216
1-24-77

MASTER

Dr 636
UCRL-52156

ADVISABILITY OF SEISMIC SCRAM

G. E. Cummings, J. E. Wells,
H. E. Lambert, and G. St. Leger-Barter

June 30, 1976

Prepared for U.S. Energy Research & Development
Administration under contract No. W-7405-Eng-48

LAWRENCE
LIVERMORE
LABORATORY
University of California/Livermore

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Energy Research & Development Administration, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately-owned rights.

NOTICE

Reference to a company or product name does not imply approval or recommendation of the product by the University of California or the U.S. Energy Research & Development Administration to the exclusion of others that may be suitable.

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161
Price: Printed Copy \$: Microfiche \$3.00

Page Range	Domestic Price	Page Range	Domestic Price
001-025	\$ 3.50	326-350	10.00
026-050	4.00	351-375	10.50
051-075	4.50	376-400	10.75
076-100	5.00	401-425	11.00
101-125	5.50	426-450	11.75
126-150	6.00	451-475	12.00
151-175	6.75	476-500	12.50
176-200	7.50	501-525	12.75
201-225	7.75	526-550	13.00
226-250	8.00	551-575	13.50
251-275	9.00	576-600	13.75
276-300	9.25	601-up	*
301-325	9.75		

*Add \$2.50 for each additional 100 page increment from 601 to 1,000 pages;
add \$4.50 for each additional 100 page increment over 1,000 pages.

LAWRENCE LIVERMORE LABORATORY
University of California Livermore, California 94550

UCRL-52156

ADVISABILITY OF SEISMIC SCRAM

G. E. Cummings, J. E. Wells,
H. E. Lambert, and G. St. Leger-Barter

MS. date: June 30, 1976

NOTICE
This report was prepared as an account of work
sponsored by the United States Government
and therefore is subject to the United States Copyright
Law and the Department of Energy, and one of
those employees, and one of those contractors,
subcontractors, or their employees, shall not
assert any patent or copyright in any invention
or discovery made in the course of, or as a
result of, the performance of the work under
this contract, or in any process, product, or
information developed or derived therefrom, or
in any process, product, or information developed
or derived therefrom, or in any process, product,
or information developed or derived therefrom.

Foreword

This report was originally prepared for the Office of Standards Development, Nuclear Regulatory Commission.

Contents

Abstract	1
1. Introduction	1
2. Background	2
Existing Seismic Instrumentation and Regulations	2
Proposed Seismic Standards	3
3. Areas of Study	5
Likelihood of Existing Plant Instrumentation Tripping the Reactor During an Earthquake	5
Spurious Reactor Trips	10
Existing Seismic Trip Systems and Their History	10
Variables Affecting the Frequency of Seismic Trips	10
Costs of Seismic Trip Systems	14
Increased Accident Probability Due to Spurious Trips	15
Consequences of Scramming During an Earthquake	16
Advantages of a Controlled Shutdown During an Earthquake	17
Grid Stability	17
Human Judgement	18
Aftershocks	18
Desirability of Allowing the Reactor to Continue Generating Power During an Earthquake	18
Emergency Services	18
Power for Reactor Emergency and Decay Heat Loads	21
4. Risk Assessment	22
Fault Tree Description	22
Fault Tree Quantification	25
Conclusions Based on Computer Output	29
5. Discussion	31
Advantages of a Seismic Trip System	31
Disadvantages of a Seismic Trip System	32
6. Summary	33
7. References	35
Appendix A: Time Interval Analysis	37
References	46
Appendix B: Output and Conclusions from IMPORTANCE Computer Code	47

ADVISABILITY OF SEISMIC SCRAM

Abstract

This study for the Nuclear Regulatory Commission assesses the value of seismic trip (scram) systems on commercial nuclear power plants. Five topics were specifically addressed: 1) the likelihood that existing instrumentation would trip the plant; 2) the consequences of spurious trips; 3) the consequences of tripping during an earthquake; 4) the advantages of a controlled

shutdown; and 5) the desirability of continuing operation. A comparative risk assessment was made based on our findings by evaluation of fault trees constructed for plants with and without seismic trip systems. It was found that seismic trip systems would have a small and undetermined effect on risk due to nuclear reactor accidents and may indeed increase the risk to society from an earthquake.

1. Introduction

This is the final report of a study by the Lawrence Livermore Laboratory on the advisability of requiring a seismic trip (scram) on commercial nuclear power plants. Previous studies have addressed the feasibility of providing a seismic trip system, and the present report is limited to a consideration of the advisability of such systems.¹

In assessing the value of a seismic trip system, we studied several specific areas:

- The likelihood that existing plant instrumentation will cause a trip during an earthquake.
- The consequences of spurious reactor trips caused by a seismic trip system.

- The consequences of tripping during an earthquake.
- The advantages of a controlled shutdown during an earthquake.
- The desirability of allowing nuclear power plants to continue to generate power during an earthquake.

Through a generalized fault tree analysis we assessed the risk in each of these specific areas. We compared fault trees for nuclear reactors with and without seismic trip systems, using estimated data. The quantification of the fault trees made possible a comparison between the risks of plants with and without seismic trip systems, and between earthquake caused risk and other accident risks.

From the risk assessment and information developed in the five areas of study, several advantages and disadvantages of seismic trip systems are listed. Since the degree to which such systems would protect a nuclear plant are somewhat dependent on the amount of time between trip and strong motion arrival afforded by

a trip system, a special appendix was added that estimates from historical data the advance shutdown time inherently available. Computer output in another appendix ranks the importance of the basic events on the fault trees to provide an easy means of establishing critical contributors to risk.

2. Background

EXISTING SEISMIC INSTRUMENTATION AND REGULATIONS

Although there are presently no requirements for a seismic trip on commercial power reactors, certain seismic instrumentation has to be provided by the operating organization for sensing and recording purposes. It may help put the seismic trip question in the proper perspective to review the existing requirements and also some currently proposed.

Seismic instrumentation at existing power plants consists mainly of triaxial time-history accelerographs, peak accelerographs, seismic switches, and response-spectrum recorders. The purpose of this instrumentation is to record the intensity of an earthquake, should one occur, and monitor the plant's response. Information from the instrumentation can then be used to help establish whether the

operating basis earthquake (OBE)* has been exceeded and comply with the requirement that the plant be shut down if it has.² Also, the instrumentation helps in validating mathematical models and prediction of seismic response.

The response-spectrum recorder, with limit switches attached to its vibrating reeds, can be used to indicate an earthquake to the control room. Other instruments either

*The OBE is the most powerful earthquake that, considering the regional and local geology and seismology and specific characteristics of local sub-surface material, could reasonably be expected to occur in the vicinity of the plant site during the operating life of the plant. Those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to withstand the vibrations of this earthquake.

actively or passively record at their location. At a minimum, instrumentation is positioned to read the free field response and is also located on the containment foundation, containment walls, and sensitive equipment inside the containment. The amount of instrumentation depends on the maximum ground acceleration predicted for the safe shutdown earthquake.[†]

There are several systems used for detecting seismic motion. Included in all are switches to detect motion greater than some threshold. The systems could be used to initiate or inhibit some action when motion (displacement, velocity, or acceleration, but normally velocity and acceleration) exceeds some threshold. The most common type is the oil damped pendulum and variations of it, such as a metal ball making contact between adjacent conductors. This type of switching is used to initiate, for example, the recording of ground acceleration with a strong motion accelerometer. It can also be

used to inhibit actions such as elevator operation in tall buildings, where required by building codes. Newer switches are basically an accelerometer with sensitivity in the bandwidth chosen as more representative of the earthquake spectrum than of civilization's background noise. These switches can also be selected and used for any axis -- vertical or horizontal. Being an electromechanical device, it is a straightforward matter to adjust mechanical constants and circuit parameters to select spectrum sensitivity. In addition to sensitivity in a select bandwidth, coincident logic from separate switches could be used to minimize spurious seismic trips where spurious trips would be costly. This coincident logic might be necessary because an individual sensor could be jarred by some broadband source (e.g., a dropped wrench) with enough energy in the frequency of interest to exceed the set threshold. This problem can also be circumvented through the use of significantly higher thresholds, but this may defeat the original intent of the switch.

PROPOSED SEISMIC STANDARDS

The regulations and standards that dictate the present seismic instrumentation requirements are Regulatory

[†] The safe shutdown earthquake is based on an evaluation of the maximum earthquake potential, considering the regional and local geology and seismology and specific characteristics of local subsurface material). It is that earthquake that produces the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional.

Guide 1.12³ and ANSI standard N18.5.⁴ A proposed standard, ANS 2.10/N643,⁵ is being prepared to specify the proper response the plant operator should make based on the instrument readings following a seismic event. Basically this standard relying on controlled, manual shutdown proposes a two step approach. In step 1 it is determined whether the OBE has been exceeded. This will be indicated if a minimum number of points of the recorded response spectrum have exceeded the OBE design response spectrum or if there is an equipment malfunction. If either is the case, the reactor is shut down in a controlled manner (may take five to ten hours). Restarting the plant may then require the concurrent approval of the Nuclear Regulatory Commission and the plant owner. There is no clear criterion on which to assess the advisability of restarting the plant, so once a plant is shut down it may be shut down for a substantial time.

If in step 1 it is determined that the OBE was not exceeded, but the plant operator has an indication that an earthquake did occur then he continues operation and proceeds to step 2. In step 2 the recorded data is analyzed to see if the design floor-response spectrum has been exceeded. If, after analysis, it

appears that the floor-response spectrum for the OBE has been exceeded, the plant is shut down. This analysis may take from one day to one week.

Note that the intent of this proposed standard is to enable the operator to determine whether the plant should be shut down and to shut down if it appears the OBE has been exceeded. In judging this, the operator is only to take into account the condition of the plant and is not to assess the need for the electric generating capacity of the plant, i.e., to consider mitigating the consequences of the earthquake in the surrounding community. Also, once plant shutdown commences, there is no established procedure for restart. Therefore, as it now stands, whether ANS 2.10 is adopted or not, if the plant is shut down because of an earthquake it may stay out of service during the earthquake emergency period including post-earthquake recovery. If a seismic trip of low trip level were used, this would assuredly be the case unless regulations are further clarified. Without a seismic trip, the operator would have some discretion, although legally he should shut down as soon as he is convinced the OBE has been exceeded.

3. Areas of Study

LIKELIHOOD OF EXISTING PLANT INSTRUMENTATION TRIPPING THE REACTOR DURING AN EARTHQUAKE

The sensitivity of the normal plant instrumentation to the seismic motion portion of the mechanical vibration and shock spectrum is determinable but of questionable value. Unless detectors are mounted such that they couple only to the free field earth motion, any signal response they may give as a direct function of any seismic input would be masked. This masking would be due to motion attenuation or amplification because of plant structures, mounting techniques, etc.

To fully understand the response of an instrumentation channel to some seismic input, that is whether it will initiate some safety action, one must know its vibration characteristics during normal operation and the response spectrum coupling to the detector. All class I electrical equipment is supposed to be qualified to ANSI N41.7 Seismic Qualification of Class I Electric Equipment.⁶ This basically requires that the equipment perform its safety function during and after the disturbance caused by a design-basis earthquake and does not really address possible vibration induced (spurious) trips. By vibration induced trips, we mean a

trip that occurs when the sensed variable does not change; for example, a trip caused by mechanical shock of an ion chamber. An example of a genuine response would be fluid level sloshing in the pressurizer of a pressurized water reactor (PWR), which results in a water level trip. If the level detectors are qualified to ANSI N41.7, they will sense high or low fluid level and operate accordingly, as the variable undergoes a genuine change. We will be primarily concerned with vibration induced, spurious instrument response in this section.

Tabulating the "at-power-operation" trip channels for a typical PWR and going through them to determine the likelihood that a trip could be induced seismically indicates many uncertainties. Typical trip channels are:

- Neutron flux
- Coolant temperature (both absolute temperature and temperature differences)
- Coolant system pressure
- Pressurizer water level
- Coolant flow
- Coolant pump status
- Steam-generator feedwater flow
- Steam-generator water level
- Turbine-generator operational status

Operator

Both reactor-protection system logic trains bypassed.

Skipping over the individual detectors for each of these for a moment and looking at the operation of the train of logic (assumed here to be relay logic) connected to the detectors, allows some generalization.

A simplified sketch of a portion of the logic train is shown in Fig. 1.⁷ This figure indicates that there are several relays in series for each channel, some requiring coincident action. Appropriate combinatorial probabilities are required, in conjunction with the actual relay qualifications and logic implementation scheme, to make any quantitative statement on potential seismic response.

However, the relays and circuit breakers probably have the same sensitivity to shake and vibration, and, since they are all very similar, all this relaying can be considered together. There is a basic MIL specification for relays⁸ and, should the relays and circuit breakers be qualified to any seismic shock approaching this specification (around 10 g), they are not likely to give spurious signals.

The detectors likely to be used in the trip channels given above are tabulated in Table 1, along with some

estimate of the detectors' individual motion sensitivities. While not absolutely complete, the table is general enough to cover the majority of the detectors in the trip channels.

As far as spurious trips from detector response to a genuine variable change, the most obviously susceptible detector is the turbine vibration monitor. This monitor's response spectrum overlaps (on its sensitive axis) the spectrum of most seismic motion. Some fossil-power units have actually experienced a turbine trip during an earthquake; Pacific Gas and Electric Company's Kern County Unit tripped during the 1952 Bakersfield earthquake because of a turbine thrust motion sensor. If the seismic motion is large enough the vibration monitor will respond, and if this monitor is connected to an automatic turbine trip that would close the turbine stop valve, the reactor would trip from fast closure of the turbine control or stop valve. It should be noted that each nuclear plant operator has the option of connecting turbine vibration monitors to the turbine trip system, and that several have chosen not to, i.e., Pacific Gas and Electric (at Diablo Canyon) and Northern States Power (at Monticello).

To summarize, a generalization on the suitability of current plant

Fig. 1. Simplified logic train for steam generator water level and feedwater flow portion of plant trip system.

Table 1. Motion sensitivity of typical trip channel detectors.

Variable	Detector	Sensitivity to motion
Neutron flux	Ionization Chamber	Some sensitivity to motion gives spurious signals. If qualified to MIL-S-901 and MIL-Std-167 for structural integrity they would probably be much less sensitive. Electronic comparison for trip reference has low sensitivity to seismic input.
Coolant temperature	Thermocouples	Low sensitivity to motion, some sensitivity to flexing near junction. Low probability of spurious output.
	Resistance thermometer	Low sensitivity.
Coolant system pressure	Pressure transducers using: Linear variable differential transformer (LVDT) Force Balance Strain gauge	Mostly low sensitivity but a function of the mass of the sensing element such as a diaphragm or bourdon tube.
	Pressure switches:	
	Snap action	Low sensitivity.
	Mercury switch	Sensitivity to tilt from less than 5° to as much as 20°, useful as inertial switch. Very sensitive to acceleration in some axes.
Pressurizer water level	Conductivity gauge	Low sensitivity of sensor itself.
	Float gauge	Some sensitivity as a function of float mass.
Coolant flow	Turbine mass flow-meter	Low sensitivity.
	Orifice differential pressure	Low sensitivity similar to pressure transducer.
Coolant pump status	Closures and limit sensors	Low sensitivity of relays and switches unless very close to actuation.
Steam-generator water level	Similar to pressurizer water level	
Turbine-generator operational status		This is part of balance of plant instrumentation and as such it may or may not have to meet the Class I equipment requirement. However, it does enter the scram train.

Table 1. (Continued)

Variable	Detector	Sensitivity to motion
Manual turbine trip	Closures and limit sensors	Low sensitivity.
Condenser vacuum	Pressure switch: Snap action Mercury switch	Low sensitivity. Sensitive to tilt and shake.
Overspeed	Tachometer Frequency counter	Low sensitivity. Low sensitivity.
Thrust bearing failure	Vibration pickup Clearance capacity gauge	Some sensitivity. Some sensitivity.
Operator	Closure and manual switching	Switch sensitivity low, presentations of variables to the operators may give false cues and at their discretion the operators may trip the plant.
Both logic trains bypassed	Circuit breaker and/or relay closures	Low sensitivity.

instrumentation for the secondary purpose of seismic tripping cannot be made. The question requires detailed study of the specific plant and instrument installations, and the results would be applicable to that plant only. The instrument system will probably trip the plant during an earthquake more severe than the OBE, particularly if the plant is designed for a large OBE. However, it appears to be only marginally probable that the lower amplitude precursor motion (see Variables Affecting the Frequency, under Spurious Reactor Trips) would trip an

instrument system designed to ANSI N41.7. Thus, such a trip would occur during strong motion and give no advance warning.

Should tripping the plant be desired for seismic disturbances, two alternatives present themselves: 1) use present seismic or earth motion sensors to properly alert the operators and leave the actual tripping to their discretion, or 2) use a well designed, redundant and coincident triaxial seismic trip system to trip at a chosen threshold tailored to an earthquake "signature."

SPURIOUS REACTOR TRIPS

Existing Seismic Trip Systems and Their History

No commercial power reactors currently employ a seismic trip system. However, a number of research, test, and production reactors are now using or have used such systems, and these are listed in Table 2 with brief notes about their performances. In general, the instrumentation used to sense the earthquake is a seismic switch sensitive to horizontal accelerations and set to trip the reactor at a given peak acceleration. In certain cases, coincident logic is used in the trip circuitry to help minimize spurious trips.

