IAEA-TECDOC-590

Case study on the use of PSA methods: Determining safety importance of systems and components at nuclear power plants

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CASE STUDY ON THE USE OF PSA METHODS:
DETERMINING SAFETY IMPORTANCE OF SYSTEMS AND COMPONENTS
AT NUCLEAR POWER PLANTS
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FOREWORD

Probabilistic Safety Assessment (PSA) is increasingly being used to complement the deterministic approach to nuclear safety. From the traditional discipline of reliability engineering, PSA developed as a structured method to identify potential accident sequences from a broad range of initiating events and to quantify their frequency of occurrence.

PSAs use inductive (event tree) and deductive (fault tree) logic and plant specific as well as generic component failure rates and frequencies of initiating events. Plant specific test and maintenance schedules, human errors and common cause failures are also considered in the probabilistic models.

PSA is nowadays a fundamental tool that provides guidance to safety related decision-making. By its very nature PSA recognizes the uncertainties associated with the logic models used to represent reality and quantifies the variability in the data of the parameters in the models.

The IAEA is promoting the conduct of PSA studies through standardization of the methodology, co-ordination of research, assistance through its Technical Co-operation Programme, and development of PSA software (PSAPACK). In addition it offers International Peer Review Services (IPERS) to review PSAs at various stages of completeness.

Emphasis at present is concentrated on "level-1" PSAs which quantify accident sequences up to estimates of core-damage probability. Level-2 (releases of radioactivity) and level-3 (off-site impacts) will be addressed at a later stage.

The work described above on the conduct of PSA is complemented by a programme on how to use the results of PSA in nuclear safety. For this purpose a series of CASE STUDIES has been prepared. The objective is to provide those who have performed PSAs with practical examples on how PSA results have been used. Those authorities and utilities still reluctant to request or perform PSAs will find convincing evidence on the benefits of such studies for nuclear safety.

With these objectives in mind, the IAEA requested a number of internationally recognized experts to document, in a uniform and suitable format, actual experience with the use of PSA for safety decisions. The documents were peer reviewed by an Oversight Committee for quality and completeness.

It is hoped that this series of CASE STUDIES will significantly contribute to the use of PSA to improve nuclear safety.

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PREFACE

A series of CASE STUDIES has been prepared to summarize practical examples on how the results of PSA studies have been used in nuclear safety. They draw from the experience of major studies and, to the extent possible, use a similar format to guide the reader. The studies illustrate the range of applications in a specific topical area. It is the objective to take examples which are using level-1 PSAs rather than individual accident sequences or systems reliability. Emphasis is given to a logical step-by-step description of the analysis and documentation of calculational procedures and data. The interpretation of the results explicitly addresses the problem of uncertainties and limitations of the studies, and includes the results of Peer Reviews.

This CASE STUDY addresses the ranking of safety significance of systems or components. There are many options to improve the safety level of a nuclear plant. Using the results of a PSA it is possible to quantify the contribution of systems and components to safety. This information can be used to rank the relative significance of planned plant modifications according to impact on core melt frequency and serious release frequency.

The purpose of this CASE STUDY is thus to systematically describe the methodology to rank safety significance and to demonstrate which type of results can be obtained using the example of a PSA for a particular plant.

The following additional Case Study documents are available:

IAEA-TECDOC-522	A Probabilistic Safety Assessment Peer Review: Case Study on the Use of Probabilistic Safety Assessment for Safety Decisions (1989)
IAEA-TECDOC-543	Procedures for Conducting Independent Peer Reviews of Probabilistic Safety Assessment (1990)
IAEA-TECDOC-547	The Use of Probabilistic Safety Assessment in the Relicensing of Nuclear Power Plants for Extended Lifetimes (1990)
IAEA-TECDOC-591	Case Study on the Use of PSA Methods: Backfitting Decisions (1991)
IAEA-TECDOC-592	Case Study on the Use of PSA Methods: Human Reliability Analysis (1991)
IAEA-TECDOC-593	Case Study on the Use of PSA Methods: Station Blackout Risk at the Millstone Unit 3 (1991)

CONTENTS

1. PROBLEM DEFINITION

A probabilistic risk or safety assessment, PSA, includes the following series of steps:

1. Identification of undesired events
2. System understanding
3. Logic model generation
4. Qualitative evaluation of the logic model
5. Data analysis
6. Quantitative or probabilistic evaluation of the logic model
7. Sensitivity or Importance analysis
8. Consequence analysis
9. Uncertainty analysis

Depending upon the scope and extent of the assessment, all or part of the above steps can be conducted. This paper emphasizes Step 7, identification of systems and components important to plant safety. An importance analysis combines the information given in Steps 1 through 6, i.e., importance analysis involves combining information that is both qualitative and probabilistic in nature. One purpose of an importance analysis is to generate a numerical ranking to determine the system and/or component failures that dominate the risk. Such a ranking can suggest where hardware, software, human factors and component design changes can be implemented to improve plant safety. Although this paper emphasizes nuclear power plant applications, the methodology described has been applied to other industries such as the chemical processing industry, Ref. (1).

The concept of importance or risk significance is applicable to nearly every other case study considered in this series. For example, an important element in implementing a backfit is the risk reduction resulting from the backfit and the cost of the backfit. In this paper, we consider both factors in formulating a risk reduction function. As described in Ref. (4), we can use measures of risk impacts of testing and maintenance activities for the technical specifications requirements. Also, the concept of risk significance can assist regulators in applying deterministic licensing criteria more objectively.

For example, the U.S. Nuclear Regulatory Commission, NRC, uses the same standard (IEEE-279) Ref. (2) for design of both the reactor protection system and the ECCS initiation system. From a risk viewpoint, it is obvious that the challenges to the reactor protection system are at least a factor of 1,000 greater than the challenges to the ECCS initiation system. Therefore,

probabilistically, one would expect the systems to have substantially different reliability needs assuming that both prevent similar consequence accidents (core melt). However, from an NRC deterministic licensing criteria standpoint, both systems could have similar reliability, i.e., that level supplied by the single failure criterion. This approach in establishing safety has certain major advantages such as ease in licensing review and litigation since showing compliance with most of the criteria is rather straightforward. However, certain technical issues are extremely difficult to resolve with this approach and many have been highlighted in the last several years. This is one of the major reasons that risk assessment is attracting substantial attention from both the NRC and the nuclear industry. Examples of these weaknesses are the inability of licensing criteria to deal with human involvement and the difficulty in judging the "significance" of deviations from these licensing criteria. Also, the system modeling techniques normally used in risk assessment are more powerful than the licensing criteria approach in dealing with multiple failures or interaction with various systems.

In probabilistic approaches, one does not necessarily limit the evaluation to a certain set of "deterministic" failures or accidents. The engineering judgement that provides most of the bases for licensing criteria is replaced by complete and logically derived descriptions of all hazards that can affect the nuclear power plant. This, of course, creates a separate set of problems, namely showing that all hazards are included. However, as a complement to the deterministic licensing approach, past risk assessment studies have uncovered many insights and, as discussed in the case study, can provide a ranking of important (and unimportant) risk areas.

The purpose of the study was twofold:

- o Develop a methodology for ranking the relative significance of planned plant modifications and design changes for the Grand Gulf Nuclear Power Plant (GGNS), and
- o Implement the methodology for GGNS and rank systems and components that are important to plant safety.

This methodology was developed in part from the concepts described in Ref. (1). The report entitled "Reactor Safety Study Methodology Application Program: Grand Gulf #1 BWR Plant", Ref. (3) (referred to as RSSMAP in this report) provided much of the plant specific information to probabilistically rank systems and components, e.g., information regarding the dominant accident sequences. GGNS is a BWR/6 with a Mark III containment.

In Section 4 we describe the methodology that was developed for the TENERA study in terms of a simplified example. In Section 5, we show how the methodology can be used for nuclear power plant applications and use the methodology to rank systems and components at GGNS according to core melt frequency and serious release frequency. In Section 5, we also discuss a qualitative ranking criteria for components and systems that are not included in a PSA.

2. OBJECTIVES

One objective of this case study is to show how the results of a nuclear power plant PSA can be used in ranking the importance of systems and components that are important to plant safety. It is important to note that a plant manager can use the results in this study without intimate knowledge of the PSA. In this study, we aggregate min cut sets, i.e., accident scenarios, according to one outcome: core melt. Core melt is the outcome typically considered in a level 1 PSA. We formulate a risk reduction function that determines the reduction in the (expected value) of release for a given design change. Radiological release is typically considered in a level 2 or level 3 PSA.

Another objective of this case study is to develop a methodology for formulating a risk reduction function for proposed design and procedural changes. As described in Ref. (5), it is important that we consider finite changes in risk and develop reliability expressions based on finite-difference functions. Also, as described in Ref. (1), we must distinguish between two types of events when considering accident causation, i.e., initiating events which cause the accident to occur and enabling events which represent failure of system mitigative features when the initiating event occurs. As discussed in this study, the risk difference expression is different for the two event types.

The case study was conducted within a short time period and with a limited budget. The objective of the case study was to take an existing PSA and use it to rank components and systems important to plant safety and availability. This ranking would assist the utility in prioritizing planned plant modifications and design changes. It is important to note for this case study that the results of the PSA would not replace the utility decision-making process but supplement it.

RSSMAP did not consider external events nor did RSSMAP conduct detailed human reliability assessments or uncertainty analyses. As a result, these issues were not considered in this case study. In addition, it is important to note that several systems considered in this case study were not analyzed in RSSMAP. Ranking of these systems was conducted by a qualitative ranking scheme described in section 5.9 of this paper.

One member of the study team (not an author of this paper) conducted a peer review of the RSSMAP study. This member was aware of the limitations of the RSSMAP study which are discussed in this paper. His input was important to the development of the qualitative ranking scheme.

3. OVERVIEW OF THE ANALYSIS

Figure 1 is a flowchart which gives an overview of the analysis. The starting point is to define the Top Event. For example, the Top Event can represent core melt which is the Boolean union of all accident sequences on the event tree resulting in core melt. The second step is to construct the logic model. In general, event trees define Top Events to fault trees. The

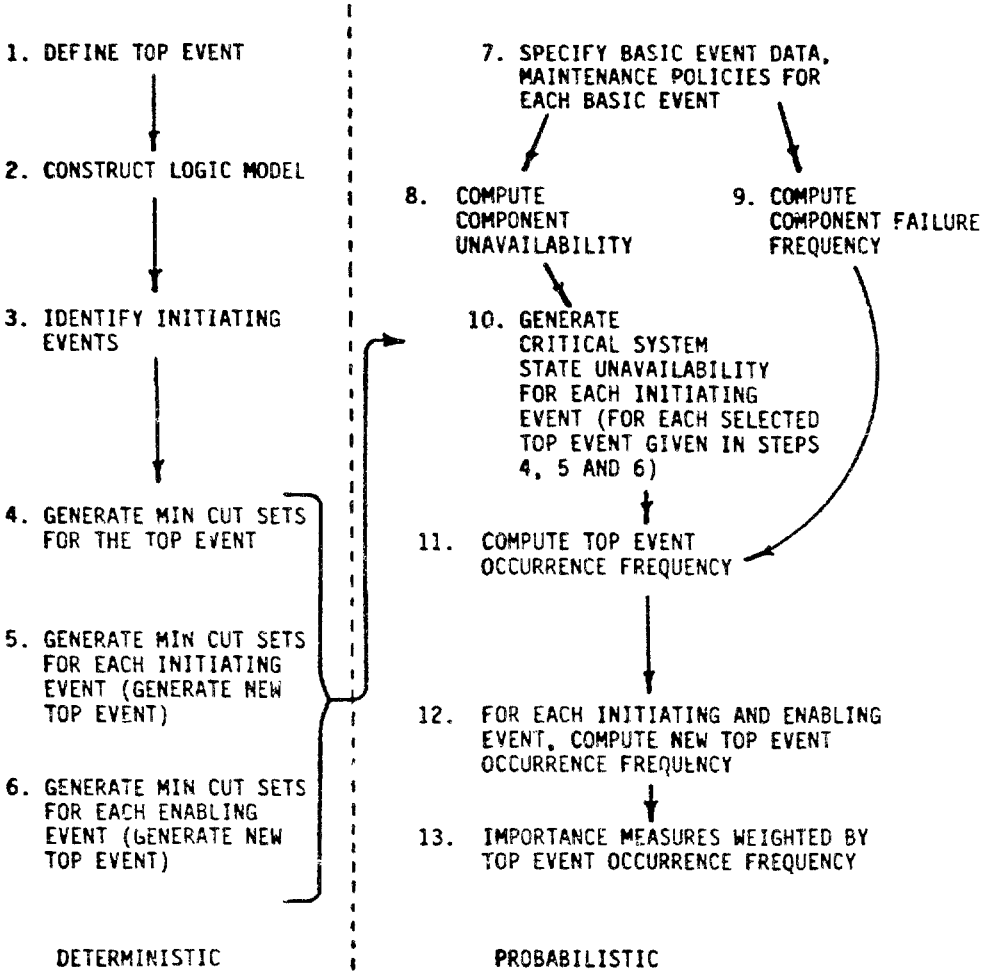


FIG. 1. Flowsheet for computational procedure.

fault trees are constructed with a Top Event in mind. The third step is to identify initiating events in the logic model. Commonly, initiating events in the event tree-fault tree analysis of nuclear reactor systems consist of events such as:

- o Transient events
- o LOCAs.

The fourth step is to find the min cut sets for the Top Event. In order to obtain a measure of participation for each initiating and enabling event, we take the Boolean union of all min cut sets containing each initiating and enabling event in the logic model. The Boolean union of all the min cut sets containing an initiating event, with the initiating event set equal to true, defines the critical system states for an initiating event. A critical system state for an initiating event defines a possible set of system mitigative features that must fail in order for the initiating event to cause the Top Event to occur.

Steps 1 through 6 entail qualitative or deterministic analyses. Steps 7 through 13 involve probabilistic calculations and require as input the min cut sets from Steps 4, 5, and 6.

In Step 7, we determine the following reliability data for the basic events:

- o Failure rates
- o Repair times
- o Human error probabilities
- o Test and maintenance frequencies.

The data from Step 7 is input for Steps 8 and 9. In Step 8, component unavailability for enabling events is computed. In Step 9, the failure frequency for initiating events is computed. Step 10 involves computing the critical system state unavailability for each initiating event.

Step 11 is the computation of the Top Event occurrence frequency which is the sum of the frequencies at which the initiating event causes the Top Event to occur. The Top Event occurrence frequency expression is a sum-of-products form, i.e., the product of the initiating event failure frequency and the critical system state unavailability. Steps 12 and 13 involve computing the failure frequency of a new Top Event which is the Boolean union of the min cut sets containing either the initiating or enabling event. Computation of a new failure frequency allows importance measures to be computed for initiating and enabling events. These measures are weighting functions which are simply the new Top Event occurrence frequency divided by the Top Event occurrence frequency.

In terms of the importance expressions given in Steps 12 and 13, risk reduction functions can be computed which assess the quantitative impact of component or procedural design changes. In addition, as described in Section 4 of this paper, risk reduction functions and importance measures can be formulated in terms of expected man-rem release using the procedure delineated in Figure 1.

4. CALCULATIONAL PROCEDURES AND METHODS

In the generation and analysis of fault trees and event trees, it is important to distinguish between two types of events:

- o Initiating events which cause system upset conditions and challenge system mitigative features to respond
- o Failure of system mitigative features that result in a serious accident such as a core melt and a radiological release given the occurrence of the initiating event [called enabling events in Ref. (1)].

Initiating and enabling events are defined with a Top Event in mind.

In this section, we show that by factoring the Top Event Boolean expression according to initiating and enabling events, we can obtain the physical meaning of the Boolean expression with regard to accident causation. Also by this factoring, the reader will readily understand how the following expressions are derived:

- o Accident frequency
- o Importance expressions for ranking and components.

The technical approach is described in terms of a simplified system of a pressure tank system [described in Ref.(1) and below]. Section 5 discusses how expressions such as core melt frequency, radiological release frequency, and importance expressions are derived.

4.1 PRESSURE TANK EXAMPLE

The system shown in Figure 2 discharges gas from a reservoir into a pressure tank. The pumping cycle is initiated by an operator who manually resets the timer, the timer contacts close and the pump starts. The manual switch is normally closed. Later (well before any overpressure condition can exist) the timer times out and the timer contacts open. Current is denied to the pump and pumping ceases. If the timer contacts do not open, the operator is instructed to observe the pressure gauge and to open the manual switch, thus causing pumping to cease. After each cycle, the compressed gas is discharged by opening the valve and then closing the valve before the next cycle begins. At the end of the operating cycle, the operator is instructed to verify the operability of the pressure gauge by observing a decrease in the tank pressure as the discharge valve is opened. To simplify the analysis, we assume that the tank is unpressurized before the cycle begins. An undesired event analyzed from a safety viewpoint in this paper is pressure tank rupture, either under load or by overpressure.

