RELAY K1A FAILS	1.9E-5
			GEL-TLR-FF-K1C	RELAY K1C FAILS	1.9E-5
61	0.0	2.5E-13	GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1B	RELAY K1B FAILS	1.9E-5
			GEL-TLR-FF-K1D	RELAY K1D FAILS	1.9E-5
62	0.0	2.4E-13	GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
63	0.0	2.3E-13	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHA	CH-A PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			GEL-CPR-FF-PCHC	CH-C PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
64	0.0	2.3E-13	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHB	CH-B PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			GEL-CPR-FF-PCHD	CH-D PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
65	0.0	1.3E-13	GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6C	CH-C WATER LEVEL RELAY K6C FAILS	1.9E-5
66	0.0	9.0E-14	GEL-CBI-FF-LCHA	CH-A WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
67	0.0	9.0E-14	GEL-CBI-FF-LCHC	CH-C WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
68	0.0	9.0E-14	GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
69	0.0	7.8E-14	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHA	CH-A PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5C	CH-C PRESSURE RELAY K5C FAILS	1.9E-5
70	0.0	7.8E-14	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHB	CH-B PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5D	CH-D PRESSURE RELAY K5D FAILS	1.9E-5
71	0.0	7.8E-14	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHC	CH-C PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5A	CH-A PRESSURE RELAY K5A FAILS	1.9E-5
72	0.0	7.8E-14	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHD	CH-D PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5B	CH-B PRESSURE RELAY K5B FAILS	1.9E-5
73	0.0	7.2E-14	GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6C	CH-C WATER LEVEL RELAY K6C FAILS	1.9E-5
74	0.0	7.2E-14	GEL-CPL-FF-LCHB	CH-B WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6D	CH-D WATER LEVEL RELAY K6D FAILS	1.9E-5
75	0.0	7.2E-14	GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6A	CH-A WATER LEVEL RELAY K6A FAILS	1.9E-5
76	0.0	7.2E-14	GEL-CPL-FF-LCHD	CH-D WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6B	CH-B WATER LEVEL RELAY K6B FAILS	1.9E-5
77	0.0	5.0E-14	GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
78	0.0	5.0E-14	GEL-CBI-FF-PCHB	CH-B PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHD	CH-D PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHB	CH-B WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPL-FF-LCHD	CH-D WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
79	0.0	4.8E-14	GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-FF-PCHC	CH-C PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
80	0.0	3.4E-14	GEL-CBI-FF-LCHA	CH-A WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-LCHC	CH-C WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
81	0.0	3.4E-14	GEL-CBI-FF-LCHC	CH-C WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
82	0.0	2.7E-14	GEL-CBI-FF-LCHA	CH-A WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6C	CH-C WATER LEVEL RELAY K6C FAILS	1.9E-5

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
83	0.0	2.7E-14	GEL-CBI-FF-LCHB	CH-B WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6D	CH-D WATER LEVEL RELAY K6D FAILS	1.9E-5
84	0.0	2.7E-14	GEL-CBI-FF-LCHC	CH-C WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6A	CH-A WATER LEVEL RELAY K6A FAILS	1.9E-5
85	0.0	2.7E-14	GEL-CBI-FF-LCHD	CH-D WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6B	CH-B WATER LEVEL RELAY K6B FAILS	1.9E-5
86	0.0	2.6E-14	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5A	CH-A PRESSURE RELAY K5A FAILS	1.9E-5
			GEL-TLR-FF-K5C	CH-C PRESSURE RELAY K5C FAILS	1.9E-5
87	0.0	2.6E-14	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5B	CH-B PRESSURE RELAY K5B FAILS	1.9E-5
			GEL-TLR-FF-K5D	CH-D PRESSURE RELAY K5D FAILS	1.9E-5
88	0.0	2.2E-14	GEL-PWR-FF-SOVA	125 VDC POWER FAILS (SOV A)	6.0E-5
			GEL-TLR-FF-K14E	TRIP SYSTEM A RELAY K14E FAILS	1.9E-5
			GEL-TLR-FF-K14G	TRIP SYSTEM A RELAY K14G FAILS	1.9E-5
89	0.0	2.2E-14	GEL-PWR-FF-SOVA	125 VDC POWER FAILS (SOV A)	6.0E-5
			GEL-TLR-FF-K14F	TRIP SYSTEM B RELAY K14F FAILS	1.9E-5
			GEL-TLR-FF-K14H	TRIP SYSTEM B RELAY K14H FAILS	1.9E-5
90	0.0	2.2E-14	GEL-PWR-FF-SOVB	125 VDC POWER FAILS (SOV B)	6.0E-5
			GEL-TLR-FF-K14A	TRIP SYSTEM A RELAY K14A FAILS	1.9E-5
			GEL-TLR-FF-K14C	TRIP SYSTEM A RELAY K14C FAILS	1.9E-5
91	0.0	2.2E-14	GEL-PWR-FF-SOVB	125 VDC POWER FAILS (SOV B)	6.0E-5
			GEL-TLR-FF-K14B	TRIP SYSTEM B RELAY K14B FAILS	1.9E-5
			GEL-TLR-FF-K14D	TRIP SYSTEM B RELAY K14D FAILS	1.9E-5
92	0.0	1.9E-14	GEL-CBI-FF-LCHA	CH-A WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
93	0.0	1.9E-14	GEL-CBI-FF-LCHB	CH-B WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHB	CH-B PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHD	CH-D PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHD	CH-D WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
94	0.0	1.9E-14	GEL-CBI-FF-LCHC	CH-C WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
95	0.0	1.9E-14	GEL-CBI-FF-LCHD	CH-D WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHB	CH-B PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHD	CH-D PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHB	CH-B WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
96	0.0	1.8E-14	GEL-CBI-FF-LCHA	CH-A WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-FF-PCHC	CH-C PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
97	0.0	1.8E-14	GEL-CBI-FF-LCHC	CH-C WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-FF-PCHC	CH-C PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
98	0.0	1.8E-14	GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-FF-PCHC	CH-C PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
99	0.0	1.8E-14	GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-FF-PCHA	CH-A PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
100	0.0	1.6E-14	GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
			GEL-TLR-FF-K5C	CH-C PRESSURE RELAY K5C FAILS	1.9E-5

a. A / as the first character in a basic event name indicates a complemented event (Success = 1 - Failure). For example, the basic event for reactor low water level trip signal channel A in test and maintenance (T&M) is GEL-RPS-TM-ALVL (Failure = 1.40E-03). Thus, the basic event name for reactor low water level trip signal channel A not in T&M is /GEL-RPS-TM-ALVL (Success = 9.986E-01). The event description for complemented events remains the same as the description used for the failure event.

Table F-2. Importance measures sorted on Fussell-Vesely for case with RPS mincut = 5.8E-06.

Basic Event	Number of Occur.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
GEL-CBI-CF-TU4-8	1	3.07E-6	5.252E-01	2.106	1.711E+5	9.972E-01
GEL-SOV-CF-PSOVS	1	1.69E-6	2.899E-01	1.408	1.716E+5	1.000E+00
GEL-TLR-CF-TRP4-8	1	3.80E-7	6.519E-02	1.070	1.716E+5	1.000E+00
GEL-TLR-CF-CHACBD	1	2.75E-7	4.705E-02	1.049	1.711E+5	9.972E-01
GEL-ROD-CF-CRD	1	2.50E-7	4.289E-02	1.045	1.716E+5	1.000E+00
GEL-ACC-CF-HCU	1	1.09E-7	1.870E-02	1.019	1.716E+5	1.000E+00
GEL-SDL-CF-HWL2-4	2	3.09E-5	7.061E-03	1.007	2.294E+2	1.332E-03
GEL-SDV-HL-WTRL2	10	6.66E-4	3.777E-03	1.004	6.668E+0	3.306E-05
GEL-SDV-HL-WTRL1	10	6.66E-4	3.777E-03	1.004	6.668E+0	3.306E-05
GEL-AOV-CF-HCU	1	6.94E-9	1.191E-03	1.001	1.716E+5	1.000E+00
GEL-CBI-CF-TMP3-7	1	4.15E-6	9.954E-04	1.001	2.409E+2	1.398E-03
GEL-CBI-CF-TML3-7	1	4.15E-6	9.954E-04	1.001	2.409E+2	1.398E-03
GEL-TLR-CF-K1-2-4	2	1.36E-6	3.108E-04	1.000	2.294E+2	1.332E-03
GEL-RPS-TM-APRES	3651	1.40E-3	2.922E-04	1.000	1.208E+0	1.216E-06
GEL-RPS-TM-ALVL	3651	1.40E-3	2.872E-04	1.000	1.205E+0	1.196E-06
GEL-TLR-CF-TM-PR	1	3.93E-7	9.426E-05	1.000	2.409E+2	1.398E-03
GEL-TLR-CF-TM-LV	1	3.93E-7	9.426E-05	1.000	2.409E+2	1.398E-03
GEL-SDL-FC-LDMB	4	6.13E-4	8.858E-05	1.000	1.144E+0	8.422E-07
GEL-SDL-FC-LCMB	4	6.13E-4	8.858E-05	1.000	1.144E+0	8.422E-07
GEL-SDL-FC-LBMA	4	6.13E-4	8.858E-05	1.000	1.144E+0	8.422E-07
GEL-SDL-FC-LAMA	4	6.13E-4	8.858E-05	1.000	1.144E+0	8.422E-07
GEL-CPL-CF-L2-4	275	7.10E-5	6.861E-05	1.000	1.966E+0	5.633E-06
GEL-CPR-CF-P2-4	275	4.86E-6	6.238E-05	1.000	1.383E+1	7.481E-05
GEL-CBI-FF-PCHC	499	2.89E-4	6.338E-06	1.000	1.022E+0	1.278E-07
GEL-TLR-FF-K1D	4	1.93E-5	2.789E-06	1.000	1.144E+0	8.422E-07
GEL-TLR-FF-K1C	4	1.93E-5	2.789E-06	1.000	1.144E+0	8.422E-07
GEL-TLR-FF-K1B	4	1.93E-5	2.789E-06	1.000	1.144E+0	8.422E-07
GEL-TLR-FF-K1A	4	1.93E-5	2.789E-06	1.000	1.144E+0	8.422E-07
GEL-CPL-FF-LCHC	499	7.72E-4	1.713E-06	1.000	1.002E+0	1.293E-08
GEL-CBI-FF-PCHA	255	2.89E-4	1.338E-06	1.000	1.005E+0	2.699E-08
GEL-CBI-FF-PCHD	474	2.89E-4	1.310E-06	1.000	1.005E+0	2.643E-08
GEL-CBI-FF-PCHB	474	2.89E-4	1.310E-06	1.000	1.005E+0	2.643E-08
GEL-CPR-FF-PCHC	499	5.71E-5	1.252E-06	1.000	1.022E+0	1.278E-07
GEL-CPL-FF-LCHA	255	7.72E-4	7.937E-07	1.000	1.001E+0	5.993E-09
GEL-CPL-FF-LCHD	474	7.72E-4	7.161E-07	1.000	1.001E+0	5.408E-09
GEL-CPL-FF-LCHB	474	7.72E-4	7.161E-07	1.000	1.001E+0	5.408E-09
GEL-CBI-FF-LCHC	499	2.89E-4	6.411E-07	1.000	1.002E+0	1.293E-08
GEL-TLR-FF-K5C	499	1.93E-5	4.233E-07	1.000	1.022E+0	1.278E-07
GEL-CBI-FF-LCHA	255	2.89E-4	2.970E-07	1.000	1.001E+0	5.993E-09
GEL-CBI-FF-LCHD	474	2.89E-4	2.680E-07	1.000	1.001E+0	5.408E-09
GEL-CBI-FF-LCHB	474	2.89E-4	2.680E-07	1.000	1.001E+0	5.408E-09
GEL-CPR-FF-PCHA	255	5.71E-5	2.643E-07	1.000	1.005E+0	2.699E-08
GEL-CPR-FF-PCHD	474	5.71E-5	2.588E-07	1.000	1.005E+0	2.643E-08
GEL-CPR-FF-PCHB	474	5.71E-5	2.588E-07	1.000	1.005E+0	2.643E-08
GEL-CPL-CF-TML2-3	172	1.22E-4	1.501E-07	1.000	1.001E+0	7.168E-09
GEL-CPR-CF-TMP2-3	172	6.35E-6	1.117E-07	1.000	1.018E+0	1.025E-07
GEL-SOV-FF-SOVB	154	6.97E-4	9.779E-08	1.000	1.000E+0	8.189E-10
GEL-SOV-FF-SOVA	154	6.97E-4	9.779E-08	1.000	1.000E+0	8.189E-10
GEL-TLR-FF-K5A	255	1.93E-5	8.933E-08	1.000	1.005E+0	2.699E-08
GEL-TLR-FF-K5D	474	1.93E-5	8.752E-08	1.000	1.005E+0	2.643E-08
GEL-TLR-FF-K5B	474	1.93E-5	8.752E-08	1.000	1.005E+0	2.643E-08
GEL-TLR-FF-K14G	552	1.93E-5	5.495E-08	1.000	1.003E+0	1.664E-08
GEL-TLR-FF-K14C	552	1.93E-5	5.495E-08	1.000	1.003E+0	1.664E-08
GEL-TLR-FF-K14H	742	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08