Most of the reactors equipped with seismic trip systems are in seismically active areas, e.g. California. Most of the trip levels are set so the reactor will shut down if an earthquake of modified-Mercalli intensity V (~ 0.01 g) is experienced at the reactor site. An earthquake of this intensity would be felt by most people but not cause significant damage. Some of the California reactors, where earthquakes of this intensity are fairly common, have their trip levels set at modified-Mercalli intensity VIII (~ 0.1 g), an intensity that should

heavily damage conventional structures, but one that a well designed reactor should survive. As can be seen from Table 2, many installations have experienced unwarranted trips from these systems, with the frequency depending on trip level, circuit logic, and the seismicity of the site.

Variables Affecting the Frequency of Seismic Trips

An unnecessary shutdown caused by a seismic trip system is not nearly the problem for the research reactors listed in Table 2 as it would be for a commercial power reactor. Research reactors are designed for up and down operation, while power reactors are designed for steady operation. Thus, a seismic trip system for a power reactor would have to be much more reliable than those for research reactors.

Unnecessary trips can result from two causes: poor instrument performance (noise or malfunction), and oversensitivity of the trip system to low-intensity earthquakes. The first cause can be minimized by use of reliable instruments, redundant channel logic, or discrimination in the sensor to allow response only to earthquake spectra. The EBR-II experience of Experimental Breeder

Table 2. Seismic trip systems employed on reactors.

Facility	Location/ zone*	Trip level, gravities	Performance
1. Materials Test Reactor Engineering Test Reactor	Idaho zone 3	0.01 - 0.02	No spurious trips ⁹
2. Hanford Production Reactors	Washington zone 2	0.01 - 0.02	No excessive spurious trips (used 2/2 logic) ¹⁰
3. Savannah River Production Reactors	South Carolina zone 3	0.02 - 0.03	No problems ¹⁰
4. Experimental Breeder Reactor II	Idaho zone 3	0.005	Two spurious trips in five years ^{9,11} (now use 2/3 logic)
5. Sodium Reactor Experiment and various other Atomic International experimental reactors	California zone 3	0.01 - 0.02	Several spurious trips ¹⁰
6. UCLA Research Reactor	California zone 3	---	---
7. AGN-201 Research Reactor	California zone 3	0.01	Two trips in five years ¹⁰ caused by minor earthquakes
3. Vallecitos Boiling Water Reactor General Electric Test Reactor	California zone 3	0.005 - 0.01	Several trips caused by earthquakes and a series of spurious trips caused by building settling ¹⁰
9. UTR-1 Research Reactor	California zone 3	0.01 - 0.02	No trips over several years ¹⁰
10. Water Boiler Neutron Source (Lawrence Livermore Laboratory)	California zone 3	0.07 - 0.15	No trips over five years
11. Light Water Breeder Reactor (Shippingport)	Pennsylvania zone 1	0.005 - 0.033 (2/3 logic)	Not yet installed ¹²
12. Berkeley Research Reactor	California zone 3	0.02 - 0.03	No spurious trips in ten years

*Uniform building code seismic zones.

Reactor II (see Table 2) suggests the triaxial seismic triggers and 2/3 channel logic can reduce the frequency of spurious trips to less than one per five years even with a very low trip level setting.¹¹ This is to be compared to an expected spurious trip frequency of ten per year at a modern power reactor. The use of a response spectrum recorder should improve on the EBR-II experience even more.

Discriminating out low intensity earthquakes (the second cause of unnecessary trips) may be more of a problem. The first thought would be to set the trip level high, so that only damaging earthquakes trip the plant. Since the plant is designed to accept an OBE and remain operable, this may be the appropriate level. However, if this were done the plant would not trip on precursor motion, and therefore would shut down during strong motion. On the other hand, setting the trip level very low would maximize the chance of shutting down before the strong motion arrived but, because of aftershocks, might result in long periods of downtime after a major quake or during an earthquake swarm. The Hebgen Lake earthquake (magnitude 7.2) resulted in 22 aftershocks during the six months following the main earthquake, and 4 of the aftershocks occurred in the same week. Danville, California,

recorded several hundred measurable tremors during an earthquake swarm in 1970.

Since power reactors take many hours to restart after they are tripped, the high frequency of aftershocks immediately following a major earthquake might result in long-term unavailability of all nuclear units in a large area if the trip level was set low. This would be the critical period when electric power to the surrounding community is most needed and when the less stringently designed fossil units would probably be out of service also.

Certain characteristics of the waves from an earthquake indicate its source distance from an instrument. These waves are predominantly the P and S waves, and they are useful in locating the source. Their differing velocities may give a time margin, in which some action beneficial to plant safety might be taken. The later arriving S wave is usually associated with the strong motion, and the time interval between the P and S wave arrivals could be enough that some motion sensitive equipment could be put into a mode to withstand strong ground motion.

With an earthquake source large enough and close enough to do damage to a power plant (i.e., greater than the safe shutdown earthquake), the

time between the P and S wave arrival would be quite short. With propagation velocities on the order of one to three kilometers per second the time interval is probably less than one to two seconds, but the buildup to peak acceleration (based on previous strong motion records) can be a few seconds. As shown in Appendix A, the mean time to strong motion after P-wave arrival is three to five seconds from P-wave tremors of 0.1 g or greater intensity. This warning time is an average value and would hold only for sites where S-wave motion is significantly larger than and separable from P-wave motion. There is no general scale factor relating the P-wave motion to the S-wave motion. If a P-to-S relationship has been determined for one event, then other earthquakes from the same source area will have approximately the same P-to-S amplitude relationship. For earthquake swarms originating in the same area, the P-to-S amplitude ratio may be said to be approximately constant. The same may be said for aftershocks. These magnitudes of motion are from strong motion recordings, regardless of the spectrum. The frequency content of the earlier P wave might identify a strong earthquake, so that a frequency sensitive seismic trip, even if set at a low level, would

only activate for strong earthquakes. Insufficient information is available to determine the feasibility of this approach.

With what is currently known, a low level trip (set at less than 0.05 g) would have to be used to make use of the time interval between the P and S waves (see Appendix A). A "sufficient time" interval would allow active components of the power plant to complete their shutdown action before the strong motion arrives. Clearly, in three to five seconds the plant will not have completely shut down. However, such systems as the control rods, main steam-isolation valves, and turbine stop valves would complete or partially complete their shutdown action. Since the emergency core cooling system (ECCS) is not activated until an accident occurs, this warning time would not affect its operability.

One of the main concerns about an earthquake is that it might rupture the reactor coolant system, leading to a loss of coolant accident. If this were to occur, a trip of the reactor three to five seconds before the accident would allow equalization and some removal of the stored energy in the fuel rods, resulting in a decrease in the subsequent peak temperature of the cladding. Assuming the ECCS works, this may

eliminate any clad perforation due to the accident and minimize the off site release.¹³ However, this is a small matter when compared to a core melt, and the core would melt if we had ECCS failure, regardless of the warning time. Note that according to Newmark, the probability of ECCS failure resulting from ground motion exceeding that of the safe shutdown earthquake by a factor of 2.5 is only 0.02.¹⁴

Costs of Seismic Trip Systems

In considering spurious trips, there are three main costs to be looked at. The first is the initial capital cost of a seismic trip system. At present, reactor sites are required to have a prescribed amount of seismic instrumentation. The increased cost to the utilities of a seismic trip system will not be great if the system incorporates some of the existing instrumentation. However, requiring this instrumentation to meet ANSI N41.7 would incur added costs. The second cost is that of maintenance associated with the seismic trip system. This cost should not significantly increase the normal maintenance cost associated with reactor instrumentation.

The third cost is that due to spurious trips of the system. Unlike the other two costs, this one could

have a significant monetary effect on the utilities. While the costs associated with spurious trips differ with location and size of reactor, these costs could have an impact on the utility if the spurious trip rate was substantial. When a large nuclear power plant such as one of Florida Power and Light's 700 MWe Turkey Point units is unable to operate, the extra cost of using fossil fuel to replace the nuclear generated power is upwards of a half-million dollars per day.¹⁵

Another example is that of Browns Ferry nuclear power plant. The Browns Ferry fire in March, 1975, shut down both reactors at the site, when they were producing 1100 MW of electrical power. Prior to the fire, the Browns Ferry reactors were saving the Tennessee Valley Authority about \$240,000 per day in fuel costs.¹⁶ Other power companies would have comparable costs.

Figure 2 illustrates the effect an increase in trip frequency will have on the plant economics. To evaluate the dollar amounts involved, assume it costs \$500,000 per day to replace the nuclear-generated power with fossil fuels, and that it takes six hours to bring a reactor back on line after a reactor trip. The cost per reactor trip would then be about \$125,000, and an increase in

Fig. 2. Increased costs due to increase in trip frequency, assuming the following: outage time due to trip equals 6 h, and cost is \$500,000/day-outage or \$125,000/trip.

spurious trip frequency of one per year would add five million dollars to plant operating costs over a 40-year life. Although this is a small amount in comparison to total generating costs, it could become significant in areas where reactors are subject to earthquake swarms, e.g., California. Also, we have not included other economic consequences to the utility of having large power producing units with increased probability of sudden shutdown, e.g., increased excess capacity requirement.

The location of a reactor becomes critical when considering the monetary effects of seismic trips. If a utility decides to build four reactors on one site, then that utility would run the risk of having all four reactors shut down at one time if a trip were caused by a small

earthquake. High trip level settings could alleviate this problem but, as mentioned in Sec. 2, this could eliminate the chance of having any advanced warning time between trip and arrival of the strong motion.

Increased Accident Probability Due to Spurious Trips

Spurious trips during normal operation may produce a greater accident risk than the earthquake itself. For example, WASH 1400 shows that the greatest accident risk for a boiling water reactor (BWR) is a transient event causing reactor shutdown followed by loss of the heat removal systems.¹⁷ WASH 1400 also states that the probability that BWR heat removal systems will fail following a reactor shutdown is 1.6×10^{-6} , assuming offsite power to be available. This probability is much lower for a PWR if a steam driven, auxiliary feedwater pump is available. Regardless, if the spurious scram rate due to a seismic trip system is 0.2/yr, i.e., once every five years (from Ref. 11), then the probability that a spurious trip will cause a fuel meltdown in a BWR is $(0.2)(1.6 \times 10^{-6}) = 3.2 \times 10^{-7}$ per reactor year. From WASH 1400, the probability that an earthquake will cause a fuel meltdown in a reactor without a seismic trip system is of the order 10^{-7} . Thus,

if all these probability values are correct, a small increase in spurious trip rate of a BWR's seismic trip system would pose a greater risk of core melt than the earthquake itself.

Also, even an incremental increase in trip frequency will increase the total number of thermal stress cycles a plant will see during a period of time. This increased number of cycles could possibly have an effect on the plant life. Note that the currently expected trip frequency of a nuclear power reactor with a seismic trip system is ten per year.¹⁷

CONSEQUENCES OF SCRAMMING DURING AN EARTHQUAKE

Without a seismic trip system, the reactor may or may not be shut down during an earthquake, depending on the performance of existing instrumentation and the plant operators. With a seismic trip system, the reactor will be shut down either just prior to or during the strong motion. If the reactor were to shut down during the strong motion, additional stress would be put on some components. The thermal transient may impose thermal stresses. Also, active components may bind up because of their movement during earthquake motion. Since critical components are designed to withstand the combined stresses of accident and

earthquake loads, any increased stress in shutdown due to the thermal transient should not cause system failure.

Possible effects on active components required to change state during an earthquake may be more significant. Two classes of active components need to be examined. One class is the seismic-category-I components, such as BWR scram valves or relays in the reactor protective system that change state during a scram. Since such components are specifically designed to perform safely during a safe shutdown earthquake, they can be expected to perform properly. The other class of active components is the non-category-I active components that must change state during a scram. Typical of this class are the feedwater- and main steam-isolation valves and valve controllers. Since these items are not classed as category I, they might malfunction during a strong earthquake and lead to an unpredictable thermal transient in shutdown. A detailed study of these systems would be required to determine this.

Shutting down prior to strong motion would allow active components to change state before earthquake motion and help to assure a normal thermal transient. However, since

the expected advanced warning times are on the order of a few seconds (see Appendix A), not all systems will have completed their action by the time the strong motion arrives, and tripping during an earthquake may affect their action. Some of the safety systems, e.g., control rods and main steam-isolation valves, might complete their action if the trip level is set low enough, and to this extent chances for a safe shutdown will be enhanced.

A seismic trip system would have little effect on the operability of the ECCS and other engineered safety systems called into play during an accident. Since these are initiated in response to an accident it would be expected that they would have to operate during strong motion if that strong motion caused the accident. Since they are specifically designed to withstand the SSE they should not be of as much concern as the non-category-1 components.

ADVANTAGES OF A CONTROLLED SHUTDOWN DURING AN EARTHQUAKE

Controlled shutdown during or following an earthquake offers some advantages over an automatic seismic scram system. These advantages pertain to grid stability, human judgements, and aftershocks. However,

it would not be possible to shut down prior to strong motion if this strategy was used.

Grid Stability

It has been estimated that the probability of losing off-site power due to a power-grid transient caused by a nuclear plant trip is 10^{-3} per trip.¹⁸ This estimate is made for a normally operating grid and single nuclear-unit trip. Since 10^{-3} is an average, the probability could be higher or lower, depending on the site of the reactor, amount of installed capacity and number of grid interconnections with other large power companies. The probability of power failure would be larger for a grid that has experienced a major earthquake and simultaneous loss of several nuclear units. For instance, four to six nuclear units are planned for the Diablo Canyon site along the California coast. If these units were all tripped simultaneously by seismic trip systems during a major earthquake, it would have a severe impact on Pacific Gas & Electric's grid in central California. This could lead to the unavailability of power to the population when it's needed to cope with the problems that arise after an earthquake.

Human Judgement

This subsection addresses the issue of allowing the operator to effect a controlled shutdown if in his opinion a damaging or larger-than-operating-basis earthquake has occurred. This issue is also exemplified by the ANS 2.10 standards subcommittee guidelines.⁵ Appropriate seismic instrumentation would be available to aid the operator in making a decision. During the shutdown, the operator could assess any possible damage to the plant and check the condition of the off-site power grid. The controlled shutdown also provides time to check the diesels providing emergency power to the site and bring them into operation in case off-site power is lost. Based on these items plus any others unique to a specific facility, the operator can decide whether to resume operation or continue the shutdown.

If this strategy is employed, procedures should be written to give the operator a specific guideline for making his decision. This is needed because of the concern and confusion that can be anticipated during and after an earthquake. These procedures should also prepare the operator for possible malfunctions of automatic protection devices.

Aftershocks

If a reactor containing a seismic trip system is shut down, inspected, and brought back on line again after an earthquake, it may be shut down several more times because of aftershocks. Allowing the operator to perform a controlled shutdown minimizes the possibility of repeated trips due to these aftershocks since he could ignore them if they are of sufficiently low level. There would then be less chance that electric power would be disrupted for a long period following an earthquake.

DESIRABILITY OF ALLOWING THE REACTOR
TO CONTINUE GENERATING POWER DURING
AN EARTHQUAKE

Emergency Services

If the utility grid and the intertie from the nuclear power plant to the grid (neither of which are seismic-category-I structures) is intact, then the total risk to the public might be less if the plant continued to supply electric power for emergency services than if it shut down. The degree of risk, of course, depends on how much of the power of the utility grid comes from nuclear units and how many of these are shut down by the earthquake.

In discussions with county offices of emergency services and the utility

Coordinator for the State of California Office of Emergency Services, the following information was obtained concerning the effect on California emergency services if utility power is lost:

- Each county office has at least two control centers. These control centers are used to control emergency services during any emergency situation. Each control center has its own source of power and can operate for several days without outside assistance.

- Hospitals have an emergency power source. This power source is not designed to completely provide the power needs of the hospital but is designed to provide enough power to handle all emergency situations.

- All local, state and federal law enforcement agencies have emergency power available when utility power is not available. These agencies also have a large number of two-way, self-powered radios available to them in emergency situations.

- Most public buildings and large department stores have emergency lighting for stairways and other critical areas.

- Telephone companies have emergency backup power in the event that utility power is lost.

All the items mentioned so far indicate utility power may not be of vital need during an earthquake emergency. However, the following points will temper that conclusion, under the assumption that the earthquake affects a large area:

- While hospitals do in fact have their own source of emergency power, most of these sources appear not to have all of their components and interties seismically qualified. Therefore, after a severe earthquake this power may not be available.

- Vehicle travel becomes dangerous and slow because traffic lights become inoperative without utility power. This problem is aggravated during the hours of darkness.

- Electrically powered street lighting will not be available because no emergency source will generally be available.

- Department store lighting would not be available except for the emergency stairway lighting mentioned earlier.

- While most homes are gas heated, the thermostat controls are electrically powered and furnaces would become inoperative without utility power.

- Many alarm systems require utility power and will not operate without it.

Fig. 3-8. Example of the type of record obtained from a cyclic triaxial test on an isotropically consolidated specimen.

Therefore, a failure criterion based on axial strain has been adopted. The number of cycles producing a given strain is defined by two methods:

- A single amplitude strain—for example, a 2.5% strain is defined as the point where the double amplitude strain (peak-to-peak) equals 5.0%, or where the maximum tension or compression excursion from the static condition equals 2.5%.
- Failure strain—for example, a 2.5 or 5% peak-to-peak strain must be defined.