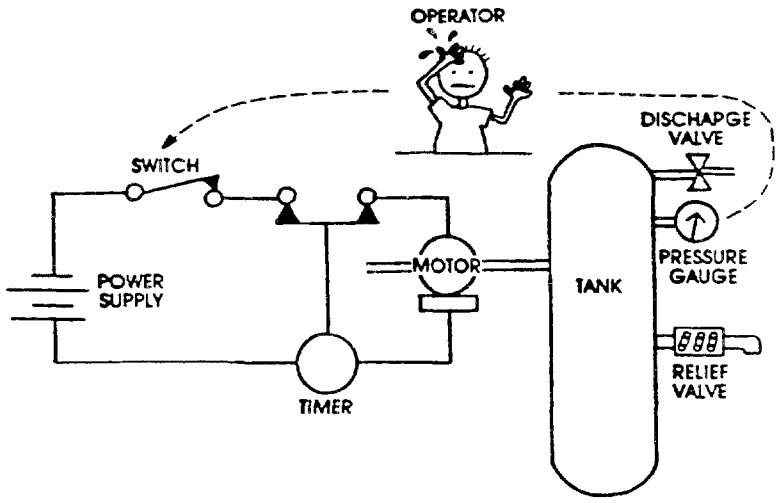


FIG. 2. Pressure tank system.

4.2 LOGIC MODEL DEVELOPMENT

A logic model describing tank rupture can be generated in terms of either (1) a fault tree, or (2) an event tree in combination with fault trees. Both approaches are used in a nuclear PSA and are discussed below.

Figure 3 displays a fault tree with Top Event "Pressure Tank Rupture". The fault tree consists of gate events and basic events. Gate events are output of logic gates, either AND or OR; basic events appear at the bottom of the fault tree and represent the limit of resolution of the fault tree. Basic events include:

- o Human error
- o Random equipment failure
- o Environmental conditions.

Basic events can include common cause events such as failure in support systems. The event "pressure tank rupture under normal load," is a single event leading to rupture of the tank. In nuclear power plant PSA applications, this event is analogous to "rupture of the reactor pressure vessel" which leads directly to core melt. We focus our attention on the gate event "tank rupture due to overpressure." The cause of overpressure is the

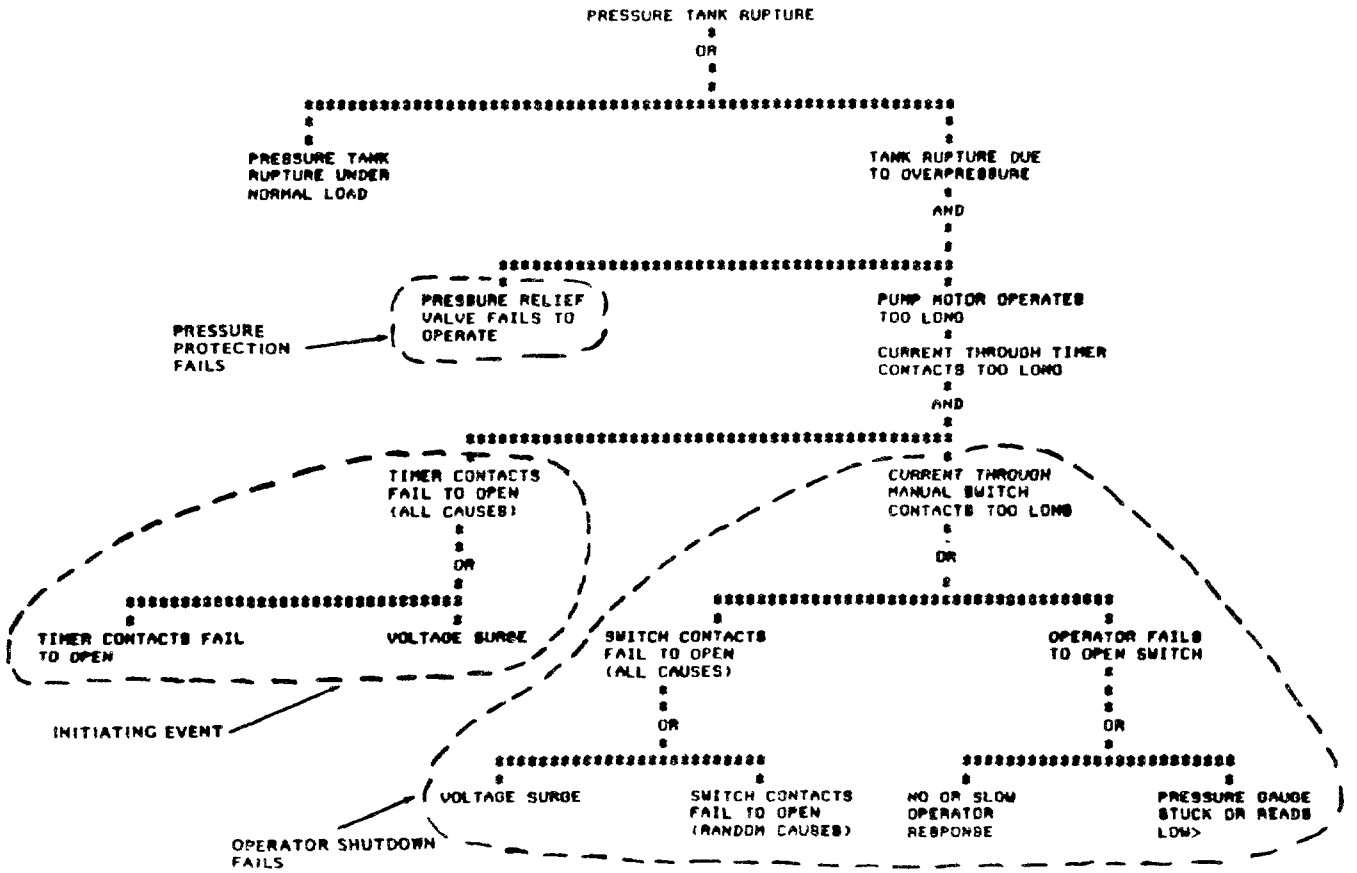


FIG. 3. Fault tree for pressure tank system.

gate event "timer contacts fail to open," which causes the pump motor to continue to operate: The basic event "pressure relief valve fails to operate" represents failure of pressure protection when the pump motor continues to operate. The gate event "current through manual switch contacts too long," represents failure of the operator shutdown function. The basic event "voltage surge" is a common-cause initiating event (also referred to as special initiators), i.e., it is an event which causes a system upset condition and simultaneously fails system mitigative features. Loss of offsite power is a common-cause initiating event in nuclear power plant PSA analysis. External events, such as flood, fire or earthquakes, can also be common-cause initiating events.

Two important points to be made are that (1) initiating events trigger the occurrence of the Top Event and, (2) the remaining basic events in a min cut set are irrelevant unless the initiating event occurs.

Figure 4 shows the event tree for pressure tank rupture due to overpressurization. The event tree starts with an initiating event and describes combinations of failure of system mitigative features that can lead to undesired system or plant states. In Figure 3, PO denotes the event "pump overrun," the initiating event. OS denotes the failure of the operator shutdown system, PP denotes failure of the pressure protection system. There are three sequences displayed in Figure 4. The sequence labeled PO*OS*PP causes overpressure and tank rupture, * denotes logical intersection, (AND).

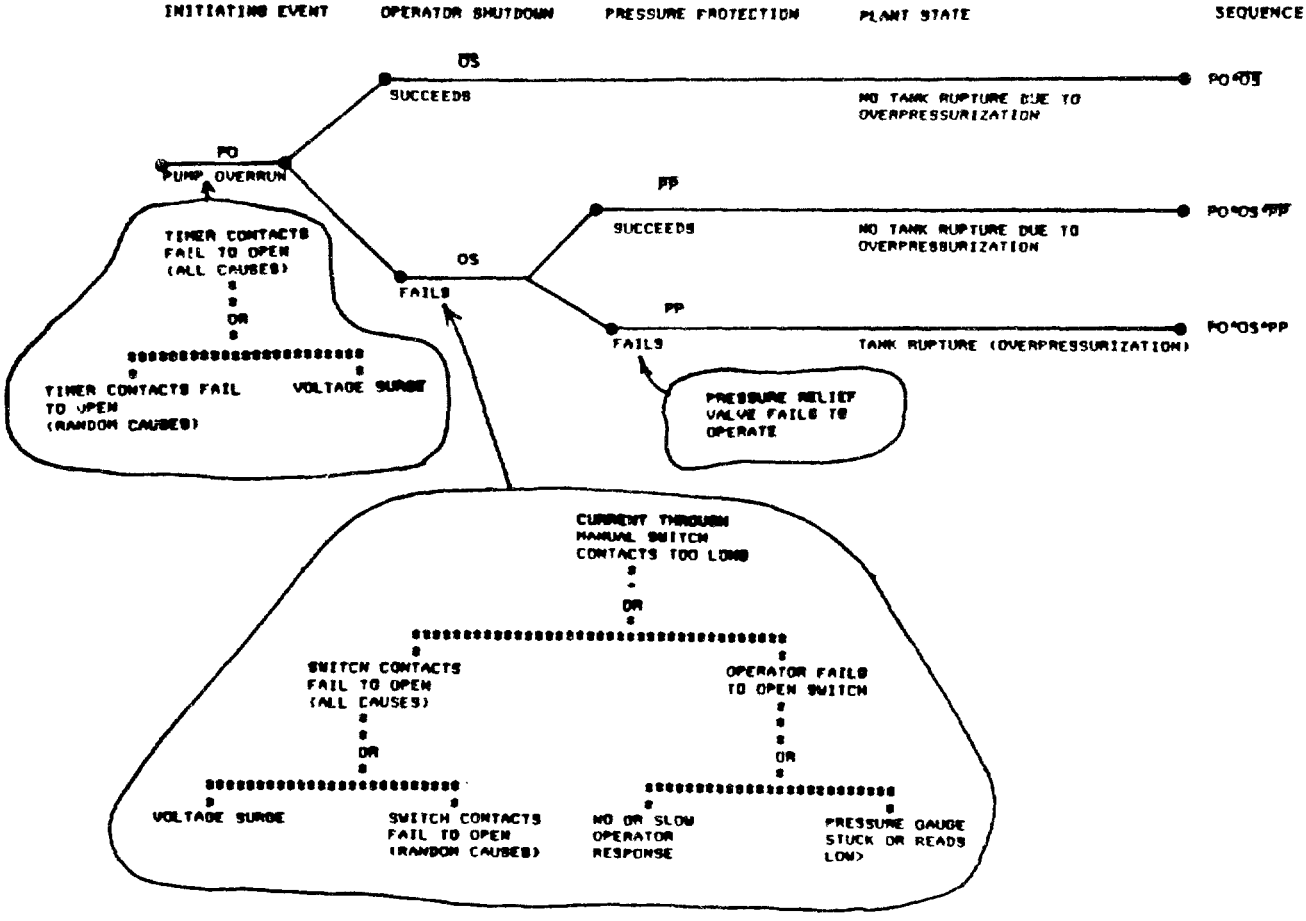


FIG. 4. Event tree for pressure tank system.

The other two sequences lead to safe results. The event tree defines Top Events to fault trees. We see that portions of the fault tree described for Figure 3 appear in Figure 4. Note that the event tree in Figure 4 contains an initiating event fault tree.

As described in ref.(1), initiating event fault trees can become very complex if control system failures are considered. The first author has found that fault trees are good in general for describing how an initiating event can occur such as loss of main feedwater. Event trees are good for describing complex relationships involving accident mitigation which is the traditional way that event trees are used in nuclear power plant PSA applications.

4.3 QUALITATIVE ANALYSIS

The next step after the logic model is qualitative analysis which entails finding the min cut sets and performing other Boolean algebraic operations. Min cut sets are sets of basic events whose occurrence ensures the occurrence of the Top Event.

4.3.1 Min Cut Sets

As shown is Table 1 , there are a total of five min cut sets to the fault tree in Figure 3. These min cut sets also describe how accident sequence PO*OS*PP can occur. Each min cut set contains one initiating event, which implies there is only one time sequence by which a minimal cut set can cause system failure. Three initiating events are listed:

- Tank ruptures under normal load
- Voltage surge (a common-cause initiating event)
- Timer contacts fail to open.

There are numerous computer codes that can find min cut sets. Consult Refs. (6) and (7).

4.3.2 Boolean Factorization

In Table 2, we factor the min cut sets in two ways:

- In terms of basic events (expression 1)
- In terms of system failures (expression 2).

System failure represents the aggregation of basic events (i.e., component failures) for that system. This aggregation describes how the system fails to perform its intended task or function.

It is important to note that the aggregation excludes basic events representing failure of support systems which affect more than one system. These basic events should be considered separately. Failure in support

TABLE 1. LISTING OF MIN CUT SETS
(for fault tree in Figure 1)

Min Cut Set	Description
1	o Tank Ruptures Under Normal Load (i)
2	o Voltage Surge (i) o Relief Valve Fails to Operate (e)
3	o Timer Contacts Fail to Open (i) o Relief Valve Fails to Operate (e) o Pressure Gauge Stuck (e)
4	o Timer Contacts Fail to Open (i) o Relief Valve Fails to Operate (e) o No or Slow Operator Response (e)
5	o Timer Contacts Fail to Open (i) o Relief Valve Fails to Operate (e) o Manual Switch Fails to Open (e)

(i) Denotes an initiating event
(e) Denotes an enabling event

TABLE 2. BOOLEAN FACTORIZATION OF MIN CUT SETS IN TABLE 1

Expression Number	Pressure Tank Rupture =
1	$\text{Tank} + \{\text{R-Valve}\} * \text{V-Surge} + \{\text{R-Valve} * [\text{Gauge} + \text{Operator} + \text{Switch}]\} * \text{Timer}$
2	$= \text{Tank} + \{\text{Press-Protect}\} * \text{V-Surge} + \{\text{Press-Protect} * \text{Op-Shutdown}\} * \text{Timer}$

Where

$+$ = Boolean Union (OR)

$*$ = Boolean Intersection (AND)

Press-Protect = R-Valve

Op-Shutdown = Gauge + Operator + Switch

NOTES:

1. Expression 1 is the top event Boolean expression factored according to basic events.
2. Expression 2 is the top event Boolean expression factored according to failure of system mitigative features.
3. Boolean terms in braces define the critical system state for each initiating event.

systems, such as electric power, can result in failure of more than one system function. For example, in the pressure tank system, we can think of voltage surge as a power supply failure which causes both overpressure (a system disturbance), and simultaneously the failure of the operator shutdown function. In this case, system failures can not be uniquely factored with their support system failures. (Section 5.2 further elaborates on this point.)

In essence, we are aggregating independent component failures within systems, i.e., these failures do not appear elsewhere in the fault tree(s) and/or the event trees.

Aggregation of independent component failures is useful for two reasons:

- o substantially reduces the number of min cut sets
- o allows assessment of the collective contribution of independent component failures to the total risk.

As described in Section 5.2, RSSMAP performed such an aggregation for all safety systems.

In Table 2, the terms in brackets define the critical system states for the occurrence of the initiating event. Stated qualitatively, critical system states describe the vulnerability of the plant to the occurrence of the initiating event.

Examining Table 1, we see qualitatively that the tank rupturing underload is the most important event since it is a single event leading to tank rupture. However, as described in Section 4.8, this event is a passive failure and, from a probabilistic viewpoint, is not the most important failure.

4.4 DATA ANALYSIS

It is important to note that expressions 1 and 2 in Table 2 are in an exact form for the computation of accident frequency. These expressions imply that we model the occurrence of the Top Event (or more generally, the occurrence of an accident) as follows:

- o Initiating event occurs
- o System is in a critical system state for the occurrence of the initiating event.

Since initiating events place a demand on system mitigative features to respond, we are interested in computing two quantities from a reliability viewpoint:

- o Initiating event failure frequency
- o Probability that system mitigative features fail to operate when the initiating event occurs, (enabling event unavailability).

To compute these quantities, we must know the maintenance policies to which system components are subjected. Ref.(1) discusses maintenance policies. For reliable systems, the component failure rate, λ, the conditional probability of failure per unit time is an accurate approximation to the failure frequency, $w_f(t)$. Enabling event unavailability, q, is a function of the following reliability parameters:

- o The component failure rate, λ
- o Inspection interval, θ
- o Repair time, τ.

From an importance ranking viewpoint, it is important to note that changes in these parameters can affect component unavailability and hence system unavailability. Figure 5 displays component unavailability and failure frequency for the following maintenance policies:

- o No repair
- o Repair, announced failure
- o Repair, unannounced failure.

It is assumed in Figure 5 that λ and τ are constant. We see that for all three maintenance policies, λ is an accurate approximation upper bound for reliable systems. See Refs.(1) and (8) for a more detailed description of maintenance policies.

Table 3 lists the basic event data for the pressure tank system.

In modeling operator recovery in PSA, it is important to include human factors analysis in predicting human error failure probabilities. Consult Refs. (9), (10), (11) and (18) for a discussion of human factors analysis as it pertains to nuclear power plant PSA applications.

In addition, where there is little or no plant-specific data available, an analyst must use generic sources of data. Consult Refs. (7) and (12) for sources of reliability data for nuclear power plant components.