Basic Event	Number of Occur.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
GEL-TLR-FF-K14F	742	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14E	836	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14D	742	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14B	742	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14A	836	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K6C	499	1.93E-5	4.270E-08	1.000	1.002E+0	1.293E-08
GEL-TLR-FF-K6A	255	1.93E-5	1.973E-08	1.000	1.001E+0	5.993E-09
GEL-TLR-FF-K6D	474	1.93E-5	1.783E-08	1.000	1.001E+0	5.408E-09
GEL-TLR-FF-K6B	474	1.93E-5	1.783E-08	1.000	1.001E+0	5.408E-09
GEL-PWR-FF-SOVB	154	6.00E-5	8.286E-09	1.000	1.000E+0	8.189E-10
GEL-PWR-FF-SOVA	154	6.00E-5	8.286E-09	1.000	1.000E+0	8.189E-10
GEL-PWR-CF-PWRAB	308	2.13E-6	5.333E-10	1.000	1.000E+0	1.638E-09

Table F-3. Importance measures sorted on Risk Increase for case with RPS mincut = 5.8E-06.

Basic Event	Number of Occur.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
GEL-TLR-CF-TRP4-8	1	3.80E-7	6.519E-02	1.070	1.716E+5	1.000E+00
GEL-SOV-CF-PSOVS	1	1.69E-6	2.899E-01	1.408	1.716E+5	1.000E+00
GEL-ROD-CF-CRD	1	2.50E-7	4.289E-02	1.045	1.716E+5	1.000E+00
GEL-AOV-CF-HCU	1	6.94E-9	1.191E-03	1.001	1.716E+5	1.000E+00
GEL-ACC-CF-HCU	1	1.09E-7	1.870E-02	1.019	1.716E+5	1.000E+00
GEL-TLR-CF-CHACBD	1	2.75E-7	4.705E-02	1.049	1.711E+5	9.972E-01
GEL-CBI-CF-TU4-8	1	3.07E-6	5.252E-01	2.106	1.711E+5	9.972E-01
GEL-TLR-CF-TM-PR	1	3.93E-7	9.426E-05	1.000	2.409E+2	1.398E-03
GEL-TLR-CF-TM-LV	1	3.93E-7	9.426E-05	1.000	2.409E+2	1.398E-03
GEL-CBI-CF-TMP3-7	1	4.15E-6	9.954E-04	1.001	2.409E+2	1.398E-03
GEL-CBI-CF-TML3-7	1	4.15E-6	9.954E-04	1.001	2.409E+2	1.398E-03
GEL-TLR-CF-K1-2-4	2	1.36E-6	3.108E-04	1.000	2.294E+2	1.332E-03
GEL-SDL-CF-HWL2-4	2	3.09E-5	7.061E-03	1.007	2.294E+2	1.332E-03
GEL-CPR-CF-P2-4	275	4.86E-6	6.238E-05	1.000	1.383E+1	7.481E-05
GEL-SDV-HL-WTRL2	10	6.66E-4	3.777E-03	1.004	6.668E+0	3.306E-05
GEL-SDV-HL-WTRL1	10	6.66E-4	3.777E-03	1.004	6.668E+0	3.306E-05
GEL-CPL-CF-L2-4	275	7.10E-5	6.861E-05	1.000	1.966E+0	5.633E-06
GEL-RPS-TM-APRES	3651	1.40E-3	2.922E-04	1.000	1.208E+0	1.216E-06
GEL-RPS-TM-ALVL	3651	1.40E-3	2.872E-04	1.000	1.205E+0	1.196E-06
GEL-TLR-FF-K1D	4	1.93E-5	2.789E-06	1.000	1.144E+0	8.422E-07
GEL-TLR-FF-K1C	4	1.93E-5	2.789E-06	1.000	1.144E+0	8.422E-07
GEL-TLR-FF-K1B	4	1.93E-5	2.789E-06	1.000	1.144E+0	8.422E-07
GEL-TLR-FF-K1A	4	1.93E-5	2.789E-06	1.000	1.144E+0	8.422E-07
GEL-SDL-FC-LDMB	4	6.13E-4	8.858E-05	1.000	1.144E+0	8.422E-07
GEL-SDL-FC-LCMB	4	6.13E-4	8.858E-05	1.000	1.144E+0	8.422E-07
GEL-SDL-FC-LBMA	4	6.13E-4	8.858E-05	1.000	1.144E+0	8.422E-07
GEL-SDL-FC-LAMA	4	6.13E-4	8.858E-05	1.000	1.144E+0	8.422E-07
GEL-TLR-FF-K5C	499	1.93E-5	4.233E-07	1.000	1.022E+0	1.278E-07
GEL-CPR-FF-PCHC	499	5.71E-5	1.252E-06	1.000	1.022E+0	1.278E-07
GEL-CBI-FF-PCHC	499	2.89E-4	6.338E-06	1.000	1.022E+0	1.278E-07
GEL-CPR-CF-TMP2-3	172	6.35E-6	1.117E-07	1.000	1.018E+0	1.025E-07
GEL-TLR-FF-K5D	474	1.93E-5	8.752E-08	1.000	1.005E+0	2.643E-08
GEL-TLR-FF-K5B	474	1.93E-5	8.752E-08	1.000	1.005E+0	2.643E-08
GEL-TLR-FF-K5A	255	1.93E-5	8.933E-08	1.000	1.005E+0	2.699E-08
GEL-CPR-FF-PCHD	474	5.71E-5	2.588E-07	1.000	1.005E+0	2.643E-08
GEL-CPR-FF-PCHB	474	5.71E-5	2.588E-07	1.000	1.005E+0	2.643E-08
GEL-CPR-FF-PCHA	255	5.71E-5	2.643E-07	1.000	1.005E+0	2.699E-08
GEL-CBI-FF-PCHD	474	2.89E-4	1.310E-06	1.000	1.005E+0	2.643E-08
GEL-CBI-FF-PCHB	474	2.89E-4	1.310E-06	1.000	1.005E+0	2.643E-08
GEL-CBI-FF-PCHA	255	2.89E-4	1.338E-06	1.000	1.005E+0	2.699E-08
GEL-TLR-FF-K14H	742	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14G	552	1.93E-5	5.495E-08	1.000	1.003E+0	1.664E-08
GEL-TLR-FF-K14F	742	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14E	836	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14D	742	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14C	552	1.93E-5	5.495E-08	1.000	1.003E+0	1.664E-08
GEL-TLR-FF-K14B	742	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14A	836	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K6C	499	1.93E-5	4.270E-08	1.000	1.002E+0	1.293E-08
GEL-CPL-FF-LCHC	499	7.72E-4	1.713E-06	1.000	1.002E+0	1.293E-08
GEL-CBI-FF-LCHC	499	2.89E-4	6.411E-07	1.000	1.002E+0	1.293E-08
GEL-TLR-FF-K6D	474	1.93E-5	1.783E-08	1.000	1.001E+0	5.408E-09
GEL-TLR-FF-K6B	474	1.93E-5	1.783E-08	1.000	1.001E+0	5.408E-09
GEL-TLR-FF-K6A	255	1.93E-5	1.973E-08	1.000	1.001E+0	5.993E-09

Basic Event	Number of Occur.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
GEL-CPL-FF-LCHD	474	7.72E-4	7.161E-07	1.000	1.001E+0	5.408E-09
GEL-CPL-FF-LCHB	474	7.72E-4	7.161E-07	1.000	1.001E+0	5.408E-09
GEL-CPL-FF-LCHA	255	7.72E-4	7.937E-07	1.000	1.001E+0	5.993E-09
GEL-CPL-CF-TML2-3	172	1.22E-4	1.501E-07	1.000	1.001E+0	7.168E-09
GEL-CBI-FF-LCHD	474	2.89E-4	2.680E-07	1.000	1.001E+0	5.408E-09
GEL-CBI-FF-LCHB	474	2.89E-4	2.680E-07	1.000	1.001E+0	5.408E-09
GEL-CBI-FF-LCHA	255	2.89E-4	2.970E-07	1.000	1.001E+0	5.993E-09
GEL-SOV-FF-SOVB	154	6.97E-4	9.779E-08	1.000	1.000E+0	8.189E-10
GEL-SOV-FF-SOVA	154	6.97E-4	9.779E-08	1.000	1.000E+0	8.189E-10
GEL-PWR-FF-SOVB	154	6.00E-5	8.286E-09	1.000	1.000E+0	8.189E-10
GEL-PWR-FF-SOVA	154	6.00E-5	8.286E-09	1.000	1.000E+0	8.189E-10
GEL-PWR-CF-PWRAB	308	2.13E-6	5.333E-10	1.000	1.000E+0	1.638E-09

Table F-4. Importance measures sorted on Birnbaum for case with RPS mincut = 5.8E-06.