As a first step in analyzing the results of testing, we have summarized the pore pressure increase and strain vs cycles for specimens subjected to the same effective confining pressure (Fig. 3-9). These data are then replotted (Fig. 3-10) to indicate the number of cycles required to obtain either initial liquefaction or a given level of strain in terms of the stress ratio, $\sigma_{dp}/2\sigma_{3c}$. The use of this stress ratio

minimizes the effects of small variations in confining pressure.

Significant differences exist between the stress conditions in the isotropically consolidated laboratory specimens and *in situ* specimens.⁴⁴ Several methods have been proposed to account for these differences.^{43,52,53} While these methods differ in form, they indicate that the laboratory results (for example, the stress ratio in Fig. 3-10), should be multiplied by a factor ranging from about 0.5 to 0.7 for normally consolidated soils. These results could then be applied to field stress conditions. This factor is mainly applicable to clean sands. For soils containing an appreciable amount of fines, the relationship has not yet been established.^{44,54}

Anisotropic ($K_f > 1.0$) Tests. The *in situ* strength of soil subjected to initial (static) shear stresses is evaluated by anisotropically consolidated triaxial

Fig. 3. Theoretical areas affected at modified-Mercalli intensity V or greater by earthquakes of varying Richter magnitude (R-M) centered in San Francisco.¹⁰

earthquake, and any automatic seismic trip systems would likely have a wide impact.

Power for Reactor Emergency and Decay Heat Loads

If an earthquake were to cause the loss of off-site power, then on-site

ac power would be required to remove decay heat or cope with a loss of coolant accident. If all reactor units at the site have also been shut down by their seismic trip systems, then the diesel generators would be the sole source of ac power. Under these conditions two probability values are required to determine the overall probability that the power requirements of the emergency or decay-heat removal systems will be met following an earthquake. The first is the probability of losing the electrical grid or intertie to the grid in an earthquake. Neither the grid nor intertie is a seismic-category-I system. Since the probability that off-site power will be lost because of reactor trip under a non-earthquake situation is 10^{-3} (Ref. 18), we can use this as a lower bound for the earthquake situation.

The second probability of concern is the probability that the diesel-generator sets will fail to provide a sufficient amount of power after an earthquake. All safety related diesel-generator sets at reactor facilities have redundant backup and are seismic-category-I, i.e., they are designed to withstand the effects of a safe shutdown earthquake. Since the probability that a diesel-generator set will not provide a sufficient amount of power when needed under a

non-earthquake situation is 3×10^{-2} ,¹⁷ this will be used as a lower bound. (The Crooks and Vissing report stated that at a 50% confidence level, the probability of successful diesel-generator set performance is less than 95%,²⁰ which is equivalent to a probability of failure of 5×10^{-2} or more. This value is in rough agreement with the 3×10^{-2} value.) Combining the two probabilities gives a minimum probability for not meeting power requirements of the decay-heat-removal system after an earthquake.

This value is 3×10^{-5} per earthquake. If we use the WASH 1400 value of 7×10^{-4} earthquakes per year, the probability of a core melt from lack of emergency power is about 2×10^{-8} per year. Even though this is a minimum value, it is a very low number when compared to the normal core melt probability of 5×10^{-5} per year. Thus this risk of seismic trip systems is probably not significant.

In addition, both BWR's and PWR's are capable of removing decay heat for several hours, using steam driven pumps independent of ac power.

4. Risk Assessment

At this point, we perform a comparative risk assessment to help establish whether seismic trip instrumentation at nuclear power plants is desirable. We use fault tree analysis to conduct a tradeoff study evaluating the risk of requiring seismic trip systems versus the risk of not requiring them. We define risk as the probability that a loss of life or property occurs. The fault tree in Fig. 4 considers the risk of nuclear power plant operation without seismic trip systems while that in Fig. 5 considers the risk with seismic trip systems. In this section we first qualitatively describe the fault trees in Figs. 4

and 5. Then we quantitatively evaluate them.

FAULT TREE DESCRIPTION

The fault trees in Figs. 4 and 5 are constructed in a similar manner. The left side of each fault tree considers accident sequences that lead to core melt during non-earthquake conditions. These accident sequences are categorized into physically distinct processes by which core melt can occur as was done in the Reactor Safety Study.¹⁷ The left side of Fig. 5 illustrates that because of spurious trips, nuclear power plants with seismic trip

Fig. 4. Fault tree for nuclear plants without seismic scram system.

Fig. 5. Fault tree for nuclear plants with seismic scram system.

instrumentation have increased non-earthquake accident frequencies.

The right side of each fault tree shows accident sequences due to nuclear power plant operation when an earthquake has occurred. This side is further divided into the risk of plant shutdown on public emergency services and the risk of continued plant operation, which might lead to a large fission-product release due to the earthquake's effects. In reference to accidents caused by earthquakes, the fault trees in Figs. 4 and 5 differ in two ways:

- The fault tree without seismic scram (Fig. 4) includes the possibility that a destructive power excursion can occur if the control rods are not fully inserted when an earthquake occurs (see Fig. 4, event 21).
- The fault tree with seismic scram (Fig. 5) includes plant shutdown when a strong motion earthquake occurs and the plant's inability to supply power immediately after the earthquake (see Fig. 5, basic event 49).

FAULT TREE QUANTIFICATION

The fault trees given in Figs. 4 and 5 include the dominant accident sequences leading to a possible core melt. These sequences were taken

directly from the Reactor Safety Study. In addition, these fault trees include earthquake caused events that could lead to core melt for nuclear power plants with and without seismic trip systems. We probabilistically evaluate both fault trees to: 1) establish whether core melts caused by earthquakes are significant contributors to the risk of nuclear power plant operation, 2) determine what difference seismic scram makes to this risk, and 3) conduct a sensitivity analysis to determine which portions of the system are of the greatest concern. Each probability of occurrence of the basic events will be assigned a mean value with uncertainty factors, so that probability confidence intervals can be computed.

The first step in the quantification of fault trees is the assignment of failure probabilities to the basic events. The estimates of the failure probabilities of the basic events with associated uncertainty factors is given in Table 3. The error factor expresses the upper 90% confidence limit to the basic event failure probability by the following relationship:

$q_{upper} = q_{mean} + (\text{uncertainty factor})(q_{mean})$, where q_{mean} is the expected value.

Table 3. Basic event probabilities.

Basic event number	Description of basic event	Reactor type	Expected rate of occurrence	Uncertainty factor
1	Loss of heat removal systems following a reactor trip causes a fuel meltdown	BWR PWR	$3 \times 10^{-5}/y$ $1.6 \times 10^{-5}/y$	10 10
2	Loss of coolant accident occurs during normal operation leads to a core meltdown	BWR PWR	$1.1 \times 10^{-6}/y$ $3.6 \times 10^{-5}/y$	10 10
3	Other initiating events lead to core melt	BWR PWR	$2.9 \times 10^{-6}/y$ $4.9 \times 10^{-6}/y$	10 10
	Damaging earthquake occurs:			
4	PGA = 0.2g (0.1g PGA = 0.25g) ^a		$6.3 \times 10^{-4}/y$	10
5	PGA = 0.5g (0.25g PGA = 0.75g)		$4.6 \times 10^{-5}/y$	10
6	PGA = 1.0g (0.75g PGA)		$1.0 \times 10^{-5}/y$	10
7	Spurious trip caused by seismic scram system		1/y	10
8	Loss of heat removal systems given a reactor trip	BWR PWR	$30 \times 10^{-6}/demand$ $1.6 \times 10^{-6}/demand$	10 10
	Other earthquake caused events leading to a core melt			
9 ^b	PGA = 0.2g		1.2×10^{-7}	30
10 ^b	PGA = 0.5g		4×10^{-6}	7
11 ^b	PGA = 1.0g		4×10^{-4}	3
	Other earthquake caused events leading to a core melt			
12 ^c	PGA = 0.2g		6×10^{-8}	30
13 ^c	PGA = 0.5g		2×10^{-6}	7
14 ^c	PGA = 1.0g		2×10^{-3}	3
	Earthquake disables main feedwater system:			
15	PGA = 0.2g		$5.5 \times 10^{-3}/event$	30
16	PGA = 0.5g		$5.5 \times 10^{-2}/event$	7
17	PGA = 1.0g		$3.2 \times 10^{-1}/event$	3
	Earthquake disables auxiliary feedwater or RHR system:			
18	PGA = 0.2g		3×10^{-3}	30
19	PGA = 0.5g		3×10^{-3}	7
20	PGA = 1.0g		3×10^{-2}	3
	Earthquake prevents control rod insertion or causes destructive power excursion:			
21 ^b	PGA = 0.2g		$1 \times 10^{-5}/event$	100
22 ^b	PGA = 0.5g		$6 \times 10^{-4}/event$	50
23 ^b	PGA = 1.0g		$1 \times 10^{-2}/event$	20

^a PGA stands for the peak ground acceleration of the earthquake.

^b Appears in Fig. 2 fault tree only (i.e., that without seismic scram).

^c Appears in Fig. 3 fault tree only (i.e., that with seismic scram).

Table 3. (Continued).

Basic event number	Basic event description	Expected rate of occurrence	Uncertainty factor
	Earthquake causes loss of coolant accident:		
24	PGA = 0.2g	5.5×10^{-3}	10
25	PGA = 0.5g	5.5×10^{-2}	5
26	PGA = 1.0g	3.2×10^{-1}	3
	ESF's fail because of earthquake damage pipes, valves, etc.:		
27	PGA = 0.2g	$3 \times 10^{-5}/\text{event}$	50
28	PGA = 0.5g	$3 \times 10^{-3}/\text{event}$	7
29	PGA = 1.0g	$3 \times 10^{-2}/\text{event}$	3
	Diesels fail to operate after an earthquake:		
30	PGA = 0.2g	0.05/demand	30
31	PGA = 0.5g	0.1/demand	7
32	PGA = 1.0g	0.25/demand	3
	Interties between nuclear plant and offsite grid are faulted:		
33	PGA = 0.2g	0.05/event	30
34	PGA = 0.5g	0.1/event	7
35	PGA = 1.0g	0.5/event	3
	Loss of emergency backup power systems:		
36	PGA = 0.2g	0.01/demand	30
37	PGA = 0.5g	0.05/demand	7
38	PGA = 1.0g	0.1/demand	1
	Grid lost for other reasons:		
39	PGA = 0.2g	0.05/event	30
40	PGA = 0.5g	0.1/event	7
41	PGA = 1.0g	0.5/event	1
42	Loss of nuclear power plants causes grid to go down	0.1/event	10
	Nuclear power plants are tripped by normal instrumentation:		
43 ^b	PGA = 0.2g	0.01/event	10
44 ^b	PGA = 0.5g	0.1/event	3
45 ^b	PGA = 1.0g	0.5/event	1
	Nuclear power plants are tripped by plant operators:		
46 ^b	PGA = 0.2g	0.01/event	10
47 ^b	PGA = 0.5g	0.05/event	3
48 ^b	PGA = 1.0g	0.1/event	2
49 ^b	Seismic scram system trips nuclear power plants	1.0/event	

The failure probabilities for basic events 1, 2, 3, and 8 appear on the left side of each fault tree and are taken directly from the Reactor Safety Study. For earthquake caused events, three levels of peak ground acceleration (PGA) are considered in the fault trees:

1. $\text{PGA} = 0.2 \text{ g}$ [$0.15 \text{ g} < \text{PGA} < 0.25 \text{ g}$],
2. $\text{PGA} = 0.5 \text{ g}$ [$0.25 \text{ g} < \text{PGA} < 0.75 \text{ g}$],
3. $\text{PGA} = 1.0 \text{ g}$ [$\text{PGA} > 0.75 \text{ g}$].

The safe shutdown earthquake is taken to give a PGA of 0.2 g.

From Hsieh's work,¹⁷ the probability per year that an average reactor site in the Eastern U.S. experiences a PGA in the interval specified above is given by:

1. $P(0.15 \text{ g} < \text{PGA} < 0.25 \text{ g}) = 6.5 \times 10^{-4}$,
2. $P(0.25 \text{ g} < \text{PGA} < 0.75 \text{ g}) = 4.6 \times 10^{-5}$,
3. $P(0.75 \text{ g} < \text{PGA}) = 1.0 \times 10^{-5}$.

Newmark's values, as cited in the Reactor Safety Study, for the probability that a single reactor system will fail when subjected to these accelerations are given in Table 4. The probability of two systems failing is the square of these values if the failures are independent, as shown in column 3 of Table 4. If two failures are completely dependent, the probability of failure is the same as for a single system. The Reactor Safety Study stated that it is reasonable to assume that failure of two systems is dependent, i.e., coupled. The

Table 4. Probability of failure of reactor system(s) subjected to various levels of ground acceleration.*

1 Peak ground acceleration, gravities	2 Probability of a single reactor system failing	3 Probability of two independent reactor systems failing	4 Probability of two coupled reactor systems failing	5 Uncertainty factor
0.2	0.001	10^{-6}	3×10^{-5}	30
0.5	0.02	4×10^{-4}	3×10^{-3}	7
1.0	0.1	10^{-2}	3×10^{-2}	3

*For reactors designed to an SSE of 0.2 g; table taken from Reactor Safety Study.

probability of the coupled failure of two systems is given by the relationship

$$P_{\text{coupled}} = \sqrt{P_{\text{independent}} \times P_{\text{dependent}}},$$

i.e., the square root of the product of the entries in columns 2 and 3 of Table 4. The uncertainty factors result from the fact that the independent probabilities in column 2 represent an upper bound for failure of two systems. We see that the probability of coupled damage in Table 4 differs by a factor of approximately 100 between the 0.2-g and 0.5-g levels and by a factor of 10 between the 0.5-g and 1.0-g levels. These spreads were taken into account when assigning failure probabilities for basic events 9 through 48 in Table 3. Most of the failure probabilities in Table 3 were taken from the Reactor Safety Study or were estimated based on information in Sec. 3 of this report. The engineered safety systems and the auxiliary feedwater system are redundant and seismically qualified. The probability of failure of these systems when subjected to an earthquake is given by P_{coupled} , the probability of failure of two systems that are coupled. The power conversion system, though redundant, is not seismically qualified, and its probability of failure when subjected to an earthquake is conservatively

assigned the probability of failure of a single system.

The earthquake frequencies for basic events 4, 5 and 6 and the associated failures probabilities for events 9 through 48 apply to reactors that are located in the eastern U.S. Reactors located in the western U.S. experience more frequent earthquakes, however they are designed to a larger safe shutdown earthquake. It is assumed, here as in the Reactor Safety Study, that these two effects cancel and that the failure data in Table 3 can be applied to all reactor sites in the U.S.

CONCLUSIONS BASED ON COMPUTER OUTPUT

From the basic event data, we calculate with the IMPORTANCE computer code,²¹ the probability of the top event for the fault trees in Figs. 4 and 5. In addition we compute the probability of the top event for the right side of each fault tree to establish if earthquake caused events are significant contributors to risk. The mean and median values and 90% confidence limits of these probabilities are shown in Table 5 and Fig. 6. We also use the IMPORTANCE computer code to conduct a sensitivity analysis, and the computer output and conclusions based on the output are discussed in detail in Appendix B.

Table 5. Mean values of risk of nuclear power plant operation for various cases.

Case	Associated fault tree	Mean value of risk
I. Top event probability without seismic scram systems	Fig. 4	3.6×10^{-5}
II. Top event probability with seismic scram systems	Fig. 5	4.1×10^{-5}
III. Earthquake caused events only, without seismic scram systems	Right side, Fig. 4	2.4×10^{-6}
IV. Earthquake caused events only, with seismic scram systems	Right side, Fig. 5	4.2×10^{-6}

Fig. 6. Median values of risk at 90% confidence levels for the cases presented in Table 5.

On the basis of the data presented in Table 5 and Fig. 6, we see that seismic trip systems make little difference in the overall risk of nuclear power plant operation. The risk due to earthquake caused events in either case is of the order 10^{-6} , which is greater than reported in the Reactor Safety Study. We cannot say in this case, however, that there is a statistically significant difference between our results and the results in the Reactor Safety Study because of the following two reasons:

- Risk in this report was defined in terms of two basic categories:

- core meltdown accidents resulting from earthquakes and other causes,

- loss of emergency services during an earthquake.

- Engineering judgement was used in many cases to assign probabilities to earthquake caused events in Table 3. The wide error bands on Fig. 6 emphasize the uncertainties involved.

Analysis of the results presented in Appendix B show that core melt-

downs contribute significantly to the risk from earthquake caused events. We assumed the consequences of both core meltdowns and loss of emergency services to be the same when defining risk in this analysis (hence, if they were of equal probability, they gave the same risk), while in fact the consequences are different.

5. Discussion

From the preceding, we may tabulate the key points concerning the use of seismic trips on nuclear power plants. Although not exhaustive, this list covers the most important issues that we discovered have a bearing on the advisability or unadvisability of seismic trips.

ADVANTAGES OF A SEISMIC TRIP SYSTEM

Probably the traditional reason for requiring seismic trips on nuclear reactors was to allow movement of those components necessary for shutdown, e.g., control rods, before they could be affected by the strong motion. Present reactor core and internals are designed for severe loss of coolant accident blowdown loads, so this concern about control rods would appear to be not as pertinent today. However, other active

components are involved in the shutdown of nuclear power plants, and they also might benefit if allowed to complete their action before strong motion arrives. It appears that unless an expensive early warning seismic network is used, from zero to six seconds of advanced warning time is all that will be available. This may allow enough time for the control rods, turbine stop valves, and main steam-line isolation valves to operate but not enough for other active components.