4.5 ACCIDENT FREQUENCY EXPRESSION

In the formulation of an accident frequency expression, we condition in a statistical sense on the occurrence of the initiating event. Then we define the fault duration times of pre-existing conditions (i.e., enabling events) relative to the occurrence of the initiating event. Pre-existing events can be events such as:

- o latent failures
- o failures resulting in loss of system redundancy

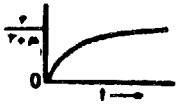
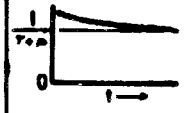
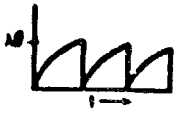
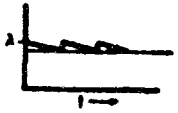
MAINTENANCE POLICY	COMPONENT UNAVAILABILITY	ASYMPTOTIC VALUE	COMPONENT UNAVAILABILITY VERSUS TIME	COMPONENT FAILURE FREQUENCY	ASYMPTOTIC VALUE	COMPONENT FAILURE FREQUENCY VERSUS TIME
1. No Repair	$1-\exp(-\lambda t) \leq \lambda t$	1		$\lambda\exp(-\lambda t)$	0	
2. Repair Announced Failure	$\frac{\tau}{\mu+\tau}[1-\exp[(-\frac{\mu+\tau}{\mu\tau})t]$	$\frac{\tau}{\mu+\tau} \leq \lambda\tau$		$\frac{1}{\mu+\tau}[1-\frac{\tau}{\mu}\exp(-\frac{\mu+\tau}{\mu\tau}t)]$	$\frac{1}{\mu+\tau} < \lambda$	
3. Repair Unannounced Failure	$1-\exp[-\lambda(t-(n-1)\theta)]$ $(n-1)\theta \leq t \leq n\theta$ $n = 1, 2, \ldots$	$1-(1-\exp(-\lambda\theta))/\lambda(\tau+\theta)$ $\approx \lambda\theta/2$ for $\tau << \theta$ (Average Unavailability)		$-\lambda\exp[-\lambda(t-(n-1)\theta)]$ $(n-1)\theta \leq t \leq n\theta$ $n = 1, 2, 3 \ldots$	$-\lambda\exp[-\frac{\lambda\theta}{2}] < \lambda$	

FIG. 5. Component unavailability and failure frequency constant λ and τ.

TABLE 3. DATA FOR BASIC EVENTS

Component Failure Mode	Basic Event Type	Failure Rate, λ, or Enabling Event Unavailability, q
Tank Rupture Under Normal Load, PT	Initiator	$\lambda_{PT} = 10^{-8}$/cycle
Timer Contacts Fail to Open, T	Initiator	$\lambda_{T} = 10^{-4}$/cycle
Voltage Surge, VS	Initiator	$\lambda_{VS} = 10^{-8}$/cycle
Relief Valve Fails to Operate, R	Enabler	$\lambda_{R} = 3 \times 10^{-4}$/hour
		$\theta_{R} = 1$ year
		$q_{R} = 0.65$*
No or Slow Operator Response, O	Enabler	$q_{O} = 10^{-2}$/demand
Manual Switch Fails to Open, S	Enabler	$q_{S} = 10^{-4}$/demand
Pressure Gauge	Enabler	$q_{G} = 10^{-5}$/hour

* It is assumed that $\tau << \theta$.
Expression $q = 1 - (1 - \exp(-\lambda\theta)) / \lambda\theta$
is used since $\lambda\theta$ is not small.

- conditions for fire and explosion such as:

  i) explosive concentration present

  or

  ii) ignition source present.

Enabling events can be demand failures such as "operator failing to respond." Hence enabling events can occur before, during or after the occurrence of the initiating event. If we assume that initiating events are randomly occurring events and that the occurrence of two initiating events in a differential time is zero (which is always the case for reliable systems), then the Top Event Occurrence Frequency, W(t), is the sum of the frequencies at which initiating events cause system failure, i.e.,

$$W(t) = \sum_{i=1}^{n_i} \Pr\left\{\begin{array}{l}\text{System is in a critical system state}\\ \text{for the occurrence of initiating}\\ \text{event } i\end{array}\right\} w_{f,i}(t)$$

$$= \sum_{i=1}^{n_i} \Pr\left\{\bigcup_i E_{i,k}\right\} w_{f,i}(t) \qquad (1)$$

where

$\Pr$ = probability

$E_{i,k}$ = event that min cut set k containing initiating event i occurs (with event i set equal to true)

$\bigcup_i$ = Boolean union of min cut sets containing initiating event i

n_i = number of initiating events in the fault tree or logic model

$w_{f,i}(t)$ = failure frequency of initiating event i

To probabilistically evaluate the terms in parenthesis in expression 1, assumptions must be made with regard to statistical dependency of basic events. With no loss in generality, we make the following assumptions:

- o System is reliable (i.e., the probability of the simultaneous occurrence of two or more min cut sets is small)
- o Basic events are statistically independent
- o λ, the conditional probability of failure per unit time, is an accurate approximation for failure frequency.

Expression (1) becomes

$$W(t) = \sum_{i=1}^{n_i} \left(\sum_{\substack{j\\ i\epsilon K_j}} \prod_{\substack{l\epsilon K_j\\ i\epsilon K_j\\ i=l}} q_l\right) \lambda_i \qquad (2)$$

Notation:

j	is an index for min cut sets
K_j	denotes the jth min cut set
ϵ	means belongs to
q	denotes enabling event probability
l	is an index for enabling events
i	is an index for initiating events
n_j	denotes the number of initiating events in all min cut sets

The term in parenthesis in expression (2) is a first order approximation for the critical state unavailability for initiating event i. This term is the sum of the conditional min cut set probabilities containing the initiating event i with i set equal to true. Expression 2 is generally an accurate expression for most risk calculations.

For the pressure tank system, expression (2) becomes (see Table 3 for notation)

$$W(t) = \lambda_{PT} + q_R\lambda_{VS} + \{q_Rq_G + q_Rq_O + q_Rq_S\}\lambda_T \quad (3)$$

$$= \lambda_{PT} + q_R\lambda_{VS} + q_R\{q_G + q_C + q_S\}\lambda_T \quad (4)$$

$$= \lambda_{PT} + Q_{PP}\lambda_{VS} + Q_{PP}Q_{OS}\lambda_T \quad (5)$$

Note that small "q" denotes component unavailability, capital "Q" denotes system unavailability. We see that expression 3 is simply the sum of the min cut set frequencies (see Table 1). Expression (5) is factored according to system unavailabilities.

If we assume there is on the average one operating cycle per hour and if we use the basic event data in Table 3, expression (3) becomes

$$W(t) = 10^{-8} + 0.65 \times 10^{-8} + 0.65\{10^{-5} + 10^{-4} + 10^{-4}\}10^{-4}/hr$$

$$= 6.7 \times 10^{-7}/hr \quad (6)$$

$$= 5.9 \times 10^{-3}/yr$$

The mean time to the occurrence of the Top Event is the reciprocal of W(t), i.e., 170 years. Note that system failure probabilities are $Q_{PP} = 0.65$ and $Q_{OS} = 1.011 \times 10^{-2}$.

Thus far, we have discussed failure rates and repair rates that are constant in time. In some situations, failure rates may exhibit a burn-in as well as a wear-out phenomenem. As discussed in Ref. (13), it is a straightforward procedure to include such phenomena in the reliability calculations. In this case, the Top Event occurrence frequency is not constant in time and must be integrated over time to obtain the expected number of occurrences (of the Top Event) per unit time.

4.6 IMPORTANCE EXPRESSIONS

In this paper, importance expressions are weighting functions, The development of the importance expression for either a component or a system is straightforward. There are three basic steps in the computation:

- Formation of a new Top Event that is the Boolean union of the min cut sets containing either the initiating or enabling event
- Use of expression (1) to compute the frequency of occurrence for the new Top Event (for initiating-event importance expressions, only one event can function as the initiating event for the new Top Event)
- Divide the results in Step 2 by the accident frequency.

Stated mathematically, the importance expression for basic events (or systems) weighted according to accident frequency is I_{AF}

$$I_{AF} = \frac{\text{Frequency of the Boolean Union of min cut sets containing the event of interest}}{\text{Top Event Occurrence Frequency, W(t)}} \qquad (7)$$

The above importance expression is simply the fractional contribution of min cut sets (containing either the initiating or the enabling event) to the total accident frequency.

Table 4 lists the importance expressions and values for the basic events and systems for the pressure tank system. The weighting is according to Top Event occurrence frequency, W(t). In this case W(t) is constant. In many cases in risk assessment W(t) is constant or can be accurately represented by a constant. It is assumed that first order approximations are valid, i.e., use of expression 2 results in an accurate calculation. Examining the expressions in Table 4, we see that for initiating events, the numerator is a linear function of the failure frequency; for enabling events, the numerator is a linear function of the enabling event unavailability. Conceptually, enabling event importance is a contributory measure of importance since enabling events do not cause the Top Event to occur.

4.7 SAFETY RANKING EXPRESSIONS

If we incorporate system design changes or component reliability improvements that result in infinitesimal changes in system unavailability or component reliability (and hence infinitesimal changes in system risk), then expression (7) can be used directly to rank components or systems. Usually, we incorporate changes or improvements that result in finite changes. In this case, we must develop an importance expression for systems and components when finite changes in system risk occur. As described below, we can develop this importance expression in terms of expression (7) given above.

TABLE 4. IMPORTANCE RANKINGS FOR PRESSURE TANK SYSTEM

Component Failure Mode or System Failure Mode	Mathematical Expression*	Value
Pressure Tank Rupture Under Load, PT	$\lambda_{PT}/W(t)$	1.5×10^{-2}
Timer Contacts Fail to Open, T	$q_R \{q_G + q_O + q_S\} \lambda_T / W(t)$	0.97
Voltage Surge, VS	$q_R \lambda_{VS}/W(t)$	9.7×10^{-3}
Relief Valve Fails to Operate, R, or Pressure Protection Fails, PP	$(q_R \lambda_{VS} + q_R \{q_G + q_O + q_S\} \lambda_T)/W(t)$ $= \{Q_{PP} \lambda_{VS} + Q_{PP} Q_{OS} \lambda_T\}/W(t)$	0.98
No or Slow Operator Response, O	$q_R q_O \lambda_T/W(t)$	0.97
Manual Switch Fails to Open, S	$q_R q_S \lambda_T/W(t)$	9.7×10^{-3}
Pressure Gauge Stuck, G	$q_R q_G \lambda_T/W(t)$	9.7×10^{-4}
Operator Shutdown System Fails, OS	$q_R \{q_G + q_O + q_S\} \lambda_T/W(t)$ $= Q_{PP} Q_{OS} \lambda_T/W(t)$	0.98

* $W(t) = \lambda_{PT} + q_R \lambda_{VS} + (q_R \{q_O + q_S + q_G\}) \lambda_T$

In this section, we consider importance rankings weighted according to accident frequency, W(t). To show mathematically how accident frequency, W(t), decreases when component unavailability decreases, W(t) is written in terms of two sums, the sum of min cut set frequencies not containing i, and the sum of min cut set frequencies containing component i.

Stated mathematically,

$$W(t) = \{ (1 - I_i) + I_i \} \; W(t) \qquad (8)$$

where

I_i = I_{AF}, expression (7) given above.

The first term in expression (8) is the contribution of other min cut sets not containing basic event i to the accident frequency. The first term times W(t) can be thought of as the residual risk when component i works perfectly. For

example, consider a component failure which is an enabling event. If an improvement in component i is made to decrease the component unavailability to q_i (new), then the component availability improvement ratio, r_i, can be defined as:

$$r_i = q_i \text{ (new)}/ q_i \text{ (old)}. \quad (9)$$

It can be shown that the new accident frequency, W(new), can be written:

$$W(\text{new}) = (1 - I_i)W(\text{old}) + r_i I_i W(\text{old}) \quad (10)$$

When the component unavailability is changed to its new value, the risk reduction ratio for component i, R_i, is defined as

$$R_i = W(\text{new})/ W(\text{old}) \quad (11)$$

Mathematically,

$$R_i = (1 - I_i) + r_i I_i \quad (12)$$

The maximum improvement in safety that can be obtained by improving the unavailability of component i is when the residual risk ratio is reduced to:

$$1 - I_i \quad (13)$$

which corresponds to the case in which $r_i = 0$ or q_i (new) = 0, i.e., as a result of the design change, component i never fails. Note that the component of highest importance has the lowest residual risk ratio. Stated in other terms, when the component of highest importance is improved, so that it works perfectly, the remaining residual risk is the smallest. Vesely, et al, Ref. (2), calls the reciprocal of expression (13), the risk reduction worth.

It is important to note that expression (12) applies for independent system failures as well as for initiating events. For independent system failures, the system availability improvement ratio can be defined as

$$r_i = Q_i \text{ (new)}/ Q_i \text{ (old)} \quad (14)$$

For initiating events, the initiating event improvement ratio can be defined as

$$r_i = \lambda_i \text{ (new)}/ \lambda_i \text{ (old)} \quad (15)$$

Another useful measure is the fractional reduction in risk defined as

$$[W(\text{old}) - W(\text{new})] / W(\text{old})$$

$$= 1 - R_i$$

$$= i_i(1 - r_i) \quad (16)$$

The percentage reduction in risk is

$$I_i(1 - r_i) \times 100\% \quad (17)$$

The advantage in using expression (16) or (17) is that the change or improvement with the greatest risk reduction will generate the largest value. Expression (12) generates the smallest value.

It is important to note that we can take expression (16) and divide by the cost of the design change to obtain an expression that gives up the maximum risk reduction for minimum cost. Such an expression could be used for backfit decisions.

4.8 EXAMPLE OF SAFETY RANKING METHODOLOGY

We use the pressure tank example to show how to prioritize system design changes and upgrades.* Examining Tables 4 and 5, we see that the following events are important:

- o Timer contacts fail to open
- o Relief valve fails to operate

TABLE 5. RISK REDUCTION RATIOS FOR PRESSURE TANK SYSTEM

Component Failure Mode or System Failure Mode	Importance Value (from Table 6)	Risk Reduction Ratio $R_i = (1 - I_i) + I_i r_i$
Pressure Tank Rupture Under Load, PT	1.5 E-2	0.985 + 1.5 E-2 r_i
Timer Contacts Fail to Open, T	0.97	0.03 + 0.97 r_i
Voltage Surge, VS	9.7 E-3	0.99 + 9.7 E-3 r_i
Relief Valve Fails to Operate, R, or Pressure Protection Fails, PP	0.98	0.02 + 0.98 r_i
Manual Switch Fails to Open, S	9.7 E-3	0.99 + 9.7 E-3 r_i
No or Slow Operator Response, O	0.97	0.03 + 0.97 r_i
Pressure Gauge Stuck, G	9.7 E-4	0.999 + 9.7 E-4 r_i
Operator Shutdown System Fails, OS	0.98	0.02 + 0.98 r_i

*Section 5.12 describes application of the GGNS safety ranking methodology for GGNS

- o No or slow operator response.

Also the failures of the following system are important:

- o Operator shutdown
- o Pressure protection.

To mitigate the effects of these failures, we can incorporate changes in reliability parameters of the above components, e.g., we can incorporate the following potential improvements, (new values of reliability parameters are indicated in parenthesis):

- o Install a timer that fails less frequently (a failure rate of 1 E-5 per cycle)
- o Special operator procedures (operator failure probability 1 E-3)
- o Shorten inspection interval of the relief valve (one month)
- o Install an identical redundant timer in series with the first one (inspect each timer once a month).

For the first three improvements listed above, only a component's reliability parameter changes. It is straightforward to compute the component's availability or failure frequency improvement ratio, r_i, and hence, the risk reduction ratio R_i. The last improvement, however, requires different treatment. In this case, the fault tree logic will change since the initiating event fault tree will change and new min cut sets as shown in Table 6 will be generated. In this case, there is functional redundancy in preventing the occurrence of the initiating event. Min cut sets 3, 4 and 5 in Table 2 can have two basic events which can function as an initiating event. In other words, there are two time sequences defined by one min cut set. As described in Section 4.5, min cut sets defining conditions for fire and explosion have this property. (An example of improving nuclear power plant safety by functional redundancy would be the incorporation of an extra offsite power line to prevent loss of offsite power. Another example is the incorporation of an extra main feedwater train to prevent the loss of main feedwater.) For this special case, we can still compute a new failure frequency for the random failure of both timers, timer 1 and timer 2, as

$$q_1\lambda_2 + q_2\lambda_1 = 2q_T\lambda_T \qquad (18)$$

where

$$q_T = \lambda_T\theta_T/2$$

$$= 1\text{ E-}4 \times 720 / 2$$

$$= 3.6\text{ E-}2$$

TABLE 6. LISTING OF MIN CUT SETS FOR PRESSURE TANK SYSTEM WITH TWO TIMERS

Min Cut Set	Description
1	o Tank Ruptures Under Normal Load (i)
2	o Voltage Surge (i) o Relief Valve Fails to Operate (e)
3	o Timer Contacts 1 Fail to Open (i) o Timer Contacts 2 Fail to Open (i) o Relief Valve Fails to Operate (e) o Pressure Gauge Stuck (e)
4	o Timer Contacts 1 Fail to Open (i) o Timer Contacts 2 Fail to Open (i) o Relief Valve Fails to Operate (e) o No or Slow Operator Response (e)
5	o Timer Contacts 1 Fail to Open (i) o Timer Contacts 2 Fail to Open (i) o Relief Valve Fails to Operate (e) o Manual Switch Fails to Open (e)

(i) Denotes an event which can function as an initiating event
(e) Denotes an enabling event

$\lambda_T \theta_T/2$ is the first order expansion of the expression given in the footnote of Table 3. This expression is the average unavailability of a component in standby.