Basic Event	Number of Occur.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
GEL-TLR-CF-TRP4-8	1	3.80E-7	6.519E-02	1.070	1.716E+5	1.000E+00
GEL-SOV-CF-PSOVS	1	1.69E-6	2.899E-01	1.408	1.716E+5	1.000E+00
GEL-ROD-CF-CRD	1	2.50E-7	4.289E-02	1.045	1.716E+5	1.000E+00
GEL-AOV-CF-HCU	1	6.94E-9	1.191E-03	1.001	1.716E+5	1.000E+00
GEL-ACC-CF-HCU	1	1.09E-7	1.870E-02	1.019	1.716E+5	1.000E+00
GEL-TLR-CF-CHACBD	1	2.75E-7	4.705E-02	1.049	1.711E+5	9.972E-01
GEL-CBI-CF-TU4-8	1	3.07E-6	5.252E-01	2.106	1.711E+5	9.972E-01
GEL-TLR-CF-TM-PR	1	3.93E-7	9.426E-05	1.000	2.409E+2	1.398E-03
GEL-TLR-CF-TM-LV	1	3.93E-7	9.426E-05	1.000	2.409E+2	1.398E-03
GEL-CBI-CF-TMP3-7	1	4.15E-6	9.954E-04	1.001	2.409E+2	1.398E-03
GEL-CBI-CF-TML3-7	1	4.15E-6	9.954E-04	1.001	2.409E+2	1.398E-03
GEL-TLR-CF-K1-2-4	2	1.36E-6	3.108E-04	1.000	2.294E+2	1.332E-03
GEL-SDL-CF-HWL2-4	2	3.09E-5	7.061E-03	1.007	2.294E+2	1.332E-03
GEL-CPR-CF-P2-4	275	4.86E-6	6.238E-05	1.000	1.383E+1	7.481E-05
GEL-SDV-HL-WTRL2	10	6.66E-4	3.777E-03	1.004	6.668E+0	3.306E-05
GEL-SDV-HL-WTRL1	10	6.66E-4	3.777E-03	1.004	6.668E+0	3.306E-05
GEL-CPL-CF-L2-4	275	7.10E-5	6.861E-05	1.000	1.966E+0	5.633E-06
GEL-RPS-TM-APRES	3651	1.40E-3	2.922E-04	1.000	1.208E+0	1.216E-06
GEL-RPS-TM-ALVL	3651	1.40E-3	2.872E-04	1.000	1.205E+0	1.196E-06
GEL-TLR-FF-K1D	4	1.93E-5	2.789E-06	1.000	1.144E+0	8.422E-07
GEL-TLR-FF-K1C	4	1.93E-5	2.789E-06	1.000	1.144E+0	8.422E-07
GEL-TLR-FF-K1B	4	1.93E-5	2.789E-06	1.000	1.144E+0	8.422E-07
GEL-TLR-FF-K1A	4	1.93E-5	2.789E-06	1.000	1.144E+0	8.422E-07
GEL-SDL-FC-LDMB	4	6.13E-4	8.858E-05	1.000	1.144E+0	8.422E-07
GEL-SDL-FC-LCMB	4	6.13E-4	8.858E-05	1.000	1.144E+0	8.422E-07
GEL-SDL-FC-LBMA	4	6.13E-4	8.858E-05	1.000	1.144E+0	8.422E-07
GEL-SDL-FC-LAMA	4	6.13E-4	8.858E-05	1.000	1.144E+0	8.422E-07
GEL-TLR-FF-K5C	499	1.93E-5	4.233E-07	1.000	1.022E+0	1.278E-07
GEL-CPR-FF-PCHC	499	5.71E-5	1.252E-06	1.000	1.022E+0	1.278E-07
GEL-CBI-FF-PCHC	499	2.89E-4	6.338E-06	1.000	1.022E+0	1.278E-07
GEL-CPR-CF-TMP2-3	172	6.35E-6	1.117E-07	1.000	1.018E+0	1.025E-07
GEL-TLR-FF-K5A	255	1.93E-5	8.933E-08	1.000	1.005E+0	2.699E-08
GEL-CPR-FF-PCHA	255	5.71E-5	2.643E-07	1.000	1.005E+0	2.699E-08
GEL-CBI-FF-PCHA	255	2.89E-4	1.338E-06	1.000	1.005E+0	2.699E-08
GEL-TLR-FF-K5D	474	1.93E-5	8.752E-08	1.000	1.005E+0	2.643E-08
GEL-TLR-FF-K5B	474	1.93E-5	8.752E-08	1.000	1.005E+0	2.643E-08
GEL-CPR-FF-PCHD	474	5.71E-5	2.588E-07	1.000	1.005E+0	2.643E-08
GEL-CPR-FF-PCHB	474	5.71E-5	2.588E-07	1.000	1.005E+0	2.643E-08
GEL-CBI-FF-PCHD	474	2.89E-4	1.310E-06	1.000	1.005E+0	2.643E-08
GEL-CBI-FF-PCHB	474	2.89E-4	1.310E-06	1.000	1.005E+0	2.643E-08
GEL-TLR-FF-K14G	552	1.93E-5	5.495E-08	1.000	1.003E+0	1.664E-08
GEL-TLR-FF-K14C	552	1.93E-5	5.495E-08	1.000	1.003E+0	1.664E-08
GEL-TLR-FF-K14H	742	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14F	742	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14E	836	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14D	742	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14B	742	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K14A	836	1.93E-5	4.952E-08	1.000	1.003E+0	1.498E-08
GEL-TLR-FF-K6C	499	1.93E-5	4.270E-08	1.000	1.002E+0	1.293E-08
GEL-CPL-FF-LCHC	499	7.72E-4	1.713E-06	1.000	1.002E+0	1.293E-08
GEL-CBI-FF-LCHC	499	2.89E-4	6.411E-07	1.000	1.002E+0	1.293E-08
GEL-CPL-CF-TML2-3	172	1.22E-4	1.501E-07	1.000	1.001E+0	7.168E-09
GEL-TLR-FF-K6A	255	1.93E-5	1.973E-08	1.000	1.001E+0	5.993E-09
GEL-CPL-FF-LCHA	255	7.72E-4	7.937E-07	1.000	1.001E+0	5.993E-09

Basic Event	Number of Occur.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
GEL-CBI-FF-LCHA	255	2.89E-4	2.970E-07	1.000	1.001E+0	5.993E-09
GEL-TLR-FF-K6D	474	1.93E-5	1.783E-08	1.000	1.001E+0	5.408E-09
GEL-TLR-FF-K6B	474	1.93E-5	1.783E-08	1.000	1.001E+0	5.408E-09
GEL-CPL-FF-LCHD	474	7.72E-4	7.161E-07	1.000	1.001E+0	5.408E-09
GEL-CPL-FF-LCHB	474	7.72E-4	7.161E-07	1.000	1.001E+0	5.408E-09
GEL-CBI-FF-LCHD	474	2.89E-4	2.680E-07	1.000	1.001E+0	5.408E-09
GEL-CBI-FF-LCHB	474	2.89E-4	2.680E-07	1.000	1.001E+0	5.408E-09
GEL-PWR-CF-PWRAB	308	2.13E-6	5.333E-10	1.000	1.000E+0	1.638E-09
GEL-SOV-FF-SOVB	154	6.97E-4	9.779E-08	1.000	1.000E+0	8.189E-10
GEL-SOV-FF-SOVA	154	6.97E-4	9.779E-08	1.000	1.000E+0	8.189E-10
GEL-PWR-FF-SOVB	154	6.00E-5	8.286E-09	1.000	1.000E+0	8.189E-10
GEL-PWR-FF-SOVA	154	6.00E-5	8.286E-09	1.000	1.000E+0	8.189E-10

Table F-5. GE RPS mincut = 5.8E-6 basic event failure probability and uncertainty data.

	Basic Event Name	Prob.	Distr. Type	Uncert. Value[a]	Correlation Class	Basic Event Description
1	GEL-ACC-CF-HCU	1.09E-7	Lognormal	4.41	-	CCF 33% OR MORE HCU ACCUMULATORS FAIL
2	GEL-AOV-CF-HCU	6.94E-9	Lognormal	1.07	-	CCF 33% OR MORE HCU SCRAM INLET/OUTLET AOVs FAIL TO OPEN
3	GEL-CBI-CF-TML3-7	4.15E-6	Lognormal	3.06	CBI2	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (LEVEL T&M)
4	GEL-CBI-CF-TMP3-7	4.15E-6	Lognormal	3.06	CBI2	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (PRES T&M)
5	GEL-CBI-CF-TU4-8	3.07E-6	Lognormal	3.67	-	CCF SPECIFIC 4 OR MORE TRIP UNITS
6	GEL-CBI-FF-LCHA	2.89E-4	Lognormal	6.24	CBI1	CH-A WATER LEVEL TRIP UNIT FAILS
7	GEL-CBI-FF-LCHB	2.89E-4	Lognormal	6.24	CBI1	CH-B WATER LEVEL TRIP UNIT FAILS
8	GEL-CBI-FF-LCHC	2.89E-4	Lognormal	6.24	CBI1	CH-C WATER LEVEL TRIP UNIT FAILS
9	GEL-CBI-FF-LCHD	2.89E-4	Lognormal	6.24	CBI1	CH-D WATER LEVEL TRIP UNIT FAILS
10	GEL-CBI-FF-PCHA	2.89E-4	Lognormal	6.24	CBI1	CH-A PRESSURE TRIP UNIT FAILS
11	GEL-CBI-FF-PCHB	2.89E-4	Lognormal	6.24	CBI1	CH-B PRESSURE TRIP UNIT FAILS
12	GEL-CBI-FF-PCHC	2.89E-4	Lognormal	6.24	CBI1	CH-C PRESSURE TRIP UNIT FAILS
13	GEL-CBI-FF-PCHD	2.89E-4	Lognormal	6.24	CBI1	CH-D PRESSURE TRIP UNIT FAILS
14	GEL-CPL-CF-L2-4	7.10E-5	Lognormal	12.11	-	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS
15	GEL-CPL-CF-TML2-3	1.22E-4	Lognormal	12.01	-	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTER (LEVEL T&M)
16	GEL-CPL-FF-LCHA	7.72E-4	Lognormal	11.04	CPL1	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS
17	GEL-CPL-FF-LCHB	7.72E-4	Lognormal	11.04	CPL1	CH-B WATER LEVEL SENSOR/TRANSMITTER FAILS
18	GEL-CPL-FF-LCHC	7.72E-4	Lognormal	11.04	CPL1	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS
19	GEL-CPL-FF-LCHD	7.72E-4	Lognormal	11.04	CPL1	CH-D WATER LEVEL SENSOR/TRANSMITTER FAILS
20	GEL-CPR-CF-P2-4	4.86E-6	Lognormal	7.05	-	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS
21	GEL-CPR-CF-TMP2-3	6.35E-6	Lognormal	6.99	-	CCF SPECIFIC 2 OR MORE PRES SENSOR/TRANSMITTER (PRES T&M)
22	GEL-CPR-FF-PCHA	5.71E-5	Lognormal	5.61	CPR1	CH-A PRESSURE SENSOR/TRANSMITTER FAILS
23	GEL-CPR-FF-PCHB	5.71E-5	Lognormal	5.61	CPR1	CH-B PRESSURE SENSOR/TRANSMITTER FAILS
24	GEL-CPR-FF-PCHC	5.71E-5	Lognormal	5.61	CPR1	CH-C PRESSURE SENSOR/TRANSMITTER FAILS
25	GEL-CPR-FF-PCHD	5.71E-5	Lognormal	5.61	CPR1	CH-D PRESSURE SENSOR/TRANSMITTER FAILS
26	GEL-MSW-CF-MSSAB	TRUE	-	-	-	CCF MANUAL SCRAM SWITCH A AND B
27	GEL-MSW-FF-MSSA	TRUE	-	-	-	MANUAL SCRAM SWITCH A FAILS
28	GEL-MSW-FF-MSSB	TRUE	-	-	-	MANUAL SCRAM SWITCH B FAILS
29	GEL-PWR-CF-PWRAB	2.13E-6	Lognormal	17.9	-	CCF 125 VDC POWER (SOV A AND SOV B)
30	GEL-PWR-FF-SOVA	6.00E-5	Lognormal	10.0	PWR1	125 VDC POWER FAILS (SOV A)
31	GEL-PWR-FF-SOVB	6.00E-5	Lognormal	10.0	PWR1	125 VDC POWER FAILS (SOV B)
32	GEL-ROD-CF-CRD	2.50E-7	Lognormal	5.81	-	CCF 33% OR MORE CRD/RODS FAIL TO INSERT
33	GEL-RPS-TM-ALVL	1.40E-3	Uniform	2.8E-3	RPS1	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL
34	GEL-RPS-TM-APRES	1.40E-3	Uniform	2.8E-3	RPS1	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL
35	GEL-SDL-CF-HWL2-4	3.09E-5	Lognormal	8.24	-	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCHES

	Basic Event Name	Prob.	Distr. Type	Uncert. Value[a]	Correlation Class	Basic Event Description
36	GEL-SDL-FC-LAMA	6.13E-4	Lognormal	5.99	SDL1	LEVEL SWITCH A (MANUFACTURER A) FAILS
37	GEL-SDL-FC-LBMA	6.13E-4	Lognormal	5.99	SDL1	LEVEL SWITCH B (MANUFACTURER A) FAILS
38	GEL-SDL-FC-LCMB	6.13E-4	Lognormal	5.99	SDL1	LEVEL SWITCH C (MANUFACTURER B) FAILS
39	GEL-SDL-FC-LDMB	6.13E-4	Lognormal	5.99	SDL1	LEVEL SWITCH D (MANUFACTURER B) FAILS
40	GEL-SDV-HL-WTRL1	6.66E-4	Lognormal	3.24	SDV1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH
41	GEL-SDV-HL-WTRL2	6.66E-4	Lognormal	3.24	SDV1	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH
42	GEL-SOV-CF-PSOVS	1.69E-6	Lognormal	10.47	-	CCF 33% OR MORE HCU SCRAM PILOT SOVs OR BACKUP SOVs FAIL
43	GEL-SOV-FF-SOVA	6.97E-4	Lognormal	10.44	SOV1	BACKUP SCRAM SOV A FAILS TO ACTUATE
44	GEL-SOV-FF-SOVB	6.97E-4	Lognormal	10.44	SOV1	BACKUP SCRAM SOV B FAILS TO ACTUATE
45	GEL-TLR-CF-CHACBD	2.75E-7	Lognormal	9.05	-	CCF CHANNEL RELAYS (NO T&M)
46	GEL-TLR-CF-K1-2-4	1.36E-6	Lognormal	8.09	-	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCH RELAYS
47	GEL-TLR-CF-TM-LV	3.93E-7	Lognormal	8.11	TLR2	CCF CHANNEL RELAYS (LEVEL T&M)
48	GEL-TLR-CF-TM-PR	3.93E-7	Lognormal	8.11	TLR2	CCF CHANNEL RELAYS (PRES T&M)
49	GEL-TLR-CF-TRP4-8	3.80E-7	Lognormal	8.47	-	CCF SPECIFIC 4 OR MORE TRIP SYSTEM RELAYS
50	GEL-TLR-FF-K14A	1.93E-5	Lognormal	6.11	TLR1	TRIP SYSTEM A RELAY K14A FAILS
51	GEL-TLR-FF-K14B	1.93E-5	Lognormal	6.11	TLR1	TRIP SYSTEM B RELAY K14B FAILS
52	GEL-TLR-FF-K14C	1.93E-5	Lognormal	6.11	TLR1	TRIP SYSTEM A RELAY K14C FAILS
53	GEL-TLR-FF-K14D	1.93E-5	Lognormal	6.11	TLR1	TRIP SYSTEM B RELAY K14D FAILS
54	GEL-TLR-FF-K14E	1.93E-5	Lognormal	6.11	TLR1	TRIP SYSTEM A RELAY K14E FAILS
55	GEL-TLR-FF-K14F	1.93E-5	Lognormal	6.11	TLR1	TRIP SYSTEM B RELAY K14F FAILS
56	GEL-TLR-FF-K14G	1.93E-5	Lognormal	6.11	TLR1	TRIP SYSTEM A RELAY K14G FAILS
57	GEL-TLR-FF-K14H	1.93E-5	Lognormal	6.11	TLR1	TRIP SYSTEM B RELAY K14H FAILS
58	GEL-TLR-FF-K15A	TRUE	-	-	-	CH-A MANUAL SCRAM SWITCH RELAY K15A FAILS
59	GEL-TLR-FF-K15B	TRUE	-	-	-	CH-B MANUAL SCRAM SWITCH RELAY K15B FAILS
60	GEL-TLR-FF-K15C	TRUE	-	-	-	CH-C MANUAL SCRAM SWITCH RELAY K15C FAILS
61	GEL-TLR-FF-K15D	TRUE	-	-	-	CH-D MANUAL SCRAM SWITCH RELAY K15D FAILS
62	GEL-TLR-FF-K1A	1.93E-5	Lognormal	6.11	TLR1	RELAY K1A FAILS
63	GEL-TLR-FF-K1B	1.93E-5	Lognormal	6.11	TLR1	RELAY K1B FAILS
64	GEL-TLR-FF-K1C	1.93E-5	Lognormal	6.11	TLR1	RELAY K1C FAILS
65	GEL-TLR-FF-K1D	1.93E-5	Lognormal	6.11	TLR1	RELAY K1D FAILS
66	GEL-TLR-FF-K5A	1.93E-5	Lognormal	6.11	TLR1	CH-A PRESSURE RELAY K5A FAILS
67	GEL-TLR-FF-K5B	1.93E-5	Lognormal	6.11	TLR1	CH-B PRESSURE RELAY K5B FAILS
6	GEL-TLR-FF-K5C	1.93E-5	Lognormal	6.11	TLR1	CH-C PRESSURE RELAY K5C FAILS

	Basic Event Name	Prob.	Distr. Type	Uncert. Value[a]	Correlation Class	Basic Event Description
8						
69	GEL-TLR-FF-K5D	1.93E-5	Lognormal	6.11	TLR1	CH-D PRESSURE RELAY K5D FAILS
70	GEL-TLR-FF-K6A	1.93E-5	Lognormal	6.11	TLR1	CH-A WATER LEVEL RELAY K6A FAILS
71	GEL-TLR-FF-K6B	1.93E-5	Lognormal	6.11	TLR1	CH-B WATER LEVEL RELAY K6B FAILS
72	GEL-TLR-FF-K6C	1.93E-5	Lognormal	6.11	TLR1	CH-C WATER LEVEL RELAY K6C FAILS
73	GEL-TLR-FF-K6D	1.93E-5	Lognormal	6.11	TLR1	CH-D WATER LEVEL RELAY K6D FAILS
74	GEL-XHE-XE-SCRAM	TRUE	-	-	-	OPERATOR FAILS TO INITIATE SCRAM

a. The uncertainty (Uncert.) value is the parameter that is used to describe the uncertainty distribution for the associated basic event. The lognormal and uniform distributions are the only two distributions used for the RPS basic events. The lognormal distribution is described by the mean and the upper 95% error factor. The uniform distribution is described by the mid point (mean probability) and the upper endpoint (Uncert. Value).

Table F-6. Top 100 cut sets (operator fails to initiate scram = 0.01) RPS mincut = 2.6E-06.

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
1	64.4	1.7E-06	GEL-SOV-CF-PSOVS	CCF 33% OR MORE HCU SCRAM PILOT SOVS OR BACKUP SOVS FAIL	1.7E-6
2	14.5	3.8E-07	GEL-TLR-CF-TRP4-8	CCF SPECIFIC 4 OR MORE TRIP SYSTEM RELAYS	3.8E-7
3	9.5	2.5E-07	GEL-ROD-CF-CRD	CCF 33% OR MORE CRD/RODS FAIL TO INSERT	2.5E-7
4	4.3	1.1E-07	/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-CF-CHACBD	CCF CHANNEL RELAYS (NO T&M)	1.1E-7
5	4.2	1.1E-07	GEL-ACC-CF-HCU	CCF 33% OR MORE HCU ACCUMULATORS FAIL	1.1E-7
6	1.2	3.1E-08	GEL-CBI-CF-TU4-8	CCF SPECIFIC 4 OR MORE TRIP UNITS	3.1E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
7	0.8	2.1E-08	GEL-SDL-CF-HWL2-4	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCHES	3.1E-5
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
8	0.8	2.1E-08	GEL-SDL-CF-HWL2-4	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCHES	3.1E-5
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
9	0.3	6.9E-09	GEL-AOV-CF-HCU	CCF 33% OR MORE HCU SCRAM INLET/OUTLET AOVS FAIL TO OPEN	6.9E-9
10	0.0	9.1E-10	GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-CF-K1-2-4	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCH RELAYS	1.4E-6
11	0.0	9.1E-10	GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-CF-K1-2-4	CCF SPECIFIC 2 OR MORE SDV HIGH WATER LEVEL SWITCH RELAYS	1.4E-6
12	0.0	2.5E-10	GEL-SDL-FC-LAMA	LEVEL SWITCH A (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDL-FC-LCMB	LEVEL SWITCH C (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
13	0.0	2.5E-10	GEL-SDL-FC-LAMA	LEVEL SWITCH A (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDL-FC-LCMB	LEVEL SWITCH C (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
14	0.0	2.5E-10	GEL-SDL-FC-LBMA	LEVEL SWITCH B (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDL-FC-LDMB	LEVEL SWITCH D (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
15	0.0	2.5E-10	GEL-SDL-FC-LBMA	LEVEL SWITCH B (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDL-FC-LDMB	LEVEL SWITCH D (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
16	0.0	1.9E-10	GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-CF-TM-LV	CCF CHANNEL RELAYS (LEVEL T&M)	1.3E-7
17	0.0	1.9E-10	/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
			GEL-TLR-CF-TM-PR	CCF CHANNEL RELAYS (PRES T&M)	1.3E-7
18	0.0	5.8E-11	GEL-CBI-CF-TML3-7	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (LEVEL T&M)	4.2E-6
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
19	0.0	5.8E-11	GEL-CBI-CF-TMP3-7	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (PRES T&M)	4.2E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
20	0.0	4.0E-11	GEL-CBI-CF-TU4-8	CCF SPECIFIC 4 OR MORE TRIP UNITS	3.1E-6
			GEL-MSW-FF-MSSA	MANUAL SCRAM SWITCH A FAILS	1.3E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
21	0.0	4.0E-11	GEL-CBI-CF-TU4-8	CCF SPECIFIC 4 OR MORE TRIP UNITS	3.1E-6
			GEL-MSW-FF-MSSB	MANUAL SCRAM SWITCH B FAILS	1.3E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
22	0.0	7.9E-12	GEL-SDL-FC-LAMA	LEVEL SWITCH A (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1C	RELAY K1C FAILS	1.9E-5
23	0.0	7.9E-12	GEL-SDL-FC-LAMA	LEVEL SWITCH A (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1C	RELAY K1C FAILS	1.9E-5
24	0.0	7.9E-12	GEL-SDL-FC-LBMA	LEVEL SWITCH B (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1D	RELAY K1D FAILS	1.9E-5