In addition, for severe earthquakes with durations on the order of 30 s, an early shutdown will minimize the number of seismic motion cycles the plant experiences in the operating state. In this way, a seismic trip may decrease risk. Note, however, that nuclear plants are designed to safely shut down during earthquakes

of such severity that they are expected to occur only once in many thousands of years (the safe shutdown earthquake). Thus, the decrease in risk is probably slight.

If a severe earthquake were to cause an accident, it would be an advantage to have the plant shut down when the accident occurred. The most notable example would be an earthquake caused loss of coolant accident. If the plant were shut down a few seconds before the accident, some of the energy stored in the fuel rods would be removed before blowdown and the resulting loss of heat-transfer capability. This would reduce the temperature transient on the cladding and minimize the chance of cladding damage.⁷

The reactor coolant system is designed to withstand a very severe earthquake — again the safe shutdown earthquake. Newmark¹⁴ estimates that if twice the ground motion of a safe shutdown earthquake is experienced, the probability of exceeding design limits is 0.01. Thus, according to his analysis, the probability of an earthquake causing a loss of coolant accident is remote. In addition, the Reactor Safety Study showed that earthquakes were not significant contributors to the probability of a core meltdown accident.

DISADVANTAGES OF A SEISMIC TRIP SYSTEM

A possible disadvantage of seismic trips is that they would cause unwanted spurious trips. By use of spectrum-sensitive detectors and redundant channel logic, electronic spurious trips can probably be satisfactorily minimized. However, in seismically active areas, unwanted trips from small earthquakes could be a problem, particularly if the trip level is set low. Low trip levels are desirable since they maximize the chance of tripping a few seconds before the arrival of strong motion. What would be required is a determination of an appropriate trip level for a given reactor site, based on its earthquake history. In this way, if a seismic trip system were employed, spurious trips could be minimized while the advantages of the system maximized.

Another disadvantage of a seismic trip system is the effect it would have on a utility's electrical generation and transmission capability following an earthquake. The sudden loss of nuclear units on a utility grid might cause a severe electrical transient, resulting in prolonged outages, particularly if the less stringently designed fossil units were kicked off by the same earthquake. While most public emergency service

coordinators anticipate power outages of several hours following a severe earthquake, outages of longer duration could severely hamper recovery efforts. Since it is not clear how long a nuclear plant would remain down after a seismic trip, this tripping action

could produce more, not less of a risk to the public. This disadvantage of seismic trips would require a high trip-level setting (to minimize trips from aftershocks) and well defined restart criteria (to allow rapid restart when safe to do so).

6. Summary

Based on the arguments developed in Sec. 3 and the risk assessment of Sec. 4, the general statement can be made that seismic trip systems on commercial nuclear power reactors would have a small and somewhat uncertain effect on plant safety. When viewed from the standpoint of risk to society as a whole, the seismic trip systems might actually increase the risk by denying a source of much needed electric power during the time of an earthquake emergency.

In assessing the advisability of seismic trip systems, we must keep in mind that since nuclear power plants are carefully designed to withstand an earthquake, they would be expected to affect risk only during a very severe earthquake. The results of such an event would be serious in any case and the nuclear power plant's response to the earthquake could either add to or help mitigate an already serious situation.

If, however, a seismic trip is deemed necessary, existing seismic instrumentation could be upgraded to provide the trip capability without incurring a severe cost. Instead of a single-frequency accelerometer switch sensing the P wave or early arrivals, a series of seismic switches of varying frequency sensitivity could be used. These switches could be used at selected points of the nuclear power plant's structure to approximate the design response spectrum of the structure. Should the ground motion input approach the response spectrum of the structure, appropriate action may be taken at whatever level deemed proper. It is possible to use a response spectrum recorder that uses discrete reeds for selected frequencies for this purpose.

Should this seismic trip be incorporated into the plant safety network, the design and installation would have to meet the criteria in

ANSI N42.7²² and be qualified to ANSI 41.5.²³ The frequency of spurious trips could be reduced by using 2/3 channel logic or some similar scheme.

If a seismic trip system were to be employed, a key question would be the determination of the trip level setting. As low as possible a setting would assure the maximum possible time between trip and strong motion arrival (see Fig. A-3 in Appendix A). In seismically inactive areas, settings in the 0.01-g range might be acceptable to assure this. On the other hand this would probably be an unacceptably low trip level in seismically active areas, such as California. Here a higher trip level would be desirable to minimize unnecessary trips. Since nuclear

plants in seismically active areas are designed to withstand larger earthquakes, such an increase in trip level could not result in an increased risk. For any trip level, an advanced warning time of at most a few seconds would be expected, and such a short time would not have a dramatic effect on plant safety.

Although it may be likely that existing instrumentation or the operator will trip the plant during a damaging earthquake, there is no assurance of this, and in any event the trip would occur during the strong motion with no benefit of advanced warning. If seismic trip is deemed desirable, a specifically designed instrumentation system for this purpose would be preferable.

7. References

1. F. Tokarz, M. Zaslawsky, and G. St. Leger-Barter, *Evaluation of the Use of Seismic Design Systems for Power Reactors*, Lawrence Livermore Laboratory, Rept. UCRL-51619 (1974).
2. *Seismic Design Criteria: Appendix A, Seismic and Geologic Siting Criteria for Nuclear Power Plants* (10CFR100), U.S. Atomic Energy Commission (Federal Register, Washington, D.C., April 30, 1975).
3. *Instrumentation for Earthquakes*, Nuclear Regulatory Commission, Washington, D.C., regulatory guide 1.12 (1974).
4. *Earthquake Documentation Criteria for Nuclear Power Plants*, American National Standards Institute, NY, Standard N18.5 (1974).
5. *Guidelines for Retrieval, Review, Processing, and Evaluation of Records Obtained from Seismic Instrumentation*, American Nuclear Society, proposed standard N643, Draft 2 (1976).
6. *IEEE Std. 386-1971 Qualification of Class 1 Electric Equipment for Nuclear Generating Stations*, American National Standards Institute, standard N41.7 (1971).
7. Portland General Electric Co., Oreg., *Trojan Nuclear Plant, Final Safety Analysis Report, Vol. 7, docket-50344-44* (Feb. 5, 1975).
8. *General Specification for Electrical Relays*, military specification MIL-R-5757F (May 8, 1970).
9. M. Shaw, AEC Division of Reactor Technology, private communication to J. Ramey, AEC Commissioner (Jan. 1970).
10. T. H. Thomas et al., *Nuclear Reactors and Earthquakes*, AEC Div. of Tech. Info., Washington, Rept. TID-7024 (1963).
11. R. Curran, Argonne National Laboratory, private communication (Nov. 1975).
12. Office of Naval Reactors, U.S. Energy Research and Development Administration, *Preliminary Safety Analysis Report, Light Water Breeder Reactor* (Shipping Port reactor) (1975).
13. Duke Power Co., Charlotte, N.C., *Preliminary Safety Analysis Report, Oconee Pressurized Water Reactor*, docket 50-269 (1968).
14. N. M. Newmark, *Probability of Predicted Seismic Damage in Relation to Nuclear Reactor Facility Design* (Draft), Consulting Engineering Services, Urbana, Illinois, prepared for the Office of Nuclear Regulatory Research, NRC (1975).
15. H. E. Siebe, "Nuclear Industry - Approach to Performance Assurance," in *Proc. Reliability and Maintainability Symposium* (IEEE, NY., 1976), p. 430.

16. R. L. Scott, *Nuclear Safety* 17 (5), 562 (1976).
17. *Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, Rept. WASH 1400 (1975).
18. J. Wells and J. Savage, *Fault Tree Analysis of the Emergency Power System of Two Light-Water Nuclear Power Reactors*, Lawrence Livermore Laboratory, Rept. UCID-16666 (1975).
19. J. L. Mulloy, *San Fernando Earthquake of February 9, 1971 - Effects on Power System Operation and Facilities*, Dept. of Water and Power of the City of Los Angeles (1974).
20. J. L. Crook and G. S. Vissing, *Nuclear Safety* 16 (2), 224 (1975).
21. H. Lambert, *Fault Trees for Decision Making in Systems Analysis*, Lawrence Livermore Laboratory, Rept. UCRL-51829 (1975).
22. *Criteria for Protection Systems for Nuclear Power Generating Stations*, American National Standards Institute, standard N42.7 (IEEE 279) (1971).
23. *Qualifying Class I Electric Equipment for Nuclear Power Generating Stations*, American National Standards Institute, standard 41.5 (IEEE 323) (1971).

Appendix A: Time Interval Analysis

If sufficient time is available between a trip signal and the arrival of strong motion, the advantage of a seismic trip system is increased. A discussion of "sufficient time" is included under "Variables Affecting the Frequency of Seismic Trips" in this report. Other methods can be used to enhance this time interval.^{A1} The objective of this appendix is to determine, from inherent ground-wave characteristics, what the time interval will be between a trip signal and the arrival of strong motion. To accomplish this task, we collected and analyzed 24 accelerograms. These strong motion accelerograms are listed in Table A-1.^{A2,A3,A4,A5} All the accelerograms analyzed were from sites in the western U.S. This is due to the lack of accelerograms from the eastern portion of the U.S. The 24 accelerograms listed in Table A-1 are from 17 different earthquakes. An analysis was performed to determine if the data appeared biased because of this fact. The analysis result using only 17 accelerogram records, one from each earthquake, vs the 24 listed in Table A-1 did not differ significantly.

Prior to determining the time interval in question, we developed a relationship between range to epicenter and time to strong motion. Since the definition of strong motion is not fixed, the analysis employed several values. Since the accelerograms analyzed in this study are taken from the time period 1933 to 1971, the accelerographs used to record these accelerograms were not all identical, and the starting characteristics of each are a little different. However, it can be said that, in general, these accelerograms were triggered at approximately 0.01 g. Note also that the interval between the arrival of a motion and the time the accelerograph actually starts is 0.1 to 0.2 s.

Figure A-1 illustrates the relationship between range to epicenter and time to strong motion when strong motion is defined as 0.1 g. The figure gives the equation for a linear fit through the data points shown. Several forms other than linear were tried, but the linear relationship provided a "best" fit and seems logical on physical grounds. The correlation coefficient associated with that regression is 0.82. Figure A-2 also illustrates this relationship, but for the case when strong motion is defined as 0.15 g.

Figures A-1 and A-2 show that the time interval of interest increases as the range increases. This relationship reflects the fact that P and S waves

Table A-1. Pertinent salient characteristics of accelerograms.

Accelerogram No.	Earthquake; date	Recording station	Richter magnitude	Ground motion
1	El Centro; May 18, 1940	El Centro, CA	7.0	S W
2	Northwest Calif./ Oct. 7, 1951	Ferndale, CA	6.0	S44W N46W
3	Kern County; July 21, 1952	Santa Barbara, CA	7.7	N42E S48E
4	Kern County; July 21, 1952	Taft, CA	7.7	N21E S69E
5	Eureka; Dec. 21, 1954	Eureka, CA	6.6	N11W N79E
6	Eureka; Dec. 21, 1954	Ferndale, CA	6.6	N44E N46W
7	San Jose; Sept. 4, 1955	San Jose, CA	5.8	N59E N31W
8	San Francisco; March 22, 1957	Golden Gate Park, S.F.	5.3	N10E S80E
9	San Francisco; March 22, 1957	S.F. State Bldg., S.F.	5.3	N09E N81W
10	Hollister; April 8, 1961	Hollister, CA	5.5	S01N N89W
11	Borrego Mountain; April 8, 1968	El Centro, CA	6.5	S W
12	Lythe Creek; Sept. 12, 1970	San Bernardino Hall of Records, CA	5.4	N E
13	San Fernando; Feb. 9, 1971	Orion Blvd., L.A.	6.4	N W
14	Lytle Creek; Sept. 12, 1970	Wrightwood, CA	5.4	S65E S25W
15	Lytle Creek; Sept. 12, 1970	Devils Canyon, CA	5.4	S E
16	Parkfield; June 27, 1966	Cholame-Shandon, CA	5.6	N65E N25N
17	San Fernando; Feb. 9, 1971	Pacoima Da, CA	6.4	S74W S16E
18	Parkfield; June 27, 1966	Cholame-Shandon, CA	5.6	N05W N85E
19	Long Beach; March 10, 1933	Vernon, CA	6.3	S08W N82W

Table A-1. (Continued)

Accelerogram No.	Earthquake; date	Recording station	Richter magnitude	Ground motion
20	El Centro; Dec. 30, 1934	El Centro, CA	6.5	S W
21	Helena; Oct. 31, 1935	Helena, MT	6.0	S W
22	Ferndale; Sept. 11, 1938	Ferndale, CA	5.5	N45E S45E
23	Olympia; April 13, 1949	Olympia, WA	7.1	N04W N86E
24	Olympia; April 29, 1965	Olympia, WA	6.5	S04E S86W

Fig. A-1. Time interval vs range to epicenter (strong motion = 0.1 g).

Fig. A-2. Time interval vs range to epicenter (strong motion = 0.15 g).

travel at different velocities. Therefore, as Newmark and Rosenblueth have stated, the time between the arrival of the P and S waves is variable as a function of distance to epicenter.^{A6} This time interval is used by the seismologist to locate the foci of earthquakes, considering the type of soil and rock the waves must travel through. The interval helps in our case, since the slower S waves usually cause the strong motion. The data scatter in Figs. A-1 and A-2 can be attributed to the varied type of soil through which the waves had to travel to reach the accelerographs.

Table A-2 lists the data points that were derived from the selected 24 accelerographs. The data points with strong motion defined at 0.25, 0.30 and 0.35 g are very dependent on each other because they came from the same accelerographs.

One problem that has been discussed elsewhere in this report is the problem of determining at what level of acceleration the seismic trip system should be set. While the final decision as to what acceleration this should be involves many different factors, Fig. A-3 helps to provide a feeling for the relationships involved between the level the system is set at and the time

Table A-2. Times between start of accelerogram and arrival of strong motion.

Accelerogram No.	Time, s					
	until arrival of strong motion of magnitude (g):					
	0.1	0.15	0.20	0.25	0.30	0.35
1	0.7	1.4	1.7	2.1	2.1	2.2
2	4.1					
3	3.6	3.7				
4	6.9					
5	3.1	3.5	3.5	3.5		
6	6.6	6.7	7.1			
7	0.9					
8	1.4					
9	1.9					
10	1.0	1.2				
11	7.9					
12	1.5	2.1				
13	2.0	3.4				
14	1.7	1.9				
15	2.2	2.2				
16	0.9	6.1				
17	5.0	6.3	6.3			
18	2.2	3.8	3.8	3.9	3.9	3.9
19	5.5	6.8	7.3	7.3	7.4	7.4
20	0.2	1.9	2.2	2.2	2.2	2.3
21	1.0	1.1				
22	0.2	0.3	2.0			
23	2.5	4.3	12.5			
24	1.3					
μ^*	2.68	3.34	5.16	3.80	3.90	3.95
σ	2.21	2.10	3.52	2.11	2.48	2.43
cv	0.83	0.63	0.68	0.56	0.63	0.61
n	24	17	9	5	4	4

* μ gives sample mean, σ is standard deviation, cv is the coefficient of variation, n is the number of data points.

that elapses between such acceleration and strong motion. Note that Fig. A-3 does not take into account the time between receiving the signal and starting the accelerograph. Figure A-3 also assumes that the 24 accelerograms used in this analysis are representative of what a typical reactor site might experience.

To assess the likelihood of exceeding a specified time interval, we need a relationship between distance and peak ground acceleration. Mickey's correlation^{A7} and the Esteva-Rosenblueth equation^{A2} are two such relationships. Another is the relationship presented by Matthiesen, Howard, and Smith.^{A8} For the purpose of this analysis, the Esteva-Rosenblueth equation shown in Fig. A-4 will be used. This equation is:

$$a = \frac{0.778 \exp(0.8M)}{R_h^2}, \quad (A-1)$$

where

a = peak ground acceleration (g),

M = Richter magnitude of the earthquake,

R_h = hypocentral distance (mi).

Fig. A-3. Relationships between trip level and time to strong motion.

Fig. A-4. Estimated relationships between acceleration, distance, and magnitude.^{A2} Focal depth is 15 mi.

The hypocentral distance R_h relates the focal depth h and the epicentral distance R by the following equation:

$$R_h = \sqrt{R^2 + h^2} . \quad (A-2)$$

The focal depth h was assumed to be 15 mi in our analysis. This is a typical focal depth of earthquakes along the west coast. The relationships we need relating R to time interval are given in Figs. A-1 and A-2 for strong motion defined as 0.10 g and 0.15 g, respectively. The Richter magnitude, M , was related to the number of earthquakes per year per square mile, N , by using^{A7}:

$$\log_{10} N = -1.72 - 0.7M \quad (A-3)$$

Note that Eq. (A-3) is referred to in Fig. A-5. Equations (A-1) and (A-2) were substituted into Eq. (A-3), giving the number of earthquakes per year per square mile, N:

$$N = 10^{-1.72 - 0.88 \ln \left[\frac{a(R^2 + 225)}{0.778} \right]} \quad (A-4)$$

For the case of strong motion defined as 0.15 g, the following equations based on the 24 data points of Table A-1 are appropriate:

$$a = 0.15 \text{ g}, \quad (A-5)$$

$$R = \frac{T - 1.12}{0.05} \text{ mi}, \quad (A-6)$$

where T is the time interval in seconds.