When considering random failures for both timers, it is important to note that the timer that fails first does not cause pump overrun, it is the second timer failure that causes pump overrun. Expression (18) represents two possible sequences of events. In the first term, timer 1 fails first (it is the enabling event) and timer 2 fails second (it is the initiating event). There is the reverse ordering of events for the second term. The initiating event improvement ratio, expression (15) for the timer in Table 5 is simply:

$$r_T = 2q_T\lambda_T/\lambda_T = 2q_T \qquad (19)$$

In general, a concise term such as expression (19) cannot be computed when the fault tree or event tree logic changes. In this case, it becomes necessary to compute a new accident frequency from the new min cut sets. Also note in Table 7, that the operator shutdown system improvement ratio is given for the case where special operator procedures are employed. To compute the system improvement ratio requires that the min cut sets for system failure be known. Otherwise, one cannot assess the change in system availability when a change in component reliability occurs.

TABLE 7. EFFECT OF POTENTIAL IMPROVEMENTS FOR THE PRESSURE TANK SYSTEM

Potential Improvement (Change in reliability parameter)	Component (C) or System (S) Improvement Ratio, r_i *		Risk Reduction Ratio(See Table 5**	Percentage Risk Reduction ***
Install a more reliable timer (failure rate decreases by a factor of 10)	1 E-5/1 E-4 = 0.1	(C)	0.13	87%
Special Operator procedures (failure probability decreases by a factor of 10)	1 E-3/1 E-2 = 0.1	(C)	0.13	87%
	1.1 E-3/1.01E-2 = 0.11	(S)*	0.13	87%
Shorten Inspection interval of relief valve (inspection interval decreases by a factor of 12)	0.21/0.65 = 0.32	(C)	0.33	67%
Incorporate a redundant timer in series with the first (inspection interval of one month)	$2q_T$ = 7.2E-2**	(C)	0.10	90%

* $r_i = (Q_{OS})_{new}/(Q_{OS})_{old}$

$=(q_G + q_O + q_S)_{new}/(q_G + q_O + q_S)_{old}$

where OS denotes operator shutdown system

** Expression (12), Section 4.7

*** Expression (17), Section 4.7

One last point, as significant design changes are actually incorporated into the plant, new importance values must be computed and new rankings as shown in Table 5 must be generated.

5. NUCLEAR POWER PLANT PSA APPLICATIONS

In this section, we show how the concepts presented in Section 3 for the pressure tank example can be extended to derive the following risk expressions:

- Core melt frequency
- Radiological release frequency
- Importance expressions for ranking of systems and components.

We discuss derivation of the above expressions in terms of the results of the GGNS RSSMAP study Ref. (4). In this section, we first discuss the GGNS RSSMAP study. Later, we discuss derivation of the above expressions.

5.1 GGNS RSSMAP STUDY

The Grand Gulf PSA was conducted as part of the Reactor Safety Study Applications Program, RSSMAP, and was a follow-up study to the Reactor Safety Study, RSS. RSSMAP was conducted with the following objectives:

- Identify the risk dominating accident sequences for a wider range of reactor designs than considered in the RSS
- Compare those accident sequences with those identified for the reactors studied in the RSS
- Based on this comparison, identify design differences between the plants which have a significant impact on risk.

RSSMAP studied the following nuclear power plants:

- Sequoya #1, Westinghouse pressurized water reactor, PWR, with ice condenser containment
- Oconee #3, Babcock and Wilcox PWR with a dry containment
- Calvert Cliffs #2, Combustion Engineering PWR with a dry containment.
- Grand Gulf #1, GE boiling water reactor, BWR/6 with a Mark 3 containment.

This report describes the Grand Gulf #1 RSSMAP study. The RSSMAP study consisted of two tasks:

- A systems analysis task
- An accident process task

We discuss each task below as it relates to the Grand Gulf RSSMAP study.

5.2 SYSTEMS ANALYSIS TASK

The Grand Gulf RSSMAP study considered two types of initiating events:

- Loss of coolant accidents, LOCAs
- Transients.

Two break sizes were considered for LOCAs (assumed frequencies by RSSMAP are indicated in parenthesis):

- o Small LOCA, break size less than a 13.5 inch diameter hole denoted by the letter S (1.4×10^{-3}/yr)
- o Large LOCA, break size greater than a 13.5 inch diameter hole denoted by the letter A (1.0×10^{-4}/yr)

Two transients were modeled for Grand Gulf, one depicting the loss of offsite power denoted at T_1 with frequency of 0.2 events per year, and the other describing all other transients as T_{23}. T_{23} which includes events such as the loss of main feedwater and is given a frequency of 7 per year. The RSSMAP study did not consider external events such as flood, fire or earthquake, nor did it consider sabotage-caused events.

The Grand Gulf LOCA event tree is displayed in Figure 6. RSSMAP judged this event tree to be adequate in representing the entire spectrum of break sizes. The success criteria for the emergency coolant injection and residual heat removal are different for the two LOCA sizes; nevertheless, the event tree structure does not change. The event definitions for the event tree headings are given in Table 8.

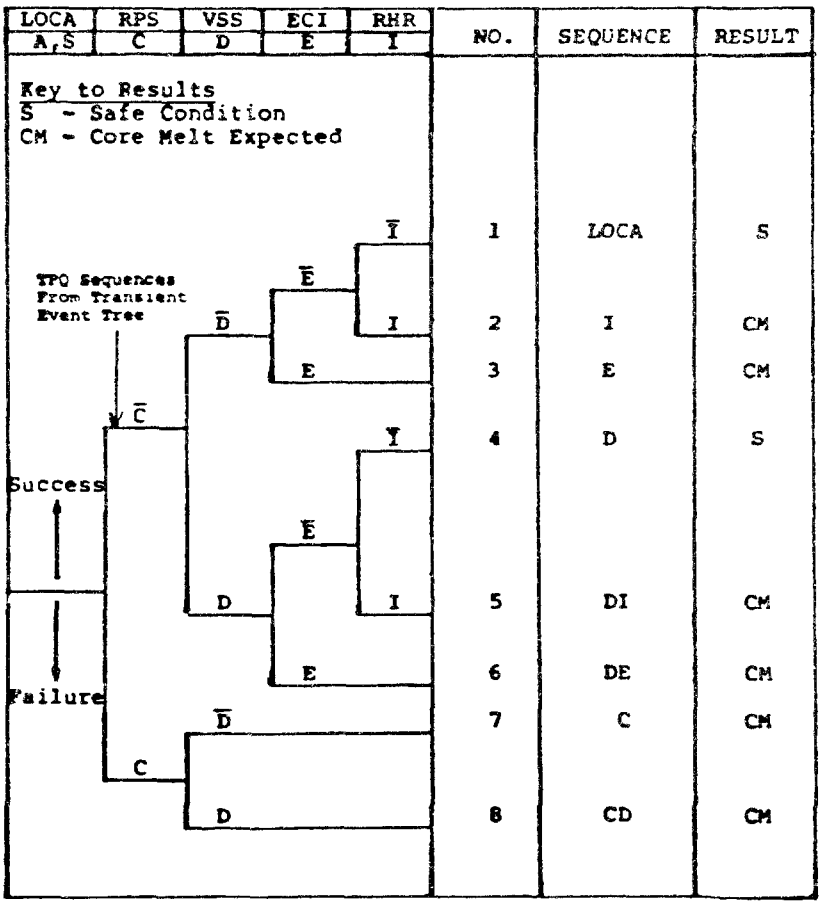


FIG. 6. Grand Gulf LOCA event tree (from Ref. 4).

TABLE 8. EVENT DEFINITION FOR LOCA EVENT TREE

LOCA - A breach of the pressure boundary of the Reactor Coolant System (RCS) which causes an uncontrollable loss of water inventory. there are two LOCA categories.

A - Large LOCA - A breach of the RCS with a flow area greater than 1 ft^2 (A > 13.5" diameter).

S - Small LOCA - A breach of the RCS with a flow area less than 1 ft^2 (S ≤ 13.5" diameter).

C - Reactor Protection System (RPS) - Failure of the Reactor Protection System to obtain and maintain reactor subcriticality.

D - Vapor Suppression System (VSS) - Failure of the suppression pool or containment sprays to condense steam produced by a LOCA.

E - Emergency Coolant Injection (ECI) - Failure to provide sufficient water to the core to prevent core melt.
- ECI for large (A) LOCAs - Failure to provide flow to the RCS from the HPCS or the LPCS of 3 out of 3 LPCI trains.
- ECI for Small (S) LOCAs - Failure to provide flow to the RCS from the HPCS, RCICS, LPCS, or 2 out of 3 LPCI trains. The ADS is required for successful LPCS or LPCI operation to reduce system pressure.

I - Residual Heat Removal (RHR) - Failure of the Residual Heat Removal System (RHRS) in conjunction with Standby Service Water System (SSWS) to remove decay heat from the containment. The SSWS is required to supply cooling water to the secondary sides of the heat exchangers. The RHRS can successfully remove heat using either train A or B in the suppression pool cooling mode.

The Grand Gulf transient event tree is displayed in Figure 7. Again, RSSMAP judged this event tree to be adequate in representing the two transient events analyzed in the RSSMAP study. The systems which mitigate the effects of the transients are displayed in Table 9.

Elaborate fault trees were not generated for the RSSMAP study. Instead, a "survey and analysis" technique was used to determine system failure modes. Boolean equations were generated to include failures due to hardware as well as downtimes due to tests and maintenance. Recovery, such as repair of failed components, was considered. Operator error however, was not included.

In the RSSMAP study, system failures are represented by the aggregation of basic events that include:

- o Random equipment failure
- o Downtime due to testing and maintenance.

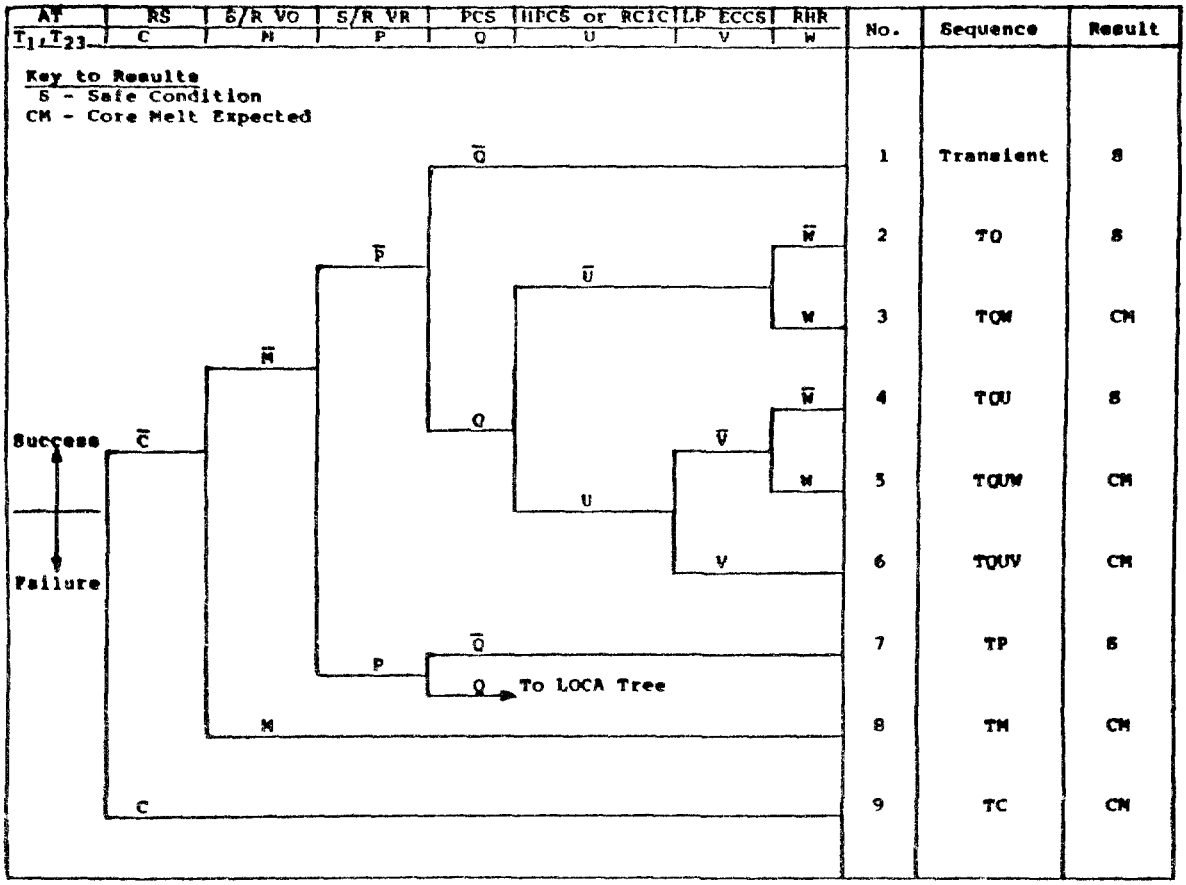


FIG. 7. Grand Gulf transient event tree (from Ref. 4).

Generally, the aggregation was done on either per train or per system basis. This aggregation for the reactor core isolation cooling system (RCIC) is presented in Table 16. As described in section 4.3.2, it is important to note that the aggregation excludes basic events representing failure of support systems which fail more than one system. To further illustrate this point, there are three residual heat removal (RHR) pumps, A, B and C used for heat removal and coolant makeup at GGNS. RHR pumps B and C are fed from the same electrical division, division II. What this implies is that the min cut sets cannot be factored uniquely according to RHR trains with their supported systems since failure of electrical division II results in failure of both RHR trains as well as other system functions.

Most nuclear power PSA analyses to date have used generic data for the initiating event frequency. It is important to note that plants may have different power conversion systems or offsite power grids. In this case, it is necessary to construct an initiating event fault tree since the generic data are not applicable. Also, it may be necessary to construct an initiating event fault tree if there are common-cause initiating events that are significant risk contributors, e.g., a failure of a support system which causes plant shutdown and simultaneously fails engineered safety features. The study team conducted a reliability study of the AC power systems at a nuclear power plant in which an initiating event fault tree was constructed for the offsite power grid.

We are emphasizing here that the analyst should be aware of common-cause initiating events and include them in the analysis if these events are risk significant.

TABLE 9. EVENT DEFINITIONS FOR TRANSIENT EVENT TREE

T_1 or T_{23} Transients - Any abnormal condition in the plant which requires that the plant be shut down, but does not directly breach RCS integrity.

- T_1 - Shutdown initiated by a loss of offsite power.
- T_{23} - Shutdown initiated by a loss of main feedwater caused by other than a loss of offsite power, and shutdowns with main feedwater initially available.

C - Reactor Subcriticality (RS) - Failure of the Reactor Protection System or the Standby Liquid Control System in conjunction with a recirculation pump trip to obtain and maintain reactor subcriticality.

M - Safety/Relief Valves Open (S/R VO) - Failure of sufficient S/RVs to open and relieve excess primary system pressure.

P - Safety/Relief Valves Reseat (S/R VR) - Failure of any open S/RVs to reseat.

Q - Power Conversion System (PCS) - Failure of the PCS to start removing decay heat in the required time (one-half hour when ECCS injection fails and about 30 hours when injection succeeds).

U - High Pressure Core Spray or Reactor Core Isolation Cooling System (HPCS or RCICS) - Failure of the HPCS or RCICS to provide high pressure makeup to the reactor vessel.

V - Low Pressure Emergency Core Cooling Systems (LP ECCS) - Failure of the Low Pressure Core Spray (LPCS) or two of three Low Pressure Coolant Injection (LPCI) trains to provide low pressure makeup to the reactor core. The Automatic Depressurization System (ADS) is required for successful LPCS and LPCIS operation.

W - Residual Heat Removal (RHR) - Failure of the Residual Heat Removal System (RHRS) in conjunction with the Standby Service Water System (SSWS) to start removing decay heat from the containment within about 30 hours. The SSWS is required to supply cooling water to the secondary sides of the RHR heat exchangers. For transients where ECCS injection succeeds, one of two RHRS loops operating in either the suppression pool cooling mode or steam condensing mode will provide RHR.