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
25	0.0	7.9E-12	GEL-SDL-FC-LBMA	LEVEL SWITCH B (MANUFACTURER A) FAILS	6.1E-4
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1D	RELAY K1D FAILS	1.9E-5
26	0.0	7.9E-12	GEL-SDL-FC-LCMB	LEVEL SWITCH C (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1A	RELAY K1A FAILS	1.9E-5
27	0.0	7.9E-12	GEL-SDL-FC-LCMB	LEVEL SWITCH C (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1A	RELAY K1A FAILS	1.9E-5
28	0.0	7.9E-12	GEL-SDL-FC-LDMB	LEVEL SWITCH D (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1B	RELAY K1B FAILS	1.9E-5
29	0.0	7.9E-12	GEL-SDL-FC-LDMB	LEVEL SWITCH D (MANUFACTURER B) FAILS	6.1E-4
			GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1B	RELAY K1B FAILS	1.9E-5
30	0.0	3.4E-12	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
31	0.0	2.4E-12	GEL-CBI-CF-TU4-8	CCF SPECIFIC 4 OR MORE TRIP UNITS	3.1E-6
			GEL-MSW-CF-MSSAB	CCF MANUAL SCRAM SWITCH A AND B	7.7E-7
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
32	0.0	2.9E-13	GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
33	0.0	2.6E-13	GEL-SOV-FF-SOVA	BACKUP SCRAM SOV A FAILS TO ACTUATE	7.0E-4
			GEL-TLR-FF-K14E	TRIP SYSTEM A RELAY K14E FAILS	1.9E-5
			GEL-TLR-FF-K14G	TRIP SYSTEM A RELAY K14G FAILS	1.9E-5
34	0.0	2.6E-13	GEL-SOV-FF-SOVA	BACKUP SCRAM SOV A FAILS TO ACTUATE	7.0E-4
			GEL-TLR-FF-K14F	TRIP SYSTEM B RELAY K14F FAILS	1.9E-5
			GEL-TLR-FF-K14H	TRIP SYSTEM B RELAY K14H FAILS	1.9E-5
35	0.0	2.6E-13	GEL-SOV-FF-SOVB	BACKUP SCRAM SOV B FAILS TO ACTUATE	7.0E-4
			GEL-TLR-FF-K14A	TRIP SYSTEM A RELAY K14A FAILS	1.9E-5
			GEL-TLR-FF-K14C	TRIP SYSTEM A RELAY K14C FAILS	1.9E-5
36	0.0	2.6E-13	GEL-SOV-FF-SOVB	BACKUP SCRAM SOV B FAILS TO ACTUATE	7.0E-4
			GEL-TLR-FF-K14B	TRIP SYSTEM B RELAY K14B FAILS	1.9E-5
			GEL-TLR-FF-K14D	TRIP SYSTEM B RELAY K14D FAILS	1.9E-5
37	0.0	2.5E-13	GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1A	RELAY K1A FAILS	1.9E-5
			GEL-TLR-FF-K1C	RELAY K1C FAILS	1.9E-5
38	0.0	2.5E-13	GEL-SDV-HL-WTRL1	SCRAM DISCHARGE VOLUME HEADER 1 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1B	RELAY K1B FAILS	1.9E-5
			GEL-TLR-FF-K1D	RELAY K1D FAILS	1.9E-5
39	0.0	2.5E-13	GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1A	RELAY K1A FAILS	1.9E-5
			GEL-TLR-FF-K1C	RELAY K1C FAILS	1.9E-5
40	0.0	2.5E-13	GEL-SDV-HL-WTRL2	SCRAM DISCHARGE VOLUME HEADER 2 WATER LEVEL HIGH	6.7E-4
			GEL-TLR-FF-K1B	RELAY K1B FAILS	1.9E-5
			GEL-TLR-FF-K1D	RELAY K1D FAILS	1.9E-5
41	0.0	7.5E-14	GEL-CBI-CF-TML3-7	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (LEVEL T&M)	4.2E-6
			GEL-MSW-FF-MSSA	MANUAL SCRAM SWITCH A FAILS	1.3E-5
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
42	0.0	7.5E-14	GEL-CBI-CF-TML3-7	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (LEVEL T&M)	4.2E-6
			GEL-MSW-FF-MSSB	MANUAL SCRAM SWITCH B FAILS	1.3E-5
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
43	0.0	7.5E-14	GEL-CBI-CF-TMP3-7	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (PRES T&M)	4.2E-6
			GEL-MSW-FF-MSSA	MANUAL SCRAM SWITCH A FAILS	1.3E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
44	0.0	7.5E-14	GEL-CBI-CF-TMP3-7	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (PRES T&M)	4.2E-6
			GEL-MSW-FF-MSSB	MANUAL SCRAM SWITCH B FAILS	1.3E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
45	0.0	5.9E-14	GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
46	0.0	5.9E-14	GEL-CBI-FF-PCHB	CH-B PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHD	CH-D PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
47	0.0	5.7E-14	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHC	CH-C PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
48	0.0	5.3E-14	GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
49	0.0	2.9E-14	GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
50	0.0	2.9E-14	GEL-CPL-FF-LCHB	CH-B WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPL-FF-LCHD	CH-D WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
51	0.0	2.2E-14	GEL-PWR-FF-SOVA	125 VDC POWER FAILS (SOV A)	6.0E-5
			GEL-TLR-FF-K14E	TRIP SYSTEM A RELAY K14E FAILS	1.9E-5
			GEL-TLR-FF-K14G	TRIP SYSTEM A RELAY K14G FAILS	1.9E-5
52	0.0	2.2E-14	GEL-PWR-FF-SOVA	125 VDC POWER FAILS (SOV A)	6.0E-5
			GEL-TLR-FF-K14F	TRIP SYSTEM B RELAY K14F FAILS	1.9E-5
			GEL-TLR-FF-K14H	TRIP SYSTEM B RELAY K14H FAILS	1.9E-5
53	0.0	2.2E-14	GEL-PWR-FF-SOVB	125 VDC POWER FAILS (SOV B)	6.0E-5
			GEL-TLR-FF-K14A	TRIP SYSTEM A RELAY K14A FAILS	1.9E-5
			GEL-TLR-FF-K14C	TRIP SYSTEM A RELAY K14C FAILS	1.9E-5
54	0.0	2.2E-14	GEL-PWR-FF-SOVB	125 VDC POWER FAILS (SOV B)	6.0E-5
			GEL-TLR-FF-K14B	TRIP SYSTEM B RELAY K14B FAILS	1.9E-5
			GEL-TLR-FF-K14D	TRIP SYSTEM B RELAY K14D FAILS	1.9E-5
55	0.0	2.0E-14	GEL-CBI-FF-LCHC	CH-C WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
56	0.0	1.9E-14	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
			GEL-TLR-FF-K5C	CH-C PRESSURE RELAY K5C FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
57	0.0	1.2E-14	GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHC	CH-C PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
58	0.0	1.2E-14	GEL-CBI-FF-PCHB	CH-B PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHD	CH-D PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
59	0.0	1.2E-14	GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHA	CH-A PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
60	0.0	1.2E-14	GEL-CBI-FF-PCHD	CH-D PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHB	CH-B PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
61	0.0	1.1E-14	GEL-CBI-FF-LCHA	CH-A WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
62	0.0	1.1E-14	GEL-CBI-FF-LCHB	CH-B WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHD	CH-D WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
63	0.0	1.1E-14	GEL-CBI-FF-LCHC	CH-C WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
64	0.0	1.1E-14	GEL-CBI-FF-LCHD	CH-D WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHB	CH-B WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
65	0.0	8.3E-15	GEL-CPL-CF-TML2-3	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTER (LEVEL T&M)	1.2E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
66	0.0	6.3E-15	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-CF-TMP2-3	CCF SPECIFIC 2 OR MORE PRES SENSOR/TRANSMITTER (PRES T&M)	6.4E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
67	0.0	4.5E-15	GEL-CBI-CF-TML3-7	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (LEVEL T&M)	4.2E-6
			GEL-MSW-CF-MSSAB	CCF MANUAL SCRAM SWITCH A AND B	7.7E-7
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
68	0.0	4.5E-15	GEL-CBI-CF-TMP3-7	CCF SPECIFIC 3 OR MORE CHANNEL TRIP UNITS (PRES T&M)	4.2E-6
			GEL-MSW-CF-MSSAB	CCF MANUAL SCRAM SWITCH A AND B	7.7E-7
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
69	0.0	4.5E-15	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			GEL-MSW-FF-MSSA	MANUAL SCRAM SWITCH A FAILS	1.3E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
70	0.0	4.5E-15	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			GEL-MSW-FF-MSSB	MANUAL SCRAM SWITCH B FAILS	1.3E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
71	0.0	4.1E-15	GEL-CBI-FF-LCHA	CH-A WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-LCHC	CH-C WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
72	0.0	4.1E-15	GEL-CBI-FF-LCHB	CH-B WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-LCHD	CH-D WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
73	0.0	4.0E-15	GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5C	CH-C PRESSURE RELAY K5C FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
74	0.0	4.0E-15	GEL-CBI-FF-PCHB	CH-B PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5D	CH-D PRESSURE RELAY K5D FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
75	0.0	4.0E-15	GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5A	CH-A PRESSURE RELAY K5A FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
76	0.0	4.0E-15	GEL-CBI-FF-PCHD	CH-D PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5B	CH-B PRESSURE RELAY K5B FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
77	0.0	2.4E-15	GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
78	0.0	2.3E-15	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHA	CH-A PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			GEL-CPR-FF-PCHC	CH-C PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
79	0.0	2.3E-15	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHB	CH-B PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			GEL-CPR-FF-PCHD	CH-D PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
80	0.0	1.3E-15	GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6C	CH-C WATER LEVEL RELAY K6C FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
81	0.0	1.1E-15	GEL-CBI-CF-TU4-8	CCF SPECIFIC 4 OR MORE TRIP UNITS	3.1E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K15A	CH-A MANUAL SCRAM SWITCH RELAY K15A FAILS	1.9E-5
			GEL-TLR-FF-K15C	CH-C MANUAL SCRAM SWITCH RELAY K15C FAILS	1.9E-5
82	0.0	1.1E-15	GEL-CBI-CF-TU4-8	CCF SPECIFIC 4 OR MORE TRIP UNITS	3.1E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K15B	CH-B MANUAL SCRAM SWITCH RELAY K15B FAILS	1.9E-5
			GEL-TLR-FF-K15D	CH-D MANUAL SCRAM SWITCH RELAY K15D FAILS	1.9E-5
83	0.0	9.0E-16	GEL-CBI-FF-LCHA	CH-A WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
84	0.0	9.0E-16	GEL-CBI-FF-LCHC	CH-C WATER LEVEL TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
85	0.0	9.0E-16	GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.4E-3
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
86	0.0	7.9E-16	GEL-PWR-CF-PWRAB	CCF 125 VDC POWER (SOV A AND SOV B)	2.1E-6
			GEL-TLR-FF-K14A	TRIP SYSTEM A RELAY K14A FAILS	1.9E-5
			GEL-TLR-FF-K14C	TRIP SYSTEM A RELAY K14C FAILS	1.9E-5
87	0.0	7.9E-16	GEL-PWR-CF-PWRAB	CCF 125 VDC POWER (SOV A AND SOV B)	2.1E-6
			GEL-TLR-FF-K14B	TRIP SYSTEM B RELAY K14B FAILS	1.9E-5
			GEL-TLR-FF-K14D	TRIP SYSTEM B RELAY K14D FAILS	1.9E-5
88	0.0	7.9E-16	GEL-PWR-CF-PWRAB	CCF 125 VDC POWER (SOV A AND SOV B)	2.1E-6
			GEL-TLR-FF-K14E	TRIP SYSTEM A RELAY K14E FAILS	1.9E-5
			GEL-TLR-FF-K14G	TRIP SYSTEM A RELAY K14G FAILS	1.9E-5
89	0.0	7.9E-16	GEL-PWR-CF-PWRAB	CCF 125 VDC POWER (SOV A AND SOV B)	2.1E-6
			GEL-TLR-FF-K14F	TRIP SYSTEM B RELAY K14F FAILS	1.9E-5
			GEL-TLR-FF-K14H	TRIP SYSTEM B RELAY K14H FAILS	1.9E-5
90	0.0	7.8E-16	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHA	CH-A PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5C	CH-C PRESSURE RELAY K5C FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
91	0.0	7.8E-16	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHB	CH-B PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5D	CH-D PRESSURE RELAY K5D FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
92	0.0	7.8E-16	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHC	CH-C PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5A	CH-A PRESSURE RELAY K5A FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
93	0.0	7.8E-16	GEL-CPL-CF-L2-4	CCF SPECIFIC 2 OR MORE LEVEL SENSOR/TRANSMITTERS	7.1E-5
			GEL-CPR-FF-PCHD	CH-D PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5

Cut Set	Cut Set Percent	Cut Set Prob.	Basic Event[a]	Description	Prob.
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K5B	CH-B PRESSURE RELAY K5B FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
94	0.0	7.2E-16	GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6C	CH-C WATER LEVEL RELAY K6C FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
95	0.0	7.2E-16	GEL-CPL-FF-LCHB	CH-B WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6D	CH-D WATER LEVEL RELAY K6D FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
96	0.0	7.2E-16	GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6A	CH-A WATER LEVEL RELAY K6A FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
97	0.0	7.2E-16	GEL-CPL-FF-LCHD	CH-D WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-CF-P2-4	CCF SPECIFIC 2 OR MORE PRESSURE SENSOR/TRANSMITTERS	4.9E-6
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-TLR-FF-K6B	CH-B WATER LEVEL RELAY K6B FAILS	1.9E-5
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
98	0.0	5.0E-16	GEL-CBI-FF-PCHA	CH-A PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHC	CH-C PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
99	0.0	5.0E-16	GEL-CBI-FF-PCHB	CH-B PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CBI-FF-PCHD	CH-D PRESSURE TRIP UNIT FAILS	2.9E-4
			GEL-CPL-FF-LCHB	CH-B WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPL-FF-LCHD	CH-D WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			/GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.0E+0
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2
100	0.0	4.8E-16	GEL-CPL-FF-LCHA	CH-A WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPL-FF-LCHC	CH-C WATER LEVEL SENSOR/TRANSMITTER FAILS	7.7E-4
			GEL-CPR-FF-PCHC	CH-C PRESSURE SENSOR/TRANSMITTER FAILS	5.7E-5
			/GEL-RPS-TM-ALVL	T&M CH-A REACTOR LOW WATER LEVEL TRIP SIGNAL	1.0E+0
			GEL-RPS-TM-APRES	T&M CH-A REACTOR HIGH PRESSURE TRIP SIGNAL	1.4E-3
			GEL-XHE-XE-SCRAM	OPERATOR FAILS TO INITIATE SCRAM	1.0E-2

a. A / as the first character in a basic event name indicates a complemented event (Success = 1 - Failure). For example, the basic event for reactor low water level trip signal channel A in test and maintenance (T&M) is GEL-RPS-TM-ALVL (Failure = 1.40E-03). Thus, the basic event name for reactor low water level trip signal channel A not in T&M is /GEL-RPS-TM-ALVL (Success = 9.986E-01). The event description for complemented events remains the same as the description used for the failure event.

Table F-7. Importance measures sorted on Fussell-Vesely for case with RPS mincut = 2.6E-06.