The number of earthquakes per year per square mile was converted to a relative probability that has been weighted by the circumference of a circle with a radius equal to the epicentral distance, R. Since R is a function of the time interval T, a relationship between T and relative probability of T could be obtained. This relationship is illustrated in Fig. A-6.

After performing this analysis of earthquake data, we can make the following generalization. Based on the data in Table A-2 and the relationships provided in Figs. A-1 and A-2, the expected interval between the starting of the accelerograph record and the arrival of the strong motion is seen to be in the range of three to five seconds. Figure A-6 shows the distribution of these time intervals.

Fig. A-5. The seismicity of the Western U.S.^{A7}

Fig. A-6. Relative probability values for various time intervals between seismic trip and a strong motion of 0.15 g.

REFERENCES

- A1. F. Tokarz, M. Zaslowsky, and G. St. Leger-Barter, *Evaluation of the Use of Seismic Scram Systems for Power Reactors*, Lawrence Livermore Laboratory, Rept. UCRL-57619 (1974).
- A2. J. Dalal, *Probabilistic Seismic Exposure and Structural Risk Evaluation*, Ph.D. thesis, Stanford University, Palo Alto, CA (1973).
- A3. Robert L. Wiegel, *Earthquake Engineering*, (Prentice-Hall, Inc., Englewood Cliffs, N.J., 1970), p. 83.
- A4. J. L. Coffman and C. A. Von Hake, *Earthquake History of the United States*, Department of Commerce, Environmental Data Services, Boulder, Colorado, Publication 41-1 (1973).
- A5. *Strong Motion Earthquake Accelerograms*, California Institute of Technology, Pasadena, CA, Rept. EERL 71-20, Vol. I (1971).
- A6. N. Newmark and E. Rosenblueth, *Fundamentals of Earthquake Engineering* (Prentice-Hall, Inc., Englewood Cliffs, N.J., 1971), p. 84.
- A7. T. Hsieh, D. Okrent, and G. E. Apostolakis, *On the Average Probability Distribution of Peak Ground Acceleration in the U.S. Continent Due to Strong Earthquakes*, University of California at Los Angeles, Rept. UCLA-ENG-7516 (1975).
- A8. R. Matthiesen, G. Howard, and C. B. Smith, *Seismic Safety of Power Plants - Appendix IV, Public Health Risks of Thermal Power Plants*, University of California at Los Angeles, Rept. UCLA-ENG-7242 (1972).

Appendix B: Output and Conclusions from IMPORTANCE Computer Code

We used the probabilities of basic event failures as input to the IMPORTANCE* computer code to calculate the probability of the top event in the fault trees in Figs. 4 and 5. We also calculated the probabilistic importance of the basic events and of min cut sets in both fault trees. (A cut set is a set of basic events whose occurrence causes the top event to occur; a min cut set is a cut set that cannot be reduced and still ensure occurrence of the top event.) In addition we compute the right hand side of each fault tree to establish if earthquake caused events contribute significantly to the total risk.

Probabilistic importance for a basic event is defined as the probability that a cut set containing the basic event occurs, conditioned on the probability of the top event. The importance of a min cut set is defined as the probability that a min cut set occurs, conditioned on the probability of the top event. In the reference given below, these importance measures are called the Fussell-Vesely measures of probabilistic importance.*

The IMPORTANCE computer code was used in computing these importance measures. The output is shown in Tables B-1, B-2, B-3, and B-4. In each table the probability of the top event, basic event importances, and cut set importance are shown. In the first column of each table, the mean probability is given (i.e., the probability when all basic events are calculated at their expected values). In addition 98%, 90%, and 80% confidence intervals are given. The column labeled 500 gives the median probability. A log normal distribution was used for all basic events in expressing the uncertainty in the failure probabilities. Examination of these tables show that for the probability of the top event, the median value exceeds the mean value. The reverse is true for the importance values, since they are the ratios of probabilities with the probability of the top event appearing in the denominator.

The median and 90% confidence levels for the probability of the top event in Tables B-1, B-2, B-3 and B-4 are plotted in Fig. 6 of Sec 4. Wide uncertainty bands for each probability are due to the uncertainties in the failure-rate data.

*See: H. E. Lambert, *Fault Trees for Decision Making in Systems Analysis*, Lawrence Livermore Laboratory, Rept. UCRL-51829 (1975).

Because of the wide overlap of these uncertainty bands, we cannot say with a high degree of statistical confidence that seismic scram systems make any difference in the risk of nuclear power plant operation.

In Table B-2 (for plants with seismic trip systems), we see that the mean probability of the top event is 4.1×10^{-5} , which is slightly greater than the mean probability of 3.6×10^{-5} given for plants without seismic trip systems in Table B-3. This difference is due in part to basic event 49 (seismic scram system trips nuclear power plants), which is contained in the fault tree in Fig. 4 but not in Fig. 5. We see in Table B-2 that basic event 49 contributes about 6% of the total probability; hence we conclude tripping the power plant at the time of the earthquake increases not decreases the total risk. In addition we see from Table B-1 that min cut set No. 4, basic event 7 and basic event 8, contributes 7% of the total probability. From this observation, we conclude that spurious scrams due to seismic scram instrumentation also tend to increase the total risk. However, as stated above, these differences may not be of statistical significance. Also if we compare the probability of the top event in Table B-1, 4.1×10^{-5} , to that in Table B-2, 4.2×10^{-6} , we see that in this analysis earthquake caused events contribute approximately 10% of the total risk. The same conclusion is obtained comparing Tables B-3 and B-4. The probabilities of the earthquake caused events are approximately a factor of ten greater than earthquake caused events that lead to core melt cited in the Reactor Safety Study. As stated in Sec. 4 of this report, this increase is due to the inclusion of loss of emergency services in the definition of risk in this report, while it is not included in the Reactor Safety Study.

TABLE B-1

IMPORTANCE LISTING FOR FAULT TREE WITH SEISMIC SCRAM

MONTE CARLO SIMULATION RESULTS	MCAN	10	50	100	500	900	950	980
CONFIDENCE LEVEL IN PER CENT		(98)	(90)	(80)		(80)	(90)	(98)
PROBABILITY OF TOP EVENT	4.120E-05	1.206E-05	2.167E-05	3.090E-05	7.806E-05	4.804E-04	8.726E-04	2.740E-03

1

MONTE CARLO SIMULATION RESULTS	MCAN	10	50	100	500	900	950	980
CONFIDENCE LEVEL IN PER CENT		(98)	(90)	(80)		(80)	(90)	(98)
RANK	BASIC EVENT							
1	BE 1	7.262E-01	3.503E-03	1.702E-02	3.670E-02	3.501E-01	6.544E-01	9.077E-01
2	DE 6	7.262E-02	7.144E-05	7.181E-04	1.035E-03	2.621E-02	4.004E-01	6.489E-01
3	BE 7	7.262E-02	7.144E-05	7.181E-04	1.035E-03	2.621E-02	4.004E-01	6.489E-01
4	BE 3	7.039E-02	3.105E-04	1.548E-03	2.219E-03	2.749E-02	2.317E-01	3.813E-01
5	BE 49	6.479E-02	1.051E-03	6.757E-03	1.699E-02	1.569E-01	6.835E-01	7.801E-01
6	BE 6	4.814E-02	2.572E-04	1.044E-03	2.222E-03	3.135E-02	2.830E-01	4.103E-01
7	BE 42	4.059E-02	5.127E-04	2.633E-03	5.090E-03	7.257E-02	5.291E-01	6.998E-01
8	BE 3	3.374E-02	7.572E-05	4.675E-04	9.262E-04	1.648E-02	1.735E-01	2.918E-01
9	BE 4	3.115E-02	1.404E-04	6.399E-04	1.741E-03	5.378E-02	6.035E-01	6.424E-01
10	BE 2	2.670E-02	7.907E-05	3.582E-04	9.815E-04	3.378E-03	8.905E-02	1.695E-01
11	BE 5	2.480E-02	1.270E-04	8.316E-04	1.502E-03	2.369E-04	7.973E-01	4.448E-01
12	BE 26	2.367E-02	3.300E-05	1.458E-04	4.613E-04	1.991E-02	4.159E-01	6.556E-01
13	BE 32	2.209E-02	6.328E-05	3.043E-04	6.835E-04	7.841E-03	1.104E-01	1.944E-01
14	BE 38	1.456E-02	1.430E-05	1.725E-04	3.780E-04	6.883E-03	7.437E-02	1.370E-01
15	BE 31	1.249E-02	4.780E-05	1.603E-04	3.636E-04	6.103E-03	1.339E-01	2.596E-01
16	BE 37	1.117E-02	2.223E-05	1.425E-04	3.701E-04	6.719E-03	1.228E-01	2.070E-01
17	BE 40	1.104E-02	2.954E-05	1.337E-04	3.192E-04	6.780E-03	1.182E-01	1.939E-01
18	BE 39	1.093E-02	6.768E-06	1.187E-04	4.051E-04	1.147E-02	2.970E-01	4.849E-01
19	BE 28	8.937E-03	2.405E-05	1.286E-04	3.247E-04	3.142E-03	4.003E-02	7.448E-02
20	BE 30	7.400E-03	6.279E-06	4.377E-05	1.811E-04	5.378E-03	2.170E-01	3.397E-01
21	BE 20	6.691E-03	2.126E-05	9.334E-05	2.051E-04	3.440E-03	3.434E-02	5.751E-02
22	BE 17	3.107E-03	1.031E-05	3.936E-05	9.641E-05	5.858E-04	1.697E-02	3.507E-02
23	BE 29	2.319E-03	4.602E-06	2.208E-05	5.789E-05	8.118E-04	1.021E-02	1.771E-02
24	BE 35	1.942E-03	3.144E-05	1.428E-05	3.607E-05	3.488E-04	6.299E-03	1.840E-02
25	BE 19	8.545E-04	2.063E-06	1.273E-05	2.691E-05	5.411E-04	6.309E-03	1.807E-02
26	BE 25	8.291E-04	1.757E-06	1.269E-05	3.040E-05	6.937E-04	8.937E-03	2.032E-02
27	BE 16	4.713E-04	9.430E-07	6.110E-06	1.400E-05	3.244E-04	7.666E-03	1.418E-02
28	BE 14	4.713E-04	9.430E-07	6.083E-06	1.396E-05	1.912E-04	1.785E-03	3.347E-03
29	BE 20	1.542E-04	1.560E-07	1.232E-06	3.324E-06	7.611E-05	1.368E-03	2.936E-03
30	BE 18	7.305E-05	5.917E-08	5.343E-07	1.336E-06	4.087E-05	1.075E-03	3.198E-03
31	BE 24	4.730E-05	3.358E-08	3.915E-07	1.059E-06	4.395E-05	1.812E-03	4.136E-03
32	BE 34	3.071E-05	6.390E-08	7.197E-08	2.312E-07	1.162E-05	4.186E-04	9.401E-04
33	BE 15	2.774E-05	9.131E-08	8.019E-08	2.769E-07	1.425E-05	1.051E-03	3.014E-03
34	BE 27	2.603E-06	4.993E-10	7.607E-09	1.085E-08	6.357E-07	5.804E-05	1.634E-04
35	BE 13	2.033E-06	6.918E-09	2.274E-08	5.831E-08	8.940E-07	1.295E-05	2.349E-05
36	BE 12	9.467E-07	9.303E-10	4.959E-09	1.573E-08	4.157E-07	9.542E-06	2.333E-05
37	BE 33	6.508E-07	1.170E-11	3.054E-10	1.201E-09	1.635E-07	3.774E-05	1.213E-04

TABLE B-1 (cont'd.)

MONTE CARLO SIMULATION RESULTS		10	50	100	500	900	950	990
CONFIDENCE LEVEL IN PER CENT		(95)	(90)	(80)		(80)	(90)	(98)
RANK	MIN CUT SET NO.							
1	1	7.262E-01	3.593E-03	1.582E-02	3.670E-02	3.501E-01	8.544E-01	9.714E-01
2	4	7.282E-02	7.144E-05	7.181E-04	1.635E-03	2.621E-02	4.004E-01	8.784E-01
3	3	7.039E-02	3.105E-04	1.048E-03	2.219E-03	2.749E-02	2.317E-01	6.931E-01
4	2	2.670E-02	7.907E-03	3.582E-04	9.815E-04	3.376E-03	8.905E-02	4.680E-01
5	28	1.578E-02	3.120E-06	4.828E-05	1.044E-04	6.678E-03	1.653E-01	7.786E-01
6	17	1.214E-02	2.197E-05	1.091E-04	2.681E-04	4.420E-03	8.131E-02	8.825E-02
7	16	1.214E-02	1.188E-05	1.121E-04	2.331E-04	4.980E-03	5.572E-02	1.179E-01
8	14	7.889E-03	1.139E-06	1.257E-05	4.861E-05	3.273E-03	1.451E-01	3.208E-01
9	27	5.583E-03	3.920E-06	5.979E-05	9.053E-05	2.513E-03	5.240E-02	1.034E-01
10	30	5.583E-03	7.782E-06	4.094E-05	7.670E-05	2.214E-03	4.159E-02	8.791E-02
11	21	5.583E-03	5.915E-06	3.261E-05	8.384E-05	2.221E-03	4.367E-02	8.017E-02
12	18	5.583E-03	5.084E-06	3.913E-05	9.985E-05	2.145E-03	5.083E-02	9.677E-02
13	29	4.733E-02	2.198E-06	1.541E-05	5.454E-05	1.623E-03	8.793E-02	2.007E-01
14	13	3.641E-03	8.134E-06	2.971E-05	7.354E-05	1.397E-03	1.834E-02	2.695E-02
15	43	3.884E-03	6.635E-06	2.907E-05	8.075E-05	5.966E-04	1.766E-02	2.947E-02
16	31	2.427E-03	1.789E-06	1.846E-05	4.643E-05	8.815E-04	1.392E-02	2.557E-02
17	20	2.367E-03	2.372E-07	3.254E-06	1.922E-05	7.992E-04	5.337E-02	1.350E-01
18	10	2.330E-03	6.132E-06	2.271E-05	5.032E-05	5.490E-04	1.301E-02	2.259E-02
19	24	2.330E-03	4.602E-06	2.208E-05	5.789E-05	6.118E-04	1.021E-02	5.686E-02
20	28	2.427E-03	2.335E-06	2.133E-05	4.461E-05	6.561E-04	1.473E-02	3.002E-02
21	40	1.942E-03	3.144E-06	1.428E-05	3.807E-05	3.468E-04	8.259E-03	1.640E-02
22	34	7.767E-04	7.226E-07	4.490E-06	1.165E-05	9.902E-05	5.043E-03	9.809E-03
23	25	7.282E-04	6.621E-07	4.505E-06	1.323E-05	1.950E-04	4.339E-03	7.938E-03
24	48	7.767E-04	6.124E-07	4.457E-06	1.146E-05	2.326E-04	4.776E-03	8.520E-03
25	7	4.855E-04	9.632E-07	6.003E-06	1.396E-05	1.912E-04	1.785E-03	3.347E-03
26	12	3.350E-04	1.745E-07	2.241E-06	5.315E-06	1.274E-04	2.608E-03	6.440E-03
27	33	3.350E-04	4.019E-07	2.485E-06	5.704E-06	1.296E-04	2.624E-03	2.888E-02
28	24	3.071E-04	1.712E-07	1.231E-06	3.571E-06	1.302E-04	4.905E-03	9.714E-03
29	42	3.071E-04	1.645E-07	1.190E-06	3.734E-06	1.169E-04	2.770E-03	6.357E-03
30	45	3.071E-04	1.998E-07	1.448E-06	3.821E-06	1.211E-04	2.985E-03	6.957E-03
31	23	2.781E-04	8.114E-08	6.331E-07	1.970E-06	1.087E-04	5.286E-03	1.589E-02
32	18	1.842E-04	1.808E-07	1.232E-06	3.324E-06	7.611E-05	1.368E-03	2.936E-03
33	9	1.842E-04	1.546E-07	9.542E-07	3.129E-06	7.874E-05	1.587E-03	4.140E-03
34	37	1.214E-04	1.449E-08	2.390E-07	8.135E-07	4.298E-05	1.803E-03	3.826E-03
35	35	2.367E-04	2.153E-08	2.211E-07	9.771E-07	8.013E-05	6.958E-03	2.235E-02
36	27	4.733E-05	1.265E-08	1.721E-07	3.571E-07	1.810E-05	6.272E-04	1.649E-03
37	39	3.071E-05	6.390E-09	7.497E-08	2.312E-07	1.162E-05	4.166E-04	9.401E-04
38	44	2.603E-05	4.003E-09	3.589E-08	1.560E-07	1.079E-05	6.054E-04	1.632E-03
39	32	2.603E-05	1.580E-09	2.621E-08	1.013E-07	9.703E-06	7.387E-04	2.911E-03
40	11	2.387E-05	2.375E-09	4.470E-08	1.670E-07	7.928E-06	8.834E-04	1.243E-03
41	41	1.302E-06	4.786E-10	1.274E-08	3.875E-08	4.935E-06	4.733E-04	1.399E-03
42	17	2.603E-06	4.062E-10	7.607E-09	1.985E-08	8.857E-07	5.904E-05	1.634E-04
43	8	2.603E-06	3.988E-10	3.997E-09	6.802E-08	8.511E-07	2.221E-05	1.607E-04
44	6	2.233E-06	6.918E-09	2.274E-08	6.631E-08	9.940E-07	1.295E-05	2.349E-05
45	5	9.467E-07	9.383E-10	4.595E-09	1.573E-08	4.157E-07	9.542E-06	2.333E-05
46	38	6.008E-07	1.170E-11	3.054E-10	1.201E-09	1.635E-07	3.764E-05	1.219E-04