5.3 ACCIDENT PROCESS TASK

The containment failure modes considered in the Grand Gulf RSSMAP study are shown in Table 10. The computer codes MARCH and CORRAL were used to predict the containment response and magnitude of release following a core melt accident sequence with a specified containment failure mode. MARCH performs the thermal hydraulics associated with the successive stages of core meltdown and containment response. CORRAL describes the fission product transport and deposition within the containment and determines the leakage to the environment. For blowdown through the suppression pool, RSSMAP assumed a decontamination factor of 100 for removal of molecular iodine and particulates. As in the RSS, RSSMAP considered five release categories with category 1 being the most severe. As shown in Table 10, accident sequence frequencies in each category were then summed in order to assess the release frequency per year. Table 10 is taken from Ref. (14), which is an unsmoothed result, i.e., the contribution from adjacent release categories is not considered. As was done in the RSS, RSSMAP performed smoothing of release categories. We use unsmoothed results in Table 4 for importance calculations in this paper.

5.4 CORE MELT FREQUENCY EXPRESSION

In general, for a level 1 PSA, we can develop a Boolean expression for core melt by taking the Boolean union of all accident sequences on the event tree leading to core melt. For the GGNS RSSMAP study, we can develop this expression by taking the Boolean union of all min cut sets in Appendix A. It is important to note that we can use expression (1) to compute core melt frequency, CMF, and use expression (7) to compute the importance of systems and components. In addition, the safety ranking expressions given in Section 4.7 are directly applicable.

All the accident sequences in Appendix A describe core melt from full power. It is important to note that we can include other accident sequences that are not from full power, e.g., a drop of a heavy load during refueling that damages safe shutdown equipment which results in core damage.

5.5 RADIOLOGICAL RELEASE FREQUENCY

Each accident sequence can result in a different radiological exposure due to factors such as:

- Containment failure mode, CFM
- Weather
- Evacuation
- Population.

In the TENERA study, the information in Tables 10 and 11 was used to rank sequences, systems, and components according to radiological release frequency. Table 10 lists the dominant accident sequences found in the RSSMAP study according to CFM probability and release category frequency.

The release categories are taken from WASH 1400.

TABLE 10. GRAND GULF DOMINANT CORE MELT ACCIDENT SEQUENCES

DOMINANT ACCIDENT SEQUENCES	BWR CORE MELT RELEASE CATEGORIES				CONTAINMENT FAILURE MODE PROBABILITIES(2)	
	1	2	3	4		
T_1PQI	$\alpha 1.6 \times 10^{-8}$	$\delta 1.6 \times 10^{-6}$			$\alpha = .01$	$\delta = 1$
$T_{23}PQI$	$\alpha 3.7 \times 10^{-8}$	$\delta 3.7 \times 10^{-6}$				$\delta = 1$
T_1PQE			$\gamma 1.2 \times 10^{-7}$	$\delta 1.2 \times 10^{-7}$	$\gamma = .5$	$\delta = .5$
$T_{23}PQE$			$\gamma 2.7 \times 10^{-7}$	$\delta 2.7 \times 10^{-7}$	$\gamma = .5$	$\delta = .5$
SI	$\alpha 4.6 \times 10^{-8}$	$\delta 4.6 \times 10^{-6}$			$\alpha = .01$	$\delta = 1$
T_1QW		$\delta 6.2 \times 10^{-6}$				$\delta = 1$
$T_{23}QW$		$\delta 1.2 \times 10^{-5}$				$\delta = 1$
$T_{23}C$		$\delta 5.4 \times 10^{-6}$				$\delta = 1$
T_1QUV			$\gamma 7.5 \times 10^{-7}$	$\delta 7.5 \times 10^{-7}$	$\gamma = .5$	$\delta = .5$
CATEGORY TOTAL (1)	1.1×10^{-7}	3.4×10^{-5}	1.2×10^{-6}	1.4×10^{-6}		

(1) This is an unsmoothed total which includes the contribution from all the nondominant sequences not shown.

(2) Containment Failure Modes

α - Steam Explosion
β - Containment Leakage
γ - Hydrogen Burning
δ - Overpressurization

(3) Source of information - NUREG/CR-1659/4 of 4

Table 11 lists the dose in man-rems for the four release categories listed in Table 10. The information in Table 11 is taken from NUREG/CR-2800, "Guidelines for Nuclear Power Safety Issue Prioritization Information Development," Ref. (14). The doses in Table 11 are for the Braidwood site in Illinois. Ref. (14) states that the doses given in Table 11 can be used for other reactor sites since the calculated doses are nearly independent of reactor site.

TABLE 11. DOSE FOR THE FOUR RELEASE CATEGORIES GIVEN IN WASH-1400*

Release Category	Dose (Man-Rems)	Normalized Dose
BWR 1	1.6 E+6	.76
BWR 2	2.1 E+6	1.0
BWR 3	1.5 E+6	.71
BWR 4	1.8 E+5	.086

* From NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issues Prioritization Information Development"

For the GGNS site, the following expression for radiological release frequency, Man-Rems/yr, can be derived in terms of expression (1) and the information given in Tables 10 and 11:

$$RF = \sum_{RC} \sum_{CFM} \sum_{AS} ASF \times Pr(CFM/AS) \times Pr(RC/AS * CFM) \times D_{RC} \qquad (20)$$

where

RF	=	Release Frequency in Man-Rems/yr
CFM	=	Containment Failure Mode
RC	=	Release Category
AS	=	Accident Sequence
ASF	=	Accident Sequence Frequency yr^{-1}
Pr	=	Probability
D_{RC}	=	Man Rem Dose for release category RC
*	=	logical intersection (AND)

The accident sequence frequency, ASF is obtained by using expressions (1) or (2). The last row in Table 10 lists the release category frequency for the GGNS plant and is in essence the inner two summations in expression (18). We can obtain the release frequency for the GGNS site by taking the final outer summation, i.e.,

$$RF = 1.1 \times 10^{-7} \times 5.4 \times 10^{6} + 3.4 \times 10^{-5} \times 7.1 \times 10^{6} + 1.2 \times 10^{-6} \times 5.1 \times 10^{6} + 1.4 \times 10^{-6} \times 6.1 \times 10^{5}$$
$$= 249 \text{ Man-Rem/yr}$$

We can develop an importance expression based upon radiological frequency in the same manner as we did for accident frequency and core melt frequency. The importance expression for weighting according to radiological release frequency is:

$$I_{RF} = \frac{\text{Radiological release frequency of min cut sets containing the event of interest}}{\text{Radiological Release Frequency, RF}} \qquad (21)$$

The above importance expression is simply the fractional contribution of the min-cut-set release frequency (min cut sets containing either the initiating or enabling event) to the total accident frequency or radiological release frequency.

In reference to Section 4.7, we can develop safety ranking expressions using expressions (8) through (17). In this case, I_{RF} is substituted for I_j.

5.6 GENERAL METHODOLOGY FOR SAFETY RANKING

The purpose of the TENERA study, in part, was to develop a methodology for ranking the relative significance of planned plant modifications and design changes at the Grand Gulf Nuclear Station, GGNS. Part of the methodology is described in Sections 4 and 5 of this report.

The general approach in the TENERA study was to use quantitative ranking criteria, where possible, supplemented by established qualitative considerations, as necessary.

The most important accident sequences were selected from the RSSMAP study of GGNS, Ref. (3). Core melt frequency was computed using expression (2) in Section 4.4. The radiological release frequency was computed using expression (20). The importance of each basic event and system was ranked according to core melt frequency and radiological release frequency by using expressions (7) and (21). In addition, the risk reduction factor, expression (13), Section 2.7 was computed for each basic event and system for both core melt frequency and radiological release frequency.

The fault tree computer codes FTAP, Ref. (15) and IMPORTANCE, Ref. (16) were used to perform the rankings.

5.7 RANKINGS BASED ON CORE MELT FREQUENCY

Appendix A lists the dominant min cut sets for each accident sequence in the GGNS RSSMAP study. The basic events of Appendix A are designated by alphanumeric names. Table 12 provides a basic event description for each of these alphanumeric designations. As listed below, three initiating events appeared in the dominant min cut sets in the RSSMAP study:

TABLE 12. BASIC EVENTS AND SYSTEM FAILURE DESCRIPTIONS

Basic Event	System Failure Description
A-.01*	Steam explosion
BATA	DC battery A
BATB	DC battery B
BCACT	Residual heat removal, RHR, initiation logic circuit Loops B and C
C	Reactor Protection System
D-.5*	Containment failure -- overpressurization
D-1.*	Containment failure -- overpressurization
DIESEL1	TDI diesel 11
DIESEL2	TDI diesel 12
DIESEL3	High Pressure Core Spray, HPCS, diesel
G-.5*	Hydrogen burning
H	High Pressure Core Spray, HPCS, hardware, test and maintenance
HACT	High Pressure Core Spray, HPCS, initiation circuit
L	Low Pressure Core Spray, LPCS, hardware, test and maintenance
LA2	Low Pressure Coolant Injection, LPCI, Loop A, hardware, test and maintenance
LB1, 2	Low Pressure Coolant Injection, LPCI, Loop B, hardware, test and maintenance
LC	Low Pressure Coolant Injection, LPCI, Loop C, hardware, test and maintenance
LOPNRE	Nonrecovery of offsite power .5 hour
LOPNRL	Nonrecovery of offsite power 28 hours
LRACT	Low Pressure Core Spray, LPCS, and Residual Heat Removal, RHR, A initiation logic circuit
OP	Automatic Depressurization System, ADS, manual initiation
P	S/R valves
PA27	Residual Heat Removal, RHR, Pump A, hardware, test and maintenance
PB27	Residual Heat Removal, RHR, Pump B, hardware, test and maintenance
Q	Power conversion system
R	Reactor Core Isolation Cooling System, RCIC, hardware, test and maintenance

* Containment Failure Mode

** Dose Release

TABLE 12. (cont.)

Basic Event	System Failure Description
RACT	Reactor Core Isolation Cooling System, RCIC, initiation circuit
RECOVERY	Operator mitigation actions within 28 hours
R-1**	Units of release Category 1
R-2**	Units of release Category 2
R-3**	Units of release Category 3
R-4**	Units of release Category 4
S	Reactor Coolant Pressure Boundary, RCPB, piping (break size less than .1 ft^2)
SA	Upper pool dump Line A valve
SAACC	Upper pool dump Line A actuation and control circuit
SAC	Standby Service Water System, SSWS, actuation and control circuit Loop A
SB	Upper pool dump Line B valve
SBACC	Upper pool dump Line B actuation and control circuit
SBC	Standby service water system, SSWS, actuation and control circuit Loop B
SCC	Standby service water system, SSWS, actuation and control circuit Loop C
SCVA	Residual Heat Removal, RHR, Loop A steam cond. mode
SCVB	Residual Heat Removal, RHR, Loop B steam cond. mode
SSWA	Standby Service Water System, SSWS, Loop A
SSWB	Standby Service Water System, SSWS, Loop B
SSWC	Standby Service Water System, SSWS, Loop C
T1	Loss of offsite power, LOSP, transient
T23	Transient other than LOSP
V1	Standby Service Water System, SSWS, valves to TD1 diesel 11 cooling
V2	Standby Service Water System, SSWS, valves to TD1 diesel 12 cooling
V3	Standby Service Water System, SSWS, valves to HPCS DG cooling
VGA1, 2	Residual Heat Removal, RHR, Train A
VGB1, 2	Residual Heat Removal, RHR, Train B

* Containment Failure Mode

** Dose Release

Initiating Event	Frequency
T_1 (loss of offsite power, LOSP)	.2/yr
T_{23} (transient other than LOSP)	7/yr (loss of power conversion system)
S (small LOCA, less than .1ft^2)	1.4 E-3/yr

The dominant min cut sets excluded large LOCAs. A Boolean expression for core melt was obtained by taking the Boolean union of all min cut sets in Appendix A, (as described in Section 5.4). The total number of min cut sets in Appendix A is 349. RSSMAP generated a core melt frequency for GGNS as 3.4 E-5 per year. Ranking of basic events, i.e., systems, according to core melt frequency and the risk reduction ratio, R_j, are displayed in Table 13.

5.8 RANKINGS ACCORDING TO RADIOLOGICAL RELEASE FREQUENCY

The problem with ranking, using importance values or risk reduction ratios based on core melt frequency, is that sequences have different radiological releases and, hence, different consequences. For example, an early core melt in general results in a larger release than does a later core melt. An early core melt, for example, can be due to coolant makeup failure following a transient or LOCA. A late core melt can be due to failure of containment heat removal. Therefore, an additional ranking methodology based on radiological release frequency was established to address this issue.

In this section, we rank basic events according to radiological release frequency (release given in man-rems). Table 10 lists the dominant sequences according to the release categories given in WASH 1400. The containment failure modes for the accident sequences and the probability of the failure mode occurrence are also described in Table 10. Table 11 lists the man-rem doses for the four BWR release categories given in Table 10. The normalized doses in Table 11 are obtained by dividing the doses in column 1 by the largest release category dose, i.e., BWR 2 release category. The normalized doses were used in the importance rankings for release and correspond to the values for the events described as R-1, R-2, R-3 and R-4 in Table 12. The motivation for normalization of doses is that the computer code IMPORTANCE will not accept basic event probabilities that exceed unity; however, the relative ranking of basic events will not change using normalized doses.

The importance rankings of basic events based on normalized doses is given in Table 14. In addition, the risk reduction ratio with respect to normalized dose frequency is also given in Table 14.

5.9 METHODOLOGY BASED ON RANKING FACTORS

Beyond the ranking of potential plant changes based on risk related importance, a more qualitative approach can also be used to rank any remaining potential changes which may not be easily evaluated by the ranking methodology described above, the following more qualitative methodology is established.

TABLE 13. RANKING OF SYSTEMS ACCORDING TO CORE MELT FREQUENCY*

RANK	BASIC EVENT	SYSTEM FAILURE DESCRIPTION	SYSTEM IMPORTANCE I_i	RISK REDUCTION RATIO $R_i = (1 - I_i) + I_i r_i$
1	Recovery	Operator mitigation	.635	3.65 E-1 + .635 r_i
2	T23	Transient other than LOSP (Main Feedwater)	.6	4.0 E-1 + .6 r_i
3	Q	Power Conversion System	.449	5.51 E-1 + .449 r_i
4	VGA1, 2	RHR Train A	.301	6.99 E-1 + .301 r_i
4	VGB1, 2	RHR Train B	.301	6.99 E-1 + .301 r_i
5	SSWA	SSWS Loop A	.285	7.15 E-1 + .285 r_i
6	SSWB	SSWS Loop B	.284	7.16 E-1 + .284 r_i
7	T1	Loss of offsite power, LOSP, transient	.266	7.34 E-1 + .266 r_i
7	LOPNRE	Nonrecovery of offsite power .5 hour	.266	7.34 E-1 + .266 r_i
8	LOPNRL	Nonrecovery of offsite power 28 hours	.207	7.93 E-1 + .207 r_i
9	P	S/R valves	.171	8.29 E-1 + .171 r_i
10	C	Reactor Protection System	.157	8.430 E-1 + .157 r_i
11	S	RCPB piping (break size less than .1 ft^2)	.134	8.660 E-1 + .134 r_i
12	R	RCIC	.121	8.790 E-1 + .121 r_i
13	LB1, 2	LPCI Loop B	.107	8.930 E-1 + .107 r_i
13	DIESEL 1	TDI diesel 11	.107	8.930 E-1 + .107 r_i
14	DIESEL 2	TDI diesel 12	.105	8.950 E-1 + .105 r_i
15	LA2	LPCI Loop A	.481 E-1	9.519 E-1 + .481 E-1 r_i
16	H	HPCS	.308 E-1	9.692 E-1 + .308 E-1 r_i
17	DIESEL 3	HPCS diesel	.293 E-1	9.707 E-1 + .293 E-1 r_i
18	V1	SSWS valves to TDI diesel 11 cooling	.221 E-1	9.779 E-1 + .221 E-1 r_i
19	V2	SSWS valves to TDI diesel 12 cooling	.220 E-1	9.780 E-1 + .220 E-1 r_i
20	OP	ADS manual initiation	.212 E-1	9.788 E-1 + .212 E-1 r_i
21	SBC	SSWS actuation and control circuit Loop B	.206 E-1	9.794 E-1 + .206 E-1 r_i
21	SAC	SSWS actuation and control circuit Loop A	.206 E-1	9.794 E-1 + .206 E-1 r_i
22	BATA	DC battery A	.114 E-1	9.886 E-1 + .114 E-1 r_i
23	SSWC	SSWS Loop C	.112 E-1	9.888 E-1 + .112 E-1 r_i
24	LC	LPCI Loop C	.938 E-2	9.906 E-1 + .938 E-2 r_i
25	SA	Upper pool dump Line A valve	.869 E-2	9.913 E-1 + .869 E-2 r_i
25	SB	Upper pool dump Line B valve	.869 E-2	9.913 E-1 + .869 E-2 r_i
26	LRACT	LPCS and RHR A initiation logic circuit	.780 E-2	9.922 E-1 + .780 E-2 r_i
27	L	LPCS	.706 E-2	9.929 E-1 + .706 E-2 r_i
28	BCACT	RHR initiation logic circuit Loops B and C	.705 E-2	9.930 E-1 + .705 E-2 r_i
29	SCVA	RHR Loop A steam cond. mode	.561 E-2	9.944 E-1 + .561 E-2 r_i
29	SCVB	RHR Loop B steam cond. mode	.561 E-2	9.944 E-1 + .561 E-2 r_i
30	PA27	RHR Pump A	.267 E-2	9.973 E-1 + .267 E-2 r_i

*Core melt frequency = 3.4×10^{-5} per year.