Basic Event	Number of Occur.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
GEL-SOV-CF-PSOVS	1	1.69E-6	6.443E-01	2.812	3.813E+5	1.000E+00
GEL-TLR-CF-TRP4-8	1	3.80E-7	1.449E-01	1.169	3.813E+5	1.000E+00
GEL-ROD-CF-CRD	1	2.50E-7	9.532E-02	1.105	3.813E+5	1.000E+00
GEL-TLR-CF-CHACBD	1	1.12E-7	4.258E-02	1.044	3.802E+5	9.972E-01
GEL-ACC-CF-HCU	1	1.09E-7	4.156E-02	1.043	3.813E+5	1.000E+00
GEL-SDL-CF-HWL2-4	2	3.09E-5	1.569E-02	1.016	5.087E+2	1.332E-03
GEL-XHE-XE-SCRAM	3648	1.00E-2	1.172E-02	1.012	2.160E+0	3.073E-06
GEL-CBI-CF-TU4-8	50	3.07E-6	1.170E-02	1.012	3.813E+3	9.998E-03
GEL-SDV-HL-WTRL2	10	6.66E-4	8.395E-03	1.008	1.360E+1	3.306E-05
GEL-SDV-HL-WTRL1	10	6.66E-4	8.395E-03	1.008	1.360E+1	3.306E-05
GEL-AOV-CF-HCU	1	6.94E-9	2.646E-03	1.003	3.813E+5	1.000E+00
GEL-TLR-CF-K1-2-4	2	1.36E-6	6.907E-04	1.001	5.087E+2	1.332E-03
GEL-SDL-FC-LDMB	4	6.13E-4	1.968E-04	1.000	1.321E+0	8.422E-07
GEL-SDL-FC-LCMB	4	6.13E-4	1.968E-04	1.000	1.321E+0	8.422E-07
GEL-SDL-FC-LBMA	4	6.13E-4	1.968E-04	1.000	1.321E+0	8.422E-07
GEL-SDL-FC-LAMA	4	6.13E-4	1.968E-04	1.000	1.321E+0	8.422E-07
GEL-TLR-CF-TM-PR	1	1.34E-7	7.142E-05	1.000	5.340E+2	1.398E-03
GEL-TLR-CF-TM-LV	1	1.34E-7	7.142E-05	1.000	5.340E+2	1.398E-03
GEL-CBI-CF-TMP3-7	50	4.15E-6	2.218E-05	1.000	6.344E+0	1.402E-05
GEL-CBI-CF-TML3-7	50	4.15E-6	2.218E-05	1.000	6.344E+0	1.402E-05
GEL-RPS-TM-APRES	18747	1.40E-3	1.751E-05	1.000	1.012E+0	3.280E-08
GEL-RPS-TM-ALVL	18747	1.40E-3	1.740E-05	1.000	1.012E+0	3.259E-08
GEL-MSW-FF-MSSB	4833	1.30E-5	1.523E-05	1.000	2.172E+0	3.073E-06
GEL-MSW-FF-MSSA	4389	1.30E-5	1.523E-05	1.000	2.172E+0	3.073E-06
GEL-TLR-FF-K1D	4	1.93E-5	6.197E-06	1.000	1.321E+0	8.422E-07
GEL-TLR-FF-K1C	4	1.93E-5	6.197E-06	1.000	1.321E+0	8.422E-07
GEL-TLR-FF-K1B	4	1.93E-5	6.197E-06	1.000	1.321E+0	8.422E-07
GEL-TLR-FF-K1A	4	1.93E-5	6.197E-06	1.000	1.321E+0	8.422E-07
GEL-CPL-CF-L2-4	1474	7.10E-5	1.529E-06	1.000	1.022E+0	5.647E-08
GEL-CPR-CF-P2-4	1474	4.86E-6	1.390E-06	1.000	1.286E+0	7.501E-07
GEL-MSW-CF-MSSAB	3648	7.71E-7	9.034E-07	1.000	2.172E+0	3.073E-06
GEL-SOV-FF-SOVB	638	6.97E-4	1.980E-07	1.000	1.000E+0	7.457E-10
GEL-SOV-FF-SOVA	638	6.97E-4	1.980E-07	1.000	1.000E+0	7.457E-10
GEL-CBI-FF-PCHC	2725	2.89E-4	1.410E-07	1.000	1.000E+0	1.280E-09
GEL-TLR-FF-K14H	3971	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14G	2969	1.93E-5	1.078E-07	1.000	1.006E+0	1.467E-08
GEL-TLR-FF-K14F	3971	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14E	4625	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14D	3971	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14C	2969	1.93E-5	1.078E-07	1.000	1.006E+0	1.467E-08
GEL-TLR-FF-K14B	3971	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14A	4625	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-CPL-FF-LCHC	2725	7.72E-4	3.801E-08	1.000	1.000E+0	1.295E-10
GEL-CBI-FF-PCHA	1356	2.89E-4	2.967E-08	1.000	1.000E+0	2.702E-10
GEL-CBI-FF-PCHD	2468	2.89E-4	2.908E-08	1.000	1.000E+0	2.647E-10
GEL-CBI-FF-PCHB	2468	2.89E-4	2.908E-08	1.000	1.000E+0	2.647E-10
GEL-CPR-FF-PCHC	2725	5.71E-5	2.785E-08	1.000	1.000E+0	1.280E-09
GEL-CPL-FF-LCHA	1356	7.72E-4	1.748E-08	1.000	1.000E+0	6.002E-11
GEL-PWR-FF-SOVB	638	6.00E-5	1.702E-08	1.000	1.000E+0	7.457E-10
GEL-PWR-FF-SOVA	638	6.00E-5	1.702E-08	1.000	1.000E+0	7.457E-10
GEL-CPL-FF-LCHD	2468	7.72E-4	1.575E-08	1.000	1.000E+0	5.415E-11
GEL-CPL-FF-LCHB	2468	7.72E-4	1.575E-08	1.000	1.000E+0	5.415E-11
GEL-CBI-FF-LCHC	2725	2.89E-4	1.418E-08	1.000	1.000E+0	1.295E-10
GEL-TLR-FF-K5C	2725	1.93E-5	9.397E-09	1.000	1.000E+0	1.280E-09

Basic Event	Number of Occur.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
GEL-CBI-FF-LCHA	1356	2.89E-4	6.476E-09	1.000	1.000E+0	6.002E-11
GEL-CBI-FF-LCHD	2468	2.89E-4	5.841E-09	1.000	1.000E+0	5.415E-11
GEL-CBI-FF-LCHB	2468	2.89E-4	5.841E-09	1.000	1.000E+0	5.415E-11
GEL-CPR-FF-PCHA	1356	5.71E-5	5.799E-09	1.000	1.000E+0	2.702E-10
GEL-CPR-FF-PCHD	2468	5.71E-5	5.672E-09	1.000	1.000E+0	2.647E-10
GEL-CPR-FF-PCHB	2468	5.71E-5	5.672E-09	1.000	1.000E+0	2.647E-10
GEL-CPL-CF-TML2-3	869	1.22E-4	3.259E-09	1.000	1.000E+0	7.187E-11
GEL-CPR-CF-TMP2-3	869	6.35E-6	2.413E-09	1.000	1.000E+0	1.028E-09
GEL-TLR-FF-K5A	1356	1.93E-5	1.947E-09	1.000	1.000E+0	2.702E-10
GEL-TLR-FF-K5D	2468	1.93E-5	1.905E-09	1.000	1.000E+0	2.647E-10
GEL-TLR-FF-K5B	2468	1.93E-5	1.905E-09	1.000	1.000E+0	2.647E-10
GEL-PWR-CF-PWRAB	1276	2.13E-6	1.185E-09	1.000	1.001E+0	1.491E-09
GEL-TLR-FF-K6C	2725	1.93E-5	9.312E-10	1.000	1.000E+0	1.295E-10
GEL-TLR-FF-K6A	1356	1.93E-5	4.233E-10	1.000	1.000E+0	6.002E-11
GEL-TLR-FF-K15D	2646	1.93E-5	4.233E-10	1.000	1.000E+0	5.942E-11
GEL-TLR-FF-K15C	3042	1.93E-5	4.233E-10	1.000	1.000E+0	5.942E-11
GEL-TLR-FF-K15B	2646	1.93E-5	4.233E-10	1.000	1.000E+0	5.942E-11
GEL-TLR-FF-K15A	1866	1.93E-5	4.233E-10	1.000	1.000E+0	5.948E-11
GEL-TLR-FF-K6D	2468	1.93E-5	3.810E-10	1.000	1.000E+0	5.415E-11
GEL-TLR-FF-K6B	2468	1.93E-5	3.810E-10	1.000	1.000E+0	5.415E-11

Table F-8. Importance measures sorted on Risk Increase for case with RPS mincut = 2.6E-06.

Basic Event	Number of Occur.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
GEL-TLR-CF-TRP4-8	1	3.80E-7	1.449E-01	1.169	3.813E+5	1.000E+00
GEL-SOV-CF-PSOVS	1	1.69E-6	6.443E-01	2.812	3.813E+5	1.000E+00
GEL-ROD-CF-CRD	1	2.50E-7	9.532E-02	1.105	3.813E+5	1.000E+00
GEL-AOV-CF-HCU	1	6.94E-9	2.646E-03	1.003	3.813E+5	1.000E+00
GEL-ACC-CF-HCU	1	1.09E-7	4.156E-02	1.043	3.813E+5	1.000E+00
GEL-TLR-CF-CHACBD	1	1.12E-7	4.258E-02	1.044	3.802E+5	9.972E-01
GEL-CBI-CF-TU4-8	50	3.07E-6	1.170E-02	1.012	3.813E+3	9.998E-03
GEL-TLR-CF-TM-PR	1	1.34E-7	7.142E-05	1.000	5.340E+2	1.398E-03
GEL-TLR-CF-TM-LV	1	1.34E-7	7.142E-05	1.000	5.340E+2	1.398E-03
GEL-TLR-CF-K1-2-4	2	1.36E-6	6.907E-04	1.001	5.087E+2	1.332E-03
GEL-SDL-CF-HWL2-4	2	3.09E-5	1.569E-02	1.016	5.087E+2	1.332E-03
GEL-SDV-HL-WTRL2	10	6.66E-4	8.395E-03	1.008	1.360E+1	3.306E-05
GEL-SDV-HL-WTRL1	10	6.66E-4	8.395E-03	1.008	1.360E+1	3.306E-05
GEL-CBI-CF-TMP3-7	50	4.15E-6	2.218E-05	1.000	6.344E+0	1.402E-05
GEL-CBI-CF-TML3-7	50	4.15E-6	2.218E-05	1.000	6.344E+0	1.402E-05
GEL-MSW-FF-MSSB	4833	1.30E-5	1.523E-05	1.000	2.172E+0	3.073E-06
GEL-MSW-FF-MSSA	4389	1.30E-5	1.523E-05	1.000	2.172E+0	3.073E-06
GEL-MSW-CF-MSSAB	3648	7.71E-7	9.034E-07	1.000	2.172E+0	3.073E-06
GEL-XHE-XE-SCRAM	3648	1.00E-2	1.172E-02	1.012	2.160E+0	3.073E-06
GEL-TLR-FF-K1D	4	1.93E-5	6.197E-06	1.000	1.321E+0	8.422E-07
GEL-TLR-FF-K1C	4	1.93E-5	6.197E-06	1.000	1.321E+0	8.422E-07
GEL-TLR-FF-K1B	4	1.93E-5	6.197E-06	1.000	1.321E+0	8.422E-07
GEL-TLR-FF-K1A	4	1.93E-5	6.197E-06	1.000	1.321E+0	8.422E-07
GEL-SDL-FC-LDMB	4	6.13E-4	1.968E-04	1.000	1.321E+0	8.422E-07
GEL-SDL-FC-LCMB	4	6.13E-4	1.968E-04	1.000	1.321E+0	8.422E-07
GEL-SDL-FC-LBMA	4	6.13E-4	1.968E-04	1.000	1.321E+0	8.422E-07
GEL-SDL-FC-LAMA	4	6.13E-4	1.968E-04	1.000	1.321E+0	8.422E-07
GEL-CPR-CF-P2-4	1474	4.86E-6	1.390E-06	1.000	1.286E+0	7.501E-07
GEL-CPL-CF-L2-4	1474	7.10E-5	1.529E-06	1.000	1.022E+0	5.647E-08
GEL-RPS-TM-APRES	18747	1.40E-3	1.751E-05	1.000	1.012E+0	3.280E-08
GEL-RPS-TM-ALVL	18747	1.40E-3	1.740E-05	1.000	1.012E+0	3.259E-08
GEL-TLR-FF-K14H	3971	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14G	2969	1.93E-5	1.078E-07	1.000	1.006E+0	1.467E-08
GEL-TLR-FF-K14F	3971	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14E	4625	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14D	3971	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14C	2969	1.93E-5	1.078E-07	1.000	1.006E+0	1.467E-08
GEL-TLR-FF-K14B	3971	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14A	4625	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-PWR-CF-PWRAB	1276	2.13E-6	1.185E-09	1.000	1.001E+0	1.491E-09
GEL-TLR-FF-K6D	2468	1.93E-5	3.810E-10	1.000	1.000E+0	5.415E-11
GEL-TLR-FF-K6C	2725	1.93E-5	9.312E-10	1.000	1.000E+0	1.295E-10
GEL-TLR-FF-K6B	2468	1.93E-5	3.810E-10	1.000	1.000E+0	5.415E-11
GEL-TLR-FF-K6A	1356	1.93E-5	4.233E-10	1.000	1.000E+0	6.002E-11
GEL-TLR-FF-K5D	2468	1.93E-5	1.905E-09	1.000	1.000E+0	2.647E-10
GEL-TLR-FF-K5C	2725	1.93E-5	9.397E-09	1.000	1.000E+0	1.280E-09
GEL-TLR-FF-K5B	2468	1.93E-5	1.905E-09	1.000	1.000E+0	2.647E-10
GEL-TLR-FF-K5A	1356	1.93E-5	1.947E-09	1.000	1.000E+0	2.702E-10
GEL-TLR-FF-K15D	2646	1.93E-5	4.233E-10	1.000	1.000E+0	5.942E-11
GEL-TLR-FF-K15C	3042	1.93E-5	4.233E-10	1.000	1.000E+0	5.942E-11
GEL-TLR-FF-K15B	2646	1.93E-5	4.233E-10	1.000	1.000E+0	5.942E-11
GEL-TLR-FF-K15A	1866	1.93E-5	4.233E-10	1.000	1.000E+0	5.948E-11
GEL-SOV-FF-SOVB	638	6.97E-4	1.980E-07	1.000	1.000E+0	7.457E-10
GEL-SOV-FF-SOVA	638	6.97E-4	1.980E-07	1.000	1.000E+0	7.457E-10