TABLE B-2

IMPORTANCE LISTING FOR FAULT TREE WITH SEISMIC SCRAM, RHS ONLY

MONTE CARLO SIMULATION RESULTS									
CONFIDENCE LEVEL IN PER CENT	MEAN	10	50	100	500	900	950	990	
	(98)	(98)	(90)	(80)	(80)	(80)	(90)	(98)	
PROBABILITY OF TOP EVENT	4.198E-08	1.072E-06	2.207E-06	3.662E-06	2.325E-09	2.086E-04	4.278E-04	1.882E-03	
1									
MONTE CARLO SIMULATION RESULTS									
CONFIDENCE LEVEL IN PER CENT	MEAN	10	50	100	500	900	950	990	
	(98)	(98)	(90)	(80)	(80)	(80)	(90)	(98)	
RANK	BASIC EVENT								
1	BE 48	6.059E-01	9.659E-02	2.171E-01	3.013E-01	6.155E-01	9.424E-01	9.748E-01	9.927E-01
2	BE 8	4.529E-01	7.763E-04	3.266E-03	9.352E-03	2.152E-01	8.624E-01	9.526E-01	9.910E-01
3	BE 42	3.979E-01	8.100E-03	2.212E-02	4.071E-02	3.439E-01	8.805E-01	9.336E-01	9.836E-01
4	BE 41	3.311E-01	2.387E-04	1.177E-03	4.047E-03	1.177E-01	6.788E-01	7.954E-01	9.201E-01
5	BE 4	3.057E-01	3.911E-04	4.898E-03	1.376E-02	3.008E-01	9.271E-01	9.814E-01	9.960E-01
6	BE 5	2.415E-01	5.637E-04	2.658E-03	6.332E-03	1.304E-01	7.513E-01	8.943E-01	9.774E-01
7	BE 32	2.165E-01	2.634E-04	1.386E-03	3.650E-03	7.775E-02	4.422E-01	5.961E-01	7.732E-01
8	BE 38	1.429E-01	1.246E-04	8.205E-04	2.255E-03	4.208E-02	2.687E-01	3.910E-01	5.901E-01
9	BE 36	2.323E-01	9.988E-04	7.243E-04	2.713E-03	7.121E-02	7.853E-01	9.176E-01	9.800E-01
10	BE 31	1.217E-01	2.052E-04	7.320E-04	1.575E-03	3.849E-02	4.227E-01	5.863E-01	8.709E-01
11	BE 40	1.162E-01	3.538E-05	4.143E-04	1.028E-03	3.395E-02	4.545E-01	6.401E-01	9.159E-01
12	BE 37	1.096E-01	9.168E-05	4.528E-04	1.194E-03	3.268E-02	3.344E-01	4.958E-01	7.088E-01
13	BE 39	1.073E-01	1.486E-04	9.894E-04	2.190E-03	6.474E-02	6.561E-01	8.340E-01	9.614E-01
14	BE 28	8.767E-02	6.807E-05	3.821E-04	1.252E-03	3.076E-02	1.977E-01	2.926E-01	5.099E-01
15	BE 30	7.585E-02	3.762E-05	2.958E-04	7.021E-04	2.891E-02	5.985E-01	7.516E-01	9.339E-01
16	BE 20	6.875E-02	5.036E-05	3.297E-04	7.920E-04	2.116E-02	1.416E-01	2.037E-01	3.999E-01
17	BE 29	2.267E-02	7.859E-06	6.199E-05	1.855E-04	5.716E-03	4.982E-02	8.498E-02	2.085E-01
18	BE 17	3.049E-02	2.294E-05	1.334E-04	4.409E-04	9.032E-03	7.088E-02	1.241E-01	2.134E-01
19	BE 35	1.906E-02	6.648E-06	5.200E-05	1.672E-04	4.712E-03	4.488E-02	6.995E-02	1.486E-01
20	BE 19	6.393E-03	8.789E-06	4.581E-05	1.031E-04	2.597E-03	4.016E-02	7.111E-02	1.954E-01
21	BE 25	6.137E-03	6.738E-06	4.462E-05	1.116E-04	2.908E-03	4.692E-02	8.006E-02	2.006E-01
22	BE 16	4.822E-03	3.879E-06	2.147E-05	4.748E-05	1.421E-03	2.062E-02	4.063E-02	1.424E-01
23	BE 14	4.764E-03	1.397E-06	2.170E-05	5.557E-05	1.003E-03	9.610E-03	1.435E-02	2.495E-02
24	BE 26	1.808E-03	5.399E-07	2.392E-06	1.296E-05	3.373E-04	6.654E-03	2.057E-02	9.376E-02
25	BE 16	7.224E-04	6.035E-07	3.514E-06	5.377E-06	2.394E-04	6.607E-03	1.984E-02	6.473E-02
26	BE 24	4.152E-04	2.084E-07	2.012E-06	5.085E-06	2.229E-04	5.364E-03	9.563E-03	3.690E-02
27	BE 34	3.014E-04	2.525E-08	2.509E-07	7.501E-07	5.559E-05	2.334E-03	5.565E-03	3.083E-02
28	BE 15	2.810E-04	6.759E-08	3.670E-07	1.086E-06	8.064E-05	4.733E-03	1.200E-02	5.438E-02
29	BE 27	2.553E-05	2.725E-09	2.433E-08	6.751E-08	4.733E-06	2.172E-04	8.401E-04	6.590E-03
30	BE 13	2.192E-05	1.166E-08	7.664E-08	1.624E-07	4.756E-06	5.611E-05	1.017E-04	3.361E-04
31	BE 12	9.251E-06	2.816E-09	1.701E-08	4.983E-08	1.412E-06	4.414E-05	1.049E-04	6.868E-04
32	BE 33	6.307E-06	1.362E-10	1.059E-09	4.736E-09	1.151E-06	1.477E-04	4.148E-04	3.593E-03

TABLE 9-2 (cont'd.)

MONTE CARLO SIMULATION RESULTS			10	50	100	500	900	950	990
CONFIDENCE LEVEL IN PER CENT			(98)	(90)	(80)		(80)	(90)	(98)
RANK	MIN CUT SET NO.								
1	22	1.540E-01	2.208E-05	2.941E-04	9.021E-04	2.035E-02	5.487E-01	8.997E-01	9.190E-01
2	18	1.191E-01	7.006E-05	4.053E-04	1.116E-03	3.245E-02	2.173E-01	2.867E-01	4.829E-01
2	12	1.191E-01	4.850E-05	5.012E-04	1.188E-03	2.956E-02	2.293E-01	3.566E-01	5.203E-01
3	10	7.742E-02	4.047E-06	8.375E-05	2.236E-04	2.141E-02	3.802E-01	5.721E-01	8.441E-01
4	26	5.479E-02	2.713E-05	1.647E-04	4.069E-04	1.110E-02	1.241E-01	2.474E-01	4.650E-01
4	23	5.479E-02	1.623E-05	1.244E-04	3.895E-04	1.169E-02	1.270E-01	2.178E-01	4.203E-01
4	11	8.478E-02	1.216E-05	9.735E-05	2.915E-04	1.078E-02	1.944E-01	3.354E-01	6.124E-01
4	17	5.479E-02	1.355E-05	3.372E-05	3.182E-04	1.112E-02	1.623E-01	3.270E-01	5.704E-01
5	25	4.645E-02	1.511E-05	8.802E-05	2.621E-04	7.709E-03	2.004E-01	3.298E-01	6.988E-01
6	39	3.812E-02	1.858E-05	9.739E-05	3.644E-04	1.003E-02	6.889E-02	1.388E-01	2.411E-01
7	9	3.573E-02	1.380E-05	9.690E-05	2.950E-04	8.808E-03	7.224E-02	1.136E-01	2.043E-01
8	27	2.362E-02	1.377E-05	6.567E-05	2.205E-04	5.738E-03	4.963E-02	8.137E-02	1.705E-01
8	24	2.362E-02	1.050E-05	7.745E-05	2.445E-04	4.921E-03	5.496E-02	8.547E-02	2.172E-01
9	16	2.320E-02	1.033E-06	2.600E-05	6.758E-05	4.260E-03	2.099E-01	4.037E-01	7.845E-01
10	15	2.297E-02	7.059E-06	8.189E-05	1.855E-04	5.716E-03	4.982E-02	8.408E-02	2.035E-01
10	6	2.297E-02	1.032E-05	7.003E-05	1.935E-04	4.898E-03	5.519E-02	9.238E-02	1.908E-01
11	30	7.623E-03	2.814E-06	1.860E-05	6.456E-05	1.608E-03	1.742E-02	3.948E-02	9.850E-02
11	42	7.623E-03	3.474E-06	2.211E-05	6.300E-05	1.880E-03	1.855E-02	2.998E-02	7.688E-02
12	21	7.147E-03	2.763E-06	1.986E-05	5.974E-05	1.639E-03	1.505E-02	2.701E-02	5.236E-02
13	36	1.906E-02	6.548E-05	5.200E-05	1.672E-04	4.712E-03	4.466E-02	6.995E-02	1.466E-01
14	3	4.764E-03	1.397E-06	2.170E-05	5.557E-05	1.003E-03	9.810E-03	1.435E-02	2.495E-02
15	20	3.298E-03	1.314E-06	8.422E-06	2.605E-05	6.689E-04	9.743E-03	1.896E-02	6.226E-02
15	8	3.298E-03	7.248E-07	6.724E-06	1.853E-05	6.043E-04	1.310E-02	3.311E-02	8.564E-02
16	29	3.014E-03	7.659E-07	5.319E-06	1.575E-05	5.680E-04	9.648E-03	2.118E-02	5.467E-02
16	38	3.014E-03	3.562E-07	4.805E-06	1.260E-05	6.346E-04	1.430E-02	2.472E-02	8.228E-02
16	41	3.014E-03	4.933E-07	5.590E-06	1.554E-05	5.760E-04	1.080E-02	2.418E-02	6.987E-02
17	32	2.740E-03	3.059E-07	3.167E-06	9.953E-06	5.241E-04	1.619E-02	3.688E-02	1.287E-01
18	14	1.106E-03	5.399E-07	2.392E-06	1.296E-05	3.373E-04	8.854E-03	2.057E-02	0.376E-02
18	5	1.607E-03	7.335E-07	2.547E-06	1.029E-05	3.102E-04	7.923E-03	1.843E-02	1.113E-01
19	37	2.201E-03	1.761E-07	2.764E-06	6.919E-06	4.809E-04	1.956E-02	3.957E-02	8.842E-02
20	33	1.191E-03	2.713E-07	2.342E-06	6.160E-06	2.058E-04	6.201E-03	1.213E-02	5.385E-02
21	19	4.215E-04	1.184E-07	9.270E-07	2.431E-06	6.879E-05	2.149E-03	5.902E-03	3.458E-02
22	35	3.014E-04	2.525E-08	2.500E-07	7.501E-07	5.859E-05	2.334E-03	5.655E-03	3.083E-02
23	28	2.555E-04	1.272E-08	1.054E-07	4.334E-07	4.860E-05	3.237E-03	7.992E-03	5.483E-02
23	40	2.555E-04	1.437E-08	3.381E-07	8.412E-07	5.514E-05	1.505E-03	4.308E-03	1.803E-02
24	7	2.323E-04	2.064E-08	1.763E-07	7.561E-07	3.981E-05	2.183E-03	7.052E-03	3.903E-02
25	37	1.277E-04	6.940E-09	6.219E-08	2.244E-07	2.840E-05	1.742E-03	3.566E-03	1.117E-02
26	4	2.515E-05	2.213E-09	1.890E-08	5.683E-08	3.732E-05	3.231E-04	1.105E-03	1.101E-02
26	13	2.515E-05	2.729E-09	2.490E-08	6.751E-08	4.239E-06	2.172E-04	8.401E-04	6.598E-03
27	2	2.192E-05	1.156E-08	7.366E-08	1.655E-07	4.056E-06	5.611E-05	1.017E-04	3.361E-04
28	1	9.291E-06	2.346E-09	1.701E-08	4.908E-08	1.412E-06	4.414E-05	1.049E-04	6.368E-04
29	34	6.397E-06	1.362E-10	1.059E-09	4.776E-09	1.151E-06	1.477E-04	4.148E-04	3.593E-03

TABLE B-2 (cont'd.)

1 REFERENCE TABLE FOR MIN LUT SETS

OUT SET NO ORDER BASIC EVENTS

1	2	BE 4	BE 12		
2		DE 6	BE 13		
3	00000000	DE 6	BE 14	BE 18	
4		BE 4	BE 15	BE 19	
5		BE 5	BE 16	BE 20	
6		BE 6	BE 17	BE 21	
7	00000000	DE 4	BE 18	BE 22	
8		BE 5	BE 19	BE 23	
9		BE 6	BE 20	BE 24	
10	00000000	BE 4	BE 21	BE 25	
11		BE 5	BE 22	BE 26	
12	00000000	BE 6	BE 23	BE 27	
13		BE 4	BE 24	BE 28	
14		BE 5	BE 25	BE 29	
15		BE 6	BE 26	BE 30	
16	00000000	BE 4	BE 27	BE 31	
17		BE 5	BE 28	BE 32	
18	00000000	BE 6	BE 29	BE 33	
19		BE 4	BE 30	BE 34	
20	00000000	BE 5	BE 31	BE 35	
21		BE 6	BE 32	BE 36	
22	00000000	BE 4	BE 33	BE 37	
23		BE 5	BE 34	BE 38	
24	00000000	BE 6	BE 35	BE 39	
25		BE 4	BE 36	BE 40	
26	00000000	BE 5	BE 37	BE 41	
27		BE 6	BE 38	BE 42	
28	00000000	BE 4	BE 39	BE 43	
29		BE 5	BE 40	BE 44	
30	00000000	BE 6	BE 41	BE 45	
31		BE 4	BE 42	BE 46	
32	00000000	BE 5	BE 43	BE 47	
33		BE 6	BE 44	BE 48	
34	00000000	BE 4	BE 45	BE 49	
35		BE 5	BE 46	BE 50	
36	00000000	BE 6	BE 47	BE 51	
37		BE 4	BE 48	BE 52	
38	00000000	BE 5	BE 49	BE 53	
39		BE 6	BE 50	BE 54	
40	00000000	BE 4	BE 51	BE 55	
41		BE 5	BE 52	BE 56	
42	00000000	BE 6	BE 53	BE 57	

TABLE B-3

IMPORTANCE LISTING FOR FAULT TREE 001001: SEISMIC SCRAM

MONTE CARLO SIMULATION RESULTS MEAN 10 50 100 300 900 950 990
 CONFIDENCE LEVEL IN PER CENT (90) (90) (80) (80) (80) (80) (98)
 PROBABILITY OF TOP EVENT 5.62E-05 4.07E-05 1.470E-05 2.024E-05 6.740E-05 5.191E-04 5.503E-04 1.477E-03

1

MONTE CARLO SIMULATION RESULTS MEAN 10 50 100 300 900 950 990
 CONFIDENCE LEVEL IN PER CENT (90) (90) (80) (80) (80) (80) (98)