TABLE 13. (cont.)

RANK	BASIC EVENT	SYSTEM FAILURE DESCRIPTION	SYSTEM IMPORTANCE I_i	RISK REDUCTION RATIO $R_i = (1 - I_i) + I_i r_i$
30	PB27	RHR Pump B	.267 E-2	9.973 E-1 + .267 E-2 r_i
31	HACT	HPCS, initiation circuit	.202 E-2	9.980 E-1 + .202 E-2 r_i
31	V3	SSWS valves to HPCS DG cooling	.202 E-2	9.980 E-1 + .202 E-2 r_i
32	SAACC	Upper pool dump Line A act. & control cct.	.125 E-2	9.988 E-1 + .125 E-2 r_i
32	SBACC	Upper pool dump Line B act. & control cct.	.125 E-2	9.988 E-1 + .125 E-2 r_i
33	BATB	DC battery B	.872 E-3	9.991 E-1 + .872 E-3 r_i
34	RACT	RCIC initiation circuit	.755 E-3	9.992 E-1 + .755 E-3 r_i
35	SCC	SSWS actuation and control circuit Loop C	.108 E-3	9.999 E-1 + .108 E-3 r_i

TABLE 14. RANKING OF SYSTEMS ACCORDING TO RELEASE FREQUENCY*

RANK	BASIC EVENT	SYSTEM FAILURE DESCRIPTION	SYSTEM IMPORTANCE I_i	RISK REDUCTION RATIO $R_i = (1 - I_i) + I_i r_i$
1	Recovery	Operator mitigation actions	.664	.336 + .664 r_i
2	T23	Transient other than LOSP (Main Feedwater)	.618	.382 + .618 r_i
3	Q	Power Conversion System	.457	.543 + .457 r_i
4	VGA1, 2	RHR Train A	.315	.685 + .315 r_i
4	VGB1, 2	RHR Train B	.315	.685 + .315 r_i
5	SSWA	SSWS Loop A	.291	.709 + .291 r_i
6	SSWB	SSWS Loop B	.290	.710 + .290 r_i
7	T1	Loss of offsite power, LOSP, transient	.241	.759 + .241 r_i
7	LOPNRE	Nonrecovery of offsite power .5 hour	.241	.759 + .241 r_i
8	LOPNRL	Nonrecovery of offsite power 28 hours	.216	.784 + .216 r_i
9	P	S/R valves	.166	.834 + .166 r_i
10	C	Reactor Protection System	.164	.836 + .164 r_i
11	S	RCPB piping (break size less than .1 ft^2)	.141	.859 + .141 r_i
12	DIESEL 1	TDI diesel 11	.992 E-1	.9008 + .992 E-1 r_i
13	DIESEL 2	TDI diesel 12	.987 E-1	.9013 + .987 E-1 r_i
14	R	RCIC	.863 E-1	.9137 + .863 E-1 r_i
15	LB 1, 2	LPCI Loop B	.553 E-1	.9447 + .553 E-1 r_i
16	LA2	LPCI Loop A	.506 E-1	.9494 + .506 E-1 r_i

*Normalized dose frequency = 3.2×10^{-5} per year. Unit dose = 7.1×10^{6} man·rem.

TABLE 14. (cont.)

RANK	BASIC EVENT	SYSTEM FAILURE DESCRIPTION	SYSTEM IMPORTANCE I_i	RISK REDUCTION RATIO $R_i = (1 - I_i) + I_i r_i$
17	SBC	SSWS actuation and control circuit Loop B	.216 E-1	.9784 + .216 E-1 r_i
17	SAC	SSWS actuation and control circuit Loop A	.216 E-1	.9784 + .216 E-1 r_i
18	V1	SSWS valves to TDI diesel 11 cooling	.207 E-1	.9793 + .207 E-1 r_i
18	V2	SSWS valves to TDI diesel 12 cooling	.207 E-1	.9793 + .207 E-1 r_i
19	H	HPCS	.130 E-1	.987 + .130 E-1 r_i
20	DIESEL 3	HPCS diesel	.124 E-1	.9876 + .124 E-1 r_i
21	SA	Upper pool dump Line A valve	.914 E-2	9.909 E-1 + .914 E-2 r_i
21	SB	Upper pool dump Line B valve	.914 E-2	9.909 E-1 + .914 E-2 r_i
22	OP	ADS manual initiation	.891 E-2	9.911 E-1 + .891 E-2 r_i
23	LRACT	HPCS and RHR A initiation logic circuit	.738 E-2	9.926 E-1 + .738 E-2 r_i
24	BCACT	RHR initiation logic circuit Loops B and C	.706 E-2	9.929 E-1 + .706 E-2 r_i
25	SCVA	RHR Loop A steam cond. mode	.586 E-2	9.941 E-1 + .586 E-2 r_i
25	SCVB	RHR Loop B steam cond. mode	.586 E-2	9.941 E-1 + .586 E-2 r_i
26	BATA	DC battery A	.533 E-2	9.947 E-1 + .533 E-2 r_i
27	SSWC	SSWS Loop C	.471 E-2	9.953 E-1 + .471 E-2 r_i
28	LC	LPCI Loop C	.395 E-2	9.961 E-1 + .395 E-2 r_i
29	L	LPCS	.297 E-2	9.970 E-1 + .297 E-2 r_i
30	PA27	RHR Pump A	.281 E-2	9.972 E-1 + .281 E-2 r_i
30	PB27	RHR Pump B	.281 E-2	9.972 E-1 + .281 E-2 r_i
31	SAACC	Upper pool dump Line A act. and control cct.	.131 E-2	9.987 E-1 + .131 E-2 r_i
31	SBACC	Upper pool dump Line B act. and control cct.	.131 E-2	9.987 E-1 + .131 E-2 r_i
32	BATB	DC battery B	.910 E-3	9.991 E-1 + .910 E-3 r_i
33	HACT	HPCS, initiation circuit	.850 E-3	9.992 E-1 + .850 E-3 r_i
34	V3	SSWS valves to HPCS DG cooling	.623 E-3	9.994 E-1 + .623 E-3 r_i
35	RACT	RCIC initiation circuit	.318 E-3	9.997 E-1 + .318 E-3 r_i
36	SCC	SSWS actuation and control circuit Loop C	.453 E-4	1 + .453 E-4 r_i

In developing this methodology, the safety and power generation design bases from the GGNS Final Safety Analysis Report, FSAR, were reviewed. From this review, a list was developed that can be used to identify plant functions that may be affected by potential plant changes.

This list was evaluated by qualified senior engineers, and each plant function in the list was assigned a ranking factor based on its perceived importance to plant safety, power generation, reliability and operability. Safety functions generally were assigned a higher ranking factor than power generation or reliability/operability functions; however, an attempt was made to indicate the relative importance of changes having a significant effect on plant reliability/operability. Based on the ranking factors determined, these functions were placed into individual groups which were then assigned ranking factors.

These group ranks and ranking factors are used to establish the relative ranking of planned changes with respect to selected criteria. The details of this methodology are described next.

5.10 RELATIVE RANKING OF PROPOSED/PLANNED MODIFICATIONS OR DESIGN CHANGES

In this section, we establish a safety ranking procedure for potential plant changes. Three categories for safety significance ranking are established, as described below.

CATEGORY 1

Category 1 changes have the highest priority in terms of relative safety significance and can be ranked by importance values or risk reduction ratios. The methodology described in Sections 5.5 and 5.6 allows an engineer to rank potential changes based on their importance to plant safety and their contribution to the decrease in system unavailability.

For category 1 changes, an engineer would review the potential change, identify the system/subsystem involved, and determine the importance value (or the risk reduction ratio) associated with that potential change. Once this has been completed, all potential changes can be ranked with Category 1.

CATEGORY 2

Ranked in importance below Category 1 are Category 2 changes. Category 1 encompasses the remaining safety-related changes and non-safety related changes important to safety, power generation, and plant reliability/operability which are not covered in Category 1.

Basically, an engineer would review the change, identify the plant function effected using the approach described in Section 5.9, and rank the change using the predetermined factors provided. Within Category 2, changes with the largest ranking factors would receive the highest priority.

Systems in Table 15 are ranked according to these criteria. The system with the highest ranking factor has the most important group rank in Table 15.

CATEGORY 3

Category 3 changes are other less significant, non-safety related changes which are not included in Category 1 or 2. These changes should be ranked after all Category 1 and 2 items. Engineering judgement and a determination of the availability of resources should be all that is necessary to rank these changes.

TABLE 15. CATEGORY 2 RANKING CRITERIA

Group Rank	Function Affected By Proposed Change	Ranking Factor
1	Improves the ability to remove energy from the primary containment to maintain the integrity of the containment system following accidents that release energy to the containment.	89
1	Improves the ability to automatically initiate the emergency core cooling systems, when required, regardless of the availability of offsite power supplies and the normal generating system of the station.	89
2	Improves any portion of the nuclear system that forms part of the reactor coolant pressure boundary, designed to retain integrity as a radioactive material containment barrier following abnormal operational transients and accidents.	84
2	Maintains the reactor coolant pressure boundary and core cooling capabilities.	84
2	Improves automatic actions immediately required in response to abnormal operational transients and accidents.	84
2	Improves control of active components of nuclear safety systems and engineered safety features from the control room.	84
2	Improves the ability of emergency core cooling systems to limit fuel cladding temperature and/or maintain fuel cladding integrity.	84
3	Improves ability to determine or prevent operations exceeding Safety Limits (Limiting Safety Systems Settings).	78
3	Improves the capacity of standby electrical power sources to power all nuclear safety systems and engineered safety features requiring electrical power.	78
3	Improves the availability of the standby electrical power sources to allow prompt reactor shutdown and decay heat removal under circumstances where normal auxiliary power is not available.	78

TABLE 15. (cont.)

Group Rank	Function Affected By Proposed Change	Ranking Factor
3	Improves the primary containment design and/or integrity.	78
3	Improves the ability of the emergency core cooling systems to provide for core cooling over the complete range of postulated break sizes in the reactor coolant pressure boundary.	78
3	Improves systems provided to remove decay heat from the containment.	78
3	Provides a major plant reliability/operability improvement.	78
4	Improves the ability to maintain the release of radioactive materials resulting from abnormal transients and accidents less than the requirements of 10 CFR 100.	74
4	Improves the capability to isolate piping that penetrates the primary containment and which could serve as a path for the uncontrolled release of radioactive material to the environs.	74
4	Improves reactor controls, including alarms, that allow the operator to rapidly assess the condition of the reactor system and locate system malfunctions.	74
5	Assures that essential safety actions are provided by equipment of sufficient redundance and independence. (i.e., no single failure of active components should prevent required actions)	70
5	Improves the design of nuclear safety systems and engineered safety features to accommodate natural environmental disturbances such as earthquakes, floods, and storms at the station site.	70
5	Improves control equipment provided to allow the reactor to respond automatically to minor load changes, major load changes and abnormal operational transients.	70
5	Improves the ability, on control room evacuation, to bring the reactor to hot shutdown conditions by using the local controls and equipment available outside the control room.	70
5	Improves the design of nuclear safety systems and engineered safety features to accommodate accident or transient induced dynamic loadings.	70

TABLE 15. (cont.)

Group Rank	Function Affected By Proposed Change	Ranking Factor
6	Improves the means by which plant operators are alerted when limits on the release of radioactive material are approached.	64
6	Improves the ability, on control room evacuation, to bring the reactor to cold shutdown conditions by utilizing the local controls and equipment available outside the control room.	64
6	Improves backup reactor shutdown capability.	64
6	Improves backup heat removal systems provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative.	64
6	Improves the ability of the fuel cladding, in conjunction with other plant systems, to retain integrity throughout the range of normal operational conditions and abnormal operational transients.	64
7	Assures that the reactor core is designed so its nuclear characteristics do not contribute to a divergent power transient.	60
7	Improves the secondary containment design and/or integrity to control release of radioactive materials from the primary containment.	60
7	Improves the control room shielding against radiation so that continued occupancy under accident conditions is possible.	60
7	Improves the design of the fuel handling and storage facilities to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel.	60
7	Enhances the ability to manually control reactor power level.	60
8	Improves the ability to maintain the normal release of radioactive materials significantly less than the requirements of 10 CFR 20.	55

TABLE 15. (cont.)

Group Rank	Function Affected By Proposed Change	Ranking Factor
8	Provides a moderate plant reliability/operability improvement or provides a major system reliability/ operability improvement.	55
8	Improves steam production for direct use in a turbine-generator unit.	55
9	Improves the ability to demonstrate functional performance requirements for nuclear safety systems and engineered safety features.	50
9	Improves the ability of the fuel cladding to accommodate, without loss of integrity, the pressures generated by fission gases throughout the design life of the fuel.	50
9	Enhances the ability to control the reactor from a single location.	50
10	Improves the gaseous disposal facilities to enhance the ability to discharge radioactive effluents or ship radioactive materials offsite in accordance with applicable regulations.	45
10	Improves the liquid disposal facilities to enhance the ability to discharge radioactive effluents or ship radioactive materials offsite in accordance with applicable regulations.	45
10	Improves the ability to test primary containment integrity and leak tightness at periodic intervals.	45
11	Improves the solid waste disposal facilities to enhance the ability to discharge radioactive effluents or ship radioactive materials offsite in accordance with applicable regulations.	40
11	Improves on access control established to allow control of radiation doses within the limits of applicable regulations.	40
11	Improves ability to prevent sabotage or physical security threat to the plant.	40
12	Provides a minor plant or system reliability/operability improvement.	27

5.11 APPLICATION OF THE SAFETY RANKING METHODOLOGY

As described in Section 4.7, the concept of risk reduction is useful for conducting tradeoff studies. Computation of risk reduction ratios generate numerical rankings that determine the system and/or component failures that dominate the risk. Such a ranking can suggest where hardware, software, human factors component design changes or maintenance policy changes can be implemented to improve plant safety.

As an example of the use of risk reduction ratios, we choose two systems, the reactor core isolation cooling system (RCIC) and the division I diesel denoted respectively in Table 12 by the alphanumeric names R and DIESEL1. Examining Table 13, we see that RCIC is more important than diesel 1 when ranked according to core melt frequency. However, Table 14 tells us that the ranking of diesel 1 is more important when ranked according to release frequency. Tables 16 and 17 list respectively the component unavailabilities for RCIC and for diesel 1.

For this example, we choose two options which would reduce the risk at GGNS--

- o delay maintenance on all motor operated valves within RCIC until the refueling outage (option 1)

- o delay maintenance on diesel 1 until the refueling outage (option 2).

These options eliminate the maintenance contribution of these components to the total system unavailability. There are a total of six motor operated valves within RCIC. It is assumed that a core melt accident at full power can not occur during refueling. We assume that the cost to implement these options is negligible.

The system improvement ratio for RCIC is

$$Q(new)/Q(old) = (5.1 \times 10^{-2} - 6 \times 5.8 \times 10^{-3})/5.1\ E\text{-}2$$

$$= 0.31.$$

As demonstrated in Table 16, the maintenance contribution for one MOV is 5.8 E-3.

For diesel 1 the system improvement ratio is

$$Q(new)/Q(old) = (3.6 \times 10^{-2} - 6.4 \times 10^{-3})/\ 3.6 \times 10^{-3}$$

$$= 0.82.$$

Table 18 displays the results of these options. It is seen that option 1 has a greater effect of reducing either the core melt frequency or release frequency and should be the policy implemented if the utility seeks the greatest reduction in risk.