Basic Event	Number of Occur.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
GEL-PWR-FF-SOVB	638	6.00E-5	1.702E-08	1.000	1.000E+0	7.457E-10
GEL-PWR-FF-SOVA	638	6.00E-5	1.702E-08	1.000	1.000E+0	7.457E-10
GEL-CPR-FF-PCHD	2468	5.71E-5	5.672E-09	1.000	1.000E+0	2.647E-10
GEL-CPR-FF-PCHC	2725	5.71E-5	2.785E-08	1.000	1.000E+0	1.280E-09
GEL-CPR-FF-PCHB	2468	5.71E-5	5.672E-09	1.000	1.000E+0	2.647E-10
GEL-CPR-FF-PCHA	1356	5.71E-5	5.799E-09	1.000	1.000E+0	2.702E-10
GEL-CPR-CF-TMP2-3	869	6.35E-6	2.413E-09	1.000	1.000E+0	1.028E-09
GEL-CPL-FF-LCHD	2468	7.72E-4	1.575E-08	1.000	1.000E+0	5.415E-11
GEL-CPL-FF-LCHC	2725	7.72E-4	3.801E-08	1.000	1.000E+0	1.295E-10
GEL-CPL-FF-LCHB	2468	7.72E-4	1.575E-08	1.000	1.000E+0	5.415E-11
GEL-CPL-FF-LCHA	1356	7.72E-4	1.748E-08	1.000	1.000E+0	6.002E-11
GEL-CPL-CF-TML2-3	869	1.22E-4	3.259E-09	1.000	1.000E+0	7.187E-11
GEL-CBI-FF-PCHD	2468	2.89E-4	2.908E-08	1.000	1.000E+0	2.647E-10
GEL-CBI-FF-PCHC	2725	2.89E-4	1.410E-07	1.000	1.000E+0	1.280E-09
GEL-CBI-FF-PCHB	2468	2.89E-4	2.908E-08	1.000	1.000E+0	2.647E-10
GEL-CBI-FF-PCHA	1356	2.89E-4	2.967E-08	1.000	1.000E+0	2.702E-10
GEL-CBI-FF-LCHD	2468	2.89E-4	5.841E-09	1.000	1.000E+0	5.415E-11
GEL-CBI-FF-LCHC	2725	2.89E-4	1.418E-08	1.000	1.000E+0	1.295E-10
GEL-CBI-FF-LCHB	2468	2.89E-4	5.841E-09	1.000	1.000E+0	5.415E-11
GEL-CBI-FF-LCHA	1356	2.89E-4	6.476E-09	1.000	1.000E+0	6.002E-11

Table F-9. Importance measures sorted on Birnbaum for case with RPS mincut = 2.6E-06.

Basic Event	Number of Occur.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
GEL-TLR-CF-TRP4-8	1	3.80E-7	1.449E-01	1.169	3.813E+5	1.000E+00
GEL-SOV-CF-PSOVS	1	1.69E-6	6.443E-01	2.812	3.813E+5	1.000E+00
GEL-ROD-CF-CRD	1	2.50E-7	9.532E-02	1.105	3.813E+5	1.000E+00
GEL-AOV-CF-HCU	1	6.94E-9	2.646E-03	1.003	3.813E+5	1.000E+00
GEL-ACC-CF-HCU	1	1.09E-7	4.156E-02	1.043	3.813E+5	1.000E+00
GEL-TLR-CF-CHACBD	1	1.12E-7	4.258E-02	1.044	3.802E+5	9.972E-01
GEL-CBI-CF-TU4-8	50	3.07E-6	1.170E-02	1.012	3.813E+3	9.998E-03
GEL-TLR-CF-TM-PR	1	1.34E-7	7.142E-05	1.000	5.340E+2	1.398E-03
GEL-TLR-CF-TM-LV	1	1.34E-7	7.142E-05	1.000	5.340E+2	1.398E-03
GEL-TLR-CF-K1-2-4	2	1.36E-6	6.907E-04	1.001	5.087E+2	1.332E-03
GEL-SDL-CF-HWL2-4	2	3.09E-5	1.569E-02	1.016	5.087E+2	1.332E-03
GEL-SDV-HL-WTRL2	10	6.66E-4	8.395E-03	1.008	1.360E+1	3.306E-05
GEL-SDV-HL-WTRL1	10	6.66E-4	8.395E-03	1.008	1.360E+1	3.306E-05
GEL-CBI-CF-TMP3-7	50	4.15E-6	2.218E-05	1.000	6.344E+0	1.402E-05
GEL-CBI-CF-TML3-7	50	4.15E-6	2.218E-05	1.000	6.344E+0	1.402E-05
GEL-XHE-XE-SCRAM	3648	1.00E-2	1.172E-02	1.012	2.160E+0	3.073E-06
GEL-MSW-FF-MSSB	4833	1.30E-5	1.523E-05	1.000	2.172E+0	3.073E-06
GEL-MSW-FF-MSSA	4389	1.30E-5	1.523E-05	1.000	2.172E+0	3.073E-06
GEL-MSW-CF-MSSAB	3648	7.71E-7	9.034E-07	1.000	2.172E+0	3.073E-06
GEL-TLR-FF-K1D	4	1.93E-5	6.197E-06	1.000	1.321E+0	8.422E-07
GEL-TLR-FF-K1C	4	1.93E-5	6.197E-06	1.000	1.321E+0	8.422E-07
GEL-TLR-FF-K1B	4	1.93E-5	6.197E-06	1.000	1.321E+0	8.422E-07
GEL-TLR-FF-K1A	4	1.93E-5	6.197E-06	1.000	1.321E+0	8.422E-07
GEL-SDL-FC-LDMB	4	6.13E-4	1.968E-04	1.000	1.321E+0	8.422E-07
GEL-SDL-FC-LCMB	4	6.13E-4	1.968E-04	1.000	1.321E+0	8.422E-07
GEL-SDL-FC-LBMA	4	6.13E-4	1.968E-04	1.000	1.321E+0	8.422E-07
GEL-SDL-FC-LAMA	4	6.13E-4	1.968E-04	1.000	1.321E+0	8.422E-07
GEL-CPR-CF-P2-4	1474	4.86E-6	1.390E-06	1.000	1.286E+0	7.501E-07
GEL-CPL-CF-L2-4	1474	7.10E-5	1.529E-06	1.000	1.022E+0	5.647E-08
GEL-RPS-TM-APRES	18747	1.40E-3	1.751E-05	1.000	1.012E+0	3.280E-08
GEL-RPS-TM-ALVL	18747	1.40E-3	1.740E-05	1.000	1.012E+0	3.259E-08
GEL-TLR-FF-K14G	2969	1.93E-5	1.078E-07	1.000	1.006E+0	1.467E-08
GEL-TLR-FF-K14C	2969	1.93E-5	1.078E-07	1.000	1.006E+0	1.467E-08
GEL-TLR-FF-K14H	3971	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14F	3971	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14E	4625	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14D	3971	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14B	3971	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-TLR-FF-K14A	4625	1.93E-5	1.078E-07	1.000	1.006E+0	1.465E-08
GEL-PWR-CF-PWRAB	1276	2.13E-6	1.185E-09	1.000	1.001E+0	1.491E-09
GEL-TLR-FF-K5C	2725	1.93E-5	9.397E-09	1.000	1.000E+0	1.280E-09
GEL-CPR-FF-PCHC	2725	5.71E-5	2.785E-08	1.000	1.000E+0	1.280E-09
GEL-CBI-FF-PCHC	2725	2.89E-4	1.410E-07	1.000	1.000E+0	1.280E-09
GEL-CPR-CF-TMP2-3	869	6.35E-6	2.413E-09	1.000	1.000E+0	1.028E-09
GEL-SOV-FF-SOVB	638	6.97E-4	1.980E-07	1.000	1.000E+0	7.457E-10
GEL-SOV-FF-SOVA	638	6.97E-4	1.980E-07	1.000	1.000E+0	7.457E-10
GEL-PWR-FF-SOVB	638	6.00E-5	1.702E-08	1.000	1.000E+0	7.457E-10
GEL-PWR-FF-SOVA	638	6.00E-5	1.702E-08	1.000	1.000E+0	7.457E-10
GEL-TLR-FF-K5A	1356	1.93E-5	1.947E-09	1.000	1.000E+0	2.702E-10
GEL-CPR-FF-PCHA	1356	5.71E-5	5.799E-09	1.000	1.000E+0	2.702E-10
GEL-CBI-FF-PCHA	1356	2.89E-4	2.967E-08	1.000	1.000E+0	2.702E-10
GEL-TLR-FF-K5D	2468	1.93E-5	1.905E-09	1.000	1.000E+0	2.647E-10
GEL-TLR-FF-K5B	2468	1.93E-5	1.905E-09	1.000	1.000E+0	2.647E-10
GEL-CPR-FF-PCHD	2468	5.71E-5	5.672E-09	1.000	1.000E+0	2.647E-10

Basic Event	Number of Occur.	Prob. of Failure	Fussell-Vesely Importance	Risk Reduction Ratio	Risk Increase Ratio	Birnbaum Importance
GEL-CPR-FF-PCHB	2468	5.71E-5	5.672E-09	1.000	1.000E+0	2.647E-10
GEL-CBI-FF-PCHD	2468	2.89E-4	2.908E-08	1.000	1.000E+0	2.647E-10
GEL-CBI-FF-PCHB	2468	2.89E-4	2.908E-08	1.000	1.000E+0	2.647E-10
GEL-TLR-FF-K6C	2725	1.93E-5	9.312E-10	1.000	1.000E+0	1.295E-10
GEL-CPL-FF-LCHC	2725	7.72E-4	3.801E-08	1.000	1.000E+0	1.295E-10
GEL-CBI-FF-LCHC	2725	2.89E-4	1.418E-08	1.000	1.000E+0	1.295E-10
GEL-CPL-CF-TML2-3	869	1.22E-4	3.259E-09	1.000	1.000E+0	7.187E-11
GEL-TLR-FF-K6A	1356	1.93E-5	4.233E-10	1.000	1.000E+0	6.002E-11
GEL-CPL-FF-LCHA	1356	7.72E-4	1.748E-08	1.000	1.000E+0	6.002E-11
GEL-CBI-FF-LCHA	1356	2.89E-4	6.476E-09	1.000	1.000E+0	6.002E-11
GEL-TLR-FF-K15A	1866	1.93E-5	4.233E-10	1.000	1.000E+0	5.948E-11
GEL-TLR-FF-K15D	2646	1.93E-5	4.233E-10	1.000	1.000E+0	5.942E-11
GEL-TLR-FF-K15C	3042	1.93E-5	4.233E-10	1.000	1.000E+0	5.942E-11
GEL-TLR-FF-K15B	2646	1.93E-5	4.233E-10	1.000	1.000E+0	5.942E-11
GEL-TLR-FF-K6D	2468	1.93E-5	3.810E-10	1.000	1.000E+0	5.415E-11
GEL-TLR-FF-K6B	2468	1.93E-5	3.810E-10	1.000	1.000E+0	5.415E-11
GEL-CPL-FF-LCHD	2468	7.72E-4	1.575E-08	1.000	1.000E+0	5.415E-11
GEL-CPL-FF-LCHB	2468	7.72E-4	1.575E-08	1.000	1.000E+0	5.415E-11
GEL-CBI-FF-LCHD	2468	2.89E-4	5.841E-09	1.000	1.000E+0	5.415E-11
GEL-CBI-FF-LCHB	2468	2.89E-4	5.841E-09	1.000	1.000E+0	5.415E-11

Table F-10. Failure probability and uncertainty data changes for RPS mincut = 2.6E-6.