RANK	BASIC EVENT	MEAN	10 (90)	50 (90)	100 (80)	300 (80)	900 (80)	950 (80)	990 (98)
1	BE 1	0.240E-01	1.056E-02	5.045E-02	9.798E-02	5.872E-01	2.240E-01	9.567E-01	9.056E-01
2	BE 3	7.971E-02	4.712E-04	1.719E-03	5.093E-03	4.261E-02	2.956E-01	4.795E-01	7.763E-01
3	BE 6	4.459E-02	7.703E-04	1.277E-03	2.168E-03	3.679E-02	3.716E-01	6.114E-01	8.465E-01
4	BE 41	3.139E-02	1.062E-04	5.794E-04	1.204E-03	1.893E-02	2.344E-01	3.810E-01	5.870E-01
5	BE 2	3.074E-02	1.721E-04	9.480E-04	1.658E-03	1.726E-02	1.372E-01	2.213E-01	5.161E-01
6	BE 32	2.121E-02	8.640E-05	3.945E-04	6.214E-04	1.838E-02	2.034E-01	3.174E-01	6.223E-01
7	BE 45	1.893E-02	5.862E-05	3.262E-04	8.518E-04	1.356E-02	1.768E-01	2.712E-01	5.449E-01
8	BE 39	1.590E-02	3.970E-05	2.286E-04	4.777E-04	8.888E-03	1.110E-01	1.696E-01	3.448E-01
9	BE 5	1.052E-02	4.391E-05	3.708E-04	7.253E-04	1.301E-02	1.986E-01	3.705E-01	6.711E-01
10	BE 4	9.933E-02	2.729E-05	1.892E-04	5.173E-04	1.973E-02	4.118E-01	6.072E-01	9.465E-01
11	BE 36	9.291E-03	3.845E-06	1.677E-05	7.810E-05	7.985E-03	3.350E-01	5.720E-01	9.291E-01
12	BE 39	9.014E-03	1.693E-06	2.939E-05	6.698E-05	6.908E-03	3.101E-01	5.636E-01	9.207E-01
13	BE 17	8.444E-02	2.744E-05	1.621E-04	3.504E-04	6.117E-03	7.604E-02	1.147E-01	3.614E-01
14	BE 40	7.754E-03	1.908E-05	1.206E-04	2.822E-04	5.778E-03	1.080E-01	2.533E-01	5.599E-01
15	BE 20	7.297E-02	2.856E-05	1.190E-04	2.646E-04	4.237E-03	4.807E-02	7.984E-02	1.794E-01
16	BE 42	7.220E-03	9.677E-05	3.832E-04	9.791E-04	1.282E-02	1.720E-01	2.969E-01	6.091E-01
17	BE 26	5.374E-03	1.772E-05	6.379E-05	1.321E-04	3.482E-03	4.213E-02	6.889E-02	2.254E-01
18	BE 37	6.986E-03	7.014E-06	1.038E-04	1.793E-04	3.629E-03	8.485E-02	2.019E-01	4.286E-01
19	BE 46	3.942E-03	1.354E-05	6.004E-05	1.108E-04	2.861E-03	8.851E-02	6.114E-02	1.430E-01
20	BE 35	6.652E-03	1.408E-05	1.070E-04	2.881E-04	4.783E-03	6.326E-02	9.992E-02	2.473E-01
21	BE 31	2.810E-03	5.237E-06	4.779E-05	1.074E-04	2.885E-03	5.492E-02	1.312E-01	4.980E-01
22	BE 44	2.383E-03	2.153E-05	1.128E-04	1.714E-04	2.160E-03	4.348E-02	1.078E-01	3.298E-01
23	BE 11	1.099E-03	3.909E-06	1.926E-05	3.911E-05	5.174E-04	5.010E-03	1.025E-02	2.323E-02
24	BE 47	7.071E-04	1.584E-06	1.468E-05	3.512E-05	5.326E-04	1.541E-02	3.310E-02	1.161E-01
25	BE 34	7.355E-04	4.949E-07	4.514E-06	1.269E-05	5.081E-04	1.237E-02	4.146E-02	1.839E-01
26	BE 19	6.446E-04	1.038E-06	1.187E-05	2.197E-05	3.847E-04	0.465E-03	2.046E-02	9.956E-02
27	BE 43	4.421E-04	3.683E-06	2.510E-05	6.810E-05	9.195E-04	1.814E-02	3.885E-02	1.264E-01
28	BE 30	4.312E-04	2.437E-07	3.383E-06	1.092E-05	8.126E-04	3.236E-02	7.709E-02	0.842E-01
29	BE 16	3.651E-04	6.064E-07	4.688E-06	1.410E-05	2.733E-04	8.136E-03	1.608E-02	7.423E-02
30	BE 33	2.686E-04	1.396E-06	2.929E-07	1.303E-05	1.428E-04	1.144E-02	3.213E-02	2.950E-01
31	BE 46	2.600E-04	2.995E-07	2.135E-06	8.149E-06	0.825E-04	1.331E-02	2.948E-02	1.131E-01
32	BE 21	1.787E-04	4.854E-08	4.742E-07	1.093E-06	7.825E-05	4.379E-03	1.454E-02	1.627E-01
33	BE 23	1.252E-04	2.001E-07	1.167E-06	4.857E-06	0.690E-05	2.647E-03	7.799E-03	2.163E-02
34	BE 22	7.565E-05	3.336E-08	2.083E-07	7.059E-07	3.160E-05	1.606E-03	4.445E-03	2.846E-02
35	BE 18	0.082E-05	3.916E-06	2.016E-07	6.823E-07	3.105E-05	1.138E-03	3.527E-03	1.457E-02
36	BE 23	2.740E-05	2.407E-08	1.747E-07	4.671E-07	1.094E-05	2.779E-04	6.195E-04	2.304E-03
37	BE 10	5.018E-06	1.116E-08	6.408E-08	1.416E-07	2.685E-06	4.304E-05	9.604E-05	2.669E-04
38	BE 15	4.127E-06	5.098E-09	3.767E-08	1.175E-07	7.597E-06	4.358E-04	1.028E-03	8.020E-03
39	BE 9	2.144E-06	1.804E-09	1.092E-08	3.151E-08	9.713E-07	3.484E-05	7.002E-05	6.808E-04
40	BE 24	1.179E-06	5.310E-10	6.708E-09	2.472E-08	2.362E-06	1.303E-04	3.338E-04	2.176E-03

TABLE B-3 (cont'd.)

MONTE CARLO SIMULATION RESULTS			10	50	100	500	900	950	990
CONFIDENCE LEVEL IN PER CENT			(98)	(90)	(80)		(80)	(90)	(98)
RANK	MIN CUT SET NO.								
1	1	0.248E-01	1.058E-02	5.045E-02	9.788E-02	5.572E-01	9.240E-01	9.587E-01	9.055E-01
2	9	7.971E-02	4.712E-04	1.719E-03	3.682E-03	4.281E-02	2.955E-01	4.785E-01	7.383E-01
3	2	3.084E-02	1.721E-04	9.480E-04	1.858E-03	1.728E-02	1.372E-01	2.213E-01	5.181E-01
4	14	1.374E-02	2.881E-05	1.770E-04	3.883E-04	7.004E-03	8.168E-02	1.452E-01	2.575E-01
5	18	6.933E-03	1.121E-05	1.237E-05	5.050E-05	5.202E-03	2.898E-01	5.881E-01	9.173E-01
6	17	6.322E-03	6.033E-06	4.529E-05	1.189E-04	2.943E-03	6.498E-02	1.537E-01	4.032E-01
7	24	8.872E-03	1.164E-05	8.399E-05	1.591E-04	4.486E-03	4.885E-02	8.083E-02	1.884E-01
8	15	4.123E-03	6.971E-06	3.588E-05	1.000E-04	1.894E-03	2.558E-02	4.819E-02	9.220E-02
9	12	2.839E-03	4.281E-06	3.768E-05	7.412E-05	1.232E-03	1.459E-02	2.718E-02	7.593E-02
10	49	2.199E-03	2.288E-06	2.487E-05	5.640E-05	1.133E-03	1.748E-02	3.071E-02	1.093E-01
10	88	2.199E-03	4.382E-06	2.083E-05	3.883E-05	1.048E-03	1.451E-02	3.180E-02	7.880E-02
10	81	2.199E-03	2.023E-06	2.044E-05	3.178E-05	1.056E-03	1.657E-02	3.922E-02	9.522E-02
10	78	2.199E-03	2.181E-06	1.793E-05	4.809E-05	9.728E-04	1.748E-02	3.237E-02	9.059E-02
11	30	1.774E-03	9.908E-07	1.059E-05	2.830E-05	6.906E-04	1.207E-02	3.207E-02	8.039E-02
11	27	1.374E-03	2.943E-06	1.273E-05	3.321E-05	6.850E-04	9.808E-03	1.838E-02	4.220E-02
11	34	1.374E-03	9.075E-07	9.724E-06	2.602E-05	6.890E-04	9.529E-03	2.809E-02	1.282E-01
12	41	6.322E-04	5.684E-07	2.410E-06	8.837E-06	3.128E-04	6.084E-03	2.597E-02	1.011E-01
12	29	6.322E-04	2.210E-07	2.089E-06	5.510E-06	3.124E-04	9.760E-03	3.423E-02	1.694E-01
12	23	6.322E-04	4.755E-07	4.076E-06	1.135E-05	2.671E-04	7.137E-03	2.011E-02	1.310E-01
12	35	6.322E-04	5.153E-07	2.839E-06	7.829E-06	2.648E-04	8.907E-03	2.313E-02	1.193E-01
13	60	4.098E-04	7.275E-07	3.632E-06	6.312E-05	2.303E-04	3.685E-03	6.904E-03	2.688E-02
14	8	1.098E-03	3.900E-06	1.920E-05	3.811E-05	5.174E-04	5.010E-03	1.025E-02	2.323E-02
15	72	4.398E-04	7.586E-07	3.270E-06	8.619E-06	2.264E-04	3.667E-03	5.972E-03	2.558E-02
15	67	4.398E-04	6.902E-07	4.413E-06	9.589E-06	2.097E-04	3.534E-03	6.460E-03	2.283E-02
15	89	4.398E-04	5.140E-07	4.013E-06	1.040E-05	2.033E-04	3.143E-03	5.981E-03	1.878E-02
16	21	4.123E-04	2.977E-07	2.442E-06	7.902E-06	1.851E-04	3.724E-03	7.139E-03	1.630E-02
17	14	3.793E-04	5.843E-07	3.023E-06	7.740E-06	1.745E-04	4.044E-03	9.431E-03	6.098E-02
18	28	3.181E-04	2.057E-07	2.282E-06	5.690E-06	1.520E-04	4.445E-03	1.295E-02	4.481E-02
18	32	3.181E-04	2.140E-07	1.206E-06	4.092E-06	1.509E-04	3.654E-03	1.185E-02	3.411E-02
18	40	2.749E-04	2.027E-07	2.919E-06	6.209E-06	1.282E-04	4.498E-03	3.198E-03	1.118E-02
20	54	4.398E-04	1.842E-07	1.603E-06	6.097E-06	2.138E-04	4.643E-03	1.031E-02	5.102E-02
21	33	2.749E-04	1.592E-07	1.588E-06	4.808E-06	1.300E-04	2.778E-03	7.551E-03	1.687E-02
22	38	2.605E-04	9.980E-08	2.684E-07	1.345E-06	1.344E-04	1.144E-02	3.178E-02	2.842E-01
23	11	2.086E-04	1.271E-07	1.379E-06	3.872E-06	6.781E-05	2.819E-03	5.735E-03	3.433E-02
24	5	1.767E-04	4.894E-08	4.742E-07	1.093E-06	7.525E-05	4.379E-03	1.484E-02	1.627E-01
25	42	1.374E-04	1.291E-07	1.028E-06	2.798E-06	6.391E-05	1.247E-03	2.809E-03	1.082E-02
26	34	1.767E-04	1.807E-08	3.802E-07	1.089E-06	6.549E-05	4.471E-03	1.200E-02	6.104E-02
27	63	8.798E-05	3.399E-08	3.359E-07	1.093E-06	3.892E-05	1.089E-03	2.382E-03	9.519E-03
28	43	1.767E-04	2.658E-08	3.771E-07	1.301E-06	6.937E-05	4.798E-03	1.205E-02	5.429E-02
29	68	2.486E-05	7.517E-08	4.958E-07	1.312E-06	2.970E-05	7.101E-04	1.698E-03	4.854E-03
30	81	8.798E-05	4.119E-08	4.317E-07	1.152E-06	4.323E-05	1.089E-03	2.644E-03	9.688E-03
31	7	7.586E-05	3.336E-08	2.089E-07	7.059E-07	3.180E-05	1.608E-03	4.445E-03	2.848E-02
32	31	5.360E-05	1.386E-08	9.241E-08	4.244E-07	3.019E-05	2.197E-03	5.172E-03	4.487E-02
32	40	5.360E-05	9.352E-09	1.161E-07	4.631E-07	2.802E-05	1.852E-03	6.460E-03	4.199E-02
32	20	7.793E-05	2.007E-08	1.868E-07	5.241E-07	1.892E-05	5.192E-04	1.139E-03	6.205E-03
34	85	3.477E-05	1.925E-08	1.265E-07	4.377E-07	1.494E-05	5.902E-04	1.295E-03	7.850E-03
34	50	3.477E-05	1.167E-08	1.809E-07	3.235E-07	1.189E-05	7.800E-04	1.690E-03	6.354E-03
34	47	3.477E-05	1.580E-08	1.188E-07	3.400E-07	1.478E-05	6.138E-04	1.530E-03	7.923E-03
34	74	3.477E-05	1.459E-08	1.568E-07	5.091E-07	1.844E-05	5.690E-04	1.483E-03	7.102E-03
34	53	3.477E-05	7.293E-09	6.418E-08	3.351E-07	1.615E-05	5.253E-04	1.810E-03	1.089E-02
35	44	3.181E-05	1.907E-08	2.179E-07	5.241E-07	1.632E-05	4.092E-04	1.269E-03	6.104E-03
37	9	2.740E-05	2.437E-08	1.747E-07	4.671E-07	1.094E-05	2.779E-04	6.195E-04	2.385E-03
37	25	2.605E-05	2.727E-09	3.379E-08	1.671E-07	1.921E-05	1.502E-03	4.280E-03	2.457E-02
37	13	2.506E-05	2.340E-09	7.218E-08	1.895E-07	1.559E-05	6.938E-04	1.887E-03	1.231E-02
37	22	2.605E-05	1.409E-09	1.714E-08	1.413E-07	1.618E-05	1.381E-03	3.774E-03	2.134E-02
38	38	1.897E-05	2.044E-08	7.547E-08	2.558E-07	1.030E-05	2.611E-04	6.779E-04	3.709E-03

TABLE B-3 (cont'd.)

1 ORIGINAL TABLE WITH CUT SETS

CUT SETS ORIGINAL BASIC EVENTS

1	1	DE 1			
2	1	DE 2			
3	1	DE 3			
4	1	DE 4	NE 9		
5	1	DE 5	NE 11		
6	1	DE 6	NE 12		
7	1	DE 7	DE 13		
8	1	DE 8	DE 14		
9	1	DE 9	DE 15	RE 16	
10	1	DE 10	DE 16	RE 17	
11	1	DE 11	DE 17	RE 18	
12	1	DE 12	DE 18	RE 19	
13	1	DE 13	DE 19	RE 20	
14	1	DE 14	DE 20	RE 21	
15	1	DE 15	DE 21	RE 22	
16	1	DE 16	DE 22	RE 23	
17	1	DE 17	DE 23	RE 24	
18	1	DE 18	DE 24	RE 25	
19	1	DE 19	DE 25	RE 26	
20	1	DE 20	DE 26	RE 27	
21	1	DE 21	DE 27	RE 28	
22	1	DE 22	DE 28	RE 29	
23	1	DE 23	DE 29	RE 30	
24	1	DE 24	DE 30	RE 31	
25	1	DE 25	DE 31	RE 32	
26	1	DE 26	DE 32	RE 33	
27	1	DE 27	DE 33	RE 34	
28	1	DE 28	DE 34	RE 35	
29	1	DE 29	DE 35	RE 36	
30	1	DE 30	DE 36	RE 37	
31	1	DE 31	DE 37	RE 38	
32	1	DE 32	DE 38	RE 39	
33	1	DE 33	DE 39	RE 40	
34	1	DE 34	DE 40	RE 41	
35	1	DE 35	DE 41	RE 42	
36	1	DE 36	DE 42	RE 43	
37	1	DE 37	DE 43	RE 44	
38	1	DE 38	DE 44	RE 45	
39	1	DE 39	DE 45	RE 46	
40	1	DE 40	DE 46	RE 47	
41	1	DE 41	DE 47	RE 48	
42	1	DE 42	DE 48	RE 49	
43	1	DE 43	DE 49	RE 50	
44	1	DE 44	DE 50	RE 51	
45	1	DE 45	DE 51	RE 52	
46	1	DE 46	DE 52	RE 53	
47	1	DE 47	DE 53	RE 54	
48	1	DE 48	DE 54	RE 55	
49	1	DE 49	DE 55	RE 56	
50	1	DE 50	DE 56	RE 57	
51	1	DE 51	DE 57	RE 58	
52	1	DE 52	DE 58	RE 59	
53	1	DE 53	DE 59	RE 60	
54	1	DE 54	DE 60	RE 61	
55	1	DE 55	DE 61	RE 62	
56	1	DE 56	DE 62	RE 63	
57	1	DE 57	DE 63	RE 64	
58	1	DE 58	DE 64	RE 65	
59	1	DE 59	DE 65	RE 66	
60	1	DE 60	DE 66	RE 67	
61	1	DE 61	DE 67	RE 68	
62	1	DE 62	DE 68	RE 69	
63	1	DE 63	DE 69	RE 70	
64	1	DE 64	DE 70	RE 71	
65	1	DE 65	DE 71	RE 72	
66	1	DE 66	DE 72	RE 73	
67	1	DE 67	DE 73	RE 74	
68	1	DE 68	DE 74	RE 75	
69	1	DE 69	DE 75	RE 76	
70	1	DE 70	DE 76	RE 77	
71	1	DE 71	DE 77	RE 78	
72	1	DE 72	DE 78	RE 79	
73	1	DE 73	DE 79	RE 80	
74	1	DE 74	DE 80	RE 81	
75	1	DE 75	DE 81	RE 82	
76	1	DE 76	DE 82	RE 83	
77	1	DE 77	DE 83	RE 84	
78	1	DE 78	DE 84	RE 85	
79	1	DE 79	DE 85	RE 86	
80	1	DE 80	DE 86	RE 87	
81	1	DE 81	DE 87	RE 88	