TABLE 16. COMPONENT UNAVAILABILITIES FOR THE REACTOR CORE ISOLATION COOLING SYSTEM (RCIC)*

Component Description	Fault Identifiers	Failure Contributors	Q/Component
Check Valve	F066 F065 F204 F011 F040	Hardware	1.0 E-4
		Q Total	1.0 E-4
Manual Valve (Normally Locked Open)	F200 F016	Operator Error	1.0 E-4
		Plugged	1.0 E-4
		Q Total	2.0 E-4
Motor Operated Valve (Normally Open)	F068-A F063-B F064-A F010-A	Plugged	1.0 E-4
		Maintenance	5.8 E-3
		Q Total(per valve)	5.9 E-3
Motor Operated Valve (Normally Closed)	F013-A F045-A	Hardware	1.0 E-3
		Plugged	1.0 E-4
		Maintenance	5.8 E-3
		Control Circuit	3.0 E-4
		Q Total(per valve)	7.2 E-3
Pump	C001	Hardware	1.0 E-3
		Control Circuit	1.0 E-3
		Maintenance	5.8 E-3
		Q Total	7.8 E-3
RCIC Turbine	C002	Fails to function	1.0 E-3
		Q Total	1.0 E-3
Trip Throttle	TTV	Fails to function	1.3 E-3
		Q Total	1.3 E-3
Turbine Governing	TGV	Fails to function	2.2 E-3
		Q Total	2.2 E-3
Total RCIC Component Unavailability			5.1 E-2

*Term R in Table 12

TABLE 17. COMPONENT UNAVAILABILITIES FOR DIVISION 1 DIESEL*

Diesel	Start Failure	3.0 E-2
	Maintenance	6.4 E-3
	Total	3.6 E-2

* Term DIESEL1 in Table 1

TABLE 18. RCIC AND DIVISION 1 DIESEL RISK REDUCTION RATIOS

System	System Improvement Ratio (r)	Risk Reduction Ratio (RRR) Expression	RRR Value	Risk Measure	Percentage Reduction in Risk Measure*
RCIC	.31	.88 + .12 r	.92	Core Melt Frequency	8%
RCIC	.31	.91 + .086 r	.94	Release Frequency	6%
Diesel 1	.82	.89 + .11 r	.98	Core Melt Frequency	2%
Diesel 1	.82	.90 + .10 r	.98	Release Frequency	2%

*Expression 17, section 4.7

6. INTERPRETATION OF RESULTS

This case study shows how a utility can make use of the results of a PSA to rank components and systems that are important to plant safety. It is important to note that this case study was conducted in a short time period and with a limited budget. The study suffered the same limitations as RSSMAP, i.e., RSSMAP did not consider external events; it did not conduct detailed human reliability assessments or uncertainty analysis. Nonetheless, the utility was able to conduct tradeoffs as described in Section 5.11. The case study gave insight as to which system modifications were to be conducted at the next refueling and which modifications could be delayed.

Examining Tables 13 and 14, we see that a PRA can provide much useful information for ranking the importance of systems and components with respect to plant safety. Much of the risk data to perform the RSSMAP PRA was taken from WASH 1400. It is important to note that the methodology described in this paper can be extended to include dependent-event analysis, i.e., quantitative common-cause analysis.

In addition, this paper places emphasis on the qualitative decision-making process which is important when no risk data are available.

We believe that the methodology described in Section 2 of this report has wide applicability to the nuclear power industry. We recommend that the safety ranking process be based on radiological release. For example, in a PWR, containment systems in general do little to mitigate core melt, but are important in reducing the consequences of radiological release.

It must be noted, however, that events due to routine release should be given a different risk significance than releases due to core melt. When considering releases from both routine release and core melt accidents, one should use a multi-attribute utility approach as described in reference (17).

In addition, there are other important issues concerning risk significance not discussed in this paper. A risk model is always sensitive to the assumptions made when constructing and evaluating the model. One should test the effect of various assumptions on the model to determine the risk significance of the assumptions. Assumptions can include issues such as (1) operator recovery; (2) operator errors of omission and commission; (3) the sensitivity of equipment to environmental conditions, and (4) success criteria, i.e., realistic versus FSAR.

Appendix A

CUT SETS FOR THE GRAND GULF DOMINANT ACCIDENT SEQUENCES

The cut sets that contribute approximately 90% or more to the total of each dominant accident sequence frequency are listed below. Maintenance contributions to the cut set frequencies which would violate technical specifications have been removed when doing so will significantly affect the results.

Sequence T_1PQI

Cut Set	Frequency
T_1*P*LOPNRE*LOPNRL*DIESEL1*DIESEL2*RECOVERY	1.2×10^{-7}
T_1*P*LOPNRE*LOPNRL*VGA2*DIESEL2*RECOVERY	7.9×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGB2*DIESEL1*RECOVERY	7.9×10^{-8}
T_1*P*LOPNRE*LOPNRL*DIESEL1*SSB*RECOVERY	7.0×10^{-8}
T_1*P*LOPNRE*LOPNRL*DIESEL2*SSA*RECOVERY	7.0×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGA2*VGB2*RECOVERY	5.3×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGA1*DIESEL2*RECOVERY	5.0×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGB1*DIESEL1*RECOVERY	5.0×10^{-8}
T_1*P*LOPNRE*LOPNRL*SB*DIESEL1*RECOVERY	4.6×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGA2*SSB*RECOVERY	4.6×10^{-8}
T_1*P*LOPNRE*LOPNRL*LA2*DIESEL2*RECOVERY	4.6×10^{-8}
T_1*P*LOPNRE*LOPNRL*SA*DIESEL2*RECOVERY	4.6×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGB2*SSA*RECOVERY	4.6×10^{-8}
T_1*P*LOPNRE*LOPNRL*LB2*DIESEL1*RECOVERY	4.6×10^{-8}
T_1*P*LOPNRE*LOPNRL*SSA*SSB*RECOVERY	4.1×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGA1*VGB2*RECOVERY	3.3×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGA2*VGB1*RECOVERY	3.3×10^{-8}
T_1*P*LOPNRE*LOPNRL*LA2*VGB2*RECOVERY	3.1×10^{-8}
T_1*P*LOPNRE*LOPNRL*LB2*VGA2*RECOVERY	3.1×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGA1*SSB*RECOVERY	2.9×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGB1*SSA*RECOVERY	2.9×10^{-8}
T_1*P*LOPNRE*LOPNRL*LA2*SSB*RECOVERY	2.7×10^{-8}
T_1*P*LOPNRE*LOPNRL*SA*SSB*RECOVERY	2.7×10^{-8}
T_1*P*LOPNRE*LOPNRL*SB*SSA*RECOVERY	2.7×10^{-8}
T_1*P*LOPNRE*LOPNRL*LB2*SSA*RECOVERY	2.7×10^{-8}
T_1*P*LOPNRE*LOPNRL*DIESEL1*V2*RECOVERY	2.6×10^{-8}
T_1*P*LOPNRE*LOPNRL*DIESEL2*V1*RECOVERY	2.6×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGA1*VGB1*RECOVERY	2.1×10^{-8}
T_1*P*LOPNRE*LOPNRL*LA2*VGB1*RECOVERY	1.9×10^{-8}
T_1*P*LOPNRE*LOPNRL*LB2*VGA1*RECOVERY	1.9×10^{-8}
T_1*P*LOPNRE*LOPNRL*SA*SB*RECOVERY	1.8×10^{-8}
T_1*P*LOPNRE*LOPNRL*LA2*LB2*RECOVERY	1.8×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGA2*V2*RECOVERY	1.8×10^{-8}

Cut Set	Frequency
T_1*P*LOPNRE*LOPNRL*VGB2*V1*RECOVERY	1.8×10^{-8}
T_1*P*LOPNRE*LOPNRL*SSA*V2*RECOVERY	1.5×10^{-8}
T_1*P*LOPNRE*LOPNRL*SSB*V1*RECOVERY	1.5×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGA1*V2*RECOVERY	1.1×10^{-8}
T_1*P*LOPNRE*LOPNRL*VGB1*V1*RECOVERY	1.1×10^{-8}
T_1*P*LOPNRE*LOPNRL*LA2*V2*RECOVERY	1.0×10^{-8}
T_1*P*LOPNRE*LOPNRL*SA*V2*RECOVERY	1.0×10^{-8}
T_1*P*LOPNRE*LOPNRL*SB*V1*RECOVERY	1.0×10^{-8}
T_1*P*LOPNRE*LOPNRL*LB2*V1*RECOVERY	1.0×10^{-9}
T_1*P*LOPNRE*LOPNRL*V1*V2*RECOVERY	5.9×10^{-8}

Sequence T_{23}PQI

T_{23}*P*Q*VGA2*VGB2*RECOVERY	5.0×10^{-7}
T_{23}*P*Q*VGB2*SSA*RECOVERY	3.4×10^{-7}
T_{23}*P*Q*VGA2*SSB*RECOVERY	3.4×10^{-7}
T_{23}*P*Q*VGA2*VGB1*RECOVERY	2.5×10^{-7}
T_{23}*P*Q*VGA1*VGB2*RECOVERY	2.5×10^{-7}
T_{23}*P*Q*VGA2*LB2*RECOVERY	2.3×10^{-7}
T_{23}*P*Q*VGB2*LA2*RECOVERY	2.3×10^{-7}
T_{23}*P*Q*SSA*SSB*RECOVERY	1.6×10^{-7}
T_{23}*P*Q*VGB1*SSA*RECOVERY	1.3×10^{-7}
T_{23}*P*Q*VGA1*SSB*RECOVERY	1.3×10^{-7}
T_{23}*P*Q*LB2*SSA*RECOVERY	1.0×10^{-7}
T_{23}*P*Q*LA2*SSB*RECOVERY	1.0×10^{-7}
T_{23}*P*Q*VGA1*VGB1*RECOVERY	1.0×10^{-8}
T_{23}*P*Q*VGA1*LB2*RECOVERY	8.5×10^{-8}
T_{23}*P*Q*VGB1*LA2*RECOVERY	8.5×10^{-8}
T_{23}*P*Q*SA*SB*RECOVERY	6.9×10^{-8}
T_{23}*P*Q*LA2*LB2*RECOVERY	6.9×10^{-8}
T_{23}*P*Q*VGA2*SBC*RECOVERY	3.2×10^{-8}
T_{23}*P*Q*VGA2*BCACT*RECOVERY	3.2×10^{-8}
T_{23}*P*Q*VGB2*SAC*RECOVERY	3.2×10^{-8}
T_{23}*P*Q*VGB2*LRACT*RECOVERY	3.2×10^{-8}
T_{23}*P*Q*SAC*SSB*RECOVERY	2.8×10^{-8}
T_{23}*P*Q*SSB*LRACT*RECOVERY	2.8×10^{-8}
T_{23}*P*Q*SBC*SSA*RECOVERY	2.8×10^{-8}
T_{23}*P*Q*SSA*BCACT*RECOVERY	2.8×10^{-8}
T_{23}*P*Q*PA27*VGB2*RECOVERY	2.2×10^{-8}
T_{23}*P*Q*PB27*VGA2*RECOVERY	2.2×10^{-8}
T_{23}*P*Q*VGA1*BCACT*RECOVERY	2.0×10^{-8}
T_{23}*P*Q*VGA1*SBC*RECOVERY	2.0×10^{-8}
T_{23}*P*Q*LRACT*VGB1*RECOVERY	2.0×10^{-8}
T_{23}*P*Q*VGB1*SAC*RECOVERY	2.0×10^{-8}
T_{23}*P*Q*PA27*SSB*RECOVERY	1.9×10^{-8}
T_{23}*P*Q*PB27*SSA*RECOVERY	1.9×10^{-9}
T_{23}*P*Q*SAACC*SB*RECOVERY	1.9×10^{-8}
T_{23}*P*Q*LRACT*LB2*RECOVERY	1.9×10^{-8}
T_{23}*P*Q*LA2*SBC*RECOVERY	1.9×10^{-8}
T_{23}*P*Q*LB2*SAC*RECOVERY	1.9×10^{-8}
T_{23}*P*Q*LA2*BCACT*RECOVERY	1.9×10^{-8}
T_{23}*P*Q*SA*SBACC*RECOVERY	1.9×10^{-8}

Sequence T_1PQE

Cut Set	Frequency
T_1*P*Q*OP*H*R	1.1×10^{-8}
T_1*P*Q*OP*LOPNRE*DIESEL3*R	1.1×10^{-8}
T_1*P*Q*LOPNRE*DIESEL1*DIESEL2*DIESEL3*R	9.5×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*DIESEL3*R*LC	5.8×10^{-9}
T_1*P*Q*LOPNRE*DIESEL2*DIESEL3*L*R	5.6×10^{-9}
T_1*P*Q*LOPNRE*SSA*DIESEL2*DIESEL3*R	5.6×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*DIESEL2*H*R	5.6×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*SSB*DIESEL3*R	5.6×10^{-9}
T_1*P*Q*LOPNRE*BATA*DIESEL2*DIESEL3	5.2×10^{-9}
T_1*P*Q*OP*LOPNRE*SSC*R	4.3×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*DIESEL2*SSC*R	3.7×10^{-9}
T_1*P*Q*LOPNRE*LB2*DIESEL1*DIESEL3*R	3.7×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*DIESEL3*R*LB1	3.4×10^{-9}
T_1*P*Q*LOPNRE*SSA*DIESEL3*R*LC	3.4×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*H*R*LC	3.4×10^{-9}
T_1*P*Q*LOPNRE*SSB*DIESEL3*L*R	3.2×10^{-9}
T_1*P*Q*LOPNRE*L*H*R*DIESEL2	3.2×10^{-9}
T_1*P*Q*LOPNRE*SSA*DIESEL2*H*R	3.2×10^{-9}
T_1*P*Q*LOPNRE*SSA*SSB*DIESEL3*R	3.2×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*SSB*H*R	3.2×10^{-9}
T_1*P*Q*BATA*LOPNRE*DIESEL3*LC	3.2×10^{-9}
T_1*P*Q*BATA*LOPNRE*DIESEL2*H	3.0×10^{-9}
T_1*P*Q*BATA*LOPNRE*SSB*DIESEL3	3.0×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*SSC*R*LC	2.3×10^{-9}
T_1*P*Q*LOPNRE*DIESEL2*SSC*L*R	2.2×10^{-9}
T_1*P*Q*LOPNRE*SSA*DIESEL2*SSC*R	2.2×10^{-9}
T_1*P*Q*LOPNRE*LB2*SSA*DIESEL3*R	2.2×10^{-9}
T_1*P*Q*LOPNRE*LB2*DIESEL1*H*R	2.2×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*SSB*SSC*R	2.2×10^{-9}
T_1*P*Q*LOPNRE*V1*DIESEL2*DIESEL3*R	2.1×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*V2*DIESEL3*R	2.1×10^{-9}
T_1*P*Q*BATA*LOPNRE*DIESEL2*SSC	2.0×10^{-9}
T_1*P*Q*BATA*LOPNRE*LB2*DIESEL3	2.0×10^{-9}
T_1*P*Q*LOPNRE*SSA*DIESEL3*R*LB1	2.0×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*H*R*LB1	2.0×10^{-9}
T_1*P*Q*LOPNRE*SSA*H*R*LC	2.0×10^{-9}
T_1*P*Q*LOPNRE*SSB*L*H*R	1.9×10^{-9}
T_1*P*Q*LOPNRE*SSA*SSB*H*R	1.9×10^{-9}
T_1*P*Q*BATA*LOPNRE*DIESEL3*LB1	1.9×10^{-9}
T_1*P*Q*OP*HACT*R	1.8×10^{-9}
T_1*P*Q*BATA*LOPNRE*H*LC	1.8×10^{-9}
T_1*P*Q*BATA*LOPNRE*SSB*H	1.8×10^{-9}
T_1*P*Q*LOPNRE*LB2*DIESEL1*SSC*R	1.4×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*SSC*R*LB1	1.3×10^{-9}
T_1*P*Q*LOPNRE*SSA*SSC*R*LC	1.3×10^{-9}
T_1*P*Q*LOPNRE*V1*DIESEL3*R*LC	1.3×10^{-9}
T_1*P*Q*LOPNRE*SSB*SSC*L*R	1.3×10^{-9}
T_1*P*Q*LOPNRE*LB2*SSA*H*R	1.3×10^{-9}
T_1*P*Q*LOPNRE*SSA*SSB*SSC*R	1.3×10^{-9}
T_1*P*Q*LOPNRE*V2*DIESEL3*L*R	1.2×10^{-9}
T_1*P*Q*LOPNRE*SSA*V2*DIESEL3*R	1.2×10^{-9}