Basic Event Name	Prob.	Distr. Type	Uncert. Value[a]	Correlation Class	Basic Event Description
GEL-MSW-CF-MSSAB	7.71E-7	Lognormal	11.23	-	CCF MANUAL SCRAM SWITCH A AND B
GEL-MSW-FF-MSSA	1.30E-5	Lognormal	5.61	MSW1	MANUAL SCRAM SWITCH A FAILS
GEL-MSW-FF-MSSB	1.30E-5	Lognormal	5.61	MSW1	MANUAL SCRAM SWITCH B FAILS
GEL-TLR-CF-CHACBD	1.12E-7	Lognormal	9.55	-	CCF CHANNEL RELAYS (NO T&M)
GEL-TLR-CF-TM-LV	1.34E-7	Lognormal	8.96	TLR2	CCF CHANNEL RELAYS (LEVEL T&M)
GEL-TLR-CF-TM-PR	1.34E-7	Lognormal	8.96	TLR2	CCF CHANNEL RELAYS (PRES T&M)
GEL-TLR-FF-K15A	1.93E-5	Lognormal	6.11	TLR1	CH-A MANUAL SCRAM SWITCH RELAY K15A FAILS
GEL-TLR-FF-K15B	1.93E-5	Lognormal	6.11	TLR1	CH-B MANUAL SCRAM SWITCH RELAY K15B FAILS
GEL-TLR-FF-K15C	1.93E-5	Lognormal	6.11	TLR1	CH-C MANUAL SCRAM SWITCH RELAY K15C FAILS
GEL-TLR-FF-K15D	1.93E-5	Lognormal	6.11	TLR1	CH-D MANUAL SCRAM SWITCH RELAY K15D FAILS
GEL-XHE-XE-SCRAM	1.00E-2	Lognormal	10.00	-	OPERATOR FAILS TO INITIATE SCRAM

a. The uncertainty (Uncert.) value is the parameter that is used to describe the uncertainty distribution for the associated basic event. The lognormal and uniform distributions are the only two distributions used for the RPS basic events. The lognormal distribution is described by the mean and the upper 95% error factor. The uniform distribution is described by the mid point (mean probability) and the upper endpoint (Uncert. Value).

Appendix G

Sensitivity Analysis

Appendix G

Sensitivity Analysis

Sensitivity analyses of the General Electric Reactor Protection System (RPS) fault tree model and quantification were performed in two general areas: data analysis and success criteria. Under the data analysis area, two specific issues were addressed. The first involves the calculation of the common-cause failure (CCF) events. As explained in Appendix E, the CCF failure probabilities were quantified using priors generated directly from the overall set of General Electric CCF data. Then the priors were updated using CCF data specific to the components and failure modes modeled in the fault tree. CCF failure probabilities resulting from that process are presented in Table E-10 in Appendix E. The base case RPS unavailability of 5.8E-6 was generated using this approach to estimating CCF failure probabilities. Another approach to calculating the CCF events is to use no prior and use only the CCF data specific to the component and failure mode in question. This process is termed the classical approach to CCF parameter estimation. This approach is very sensitive to the exclusion or inclusion of specific CCF data for components with few or no CCF data. Also, such an approach predicts a zero probability of a CCF event if no data exist for the period 1984 through 1995. CCF probabilities obtained using the classical approach are presented in Table E-12 in Appendix E. The classical approach resulted in lower CCF probabilities for 18 events and higher CCF probabilities for 5 events, compared with the approach using priors and an updating process. Requantifying the RPS fault tree using the classical CCF probabilities resulted in a point estimate unavailability of 2.4E-5, compared with the base case result of 5.8E-6. The dominant contributor for the classical approach is the CCF of the hydraulic control unit (HCU) solenoid-operated valves (SOVs) and backup scram SOVs. This event contributes 98% to the total unavailability. However, the classical CCF results predict no contributions from the control rods (RODs), control rod drives (CRDs), and accumulators (ACCs). (These events have zero probability for the classical case.)

The other data analysis sensitivity covers the use of uncertain data. As explained in Section 2.3.1 of this report, failure event data were grouped into nine categories. One of these categories includes complete failures in the non-fail-safe direction (NFS/CF). Events in this category are clearly failures with respect to the RPS fault tree model. Three other categories (NFS/UC, UKN/CF, and UKN/UC) contain events that may be NFS/CF. However, because of incomplete information in the failure event description, the data analysts were not able to determine whether such events were actually NFS/CF. Therefore, in the data analysis (Appendices A and C of this report) the component failure probabilities were obtained using all of the NFS/CF events and a fraction of the events in the other three data categories. To estimate the impact on RPS unavailability from the use of these uncertain events, the component failure rates were estimated using only the NFS/CF events. (See Table C-1 in Appendix C of this report.) The resultant RPS unavailability is 2.0E-6, which is a 66% reduction from the base case value of 5.8E-6. Also, the RPS unavailability was recalculated assuming all of the uncertain data were NFS/CF. This resulted in an increase in the RPS unavailability to 9.9E-6, a 69% increase from the base case value of 5.8E-6. Therefore, the data associated with incomplete information in the failure event descriptions contribute approximately plus or minus 68% to the uncertainty in the overall RPS unavailability.

The other sensitivity cases are categorized as success criteria related. One sensitivity case involves quantifying the RPS fault tree with the backup scram portion deleted. With this deletion, the K14 relays require only specific combinations of two failures (out of eight) to fail the RPS, rather than specific combinations of four failures. Table E-10 in Appendix E indicates that this increases the K14 relay CCF event (GEL-TLR-CF-TRP4-8) failure probability from 3.8E-7 to 2.4E-6 (event GEL-TLR-CF-2OF8). This results in an increase in the RPS unavailability from the base case value of 5.8E-6 to 7.8E-6 (35% increase).

Another success criteria sensitivity case involves the accumulators (ACCs) and the scram inlet air-operated valves (AOVs). As long as the scram outlet AOV opens in an HCU, the control rod will still insert into the core even if the ACC and scram inlet AOV fail. (If this happens, the control rod is driven into the core by the reactor vessel water pressure. The insertion time is longer for this mode of operation.) If the ACC and scram inlet AOV are not required, then the RPS unavailability drops from 5.8E-6 to 5.7E-6 (2% decrease).

Only two trip signals were included in the RPS fault tree model: reactor vessel level and pressure. If three signals were included in the model, then two of the six CCF events dominating the RPS unavailability would be affected (GEL-CBI-CF-TU4-8 and GEL-TLR-CF-CHACBD). In both cases, the events would change from specific four of eight failures to specific six of 12 failures. Given these changes, the RPS unavailability drops from 5.8E-6 to approximately 3.8E-6 (34% decrease).

The RPS fault tree logic assumes that two of four rod group signals failing will fail the main scram portion of the overall RPS. Because of the K14 relay logic, there is no affect on the RPS cut sets and unavailability if the failure criterion were changed to one of four rod group signals failing. Also, because the backup scram is modeled as part of the RPS, there is no change in the cut sets or unavailability if the failure criterion were changed to three of four rod group signals failing. Therefore, the RPS results are insensitive to the rod group failure criterion.

The final success criteria sensitivity involves the assumption that 33% (or more) of the control rods must fail to insert. This failure criterion affects three of the six dominant CCF events (SOV, ROD, and ACC CCFs). Figure G-1 illustrates the change in CCF failure probability for the SOV and ROD CCF events over the range of 10% (or more) to 50% (or more). (The ACC event, not shown in the figure, is a minor contributor to RPS unavailability.) If the 33% failure criterion were changed to 10%, the RPS unavailability increases from 5.8E-6 to 8.7E-6 (49% increase). If the failure criterion were changed to 50%, the RPS unavailability decreases from 5.8E-6 to 4.5E-6 (23% decrease).

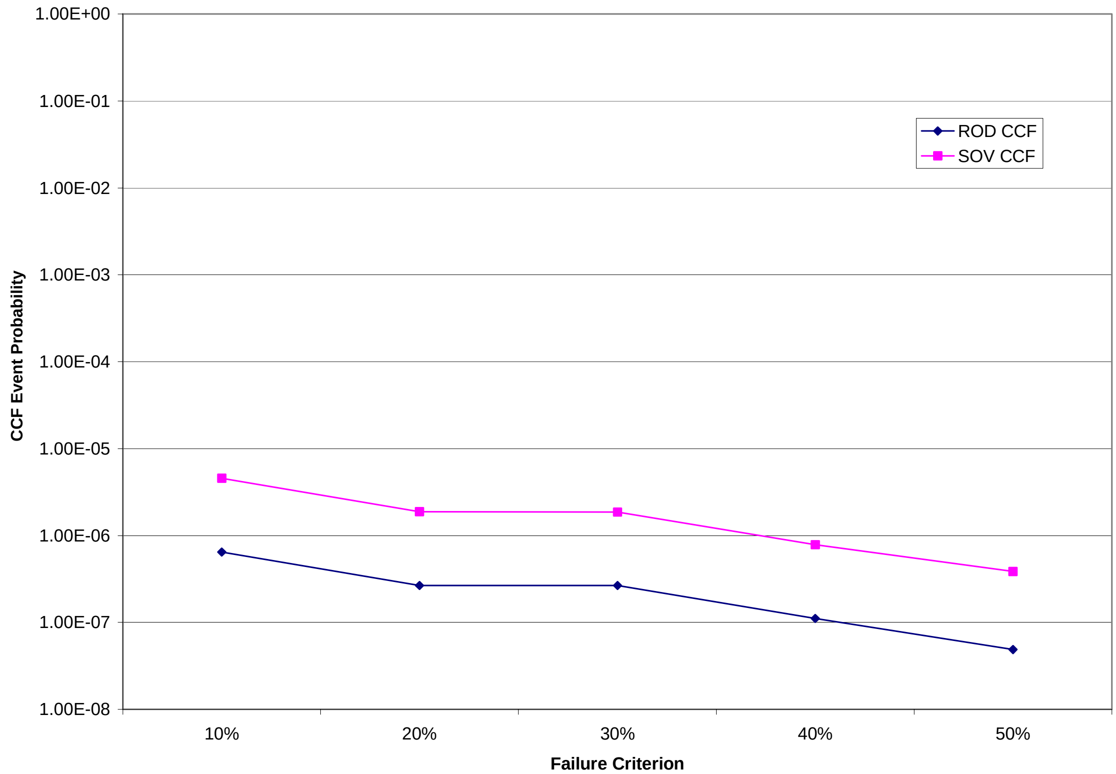


Figure G-1. Sensitivity of ROD and SOV CCF events to control rod failure criterion.

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11. ABSTRACT (200 words or less)
This report documents an analysis of the safety-related performance of the reactor protection system (RPS) at U.S. General Electric commercial reactors during the period 1984 through 1995. General Electric RPS designs analyzed in this report include those with relay-based trip systems. The analysis is based on a BWR/4 plant design. RPS operational data were collected for all U.S. General Electric commercial reactors from the Nuclear Plant Reliability Data System and Licensee Event Reports. A risk-based analysis was performed on the data to estimate the observed unavailability of the RPS, based on a fault tree model of the system. An engineering analysis of trends and patterns was also performed on the data to provide additional insights into RPS performance. RPS unavailability results obtained from the data were compared with existing unavailability estimates from Individual Plant Examinations and other reports.

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