TABLE B-4

IMPORTANCE LISTING FOR FAULT TREE WITHOUT SEISMIC SCRAM, CHS ONLY

MONTE CARLO SIMULATION RESULTS		MEAN	10	50	100	500	900	950	990
CONFIDENCE LEVEL IN PER CENT			(98)	(90)	(80)		(80)	(90)	(98)
PROBABILITY OF TOP EVENT		2.30E-05	4.878E-07	1.092E-05	1.606E-05	1.029E-05	8.151E-05	1.365E-04	9.698E-04
RANK	BASIC EVENT								
1	BE 5	8.874E-01	1.484E-03	7.610E-03	1.918E-02	4.203E-01	1.265E-01	8.734E-01	9.948E-01
2	BE 41	4.798E-01	7.125E-04	2.831E-02	6.462E-03	2.015E-01	8.378E-01	7.594E-01	8.857E-01
3	BE 32	3.241E-01	9.123E-04	2.364E-03	6.163E-03	1.448E-01	6.607E-01	6.822E-01	6.117E-01
4	BE 45	2.755E-01	2.659E-04	2.061E-03	5.499E-03	1.255E-01	4.448E-01	5.773E-01	7.450E-01
5	BE 38	2.352E-01	2.253E-04	1.815E-03	3.718E-03	6.300E-02	3.369E-01	4.408E-01	6.327E-01
6	BE 9	1.608E-01	1.440E-04	1.424E-03	4.470E-03	1.040E-01	7.382E-01	8.879E-01	9.880E-01
7	BE 4	1.518E-01	2.097E-04	1.481E-03	5.319E-03	1.803E-01	9.120E-01	9.881E-01	9.958E-01
8	BE 35	1.480E-01	1.300E-05	3.278E-04	9.114E-04	5.094E-02	6.037E-01	9.422E-01	9.932E-01
9	BE 38	1.377E-01	6.392E-06	1.778E-04	8.509E-04	8.052E-02	6.038E-01	9.389E-01	9.821E-01
10	BE 17	1.290E-01	1.122E-04	6.727E-04	2.716E-03	5.811E-02	2.488E-01	3.570E-01	5.171E-01
11	BE 40	1.105E-01	6.477E-05	5.043E-04	1.344E-03	4.089E-02	4.478E-01	6.593E-01	8.814E-01
12	BE 20	1.109E-01	1.247E-04	9.137E-04	1.891E-03	3.979E-02	1.787E-01	2.708E-01	4.837E-01
13	BE 42	1.103E-01	9.710E-04	4.960E-03	9.674E-03	1.160E-01	5.895E-01	6.608E-01	8.325E-01
14	BE 37	1.087E-01	6.201E-05	3.847E-04	1.061E-03	3.237E-02	4.084E-01	6.045E-01	8.447E-01
15	BE 35	1.019E-01	6.440E-05	8.085E-04	2.334E-03	4.141E-02	2.041E-01	2.573E-01	4.197E-01
16	BE 28	8.212E-02	7.112E-05	5.047E-04	1.177E-03	2.892E-02	1.685E-01	2.321E-01	4.010E-01
17	BE 48	6.023E-02	5.977E-05	5.538E-04	1.363E-03	2.424E-02	1.357E-01	1.958E-01	3.289E-01
18	BE 31	4.294E-02	2.102E-05	2.571E-04	6.648E-04	2.154E-02	2.753E-01	4.959E-01	7.863E-01
19	BE 44	3.642E-02	5.993E-05	4.474E-04	1.297E-03	3.355E-02	1.904E-01	3.057E-01	5.892E-01
20	BE 11	1.660E-02	1.815E-05	1.150E-04	2.815E-04	3.603E-03	2.723E-02	4.508E-02	1.068E-01
21	BE 47	1.203E-02	1.032E-05	5.461E-05	1.863E-04	5.795E-03	6.989E-02	1.194E-01	3.041E-01
22	BE 34	1.125E-02	1.922E-06	3.461E-05	9.633E-05	4.359E-03	7.318E-02	1.874E-01	4.412E-01
23	BE 19	9.853E-03	7.239E-06	4.848E-05	1.282E-04	3.329E-03	4.384E-02	7.240E-02	2.177E-01
24	BE 43	6.793E-03	9.518E-05	2.854E-04	4.398E-04	3.416E-03	8.374E-02	1.014E-01	3.379E-01
25	BE 30	6.886E-03	2.655E-06	1.845E-05	7.755E-05	4.113E-03	1.767E-01	3.815E-01	7.084E-01
26	BE 16	5.579E-03	2.878E-06	3.336E-05	8.697E-05	2.419E-03	3.570E-02	6.460E-02	2.893E-01
27	BE 33	4.101E-03	1.897E-07	1.997E-06	6.104E-06	9.839E-04	7.271E-02	1.556E-01	6.227E-01
28	BE 46	3.943E-03	1.266E-06	2.137E-05	6.567E-05	2.410E-03	6.990E-02	1.347E-01	3.078E-01
29	BE 21	2.730E-03	2.262E-07	1.821E-06	7.271E-06	4.086E-04	4.472E-02	1.300E-01	4.266E-01
30	BE 25	1.013E-03	5.540E-07	6.772E-06	1.610E-05	8.673E-04	1.836E-02	3.378E-02	1.402E-01
31	BE 22	1.155E-03	1.831E-07	1.006E-06	3.505E-06	2.811E-04	1.123E-02	3.494E-02	1.850E-01
32	BE 18	4.709E-04	1.649E-07	1.522E-06	4.425E-06	1.825E-04	7.673E-03	2.036E-02	2.317E-01
33	BE 23	4.200E-04	1.902E-07	1.195E-06	3.441E-06	8.872E-05	1.954E-03	2.900E-02	6.677E-03
34	BE 10	7.720E-05	3.748E-05	3.842E-07	8.769E-07	1.827E-05	2.743E-04	5.942E-04	1.089E-03
35	BE 15	6.396E-05	2.442E-08	2.210E-07	7.704E-07	6.292E-05	2.925E-03	8.980E-03	7.473E-02
36	BE 8	3.278E-05	6.515E-09	5.117E-08	1.600E-07	8.149E-06	2.340E-04	5.607E-04	2.736E-03
37	BE 24	1.802E-05	3.628E-09	4.107E-08	1.553E-07	1.188E-05	2.350E-04	1.795E-03	1.126E-02

TABLE B-4 (cont'd.)

MONTE CARLO SIMULATION RESULTS				10	50	100	500	900	950	990
CONFIDENCE LEVEL IN PER CENT				(98)	(90)	(80)		(80)	(90)	(98)
RANK	MIN	CUT	SET NO							
1	15	2	100E-01	1.857E-04	1.284E-03	2.523E-03	5.603E-02	2.030E-01	3.930E-01	5.927E-01
2	13	1	195E-01	4.932E-04	1.118E-04	5.615E-04	5.076E-02	7.870E-01	9.248E-01	9.916E-01
3	14	9	650E-02	2.293E-05	2.034E-04	5.608E-04	2.267E-02	3.298E-01	5.039E-01	8.143E-01
4	21	1	050E-01	7.971E-05	4.555E-04	1.426E-03	2.644E-02	1.422E-01	2.121E-01	3.351E-01
5	12	6	200E-04	7.234E-05	2.531E-04	6.922E-04	1.572E-02	9.173E-02	1.181E-01	2.483E-01
6	9	4	430E-02	1.343E-05	1.849E-04	5.652E-04	9.807E-03	7.048E-02	1.239E-01	2.908E-01
7	45	3	160E-02	7.102E-06	1.245E-04	3.185E-04	6.053E-03	6.335E-02	9.785E-02	1.779E-01
7	72	3	360E-02	2.391E-05	1.285E-04	3.985E-04	7.821E-03	6.286E-02	9.586E-02	1.808E-01
7	48	3	760E-02	1.414E-05	1.090E-04	3.472E-04	9.986E-03	6.578E-02	9.772E-02	1.910E-01
7	63	3	350E-02	1.185E-05	1.475E-04	4.541E-04	9.244E-03	6.817E-02	9.880E-02	1.583E-01
8	24	2	100E-02	1.325E-05	9.292E-05	2.401E-04	4.951E-03	3.562E-02	5.809E-02	1.170E-01
8	27	2	100E-02	7.516E-06	7.957E-05	1.034E-04	4.802E-03	4.720E-02	9.258E-02	1.748E-01
8	33	2	100E-02	6.671E-06	5.236E-05	1.819E-04	5.247E-03	5.451E-02	9.692E-02	2.419E-01
9	26	9	603E-03	7.360E-07	1.080E-05	4.605E-05	1.744E-03	5.765E-02	1.236E-01	3.718E-01
9	19	9	980E-03	1.153E-06	1.676E-05	5.931E-05	2.349E-03	4.295E-02	7.602E-02	2.080E-01
9	20	9	650E-03	1.700E-06	1.717E-05	6.209E-05	2.146E-03	3.890E-02	8.512E-02	2.378E-01
9	32	9	640E-03	1.715E-06	1.488E-05	4.721E-05	2.309E-03	4.768E-02	7.788E-02	3.117E-01
10	57	6	720E-03	2.516E-06	2.198E-05	7.506E-05	1.364E-03	1.605E-02	2.279E-02	8.059E-02
11	5	1	680E-02	1.615E-05	1.153E-04	2.815E-04	3.603E-03	2.720E-02	4.508E-02	1.088E-01
12	89	6	720E-03	4.023E-06	2.569E-05	7.173E-05	1.527E-03	1.547E-02	2.607E-02	5.985E-02
12	66	6	720E-03	5.277E-06	2.546E-05	6.699E-05	1.531E-03	1.581E-02	2.807E-02	7.298E-02
12	54	6	720E-03	2.582E-06	2.759E-05	8.955E-05	1.981E-03	1.695E-02	2.862E-02	6.298E-02
13	10	6	300E-03	1.510E-06	2.048E-05	5.678E-05	1.450E-03	1.785E-02	3.816E-02	7.894E-02
14	11	5	708E-03	1.634E-06	1.144E-05	3.989E-05	1.296E-03	2.193E-02	4.003E-02	1.073E-01
15	23	4	830E-03	1.175E-06	9.217E-06	2.005E-05	1.048E-03	2.172E-02	3.928E-02	1.285E-01
15	29	4	830E-03	1.229E-06	6.812E-06	2.900E-05	9.047E-04	2.356E-02	4.182E-02	8.496E-02
16	30	4	200E-03	6.243E-07	1.692E-05	4.016E-05	9.450E-04	1.168E-02	2.179E-02	4.515E-02
17	81	6	720E-03	1.807E-06	1.953E-05	5.150E-05	1.812E-03	1.800E-02	3.425E-02	9.512E-02
18	42	4	230E-03	2.289E-06	1.603E-05	6.137E-05	8.948E-04	1.014E-02	1.848E-02	4.808E-02
19	25	4	095E-03	1.334E-07	1.926E-05	5.493E-05	3.8E-04	6.990E-02	1.545E-01	6.221E-01
20	8	3	138E-03	4.722E-07	0.956E-05	2.369E-05	6.448E-04	6.882E-03	0.081E-02	9.750E-02
21	2	2	730E-03	7.262E-07	1.821E-06	7.271E-06	4.888E-04	4.472E-02	1.300E-01	4.260E-01
22	40	2	730E-03	1.393E-07	3.626E-06	1.189E-05	6.109E-04	2.097E-02	6.129E-02	1.899E-01
22	31	2	730E-03	1.490E-07	2.722E-06	8.528E-06	5.651E-04	2.166E-02	6.716E-02	1.704E-01
23	38	2	100E-03	4.853E-07	6.004E-06	2.189E-05	4.022E-04	7.283E-03	1.608E-02	4.990E-02
24	60	1	134E-03	2.042E-07	3.895E-06	9.470E-06	2.698E-04	4.012E-03	8.309E-03	2.118E-02
24	78	1	134E-03	4.043E-07	3.892E-06	1.277E-05	2.842E-04	4.449E-03	8.785E-03	3.051E-02
25	4	1	190E-03	1.931E-07	1.006E-06	3.508E-06	2.811E-04	1.123E-02	3.484E-02	1.885E-01
26	28	1	190E-04	3.786E-06	4.487E-07	2.120E-06	1.468E-04	9.350E-03	3.124E-02	1.238E-01
27	38	1	250E-07	2.454E-07	4.183E-06	1.078E-05	2.827E-04	3.904E-03	7.782E-03	2.085E-02
28	37	6	190E-04	1.059E-07	5.701E-07	1.928E-06	1.738E-04	6.967E-03	3.084E-02	1.594E-01
37	17	5	798E-04	1.149E-07	9.919E-07	3.180E-06	1.337E-04	2.873E-03	5.445E-03	1.897E-02
30	44	5	513E-04	6.174E-08	6.404E-07	1.820E-06	1.063E-04	3.125E-03	8.199E-03	5.709E-02
30	47	5	513E-04	8.205E-08	5.877E-07	2.301E-06	1.180E-04	2.827E-03	5.406E-03	2.645E-02
30	62	5	513E-04	3.670E-08	6.513E-07	1.619E-06	1.031E-04	3.523E-03	9.938E-03	4.738E-02
30	71	5	513E-04	5.922E-08	6.197E-07	2.384E-06	1.021E-04	2.972E-03	6.214E-03	3.402E-02
30	50	5	513E-04	3.708E-08	5.889E-07	2.029E-06	1.269E-04	3.240E-03	6.638E-03	1.616E-02
31	41	4	430E-04	1.827E-07	1.073E-06	2.822E-06	9.651E-05	2.032E-03	8.087E-03	2.818E-02
32	6	4	290E-04	1.902E-07	1.195E-06	3.441E-06	8.672E-05	1.954E-03	2.990E-03	6.877E-03
33	19	4	095E-04	1.236E-08	2.635E-07	7.602E-07	8.658E-05	5.362E-03	2.024E-02	1.269E-01
34	10	4	01E-04	3.765E-08	5.377E-07	1.463E-05	5.100E-05	4.287E-03	1.473E-02	1.975E-01
33	22	4	095E-04	1.157E-08	4.021E-07	8.039E-07	6.735E-05	6.223E-03	2.403E-02	1.852E-01
34	35	2	390E-04	5.587E-08	3.749E-07	1.505E-06	7.406E-05	1.014E-03	2.826E-03	1.180E-02
35	65	2	657E-04	2.924E-08	3.193E-07	1.161E-06	4.867E-05	1.793E-03	5.126E-03	2.047E-02
36	68	2	657E-04	2.775E-08	3.405E-07	9.051E-07	4.803E-05	1.508E-03	4.242E-03	1.757E-02
36	77	2	657E-04	2.301E-08	3.063E-07	1.127E-06	6.863E-05	1.014E-03	3.581E-03	1.454E-02
36	53	2	657E-04	3.714E-08	3.785E-07	9.283E-07	5.192E-05	2.028E-03	4.616E-03	1.670E-02
36	56	2	570E-04	3.153E-08	2.677E-07	9.031E-07	4.853E-05	1.681E-03	3.438E-03	1.411E-02
36	59	2	670E-04	2.971E-08	2.118E-07	1.082E-06	4.704E-05	1.892E-03	3.263E-03	8.029E-03
36	74	5	513E-05	3.861E-09	3.95E-08	1.405E-07	1.370E-05	4.644E-04	1.113E-03	7.602E-03
37	75	1	134E-02	2.260E-08	2.44E-07	6.637E-07	2.167E-05	6.739E-04	1.599E-03	4.354E-03

TABLE B-4 (cont'd.)

REFERENCE TABLE FOR MIN CUT SETS

CUT SET NO. ORDER BASIC EVENTS

1	2	BE 4	BE 9		
2	2	BE 4	BE 10		
3	2	BE 4	BE 11		
4	2	BE 4	BE 12		
5	2	BE 4	BE 13		
6	2	BE 4	BE 14		
7	2	BE 4	BE 15		
8	2	BE 4	BE 16		
9	2	BE 4	BE 17		
10	2	BE 4	BE 18		
11	2	BE 4	BE 19		
12	2	BE 4	BE 20		
13	2	BE 4	BE 21		
14	2	BE 4	BE 22		
15	2	BE 4	BE 23		
16	2	BE 4	BE 24		
17	2	BE 4	BE 25		
18	2	BE 4	BE 26		
19	2	BE 4	BE 27		
20	2	BE 4	BE 28		
21	2	BE 4	BE 29		
22	2	BE 4	BE 30		
23	2	BE 4	BE 31		
24	2	BE 4	BE 32		
25	2	BE 4	BE 33		
26	2	BE 4	BE 34		
27	2	BE 4	BE 35		
28	2	BE 4	BE 36		
29	2	BE 4	BE 37		
30	2	BE 4	BE 38		
31	2	BE 4	BE 39		
32	2	BE 4	BE 40		
33	2	BE 4	BE 41		
34	2	BE 4	BE 42		
35	2	BE 4	BE 43		
36	2	BE 4	BE 44		
37	2	BE 4	BE 45		
38	2	BE 4	BE 46		
39	2	BE 4	BE 47		
40	2	BE 4	BE 48		
41	2	BE 4	BE 49		
42	2	BE 4	BE 50		
43	2	BE 4	BE 51		
44	2	BE 4	BE 52		
45	2	BE 4	BE 53		
46	2	BE 4	BE 54		
47	2	BE 4	BE 55		
48	2	BE 4	BE 56		
49	2	BE 4	BE 57		
50	2	BE 4	BE 58		
51	2	BE 4	BE 59		
52	2	BE 4	BE 60		
53	2	BE 4	BE 61		
54	2	BE 4	BE 62		
55	2	BE 4	BE 63		
56	2	BE 4	BE 64		
57	2	BE 4	BE 65		
58	2	BE 4	BE 66		
59	2	BE 4	BE 67		
60	2	BE 4	BE 68		
61	2	BE 4	BE 69		
62	2	BE 4	BE 70		
63	2	BE 4	BE 71		
64	2	BE 4	BE 72		
65	2	BE 4	BE 73		
66	2	BE 4	BE 74		
67	2	BE 4	BE 75		
68	2	BE 4	BE 76		
69	2	BE 4	BE 77		
70	2	BE 4	BE 78		
71	2	BE 4	BE 79		
72	2	BE 4	BE 80		
73	2	BE 4	BE 81		
74	2	BE 4	BE 82		
75	2	BE 4	BE 83		
76	2	BE 4	BE 84		
77	2	BE 4	BE 85		
78	2	BE 4	BE 86		