Cut Set	Frequency
T_1*P*Q*LOPNRE*V1*DIESEL2*H*R	1.2×10^{-9}
T_1*P*Q*LOPNRE*V1*SSB*DIESEL3*R	1.2×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*V2*H*R	1.2×10^{-9}
T_1*P*Q*BATA*LOPNRE*SSC*LC	1.2×10^{-9}
T_1*P*Q*BATA*LOPNRE*SSB*SSC	1.2×10^{-9}
T_1*P*Q*BATA*LOPNRE*LB2*H	1.2×10^{-9}
T_1*P*Q*LOPNRE*SSA*H*R*LB1	1.2×10^{-9}
T_1*P*Q*BATA*LOPNRE*V2*DIESEL3	1.2×10^{-9}
T_1*P*Q*BATA*LOPNRE*H*LB1	1.1×10^{-9}
T_1*P*Q*LOPNRE*DIESEL1*DIESEL2*V3*R	8.7×10^{-10}
T_1*P*Q*LOPNRE*LB2*SSA*SSC*R	8.4×10^{-10}
T_1*P*Q*LOPNRE*V1*DIESEL2*SSC*R	8.2×10^{-10}
T_1*P*Q*LOPNRE*LB2*V1*DIESEL3*R	8.2×10^{-10}
T_1*P*Q*LOPNRE*DIESEL1*V2*SSC*R	8.2×10^{-10}
T_1*P*Q*BATA*LOPNRE*LB2*SSC	7.8×10^{-10}
T_1*P*Q*LOPNRE*SSA*SSC*R*LB1	7.8×10^{-10}
T_1*P*Q*LOPNRE*V1*DIESEL3*R*LB1	7.6×10^{-10}
T_1*P*Q*LOPNRE*V1*H*R*LC	7.5×10^{-10}
T_1*P*Q*BATA*LOPNRE*SSC*LB1	7.3×10^{-10}
T_1*P*Q*LOPNRE*V2*L*H*R	7.2×10^{-10}
T_1*P*Q*LOPNRE*SSA*V2*H*R	7.2×10^{-10}
T_1*P*Q*LOPNRE*V1*SSB*H*R	7.2×10^{-10}
T_1*P*Q*BATA*LOPNRE*V2*H	6.7×10^{-10}
T_1*P*Q*LRACT*H*R*LC	5.7×10^{-10}
T_1*P*Q*BCACT*L*H*R	5.4×10^{-10}
T_1*P*Q*LOPNRE*DIESEL1*V3*R*LC	5.3×10^{-10}
T_1*P*Q*LOPNRE*DIESEL2*V3*L*R	5.1×10^{-10}
T_1*P*Q*LOPNRE*SSA*DIESEL2*V3*R	5.1×10^{-10}
T_1*P*Q*LOPNRE*DIESEL1*SSB*V3*R	5.1×10^{-10}
T_1*P*Q*LOPNRE*V1*SSC*R*LC	5.0×10^{-10}
T_1*P*Q*LOPNRE*V2*SSC*L*R	4.8×10^{-10}
T_1*P*Q*LOPNRE*SSA*V2*SSC*R	4.8×10^{-10}
T_1*P*Q*LOPNRE*LB2*V1*H*R	4.8×10^{-10}
T_1*P*Q*LOPNRE*V1*SSB*SSC*R	4.8×10^{-10}
T_1*P*Q*BATA*LOPNRE*DIESEL2*V3	4.8×10^{-10}
Sequence T_{23}PQE	
T_{23}*P*Q*OP*R*H	3.8×10^{-7}
T_{23}*P*Q*OP*R*HACT	6.4×10^{-8}
T_{23}*P*Q*OP*RACT*H	2.6×10^{-8}
T_{23}*P*Q*R*LRACT*H*LC	2.0×10^{-8}
T_{23}*P*Q*R*BCACT*L*H	1.9×10^{-8}
T_{23}*P*Q*R*LRACT*LB2*H	1.3×10^{-8}
T_{23}*P*Q*R*LRACT*H*LB1	1.2×10^{-8}
Sequence SI	
S*VGA2*VGB2	6.2×10^{-7}
S*VGB2*SSA	4.2×10^{-7}
S*VGA2*SSB	4.2×10^{-7}
S*VGA2*VGB1	3.2×10^{-7}
S*VGA1*VGB2	3.2×10^{-7}
S*VGA2*LB2	2.8×10^{-7}

Cut Set	Frequency
S*LA2*VGB2	2.8×10^{-7}
S*SSA*SSB	1.9×10^{-7}
S*VGB1*SSA	1.6×10^{-7}
S*VGA1*SSB	1.6×10^{-7}
S*LB2*SSA	1.3×10^{-7}
S*LA2*SSB	1.3×10^{-7}
S*VGA1*VGB1	1.3×10^{-7}
S*VGA1*LB2	1.1×10^{-7}
S*LA2*VGB1	1.1×10^{-7}
S*SA*SB	8.6×10^{-8}
S*LA2*LB2	8.6×10^{-8}
S*VGA2*SBC	4.0×10^{-8}
S*VGA2*BCACT	4.0×10^{-8}
S*VGB2*SAC	4.0×10^{-8}
S*VGB2*LRACT	4.0×10^{-8}
S*SAC*SSB	3.5×10^{-8}
S*LRACT*SSB	3.5×10^{-8}
S*SSA*SBC	3.5×10^{-8}
S*BCACT*SSA	3.5×10^{-8}
S*PA27*VGB2	2.7×10^{-8}
S*VGA2*PB27	2.7×10^{-8}
S*VGA1*BCACT	2.5×10^{-8}
S*VGA1*SBC	2.5×10^{-8}
S*LRACT*VGB1	2.5×10^{-8}
S*VGB1*SAC	2.5×10^{-8}
S*PA27*SSB	2.4×10^{-8}
S*PB27*SSA	2.4×10^{-8}
S*SAACC*SB	2.4×10^{-8}
S*LRACT*LB2	2.4×10^{-8}
S*LA2*SBC	2.4×10^{-8}
S*LB2*SAC	2.4×10^{-8}
S*LA2*BCACT	2.4×10^{-8}
S*SA*SBACC	2.4×10^{-8}

Sequence T_1QW

T_1*LOPNRE*LOPNRL*DIESEL1*DIESEL2*RECOVERY	1.1×10^{-6}
T_1*LOPNRE*LOPNRL*SSA*DIESEL2*RECOVERY	6.4×10^{-7}
T_1*LOPNRE*LOPNRL*DIESEL1*SSB*RECOVERY	6.4×10^{-7}
T_1*LOPNRE*LOPNRL*VGB1*DIESEL1*RECOVERY	4.5×10^{-7}
T_1*LOPNRE*LOPNRL*VGA1*DIESEL2*RECOVERY	4.5×10^{-7}
T_1*LOPNRE*LOPNRL*SSA*SSB*RECOVERY	3.7×10^{-7}
T_1*LOPNRE*LOPNRL*VGB1*SSA*RECOVERY	2.6×10^{-7}
T_1*LOPNRE*LOPNRL*VGA1*SSB*RECOVERY	2.6×10^{-7}
T_1*LOPNRE*LOPNRL*V1*DIESEL2*RECOVERY	2.4×10^{-7}
T_1*LOPNRE*LOPNRL*DIESEL1*V2*RECOVERY	2.4×10^{-7}
T_1*LOPNRE*LOPNRL*VGA1*VGB1*RECOVERY	1.9×10^{-7}
T_1*LOPNRE*LOPNRL*SSA*V2*RECOVERY	1.4×10^{-7}
T_1*LOPNRE*LOPNRL*V1*SSB*RECOVERY	1.4×10^{-7}
T_1*LOPNRE*LOPNRL*VGB1*V1*RECOVERY	1.0×10^{-7}
T_1*LOPNRE*LOPNRL*VGA1*V2*RECOVERY	1.0×10^{-8}
T_1*LOPNRE*LOPNRL*V1*V2*RECOVERY	5.4×10^{-8}
T_1*LOPNRE*LOPNRL*VGA2*DIESEL2*R*RECOVERY	3.7×10^{-8}

Cut Set	Frequency
T_1*LOPNRE*LOPNRL*VGB2*DIESEL1*R*RECOVERY	3.7×10^{-8}
T_1*LOPNRE*LOPNRL*SAC*DIESEL2*RECOVERY	3.6×10^{-8}
T_1*LOPNRE*LOPNRL*DIESEL1*SBC*RECOVERY	3.6×10^{-8}
T_1*LOPNRE*LOPNRL*BATB*DIESEL1*RECOVERY	3.0×10^{-8}
T_1*LOPNRE*LOPNRL*BATA*DIESEL2*RECOVERY	3.0×10^{-8}
T_1*LOPNRE*LOPNRL*VGA2*VGB2*R*RECOVERY	2.5×10^{-8}
T_1*LOPNRE*LOPNRL*VGB2*SCVB*DIESEL1*RECOVERY	2.3×10^{-8}
T_1*LOPNRE*LOPNRL*VGA2*SCVA*DIESEL2*RECOVERY	2.3×10^{-8}
Sequence T_{23}QW	
T_{23}*Q*SSA*SSB*RECOVERY	3.2×10^{-6}
T_{23}*Q*VGB1*SSA*RECOVERY	1.9×10^{-6}
T_{23}*Q*VGA1*SSB*RECOVERY	1.9×10^{-6}
T_{23}*Q*VGA1*VGB1*RECOVERY	9.4×10^{-7}
T_{23}*Q*VGA2*VGB2*R*RECOVERY	3.0×10^{-7}
T_{23}*Q*VGA2*SSB*R*RECOVERY	2.6×10^{-7}
T_{23}*Q*VGB2*SSA*R*RECOVERY	2.6×10^{-7}
T_{23}*Q*SSA*SBC*RECOVERY	2.6×10^{-7}
T_{23}*Q*SAC*SSB*RECOVERY	2.6×10^{-7}
T_{23}*Q*VGA2*VGB1*R*RECOVERY	1.9×10^{-7}
T_{23}*Q*VGA1*VGB2*R*RECOVERY	1.9×10^{-7}
T_{23}*Q*VGB1*SAC*RECOVERY	1.9×10^{-7}
T_{23}*Q*VGA1*SBC*RECOVERY	1.9×10^{-7}
T_{23}*Q*LA2*VGB2*R*RECOVERY	1.8×10^{-7}
T_{23}*Q*VGA2*LB2*R*RECOVERY	1.8×10^{-7}
T_{23}*Q*VGB2*SCVB*SSA*RECOVERY	1.7×10^{-7}
T_{23}*Q*VGA2*SCVA*SSB*RECOVERY	1.7×10^{-7}
T_{23}*Q*LA2*SSB*R*RECOVERY	1.5×10^{-7}
T_{23}*Q*LB2*SSA*R*RECOVERY	1.5×10^{-7}
Sequence T_{23}C	
T_{23}*C	5.4×10^{-6}
Sequence T_1QUV	
T_1*LOPNRE*OP*R*DIESEL3	1.1×10^{-7}
T_1*LOPNRE*R*DIESEL1*DIESEL2*DIESEL3	9.5×10^{-8}
T_1*LOPNRE*OP*R*H	6.4×10^{-8}
T_1*LOPNRE*R*DIESEL1*DIESEL3*LC	5.8×10^{-8}
T_1*LOPNRE*R*SSA*DIESEL2*DIESEL3	5.6×10^{-8}
T_1*LOPNRE*R*DIESEL1*SSB*DIESEL3	5.6×10^{-8}
T_1*LOPNRE*R*DIESEL1*DIESEL2*H	5.6×10^{-8}
T_1*LOPNRE*R*DIESEL2*DIESEL3*L	5.6×10^{-8}
T_1*LOPNRE*BATA*DIESEL2*DIESEL3	5.2×10^{-8}
T_1*LOPNRE*OP*R*SSC	4.3×10^{-8}
T_1*LOPNRE*R*DIESEL1*DIESEL2*SSC	3.7×10^{-8}
T_1*LOPNRE*R*LB2*DIESEL1*DIESEL3	3.7×10^{-8}
T_1*LOPNRE*R*LB1*DIESEL1*DIESEL3	3.4×10^{-8}
T_1*LOPNRE*R*SSA*DIESEL3*LC	3.4×10^{-8}
T_1*LOPNRE*R*DIESEL1*H*LC	3.4×10^{-8}
T_1*LOPNRE*R*SSA*SSB*DIESEL3	3.2×10^{-8}

Cut Set	Frequency
T_1*LOPNRE*R*SSA*DIESEL2*H	3.2×10^{-8}
T_1*LOPNRE*R*SSB*DIESEL1*H	3.2×10^{-8}
T_1*LOPNRE*R*SSB*DIESEL3*L	3.2×10^{-8}
T_1*LOPNRE*R*DIESEL2*L*H	3.2×10^{-8}
T_1*LOPNRE*BATA*DIESEL3*LC	3.2×10^{-8}
T_1*LOPNRE*BATA*SSB*DIESEL3	3.0×10^{-8}
T_1*LOPNRE*BATA*DIESEL2*H	3.0×10^{-8}
T_1*LOPNRE*R*DIESEL1*SSC*LC	2.3×10^{-8}
T_1*LOPNRE*R*SSA*DIESEL2*SSC	2.2×10^{-8}
T_1*LOPNRE*R*LB2*SSA*DIESEL3	2.2×10^{-8}
T_1*LOPNRE*R*DIESEL1*SSB*SSC	2.2×10^{-8}
T_1*LOPNRE*R*LB2*DIESEL1*H	2.2×10^{-8}
T_1*LOPNRE*R*DIESEL2*SSC*L	2.2×10^{-8}
T_1*LOPNRE*R*V1*DIESEL2*DIESEL3	2.1×10^{-8}
T_1*LOPNRE*R*DIESEL1*V2*DIESEL3	2.1×10^{-8}
T_1*LOPNRE*BATA*DIESEL2*SSC	2.0×10^{-8}
T_1*LOPNRE*BATA*LB2*DIESEL3	2.0×10^{-8}
T_1*LOPNRE*R*SSA*DIESEL3*LB1	2.0×10^{-8}
T_1*LOPNRE*R*DIESEL1*H*LB1	2.0×10^{-8}
T_1*LOPNRE*R*SSA*H*LC	2.0×10^{-8}
T_1*LOPNRE*R*SSA*SSB*H	1.9×10^{-8}
T_1*LOPNRE*R*SSB*L*H	1.9×10^{-8}
T_1*LOPNRE*BATA*DIESEL3*LB1	1.9×10^{-8}
T_1*LOPNRE*BATA*H*LC	1.8×10^{-8}
T_1*LOPNRE*BATA*SSB*H	1.8×10^{-8}
T_1*LOPNRE*R*LB2*DIESEL1*SSC	1.4×10^{-8}
T_1*LOPNRE*R*DIESEL1*SSC*LB1	1.3×10^{-8}
T_1*LOPNRE*R*SSA*SSC*LC	1.3×10^{-8}
T_1*LOPNRE*R*V1*DIESEL3*LC	1.3×10^{-8}
T_1*LOPNRE*R*SSA*SSB*SSC	1.3×10^{-8}
T_1*LOPNRE*R*LB2*SSA*H	1.3×10^{-8}
T_1*LOPNRE*R*SSB*SSC*L	1.3×10^{-8}
T_1*LOPNRE*R*SSA*V2*DIESEL3	1.2×10^{-8}
T_1*LOPNRE*R*V1*SSB*DIESEL3	1.2×10^{-8}
T_1*LOPNRE*R*V1*DIESEL2*H	1.2×10^{-8}
T_1*LOPNRE*R*DIESEL1*V2*H	1.2×10^{-8}
T_1*LOPNRE*R*V2*DIESEL3*L	1.2×10^{-8}
T_1*LOPNRE*BATA*SSC*LC	1.2×10^{-8}
T_1*LOPNRE*BATA*LB2*H	1.2×10^{-8}
T_1*LOPNRE*BATA*SSB*SSC	1.2×10^{-8}
T_1*LOPNRE*R*SSA*H*LB1	1.2×10^{-8}
T_1*LOPNRE*BATA*V2*DIESEL3	1.2×10^{-8}
T_1*LOPNRE*BATA*H*LB1	1.1×10^{-8}
T_1*LOPNRE*OP*R*V3	1.0×10^{-9}
T_1*LOPNRE*R*DIESEL1*DIESEL2*V3	8.7×10^{-9}
T_1*LOPNRE*R*LB2*SSA*SSC	8.4×10^{-9}
T_1*LOPNRE*R*V1*DIESEL2*SSC	8.2×10^{-9}
T_1*LOPNRE*R*LB2*V1*DIESEL3	8.2×10^{-9}
T_1*LOPNRE*R*DIESEL1*V2*SSC	8.2×10^{-9}
T_1*LOPNRE*BATA*LB2*SSC	7.8×10^{-9}
T_1*LOPNRE*R*SSA*SSC*LB1	7.8×10^{-9}
T_1*LOPNRE*R*V1*DIESEL3*LB1	7.6×10^{-9}
T_1*LOPNRE*R*V1*H*LC	7.5×10^{-9}

Cut Set	Frequency
T_1*LOPNRE*BATA*SSC*LB1	7.3×10^{-9}
T_1*LOPNRE*R*SSA*V2*H	7.2×10^{-9}
T_1*LOPNRE*R*V1*SSB*H	7.2×10^{-9}
T_1*LOPNRE*R*V2*L*H	7.2×10^{-9}
T_1*LOPNRE*BATA*V2*H	6.7×10^{-9}
T_1*LOPNRE*R*DIESEL1*V3*LC	5.3×10^{-9}
T_1*LOPNRE*R*SSA*DIESEL2*V3	5.1×10^{-9}
T_1*LOPNRE*R*SSB*DIESEL1*V3	5.1×10^{-9}
T_1*LOPNRE*R*DIESEL2*V3*L	5.1×10^{-9}
T_1*LOPNRE*R*V1*SSC*LC	5.0×10^{-9}
T_1*LOPNRE*R*SSA*V2*SSC	4.8×10^{-9}
T_1*LOPNRE*R*V1*SSB*SSC	4.8×10^{-9}
T_1*LOPNRE*R*LB2*V1*H	4.8×10^{-9}
T_1*LOPNRE*R*V2*SSC*L	4.8×10^{-9}
T_1*LOPNRE*BATA*DIESEL2*V3	4.8×10^{-9}
T_1*LOPNRE*R*V1*V2*DIESEL3	4.7×10^{-9}
T_1*LOPNRE*BATA*V2*SSC	4.5×10^{-9}
T_1*LOPNRE*R*V1*H*LB1	4.5×10^{-9}
T_1*LOPNRE*OP*R*SCC	3.7×10^{-9}
T_1*LOPNRE*OP*R*HACT	3.7×10^{-9}
T_1*LOPNRE*R*LB2*DIESEL1*V3	3.4×10^{-9}

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